NUREG-1125
Volume 22

A Compilation of Reports of The Advisory Committee on Reactor Safeguards

2000 Annual

U. S. Nuclear Regulatory Commission

April 2001

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NUREG-1125
Volume 22

A Compilation of Reports of The Advisory Committee on Reactor Safeguards

2000 Annual

U. S. Nuclear Regulatory Commission

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ABSTRACT

This compilation contains 56 ACRS reports submitted to the U. S. Nuclear Regulatory Commission (NRC), or to the NRC Executive Director for Operations, during calendar year 2000. In addition, a report to the Commission on the NRC Safety Research Program, NUREG-1635, Volume 3, is included by reference only. 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 organized in chronological order.

PREFACE

The enclosed reports, issued during calendar year 2000, contain the recommendations and comments of the U. S. NRC's Advisory Committee on Reactor Safeguards on various regulatory matters. NUREG-1125 is published annually. Previous issues are as follows:

Volume	Inclusive Dates
1 through 6	September 1957 through December 1984
7	Calendar Year 1985
8	Calendar Year 1986
9	Calendar Year 1987
10	Calendar Year 1988
11	Calendar Year 1989
12	Calendar Year 1990
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16	Calendar Year 1994
17	Calendar Year 1995
18	Calendar Year 1996
19	Calendar Year 1997
20	Calendar Year 1998
21	Calendar Year 1999

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UNITED STATES
NUCLEAR REGULATORY COMMISSION
ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
WASHINGTON, D.C. 20555-0001

February 8, 2000

The Honorable Richard A. Meserve
Chairman
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555-0001

Dear Chairman Meserve:

SUBJECT: SECY-00-0011, "EVALUATION OF THE REQUIREMENT FOR LICENSEES TO UPDATE THEIR INSERVICE INSPECTION AND INSERVICE TESTING PROGRAMS EVERY 120 MONTHS"

During the 469th meeting of the Advisory Committee on Reactor Safeguards, February 3-5, 2000, we discussed the NRC staff's analysis of ACRS comments and recommendations regarding the 120-month update requirement for inservice inspection (ISI) and inservice testing (IST) programs, which is included in SECY-00-0011 and also in a January 13, 2000, memorandum from the NRC Executive Director for Operations. The staff continues to recommend that the update requirement be eliminated from 10 CFR 50.55a, "Codes and standards." If the update requirement is eliminated, any subsequent NRC-imposed update of Section XI of the American Society of Mechanical Engineers (ASME) Code would be subject to a backfit analysis in accordance with 10 CFR 50.109, "Backfitting."

We continue to recommend that the Commission adopt Option 2 proposed by the staff in SECY-00-0011 and retain the 120-month update requirement for ISI and IST programs in 10 CFR 50.55a.

The assurance of the integrity of the reactor coolant pressure boundary and the containment is one of the cornerstones of the NRC regulatory system. The license renewal process is predicated on the demonstration that any effects of aging on critical plant systems will be adequately managed. Effective ISI and IST programs are crucial to this demonstration and to public confidence in the license renewal process. Because of this, we believe that the ISI and IST standards are different from other industry standards for which there is no mandatory update requirement.

In support of Option 1 in SECY-00-0011, the Nuclear Energy Institute (NEI) and the staff argue that the current ASME Code requirements have reached such a level of maturity that further updating will provide little benefit. We believe that the review of the past decade of experience presented to us by the ASME demonstrated that there were significant changes to the ISI, IST, and operations and maintenance requirements that improved the effectiveness and efficiency of these programs. Indeed, both the staff and NEI recognized that the 1989 version of the Code would have to be updated to include requirements from the 1992, 1995, and 1996 versions of the Code to be considered as an acceptable baseline. The staff and NEI arguments would be

more convincing if they could identify a decade in which significant changes had not been made in the Code. Changes in the Code reflect the latest knowledge and experience in inspections and testing and sometimes provide relief from existing requirements.

Changes are not introduced into the ASME Code requirements frivolously. Approximately 30% of the Section XI membership are representatives of licensees. They have a very good understanding of the impact of any proposed changes on their operations. Any proposed changes are subject to peer review by a broad-based group of experts from the licensees, manufacturers, vendors, the NRC, and other engineering and consulting organizations. If the update requirement is eliminated, the staff may be required to demonstrate to the public, including State officials, why requirements in consensus standards should not be adopted.

Under Option 1, any mandated updates to the ISI and IST programs would have to pass the 10 CFR 50.109 backfit criteria. In SECY-00-0011, the staff argues that it can make qualitative assessments to demonstrate a substantial increase in the overall protection of the public health and safety. We continue to believe that 10 CFR 50.109 evaluations are not well suited to assess the appropriateness of defense-in-depth measures, such as the ASME Code updates. Effective ISI and IST programs based on a broad technical consensus standard are prudent to provide confidence that the effects of aging are adequately managed.

Sincerely,

Dana A. Powers

Dana A. Powers
Chairman

<u>References</u>:

1. SECY-00-0011, memorandum dated January 14, 2000, from William D. Travers, Executive Director for Operations, NRC, for the Commissioners, Subject: Evaluation of the Requirement for Licensees to Update Their Inservice Inspection and Inservice Testing Programs Every 120 Months.
2. Letter dated January 13, 2000, from William D. Travers, Executive Director for Operations, NRC, to Dana A. Powers, Chairman, ACRS, Subject: Draft Commission Paper Regarding 120-Month Update Requirement for Inservice Inspection and Inservice Testing Programs.
3. ACRS letter dated December 8, 1999, from Dana A. Powers, Chairman, ACRS, to the Honorable Richard A. Meserve, Chairman, NRC, Subject: Draft Commission Paper Regarding the 120-Month Update Requirement for Inservice Inspection and Inservice Testing Programs.
4. ACRS letter dated May 19, 1999, from Dana A. Powers, Chairman, ACRS, to the Honorable Shirley A. Jackson, Chairman, NRC, Subject: The Role of Defense In Depth in a Risk-Informed Regulatory System.
5. Memorandum dated June 24, 1999, from Annette L. Vietti-Cook, Secretary of the Commission, to William D. Travers, Executive Director for Operations, NRC, Subject: Staff Requirements - Reconsideration of SECY-99-017 (Proposed Amendment to 10 CFR 50.55a).

6. Letter dated April 19, 1999, from Dana A. Powers, Chairman, ACRS, to William D. Travers, Executive Director for Operations, NRC, Subject: SECY-99-017, "Proposed Amendment to 10 CFR 50.55a."
7. Table provided by ASME during ACRS meeting, December 2-4, 1999, "Important Section XI SG NDE Code Changes and Code Cases, 1989 Addenda through 1999 Addenda," Revision 2, November 1, 1999.
8. Memorandum dated November 12, 1999, from Ashok C. Thadani, Office of Nuclear Regulatory Research, NRC, to John T. Larkins, ACRS, Subject: Generic Safety Issue 190, "Fatigue Evaluation of Metal Components for 60-Year Plant Life."

UNITED STATES
NUCLEAR REGULATORY COMMISSION
ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
WASHINGTON, D.C. 20555-0001

February 11, 2000

The Honorable Richard A. Meserve
Chairman
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555-0001

Dear Chairman Meserve:

SUBJECT: IMPORTANCE MEASURES DERIVED FROM PROBABILISTIC RISK ASSESSMENTS

During the 469th meeting of the Advisory Committee on Reactor Safeguards, February 3-5, 2000, we met with representatives of the NRC staff and Consumers Energy Company, Southern California Edison, and STP Nuclear Operating Company regarding the use of importance measures in risk-informing 10 CFR Part 50. We also had the benefit of the documents referenced.

This report responds to the Commission request in the December 17, 1999 Staff Requirements Memorandum that the ACRS evaluate the importance measures derived from Probabilistic Risk Assessments (PRAs) that are currently being contemplated for risk-informing Part 50 and, where appropriate, provide recommended additions or alternatives.

We believe that risk-informed decisions are best made using metrics, such as core damage frequency (CDF) or large, early release frequency (LERF), to evaluate the impact of decision options. There are, however, important situations in which this impact cannot be calculated easily. These include the risk-informed determination of special treatment requirements for structures, systems, and components (SSCs). The SSCs are first categorized according to their "importance," and then a decision is made regarding special treatment requirements for each category. The impact of these requirements on CDF and LERF is not quantified.

The risk-important categories of SSCs can be determined in a number of ways. The commonly used importance measures in risk-informed applications are the Fussell-Vesely (FV) and Risk Achievement Worth (RAW), although others, such as the Birnbaum measure, are occasionally used.

In evaluating the robustness of the SSC categorization, it is important to consider two facts: (1) depending on their definition, importance measures provide different insights regarding the

SSC importance and (2) the categorization is the result of an integrated decision by an expert panel that takes into account plant information in addition to the insights provided by the importance measures.

Since the determination of what is important, i.e., the definition of the importance measures, is somewhat arbitrary, these measures have limitations that include the following:

1. Importance measures are typically evaluated for individual SSCs. Yet, some decisions may affect groups of SSCs. While individual SSCs of a group may not be risk significant, the group itself may be.

2. Importance measures are strongly affected by the scope and quality of the PRA. For example, incomplete assessments of risk contributions from low-power and shutdown operations, fires, and human performance will distort the importance measures. Even with a full-scope, high-quality PRA, the importance measures have limitations, as discussed in our report of October 12, 1999.

3. The various categories of risk significance are determined by defining threshold values for the importance measures. For example, in some applications, a SSC is in the "high" risk-significant category when FV $\geq$ 0.005 and RAW $\geq$ 2.0, or FV $\geq$ 0.1, or RAW $\geq$ 100. In other applications, the numerical values are different. Some licensees choose to emphasize one measure over the other, e.g., RAW over FV. The relationship of these choices to CDF and LERF is unknown.

Given that the analysts have freedom in determining the criteria of risk significance, we were not surprised to find out that some licensees are implementing an approach that does not use importance measures at all. The Top Event Prevention Analysis (TEPA) utilizes success paths to determine what is important. We agree that this approach may have desirable defense-in-depth characteristics.

We note that the statistical literature also contains a number of methods for determining the sensitivity of a function, in our case the CDF or LERF, to its basic inputs, e.g., the failure rates. These methods allow us to investigate the issue of importance at a more elementary level (i.e., the parameter level) than that of FV and RAW (i.e., the SSC level).

As stated above, what really matters is the robustness of the SSC categorization that the expert panel produces through its integrated decisionmaking process that includes plant information in addition to the information provided by the importance measures. Since any choice of criteria for risk significance will likely involve some arbitrariness, we believe, as stated in our report of October 12, 1999, that the expert panel that determines the categorization of SSCs should be fully aware of the limitations and constraints of the chosen method. The panel should be provided with the results of sensitivity analyses, the results of alternative approaches, and an evaluation of the impact of these results on CDF and LERF. We recommend that a project be established to identify clearly the limitations of each proposed approach to importance

determination and to provide guidance to the expert panel on its deliberations regarding these matters. We believe that useful results can be produced in a short period.

Sincerely,

Dana A. Powers
Chairman

References:

1. Report dated October 12, 1999, from D. A. Powers, Chairman, ACRS, to Greta Joy Dicus, Chairman, U.S. Nuclear Regulatory Commission, Subject: Proposed Plans for Developing Risk-Informed Revisions to 10 CFR Part 50, "Domestic Licensing of Production and Utilization Facilities."
2. M.C. Cheok, G. W. Parry, and R. R. Sherry, "Use of Importance Measures in Risk-Informed Regulatory Applications," *Reliability Engineering and System Safety*, 60, 213-226, 1998.
3. W. E. Vesely, "Supplemental Viewpoints on the Use of Importance Measures in Risk-Informed Regulatory Applications," *Reliability Engineering and System Safety,* 60, 257-259, 1998.
4. W. E. Vesely, "Reservations on ASME Risk-Based Inservice Inspection and Testing: An Outlook to the Future," *Risk Analysis,* 18, 423-425, 1998.
5. K. N. Fleming, "Developing Useful Insights and Avoiding Misleading Conclusions from Risk Importance Measures in PSA Applications," PSA '96, Park City, Utah, September 29 - October 3, 1996, pp. 215-221, American Nuclear Society.
6. R. W. Youngblood, "Applying Risk Models to Formulation of Safety Cases," *Risk Analysis,* 18, 433-444, 1998.
7. R. A. White, R. B. Worrell, and D. P. Blanchard, "Using Top Event Prevention Analysis to Select a Safety-Significant Subset of Check Valves for Testing," Presented at TopSafe98, Valencia, Spain, April 1998.
8. C. E. Nierode, T. Willemson, R. B. Worrell, and D. P. Blanchard, "Use of Top Event Prevention Analysis to Select a Safety-Significant Subset of Air-Operated Valves for Testing," Proceedings of PSAM 4, New York City, September 13-18, 1998, pp. 1358-1363, A. Mosleh and R. A. Bari, Eds., Springer-Verlag, London.

UNITED STATES
NUCLEAR REGULATORY COMMISSION
ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
WASHINGTON, D.C. 20555-0001

February 11, 2000

Dr. William D. Travers
Executive Director for Operations
U.S. Nuclear Regulatory Commission
Washington DC 20555-0001

Dear Dr. Travers:

SUBJECT: REVISION OF APPENDIX K, "ECCS EVALUATION MODELS," TO 10 CFR PART 50

During the 469th meeting of the Advisory Committee on Reactor Safeguards, February 3-5, 2000, we reviewed the proposed final revision of Appendix K to 10 CFR Part 50. During this review, we had the benefit of discussions with representatives of the NRC staff and the Caldon Corporation. We also had the benefit of the documents referenced. We had previously commented on the proposed revision to Appendix K in a letter dated July 22, 1999.

The proposed final rule will permit a reduction in the conservatism of the reactor power level assumed for loss-of-coolant accident analysis by relaxing the requirement that a licensee assume 1.02 times licensed power for the Appendix K emergency core cooling system analysis. This rulemaking is in response to requests from licensees seeking credit in safety analyses for reduction in uncertainty of reactor power resulting from the use of highly accurate flow measurement systems. This rule change will allow licensees to credit use of such measurement systems and will avoid an expected large number of exemption requests, thereby reducing regulatory burden. Licensees are expected to pursue small power increases or other cost-saving changes to plant operating parameters through license amendment requests.

Recommendations

- The Commission should approve this rule change.
- The staff should provide guidance to licensees to account appropriately for power measurement uncertainty in their safety analyses.

Discussion

This rule is an example of allowing an appropriate reduction of conservatism in the regulations when the uncertainties that led to this conservatism can be shown to have been reduced. In principle, this is a straightforward matter. Implementation of the rule will require specific

guidance about the definition of uncertainties. For example, does "x% uncertainty" imply that there is some confidence level, such as 95%, that the deviations between actual and measured values are less than x% of the measured values? How are uncertainties in several values contributing to power calculation, such as temperatures and flowrate, to be combined? Answers to these questions as well as a suitable reference should be provided in the guidance to the licensees.

In our July 22, 1999 letter on this matter, we recommended that the staff evaluate the possible impact of the proposed rule on parts of the regulations other than Appendix K. Some changes to guidance documents may be necessary, as mentioned in the Statement of Considerations accompanying the rule revision.

Sincerely,

Dana A. Powers

Dana A. Powers
Chairman

References:

1. Note dated January 19, 2000, from Joe Donoghue, Office of Nuclear Reactor Regulation, NRC, to Paul A. Boehnert, ACRS, transmitting Final Rule: Revision of Part 50, Appendix K, "ECCS Evaluation Models."
2. ACRS letter dated July 22, 1999, from Dana A. Powers, Chairman, ACRS, to William D. Travers, Executive Director for Operations, NRC, Subject: Revision of Appendix K, "ECCS Evaluation Models," to 10 CFR Part 50.
3. Letter dated August 18, 1999, from William D. Travers, Executive Director for Operations, NRC, to Dana A. Powers, Chairman, ACRS, Subject: Staff Response to ACRS Letter of July 22, 1999, on Revision of Appendix K, "ECCS Evaluation Models," to 10 CFR Part 50.
4. Letter dated December 15, 1999, from David J. Modeen, Nuclear Energy Institute, to U.S. Nuclear Regulatory Commission, Subject: Transmittal of Comments on Proposed Change to 10 CFR Part 50, Emergency Core Cooling System Evaluation Models.
5. Caldon Comments on NRC Proposed Rule ECCS Evaluation Models dated December 15, 1999.
6. Letter dated October 26, 1999, from James H. McCarthy, Virginia Power, to U. S. Nuclear Regulatory Commission, Subject: Emergency Core Cooling System Evaluation Models, 10 CFR 50.
7. Letter dated December 9, 1999, from Mark J. Burzynski, Tennessee Valley Authority, to U. S. Nuclear Regulatory Commission, Subject: NRC - Emergency Core Cooling System Evaluation Models.
8. Letter dated December 14, 1999, from James A. Hutton Jr., PECO Nuclear, to U. S. Nuclear Regulatory Commission, Subject: Comments Concerning "Emergency Core Cooling System Evaluation Models."

9. Letter dated December 20, 1999, from Harry P. Salmon, Jr., New York Power Authority, to U.S. Nuclear Regulatory Commission, Subject: Comments on Proposed Rulemaking, Emergency Core Cooling System Evaluation Models.

UNITED STATES
NUCLEAR REGULATORY COMMISSION
ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
WASHINGTON, D.C. 20555-0001

February 14, 2000

The Honorable Richard A. Meserve
Chairman
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555-0001

Dear Chairman Meserve:

SUBJECT: IMPEDIMENTS TO THE INCREASED USE OF RISK-INFORMED REGULATION

The ACRS has long advocated the transition to a risk-informed regulatory system. Over the last several years, we have discussed potential impediments along the path toward risk-informed regulation.

This report responds to the Commission's request in the December 17, 1999 Staff Requirements Memorandum that the ACRS provide examples of impediments to the increased use of risk-informed regulation, an evaluation of the significance of these impediments, and, as appropriate, proposed solutions to identified problems. In our review, we had the benefit of the documents referenced.

There can be a variety of views on what is meant by risk-informed regulation. One view, for example, could be that we start over and redo the whole body of regulations making them risk informed without having a reactor type or design in mind. We believe that this should be considered on a non-urgent long time frame.

Another view could be that there exists a body of regulations and a population of light water reactor plants whose designs have resulted from meeting these regulations. As a result, risk-informed regulation would mean using risk insights gained from performing probabilistic risk assessments (PRAs) on the existing plants to modify the regulations holistically or in selected areas to make the regulations coherent, ensure that all requirements are necessary, and provide a focus on the more risk-significant issues.

In responding to the Commission's request, we take the latter view of what the agency intends in its efforts to risk inform the regulations.

In this respect, we identify a number of conditions that we believe hinder the progress of risk informing the regulations and implementing the changes for operating reactors. We have placed these "impediments" in two separate categories, "cultural/institutional" and "technical."

The cultural/institutional impediments are characterized as being related to attitudes, impressions, institutional or organizational barriers, processes, resource limits and similar such attributes. An important cultural/institutional impediment is the perception by licensees that they will need to expend substantial resources to update their PRAs to an acceptable level, provide additional staffing and resources to utilize and maintain the PRAs, and still have to comply with the current deterministic regulations. They fear that risk considerations will be add-ons to the existing regulatory system that will impose additional burdens.

There are many more cultural/institutional impediments than there are technical ones. We have chosen not to focus on the cultural/institutional impediments because it is our view that, as we risk inform the regulations in a technically defensible manner and consistently apply these regulations, most of the cultural/institutional impediments will fade away naturally with time. On the other hand, the technical impediments will not go away by themselves but will require significant effort and research for resolution.

We consider the more significant of the technical impediments to be:

1. PRA inadequacies and incompleteness in some areas.

2. The need to revisit risk-acceptance criteria.

3. Lack of guidance on how to implement defense in depth and on how to impose sufficiency limits.

4. Lack of guidance on the significance and appropriate use of importance measures.

5. Variation of PRA quality and scope and the need for Standards.

While we consider it important that efforts be undertaken to overcome these impediments, we believe that the state-of-the-art of PRA is sufficiently advanced that the agency can proceed with efforts to become more risk informed in its regulatory activities. However, it will be necessary to craft the regulations in a conservative manner to accommodate these shortcomings and in such a way that they can be easily evolved as improvements are made in the state-of-the-art of PRA. We also believe that the agency ought not to underestimate the risk analysis capabilities that will be needed to sustain a risk-informed regulatory system.

Our views on each of the technical impediments are discussed below:

1. <u>PRA Inadequacies and Incompleteness</u>

   Most of the current PRAs are inadequate for assessing risk contributions from fires, seismic events, human performance, organizational factors, and safety culture factors. They are incapable of assessing the lifetime average risk contribution from shutdown conditions. The reliability database is weak for passive components and "nonsafety-related" systems and components.

2. Risk Acceptance Criteria

It is necessary to have risk acceptance criteria applicable to individual licensees in a risk-informed regulatory system. The initial efforts to risk inform the regulatory activities have utilized two metrics for risk acceptance - mean values of core damage frequency (CDF) and large, early release frequency (LERF). The values for CDF and LERF used in Regulatory Guide 1.174 are consistent with the Commission's safety goals. These safety goals were, however, originally intended to be goals (i.e., some things to strive for) for the average risk status of the population of plants as a whole. It is generally recognized that safety goals are not risk acceptance values that would, for example, be surrogates for adequate protection.

In a risk-informed regulatory system, it is necessary to have risk acceptance limits. If we are to have limits on CDF and LERF that are "consistent" with "adequate protection," we believe these would differ from those in Regulatory Guide 1.174. It is important at this stage of risk-informing the regulations that quantitative limits be incorporated into an expanded definition of adequate protection.

3. Defense-in-Depth

According to the Commission's White Paper (SECY-98-144):

> Defense-in-Depth is an element of NRC's Safety Philosophy that employs successive compensatory measures to prevent accidents or mitigate damage if a malfunction, accident, or naturally caused event occurs at a nuclear facility.

If defense-in-depth is viewed as measures taken to compensate for the PRA inadequacies and uncertainties, then there is a need for guidance to help quantify how many compensatory measures are necessary and how good these have to be.

4. Importance Measures

We have noted that risk-informed decisions are often based on categorizing structures, systems, and components according to their importance in influencing changes to CDF and LERF. As discussed in our report on importance measures, there is a need for guidance on the appropriate interpretation and use of importance measures.

5. Need for Standards/PRA Quality and Scope

Most PRAs for existing reactors were developed in response to Generic Letter 88-20 and Supplement 4 requesting the individual plant examination (IPE) and individual plant examination for external events (IPEEEs). It has been noted that there is much variation in the scope and quality of these PRAs.

NRC has requested the American Society of Mechanical Engineers and the American Nuclear Society to develop Standards to ensure that the technical quality of PRAs is sufficient to support the regulatory review and approval of licensee risk-informed

applications. We believe that development of appropriate PRA Standards is important to risk-informing the regulations. However, we believe it is important that standards not stifle the continuing improvement of PRA methods.

We believe that the quality of any PRA is reflected in the quantified uncertainty distribution. It is important that the Standards include guidance on the appropriate determination of uncertainties (epistemic and aleatory) and the NRC staff needs guidance on how to consistently use these in the decisionmaking process.

As stated above, even though impediments exist, the agency has the capabilities necessary to make significant progress in developing and implementing risk-informed regulations.

Sincerely,

Dana A. Powers

Dana A. Powers
Chairman

References:

1. Staff requirements Memorandum dated December 17, 1999.
2. U.S. Nuclear Regulatory Commission, Regulatory Guide 1.174, "An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis," July 1998.
3. U.S. Nuclear Regulatory Commission, Policy Statement, "Safety Goals for the Operations of Nuclear Power Plants," 10 CFR Part 50, August 21, 1986.
4. SECY-98-144, Subject: White Paper on Risk-Informed and Performance-Based Regulation, dated June 22, 1998.
5. Letter to William D. Travers, Executive Director for Operations from Dana A. Powers, Chairman, Advisory Committee on Reactor Safeguards, Subject: Proposed ASME Standard for Probabilistic Risk Assessment for Nuclear Power Plant Applications (Phase 1), dated March 27, 1999.
6. Report to Richard A. Meserve, Chairman, NRC from Dana A. Powers, Chairman, Advisory Committee on Reactor Safeguards, Subject: Importance Measures Derived from Probabilistic Risk Assessment dated February , 2000.
7. U.S. Nuclear Regulatory Commission, Generic Letter 88-20, dated November 23, 1988, Subject: Individual Plant Examination for Severe Accident Vulnerabilities.
8. U.S. Nuclear Regulatory Commission, Generic Letter 88-20, Supplement 4, dated June 28, 1991,Subject: Individual Plant Examination for External Events.

UNITED STATES
NUCLEAR REGULATORY COMMISSION
ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
WASHINGTON, D.C. 20555-0001

March 10, 2000

The Honorable Richard A. Meserve
Chairman
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555-0001

Dear Chairman Meserve:

SUBJECT: PROPOSED FINAL REGULATORY GUIDE 1.XXX, "ASSESSING AND MANAGING RISK BEFORE MAINTENANCE ACTIVITIES AT NUCLEAR POWER PLANTS"

During the 470th meeting of the Advisory Committee on Reactor Safeguards, March 1-4, 2000, we reviewed the proposed final Regulatory Guide 1.XXX, "Assessing and Managing Risk Before Maintenance Activities at Nuclear Power Plants," which was developed to supplement Regulatory Guide 1.160, "Monitoring the Effectiveness of Maintenance at Nuclear Power Plants." Regulatory Guide 1.XXX endorses the revised Section 11, "Assessment of Risk Resulting from Performance of Maintenance Activities," of NUMARC 93-01. During our review, we had the benefit of discussions with representatives of the NRC staff and of the documents referenced.

Recommendations

1. The proposed final Regulatory Guide 1. XXX, which is a supplement to Regulatory Guide 1.160 should be approved for industry use.

2. We support the staff's endorsement of Section 11 of NUMARC 93-01 guidance document.

Discussion

During our 467th meeting, November 4-6, 1999, we were briefed by the NRC staff and the Nuclear Energy Institute (NEI) on the proposed Revision 3 to Regulatory Guide 1.160 and were told that there were minor issues to be resolved, including the definition of unavailability. Subsequently, the NRC staff and NEI have resolved these issues and have agreed upon the definition of unavailability. We note that this definition reflects the contributions to unavailability of planned and unplanned down times only. This definition is acceptable for the purposes of Regulatory Guide 1.XXX.

During our discussion of Regulatory Guide 1.XXX, the staff identified additional changes proposed by NEI to Sections 11.3.2 and 11.3.8 of NUMARC 93-01. These changes are to

address temporary alterations that are necessary for maintenance during power operations. For such temporary alterations, no review would be required under 10 CFR 50.59 unless the alterations are expected to be in effect for more than 90 days during power operation.

The staff states that temporary alterations, which are in effect less than 90 days, will be assessed under the requirements of 10 CFR 50.65(a)(4). To clarify the need for these assessments, the staff proposes to add the following paragraph to the implementation Section of Regulatory Guide 1.XXX:

> The assessment does not relieve the licensee from obligations to its license or the regulations. The exemption requirements and 10 CFR 50.90 remain in effect. The intent is to eliminate overlapping requirements for assessments which could be considered to exist under 10 CFR 50.65(a)(4) and 10 CFR 50.59. This clarification applies to temporary alterations directly related to and required to support the specific maintenance activity being assessed.

We agree that licensees should not be required to perform duplicate assessments for temporary alterations during maintenance activities. We agree with the proposed additional paragraph. Regulatory Guide 1.XXX should be approved for industry use.

Sincerely,

Dana A. Powers

Dana A. Powers
Chairman

References :

1. Memorandum dated February 16, 2000, from Theodore R. Quay, Office of Nuclear Reactor Regulation, NRC, to John T. Larkins, ACRS, Subject: Request for Review of Regulatory Guidance for 10 CFR 50.65, The Maintenance Rule, transmitting:
   a. Regulatory Guide 1.XXX, "Assessing and Managing Risk Before Maintenance Activities at Nuclear Power Plants"
   b. Final Section 11 of NUMARC 93-01, "Assessment of Risk Resulting from Performance of Maintenance Activities," February 11, 2000.

2. NUMARC 93-01, Revision 2, Nuclear Energy Institute, "Industry Guideline for Monitoring the Effectiveness of Maintenance at Nuclear Power Plants," (line-in/line-out version), April 1996.
3. U. S. Nuclear Regulatory Commission, Regulatory Guide 1.160, Revision 2, "Monitoring the Effectiveness of Maintenance at Nuclear Power Plants," March 1997.

UNITED STATES
NUCLEAR REGULATORY COMMISSION
ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
WASHINGTON, D.C. 20555-0001

March 13, 2000

The Honorable Richard A. Meserve
Chairman
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555-0001

Dear Chairman Meserve:

SUBJECT: REPORT ON THE SAFETY ASPECTS OF THE LICENSE RENEWAL APPLICATION FOR THE OCONEE NUCLEAR STATION, UNITS 1, 2 AND 3

During the 470th meeting of the Advisory Committee on Reactor Safeguards, March 1-4, 2000, we completed our review of Duke Energy Corporation's application for license renewal of the Oconee Nuclear Station, Units 1, 2 and 3 and the related Final Safety Evaluation Report (FSER). Our review included a plant visit and four meetings, one of which was conducted in Clemson, South Carolina. We had the benefit of insights gained from two meetings concerning generic license renewal issues and the review of another license renewal application. We provided an interim letter dated September 13, 1999, concerning the Oconee license renewal application. During these reviews, we had the benefit of the documents referenced.

Conclusion

On the basis of our review of Duke's application, the staff's FSER, and the resolution of the open and confirmatory items identified in the June 1999 Safety Evaluation Report (SER), we conclude that:

- Duke has properly identified the structures, systems, and components (SSCs) that are subject to aging management programs according to the requirements of 10 CFR Part 54.

- Possible aging mechanisms associated with passive, long-lived SSCs have been appropriately identified.

- The programs instituted to manage aging-related degradation of the identified SSCs are appropriate and provide reasonable assurance that Oconee Units 1, 2 and 3 can be operated in accordance with their current licensing basis for the period of the extended license without undue risk to the health and safety of the public.

Background and Discussion

This report is intended to fulfill the requirement of 10 CFR 54.25 that each license renewal application be referred to the ACRS for a review and report. Duke requested renewal of the operating licenses for the Oconee Units 1, 2 and 3 for a period of 20 years beyond the current license term. The FSER documents the results of the staff's review of information submitted by Duke, including those commitments that were necessary to resolve open and confirmatory items identified by the staff in its SER. The staff's review included the verification of the completeness of the identification and categorization of the SSCs considered in the application; the validation of the integrated plant assessment process; the identification of the possible aging mechanisms associated with each passive long-lived component; and the adequacy of the aging management programs.

In the SER, the staff identified a number of open and confirmatory items. The staff and Duke have now resolved all open items and addressed all confirmatory items, in part through additional commitments made by Duke. The Duke commitments will become a part of the plant's licensing basis and will be added to the Oconee Final Safety Analysis Report (FSAR). This will make the commitments enforceable.

Several of the open items, such as the completeness of the methodology used to identify SSCs that are within the scope of Part 54 and the consideration of the effects of the reactor coolant environment on fatigue life, may have generic implications for future license renewal applications.

Because Oconee was licensed before NUREG-75/087, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants," was issued in September 1975, the safety-related SSCs at Oconee do not completely bound the set of SSCs that are relied upon to be functional during and following design basis events. Consequently, nonsafety-related components that are relied upon to perform safety-related functions are within the scope of Part 54. As noted in our interim letter, this is a generic issue for older plants. The process of identifying these additional SSCs without expanding the current licensing basis of Oconee required significant interaction between the staff and the licensee.

In accordance with the license renewal scoping criteria specified in 10 CFR 54.4 (a), the staff identified a set of additional events that had not been considered in Duke's license renewal application. Although these events were not part of the original FSAR accident analysis, Duke was asked to perform a plant-specific evaluation. We agree with the staff determination that these events should be considered in the analysis of scope. Duke evaluated these events to identify additional SSCs that should be included within the scope of license renewal. This evaluation did not identify any additional SSCs and provides further evidence that SSCs within the scope of 10 CFR Part 54 have been appropriately identified.

Insulated cables in localized areas in the Oconee containment have been identified in station problem reports as exhibiting accelerated thermal and radiation-induced aging effects due to adverse environments. Where the design and installation conditions responsible for the accelerated aging have not been corrected, the staff requested that an aging management program be instituted as part of the license renewal application. The staff also requested that

an aging management program be instituted for medium-voltage cables located in trenches or buried in the ground, where the cables are exposed to moisture.

Duke responded by instituting an Insulated Cables Aging Management Program that includes cables within the scope of license renewal that are installed in locations with adverse environments and could be subject to aging effects from radiation, heat, or moisture. The only insulated cables excluded from this program are those covered by the Environmental Qualification Program. The Insulated Cables Aging Management Program identifies inspections, parameters to be monitored, and corrective actions to be taken in accordance with the requirements of 10 CFR Part 50, Appendix B. We concur with the staff's conclusion that this comprehensive program resolves this open item.

A number of SER open items involved reactor vessel internal components. Aging effects to be addressed included changes in dimensions due to void swelling, cracking in reactor vessel internal noncast austenitic stainless steel components, cracking of baffle-former bolts, embrittlement of cast austenitic stainless steel components, thermal embrittlement of vent valves, and reduction in fracture toughness. Duke has addressed these open items in the Oconee Reactor Vessel Internals Aging Management Program (RVIAMP). This program includes participation in industry initiatives to investigate these aging effects, inspections, and reports to be provided to the NRC on a periodic basis. A final report will be submitted by Duke to the NRC near the end of the initial license period for Unit 1. The final report will contain the test results from the Babcock & Wilcox Owners Group's RVIAMP and the recommended inspection program for Oconee. On the basis of this information, Duke will implement an aging management program for the reactor vessel internals. We find the proposed program comprehensive and adequate for resolving the reactor vessel internals open items.

Duke committed to implementing a plant-specific fatigue monitoring program in which it will use correlations published in NUREG/CR-5704 to calculate environmental penalties at the high fatigue-usage locations identified in NUREG/CR-6260 to assess the effects of the reactor coolant environment on the fatigue life of components and piping. The correlations reflect data developed to resolve Generic Safety Issue (GSI) 190, "Fatigue Evaluation of Metal Components for 60-Year Plant Life." We concur with the staff's conclusion that Duke's proposed program is an acceptable plant-specific approach for resolving GSI-190 concerns.

The Oconee license renewal application described the process and the results of a time-limited aging analysis to demonstrate the adequacy of prestressing forces in the containment post-tensioning tendons during the period of extended operation. The staff requested additional information needed to support this demonstration. Duke has responded by proposing a Post-Tensioning System Loss of Prestress Aging Management Program to identify and correct degradation of the post-tensioning system prior to an unacceptable loss of prestress. This program implements the requirements of the American Society of Mechanical Engineers (ASME) Code Section XI, Subsection IWL, for in-service inspection, trending, and repair or replacement activities of the post-tensioning systems of concrete containments. We concur with the staff's assessment that the implementation of this program adequately resolves this open item.

As Oconee Units 1, 2 and 3 age, inspection and operating experience may prompt significant adjustments to their aging management programs. Duke has committed to document in the

FSAR Supplement that all components subject to an aging management program fall under the requirements of its Problem Investigation Process corrective action program. Furthermore, the staff has required that Duke include in the Oconee FSAR the license renewal application commitments that the staff relied upon to conclude that aging effects will be adequately managed for the period of extended operation. These steps ensure that future changes to the aging management programs can be controlled under the 10 CFR 50.59 process.

The staff has performed a comprehensive and thorough review of Duke's application. The additional programs required by the staff are appropriate and sufficient. Current regulatory requirements and existing Duke programs provide adequate management of aging-induced degradation for those SSCs within the scope of the license renewal rule.

Mr. John D. Sieber did not participate in the Committee's deliberations regarding this matter.

Dr. William J. Shack did not participate in the Committee's deliberations regarding aging-induced degradation.

Sincerely,

Dana A. Powers

Dana A. Powers
Chairman

References:

1. Letter dated February 3, 2000, from David B. Matthews, Office of Nuclear Reactor Regulation, to William R. McCollum, Jr., Duke Energy Corporation, Subject: Final Safety Evaluation Report.
2. ACRS letter dated September 13, 1999, from Dana A. Powers, Chairman, ACRS, to William D. Travers, Executive Director for Operations, NRC, Subject: Interim Letter Related to the License Renewal of Oconee Nuclear Station.
3. Letter dated June 16, 1999, from David B. Mathews, Office of Nuclear Reactor Regulation, to William R. McCollum, Jr., Duke Energy Corporation, Subject: Oconee Nuclear Station, Units 1, 2 and 3, License Renewal Safety Evaluation Report.
4. Letter dated April 26, 1999, from Christopher I. Grimes, Office of Nuclear Reactor Regulation, to David J. Firth, B&W Owners Group, Subject: Acceptance for Referencing of Generic License Renewal Program Topical Report Entitled, "Demonstration of the Management of Aging Effects for the Reactor Vessel," BAW-2251, June 1996.
5. Letter dated June 27, 1996, from D. K. Croneberger, B&W Owners Group, to Document Control Desk, NRC, Subject: Submittal of BAW-2251, "Demonstration of the Management of Aging Effects for the Reactor Vessel," June 1996.
6. U. S. Nuclear Regulatory Commission, Office of Nuclear Reactor Regulation Office Letter Transmittal No. 805, "License Renewal Application Review Process," June 19, 1998.
7. U. S. Nuclear Regulatory Commission Safety Evaluation Report (SER) related to the Babcock & Wilcox (BAW) Topical Report 2251, "Demonstration of the Management of Aging Effects for the Reactor Vessel," April 26, 1999.

8. U. S. Nuclear Regulatory Commission, NUREG/CR-5704, "Effects of LWR Coolant Environments on Fatigue Design Curves of Austenitic Stainless Steels," April 1999.
9. U. S. Nuclear Regulatory Commission, NUREG/CR-6260, "Application of NUREG/CR-5999 Interim Fatigue Curves to Selected Nuclear Power Plant Components," March 1995.

UNITED STATES
NUCLEAR REGULATORY COMMISSION
ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
WASHINGTON, D.C. 20555-0001

March 13, 2000

The Honorable Richard A. Meserve
Chairman
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555-0001

Dear Chairman Meserve:

SUBJECT: SECY-00-0007, "PROPOSED STAFF PLAN FOR LOW POWER AND SHUTDOWN RISK ANALYSIS RESEARCH TO SUPPORT RISK-INFORMED REGULATORY DECISION MAKING"

During the 468th and 470th meetings of the Advisory Committee on Reactor Safeguards, December 2-4, 1999, and March 1-4, 2000, we discussed the NRC staff's Low-Power and Shutdown Risk Perspectives Report. Our Subcommittee on Reliability and Probabilistic Risk Assessment met on November 18, 1999, to discuss this matter. We had the benefit of the documents referenced.

Conclusions and Recommendations

1. The staff should evaluate the adequacy of its analytical tools for independently assessing the risk significance of plant configurations during low-power and shutdown (LPSD) operations, especially during plant transitions. If the staff's analytical tools are found to be inadequate or lacking in certain areas, the staff should develop a course of action to address these inadequacies.

2. We agree with the staff's proposed continued support to the American Nuclear Society for developing an industrial standard in the area of LPSD risk assessment.

3. Assessment of human performance during LPSD operations and transition periods should be included in the ATHEANA (A Technique for Human Event Analysis) project. Human actions that initiate abnormal events should be given special attention.

Discussion

In reports dated April 18, 1997 and June 11, 1999, we commented on the importance of evaluating the significance of LPSD risks to the development of risk-informed regulations and on the need for the NRC to develop robust methods to assess these risks. In our report dated June 11, 1999, we noted that the analytical tools developed and used by licensees for

configuration risk management during outages are valuable. The tools used by licensees include the Outage Risk Assessment Management (ORAM™) software, the Equipment Out Of Service (EOOS™) methodology, Safety Monitor™, and risk monitors. These tools are based on combinations of defense-in-depth strategies and PRA insights. The extent to which PRA is used varies.

Although we are encouraged by the increased use of such methodologies by the licensees, we believe that the staff should have the capability to independently evaluate licensee analyses and activities. It is not apparent that the senior reactor analysts and inspection staff have adequate analytical tools to independently evaluate management of LPSD risk. The staff should evaluate the adequacy of its tools in comparison with those used by the industry. If the staff's analytical tools are found to be inadequate or lacking in certain areas, the staff should develop a course of action to address these inadequacies.

The first phase of the staff's program to evaluate LPSD risk resulted in a report entitled "Low Power and Shutdown Risk: A Perspectives Report." The staff's conclusions in that report substantiated our concern that the risks from LPSD operations can be comparable to those from power operations. The report also confirmed that human errors are a significant contributor to risk during LPSD operations. The report further noted that the LPSD risks were high even after configuration risk management strategies were implemented. In addition, it is not apparent to what extent the risks during plant transition periods were assessed. The NRC-sponsored LPSD risk studies did not investigate risk during plant transitions in detail.

A major conclusion in the report is that human performance issues are especially important during LPSD operations and related transition periods. The operational experience and the expert judgments cited in the report indicate that the error-forcing contexts during LPSD operations may be quite different from those anticipated during power operations. In less familiar situations, operators may have to function in a "knowledge-based" mode. This is contrary to the thinking that evolved after the 1979 accident at Three Mile Island. It is now believed that operator performance should be "rule-based," using procedures, rather than "knowledge-based." Some of the reasons for the differences in error-forcing contexts are unfamiliar plant configurations, unfamiliar indications, and limited procedural guidance.

When human performance during LPSD operations is discussed, the emphasis is usually on recovery actions. As noted in our report on ATHEANA dated December 15, 1999, there is a need to investigate human performance during normal activities that may cause a plant event. This is particularly important for LPSD operations because of the multiplicity of tasks that operators perform, the increased volume of concurrent and competing work activities, and the large number of different plant configurations with equipment out of service. Emphasis should be given to investigating human performance during transitions between plant operational states. We believe that progress in addressing these human performance issues is achievable within the context of the ATHEANA project.

We agree with the staff's proposed continued support for the development of an industrial standard by the American Nuclear Society in the area of LPSD risk assessment.

We look forward to working with the staff as it proceeds with resolving these important issues.

Sincerely,

Dana A. Powers
Chairman

References:

1. Memorandum dated January 12, 2000, from William D. Travers, Executive Director for Operations, NRC, to the Commissioners, Subject: SECY-00-0007, Proposed Staff Plan for Low Power and Shutdown Risk Analysis Research to Support Risk-Informed Regulatory Decision Making.
2. ACRS report dated June 11, 1999, from Dana A. Powers, Chairman, ACRS, to Shirley Ann Jackson, Chairman, NRC, Subject: Development of a Low-Power and Shutdown Risk Assessment Program.
3. ACRS 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.
4. ACRS letter dated December 15, 1999, from Dana A. Powers, Chairman, ACRS, to William D. Travers, Executive Director for Operations, NRC, Subject: NUREG-1624, Revision 1, "Technical Basis and Implementation Guidelines for a Technique for Human Event Analysis (ATHEANA)."
5. SAND99-1815, Sandia Report, "Summary of Information Presented at an NRC-Sponsored Low-Power Shutdown Public Workshop, April 27, 1999, Rockville, Maryland," July 1999.
6. U. S. Nuclear Regulatory Commission, NUREG/CR-6143, "Evaluation of Potential Severe Accidents During Low Power and Shutdown Operations at Grand Gulf, Unit 1," July 1995.
7. U. S. Nuclear Regulatory Commission, NUREG/CR-6144, "Evaluation of Potential Severe Accidents During Low Power and Shutdown Operations at Surry, Unit 1," October 1995.

UNITED STATES
NUCLEAR REGULATORY COMMISSION
ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
WASHINGTON, D.C. 20555-0001

March 13, 2000

Dr. William D. Travers
Executive Director for Operations
U.S. Nuclear Regulatory Commission
Washington DC 20555-0001

Dear Dr. Travers:

SUBJECT: PROPOSED RESOLUTION OF GENERIC ISSUE B-17, "CRITERIA FOR SAFETY-RELATED OPERATOR ACTIONS," AND GENERIC ISSUE 27, "MANUAL VS. AUTOMATED ACTIONS"

During the 470th meeting of the Advisory Committee on Reactor Safeguards, March 1-4, 2000, we reviewed the proposed resolution of Generic Issue (GI) B-17, "Criteria for Safety-Related Operator Actions," and GI 27, "Manual vs. Automated Actions." During this review, we had the benefit of discussions with representatives of the NRC staff and of the documents referenced. The Committee had previously reviewed a proposed resolution approach to GI B-17 in 1995.

Conclusions

- The Committee agrees with the staff's resolution approach for these Generic Issues.
- The Committee would like to review the staff's evaluation of the ANSI/ANS Standard ANSI/ANS-58.8-1994 before it is endorsed.

Discussion

GI B-17 was formulated in 1978, before the TMI accident, to address a concern about whether certain time-critical-safety-related operator actions should be automated. A time criterion was to be established as a way to resolve this GI. In 1981, GI-27 was formulated to address questions as to whether certain safety actions should be automated or if manual operator actions were acceptable. Because they address nearly identical issues, these GI's were combined.

The staff position is that the regulatory actions that have been implemented since the 1979 TMI accident provide adequate grounds for closing GIs B-17 and 27. These regulatory actions have included: enhanced operator training and licensing requirements, including use of plant-specific simulators; improved training based on the Systems Approach to Training; establishment of minimum plant staffing levels; use of symptom-based emergency operating procedures; and

completion of plant IPEs. The argument is also made that any new or revised regulatory activities to address this issue (i.e., automation of human actions) would not be cost effective or substantially increase public health and safety, given the existing regulations. We support the staff's positions in this regard.

In 1995, the staff proposed to close out GI B-17 by the endorsement of an American National Standard ANSI/ANS-58.8-1994, "Time Response Design Criteria for Safety-Related Operator Actions." The ACRS reviewed this matter during its 426th meeting, November 2-4, 1995, and advised against the use of this Standard to close out B-17 because the technical basis for the Standard was not available for review. The staff subsequently agreed to consider alternatives to the time-criterion approach advocated in this Standard for close out of this issue.

It is our understanding that the NRC staff may endorse ANSI/ANS-58.8-1994 for licensees to adopt in seeking relief from the use of automated equipment in transient situations by reliance on manual operator actions. Our concern regarding the need for an adequate review of the ANSI Standard has not been resolved. We would like to review the staff's evaluation of the Standard before it is endorsed.

Sincerely,

Dana A. Powers

D. A. Powers
Chairman

References:

1. Letter dated February 17, 2000, from Charles E. Rossi, Office of Nuclear Regulatory Research, NRC, to John T. Larkins, ACRS, Subject: Proposed Resolution of Generic Issues B-17, "Criteria for Safety-Related Operator Actions," and GI-27, "Manual Vs. Automated Actions."
2. ACRS Letter dated November 14, 1995, from T. S. Kress, Chairman, ACRS, to James M. Taylor, Executive Director for Operations, NRC, Subject: Proposed Final Regulatory Guide 1.164, "Time Response Design Criteria for Safety-Related Operator Actions," to Resolve Generic Safety Issue B-17."
3. American Nuclear Society, American National Standard, ANSI/ANS 58.8-1994, "Time Response Design Criteria for Safety-Related Operator Actions," August 23,1994.

UNITED STATES
NUCLEAR REGULATORY COMMISSION
ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
WASHINGTON, D.C. 20555-0001

March 13, 2000

MEMORANDUM TO: William D. Travers
Executive Director for Operations

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

SUBJECT: PROPOSED FINAL AMENDMENT TO 10 CFR 50.72, "IMMEDIATE NOTIFICATION REQUIREMENTS FOR OPERATING NUCLEAR POWER REACTORS," AND 10 CFR 50.73, "LICENSEE EVENT REPORT SYSTEM"

During the 470th meeting of the Advisory Committee on Reactor Safeguards, March 1-4, 2000, the Committee heard a status report concerning the proposed final amendment to 10 CFR 50.72 and 50.73. The Committee had reviewed and commented on a previous draft of the proposed amendment during the 460th ACRS meeting on March 10-13, 1999. During its 469th meeting, February 3-5, 2000, the Committee reviewed the proposed final amendment. Subsequently, the Committee has decided not to comment further on this matter.

References:

1. ACRS letter dated March 23, 1999, from Dana A. Powers, Chairman, ACRS, to William D. Travers, Executive Director for Operations, NRC, Subject: Proposed Amendment to 10 CFR 50.72, Immediate Notification and 50.73, Licensee Event Reporting System.
2. Letter dated April 19, 1999, from William D. Travers, Executive Director for Operations, NRC, to Dana A. Powers, Chairman, ACRS, Subject: Proposed Rulemaking to Modify the Reactor Event Reporting Requirements in 10 CFR 50.72 and 50.73.
3. Letter dated September 17, 1999, from James W. Davis, NEI, to the Secretary of the NRC, Subject: Proposed Rule for Reporting Requirements for Nuclear Power Reactors.
4. Memorandum dated June 15, 1999, from Annette Vietti-Cook, Secretary, NRC, to William D. Travers, Executive Director for Operations, Subject: Staff Requirements - SECY-99-119 - Rulemaking to Modify the Event Reporting Requirements for Power Reactors in 10 CFR 50.72 and 50.73.
5. Memorandum dated December 30, 1999, from David B. Matthews, Office of Nuclear Reactor Regulation, NRC, to NRC Office Directors, Subject: Review and Comments on Commission Paper Entitled "Rulemaking to Modify the Event Reporting Requirements for Power Reactors in 10 CFR 50.72 and 50.73."

cc: A. Vietti-Cook, SECY
J. Blaha, OEDO
W. Ott, OEDO
S. Collins, NRR
D. Matthews, NRR
C. Carpenter, NRR
D. Allison, NRR

UNITED STATES
NUCLEAR REGULATORY COMMISSION
ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
WASHINGTON, D.C. 20555-0001

March 15, 2000

The Honorable Richard A. Meserve
Chairman
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555-0001

Dear Chairman Meserve:

SUBJECT: REVISED REACTOR OVERSIGHT PROCESS

During the 469th and 470th meetings of the Advisory Committee on Reactor Safeguards, February 3-5 and March 1-4, 2000, we discussed technical aspects of the revised reactor oversight process, including the technical adequacy of current and proposed performance indicators (PIs) and the significance determination process (SDP).

This report responds to the Commission request in the December 17, 1999 Staff Requirements Memorandum, that the ACRS evaluate the extent to which the PIs, collectively, provide meaningful insights into those areas of plant operations that are most important to safety. Our Subcommittee on Plant Operations met on January 20, 2000, to discuss these matters. We also had the benefit of the documents referenced.

Conclusions and Recommendations

1. The Revised Reactor Oversight Process (RROP) makes NRC assessments and actions more objective, predictable, and understandable to both the public and industry.

2. Although the RROP is a work in progress, it is ready for initial implementation at all power reactors. Further adjustments in the process may be needed as more experience is gained with a larger base of plants. Because changes are expected after the initial implementation, staff should look for methods to implement the process in ways that it can be easily changed.

3. The choices of the PIs and the associated thresholds remain controversial. Alternative views of ACRS members regarding the choice of thresholds are offered in the discussion.

4. The SDP is incomplete. Further development of this process and the analytical tools it uses is required for full implementation.

5. Additional PIs will be needed for full and effective implementation of the RROP. In particular, PIs are needed to characterize the licensee's problem identification and corrective action program (CAP), human performance, safety culture, and low-power and shutdown operations.

Discussion

The RROP pilot program was completed in November 1999 and lessons learned have resulted in changes that have improved the process prior to its initial implementation at all power reactors. The process is intended to ensure that plants continue to perform at an acceptable level and to provide early warning of adverse trends.

We recognize that the RROP is a work in progress and that certain aspects could not be fully exercised and evaluated during the 6-month pilot program. We agree that the overall process, the concept of the cornerstones, and the associated framework are sound. The new process will make NRC assessments more objective, predictable, and understandable to both the public and industry and should be approved for initial implementation at all plants. The staff has stated that continued development and implementation of the process will not adversely affect initial implementation. The staff plans to assess the effectiveness of the entire process after the first year of initial implementation.

The staff has selected a set of PIs to be used as part of the RROP, which is intended to be risk informed and performance based. The PIs are defined in the expectation that they are correlated with risk, even though in some cases the implied correlation cannot be explicitly defined or quantified. Without such an explicit connection to risk, it is difficult to determine which and how many PIs are sufficient or to determine quantitative threshold values. An added practical constraint to the selection of a set of PIs is the limited ability of the staff to obtain data from the licensees.

Recognizing that there are unavoidable limitations in the chosen set of PIs, the staff has developed a baseline inspection program for each cornerstone to complement and supplement the PIs. We agree with the staff that the technical adequacy of the proposed PIs should be evaluated in the context of the overall assessment process.

Another key element of the RROP is the licensee's problem identification and CAP. A basic tenet of the RROP is that the licensee's CAP should be relied upon to correct issues that do not result in crossing safety performance thresholds. This is based on the assumption that the improved overall industry performance over the past 10 years has demonstrated the general robustness of the CAPs. Confirmation of this assumption for individual plants requires that NRC periodically assess the effectiveness of each CAP as part of the baseline inspection program.

We believe that additional PIs will be needed for full and effective implementation of the program. In particular, PIs are needed to characterize the licensee's problem identification and CAP, human performance, safety culture, and low-power and shutdown operations.

The proposed green-white PI thresholds have been selected as the 95th percentile of the values for the whole population of operating plants. Some ACRS members believe that this approach has led to the selection of PI thresholds that are too high to provide early warning of adverse trends in performance. The proposed values are such that most indicators will always be in the green, therefore, the PIs may not contribute meaningful information to the oversight process.

Because performance in the green may be interpreted as good performance, there will be a reduced incentive for improved performance by the licensees.

Some ACRS members find the staff's approach to the selection of the green-white thresholds acceptable. Current industry practices and regulatory requirements, along with the previous inspection and oversight process, have resulted in acceptable overall industry performance. Therefore, the set of current values for the PIs does represent the range of acceptable performance values, and the 95th percentile values are to identify outliers. Obviously there is some degree of arbitrariness involved, but it is an acceptable choice for initial implementation.

Some ACRS members believe that there is a fundamental flaw with the process of selecting the PI thresholds. As noted in our report dated June 10, 1999, a lesson from the probabilistic risk assessments and Individual Plant Examinations is that the risk profile of each plant is unique. The PIs and the thresholds should reflect this finding and should be plant specific. This means that the threshold for a specific PI should be selected from a distribution of values that reflects past performance with respect to this PI at that plant. A typical value that is usually selected is the 95th percentile of this plant-specific curve. The current process, however, selects the thresholds from distributions that include plant-to-plant variability. A plant-to-plant variability curve represents the distribution of the past values of a PI across all plants. The selection of the 95th percentile of these distributions could have two significant consequences. First, the thresholds are too high for the plants whose past performance placed them below the chosen threshold value. Second, the few plants with past performance above the selected threshold value may be in the "white" category without credit for other compensating features. This situation would create pressure on those licensees to "improve" their performance with respect to the PI, thereby ratcheting up the expected performance of the plant.

The same ACRS members believe that the establishment of plant-specific thresholds is feasible. The staff has agreed that, ideally, plant-specific thresholds would be desirable, but that they cannot be established at this time. An example of such an exercise, however, was the implementation of the maintenance rule and the proposed plant-specific performance criteria by the licensees. The staff has collected and published plant-specific data, including those from studies by the former Office for Analysis and Evaluation of Operational Data, e.g., NUREG/CR-5500, Volumes 4-8, and associated updates. Alternatively, it may be possible to identify groups of plants with similar design and operational characteristics that could share the same PI threshold values.

Some ACRS members are concerned that the high PI thresholds focus on equipment performance only. The staff has stated that cross-cutting issues involving human performance and safety culture will manifest themselves through the PIs or the baseline inspections. The baseline inspections may lag adverse human performance trends and not trigger action until some PI thresholds are exceeded. PI thresholds do not appear to provide timely warning of negative trends.

The SDP is designed to provide guidance for the risk characterization of inspection program findings so that the overall licensee performance assessment process can compare and evaluate the findings on a significance scale similar to that established for PIs. The SDP is still incomplete. Findings from workshops and lessons learned on the pilot program have not been

accounted for in the SDP. Because of limitations in the staff's analytical tools, very approximate risk assessment methods are used for some SDP evaluations.

It is expected that the overwhelming majority of SDP findings will be "green." We are concerned that such an outcome could mask programmatic problems. For example, weakness in a maintenance program that was manifested by the failure of an unimportant component would result in a "green" finding, but the same programmatic weakness could result in the failure of a safety-significant component. The staff recognizes the potential problem but believes that such programmatic weakness will be reflected in the PIs or identified through inspection of the problem identification and CAP. More experience with the process is needed to validate this assumption.

Notwithstanding these concerns, we believe that the staff has developed a comprehensive oversight process, which is a significant improvement over the previous one. The staff's request to proceed with initial implementation should be approved, recognizing that changes will be made to the RROP, including the SDP; that research should continue to identify better choices for PIs and associated thresholds; that the current PIs are limited in scope; and that any reduction in the baseline inspection effort will require more realistic PIs.

Once the RROP has been implemented, substantial resistance may arise toward any changes. Because changes are expected after the initial implementation, staff should look for methods to implement the process in ways that it can be easily changed.

Sincerely,

Dana A. Powers

Dana A. Powers
Chairman

<u>References</u>:

1. Memorandum dated February 24, 2000, from William D. Travers, Executive Director for Operations, NRC, for The Commissioners, Subject: SECY-00-0049, Results of the Revised Reactor Oversight Pilot Program.
2. Memorandum dated December 17, 1999, from Annette L. Vietti-Cook, Secretary, NRC, to John T. Larkins, ACRS, Subject: Staff Requirements - Meeting with Advisory Committee on Reactor Safeguards, November 4, 1999.
3. Nuclear Energy Institute, NEI 99-02, Draft Revision D, "Regulatory Assessment Performance Indicator Guideline," November 1999.
4. Letter dated November 23, 1999, from Samuel J. Collins, Director, Office of Nuclear Reactor Regulation, NRC, to Dana A. Powers, Chairman, ACRS, Subject: Advisory Committee on Reactor Safeguards Review of Revised Reactor Oversight Process Technical Components.
5. Memorandum dated June 18, 1999, from Annette Vietti-Cook, Secretary, NRC, to William D. Travers, Executive Director for Operations, NRC, Subject: Staff Requirements - SECY-99-007 - Recommendations for Reactor Oversight Process Improvements, and SECY-99-007A - Recommendations for Reactor Oversight Process Improvements (Follow-up to SECY-99-007).

6. Letter dated June 10, 1999, from Dana A. Powers, Chairman, ACRS, to William D. Travers, Executive Director for Operations, NRC, Subject: Pilot Application of the Revised Inspection and Assessment Programs, Risk-Based Performance Indicators, and Performance-Based Regulatory Initiatives and Related Matters.
7. Report dated February 23, 1999, from Dana A. Powers, Chairman, ACRS, to Shirley Ann Jackson, Chairman, NRC, Subject: Proposed Improvements to the NRC Inspection and Assessment Programs.
8. U. S. Nuclear Regulatory Commission, NUREG/CR-5500, Vol. 4, "Reliability Study: High-Pressure Coolant Injection (HPCI) System, 1987-1993," September 1999.
9. U. S. Nuclear Regulatory Commission, NUREG/CR-5500, Vol. 5, "Reliability Study: Emergency Diesel Generator Power System, 1987-1993," September 1999.
10. U. S. Nuclear Regulatory Commission, NUREG/CR-5500, Vol. 6, "Reliability Study: Isolation Condenser System, 1987-1993," September 1999.
11. U. S. Nuclear Regulatory Commission, NUREG/CR-5500, Vol. 7, "Reliability Study: Reactor Core Isolation Cooling System, 1987-1993," September 1999.
12. U. S. Nuclear Regulatory Commission, NUREG/CR-5500, Vol. 8, "Reliability Study: High Pressure Core Spray (HPCS) System, 1987-1993," September 1999.
13. U. S. Nuclear Regulatory Commission, NUREG/CR-xxx Vol. X, "Reliability Study Update: High-Pressure Coolant Injection (HPCI) System, 1987-1998" (Draft), October 1999.
14. U. S. Nuclear Regulatory Commission, NUREG/CR-xxx, Vol. X, "Reliability Study Update: Reactor Core isolation Cooling (RCIC) System, 1987-1998" (Draft), October 1999.
15. U. S. Nuclear Regulatory Commission, NUREG/CR-xxx Vol. x, "Reliability Study Update: High-Pressure Coolant Injection (HPCI) System, 1987-1998" (Draft), October 1999.

UNITED STATES
NUCLEAR REGULATORY COMMISSION
ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
WASHINGTON, D.C. 20555-0001

April 6, 2000

MEMORANDUM TO: William D. Travers
Executive Director for Operations

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

SUBJECT: FINAL RULE: "ELIMINATION OF THE REQUIREMENT FOR NONCOMBUSTIBLE FIRE BARRIER PENETRATION SEAL MATERIALS AND OTHER MINOR CHANGES" (10 CFR PART 50)

During the 471st meeting of the Advisory Committee on Reactor Safeguards, April 5-7, 2000, the Committee considered the subject final rule. The proposed rule change was initially directed by the Commission in the Staff Requirements Memorandum dated June 30, 1998. During the 460th meeting, July 8-10, 1998, the Committee agreed with the Commission's direction as indicated in its report of July 20, 1998. Since the issuance of the report, there were no significant changes made to the proposed final rule. Therefore, the Committee decided not to review this rule and has no objection to issuing the final rule for industry use.

Reference :
Memorandum dated March 9, 2000, from John N. Hannon, Office of Nuclear Reactor Regulation, to John T. Larkins, ACRS, transmitting Final Rule: "Elimination of the Requirement for Noncombustible Fire Barrier Penetration Seal Materials and Other Minor Changes" (10 CFR PART 50)

cc: A. Vietti-Cook, SECY
J. Blaha, OEDO
G. Millman, OEDO
S. Collins, NRR
J. Hannon, NRR
E. Weiss, NRR

UNITED STATES
NUCLEAR REGULATORY COMMISSION
ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
WASHINGTON, D.C. 20555-0001

April 7, 2000

MEMORANDUM TO: William D. Travers
Executive Director for Operations

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

SUBJECT: DRAFT REGULATORY GUIDE DG-1094, "FIRE PROTECTION FOR OPERATING NUCLEAR POWER PLANTS"

During the 471st meeting of the Advisory Committee on Reactor Safeguards, April 5-7, 2000, the Committee considered the subject draft regulatory guide and decided not to review it. The Committee plans to review the proposed final version of this regulatory guide after reconciliation of public comments. The Committee has no objection to issuing the draft Regulatory Guide for public comment.

Reference:
Note dated April 6, 2000, from Eric Weiss, Office of Nuclear Reactor Regulation, to John T. Larkins, ACRS, transmitting Draft Regulatory Guide DG-1094, "Fire Protection for Operating Nuclear Power Plants."

cc: A. Vietti-Cook, SECY
J. Blaha, OEDO
G. Millman, OEDO
S. Collins, NRR
J. Hannon, NRR
E. Weiss, NRR

UNITED STATES
NUCLEAR REGULATORY COMMISSION
ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
WASHINGTON, D.C. 20555-0001

April 10, 2000

MEMORANDUM TO: William D. Travers
Executive Director for Operations

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

SUBJECT: SECY-00-0061, PROPOSED REVISION TO THE ENFORCEMENT POLICY TO ADDRESS THE REVISED REACTOR OVERSIGHT PROCESS

During the 471st meeting of the Advisory Committee on Reactor Safeguards, April 5-7, 2000, the Committee considered the proposed revision to the enforcement policy and decided not to review it.

Reference:
Memorandum dated March 9, 2000, from William D. Travers, Executive Director for Operations, NRC, to the Commissioners, SECY-00-0061, "Proposed Revision to the Enforcement Policy to Address the Revised Reactor Oversight Process."

cc: A. Vietti-Cook, SECY
J. Blaha, OEDO
G. Millman, OEDO
W. Borchardt, OE
S. Collins, NRR
W. Dean, NRR

UNITED STATES
NUCLEAR REGULATORY COMMISSION
ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
WASHINGTON, D.C. 20555-0001

April 10, 2000

MEMORANDUM TO: William D. Travers
Executive Director for Operations

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

SUBJECT: SECY-00-0071, DRAFT REGULATORY GUIDE (DG-1095), "GUIDANCE FOR IMPLEMENTATION OF 10 CFR 50.59 (CHANGES, TESTS, AND EXPERIMENTS)"

During the 469th meeting of the Advisory Committee on Reactor Safeguards, February 3-5, 2000, the Committee met with representatives of the NRC staff and the Nuclear Energy Institute (NEI) to discuss NEI document 96-07, "Guidelines for 10 CFR 50.59 Evaluations." During its 471st meeting, April 5-7, 2000, the Committee considered the subject draft Regulatory Guide (DG-1095), that endorses, with clarifications, NEI 96-07.

Since the remaining items regarding this matter are process issues, the Committee decided not to review the proposed draft Regulatory Guide. The Committee will consider reviewing the proposed final version of this Regulatory Guide after reconciliation of the public comments.

<u>References</u>:

1. SECY-00-0071, dated March 24, 2000, from William D. Travers, Executive Director for Operations, for the Commissioners, Subject: Draft Regulatory Guide (DG-1095), "Guidance for Implementation of 10 CFR 50.59 (Changes, Tests, and Experiments)."
2. NEI 96-07, Revision 1 (Final Draft), Nuclear Energy Institute, "Guidelines for 10 CFR 50.59, Evaluations," dated January 18, 2000.

cc: A. Vietti-Cook, SECY
J. Blaha, OEDO
G. Millman, OEDO
S. Collins, NRR
D. Matthews, NRR
E. McKenna, NRR

UNITED STATES
NUCLEAR REGULATORY COMMISSION
ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
WASHINGTON, D.C. 20555-0001

April 13, 2000

The Honorable Richard A. Meserve
Chairman
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555-0001

Dear Chairman Meserve:

SUBJECT: DRAFT FINAL TECHNICAL STUDY OF SPENT FUEL POOL ACCIDENT RISK AT DECOMMISSIONING NUCLEAR POWER PLANTS

During the 471st meeting of the Advisory Committee on Reactor Safeguards, April 5-7, 2000, we met with representatives of the NRC staff and discussed the subject document. We also had the benefit of the documents referenced, which include the available stakeholders comments. This report is in response to the Commission's request in the Staff Requirements Memorandum dated December 21, 1999, that the ACRS perform a technical review of the validity of the draft study and risk objectives.

BACKGROUND

Decommissioning plants are subject to many of the same regulatory requirements as operating nuclear plants. Because of the expectation that the risk will be lower at decommissioning plants, particularly as time progresses to allow additional decay of fission products, some of these requirements may be inappropriate. Exemptions from the regulations are frequently requested by licensees after a nuclear power plant is permanently shut down. To increase the efficiency and effectiveness of decommissioning regulations, the staff has engaged in rulemaking activities that would reduce the need to routinely process exemptions. The staff has undertaken the technical study and risk analysis discussed here to provide a firm technical basis for rulemaking concerning several exemption issues.

In the draft study the staff has concluded that, provided certain industry decommissioning commitments are implemented at the plants, after one year of decay time the risk associated with spent fuel pool fires is sufficiently low that emergency planning requirements can be significantly reduced. It also concluded that after five years the risk of zirconium fires is negligible even if the fuel is uncovered and that requirements intended to ensure spent fuel cooling can be reduced.

RECOMMENDATIONS

1. The integrated rulemaking on decommissioning should be put on hold until the staff provides technical justification for the proposed acceptance criterion for fuel uncovery frequency. In particular, the staff needs to incorporate the effects of enhanced release of ruthenium under air-oxidation conditions and the impact of the MELCOR Accident

Consequence Code System (MACCS) code assumptions on plume-related parameters in view of the results of expert elicitation.

2. The technical basis underlying the zirconium-air interactions and the criteria for ignition needs to be strengthened. In particular, the potential impact of zirconium-hydrides in high burnup fuel and the susceptibility of the clad to breakaway oxidation need to be addressed.

3. Uncertainties in the risk assessment need to be quantified and made part of the decisionmaking process.

DISCUSSION

The staff's conclusion that the risk after one year of decay time is sufficiently low that emergency planning requirements can be reduced is based partially on the assessed value of fuel uncovery frequency (3.4 x 10^{-6}/yr) being less than the Regulatory Guide 1.174 large, early release frequency (LERF) acceptance value (1x10^{-5}/yr). This LERF risk-acceptance value was derived to be a surrogate for the Safety Goal early fatality quantitative health objectives (QHO) *for operating reactors.* The derivation from the QHO is based, however, on the fission product releases that occur under severe accident conditions which are driven by steam oxidation of the zircaloy and the fuel. These releases include only insignificant amounts of ruthenium. Under air-oxidation conditions of spent fuel fires, significant data indicate much enhanced releases of ruthenium as the very volatile oxide. Indications are that, under air oxidation conditions, the release fractions of ruthenium may be equivalent to those for iodine and cesium. In the accident at Chernobyl significant releases of ruthenium were observed and attributed to the interactions of fuel with air.

These findings have significant implications. The ruthenium inventory in spent fuel is substantial. Ruthenium has a biological effectiveness equivalent to that of Iodine-131 and has a relatively long half-life. If there are significant releases of ruthenium, the Regulatory Guide 1.174 LERF value may not be an appropriate surrogate for the prompt fatality QHO. In addition, because of the relatively long half-life of ruthenium-106, it is likely that the early fatality QHO would no longer be the controlling consequence.

In response to our concerns about the effects of substantial ruthenium release, the staff has made additional MACCS calculations in which it assumed 100 percent release of the ruthenium inventory. For a one-year decay time with no evacuation, the prompt fatalities increased by two orders of magnitude over those in the report which did not include ruthenium release, the societal dose doubled and the cancer fatalities increased four-fold.

Our concern is not just with ruthenium. We are concerned with the appropriateness of the entire source term used in the study. There is a known tendency for uranium dioxide in air to decrepitate into fine particles. The decrepitation is caused by lattice strains produced as the dioxide reacts to form U_3O_8. This decrepitation is a bane of thermogravimetric studies of air oxidation of uranium dioxide since it can cause fine particles to be entrained in the flowing air of the apparatus. This suggests that decrepitating fuel would be readily entrained in vigorous natural convection flows produced in an accident at a spent fuel pool. The decrepitation process provides a low-temperature, mechanical, release mechanism for even very refractory

radionuclides. The staff did consider the possibility that "fuel fines" could be released from fuel with ruptured cladding. It did not, however, believe these fuel fines could escape the plant site. Nevertheless, the staff considered the effect of a $6x10^{-6}$ release fraction of fines. This minuscule release fraction did not significantly affect the calculated findings. There is no reason to think that such a low release fraction would be encountered with decrepitating fuel.

Consequences of accidents involving a spent fuel pool were analyzed using the MACCS code. The staff has completed an expert opinion elicitation regarding the uncertainties associated with many of the critical features of the MACCS code. The findings of this elicitation seem not to have been considered in the analyses of the spent fuel pool accident. One of the uncertainties in MACCS identified by the experts is associated with the spread of the radioactive plume from a power plant site. The spread expected by the experts is much larger than what is taken as the default spread in the MACCS calculations. There is no indication that the staff took this finding into account in preparing the consequence analyses. In addition, the initial plume energy assumed in the MACCS calculations, which determines the extent of plume rise, was taken to be the same as that of a reactor accident rather than one appropriate for a zirconium fire. We suspect, therefore, that the consequences found by the staff tend to overestimate prompt fatalities and underestimate land contamination and latent fatalities just because of the narrow plume used in the MACCS calculations and the assumed default plume energy.

The staff needs to review the air oxidation fission products release data from Oak Ridge National Laboratory and from Canada that found large releases of cesium, tellurium, and ruthenium at temperatures lower than 1000 ^{0}C. Based on these release values for ruthenium, and incorporating uncertainties in the MACCS plume dispersal models, the consequence analyses should be redone.

Based on the results of this reevaluation of the consequences, the staff should determine an appropriate LERF for spent fuel fires that properly reflects the prompt fatality QHO and the potential for land contamination and latent fatalities associated with spent fuel pool fires.

In developing risk-acceptance criteria associated with spent fuel fires, the staff should also keep in mind such factors as the relatively small number of decommissioning plants to be expected at any given time and the short time at which they are vulnerable to a spent fuel pool fire.

We also have difficulties with the analysis performed to determine the time at which the risk of zirconium fires becomes negligible. In previous interactions with the staff on this study, we indicated that there were issues associated with the formation of zirconium-hydride precipitates in the cladding of fuel especially when that fuel has been taken to high burnups. Many metal hydrides are spontaneously combustible in air. Spontaneous combustion of zirconium-hydrides would render moot the issue of "ignition" temperature that is the focus of the staff analysis of air interactions with exposed cladding. The staff has neglected the issue of hydrides and suggested that uncertainties in the critical decay heat times and the critical temperatures can be found by sensitivity analyses. Sensitivity analyses with models lacking essential physics and chemistry would be of little use in determining the real uncertainties.

The staff analysis of the interaction of air with cladding has relied on relatively geriatric work. Much more is known now about air interactions with cladding. This greater knowledge has come in no small part from studies being performed as part of a cooperative international

program (PHEBUS FP) in which NRC is a partner. Among the findings of this work is that nitrogen from air depleted of oxygen will interact exothermically with zircaloy cladding. The reaction of zirconium with nitrogen is exothermic by about 86,000 calories per mole of zirconium reacted. Because the heat required to raise zirconium from room temperature to melting is only about 18,000 calories per mole, the reaction enthalpy with nitrogen is ample. In air-starved conditions, the reaction of air with zirconium produces a duplex film in which the outer layer is zirconium dioxide (ZrO_2) and the inner layer is the crystallographically different compound zirconium nitride (ZrN). The microscopic strains within this duplex layer can lead to exfoliation of the protective oxide layer and reaction rates that deviate from parabolic rates. These findings may well explain the well-known tendency for zirconium to undergo breakaway oxidation in air whereas no such tendency is encountered in either steam or in pure oxygen. Because of these findings, we do not accept the staff's claim that it has performed "bounding" calculations of the heatup of Zircaloy clad fuel even when it neglects heat losses.

The staff focuses its analysis of the reactions of gases with fuel cladding on a quantity they call an "ignition temperature." The claim is that this is the temperature of self-sustained reaction of gas with the clad. Gases will react with the cladding at all temperatures. In fact, at temperatures well below the "conservative ignition temperature" identified by the staff, air and oxygen will react with the cladding quite smoothly and at rates sufficient to measure. Data in these temperature ranges well below the "ignition" temperature form much of the basis for the correlations of parabolic reaction rates with temperature. We believe that the staff should look for a condition such that the increase with temperature of the heat liberation rate by the reaction of gas with the clad exceeds the increase with temperature of the rate of heat losses by radiation and convection. Finding this condition requires that there be high quality analyses of the heat losses and that the heat of reaction be properly calculated. Since staff has neglected any reaction with nitrogen and did not consider breakaway oxidation (causes for the deviations from parabolic reaction rates), it has not made an appropriate analysis to find this "ignition temperature."

In fact, the search for the ignition temperature may be the wrong criterion for the analysis. The staff should also be looking for the point at which cladding ruptures and fission products can be released. Some fraction of the cladding may be ruptured before any exposure of the fuel to air occurs. Even discounting this, one still arrives at much lower temperature criteria for concern over the possible release of radionuclides.

There are other flaws in the material interactions analyses performed as part of the study. For instance, in examining the effects of aluminum melting, the staff seems to not recognize that there is a very exothermic intermetallic reaction between molten aluminum and stainless steel. Compound formation in the Al-Zr system suggests a strong intermetallic reaction of molten aluminum with fuel cladding as well. The staff focuses on eutectic formations when, in fact, intermetallic reactions are more germane to the issues at hand.

We are concerned about the conservative treatment of seismic issues. Risk-informed decisionmaking regarding the spent fuel pool fire issues should use realistic analysis, including an uncertainty assessment.

Because the accident analysis is dominated by sequences involving human errors and seismic events which involve large uncertainties, the absence of an uncertainty analysis of the

frequencies of accidents is unacceptable. The study is inadequate until there is a defensible uncertainty analysis.

The risk posed by fuel uncovery in spent fuel pools for decommissioning plants may indeed be low, however, the technical shortcomings of this study are significant and sufficient for us to recommend that rulemaking be put on hold until the inadequacies discussed herein are addressed by the staff.

Sincerely

Dana A. Powers

Dana A. Powers
Chairman

<u>References</u>:

1. Draft For Comment, Draft Final Technical Study of Spent Fuel Pool Accident Risk at Decommissioning Nuclear Power Plants," February 2000.
2. SECY-99-168, dated June 30, 1999, memorandum from William D. Travers, Executive Director for Operations, NRC, for the Commissioners, Subject: Improving Decommissioning Regulations For Nuclear Power Plants.
3. Memorandum dated December 21, 1999, from Anette L. Vietti-Cook, Secretary of the Commission, to William D. Travers, Executive Director for Operations, NRC, Subject: Staff Requirements -SECY-99-168 - Improving Decommissioning Regulations for Nuclear Power Plants.
4. Letter dated November 12, 1999, from Dana A. Powers, Chairman, ACRS, to William D. Travers, Executive Director for Operations, NRC, Subject: Spent Fuel Fires Associated With Decommissioning.
5. Letter dated December 16, 1999, from William D. Travers, Executive Director for Operations, NRC, to Dana A. Powers, Chairman, ACRS, Subject: Spent Fuel Fires Associated With Decommissioning.
6. E-mail message dated April 5, 2000, from Alan Nelson, Nuclear Energy Institute, to M. El-Zeftawy, ACRS, transmitting NEI comments on Appendix 2.b, "Structural Integrity Seismic Loads."
7. U. S. Nuclear Regulatory Commission, NUREG/CR-6613, "Code Manual for MACCS2, May 1998.
8. U. S. Department of Commerce, "JANAF Thermochemical Tables," Second Edition, Issued June 1971.
9. U. S. Nuclear Regulatory Commission, NUREG/CP-0149, Vol. 2 "Twenty-Third Water Reactor Safety Information Meeting," October 23-25, 1995, "The Severe Accident Research Programme PHEBUS FP.: First Results and Future Tests," published March 1996.
10. U. S. Nuclear Regulatory Commission, NUREG/CR-6244, Vol. 1, "Probabilistic Accident Consequence Uncertainty Analysis," Dispersion and Deposition Uncertainty Assessment, published January 1995.
11. U. S. Nuclear Regulatory Commission, Regulatory Guide 1.174, "An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis," July 1998.

UNITED STATES
NUCLEAR REGULATORY COMMISSION
ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
WASHINGTON, D.C. 20555-0001

April 13, 2000

The Honorable Richard A. Meserve
Chairman
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555-0001

Dear Chairman Meserve:

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

During the 471st meeting of the Advisory Committee on Reactor Safeguards, April 5-7, 2000, we met with representatives of the NRC staff to discuss the NRC program for risk-based analysis of reactor operating experience. The staff presented an overview of the key elements of this program, including data sources, reliability studies, common-cause failure analyses, accident sequence precursor analyses, and risk-based performance indicators. Our Subcommittee on Reliability and Probabilistic Risk Assessment met on December 15, 1999, to discuss these matters.

Conclusions and Recommendations

1. The NRC program for risk-based analysis of reactor operating experience is appropriately focused on current and future needs of the Agency. This program is essential to validate system reliability analysis models and predictions, and is of critical importance for the successful transition to risk-informed regulation. This program should be assigned high priority and adequate resources to effectively support the transition to risk-informed regulation.

2. The staff should work with the industry to ensure that licensee reporting of reliability data for structures, systems, and components (SSCs) that perform risk-significant functions becomes an industry self-imposed requirement of the Equipment Performance and Information Exchange System (EPIX) Program.

3. The staff should perform a systematic evaluation of the equipment reliability and availability databases it needs to support risk-informed regulation.

4. The staff should develop a White Paper or other regulatory guidance to provide mathematical definitions for risk-analysis terms, such as availability and reliability, and to provide guidance on the specific raw data that need to be collected to properly estimate such terms.

5. The staff should perform a systematic comparison of the Standardized Plant Analysis Risk (SPAR) models with the licensees' probabilistic risk assessments (PRAs) or individual plant examinations (IPEs) to identify potential differences in modeling and results, understand the reasons for the differences, and use these insights to improve the SPAR models and tools for the inspection staff. A peer review of the SPAR models should also be considered.

Discussion

The NRC program for risk-based analysis of reactor operating experience includes the collection of industry data, estimation of reliability and availability parameters, industry-wide assessment and trending of systems and components reliability, study of common-cause failure analyses, accident sequence precursor analyses, and development of SPAR models for use by the NRC in risk-informed regulation. Each element of this program has clearly identified users, and in the case of SPAR, a SPAR Model Users Group. This program and its elements are appropriately focused on current and future needs of the Agency. The products generated by this program provide essential information and support to important NRC functions and initiatives.

This program is essential to validate system reliability analysis models and predictions and is of critical importance for the successful transition to risk-informed regulation. The availability of an improved and expanded database grounded on operating experience will allow the staff to better estimate reliability and availability parameters to trend industry performance. The analysis of this database is essential for the identification and industry-wide trending of common-cause failures and accident sequence precursors, the validation of risk assessment tools and their predictions, and the development of an improved and expanded set of revised reactor oversight process performance indicators and plant-specific thresholds. The need for careful review and analysis of the expanding database and for the prompt development, refinement, and maintenance of analysis tools, such as SPAR models, to support plant-specific risk evaluations and the risk-informed oversight process may require more resources than are currently allocated to this program.

Approximately 47,000 licensee event reports (LERs) submitted since 1981 constitute the operating experience database developed and maintained by the NRC and accessed through the Sequence Coding and Search System (SCSS). The nature of the events requiring an LER tends to restrict the type of data to failures that fit certain regulatory reporting requirements. For example, failure of a component in a redundant train of a safety-related system is sometimes not reportable through an LER. Therefore, the LERs do not provide a complete failure rate database. To correct this situation and to expand the database available to the NRC to include, for example, failure rates of components identified as risk significant for the maintenance rule, the Institute for Nuclear Power Operations (INPO) has agreed to make accessible to the NRC the EPIX Program that was developed by the industry to support maintenance rule implementation. The industry has agreed to modify EPIX to provide the reliability and availability data requested by the staff. Furthermore, the industry has agreed to make EPIX the industry's single, common database to support all industry uses, thereby eliminating the several

collection systems previously in place to support specific industry applications and regulatory requirements.

These actions represent significant progress in the development of a common industry and NRC operating experience database. We are concerned, however, that licensee reporting of certain reliability and availability data into EPIX remains voluntary even for key components that perform risk-significant functions. Inconsistent, selective reporting will invalidate the database and make its use in regulatory applications limited. The staff should address this concern with the industry to ensure that licensee reporting of reliability and availability data for components that perform risk-significant functions becomes an industry self-imposed requirement of the EPIX Program.

Through the years, the industry has responded to new regulatory requirements or industry initiatives to collect reliability and availability information by establishing *ad hoc* collection processes tailored to the specific uses of the information being collected. This approach has resulted in the proliferation of definitions of terms such as availability and reliability that are not always consistent with their use in PRA. Now that the decision has been made to utilize EPIX as the industry's single, common database to support all industry applications and regulatory requirements, it is essential to establish common definitions for these terms so that the appropriate data from which these terms are estimated will be collected. The staff should develop a White Paper or other regulatory guidance (e.g., an Information Notice) that will provide proper definitions of these terms and examples of how they are to be estimated from data and will discuss how they differ from the definitions currently used by the industry to fulfill specific regulatory requirements or industry initiatives.

The development of SPAR models is a significant initiative intended to provide the staff, including the regions and resident inspectors, with the tools necessary to support the significance determination process and to perform independent evaluations of events and proposed plant changes. The SPAR models are internal events models of each U.S. operating plant and model the plants down to the system or major component level, but do not model down to the subcomponent and support system level. These analyses have been compared with existing plant-specific PRAs or IPEs in only a few cases. Given the inherent limitations of the SPAR models, they should undergo a systematic comparison with all IPEs in order to understand differences in modeling and results. Such comparison will help resolve modeling differences, identify additional systems and components that need to be included in the model, identify significant omissions and errors, and develop a better understanding of IPE models. To achieve this resolution, it may be necessary for the staff to examine the licensees' fault trees as well as the IPEs.

The user-driven initiatives to verify, validate, and improve the SPAR models to support risk-informed evaluations will make the SPAR models increasingly capable and important in regulatory actions and evaluations. A peer review of the SPAR models should be considered.

The staff is planning to use the substantial availability and reliability information made available through EPIX to develop improved, risk-based performance indicators (RBPIs). The staff plans to identify RBPIs for low-power and shutdown operations and for external events, to expand the current set of performance indicators to include unavailability indicators for risk-significant SSCs, to identify plant-specific thresholds, and to provide a consistent framework for combining

the risk significance of performance indicators and inspection findings to determine the risk significance of overall plant performance. This RBPI plan is documented in an overview White Paper currently under review by the ACRS. This work addresses the concerns of some ACRS members and other stakeholders with the current set of performance indicators. We plan to review this very important work. The successful implementation of this plan depends on the soundness of the EPIX database, which in turn, depends on the industry's commitment to consistently report the required reliability and availability data into the EPIX Program.

Sincerely,

Dana A. Powers

Dana A. Powers
Chairman

References:

1. Letter dated April 3, 2000, from Thomas L. King, Nuclear Regulatory Commission, to Ralph E. Beedle, Nuclear Energy Institute, Subject: Request for Review of White Paper - Development of Risk-Based Performance Indicators, Program Overview.
2. U.S. Nuclear Regulatory Commission, NUREG/CR-5500, Vols. 4-8, Reliability Studies, 1987-1993, September 1999:
   - High-Pressure Coolant Injection (HPCI) System
   - Emergency Diesel Generator Power System
   - Isolation Condenser System
   - Reactor Core Isolation Cooling System
   - High-Pressure Core Spray (HPCS) System
3. U. S. Nuclear Regulatory Commission, NUREG/CR-XXX, Vol. X, Draft Reliability Study Updates: 1987-1998, October 1999:
   - High-Pressure Coolant Injection (HPCI) System
   - Reactor Core Isolation Cooling (RCIC) System
4. Letter dated September 28, 1999, from Tae Kim, Nuclear Regulatory Commission, to Roger O. Anderson, Northern States Power Company, Subject: Prairie Island Nuclear Generating Plant, Unit 1 - Review of Preliminary Accident Sequence Precursor Analysis of Operational Event.
5. Letter dated September 30, 1999, from Stewart N. Bailey, Nuclear Regulatory Commission, to Guy G. Campbell, First Energy Nuclear Operating Company, Subject: Review of Preliminary Accident Sequence Precursor Analysis of Operational Condition at the Davis-Besse Nuclear Power Plant.
6. Letter dated October 5, 1999, from George F. Dick, Nuclear Regulatory Commission, to Oliver D. Kingsley, Commonwealth Edison Company, Subject: Final Accident Sequence Precursor Analysis of Event of September 12, 1998: Byron Station, Unit 1.
7. Letter dated November 9, 1999, from John A. Grobe, Nuclear Regulatory Commission Region III, to R. P. Powers, American Electric Power Company, Subject: Donald C. Cook Nuclear Plant, Units 1 and 2 - Issuance of the Second Draft Risk Assessment.
8. Report dated March 15, 2000, from Dana A. Powers, Chairman, ACRS, to Richard A. Meserve, Chairman, NRC, Subject: Revised Reactor Oversight Process.

9. Letter dated November 22, 1996, from T. S. Kress, Chairman, ACRS, to James M. Taylor, Executive Director for Operations, NRC, Subject: NRC Programs for Risk-Based Analysis of Reactor Operating Experience.

UNITED STATES
NUCLEAR REGULATORY COMMISSION
ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
WASHINGTON, D.C. 20555-0001

April 17, 2000

The Honorable Richard A. Meserve
Chairman
U.S. Nuclear Regulatory Commission
Washington D.C. 20555-0001

Dear Chairman Meserve:

SUBJECT: REACTOR SAFETY GOAL POLICY STATEMENT

During the 469th through 471st meetings of the Advisory Committee on Reactor Safeguards, February 3-5, March 1-4, and April 5-7, 2000, respectively, we discussed the staff's recommendations regarding possible modifications to the Commission's Reactor Safety Goal Policy Statement (SGPS). During these meetings, we had the benefit of discussions with representatives of the NRC staff and of the documents referenced.

BACKGROUND

The staff has identified, and made recommendations on, a set of eight issues for possible consideration in a revised SGPS:

1. Plant specific usage of safety goals
2. Subsidiary objectives [e.g., elevating core damage frequency (CDF) to a fundamental goal]
3. Treatment of uncertainty
4. Use of safety goals to define "how safe is safe enough"
5. Definition of adequate protection and defense in depth
6. Societal risk goals
7. Land contamination goals
8. Temporary changes in risk

In general, the staff has recommended little fundamental change in the SGPS with respect to these eight issues. The implication we draw from this is that the policy guidance with respect to these items would probably be misplaced in a SGPS. That is, the SGPS may not be the right vehicle to meet the need for policy guidance on these particular issues, which have actually arisen in the context of risk informing the regulations. Since there is still a strong need for policy guidance on these issues, we make the following recommendations.

RECOMMENDATIONS

An entirely new policy statement on risk-informed regulation should be developed that would include the following:

- Consideration of a "three-region approach" that defines CDF and large, early release frequency (LERF) boundaries that would be consistent with "adequate protection" and that would define "how safe is safe enough."

- The concept of risk limits for individual plant applications. These risk limits would be quantitatively expressed limits on CDF and LERF and would possibly consider additional limits for societal risk, land contamination, and a cap on temporary changes in risk.

- Guidance on defense in depth to address uncertainties in the risk assessments.

DISCUSSION

Three-Region Approach

The Safety Goal Policy Statement (SGPS) expresses the NRC's policy on "how safe is safe enough" for the population of plants on the average and is not intended for application to individual plants. As such, we see few deficiencies that need rectifying. So, instead of a broad restatement of the SGPS, we believe the need exists for the development of a new policy statement related to risk informing the regulations. This new policy statement should include risk criteria that each individual plant must meet. Up to now, the risk acceptance of individual plants has been dealt with through the concept of "adequate protection," which, among other things, is defined in terms of substantially meeting the requirements in the current body of regulations but is not currently associated with quantitative risk limits. Now that the NRC has embarked on a significant program of risk-informed modifications to the body of regulations, the concept that adequate protection means meeting the regulatory requirements becomes a bit ambiguous and is not nearly so useful as it is in a "deterministic" regulatory system. To ensure coherence in the modified regulations, it will be necessary to have quantitative risk limits, particularly on CDF and LERF.

As we have recommended in previous reports, a three-region approach is a practical way to express such limits. The bottom region would represent "how safe is safe enough." Plants that meet the risk-informed regulations, which may be substantially modified from the current regulations, and whose risk status falls within this region would be considered acceptable. Plants with a risk status falling into the top region would be considered unacceptable, irrespective of whether they met the other regulatory requirements. Such plants would be required to improve their risk status so as to fall at least into the middle region where something like the traditional regulatory analysis would be made for any further improvements that may be considered.

The most likely CDF and LERF candidates for the lower boundaries are those that appear in Regulatory Guide 1.174. We are not certain of the appropriate values for the upper boundaries, but believe they should be consistent with levels achieved as a result of the current

adequate protection concept. This implies to us values about an order of magnitude above the limits in Regulatory Guide 1.174 (i.e., CDF of about 10^{-3}/yr and LERF of about 10^{-4}/yr).

To support the development of a new policy statement on risk-informed regulation, the staff should perform a study to determine the CDF and LERF limits that would be consistent with "adequate protection" and that would constitute the upper boundary. As part of this study, consideration should also be given to determine if additional limits related to societal risk (total deaths) and land contamination can be developed. One possible approach for developing such additional limits would be to set them at the cost-equivalent value of the LERF limit that is determined to be consistent with adequate protection. In principle, this approach would constitute a policy statement on the acceptable exceedance frequency of the cost consequences associated with nuclear power plant accidents. Such additional risk criteria are not likely to be expressible in terms of a surrogate LERF value. An alternative surrogate might be to express limits on exceedance frequency for fission product release which could simultaneously incorporate multiple risk acceptance objectives.

Temporary Changes in Risk

One of the limits in Regulatory Guide 1.174 focuses on the plant CDF expressed on a per-year basis. Temporary changes in CDF (i.e., spikes) that result from planned shutdown or online maintenance activities are not now included in the assessed values for the Regulatory Guide 1.174 limits. No attempts have been made to forecast over the lifetime of a plant how many such spikes to expect or how big they might be. To ensure that the contributions of such spikes do not significantly alter the assessed values that are to be compared to the Regulatory Guide 1.174 limits, there may be a need to place a cap on individual spikes. That is, the acceptability of planned maintenance activities would be contingent upon making a risk assessment for the altered configuration that shows that the spike limit will not be exceeded.

Defense in Depth and Uncertainties

Defense in depth is defined as the application of successive compensatory measures to prevent accidents or to mitigate damage. As we have stated in previous reports, there is a need for policy guidance on the proper balance among such compensatory measures (how many are necessary and how good they have to be), else the application of the defense-in-depth philosophy is subject to an arbitrariness that could hinder the progress of risk-informed regulation.

In recent reports, we have noted that the application of defense in depth can take the form of an allocated "balance" for the risk reduction to be attributed to the various successive compensatory measures for accident prevention and mitigation. Since no technical basis exists for what constitutes an appropriate balance, the establishment of such a balance becomes a matter of policy. It should be established by the Commission. If the risk-reduction contributions for each successive compensatory measure can be quantified with a probabilistic risk assessment (PRA) along with the associated uncertainties, then the PRA becomes the tool for measuring how many such measures are needed and how good they need to be to meet the overall risk objective with the specified allocation. If not, the application of successive compensatory measures becomes a matter of judgment tempered by past experience. In either case, the extent of application should reflect the overall uncertainty in the assessment of

the risk. The greater the overall uncertainty, the more extensive should be the application of compensatory measures.

This defense-in-depth philosophy calls for a requirement that the uncertainties be quantified or estimated and entered into the decision on how much to rely strictly on the PRA results (rationalist approach) and how much to fall back on the traditional judgmental application of defense in depth (structuralist approach). There is a need to tie the actual values of the limits on CDF and LERF to the uncertainties associated with their quantification. The larger the uncertainty, the lower the acceptance limit should be. The staff needs to develop guidance for how to implement such concepts as a way to place quantitative limits on defense in depth in a risk-informed regulatory system.

Additional comments by ACRS Members William J. Shack, John J. Barton, and Mario V. Bonaca are presented below.

Sincerely,

Dana A. Powers

Dana A. Powers
Chairman

Additional Comments by ACRS Members William J. Shack, John J. Barton, and Mario V. Bonaca

We do not agree with our colleagues that there is a pressing need for a more quantitative guideline for risk-informed regulation at the present time. Development of such a guideline would require a significant commitment of resources from the staff and the stakeholders that could be more productively used on activities more directly related to the management of risk such as the implementation of the revised reactor oversight process and the revised maintenance rule, with greater impact on the focusing of licensee resources on risk-significant activities such as risk-informing the classification of safety-significant components, or the assurance of reliable risk assessments through the development of PRA standards for internal and external events.

The concept of adequate protection, the backfit rule, the Safety Goal Policy Statement, and Regulatory Guide 1.174 already provide a regulatory basis for a multiregion approach akin to that proposed in the ACRS report. Additional guidance on acceptable changes in risk is also provided in Regulatory Guide 1.177, "An Approach for Plant-Specific, Risk-Informed Decisionmaking Technical Specifications," and the supplement to Regulatory Guide 1.160, "Monitoring the Effectiveness of Maintenance at Nuclear Power Plants." It is already clear to all stakeholders that the Commission considers PRA results an important element in the decisionmaking process to determine appropriate levels of regulatory action.

In addition, the prescription of quantitative limits in high-level regulatory guidance, such as rules or policy statements, should be minimized. Such limits often are taken to imply a greater precision than is warranted and can lead to an undue emphasis on a single element of the

decisionmaking process. If they are included in the rules or policy statements, especially in terms of prescribed values for even temporary changes in risk, we can envision problems with providing defensible arguments for the values determined by the "PRA of the hour." The preferred approach is that taken in the development of the new Paragraph (a)(4) of the maintenance rule requirement that the licensee assess and manage the increases in risk, but the numerical guidelines for action thresholds are set in the associated Regulatory Guide through endorsement of Section 11 to NUMARC 93–01, "Industry Guideline for Monitoring the Effectiveness of Maintenance at Nuclear Power Plants."

References:

1. Memorandum dated January 27, 2000, from Joseph A. Murphy, Nuclear Regulatory Commission, to John T. Larkins, Advisory Committee on Reactor Safeguards, Subject: ACRS Review of Draft Commission Paper on Proposed Modifications to the Reactor Safety Goal Policy Statement (Predecisional).
2. Memorandum dated October 28, 1999, from Annette Vietti-Cook, Secretary of the Commission, to William D. Travers, Executive Director for Operations, NRC, Subject: Staff Requirements - SECY-99-191 - Modifications to the Safety Goal Policy Statement.
3. Memorandum dated October 16, 1997, from John C. Hoyle, Secretary of the Commission, to L. Joseph Callan, Executive Director for Operations, NRC, Subject: Staff Requirements - SECY-97-208 - Elevation of the Core Damage Frequency Objective to a Fundamental Commission Safety Goal.
4. Report dated April 19, 1999, from Dana A. Powers, Chairman, ACRS, to Shirley A. Jackson, Chairman, NRC, Subject: Status of Efforts on Revising the Commission's Safety Goal Policy Statement.
5. Report dated May 11, 1998, from Robert L. Seale, Chairman, ACRS, to Shirley A. Jackson, Chairman, NRC, Subject: Elevation of CDF to a Fundamental Safety Goal and Possible Revision of the Commission's Safety Goal Policy Statement.
6. Letter dated May 24, 1999, from William D. Travers, Executive Director for Operations, NRC, to Dana A. Powers, Chairman, ACRS, Subject: Status of Efforts on Revising the Commission's Safety Goal Policy Statement.
7. U.S. Nuclear Regulatory Commission, Regulatory Guide 1.174, "An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis," July 1998.
8. Report dated May 19, 1999 from Dana A. Powers, Chairman, ACRS, to Shirley A. Jackson, Chairman, NRC, Subject: The Role of Defense in Depth in a Risk-Informed Regulatory System.
9. Report dated June 17, 1997, from Robert L. Seale, Chairman, ACRS, to Shirley A. Jackson, Chairman, NRC, Subject: Proposed Staff Position Regarding Inclusion of a Containment Spray System in the AP600 Design.

UNITED STATES
NUCLEAR REGULATORY COMMISSION
ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
WASHINGTON, D.C. 20555-0001

April 18, 2000

Dr. Williams D. Travers
Executive Director for Operations
U. S. Nuclear Regulatory Commission
Washington, D. C. 20555-0001

Dear Dr. Travers:

SUBJECT: PROPOSED NRC RESEARCH PLAN FOR DIGITAL INSTRUMENTATION AND CONTROL

During the 471st meeting of the Advisory Committee on Reactor Safeguards, April 5-7, 2000, we met with representatives of the NRC staff to discuss the Proposed NRC Research Plan for Digital Instrumentation and Control (I&C). We also had the benefit of the referenced documents.

This Proposed Research Plan for Digital I&C is the first formal response of the NRC to the 1997 National Research Council study entitled, "Digital Instrumentation and Control Systems in Nuclear Power Plants (Safety and Reliability Issues)." This Research Plan is essential for a definitive road map for research in the digital I&C area that is critically important to the Agency.

BACKGROUND

Digital I&C has been widely used in many high-technology fields, including some safety-critical fields, such as nuclear weapons safety and global aircraft navigation and control, for more than three decades. However, they have only been introduced into safety-related systems of nuclear power plants in the United States in the last decade. Although digital technology has the capability of improving performance and safety, it generally may introduce complexity and new failure modes that have resulted in NRC review being difficult and time consuming. The methodology and procedures prescribed in Chapter 7, "Instrumentation and Control Systems" of the Standard Review Plan (NUREG-0800) are process oriented. A year to review a topical report on a digital I&C system is not uncommon. Unfortunately, the NRC staff does not have tools and procedures to expedite the review process while providing the needed assurance of safety.

In its 1997 study, the National Research Council recommended, among other things, that NRC develop a research and development plan that would balance short-term needs and long-term anticipatory research needs. In NUREG-1635, Vol. 1, "Review and Evaluation of the Nuclear Regulatory Commission Safety Research Program," the ACRS cautioned that "Vulnerabilities of digital systems are different than those of analog systems. Failure probabilities and the failure

characteristics of these systems are also different. Appropriate methods to include digital and software systems in PRAs do not exist." These concerns were also expressed in NUREG-1635, Vol. 2 and Vol. 3. The proposed I&C Research Plan is in response to the issues raised in our reports and those identified by the National Research Council study.

Digital I&C systems are advancing in capability and complexity at a rapid rate. Increased use of automation for traditional tasks such as calibration, surveillance, and fault diagnostics are beginning to appear in nonsafety systems, and NRC must ensure that there are no inappropriate interactions with safety systems. Furthermore, the introduction of "smart systems" with additional complex features, such as self-adaptive compensation and non-mechanistic modeling capabilities provided by neural networks and fuzzy logic appears on the horizon.

CONCLUSIONS AND RECOMMENDATIONS

1. Specific anticipated output or product for each research task should be identified and the way in which this output or product meets the Agency needs should be clearly established. It is not sufficient to indicate that the task output is a report or a computer software code.

2. The approach to be taken or tools to be developed to reduce review time or to increase the assurance of the safety of digital systems being reviewed (e.g., how the proposed task accomplishes the specified goal or research result) should be stated and justified.

3. Quantitative estimates of the anticipated benefits should be given where possible.

4. The software systems program being conducted at the University of Virginia is currently the "magnum opus" of the Office of Nuclear Regulatory Research (RES) Digital I&C research effort. Showing how this program is meeting the research needs of NRC (or progressing toward this goal) could illustrate how activities proposed in this Research Plan could meet their specific objectives.

5. Each proposed task should be analyzed to determine the best approach to accomplish its goal. In some cases, buying commercial software, obtaining technology from other Government Agencies or industries, or adopting industrial standards rather than research may be adequate.

6. The priorities for the various tasks should be explicitly stated in the Proposed Research Plan.

DISCUSSION

The four research areas addressed in the Proposed Research Plan are discussed below:

1. Systems Aspects of Digital Technology

   The Plan addresses the systems aspects of digital technology, including diagnostic and fault tolerance, the computer operating systems, and systems requirements

specifications. These items are related to component and system design, and research in these areas logically is the domain of the I&C vendors. The principal issue for NRC is how to ensure that Commercial Off-The-Shelf software can safely and reliably handle safety-critical functions. The Plan needs better focus on this principal issue.

The proposed investigation of environmental stressors on digital I&C components is a continuation of an ongoing RES program involving the influence of smoke, fire, temperature, humidity, and lightning. Unless there are new unforseen aspects of these stressors, this program should be concluded expeditiously.

2. <u>Software Quality Assurance</u>

Gaining a better understanding of software faults and how to identify them is very important for the NRC. Although the NRC has guidance for software quality assurance, it does not specify the amount of testing required because there currently is no scientific basis for such a requirement. Establishing objective criteria for the adequacy of software quality assurance based on sound principles is an important task.

Current procedures for reviewing software are resource intensive for the staff. The staff urgently needs tools to expedite its review of software systems as documented in the Office of Nuclear Reactor Regulation (NRR) user need memorandum of March 17, 2000. We support elements of the Proposed Research Plan that are directed toward meeting NRR needs.

3. <u>Risk Assessment of Digital I&C Systems</u>

NRC has immediate needs for databases on failure rates of systems containing digital electronic components and the failure modes of digital components and software. These databases are needed to aid the staff reviews of digital I&C systems now being proposed by licensees. Work to develop these databases deserves priority support.

On a longer time scale, the NRC will need probabilistic methods to analyze the performance of systems with digital elements. Development of such probabilistic tools for use by the NRC line organizations should be supported.

4. <u>Emerging I&C Technologies and Applications</u>

The inclusion of emerging I&C technologies as anticipatory research was recommended by the National Research Council's study. It is particularly important that NRC understand the uses and limitations of these emerging technologies as well as the safety implications of such systems. In the long run, automated operation of nuclear power plants with advanced features, such as automatic replacement of signals from faulty sensors, self-adaption to changing conditions, and the use of non-mechanistic (data based) models, seems inevitable. NRC must be prepared to address such issues when these new systems are brought in for review. Automation, or perhaps intelligent systems that back up the operators (operator assistants), is likely to be introduced into

current generation plants to reduce operational errors within the next decade. Preparing for such predictable developments is a desirable element of anticipatory research.

Sincerely,

Dana A. Powers

Dana A. Powers
Chairman

References:
1. Memorandum dated March 17, 2000, from Sher Bahadur, Office of Nuclear Regulatory Research, to John T. Larkins, ACRS, transmitting Draft NRC Research Plan Digital Instrumentation and Control (Predecisional).
2. National Research Council, "Digital Instrumentation and Control Systems in Nuclear Power Plants (Safety and Reliability Issues)," 1997.
3. U. S. Nuclear Regulatory Commission, NUREG-1635, "Review and Evaluation of the Nuclear Regulatory Commission Safety Research Program, Volumes 1 and 2 (Volume 3 is in Print), 1988, 1999, and 2000, respectively.
4. Memorandum dated March 17, 2000, from Samuel J. Collins, Office of Nuclear Reactor Regulation, to Ashok C. Thadani, Office of Nuclear Regulatory Research, Subject: User Need for Digital Instrumentation and Controls Research.

UNITED STATES
NUCLEAR REGULATORY COMMISSION
ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
WASHINGTON, D.C. 20555-0001

May 19, 2000

MEMORANDUM TO: William D. Travers
Executive Director for Operations

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

SUBJECT: PROPOSED FINAL REVISION 1 TO REGULATORY GUIDE 1.54, "SERVICE LEVEL I, II, AND III PROTECTIVE COATINGS APPLIED TO NUCLEAR POWER PLANTS" (FORMERLY DG-1076)

During the 472nd meeting of the Advisory Committee on Reactor Safeguards, May 11-13, 2000, the Committee considered the proposed Revision 1 to Regulatory Guide 1.54 and decided not to review it. The Committee has no objection to issuing the final revision to this Regulatory Guide for industry use.

Reference:

Memorandum dated May 9, 2000, from Michael E. Mayfield, Acting Director, Division of Engineering Technology, Office of Nuclear Regulatory Research, to John T. Larkins, Subject: Request for Review and Concurrence to Issue Proposed Revision 1 to Regulatory Guide 1.54, "Service Level I, II, and III Protective Coatings Applied to Nuclear Power Plants" (formerly DG-1076)

cc: A. Vietti-Cook, SECY
J. Blaha, OEDO
G. Millman, OEDO
A.Thadani, RES
S. Collins, NRR
A. Serkiz, RES
M. Mayfield, RES

UNITED STATES
NUCLEAR REGULATORY COMMISSION
ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
WASHINGTON, D.C. 20555-0001

May 22, 2000

MEMORANDUM TO: William D. Travers
Executive Director for Operations

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

SUBJECT: PROPOSED MODIFICATIONS TO REGULATORY GUIDANCE DOCUMENTS REGARDING USE OF RISK-INFORMED DECISIONMAKING IN LICENSE AMENDMENT REVIEWS

During the 472nd meeting of the Advisory Committee on Reactor Safeguards, May 11-13, 2000, the Committee met with representatives of the NRC staff, Nuclear Energy Institute, and Union Electric Company to discuss the proposed new Appendix to NUREG-0800, Standard Review Plan Chapter 19, "Use of Probabilistic Risk Assessment in Plant-Specific, Risk-Informed Decisionmaking: General Guidance," and associated modifications to Regulatory Guide 1.174, "An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis." The Committee has no objection to issuing the draft documents for public comment. The Committee plans to review the proposed final version of these documents after reconciliation of public comments.

References:
1. Memorandum dated April 3, 2000, from Gary M. Holahan, Office of Nuclear Reactor Regulation, to John T. Larkins, Executive Director, ACRS, Subject: Modifications to Regulatory Guidance Documents Regarding Use of Risk-Informed Decisionmaking in License Amendment Reviews.
2. U.S. Nuclear Regulatory Commission, Regulatory Guide 1.174, "An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis," July 1998.
3. U.S. Nuclear Regulatory Commission, NUREG-0800, Standard Review Plan Chapter 19, "Use of Probabilistic Risk Assessment in Plant-Specific, Risk-Informed Decisionmaking: General Guidance," July 1998.
4. U. S. Nuclear Regulatory Commission, NRC Regulatory Issue Summary 2000-07, "Use of Risk-Informed Decisionmaking in License Amendment Reviews," March 28, 2000.

cc: A. Vietti-Cook, SECY
J. Blaha, EDO
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UNITED STATES
NUCLEAR REGULATORY COMMISSION
ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
WASHINGTON, D.C. 20555-0001

May 22, 2000

MEMORANDUM TO: William D. Travers
Executive Director for Operations

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

SUBJECT: DRAFT REGULATORY GUIDE DG-1096, "TRANSIENT AND ACCIDENT ANALYSIS METHODS" AND STANDARD REVIEW PLAN, SECTION 15.0.1, "REVIEW OF ANALYTICAL COMPUTER CODES"

During the 472nd meeting of the Advisory Committee on Reactor Safeguards, May 11-13, 2000, the Committee met with the NRC staff to discuss the subject draft regulatory guide and standard review plan (SRP) Section. Following the NRC staff presentation, the Committee decided that additional review of these documents by the Committee prior to issuing them for public comment is not necessary. The Committee requested that the staff provide any revisions of these documents to the ACRS prior to issuance for public comment.

The Committee plans to review the proposed final version of the draft regulatory guide and SRP Section after reconciliation of public comments. The Committee has no objection to issuing the draft regulatory guide and SRP section for public comment.

Reference:
Memorandum from Gary Holahan, Office of Nuclear Reactor Regulation, to John T. Larkins, ACRS, dated April 14, 2000, transmitting Draft Regulatory Guide DG-1096, "Transient and Accident Analysis Methods" and Standard Review Plan, Section 15.0.1, "Review of Analytical Computer Codes."

cc: A. Vietti-Cook, SECY
J. Blaha, OEDO
G. Millman, OEDO
S. Collins, NRR
A. Thadani, RES
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UNITED STATES
NUCLEAR REGULATORY COMMISSION
ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
WASHINGTON, D.C. 20555-0001

May 23, 2000

The Honorable Richard A. Meserve
Chairman
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555-0001

Dear Chairman Meserve:

SUBJECT: SECY-00-0053, "NRC PROGRAM ON HUMAN PERFORMANCE IN NUCLEAR POWER PLANT SAFETY"

During the 472nd meeting of the Advisory Committee on Reactor Safeguards, May 11-13, 2000, we completed our review of SECY-00-0053, "NRC Program on Human Performance in Nuclear Power Plant Safety." Our Subcommittee on Human Factors reviewed this matter on March 15, 2000 . During our review, we had the benefit of discussions with representatives of the NRC staff.

Observations and Recommendations

1. The staff has started to develop a framework for coordinating the agency's activities in this important area. The relevant activities of other agencies have been reviewed and operating experience has been analyzed.

2. The analysis of operating experience to identify latent conditions resulting from organizational and programmatic deficiencies and to assess their risk significance is an important element of the Program and should be expanded.

3. Activities under the Program should focus on supporting the two major agency initiatives to risk inform the regulations and to revise the reactor oversight process.

4. The coordination between the, A Technique for Human Event Analysis (ATHEANA) project and the analysis of operating experience project should be improved. ATHEANA's data needs should be considered in the analysis of operating experience. The analysis could, in turn, suggest areas of possible improvements in human reliability analysis models such as ATHEANA.

5. The work proposed in the Program to characterize the extent to which human performance is captured in the revised reactor oversight process (RROP) should be pursued. The validity of the assumption that the impact of cross-cutting issues on plant safety will be reflected in the performance indicators and the baseline inspection findings should be tested.

Discussion

The current Program provides a good step toward the coordination of the staff's activities in this very important area. We expect that the Program will evolve as the results of the current efforts are obtained and new activities are identified.

The staff reviewed the activities of other agencies in this area and performed an analysis of operating experience. Preliminary results from this analysis indicate that human performance has been an important contributor to the large majority of significant events. Latent conditions such as failure to fix known problems, inadequate attention to organizational learning, and inadequate maintenance practices figured prominently in these events. Latent conditions, in fact, outweighed active human performance errors by four to one. What remains to be done is to evaluate the significance of these observations in a probabilistic risk assessment context. The challenge here is not so much the evaluation of the risk significance of individual human errors but, rather, the significance of latent conditions that may lead to poor human performance and the potential for common-cause failures.

We agree with Commissioner Merrifield's observation in his speech at the Regulatory Information Conference that "...we and our licensees must continue to wage an aggressive campaign against the buildup of latent conditions and we simply must not forget to worry." The results of the staff's analysis of operating experience indicated that a major part of latent conditions stems from organizational deficiencies. Such organizational issues constitute an important element of what the International Nuclear Safety Advisory Group has called the "safety culture" of the plant (INSAG-4, 1991).

The agency's two major regulatory initiatives are to risk inform the regulations and to revise the reactor oversight process. We believe that an important consideration in determining which projects should be included in the Program is the degree to which these projects support the agency's initiatives. For example, it is not clear to us what needs of the RROP or ATHEANA will be satisfied by the results of the project on control room design or the Halden simulation experiments.

Risk informing the regulations requires models for human reliability analysis (HRA). There are two broad areas of human performance activities that are included in HRA: activities before an initiating event and activities after an initiating event.

The HRA for routine pre-initiator activities relies largely on the methods described in the Human Reliability Handbook. As we stated in our letter on ATHEANA dated December 15, 1999, an important omission in the analysis of pre-initiator activities is the failure to investigate how human errors during normal operations would initiate a plant event. We believe insights derived from the analysis of operating experience, as well as recent models of human error, can be used to perform this investigation.

The major HRA project for post-initiator activities is ATHEANA. There needs to be more coordination between the ATHEANA project and the analysis of operating experience project. The ATHEANA data needs should be inputs to the data analysis project and the findings from the analysis could suggest improvements to the ATHEANA model. For example, the kinds of

latent conditions identified by the analysis of operating experience should be significant inputs to the identification of the error-forcing context that is an important element of ATHEANA.

The revised reactor oversight process defines three cross-cutting issues: human performance, safety-conscious work environment, and problem identification and resolution. An assumption in the oversight process is that the impact of cross-cutting issues on plant safety will be reflected in the plant performance indicators and the baseline inspection findings. This is an untested assumption. The proposed activity to characterize the extent to which human performance is captured in the revised reactor oversight process will test this assumption and should be pursued.

We look forward to hearing from the staff on the results obtained from this Program.

Sincerely,

Dana A. Powers

Dana A. Powers
Chairman

<u>References</u>

1. SECY-00-0053, Memorandum dated February 29, 2000, from William D. Travers, Executive Director for Operations, NRC, to The Commissioners, Subject: NRC Program on Human Performance in Nuclear Power Plant Safety.
2. Remarks of Jeffrey S. Merrifield, Commissioner, at the Regulatory Information Conference, Washington, D.C., March 29, 2000.
3. International Atomic Energy Agency, Vienna, International Nuclear Safety Advisory Group, "Safety Culture," Report 75-INSAG-4, 1991.
4. Swain, A.D., and H.E. Guttmann, *Handbook of Human Reliability Analysis with Emphasis on Nuclear Power Plant Applications,* NUREG/CR-1278, Rev. 1, Sandia National Laboratories, Albuquerque, NM, August 1983.
5. Letter dated December 15, 1999, from Dana A. Powers, Chairman, ACRS, to William D. Travers, Executive Director for Operations, NRC, Subject: NUREG-1624, Revision 1, "Technical Basis and Implementation Guidelines for A Technique for Human Event Analysis (ATHEANA)."
6. Letter dated February 19, 1999, from Dana A. Powers, Chairman, ACRS, to William D. Travers, Executive Director for Operations, NRC, Subject: SECY-98-244, "NRC Human Performance Plan."

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

May 25, 2000

The Honorable Richard A. Meserve
Chairman
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555-0001

Dear Chairman Meserve:

SUBJECT: USE OF DEFENSE IN DEPTH IN RISK-INFORMING NMSS ACTIVITIES

During the 118th meeting of the Advisory Committee on Nuclear Waste (ACNW), March 27-29, 2000 and the 472nd meeting of the Advisory Committee on Reactor Safeguards (ACRS), May 11-13, 2000, the Committees completed their review of the use of defense in depth in risk-informing the activities of the Office of Nuclear Material Safety and Safeguards (NMSS). On January 13–14, 2000, the Joint Subcommittee of the ACRS/ACNW held a meeting to discuss the NRC's defense-in-depth philosophy in the regulatory process emphasizing its role in NMSS activities, particularly in the licensing of a high-level radioactive waste repository. Members of the Joint Subcommittee, invited experts Robert Bernero, Robert Budnitz, and Thomas Murley, and representatives of the NRC staff, the Nuclear Energy Institute (NEI), and Westinghouse Electric Company provided presentations and held discussions on defense in depth. We also had the benefit of the documents referenced.

OBSERVATIONS AND RECOMMENDATIONS

1. The various compensatory measures taken for the purposes of defense in depth can be graded according to the risk posed by the activity, the contribution of each compensatory measure to risk reduction, the uncertainties in the risk assessment, and the need to build stakeholders trust.

2. The treatment of defense in depth for transportation, storage, processing and fabrication should be similar to its treatment for reactors. Defense in depth for industrial and medical applications can be minimal and addressed on the basis of actuarial information.

3. Defense in depth for protecting the public and the environment from high-level waste (HLW) repositories is both a technical and a policy issue. It is important that a reasonable balance be achieved in the contribution of the various compensatory measures to the reduction of risk. The staff should develop options on how to achieve the desired balance. The opinions of experts and other stakeholders should be sought regarding the appropriateness of each option.

4. Since the balancing of compensatory measures to achieve defense in depth depends on the acceptability of the risk posed by the facility or activity, risk-acceptance criteria should be developed for all NMSS-regulated activities.

BACKGROUND

We agree that there is a need for a common understanding of defense in depth as it relates to a risk-informed regulatory system and that a good working definition is provided in the Commission's White Paper on Risk-Informed and Performance-Based Regulation (Reference 1):

> Defense-in-Depth is an element of the NRC's safety philosophy that employs successive compensatory measures to prevent accidents or mitigate damage if a malfunction, accident, or naturally caused event occurs at a nuclear facility.

As noted in Reference 2, this safety philosophy was formulated in the early days of nuclear power development when it was recognized that the probabilities of accidents with severe consequences must be kept low. At that time, the methods of probabilistic risk assessment (PRA) did not exist, therefore, representative values for these probabilities were unavailable. Although the philosophy of defense in depth has served the nuclear power industry well, two criticisms have been raised.

- Potentially significant accident sequences were overlooked due to the inability to analyze nuclear plants as integrated systems. An example is the interfacing systems loss-of-coolant accident that was identified by the Reactor Safety Study (Reference 3).
- At times, unnecessary burden was imposed on the licensees due to the inability to quantify the impact of the compensatory measures on risk.

There are ways to improve the implementation of the defense-in-depth philosophy because we now have the ability to analyze nuclear facilities as integrated systems and have improved significantly the ability to quantify risk.

The defense-in-depth philosophy remains pertinent because our ability to quantify risks is imperfect. There are uncertainties, primarily due to inadequate models, that current risk assessments do not quantify. The question "what if we are wrong?" is still valid for PRAs and performance assessments (PAs) and speaks to the need for defense in depth. Also, defense in depth is valuable to the NRC's effort to communicate with stakeholders.

The primary need for improving the implementation of defense in depth in a risk-informed regulatory system is guidance to determine how many compensatory measures are appropriate and how good these should be. To address this need, we believe that the following guiding principles are important:

- Defense in depth is invoked primarily as a strategy to ensure public safety given the unquantified uncertainty in risk assessments. The nature and extent of compensatory measures should be related, in part, to the degree of uncertainty.

- The nature and extent of compensatory measures should depend on the degree of risk posed by the licensed activity.

- How good each compensatory measure should be is, to a large extent, a value judgment and, thus, a matter of policy.

Nuclear Reactors

To demonstrate the significance of these guiding principles, we use an example from reactors. A PRA that includes determination of parameter uncertainties can result in probability distributions for the failures of various safety functions in place to control core damage frequency (CDF).

The probability distributions for each of these compensatory measures provide very useful insights into what is currently achievable regarding the performance of each measure. The distribution of the CDF is the result of the propagation of these distributions through the accident sequences. It is the CDF distribution that should determine if additional compensatory measures are needed due to inadequate models. In general, the more such measures are added, the more this distribution shifts to lower frequency values. What CDF distribution is acceptable is a matter of policy. As noted above, the current regulatory system for reactors has evolved without the benefit of these probability distributions. Consequently, the structuralist approach to defense in depth was employed that involves placing compensatory measures on important safety cornerstones to satisfy acceptance criteria for defined design-basis accidents that represent the range of important accident sequences.

The adequacy of the models that have produced probability distributions is an important consideration. Having the results of the risk assessment, we may be able to evaluate the significance of the inadequacies of the models in the context of the probability distributions that have been calculated. Although we can always express our confidence in the risk results in terms of probability curves, we know that to do so in some cases would require excessive reliance on expert judgments. Thus, it remains a matter of policy to decide what compensatory measures should be taken to account for model inadequacies.

Nuclear Materials

The issue of defense in depth and the suggested guiding principles have to be considered somewhat differently when it comes to nuclear materials. For example, there is much less experience in the application of PRA methods to nuclear materials than for nuclear reactors. Although materials systems are not as complex as those for reactors in terms of the assessment of risk, there is greater diversity in materials licensed activities. Perhaps the biggest difference relates to the basic differences in the safety issues between reactors and nuclear waste disposal, especially with regard to HLW repositories. The principal concern in the safety of such repositories is not a catastrophic release of radiation resulting from an accident, but rather the loss through contamination of a valuable life-supporting resource such as ground water or land use. Both can be pathways for radiation exposure to humans. On the other hand, both lend themselves to simple interdiction and intervention measures for the protection of public health and safety. Therefore, the concept of defense in depth for

repositories should be targeted more towards protecting resources where there are high uncertainties due to the very long time involved. Although the accident perspective is somewhat important during pre-closure operations, it is not the dominant safety issue in the area of nuclear waste. Pre-closure operations do, however, lend themselves to using risk-assessment methods similar to those applied to reactor facilities.

With respect to the issue of the diversity of nuclear materials, SECY-99-100 categorizes nuclear materials into four groups. The four groups are abbreviated here as nuclear material activities involving: (1) disposal, (2) transportation and storage, (3) processing and fabrication, and (4) industrial and medical applications.

For disposal (Group 1), the reactor example suggests an approach for considering the effectiveness of protective barriers. For waste disposal facilities, defense in depth is implemented through the use of multiple barriers. For transportation and processing facilities (Groups 2 and 3), PRA methods similar to those applied to reactors can be used and defense in depth can be treated as it is for reactors. For industrial and medical applications (Group 4), we believe that sufficient data exist for many of these nuclear materials activities so that the uncertainties in estimating risks are relatively small. For Group 4 materials, defense in depth can be minimal and can be addressed on the basis of actuarial information, an advantage not available to the same extent for Groups 1-3.

DISCUSSION

Implementation of regulations within a risk-informed framework, including the use of defense in depth, requires the establishment of risk-acceptance criteria for each regulated activity. In most cases, a facility (or a proposed design) already exists with compensatory measures in place. The questions then become (1) Are these measures sufficient for the facility or design to meet the risk-acceptance criteria? (2) Do the measures compensate sufficiently for uncertainties in their assessment? (3) Will the measures gain stakeholder acceptance? Answering these questions is the most difficult aspect of the appropriate utilization of defense in depth in a risk-informed regulatory framework and is the key to establishing limits of necessity and sufficiency.

Establishing the sufficiency and balance of compensatory measures (how many and how good) is, in our view, equivalent to an allocation of the risk reduction (to meet the acceptance criteria) among the various compensatory measures; that is, establishing a regulatory objective based on the balance between prevention and mitigation and perhaps including the balanced allocation among events.

In the power reactor area, there exists a precedent for such allocation. A CDF mean value of 10^{-4}/reactor-year and a conditional containment failure probability of 0.1 have been utilized as the appropriate allocation between prevention and mitigation. These values meet the 10^{-5}/reactor-year risk-acceptance criterion for large, early release frequency as expressed in Regulatory Guide 1.174. This allocation, as described in this Guide, is only for the purpose of evaluating proposed changes to individual plant licensing basis. The staff now has also proposed to use these allocations to guide the defense in depth and risk-informed aspects of 10 CFR Part 50, "Domestic Licensing of Production and Utilization Facilities," by evaluating the risk contribution of each event sequence class against these measures to ensure a balance in the contribution of the sequences.

For nuclear materials applications, including HLW repositories, we recommend the following pragmatic approach for selecting compensatory measures:

1. The contribution that each individual safety system makes in achieving the risk-acceptance criterion should be determined by risk assessment with quantified uncertainty distributions.

2. The adequacy of the risk-assessment models should be evaluated quantitatively where possible and qualitatively in all aspects.

3. Whether the appropriate balance has been achieved can be judged through the opinions of experts and of other stakeholders and is ultimately a policy issue.

4. Policy options should be formulated on how the appropriate balance can be achieved. The impact of each option on building stakeholder trust should be evaluated.

We look forward to working with the staff on these important matters.

Sincerely,

B. John Garrick
Chairman, ACNW

Dana A. Powers
Chairman, ACRS

References:

1. Memorandum dated February 24, 1999, from Annette Vietti-Cook, Secretary, NRC, to William D. Travers, Executive Director for Operations, NRC, Subject: Staff Requirements - SECY-98-144 - White Paper on Risk-Informed and Performance-Based Regulation.
2. J.N. Sorensen, G.E. Apostolakis, T.S. Kress, D.A. Powers, Advisory Committee on Reactor Safeguards, "On the Role of Defense in Depth in Risk-Informed Regulation," American Nuclear Society Conference, PSA '99, International Topical Meeting on Probabilistic Safety Assessment, Washington, DC (408-413), August 22-26, 1999.
3. U.S. Nuclear Regulatory Commission, NUREG-74/014, "Reactor Safety Study, An Assessment of Accident Risks in the U.S. Nuclear Power Plants, WASH-1400," October 1975.
4. SECY-99-100, Memorandum dated March 31, 1999, from William D. Travers, Executive Director for Operations, NRC, to the Commissioners, Subject: Framework for Risk-Informed Regulations in the Office of Nuclear Material Safety and Safeguards.
5. U. S. Nuclear Regulatory Commission, Regulatory Guide 1.174, "An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis," July 1998.

UNITED STATES
NUCLEAR REGULATORY COMMISSION
ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
WASHINGTON, D.C. 20555-0001

June 12, 2000

MEMORANDUM TO: William D. Travers
Executive Director for Operations

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

SUBJECT: INDUSTRY INITIATIVES IN THE REGULATORY PROCESS

During the 473rd meeting of the Advisory Committee on Reactor Safeguards, June 7-9, 2000, the Committee met with representatives of the staff and the Nuclear Energy Institute to discuss a draft Commission paper concerning guidelines for using industry initiatives in the regulatory process. The Committee has no objection to issuing these guidelines for public comment and would like the opportunity to review the proposed final guidelines after resolution of public comments.

Reference
Draft Commission paper received May 30, 2000, from William D. Travers, Executive Director for Operations, for the Commissioners, Subject: Industry Initiatives in the Regulatory Process.

cc: A. Vietti-Cook, SECY
J. Craig, OEDO
G. Millman, OEDO
S. Collins, NRR
J. Strosnider, NRR
R. Wessman, NRR
G. Carpenter, NRR
R. Hermann, NRR

UNITED STATES
NUCLEAR REGULATORY COMMISSION
ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
WASHINGTON, D.C. 20555-0001

June 20, 2000

The Honorable Richard A. Meserve
Chairman
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555-0001

Dear Chairman Meserve:

SUBJECT: PROPOSED RESOLUTION OF GENERIC SAFETY ISSUE-173A, "SPENT FUEL STORAGE POOL FOR OPERATING FACILITIES"

During the 473rd meeting of the Advisory Committee on Reactor Safeguards, June 7-9, 2000, we met with representatives of the NRC staff to discuss the proposed resolution of Generic Safety Issue (GSI)-173A, "Spent Fuel Storage Pool for Operating Facilities." We also had the benefit of the referenced documents.

Recommendations

1. The staff should defer closing out GSI-173A until the re-evaluation associated with spent fuel pool (SFP) accidents for decommissioning plants has been completed.

2. The staff should develop screening criteria for regulatory analyses that are appropriate for SFP accidents at operating reactors.

Discussion

The principal concerns of GSI-173A involve the potential for a sustained loss of SFP cooling capability and a potential for a substantial loss of SFP coolant inventory.

The staff had previously developed and implemented a generic spent fuel storage pool action plan to resolve concerns related to GSI-173A. This plan included plant-specific evaluations and regulatory analyses for safety enhancement backfits for plants that are more vulnerable to the GSI-173A concerns.

The staff has completed the review and evaluation of design features related to the SFP associated with each operating reactor. It found that existing structures, systems, and components related to storage of irradiated fuel provide adequate protection of public health and safety. Consequently,

the staff pursued regulatory analyses for safety enhancement backfits on a plant-specific basis. For these regulatory analyses, the staff used screening criteria for the frequency of "uncovery to within one foot of the top of fuel" or "loss of cooling for eight hours."

The screening criteria were:

$\leq 10^{-6}$/yr	No action justified
10^{-6}/yr to 10^{-5}/yr	Further evaluation needed
$\geq 10^{-5}$/yr	Proceed to value-impact evaluation

With this choice of screening criteria, the staff determined that no further regulatory actions were warranted.

The screening criteria, which constituted the primary basis for the staff's findings, are essentially equivalent to the criteria in the Regulatory Analysis Guidelines. The criteria in the Regulatory Analysis Guidelines are derived from the prompt fatality quantitative health objective (QHO) of the Safety Goal Policy Statement. These are appropriate surrogates for this QHO for reactor accident source terms (fission product releases) driven by steam-zircaloy oxidation. As noted in our report of April 13, 2000, which is related to SFP accident risk at decommissioning nuclear power plants, it is very likely that the source terms for SFP accidents will be significantly different from those for operating reactor accidents. The fission product release from spent fuel accidents is most likely driven by air oxidation of the zircaloy clad. Under such circumstances, there is convincing evidence that there may be substantial release of the ruthenium inventory as the volatile oxide, as well as release of significant quantities of "fuel fines" through a decrepitation process.

Such differences in source terms have significant implications. Ruthenium has relatively long half-life isotopes, its inventory in spent fuel is substantial, and its biological consequences are severe. In connection with decommissioning plants, the staff estimated that prompt fatalities due to an SFP fire could increase by as much as two orders of magnitude if the source term is assumed to include 100-percent release of ruthenium compared to essentially zero release. In addition, the societal dose could double and the cancer fatalities could increase four-fold for this estimated source term. The consequences of actinide releases associated with either fuel decrepitation or matrix-stripping have not yet been evaluated. With emergency response measures, the limiting consideration might well no longer be prompt fatalities. The staff should assess the impact of the different source term on latent fatalities and land contamination.

Because of these differences in the source term, the screening criteria used in this application appear to be inappropriate as surrogates for the prompt fatality QHO related to SFP accidents at operating reactors. A proper surrogate could lead to changes in the conclusions that the staff has reached.

Before closing out GSI-173A and developing the Standard Review Plan and regulatory guidance, the staff should await the results of the proposed re-evaluation of SFP accidents for decommissioning plants and should re-evaluate the regulatory analysis screening criteria for application to SFP accidents at operating reactors.

Sincerely,

Dana A. Powers
Chairman

References:

1. Memorandum dated July 26, 1996, from James M. Taylor, Executive Director for Operations, NRC, to NRC Chairman Jackson and Commissioners Rogers and Dicus, Subject: Resolution of Spent Fuel Storage Pool Action Plan Issues.
2. Memorandum dated September 30, 1997, from L. Joseph Callan, Executive Director for Operations, NRC, to NRC Chairman Jackson and Commissioners Diaz, Dicus, and McGaffigan, Subject: Followup Activities on the Spent Fuel Pool Action Plan.
3. Office for Analysis and Evaluation of Operational Data, NRC, AEOD/S96-02, "Assessment of Spent Fuel Cooling," September 1996.
4. Report dated April 13, 2000, from Dana A. Powers, Chairman, ACRS, to Richard A. Meserve, Chairman, NRC, Subject: Draft Final Technical Study of Spent Fuel Pool Accident Risk at Decommissioning Nuclear Power Plants.

UNITED STATES
NUCLEAR REGULATORY COMMISSION
ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
WASHINGTON, D.C. 20555-0001

June 20, 2000

Dr. William D. Travers
Executive Director for Operations
U.S. Nuclear Regulatory Commission
Washington, DC 20555-0001

Dear Dr. Travers:

SUBJECT: PROPOSED FINAL REGULATORY GUIDE AND STANDARD REVIEW PLAN SECTION ASSOCIATED WITH THE ALTERNATIVE SOURCE TERM RULE

During the 473RD meeting of the Advisory Committee on Reactor Safeguards, June 7-9, 2000, we reviewed the proposed final Regulatory Guide 1.XXX (DG-1081), "Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Plants," and Standard Review Plan Section 15.0.1, "Radiological Consequence Analyses Using Alternative Source Terms." During this review, we had the benefit of discussions with representatives of the NRC staff and of the documents referenced. We previously reviewed the draft versions of these documents and the proposed final rule on the use of alternative source term and provided a report to the Commission dated September 17, 1999.

We find that the proposed final Regulatory Guide and Standard Review Plan Section appropriately address the issues associated with the voluntary use of alternative source terms. We believe these documents are acceptable for issuance. There is, however, a need for both minor editorial changes and to purge the Regulatory Guide of several carryover items from previous regulatory guides that are not appropriate for implementation of alternative source terms. Examples include, specification of the breathing rate to three significant figures for calculation of control room dose consequences and the lack of adequate technical justification for the speciation of iodine in the fuel pin gap. These suggested changes have been provided to the staff for its consideration.

Sincerely,

Dana A. Powers

Dana A. Powers
Chairman

References:

1. Memorandum dated May 5, 2000 from G. M. Holahan, Office of Nuclear Reactor Regulation, NRC, to John T. Larkins, ACRS, Subject: Transmittal of Final Regulatory

Guide 1. XXX (DG-1081), "Alternative Radiological Source Terms for Evaluating Design-Basis Accidents at Nuclear Power Plants," and Standard Review Plan Section 15.0.1, "Radiological Consequence Analyses Using Alternative Source Terms."

2. ACRS Report dated September 17, 1999, from Dana A. Powers, Chairman, ACRS, to Greta Joy Dicus, Chairman, NRC, Subject: Proposed Final Rule on Use of Alternative Source Term at Operating Reactors, Draft Regulatory Guide, and Standard Review Plan.

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UNITED STATES
NUCLEAR REGULATORY COMMISSION
ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
WASHINGTON, D.C. 20555-0001

June 21, 2000

MEMORANDUM TO: William D. Travers
Executive Director for Operations

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

SUBJECT: AP1000 PRE-APPLICATION REVIEW

During the 473rd meeting of the Advisory Committee on Reactor Safeguards, June 7–9, 2000, the Committee considered the proposed AP1000 advanced reactor design pre-application and the issues that would need to be addressed as part of the staff's review of a license application. Attached is a list of issues that the Committee decided should be addressed by the Westinghouse Electric Company.

Attachment: ACRS Issues Related to the Review of the AP1000 Design

Reference:
Westinghouse Electric Company Slides, "AP1000 Overview," presented to the NRC staff on April 27, 2000.

cc: A. Vietti-Cook, SECY
J. Craig, OEDO
G. Millman, OEDO
S. Collins, NRR
D. Matthews, NRR
J. Wilson, NRR

ACRS ISSUES RELATED TO THE REVIEW OF THE AP1000 DESIGN

1. The staff should ensure that the Westinghouse Electric Company's application for the AP1000 design includes the following:

   a. Scope of additional analyses needed for the Standard Safety Analysis Report (SSAR) Chapter 15 accidents. (Revised codes used in the analyses may need to be revalidated.)

   b. Clear identification of the inadequacies in the NOTRUMP code and the steps taken to compensate for them. (A convincing demonstration of the applicability of the revised NOTRUMP code to the AP1000 design is needed.)

   c. Demonstration of the scalability and adequacy of the existing thermal-hydraulic integral and separate effects data.

   d. Identification of additional experiments or analyes needed to justify crediting in-vessel core debris retention as part of the licensing basis.

   e. An evaluation of core performance.

   f. An evaluation of the impact of any changes in performance ratings resulting from design changes.

   g. An evaluation of the effects of the pool of water above the containment on containment structures during seismic events.

2. The staff should ensure that the Westinghouse Electric Company's probabilistic risk analysis for the AP1000 includes the following:

   a. In-containment aerosol behavior, especially the effects of particle charging

   b. Catastrophic failure of the steel shell containment

   c. Containment bypass accident sequences, especially sequences involving steam generator tube ruptures

   d. Reactor coolant system depressurization reliability

   e. Efficacy and reliability of external cooling of the containment shell

   f. Stratification and mixing in the containment

UNITED STATES
NUCLEAR REGULATORY COMMISSION
ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
WASHINGTON, D.C. 20555-0001

June 22, 2000

Dr. William D. Travers
Executive Director for Operations
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555-0001

Dear Dr. Travers:

SUBJECT: DRAFT REPORT, "REGULATORY EFFECTIVENESS OF THE STATION BLACKOUT RULE"

During the 473rd meeting of the Advisory Committee on Reactor Safeguards, June 7-9, 2000, we reviewed the staff's draft report on its evaluation of the regulatory effectiveness of the station blackout (SBO) rule. During our review, we had the benefit of discussions with representatives of the NRC staff and of the documents referenced.

Conclusions and Recommendations

1. The initiative undertaken by the staff to evaluate selected regulations to determine whether they have been effective in achieving their objectives is valuable and should be continued.

2. Regulatory documents related to the SBO rule should be revised to eliminate identified inconsistencies in the definition of reliability. Because of these inconsistencies, the intended reliability targets for emergency diesel generators (EDGs) are not being met in some cases.

3. Acceptance of the use of trigger values in inspection documents should be discontinued.

4. The evaluation of the regulatory effectiveness of the SBO rule provides significant lessons that should be beneficial in preparing a template for the evaluation of other regulations and in the development of future regulations.

Discussion

The staff has an ongoing program to make NRC activities and decisions more effective, efficient, and realistic. As part of this program, the staff is evaluating selected regulations to determine whether the requirements imposed by such regulations are effective in achieving their intended objectives.

The SBO rule is the first to be subjected to this type of evaluation. To assess the regulatory effectiveness, the staff translated regulatory requirements of the SBO rule and other related regulations into a set of expectations on station blackout coping capability, risk reduction, EDG reliability, and value-impact. Actual outcomes from implementing the rule were reviewed to determine if these expectations were met.

The evaluation provides valuable insights on the benefits of the SBO rule and on ways to make future regulations more effective. The SBO rule has provided significant safety benefits in each area evaluated and has resulted in a mean risk reduction consistent with the Commission's objective (core damage frequency reduction of 2.6 $\times 10^{-5}$ events per reactor year). This safety improvement appears to have been cost effective, notwithstanding an implementation cost that exceeded the staff's estimated cost by a factor of four. Much of the excess cost may be associated with the addition of dedicated EDGs at some sites, which was not foreseen by the original cost-benefit evaluation performed by the staff. Safety improvements made by individual utilities that went beyond the minimum to meet the provisions of the SBO rule should not be the basis for a "cost- related" criticism of the rule.

The evaluation also shows that some of the safety improvements provided by the implementation of the SBO rule are being eroded because reliability calculations for some cases are not correct. For example, some licensees do not include EDG maintenance outage times in their reliability calculations, and some licensees do not count failures of EDG support equipment, such as the load sequencer, in their assessment of EDG reliability. This is occurring because several SBO-related regulatory documents provide inconsistent guidance on how to calculate reliability for comparison against EDG reliability targets. These documents should be revised to eliminate inconsistencies in the definition of reliability.

In addition, NRC field inspection documents allow the use of NUMARC 87-00, Revision 1, Appendix D trigger values for assessing compliance with Regulatory Guide 1.155, "Station Blackout," reliability targets. Trigger values were recognized as inappropriate by the ACRS and the staff, but have been retained inadvertently in the inspection documents and are being used by some licensees. The use of trigger values ought to be eliminated.

An important lesson learned from the evaluation of the SBO rule is that regulatory documents have to be reviewed more carefully for consistent interpretation of terms, goals, criteria, and measurements. Also, the evaluation showed the importance of establishing a risk-reduction expectation prior to the development of a new regulation. It was possible to evaluate the risk-reduction expectation for the SBO rule only because the Commission established expectations at the time the rule was issued. These lessons should be valuable in preparing a template for evaluating the effectiveness of other regulations and in developing future regulations.

Sincerely,

Dana A. Powers

Dana A. Powers
Chairman

References:
1. Letter dated April 14, 2000, from Charles E. Rossi, Office of Nuclear Regulatory Research, to David Modeen, Nuclear Energy Institute, Subject: Draft Report, "Regulatory Effectiveness of the Station Blackout Rule."
2. U. S. Nuclear Regulatory Commission, Regulatory Guide 1.155, "Station Blackout," August 1988.
3. Nuclear Management and Resources Council, "Guidelines and Technical Bases for NUMARC Initiatives Addressing Station Blackout at Light Water Reactors," NUMARC-87-00, November 1987.
4. Letter dated December 14, 1993, from J. Ernest Wilkins, Jr., Chairman, ACRS, to Ivan Selin, Chairman, NRC, Subject: ACRS Concern Over "Trigger Value" Approach Proposed by Regulatory Guide 1.160."

UNITED STATES
NUCLEAR REGULATORY COMMISSION
ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
WASHINGTON, D.C. 20555-0001

July 14, 2000

MEMORANDUM TO: William D. Travers
Executive Director for Operations

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

SUBJECT: DRAFT REGULATORY GUIDES

During the 474th meeting of the Advisory Committee on Reactor Safeguards, July 12-14, 2000, the Committee considered the staff's request that the ACRS review the proposed final revisions to the Regulatory Guides listed below after resolution of public comments. The Committee will consider reviewing the proposed final versions of these Guides subsequent to reconciliation of public comments. In the future, the Committee would appreciate receiving draft documents for use in making decisions.

1. DG-1098, "Safety-Related Concrete Structures for Nuclear Power Plants (other than Reactor Vessels and Containments)"
2. DG-1100, "Design Guidance for Radioactive Waste Management Systems, Structures, and Components Installed in Light-Water-Cooled Nuclear Power Plants"
3. DG-1099, "Anchoring Component and Structural Supports in Concrete"
4. DG-XXXX, "Site Investigations for Foundations of Nuclear Power Plants," (Proposed Revision 2 to Regulatory Guide 1.132)
5. DG-XXXX, "Procedures and Criteria for Assessing Seismic Liquefaction at Nuclear Power Plant Sites"
6. DG-XXXX, "Laboratory Investigation of Soils for Engineering Analysis and Design of Nuclear Power Plants," (Proposed Revision 1 to Regulatory Guide 1.138)

Reference:
Memorandum dated July 6, 2000, from Sher Bahadur, Office of Nuclear Regulatory Research, to John T. Larkins, ACRS, Subject: Draft Regulatory Guides.

cc: A. Vietti-Cook, SECY
J. Craig, OEDO
G. Millman, OEDO
A. Thadani, RES
S. Collins, NRR
S. Bahadur, RES
M. Mayfield, RES

UNITED STATES
NUCLEAR REGULATORY COMMISSION
ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
WASHINGTON, D.C. 20555-0001

July 17, 2000

MEMORANDUM TO: William D. Travers
Executive Director for Operations

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

SUBJECT: TOPICAL REPORT BAW-2374, "JUSTIFICATION FOR NOT INCLUDING POSTULATED BREAKS IN LARGE-BORE REACTOR COOLANT SYSTEM PIPING IN THE LICENSING BASIS FOR EXISTING AND REPLACEMENT ONCE-THROUGH STEAM GENERATORS"

During the 474th meeting of the Advisory Committee on Reactor Safeguards, July 12-14, 2000, the Committee considered the subject document and decided that it would like the opportunity to review this matter after the staff prepares the safety evaluation.

Reference:
Letter dated July 7, 2000, from David J. Firth, B&W Owners Group, to Document Control Desk, U.S. Nuclear Regulatory Commission, Subject: Submittal of Topical Report BAW-2374, "Justification for Not Including Postulated Breaks in Large-Bore Reactor Coolant System Piping in the Licensing Basis for Existing and Replacement Once-through Steam Generators."

cc: A. Vietti-Cook, SECY
J. Craig, OEDO
G. Millman, OEDO
S. Collins, NRR
S. Bajwa, NRR
S. Bailey, NRR

UNITED STATES
NUCLEAR REGULATORY COMMISSION
ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
WASHINGTON, D.C. 20555-0001

July 18, 2000

MEMORANDUM TO: William D. Travers
Executive Director for Operations

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

SUBJECT: PROPOSED RULE, 10 CFR PART 55, "OPERATOR LICENSE ELIGIBILITY AND USE OF SIMULATION FACILITIES IN OPERATOR LICENSING"

During the 474th meeting of the Advisory Committee on Reactor Safeguards, July 12-14, 2000, the Committee considered the subject proposed rule. The Committee has no objection to issuing this rule.

<u>Reference</u>:
NRC Proposed Rule, 10 CFR Part 55, "Operator License Eligibility and Use of Simulation Facilities in Operator Licensing," received July 7, 2000.

cc: A. Vietti-Cook, SECY
J. Craig, OEDO
G. Millman, OEDO
S. Collins, NRR
B. Boger, NRR
G. Tracy, NRR
D. Trimble, NRR

UNITED STATES
NUCLEAR REGULATORY COMMISSION
ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
WASHINGTON, D.C. 20555-0001

July 18, 2000

MEMORANDUM TO: William D. Travers
Executive Director for Operations

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

SUBJECT: PROPOSED REVISION 3 TO REGULATORY GUIDE 1.52, "DESIGN, INSPECTION, AND TESTING CRITERIA FOR AIR FILTRATION AND ADSORPTION UNITS OF POST-ACCIDENT ENGINEERED-SAFETY-FEATURE ATMOSPHERE CLEANUP SYSTEMS IN LIGHT-WATER-COOLED NUCLEAR POWER PLANTS."

During the 474th meeting of the Advisory Committee on Reactor Safeguards, July 12-14, 2000, the Committee considered the subject draft regulatory guide. The Committee has no objection to the publication of this regulatory guide.

Reference:
Draft Regulatory Guide DG-1102, Proposed Revision 3 to Regulatory Guide 1.52, "Design, Inspection, And Testing Criteria For Air Filtration And Adsorption Units of Post-Accident Engineered-Safety-Feature Atmosphere Cleanup Systems in Light-Water-Cooled Nuclear Power Plants," July 7, 2000.

cc: A. Vietti-Cook, SECY
J. Craig, OEDO
G. Millman, OEDO
S. Collins, NRR
G. Holahan, NRR
J. Hannon, NRR
J. Segala, NRR

UNITED STATES
NUCLEAR REGULATORY COMMISSION
ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
WASHINGTON, D.C. 20555-0001

July 20, 2000

MEMORANDUM TO: William D. Travers
Executive Director for Operations

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

SUBJECT: PROPOSED FINAL REGULATORY GUIDANCE DOCUMENTS REGARDING USE OF RISK-INFORMED DECISIONMAKING IN LICENSE AMENDMENT REVIEWS

During the 474th meeting of the Advisory Committee on Reactor Safeguards, July 12-14, 2000, the Committee considered the proposed final revisions to regulatory guidance documents regarding use of risk-informed decisionmaking in license amendment reviews. In a report dated October 8, 1999, the Committee commented on a draft Commission paper associated with this matter. During its 472nd meeting, May 11-13, 2000, the Committee discussed with representatives of the staff, the Nuclear Energy Institute, and the Union Electric Company the proposed revisions to these regulatory documents. The Committee has no additional comments or concerns on the proposed final revisions to these regulatory documents.

<u>References</u>:

1. Memorandum received July 13, 2000, from Gary M. Holahan, Office of Nuclear Reactor Regulation, to John T. Larkins, Executive Director, ACRS, Subject: Final Regulatory Guidance Documents Regarding Use of Risk-Informed Decisionmaking in License Amendment Reviews.
2. Report dated October 8, 1999, from Dana A. Powers, Chairman, ACRS, to Greta Joy Dicus, Chairman, NRC, Subject: Draft Commission Paper Regarding Proposed Guidelines for Applying Risk-Informed Decisionmaking in License Amendment Reviews.
3. Memorandum dated May 22, 2000, from John T. Larkins, Executive Director, ACRS, to William D. Travers, Executive Director for Operations, NRC, Subject: Proposed Modifications to Regulatory Guidance Documents Regarding Use of Risk-Informed Decisionmaking in License Amendment Reviews.

cc: A. Vietti-Cook, SECY
J. Craig, OEDO
G. Millman, OEDO
S. Collins, NRR
G. Holahan, NRR
R. Palla, NRR

UNITED STATES
NUCLEAR REGULATORY COMMISSION
ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
WASHINGTON, D.C. 20555-0001

July 20, 2000

The Honorable Richard A. Meserve
Chairman
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555-0001

Dear Chairman Meserve:

SUBJECT: NUCLEAR ENERGY INSTITUTE LETTER DATED JANUARY 19, 2000, ADDRESSING NRC PLANS FOR RISK-INFORMING THE TECHNICAL REQUIREMENTS IN 10 CFR PART 50

During the 474th meeting of the Advisory Committee on Reactor Safeguards, July 12-14, 2000, we discussed the subject letter to NRC Chairman Meserve. In addition, we discussed with representatives of the staff and the Nuclear Energy Institute (NEI) the NRC plans for risk-informing the technical requirements in 10 CFR Part 50. During our discussions, we had the benefit of the documents referenced.

This report responds to the Commission's request in the April 5, 2000 Staff Requirements Memorandum (SRM) that the ACRS review the subject letter.

Recommendations

1. The staff should proceed with finalizing the framework for risk-informing the technical requirements of 10 CFR 50, including the prioritization criteria, and use the information in the NEI letter, as appropriate.

2. The staff will want to interact further with the Industry to determine the benefits and burden reduction that could result from changes in rules in light of risk information.

Background

The Commission directed the staff to develop a plan for risk-informing technical requirements in 10 CFR Part 50. In response to staff activities in this area, NEI conducted an industry survey to identify regulations that are prime candidates for assessment and change or possible candidates for improvement. This was the subject of an NEI letter dated January 19, 2000, to Chairman Meserve. In an SRM dated April 5, 2000, the Commission requested that:

The ACRS review the January 19, 2000, letter from the Nuclear Energy Institute (NEI) to Chairman Meserve, that addresses NRC plans for risk-informing the technical requirements in 10 CFR Part 50. In particular, the ACRS, in coordination with the NRC staff, should evaluate the priority listing of regulatory requirements that might be modified based on consideration of risk. This includes review of interim staff reports on the activities described in SECY-99-256 and SECY-99-264.

In SECY-98-300, "Options for Risk-Informed Revisions to 10 CFR Part 50 - Domestic Licensing of Production and Utilization Facilities," the staff proposed three options for modifying regulations in 10 CFR Part 50 to make them risk informed. These options were:

1. Continue with ongoing rulemaking, but make no additional changes to Part 50.

2. Make changes to the overall scope of systems, structures, and components (SSCs) covered by those sections of Part 50 requiring special treatment (such as quality assurance, technical specifications, environmental qualification, and 10 CFR 50.59 by formulating new definitions of safety-related and important-to-safety SSCs).

3. Make changes to specific requirements in the body of regulations, including general design criteria.

In the SRM of June 8, 1999, the Commission approved proceeding with the current rulemaking in Option 1, implementing Option 2, and proceeding with a study of Option 3. For Option 3, the Commission requested that the staff determine how best to proceed and provide a detailed plan outlining its recommendations regarding specific regulatory changes that should be pursued. SECY-99-256 provides the staff's plans for implementing Option 2. SECY-99-264 provides the staff's plans with respect to the Commission request to proceed with a study of Option 3.

The letter of January 19, 2000, which is the primary subject of this report, provided the industry's initial response to SECY-99-264. In this letter, NEI stated that there is general industry support for the overall approach. NEI also reported the results of a survey to which 61 units responded. This survey identified what the industry considers as prime candidate regulations for assessment and change and provided estimates of the financial benefits expected from risk-informing each identified regulation.

Discussion

It is appropriate that the staff consider the industry's priorities and seek information from the industry on the expected benefits. The industry priority list appears to be primarily driven by burden reduction and the associated cost savings. This is an important input in the prioritization process. The industry presumably is the best judge of the burden associated with a regulation, and this input will be valuable to the staff in developing its own priority listing. Many of the NEI priority items seem to relate to the scope of SSCs important to safety, quality assurance, and in-service inspection. These items are already incorporated under Option 1 and Option 2 and, thus, are already being given priority. The staff has also accelerated its preparation of a risk-informed revision to 10 CFR 50.44, "Standards for combustible gas control system in light-water-cooled power reactors."

In SECY-00-0086, "Status Report on Risk-Informing the Technical Requirements of 10 CFR Part 50 (Option 3)," the staff proposed a framework for prioritization, consideration of defense in depth, safety margins, and uncertainties. Because this framework is still under development, it is premature for us to comment. We believe, however, that this framework is appropriate and its development should continue.

If the staff is to have reliable estimates of the benefits of risk-informing selected parts of 10 CFR Part 50, there must be some sort of determination of the possible plant changes that will result. This determination appears to require first developing the risk-informed version of the rule and then identifying the possible changes on a plant-by-plant basis. After the staff has decided on the risk-informed version of a particular rule, it may want to further interact with the industry to determine the ranges of benefits – including uncertainties. For risk/benefit decisions, uncertainties in benefits are just as important as uncertainties in risk.

The highest priority candidate in the NEI letter is 10 CFR 50.46 and Appendix K related to emergency core cooling system (ECCS). The NEI letter provided information on the potential benefit (of up to $3 million per unit per year) as one of the bases for this selection. In our view, 10 CFR 50.46 and Appendix K can be considered as a deterministic specification on how good the ECCS cooling capability must be after it is activated. Its risk implications relate primarily to success criteria – will the ECCS be good enough to provide assurance that the accident will be terminated and long-term shutdown cooling provided. Probabilistic risk assessment insights, however, also suggest that the proposed challenge to the ECCS, an instantaneous double ended guillotine break (DEGB), is an extremely unlikely event.

It is not clear that substantial changes can be made in terms of the success criteria. Successful continued cooling involves evaluation of the effects of potential local hot spots, possible geometry changes as a result of rod bowing and clad swelling, local dry out, steam-zirconium chemical reactions, and possible propagation of loss of coolant from local to substantial involvement of the core. Such phenomena are highly uncertain and, therefore, must have proper criteria to provide the required confidence to be attached to the success criteria that the accident will be terminated and the core damage frequency acceptance value will be achieved. In our view, then, this is an area with a strong defense-in-depth component related to the proper balance between prevention and mitigation in a highly uncertain phenomenological area.

There appear to be greater benefits from reconsidering changes in the definition of the challenges to the ECCS, i.e., replacement of the DEGB, with an alternative large-break loss-of-coolant accident. It has long been recognized that the DEGB has led to undesirable consequences in the structural design of piping systems. It may also have negative consequences when used as the design basis for ECCS. It could, for example, result in a greater likelihood of pressurized thermal shock and lead to unrealistic startup times for emergency equipment that can reduce reliability.

On the other hand, the use of the DEGB can be considered as a sort of margin on the acceptable performance of ECCS. A systematic assessment, therefore, of the consequences of this change must be considered. Although the staff's framework is still under development, it does include a proposed process to appropriately consider the impacts of changes to the

regulations. We look forward to interacting with the staff in its development of the final framework.

Sincerely,

Dana A. Powers

Dana A. Powers
Chairman

References:

1. Letter dated January 19, 2000, from Joe F. Colvin, President and Chief Executive Officer, NEI, to Richard A. Meserve, Chairman, NRC, regarding Proposed Staff Plan for Risk-Informing Technical Requirements in 10 CFR Part 50.
2. Memorandum dated April 5, 2000, from Annette L. Vietti-Cook, Secretary, NRC, to John T. Larkins, ACRS/ACNW, Subject: Staff Requirements - Meeting with ACRS on Risk Informing 10 CFR Part 50, March 2, 2000.
3. Memorandum dated June 8, 1999, from Annette L. Vietti-Cook, Secretary, NRC, to William D. Travers, Executive Director for Operations, NRC, Subject: Staff Requirements - SECY-98-300 - Options for Risk-Informed Revisions to 10 CFR Part 50, "Domestic Licensing of Production and Utilization Facilities."
4. Memorandum dated April 12, 2000, from William D. Travers, Executive Director for Operations, NRC, for the Commissioners, SECY-00-0086, Subject: Status Report on Risk-Informing the Technical Requirements of 10 CFR Part 50 (Option 3).
5. Memorandum dated October 29, 1999, from William D. Travers, Executive Director for Operations, NRC, for the Commissioners, SECY-99-256, Subject: Rulemaking Plan for Risk-Informing Special Treatment Requirements.
6. Memorandum dated November 8, 1999, from William D. Travers, Executive Director for Operations, NRC, for the Commissioners, SECY-99-264, Subject: Proposed Staff Plan for Risk-Informing Technical Requirements in 10 CFR Part 50.
7. Memorandum dated December 23, 1998, from William D. Travers, Executive Director for Operations, NRC, for the Commissioners, SECY-98-300, Subject: Options for Risk-Informed Revisions to 10 CFR Part 50 - "Domestic Licensing of Production and Utilization Facilities."

UNITED STATES
NUCLEAR REGULATORY COMMISSION
ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
WASHINGTON, D.C. 20555-0001

July 20, 2000

Dr. William D. Travers
Executive Director for Operations
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555-0001

Dear Dr. Travers:

SUBJECT: PROPOSED FINAL ASME STANDARD FOR PROBABILISTIC RISK ASSESSMENT FOR NUCLEAR POWER PLANT APPLICATIONS

During the 474th meeting of the Advisory Committee on Reactor Safeguards, July 12-14, 2000, we met with representatives of the American Society of Mechanical Engineers (ASME) Committee on Nuclear Risk Management (CNRM) to discuss the proposed final Standard for Probabilistic Risk Assessment (PRA) for Nuclear Power Plant Applications. Our Subcommittee on Reliability and PRA met with the ASME CNRM on June 28, 2000, to discuss this matter. We previously reviewed a draft version of the ASME Standard and commented in a letter dated March 25, 1999.

<u>**Conclusions and Recommendations**</u>

1. The proposed Standard is not a traditional "design-to" engineering standard or a procedures guide. Consequently, any argument that a PRA should be accepted by the staff simply because it meets the Standard would not be valid.

2. The Standard should be useful because it provides a framework for the systematic assessment of PRA elements. This will aid staff reviews by identifying weak elements in a PRA. Because the Standard can accommodate a wide range of PRA quality, however, the staff will still need to make a case-by-case assessment of the adequacy of the PRA.

3. The three categories of PRA requirements proposed in the Standard deal reasonably with the wide range of risk-informed decisions. The differences among the categories should be delineated more clearly, especially the treatment of uncertainties.

4. The discussion of the categories of requirements needed for particular regulatory applications that is given in Section 1.5, "Application Categories," can be misleading and should be deleted.

5. More guidance and examples should be given on the circumstances under which supplementary analyses would be needed and how they would enhance the scope and level of detail in a PRA.

Discussion

The quality of PRA is at the heart of a successful risk-informed regulatory system. The term "quality" includes many things, such as issues of scope, detail, and technical adequacy of the analyses. PRAs are very ambitious. To model everything that is relevant in a particular situation, including hardware failures, human performance, as well as physical and chemical phenomena, is extremely difficult. Defining PRA quality *a priori* is a highly subjective and very difficult task, given the varied nature of potential risk-informed decisions. Thus, PRA quality should be evaluated in the context of the decision the PRA supports. If, for instance, a particular decision is insensitive to recovery actions, a PRA that does not include such actions would not suffer in quality for that particular decision.

The Standard recognizes this difficulty and proposes three categories of requirements that determine the range of applications for which a PRA would be appropriate. The delineation of the differences among categories is not always clear and this situation is exacerbated by the fact that the Standard relies primarily on tables with limited accompanying text. More details on the differences among the categories and further elaboration on the requirements would be beneficial.

The NRC staff will ultimately have to decide whether the submitted risk information is sufficient and of adequate quality to support a particular risk-informed decision. The categories and the associated requirements will facilitate this process by helping all parties involved establish a common PRA language and by providing a framework within which potential weaknesses of the PRA could be identified early in the decisionmaking process.

The Standard should not be viewed in the same way as other, more traditional, "design-to" standards usually associated with ASME. PRAs of a wide range of quality could be said to meet the requirements of the Standard. Consequently, any argument that a PRA should be accepted by the staff simply because it meets the Standard is moot. The discussion of the categories of requirements needed for a particular regulatory application provided in Section 1.5 of the Standard should be deleted to avoid misunderstandings and misleading expectations. We were told by the ASME representatives that they would consider revising this Section to avoid these problems.

For a given application, the Standard allows the use of supplementary analyses to augment the PRA but does not provide guidance on the scope and level of detail of these analyses relative to that provided for the categories. Lack of such guidance may increase the NRC staff effort required to assess the appropriateness of the supplementary analyses in risk-informed decisionmaking.

We offered a number of detailed comments on the Standard that the ASME representatives agreed to consider. We look forward to reviewing the staff's work related to this matter.

Sincerely,

Dana A. Powers

Dana A. Powers
Chairman

References:

1. Letter dated June 14, 2000, from G. M. Eisenberg, ASME International, to M. Markley, ACRS, transmitting Draft #12 of Proposed ASME Standard for Probabilistic Risk Assessment for Nuclear Power Plant Applications, dated May 30, 2000.
2. American Society of Mechanical Engineers, "White Paper and Guidance to Reviewers of the Draft ASME Standard for Probabilistic Risk Assessment for Nuclear Power Plant Applications," dated June 13, 2000.
3. Letter dated March 25, 1999, from Dana A. Powers, Chairman, Advisory Committee on Reactor Safeguards, to William D. Travers, Executive Director for Operations, NRC, Subject: Proposed ASME Standard for Probabilistic Risk Assessment for Nuclear Power Plant Applications (Phase 1).

UNITED STATES
NUCLEAR REGULATORY COMMISSION
ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
WASHINGTON, D.C. 20555-0001

July 27, 2000

MEMORANDUM TO: William D. Travers
Executive Director for Operations

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

SUBJECT: ASSESSMENT OF THE QUALITY OF PRAs

During the 474th meeting of the Advisory Committee on Reactor Safeguards, July 12-14, 2000, the staff made a presentation on PRA Quality. At that time, the Committee was advised that the Commission Paper on PRA Quality was being revised. Therefore, the Committee decided to defer comment on the Commission Paper until a revised version of the document is received.

The Committee plans to review and provide comments on the revised version of the Commission Paper.

References:

1. Memorandum dated April 18, 2000, from Annette Vietti-Cook, Secretary, NRC, to William D. Travers, Executive Director for Operations, NRC, Subject: Staff Requirements - Briefing on Risk-Informed Regulation Implementation Plan (SECY-00-0062).
2. Draft Memorandum received June 30, 2000, from William D. Travers, Executive Director for Operations, NRC, to The Commissioners, Subject: Addressing PRA Quality in Risk-Informed Applications.
3. Memorandum dated June 19, 2000, from Samuel J. Collins, Director, NRR, to Ashok C. Thadani, Director, RES, Subject: Request for Assistance in Review of NEI 00-02, "Probabilistic Risk Assessment Peer Review Process Guidance."

cc: A. Vietti-Cook, SECY
J. Craig, OEDO
G. Millman, OEDO
S. Collins, NRR
G. Holahan, NRR
A. Thadani, RES
T. King, RES
M. Cunningham, RES
M. Drouin, RES

UNITED STATES
NUCLEAR REGULATORY COMMISSION
ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
WASHINGTON, D.C. 20555-0001

September 5, 2000

MEMORANDUM TO: William D. Travers
Executive Director for Operations

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

SUBJECT: FINAL REGULATORY GUIDE 1.18x ON 10 CFR 50.59, CHANGES, TESTS, AND EXPERIMENTS

During the 471st meeting of the Advisory Committee on Reactor Safeguards, April 5-7, 2000, the Committee discussed draft Regulatory Guide (DG-1095), "Guidance for Implementation of 10 CFR 50.59 (Changes, Tests, and Experiments)," that endorses, with clarifications, NEI document 96-07, "Guidelines for 10 CFR 50.59 Evaluation."

During the 475th meeting, August 29- September 1, 2000, the Committee decided not to review the subject final Regulatory Guide that endorses (without exception) NEI 96-07, Revision 1.

References:

1. U.S. Nuclear Regulatory Commission, Office of Nuclear Regulatory Research, Regulatory Guide 1.18x, "Guidance for Implementation of 10 CFR 50.59, Changes, Tests, and Experiments," dated September 2000.

2. Nuclear Energy Institute, NEI 96-07, Revision 1 (Pre-Publication Draft), "Guidelines for 10 CFR 50.59 Evaluations," dated July 12, 2000.

cc: A. Vietti-Cook, SECY
J. Craig, OEDO
G. Millman, OEDO
S. Collins, NRR
D. Matthews, NRR
E. McKenna, NRR

UNITED STATES
NUCLEAR REGULATORY COMMISSION
ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
WASHINGTON, D.C. 20555-0001

September 7, 2000

MEMORANDUM TO: William D. Travers
Executive Director for Operations

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

SUBJECT: DRAFT REGULATORY GUIDE DG-1075, "EMERGENCY PLANNING AND PREPAREDNESS FOR NUCLEAR POWER REACTORS," (PROPOSED REVISION 4 TO REGULATORY GUIDE 1.101)

During the 475th meeting of the Advisory Committee on Reactor Safeguards, August 29 - September 1, 2000, the Committee considered the subject draft regulatory guide and has no objections to its publication.

References:
1. U. S. Nuclear Regulatory Commission, Draft Regulatory Guide DG-1075, "Emergency Planning and Preparedness for Nuclear Power Reactors," (Proposed Revision 4 to Regulatory Guide 1.101), March 2000.
2 Nuclear Energy Institute, NEI 99-01, Final Draft Revision 4, "Methodology for Development of Emergency Action Levels," February 2000.

cc: A. Vietti-Cook, SECY
J. Craig, OEDO
G. Millman, OEDO
S. Collins, NRR
D. Matthews, NRR
C. Carpenter, NRR
J. Birmingham, NRR

UNITED STATES
NUCLEAR REGULATORY COMMISSION
ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
WASHINGTON, D.C. 20555-0001

September 7, 2000

The Honorable Richard A. Meserve
Chairman
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555-0001

Dear Chairman Meserve:

SUBJECT: ASSESSMENT OF THE QUALITY OF PROBABILISTIC RISK ASSESSMENTS

During the 475th meeting of the Advisory Committee on Reactor Safeguards, August 29-September 1, 2000, we discussed the staff's approach for addressing the issue of quality of probabilistic risk assessments (PRAs) described in SECY-00-0162. We previously met with representatives of the staff to discuss the draft Commission paper on this matter during our July 12-14, 2000 meeting. We had the benefit of the documents referenced.

Conclusions and Recommendations

1. We agree with the staff's recommendation to continue with the current process of determining the applicability of PRAs to specific regulatory applications.

2. The staff has appropriately emphasized that the quality of a PRA must be judged in the context of the regulatory decision that the PRA supports.

3. Attachment 1, "PRA Scope and Technical Attributes," to SECY-00-0162 is a useful high-level tutorial exposition of PRA elements and technical attributes. It is not a "design-to" standard, nor is it intended to be.

4. The staff should augment its collection of examples of risk-informed decisions and the requisite PRA quality to include a more diverse set of examples and should provide more details on how risk information was used. This would enable generic conclusions to be drawn regarding the role and quality of the risk information utilized in these decisions.

5. The case-study ("bottom-up") approach in Attachment 2, "PRA Quality in Risk-Informed Regulation," to SECY-00-0162 is a much needed complement to the "top-down" approach that both Attachment 1 and the American Society of Mechanical Engineers (ASME) Standard for PRAs have taken. This two-pronged approach to the issue of PRA quality is necessary for the achievement of consensus regarding this very difficult issue.

Discussion

In the Staff Requirements Memorandum dated April 18, 2000, the Commission requested the staff to provide recommendations for addressing the issue of PRA quality until the ASME and American Nuclear Society Standards have been completed, including the role of an industry PRA certification process. The staff has responded by stating that it will continue with the current process of reviewing PRAs for specific applications. The staff has provided two attachments to further elaborate on its expectations.

In our report dated July 20, 2000, we stated that the quality of PRA is at the heart of a successful risk-informed regulatory system and that PRA quality should be evaluated in the context of the decision it supports. While this recognition is realistic and appropriate, it is also the main obstacle to developing a PRA standard in the traditional sense that the engineering community normally interprets the term "standard." It is unrealistic for a standard to define a high-quality PRA as one that is of full-scope and uses detailed state-of-the-art models because many regulatory applications do not require this level of effort.

We commented on these challenges when we reviewed the proposed ASME Standard for PRA which attempted to define three categories of PRA quality. Attachment 1 of SECY-00-0162 eschews categories and provides what is necessarily a high-level description of basic PRA elements. We note that a PRA could satisfy the functional attributes listed in Attachment 1, and still be of poor quality. This is an inherent problem and is not intended as a criticism of the staff's effort.

Because the critical issue is the support for regulatory decisions, we found the discussion in Attachment 2 to be useful. The examples of PRA elements important in specific decisions were illuminating. For example, the staff states that in reviewing requests for boiling water reactor (BWR) incremental power uprates, it concluded that increased power levels would result in less time for operator actions during an accident. A PRA supporting such decisions has to include an appropriate analysis of how this shorter time would affect the progression of the relevant accidents. It would have been difficult to determine the importance of this particular PRA requirement before the need for making a decision on this issue arose.

The staff has considerable experience with a number of specific risk-informed regulatory decisions. The staff should expand Attachment 2 to provide more details on how risk information was used in such decisions and to identify common themes and frequently asked questions. Such a case-study ("bottom-up") approach is a much needed complement to the "top-down" approach that both Attachment 1 and the ASME Standard have taken. This two-

pronged approach to the issue of PRA quality is necessary to achieve consensus regarding this very difficult issue.

Sincerely,

Dana A. Powers
Chairman

References:

1. Memorandum dated July 28, 2000, from William D. Travers, Executive Director for Operations, NRC, to The Commissioners, Subject: SECY-00-0162, Addressing PRA Quality in Risk-Informed Activities.
2. Memorandum dated April 18, 2000, from Annette Vietti-Cook, Secretary, NRC, to William D. Travers, Executive Director for Operations, Subject: Staff Requirements - Briefing on Risk-Informed Regulation Implementation Plan (SECY-00-0062).
3. Letter dated July 20, 2000, from D.A. Powers, Chairman, Advisory Committee on Reactor Safeguards, to William D. Travers, Executive Director for Operations, NRC, Subject: Proposed Final ASME Standard for Probabilistic Risk Assessment for Nuclear Power Plant Applications.

UNITED STATES
NUCLEAR REGULATORY COMMISSION
ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
WASHINGTON, D.C. 20555-0001

September 8, 2000

The Honorable Richard A. Meserve
Chairman
U. S. Nuclear Regulatory Commission
Washington, D.C. 20555-0001

Dear Chairman Meserve:

SUBJECT: CAUSES AND SIGNIFICANCE OF DESIGN BASIS ISSUES AT U.S. NUCLEAR POWER PLANTS

During the 475th meeting of the Advisory Committee on Reactor Safeguards (ACRS), August 29 - September 1, 2000, we met with representatives of the Nuclear Regulatory Commission (NRC) staff to review their study of design basis issues (DBIs). The study describes trends and causes of DBIs. The ACRS had previously expressed concern that the disbanding of the Office of Analysis and Evaluation of Operational Data (AEOD) would make it difficult to retain the assessment of operational experience. Therefore, we are pleased to learn that analyses of data have been continued.

The staff examined Licensee Event Reports (LERs) to determine the level of risk of the finding, whether safety system intervention occurred, whether an actual or a potential event took place, and the consequences of the event described in the LER (failed or degraded system or train). From 1985 through 1997, the leading causes of DBIs were original design errors - 72%, procedure deficiencies - 28%, and human errors - 22% (note that more than one cause has generally contributed to each DBI).

Emergency core cooling, emergency ac/dc power, and containment and containment isolation were the safety related systems that accounted for about half of the DBIs. About 19% were potentially risk significant. Although the number of DBIs increased substantially due in part to increased scrutiny, the fraction of DBI events that qualified as accident sequence precursor events decreased from approximately 8% in 1990 to less than 1% in 1997.

The lessons learned from the data analysis support our contention that it is important not to lose the capability that resided with the former AEOD. This particular compilation of data on operational experience should have an impact on how probabilistic risk assessments (PRAs) are reviewed. The results imply that the risk contribution of design faults revealed by operational experience is limited.

The small fraction of risk significant events suggests that the criteria for what constitutes DBI-LERs should be redefined. This would reduce the burden of reporting requirements with no impact on safety.

Sincerely,

Dana A. Powers

Dana A. Powers
Chairman

Reference:
U. S. Nuclear Regulatory Commission, Office of Nuclear Regulatory Research Draft Report, "Causes and Significance of Design Basis Issues at U. S. Nuclear Power Plants," May 2000.

UNITED STATES
NUCLEAR REGULATORY COMMISSION
ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
WASHINGTON, D.C. 20555-0001

September 8, 2000

Dr. William D. Travers
Executive Director for Operations
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555-0001

SUBJECT: PROPOSED HIGH-LEVEL GUIDELINES FOR PERFORMANCE-BASED ACTIVITIES

Dear Dr. Travers:

During the 475th meeting of the Advisory Committee on Reactor Safeguards, August 29 - September 1, 2000, we met with representatives of the staff to discuss the proposed high-level guidelines for performance-based activities. Also, during our June 7-9, 2000 meeting, we discussed this matter with representatives of the staff and the Public Citizen Critical Mass Energy Project. We had the benefit of the documents referenced.

Conclusions and Recommendations

1. We support the staff's proposal to apply the guidelines for performance-based activities to an example regulation.

2. The guidelines should explicitly state that the performance levels and reliability parameters should be set at the highest practical level.

3. Guidance should be given on the extent to which multiple performance parameters that provide redundant information should be used to satisfy the defense-in-depth philosophy.

4. Expanded discussions should be provided in the guidelines of the responses to the relevant questions that appeared in the *Federal Register* Notice of May 9, 2000.

Discussion

The proposed high-level guidelines provide a systematic method to incorporate performance-based principles into regulatory activities. These guidelines are comprehensive. They incorporate important principles and strategies and Commission policy and direction. These guidelines provide consistency among new performance-based regulations and coherence with the current body of regulations. The determination of performance parameters using the proposed hierarchical structure is logical and systematic.

The guidelines are structured to, among other things, assess performance-based regulatory changes. One such guideline, II.C(2), states that:

> An assessment would be made of the performance criteria and the level in the performance hierarchy where they have been set. In general, performance criteria should be set at a level commensurate with the function being performed. In most cases, performance criteria would be expected to be set at the system level or higher.

The guidelines should explicitly state that the performance criteria should be selected at the highest practical level, and that this principle should be applied to the other guidelines. This enhances licensee flexibility and reduces regulatory burden while maintaining the appropriate level of safety.

The guidelines are also structured to assess consistency and coherence with overarching NRC goals and principles. In cases in which redundant performance parameters are identified, the staff will need to provide guidance on balancing the use of a minimum number of parameters with the use of multiple parameters consistent with the defense-in-depth philosophy. We believe that the guidelines should state the degree to which redundant, overlapping, or confirmatory performance indicators are required to validate the data or provide defense-in-depth.

The discussion of the draft guidelines in the May 9 *Federal Register* Notice contains a number of relevant questions related to the application of performance-based regulatory guidelines. In Attachment 2 to the proposed Commission paper, the staff responded to public comments related to these questions. The guidelines should more clearly identify the staff positions stated in response to these questions, because many users of the guidelines may have these same questions.

We look forward to reviewing the report to the Commission on the trial applications of these guidelines.

Sincerely,

Dana A. Powers

Dana A. Powers
Chairman

References:

1. U.S. Nuclear Regulatory Commission, "High-Level Guidelines for Performance-Based Activities [Predecisional]," Draft Commission paper from W.D. Travers, Executive Director for Operations, for the Commissioners, received August 25, 2000.
2. U.S. Nuclear Regulatory Commission, "Revised High-Level Guidelines for Performance-Based Activities (10 CFR Chapter I)," *Federal Register*, Vol. 65, No. 90, May 9, 2000, pp. 26772-26776.

3. A. Vietti-Cook, Secretary, U.S. Nuclear Regulatory Commission (NRC), memorandum to W.D. Travers, Executive Director for Operations, NRC, Subject: Staff Requirements - SECY-99-176 - Plans for Pursuing Performance-Based Initiatives, September 13, 1999.
4. D.A. Powers, Chairman, Advisory Committee for Reactor Safeguards (ACRS), NRC, letter to W.D. Travers, Executive Director for Operations, NRC, Subject: Pilot Application of the Revised Inspection and Assessment Programs, Risk-Based Performance Indicators, and Performance-Based Regulatory Initiatives and Related Matters, June 10, 1999.
5. R.L. Seale, Chairman, ACRS, NRC, Letter to L.J. Callan, Executive Director for Operations, NRC, Subject: Plans to Increase Performance-Based Approaches in Regulatory Activities, April 9, 1998.
6. L. Gue, Public Citizen's Critical Mass Energy and Environment Program, Statement to the ACRS, NRC, Subject: Revised Proposal for High-Level Guidelines for Performance-Based Regulation, June 8, 2000.
7. A. Shollenberger, Public Citizen, Critical Mass Energy Project, Letter to N.P. Kadambi, NRC, Subject: Proposed High-Level Guidelines for Performance-Based Regulation, March 7, 2000.
8. J. Riccio, Public Citizen, Critical Mass Energy Project, Letter to D.L. Meyer, NRC, Subject: High-Level Guidelines for Performance-Based Activities, March 22, 2000 .
9. S.D. Floyd, Nuclear Energy Institute, Letter to David L. Meyer, NRC, Subject: NEI Comments on Proposed Guideline for Performance-Based Activities March 24, 2000.
10. J.H. McCarthy, Virginia Power Company, Letter to D. Meyer, NRC, Subject: 10 CFR Chapter 1; High-Level Guidelines for Performance-Based Activities, March 20, 2000.
11. J.A. Hutton, Jr., PECO Energy Company, Letter to Chief, Rules and Directives Branch, NRC, Subject: Comments Concerning "High Level Guidelines for Developing Performance Based Activities," March 21, 2000.
12. D.F. Stenger, Hopkins & Sutter, Letter to D.L. Meyer, NRC, Subject: NRSG Comments on Proposed Guidelines for Performance-Based Activities, March 24, 2000.

UNITED STATES
NUCLEAR REGULATORY COMMISSION
ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
WASHINGTON, D.C. 20555-0001

September 12, 2000

Dr. William D. Travers
Executive Director for Operations
U. S. Nuclear Regulatory Commission
Washington, D.C. 20555-0001

Dear Dr. Travers:

SUBJECT: PROPOSED FINAL REGULATORY GUIDE DG-1093, "GUIDANCE AND EXAMPLES FOR IDENTIFYING 10 CFR 50.2 DESIGN BASES"

During the 475th meeting of the Advisory Committee on Reactor Safeguards, August 29 - September 1, 2000, we met with representatives of the NRC staff and the Nuclear Energy Institute (NEI) to discuss proposed final Regulatory Guide DG-1093 regarding design bases information. This guide endorses Appendix B of NEI 97-04, "Design Bases Program Guidelines," as an acceptable method of meeting NRC requirements. We also had the benefit of the documents referenced.

Recommendation

We recommend issuance of DG-1093, which endorses NEI 97-04, Appendix B.

Discussion

The term "design bases" is defined in 10 CFR 50.2 and is used in several regulations in 10 CFR Part 50, including 50.34, 50.59, Appendix A, and Appendix B. In recent years, there has been disagreement between the staff and the industry on the meaning of the10 CFR 50.2 definition. The industry effort to resolve this disagreement began with proposed guidance and examples in 1997. The NEI guidance was developed as a result of system-specific engineering inspections that showed that some licensees were not maintaining design bases information as required by NRC regulations. In response to the problems identified during these inspections and other problems identified by the licensees, many licensees initiated design bases reconstitution programs. These programs sought to identify and selectively regenerate missing documentation. During the documentation effort, it became clear that the definitions of what constituted design bases information differed from licensee to licensee. The lessons learned from events at Millstone and Maine Yankee showed that the definition of design bases should be clarified.

The ACRS briefings in October and November 1999 provided a forum for discussion of this issue. The draft regulatory guide was published for comment in April 2000. NRC staff and NEI

representatives have held numerous meetings to discuss the changes necessary to Appendix B of NEI 97-04 as a result of public comments received on the draft regulatory guide and to discuss additional changes proposed by the NRC staff in a letter to NEI dated July 18, 2000. The draft regulatory guide provides clarification of what constitutes design bases information.

We commend the staff and NEI for the completion of this difficult task through the extensive public meetings held with stakeholders in the past year.

Sincerely,

Dana A. Powers

Dana A. Powers
Chairman

References:

1. Memorandum from D. B. Matthews, Office of Nuclear Reactor Regulation, NRC, to J. T. Larkins, ACRS, Subject: Final Regulatory Guide, "Guidance and Examples for Identifying 10 CFR 50.2 Design Bases," August 7, 2000.
2. Memorandum from S. L. Magruder, Office of Nuclear Reactor Regulation, NRC, to C. A. Carpenter, Office of Nuclear Reactor Regulation, Subject: Summary of July 27, 2000 Meeting with NEI on Revision to NEI 97-04 on the Definition of 10 CFR 50.2 Design Bases, July 31, 2000.

UNITED STATES
NUCLEAR REGULATORY COMMISSION
ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
WASHINGTON, D.C. 20555-0001

September 13, 2000

The Honorable Richard A. Meserve
Chairman
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555-0001

Dear Chairman Meserve:

SUBJECT: PROPOSED RISK-INFORMED REVISIONS TO 10 CFR 50.44, "STANDARDS FOR COMBUSTIBLE GAS CONTROL SYSTEM IN LIGHT-WATER-COOLED POWER REACTORS"

During the 474th and 475th meetings of the Advisory Committee on Reactor Safeguards, July 12-14 and August 29-September 1, 2000, we met with representatives of the NRC staff, the Nuclear Energy Institute, and Performance Technology, Inc., to discuss proposed risk-informed revisions to 10 CFR 50.44 and related matters. Our Subcommittee on Reliability and Probabilistic Risk Assessment met on June 29 and July 11, 2000, to discuss these matters. We also had the benefit of the documents referenced.

<u>Background</u>

We last met with the Commission on March 2, 2000, to discuss staff plans for developing risk-informed revisions to 10 CFR Part 50 and to discuss our report dated October 12, 1999, concerning the staff's proposed Option 2 (SECY-99-256) and Option 3 (SECY-99-264) approaches. On July 20, 2000, we provided a report to the Commission on the NEI letter dated January 19, 2000, concerning the issues and priorities for NRC plans for risk-informing the technical requirements in 10 CFR Part 50.

This report responds to the Commission request in the April 5, 2000 Staff Requirements Memorandum (SRM) on these matters. It focuses on the staff's examination of 10 CFR 50.44 as a trial case for risk-informing the regulations under Option 3.

<u>Conclusions and Recommendations</u>

1. We agree with the staff's conclusion that there is little or no safety benefit associated with some of the requirements of the current 10 CFR 50.44 and that these requirements constitute unnecessary regulatory burdens.

2. The work, to date, provides sufficient basis for the development of a risk-informed 10 CFR 50.44 that can provide both a safety benefit and a reduction in unnecessary burden. We recommend that the staff be directed to proceed with rulemaking.

3. Because the study of 10 CFR 50.44 is intended to be illustrative of a general approach, the discussion of how risk information was used to develop the results on the conditional large release probabilities should be expanded.

Discussion

In the SRM dated February 3, 2000, the Commission approved the staff's plan to risk-inform the technical requirements of 10 CFR Part 50 (Option 3). In accordance with that plan, the staff has developed a draft framework document for risk-informed changes to 10 CFR 50. The staff used the processes described in the framework document to develop recommendations for risk-informed changes to 10 CFR 50.44 for the control of hydrogen and carbon monoxide that could burn or detonate, thereby challenging the integrity of the containment.

We were briefed on the development of the proposed framework document during our July 12-14, 2000 meeting. Subsequently, we received an updated draft revision 2 of the framework document. This document continues to evolve, and we have not yet had sufficient opportunity to review it. Although we wish to discuss the details of the framework with the staff, we agree that it is appropriate for the staff to begin trial application of the framework for the development of risk-informed changes to specific regulations.

The initial application of the processes described in the framework was to develop recommendations for changes to 10 CFR 50.44. The draft version of the staff study of a risk-informed approach to 10 CFR 50.44 provides an excellent discussion of the development and implementation of the current 10 CFR 50.44 and its relationship to other regulations and implementing documents. It also provides a useful summary of the risk significance of combustible gases and effectively characterizes the important issues. Because it is intended to be illustrative of how risk information can be used to develop alternatives to current regulations, the discussion of how the risk information in NUREG-1150 and NUREG-1560 was used to develop the conditional large release probabilities should be expanded. It would be helpful, for example, to identify the dominant sequences leading to containment failure due to combustible gases for a representative set of plants, to compare the findings from studies of severe accident risks (NUREG-1150) and from the individual plant examinations (NUREG-1560) and to better explain the reasoning that was used in the development of the conditional large release probabilities for the various classes of containments (Tables 4-2, 3, and 4 of the 10 CFR 50.44 study). More specific references to NUREG-1150 are also needed to make the study a proper technical basis document for the development of a risk-informed 10 CFR 50.44.

The staff presented specific recommendations for the elimination, modification, or enhancement of some of the current requirements in 10 CFR 50.44. In addition, the staff proposes to specify in the regulation a combustible gas source term based on realistic calculations for risk-significant severe accident sequences. A performance-based alternative would be provided to allow the licensee to use plant-specific analyses to demonstrate that the plant would meet specified performance criteria (e.g., maintenance of containment integrity for at least 24 hours for all risk-significant events). The staff also recommends that long-term (greater than 24

hours) combustible gas control be included as part of the Severe Accident Management Guidelines to mitigate the possibility of a large, late radionuclide release.

We agree with the staff's assessment on the risk-significance of combustible gas control for the various types of containments and believe that the work, to date, provides the basis for the development of a risk-informed 10 CFR 50.44 that can provide both a safety benefit and a reduction in unnecessary burden for licensees. The staff should be directed to proceed with rulemaking. The results of this study should assist the disposition of the petition for rulemaking that came from the submission by Performance Technology, Inc.

We look forward to reviewing revisions to the framework document. We also look forward to reviewing the staff's proposed rulemaking (Option 2) associated with the special treatment requirements for structures, systems, and components.

Sincerely,

Dana A. Powers

Dana A. Powers
Chairman

References:

1. Draft memorandum received August 18, 2000, from William D. Travers, Executive Director for Operations, NRC, to The Commissioners, Subject: Status Report on Risk-Informing the Technical Requirements of 10 CFR Part 50 (Option 3) and Recommendations on Risk-Informing 10 CFR 50.44 (Combustible Gas Control).
2. Memorandum dated February 3, 2000, from Annette Vietti-Cook, Secretary, NRC, to William D. Travers, Executive Director for Operations, NRC, Subject: Staff Requirements - SECY-99-264 - Proposed Staff Plan for Risk-Informing Technical Requirements of 10 CFR Part 50.
3. Letter dated April 18, 2000 from Steven D. Floyd, Nuclear Energy Institute, to Thomas L. King, Office of Nuclear Regulatory Research, NRC, Subject: Industry Comments on Draft NRC Framework for Risk-Informing NRC Technical Requirements, and Draft NRC Report on Risk-Informing 10 CFR 50.44, "Standards for Combustible Gas Control Systems in Light-Water-Cooled Power Reactors."
4. Report dated October 12, 1999, from Dana A. Powers, Chairman, ACRS, to Greta Joy Dicus, Chairman, NRC, Subject: Proposed Plans for Developing Risk-Informed Revisions to 10 CFR Part 50, "Domestic Licensing of Production and Utilization Facilities."
5. Report dated July 20, 2000, from Dana A. Powers, Chairman, ACRS, to Richard A. Meserve, Chairman, NRC, Subject: Nuclear Energy Institute Letter dated January 19, 2000, Addressing NRC Plans for Risk-Informing the Technical Requirements in 10 CFR Part 50.
6. U. S. Nuclear Regulatory Commission NUREG-1150, Vols. 1-3, "Severe Accident Risks: An Assessment for Five U.S. Nuclear Power Plants," Final Report, December 1990.
7. U. S. Nuclear Regulatory Commission NUREG-1560, Vols. 1-5, "Individual Plant Examination Program: Perspectives on Reactor Safety and Plant Performance," Final Report, December 1997.

UNITED STATES
NUCLEAR REGULATORY COMMISSION
ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
WASHINGTON, D.C. 20555-0001

September 14, 2000

The Honorable Richard A. Meserve
Chairman
U.S. Nuclear Regulatory Commission
Washington, DC 20555-0001

Dear Chairman Meserve:

SUBJECT: PRE-APPLICATION REVIEW OF THE AP1000 STANDARD PLANT DESIGN - PHASE I

During the 475th meeting of the Advisory Committee on Reactor Safeguards, August 29–September 1, 2000, we discussed the results of the staff's pre-application (Phase I) review of the Westinghouse Electric Company's proposed AP1000 Standard Plant Design. During this meeting, we had the benefit of discussions with representatives of the staff and of the documents referenced. A list of our issues that need to be addressed during the AP1000 pre-application review was sent to the NRC Executive Director for Operations on June 21, 2000.

Background

Westinghouse plans to seek certification of a 1000 MWe nuclear plant similar to the certified AP600 design, and seeks NRC feedback on the scope and cost for review and certification of the AP1000 design. The NRC and Westinghouse have agreed to a three-phase review approach. Phase I is to: identify the review assumptions and issues that need to be evaluated; identify the information necessary to evaluate the assumptions and issues; estimate the resources required to perform the Phase II review; and provide a schedule for the certification review.

In a letter dated May 31, 2000, Westinghouse identified five "fundamental assumptions" for evaluation by the staff during Phase II review:

1. The AP1000 Design Certification Application will reference sections of the AP600 Design Control Document that do not change for AP1000.

2. The AP1000 Design Certification Application will not require additional tests to be performed by the applicant.

3. The AP1000 Design Certification Application can utilize the AP600 analysis codes with limited modifications.

4. The AP1000 Design Certification Application can utilize the AP600 probabilistic risk assessment (PRA) supplemented with a sensitivity study to meet the requirements for a plant-specific PRA.

5. The AP1000 Design Certification Application can defer selected design activities to the Combined License (COL) applicant.

In its Phase I assessment, the staff addressed these assumptions and provided Westinghouse with expectations on information that must be provided to the staff to assess the validity of these assumptions.

Recommendations

1. The PRA should include uncertainty distributions on core damage frequency, conditional containment failure probability (CCFP), and large, early release frequency (LERF).

2. The seismic analysis should not be left solely to the COL applicant and should be included in the PRA using a representative site.

3. The applicant's results from the codes NOTRUMP, WCOBRA/TRAC, LOFTRAN, and WGOTHIC for the design basis accidents should be accompanied by uncertainty assessments.

4. The staff should obtain and exercise the above codes to assist its independent evaluation and validation of these codes.

Discussion

The staff has done a commendable job of determining the information it will need to assess the five assumptions proposed by Westinghouse, and we generally agree with the staff's initial positions on these assumptions. We are concerned, however, that the staff may not be requesting sufficient information to conduct the certification review without undue reliance on judgment. Because the applicant does not plan to perform additional tests, certification of the AP1000 will be more dependent on the results of analyses than was the case for the AP600.

In a Staff Requirements Memorandum of July 21, 1993, the Commission approved the use of a CCFP goal of 0.1 along with a containment performance goal for advanced light-water reactor designs. Westinghouse, for points of reference in development of the AP600 PRA, used a LERF goal of 10^{-6} per year as well as the CCFP goal of 0.1.

The AP600 PRA reported an overall LERF of about 10^{-8} per year and a CCFP of about 0.1. While this low value of LERF was comforting, it was based on new systems and components [passive emergency core cooling system (ECCS) combined with active systems, reactor vessel external flooding, etc.] for which there was little experience. Thus, the CCFP and LERF results for the AP1000 are likely to be subject to much greater uncertainty than that associated with current operating plant PRA results. With "reasonable" variation of parameters, the staff estimated that the AP600 CCFP could have easily been 0.5 at a reasonable confidence level. The design changes along with the increased plant size and power rating of the magnitude

proposed will negatively impact both the LERF and the CCFP as well as increase the uncertainties associated with these acceptance parameters.

Increasing the height of the containment and the quantity of water in the tank on top may well increase the vulnerability of the AP1000 containment to seismic events. Both selections of site characteristics and seismicity are challenges to the conduct of a PRA for the AP1000 that includes external event initiators. It is most important that artificial uncertainty not be injected into the PRA results by including bounding ranges of site characteristics and seismicities. A representative site and representative seismicity for the recommended PRA would be satisfactory.

We are concerned that the AP1000 defense in depth associated with a CCFP goal of 0.1 might be unduly compromised by the increase in plant size and the uncertainties could be much greater than those for the AP600. If the staff is to properly assess the AP1000 design with respect to acceptance values of risk metrics and its compliance with the defense-in-depth philosophy, the PRA will need to include an uncertainty analysis. Without such a PRA, we will be faced with insufficient information on which to base our judgment on the defense-in-depth acceptability of the AP1000 containment.

Our second concern relates to the deterministic part of the design certification. The acceptability of the AP600 for certification with respect to the design basis deterministic aspects was partially based on the use of computer codes with validation based on data from separate effects and integral tests.

The AP600 certification was also partially approved on the basis that the scaled integral experiments demonstrated the robustness of the AP600 ECCS for keeping the core covered over the entire period of the design basis accident sequences. It is likely that this level of comfort will be eroded for the AP1000 because of scaling issues that could make the integral tests no longer directly applicable to the full-scale design. Thus, for the AP1000 there will be much greater reliance on the code results. The concern involves, then, the use of codes that have not been validated for the AP1000 conditions to determine margins.

In past licensing reviews, the staff has been content to use a process in which conservative analyses were used to demonstrate that acceptance criteria (e.g., peak clad temperature) could be met. This process could be used because extensive experience and experimental data were available to substantiate the judgment that the analyses were indeed conservative. Extensive experience and data are not available for passive plants. For the AP600, correctly scaled experiments were performed that demonstrated the robustness of the emergency core cooling. If the scaling of these experiments proves to be less satisfactory for the AP1000, greater reliance on thermal-hydraulic codes will be required.

The use of the predictive codes NOTRUMP, WCOBRA/TRAC, LOFTRAN, and WGOTHIC has been approved only for the AP600, and the validity of these codes for application to the AP1000 must be determined. The available experimental data relevant to passive flow conditions may not be sufficient to validate the use of these codes for the AP1000 geometry and conditions. The applicant intends to conduct a detailed scaling analysis to demonstrate the sufficiency of these experimental data for the AP1000.

If the scaling analysis is less than satisfactory, it will be necessary to determine the uncertainties of the predictions of the codes NOTRUMP, WCOBRA/TRAC, LOFTRAN, and WGOTHIC in a technically defensible manner. This could even necessitate additional, properly scaled experiments to provide confidence that the calculated figures of merit are conservative.

In any case, it will be necessary to assess the uncertainty and validation analysis of the codes provided by Westinghouse. The staff should acquire and exercise these codes so that it can independently evaluate the sensitivity of their predictions to assumptions, model idealizations, and choices of parameters in the correlations.

Sincerely,

Dana A. Powers

Dana A. Powers
Chairman

References:

1. Memorandum dated July 27, 2000, from Samuel J. Collins, Director, Office of Nuclear Reactor Regulation, U. S. Nuclear Regulatory Commission, to W. E. Cummins, Westinghouse Electric Company, Subject: AP1000 Pre-Application Review - Phase One.
2. Memorandum dated May 31, 2000, from M. M. Corletti, Westinghouse Electric Company, to Document Control Desk, U. S. Nuclear Regulatory Commission, Subject: AP1000 Pre-Application Review Items.
3. Memorandum dated June 21, 2000, from John T. Larkins, Executive Director, Advisory Committee on Reactor Safeguards, to William D. Travers, Executive Director for Operations, NRC, Subject: AP1000 Pre-Application Review.
4. Memorandum dated July 21, 1993, from Samuel J. Chilk, Secretary of the Commission, for James M. Taylor, Executive Director for Operations, NRC, Subject: SECY-93-087 - Policy, Technical, and Licensing Issues Pertaining to Evolutionary and Advanced Light-Water Reactor (ALWR) Designs.
5. U. S. Nuclear Regulatory Commission, Final Safety Evaluation Report Related to Certification of the AP600 Design, Vol. 2, September 1998.

UNITED STATES
NUCLEAR REGULATORY COMMISSION
ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
WASHINGTON, D.C. 20555-0001

October 11, 2000

The Honorable Richard A. Meserve
Chairman
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555-0001

Dear Chairman Meserve:

SUBJECT: UNION OF CONCERNED SCIENTISTS REPORT, "NUCLEAR PLANT RISK STUDIES: FAILING THE GRADE"

During the 475th and 476th meetings of the Advisory Committee on Reactor Safeguards, August 30 - September 1 and October 5-7, 2000, we met with a representative of the Union of Concerned Scientists (UCS) concerning the report entitled, "Nuclear Plant Risk Studies: Failing the Grade," issued in August 2000 [Ref. 1]. We reviewed this report for two reasons. First, the UCS report asserts that NRC decision making is increasing risk for the American people. Second, the report criticizes a major current initiative of the agency, namely, risk-informing the regulations. In support of the Commission's objective of building and maintaining public trust and confidence in regulatory decisions [Ref. 2], we offer the following comments regarding the UCS report.

Conclusions

1. The UCS report's assertion that "the risk assessments are seriously flawed and their results are being used inappropriately to increase – not reduce – the threat to the American public" is not valid.

2. The UCS report's claim that consequences of potential reactor accidents are not evaluated is not valid. Many probabilistic risk assessments (PRAs) calculate consequences, and the NRC has sponsored PRAs that have resulted in extremely detailed assessments of consequences and their associated uncertainties.

3. The UCS description of PRA is misleading.

4. The UCS report's list of "unrealistic assumptions" is not accurate. The report exaggerates their significance and ignores the agency's ongoing efforts to assess the validity of the data used in PRAs.

5. The report correctly identifies the need for PRA quality standards, but fails to mention the significant efforts under way to develop such standards.

6. Disparate results from "sister" plants are interpreted in the report as reflecting inadequacies in PRAs, but often, in fact, reflect differences in the design of the plant and in operating practices. The sources of these differences are investigated by the NRC staff when these PRAs are used in decision making.

7. The statement that "it is not possible to properly manage risk when only *reasonable* – instead of *all possible* – measures are taken to prevent and mitigate events unless the probabilities and consequences are accurately known" is unrealistic. No risk issue is ever managed by taking "all possible measures" to prevent and mitigate risk.

8. We disagree with the recommendation that the use of risk information should be disallowed until the methodology includes the improvements recommended by the UCS report. It would be a disservice to the nation if the agency ignored the benefits provided by the continued use of this technology.

9. The author of the UCS report was forced to rely on summary results derived from Individual Plant Examination (IPE) submittals without having access to the supporting PRAs. The NRC needs to facilitate public access to PRAs and risk information used in regulatory decisions.

Background

We briefly reviewed the history of the evolution of reactor safety philosophy to allow a better understanding of the impact that PRA has had on reactor safety assessments.

In the early days of nuclear power development (in the 1950s and 1960s), both the industry and the regulators recognized that large uncertainties existed in the assessment of the consequences of potential reactor accidents. A nuclear safety philosophy to both prevent and mitigate the consequences of these potential accidents evolved, but the resulting degree of safety could only be determined by subjective judgment. The cornerstones of this safety philosophy were defense in depth and safety margins.

Defense in depth is an element of the NRC's safety philosophy that employs successive compensatory measures to prevent accidents or mitigate damage if a malfunction, accident, or naturally caused event occurs at a nuclear facility [Ref. 3]. Safety margins, i.e., the differences between the failure and the operating points, were purposely made large in order to accommodate a range of uncertainties.

The first major reactor PRA, the Reactor Safety Study, was published in 1975 [Ref. 4]. Subsequently, a number of important developments occurred. Major research programs were established to advance PRA methodology. Additional PRAs were completed both in the United States and internationally, which added greatly to the understanding of the potential risks of nuclear power systems and the maturity of the methodology. These PRA developments have changed the views on how to manage reactor safety in several fundamental ways:

1. Plants can be analyzed as integrated systems through the systematic development of accident sequences. The fundamental questions: "What can go wrong?", "How likely is it?", and "What are the consequences?" [Ref. 3] can be addressed. Unlike conventional analyses that are based on a single failure, these sequences consider multiple failures, including hardware failures and human errors, as well as physical phenomena, and any other factors that are thought to affect the progression of the accident. This approach permits a more in-depth analysis of plant behavior.

2. The analysis of facilities as integrated systems has identified a number of important safety improvements. Examples are the requirement to automate the initiation of the auxiliary feedwater systems, and, in part, the rules to address accident sequences initiated by anticipated transients without scram and station blackout.

3. Thousands of accident sequences are considered in a PRA in contrast to the relatively small number of design-basis accidents considered in conventional analyses. Even so, completeness remains an issue: are there accident sequences that have not been considered? The broad application of PRA by diverse practitioners has made it unlikely that any major contributors to risk have not been identified. Experience shows that the systematic search for accident sequences produces a far more complete picture of the way failures can occur in complex systems.

4. The probabilities of accident sequences can be quantified. This allows the estimation of risk metrics such as the frequency of severe damage to the reactor core, the frequency of release of large amounts of radioactivity, and the probability of death of an individual living near the plant. Accident sequences can be ranked according to their contribution to risk.

5. Although thousands of accident sequences are considered in a PRA, it has been found that the risk is dominated by relatively few sequences. The identification of dominant sequences and risk-significant events provides valuable insight. By focusing resources on these, risk can be more effectively managed. An example is the significant decline in the rate of common-cause failures over the years [Ref. 5].

In over two decades of development following the Reactor Safety Study, PRA reached a level of maturity that allows it to be used to identify unnecessary regulatory burden, as well as additional safety improvements. It is unfortunate that the two uses of PRA (to impose burden, when necessary, and to remove it, when unnecessary) have been separated by time, because this may create the false impression that burden reduction is the primary use of risk information.

The Commission issued the PRA Policy Statement in 1995 [Ref. 6] directing the staff to use PRA insights in all regulatory matters to the extent supported by the state of the art. Following this, Regulatory Guide 1.174 [Ref. 7] was issued establishing a framework for using PRA in risk-informed decisions on plant-specific changes to the licensing basis. Regulatory Guide 1.174 and the associated Standard Review Plan Chapter 19 [Ref. 8] enabled licensees to include risk information in the justification of license changes.

Regulatory Guide 1.174 proposes a "risk-informed" approach in which PRA insights (including quantitative results) are one set of inputs to an "integrated decision-making process." This

process must also consider traditional engineering safety analyses, defense in depth, and maintenance of adequate safety margins. Performance after such changes in the licensing basis must be monitored to detect unanticipated consequences. Regulatory Guide 1.174 also recognizes that there are uncertainties in PRAs and provides guidance for licensees to assess how these might affect the decision-making process.

Discussion of the UCS Report

The UCS report overstates the reliance of the Regulatory Guide 1.174 decision-making process on quantitative PRA results (the verb "rely" appears in numerous places in the UCS report). It gives passing mention to "risk-informed regulation" (page 22), but does not elaborate on what it is. The inclusion of additional information in the decision-making process and of the requirement to monitor performance after changes are made are not discussed.

The UCS report claims that PRAs are not full risk assessments because potential accident consequences are not evaluated. This is not true. Many PRAs calculate these consequences, e.g., the population dose and the number of prompt and latent cancer fatalities. NUREG-1150 [Ref. 9] involved very detailed assessments of the consequences of nuclear accidents and the associated uncertainties. It showed that the consequences were very site-specific and subject to large uncertainty. For these reasons, it is appropriate to introduce the core damage frequency (CDF) and the large early release frequency (LERF) as surrogate metrics appropriate for decision making regarding most plant modifications. CDF and LERF reflect plant design and operation. Core damage itself is an undesirable event and is, in fact, necessary for serious consequences to occur. Thus, preventing core damage is both wise and an appropriate application of an effective defense-in-depth philosophy. The relationship of LERF to prompt fatalities has been studied and is well understood [Ref. 10]. The numerical goal for LERF used in Regulatory Guide 1.174 has been shown to be consistent with the NRC safety goal on prompt fatalities. While we believe that CDF and LERF are useful metrics for regulatory applications, we hope that in the future it will be possible to have complete Level 3 PRAs for every plant so that complete risk profiles will be available [Ref. 11].

The UCS report provides an unsatisfactory description of PRA. The "fault" trees referred to in the report are normally called "event" trees. The fact that it is the *conditional* probabilities of the branches that are multiplied together to give the probability of an accident sequence is not mentioned. The fact that these conditional probabilities are produced from detailed fault trees that search for potential system failure modes is also not mentioned. A critical assessment of a technology should have a better discussion of its basic elements. In addition, one would expect that a report concerned with the potential uncertainties in PRA results would reference studies like NUREG–1150, which contains very detailed and comprehensive discussion of potential uncertainties and how they affect PRA results. The report's description fails to reflect the depth of analysis that goes into constructing a PRA.

It is very misleading to list the number of regulatory violations (Table 1, page 7) and the number of design problems (Table 2, page 8) in each year without providing any evaluation of their impact on safety. The NRC regularly evaluates the safety significance of violations and design deficiencies that have been identified. A recent study of design basis violations showed that in 1990 about 8% had some safety significance, i.e., could potentially result in a change in the CDF

on the order of several events per million reactor years. In 1998, only about 1% had safety significance [Ref. 12].

Although PRAs generally do not explicitly include aging, the critical issue is whether there is any evidence that the failure rates assumed in the PRAs are unrealistic. For passive components like piping, steam generator tubes, and valve bodies that are not subject to periodic testing, extensive work has been done to characterize the degradation that occurs due to fatigue, general corrosion, stress corrosion cracking, thermal aging, and erosion-corrosion. When necessary, additional inspections are performed as part of reactor aging management programs. Both analysis and experience demonstrate that these aging management programs are succeeding in maintaining values of failure rates and failure probabilities consistent with those assumed in the PRAs.

PRAs do not assume that reactor pressure vessels (RPVs) never fail as claimed in the UCS report. Conservative estimates show that as-fabricated pressure vessels have very low probabilities of failure. Because the probability is so low, sequences involving RPV failure (the "R sequences") make negligible contribution to the total CDF. Irradiation does embrittle the RPV. This embrittlement is well understood and is carefully monitored. Although the probability of vessel failure does increase with time, conservative regulations ensure that the frequency of RPV failure remains well below five failures per million reactor-years. Thus, vessel failure still makes a negligible contribution to the total CDF.

The claim that PRAs assume that "plant workers will not make serious mistakes" is false. In fact, many experts believe that PRAs do not give operators the credit they deserve. As contrasted to conventional analyses that do not consider human error, it was PRA technology that focused attention on the significance of human error. Some IPEs made very optimistic assumptions about human error probabilities, but NRC review identified those immediately. Human error analysis is still one of the larger sources of uncertainty in PRA results, and the PRA community is actively pursuing better models to describe human performance. The NRC has been a major supporter of such research efforts.

The UCS report fails to mention the agency's ongoing efforts to assess the validity of the data used in PRAs. Work sponsored by the former Office for Analysis and Evaluation of Operational Data and the Office of Nuclear Regulatory Research compared actual plant performance with IPE estimates. The conclusion has been that IPE estimates are sometimes higher and sometimes lower than the estimates based on experience. The NRC staff has investigated the reasons for these differences. It is important to note that the significance of such differences has to be evaluated in the context of the PRA accident sequences. The NRC is using the data-based estimates in its evaluations.

Current PRAs are frequently limited in scope, i.e., they only analyze the behavior of the plants for full-power operation; they do not treat the effects of some initiators, such as fire and those at shutdown operations, as completely as others. In addition, other potential accidents not involving the core, such as those involving spent-fuel pools, should also be assessed, using PRA techniques. We believe that the development and expansion of PRA technology is needed, but this should not inhibit the use of PRA for problems within the scope of the currently available PRAs. Most practitioners know that one does not always need a "perfect" PRA to gain important insights regarding plant safety.

We agree that standards establishing minimum requirements for PRA quality are necessary to reduce the staff effort required to assess the quality of PRAs used for risk-informed decision making. We are surprised that the UCS report does not mention the ongoing significant efforts by the American Society of Mechanical Engineers, the American Nuclear Society, the National Fire Protection Association, and the NRC to develop such standards. Industry is also undertaking programs to assess the quality of existing PRAs.

The report concludes that the differences in PRA results between "sister" plants raise questions about the quality of PRAs. Sister plants are not necessarily identical. For instance, St. Lucie Units 1 and 2 are sister plants that have significant differences in their cores. Unit 1 has 14 x 14 fuel rod assemblies whereas Unit 2 has 16 x 16 fuel rod assemblies. This resulted in many more control rods in Unit 2 and in associated changes in the configuration and drive systems. Sister plants also tend to have differences on the secondary side either because they were built by different architect engineering firms or because the owners chose different configurations for the supporting systems. These differences in sister plants should and do have an impact on the PRA results. These results can also be affected by differences in emergency and abnormal operating procedures, management processes and practices, and control room layout. Of course, the team of analysts performing the PRA and the approach they use also affect the results. Similar issues were raised and investigated when the NRC staff reviewed the IPEs [Ref. 14]. We anticipate that having a standard for PRA quality will reduce the current sensitivity of PRA results to the team doing the analysis.

Are there deficient PRAs out there? Yes. It is very doubtful, however, that they will be used in risk-informed regulatory decisions. The greater the reliance on risk information in regulatory decision making, the greater the scrutiny of the PRA. PRA quality is evaluated in the context of the decision the PRA supports, as it should be.

The UCS report argues that "it is not possible to manage risk when only *reasonable* – instead of *all possible* – measures are taken to prevent and mitigate events unless the probabilities and consequences are accurately known." No society, including our own, takes all "possible" measures to prevent and mitigate accidents. Societies do what they deem to be reasonable even when the relevant probabilities and consequences of accidents are not known quantitatively. To demand that these quantities be known "accurately" is meaningless as a general statement. Risk management, by its very nature, deals with uncertainty regardless of its magnitude.

The UCS report relies on summary results derived from IPE submittals by licensees. These IPE submittals are now substantially out of date. They did not have the qualification and scrutiny expected to be currently required for any risk assessment information that is submitted in support of a risk-informed request from a licensee. The author of the UCS report had to rely on the outdated IPE results because the updated PRAs are not publicly available. This raises the question of how to best provide the public with ready access to detailed risk assessments that will be used to support licensee requests. Without ready access to these risk analyses, the public may not have confidence in regulatory decisions that use risk information. This situation should be rectified.

We disagree with the UCS report's recommendation that risk information should not be used until all of the requirements listed in the report are met. Current PRAs provide the best available understanding of the potential risks. There are definite benefits to society from the use of risk

information in the regulation of nuclear reactors, and it would be a disservice to the nation if the agency ignored the valuable insights that this technology provides.

Sincerely,

Dana A. Powers

Dana A. Powers
Chairman

References:

1. Union of Concerned Scientists, "Nuclear Plant Risk Studies: Failing the Grade," David Lochbaum, August 2000.
2. U.S. Nuclear Regulatory Commission, NUREG-1614, Vol. 2, Parts 1 and 2, FY2000-2005 NRC Strategic Plan, September 2000.
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4. U.S. Nuclear Regulatory Commission, NUREG-74/014, "Reactor Safety Study, An Assessment of Accident Risks in the U.S. Nuclear Power Plants, WASH-1400," October 1975.
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7. U.S. Nuclear Regulatory Commission, Regulatory Guide 1.174, "An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis," July 1998.
8. U.S. Nuclear Regulatory Commission, NUREG-0800, Standard Review Plan Chapter 19.0, "Use of Probabilistic Risk Assessment in Plant-Specific, Risk-Informed Decisionmaking: General Guidance," July 1998.
9. U. S. Nuclear Regulatory Commission, NUREG-1150, Volume 3, "Reactor Risk Reference Document," February 1987.
10. Report dated April 11, 1997, from R. L. Seale, Chairman, Advisory Committee on Reactor Safeguards, to Shirley Ann Jackson, Chairman, NRC, Subject: Risk-Based Regulatory Acceptance Criteria for Plant-Specific Application of Safety Goals.
11. ACRS Report dated November 18, 1996, from T. S. Kress, Chairman, ACRS, to Shirley Ann Jackson, Chairman, NRC, Subject: Plant-Specific Application of Safety Goals.
12. U. S. Nuclear Regulatory Commission, Office of Nuclear Regulatory Research, Draft Report, "Causes and Significance of Design Basis Issues at U. S. Nuclear Power Plants," May 2000.
13. U. S. Nuclear Regulatory Commission, NUREG/CR-5750, "Rates of Initiating Events at U.S. Nuclear Power Plants: 1987 - 1995," February 1999.
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UNITED STATES
NUCLEAR REGULATORY COMMISSION
ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
WASHINGTON, D.C. 20555-0001

October 11, 2000

MEMORANDUM TO: William D. Travers
Executive Director for Operations

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

SUBJECT: PROPOSED REVISION TO 10 CFR 73.55, "REQUIREMENTS FOR PHYSICAL PROTECTION OF LICENSED ACTIVITIES IN NUCLEAR POWER REACTORS AGAINST RADIOLOGICAL SABOTAGE"

During the 476th meeting of the Advisory Committee on Reactor Safeguards, October 5-7, 2000, the Committee considered a proposed revision to 10 CFR 73.55. The staff presented the fundamental concepts of the proposed revision to the Committee at the 472nd ACRS meeting, May 11-13, 2000. The Committee decided not to review the proposed revision and has no objections to the staff issuing it for public comments. The Committee would like the opportunity to review the proposed final revision after the staff has resolved public comments.

References:
Memorandum from Bruce A. Boger, Office of Nuclear Reactor Regulation, to John T. Larkins, Executive Director, ACRS, Subject: ACRS Review of a Proposed Rulemaking Package to Revise 73.55, issued September 1, 2000.

cc: A. Vietti-Cook, SECY
J. Craig, OEDO
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UNITED STATES
NUCLEAR REGULATORY COMMISSION
ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
WASHINGTON, D.C. 20555-0001

October 12, 2000

Dr. William D. Travers
Executive Director for Operations
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555-0001

Dear Dr. Travers:

SUBJECT: PRESSURIZED THERMAL SHOCK TECHNICAL BASIS REEVALUATION PROJECT

During the 476th meeting of the Advisory Committee on Reactor Safeguards, October 5-7, 2000, we reviewed the status of activities associated with the Pressurized Thermal Shock (PTS) Technical Basis Reevaluation Project. During our 472nd meeting, May 11-13, 2000, we reviewed a draft Commission Paper concerning revising the technical basis for the PTS screening criterion. Our Subcommittee on Materials and Metallurgy met on September 21, 2000, to review draft reports associated with the Reevaluation Project. During our reviews we had the benefit of discussions with the NRC staff and of the documents referenced.

Recommendations and Conclusions

1. The staff should examine the implications of using a large early release frequency (LERF) acceptance guideline based on an air-oxidation source term on the acceptance value for reactor pressure vessel failure frequency.

2. The reevaluation of the PTS rule is a well thought out project that integrates the results of thermal-hydraulic, probabilistic risk assessment, radiation damage, probabilistic fracture mechanics, and materials characterization studies. The work is still in progress, but it appears to be proceeding well.

Discussion

The PTS rule 10 CFR 50.61 was issued in 1983 as an "adequate protection" rule. A class of transient events had been identified in which a rapid cooldown of the reactor vessel coincides with high internal pressure. Such events produce high stresses and relatively low temperatures at the inner surface of the vessel. Because of the high stresses and the reduced fracture toughness near the inner surface, preexisting flaws can propagate through the wall and cause vessel failure. The probability of vessel failure is the product of the probability that a PTS initiating event will occur and the conditional probability of failure given an event. The staff established a screening criterion based on critical values for the reference temperature (RT_{NDT}) that can be used to characterize the toughness of reactor vessels at temperatures associated

with PTS events. The screening criterion was intended to ensure that the frequency of throughwall vessel failure was less than 5×10^{-6} events per reactor year. This throughwall vessel failure frequency value was also used as an acceptance value in Regulatory Guide 1.154, which provides the basis for a more detailed assessment of the probability of vessel failure if the PTS screening criterion is exceeded.

As part of the reevaluation of the PTS rule, the staff is reassessing the acceptance value for vessel failure because of the additional risk assessment experience that has been developed since the PTS rule was established in 1983. The staff described some of its considerations in determining an acceptance value in SECY-00-0140. This SECY paper discusses the choice of the acceptance value in terms of the core damage frequency and LERF acceptance guidelines used in Regulatory Guide 1.174. As we noted in our report on spent fuel pool accident risk dated April 13, 2000, the LERF acceptance guideline in Regulatory Guide 1.174 was based on the prompt fatalities associated with a steam-oxidation driven source term. In the cases of a spent fuel pool fire or a reactor pressure vessel failure due to PTS, core damage will occur in air rather than steam. The staff currently is considering the implications of air oxidation on the source term in its evaluation of spent fuel pool cooling. The staff also should consider the implications of an air-oxidation source term on the choice of an acceptance value for reactor pressure vessel failure frequency.

The staff's effort to update the Fracture Analysis of Vessels: Oak Ridge (FAVOR) probabilistic fracture mechanics code is nearing completion. The updated code is now capable of realistic descriptions of neutron fluence, material variability, flaw distributions, and fracture toughness. The staff has chosen not to update the code to include azimuthal variations in temperature based on the assumption that the thermal-hydraulic experiments at the APEX facility will confirm that the temperatures in the beltline region during a PTS event are nearly axisymmetric. Updating the FAVOR code is an important achievement of the PTS reevaluation project.

The staff reported on its continuing efforts to develop more realistic descriptions of flaw distributions in reactor pressure vessels, to characterize uncertainties in fracture toughness distributions, and to validate the thermal-hydraulic models for PTS events. This is work in progress and so it is premature for us to evaluate it. It appears, however, that the project integrates probabilistic risk assessment, probabilistic fracture mechanics, and thermal hydraulics well. We look forward to additional discussions with the staff on its efforts to provide a comprehensive estimate of the uncertainties in PTS calculations by integrating the treatment of uncertainties through a PRA-type analysis.

Sincerely,

Dana A. Powers

Dana A. Powers
Chairman

References:

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to Pressurized Thermal Shock," issued in the Proceedings from the IAEA Specialists' Meeting, "Methodology and Supporting Research for the Pressurized Thermal Shock Evaluation," Rockville, MD, July 2000.

2. U. S. Nuclear Regulatory Commission, Office of Research, Draft, "Report on the Results of the Expert Judgment Process for the Generalized Flaw Size and Density Distribution for Domestic Reactor Pressure Vessels," received September 7, 2000.
3. E. D. Eason, and J. E. Wright, Modeling & Computing Services, Draft Report, "Updated Transition Temperature Shift Model," dated July 28, 2000.
4. U. S. Nuclear Regulatory Commission, report prepared by Oak Ridge National Laboratory, "Technical Basis for Statistical Models of Extended K_{Ic} and K_{Ia} Fracture Toughness Databases for RPV Steels," dated February 2000.
5. F. Li, et. al., University of Maryland, "K_{Ic} / K_{Ia} Uncertainty Characterization," dated June 23, 2000.
6. Report from Dana A. Powers, Chairman, ACRS, to Richard A. Meserve, Chairman, NRC, Subject: Draft Final Technical Study of Spent Fuel Pool Accident Risk at Decommissioning Nuclear Power Plants, dated April 13, 2000.
7. U.S. Nuclear Regulatory Commission, Regulatory Guide 1.154, "Format and Content of Plant-Specific Pressurized Thermal Shock Safety Analysis Reports for Pressurized Water Reactors," January 1987.
8. U. S. Nuclear Regulatory Commission, Regulatory Guide 1.174, "An Approach for Using Probabilistic Risk Assessment In Risk-Informed Decisions On Plant-Specific Changes to the Licensing Basis," July 1998.
9. U.S. Nuclear Regulatory Commission, SECY-00-0140, memorandum from William D. Travers, Executive Director for Operations, NRC, for the Commissioners, Subject: Reevaluation of the Pressurized Thermal Shock Rule (10 CFR 50.61) Screening Criterion, dated June 23, 2000.

UNITED STATES
NUCLEAR REGULATORY COMMISSION
ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
WASHINGTON, D.C. 20555-0001

November 8, 2000

The Honorable Richard A. Meserve
Chairman
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555-0001

Dear Chairman Meserve:

SUBJECT: DRAFT FINAL TECHNICAL STUDY OF SPENT FUEL POOL ACCIDENT RISK AT DECOMMISSIONING NUCLEAR POWER PLANTS

During the 477th meeting of the Advisory Committee on Reactor Safeguards, November 2-4, 2000, we discussed the draft final technical study of spent fuel pool accident risk at decommissioning nuclear power plants. The ACRS Reactor Fuels Subcommittee also met on October 18, 2000, to discuss this matter. During these meetings, we had the benefit of discussions with the staff and with representatives from NEI and the Institute for Resource and Security Studies. We also had the benefit of the documents referenced.

OBSERVATIONS AND RECOMMENDATIONS

1. This revised technical study provides an adequate basis for decisions on emergency preparedness (EP) requirements at decommissioning plants. The study has addressed concerns expressed in our report dated April 13, 2000.

2. Although the information is not needed for the decision on EP, the final report should include the calculated consequences for total deaths, injuries, and land contamination to provide input to decisions concerning insurance and safety at decommissioning plants.

3. The ACRS and stakeholders who appeared at our meetings on this subject agree that the staff needs to develop a better phenomenological understanding of the thermal hydraulics, chemical reactions, source terms, and physical phenomena associated with spent fuel pool fires as a basis for future risk-informed decisions concerning safety during plant decommissionings.

4. While not necessary for the intended use of this study, there is a need to reconcile the Lawrence Livermore National Laboratory (LLNL) seismic hazard curves and the Electric Power Research Institute (EPRI) seismic hazard curves so that the agency will have a consistent basis for its future seismic-related regulatory decisions.

DISCUSSION

In our April 13, 2000, report on an earlier draft of this study, we identified a number of concerns, including:

1. The inappropriate use of the Regulatory Guide (RG) 1.174 risk acceptance criterion for large early release frequency (LERF) in view of the expected differences in source terms for air-oxidation accidents and steam-oxidation accidents.

2. The use of an ignition temperature based on data from fresh (nonhydrided) cladding.

3. The lack of consideration of uncertainties in plume dispersion parameters.

4. The initial plume energy used in the atmospheric dispersion assessment could be substantially greater in a spent fuel fire than in an operating reactor accident.

5. The assessment of the seismic risk.

We are pleased that the revised technical study has addressed each of these concerns as follows.

LERF Acceptance Criterion and the Source Term

The revised technical study carried out the risk analysis for a representative site and included atmospheric transport and consequence determination. Instead of relying on a LERF surrogate, the results can be directly compared with the prompt and latent fatality Safety Goals. Based on sensitivity studies, the staff adopted a revised source term with a ruthenium release fraction of 0.75 and an actinide release fraction of 0.035. These values appear defensible for an air-oxidation source term based on the experimental data currently available.

Ignition Time for a Zirconium (Zr) Fire

Because of the lack of prototypic data and the large uncertainty associated with the requisite heat transfer analyses, particularly under obstructed air flow conditions, the study concluded that ignition of the Zr clad could not be precluded for any specific time period. We believe this to be an appropriately conservative approach until additional data and better analyses are available.

Atmospheric Dispersion Uncertainty

To estimate the uncertainty in the consequences as a result of atmospheric dispersion of released radioactivity, a total of 300 MELCOR Accident Consequence Code System (MACCS) calculations were performed using distributions of the dispersion parameters, σ_y and σ_z. This is a credible way to deal with these uncertainties and provides a better basis for decisionmaking.

Plume Energy

The staff performed a sensitivity calculation in which the plume energy dissipation rate for spent fuel pool fires was selected at three values covering the plausible range. This, too, provides technical input needed for the decisionmaking process.

Seismic Risk Assessment

The revised technical study used a less conservative (than the earlier draft study) but still simplified method that made use of a typical high confidence of low probability of failure (HCLPF) for a plant. The simplified method then combined the HCLPF with both the LLNL and the EPRI seismic hazard curves to estimate the seismic risk. We found this to be an acceptable approach for the purpose of comparing prompt fatalities and latent cancers with the Safety Goals.

For the cases using the more conservative LLNL seismic hazard curves, high ruthenium release, and significant actinide release, the overall individual risk of prompt fatalities is about a factor of 4 lower than the Safety Goal and the individual risk of latent cancer fatality is about an order of magnitude less than the Safety Goal. Emergency response measures had negligible effect on these results. Since severe seismic events dominated the risk of spent fuel fires at decommissioning plants, the staff argued that emergency response would be hindered by the collateral seismic damage to the transportation and communications infrastructure. Thus, the emergency response was equivalent to "late evacuation" (i.e., the population in the emergency response zone was modeled as being outdoors for the first 24 hours and then evacuated). Emergency response would be effective only for other-than-seismic accident sequences. Because these sequences develop slowly, there is time for effective *ad hoc* measures. These sequences, however, make a smaller contribution to risk, and therefore, emergency response planning is of marginal value. We agree with the staff's arguments and find that the technical study provides an adequate basis for decisions on potential relaxation of the EP requirements at decommissioning plants.

Regulatory decisions related to spent fuel pools should not be based solely on individual risk of prompt fatalities and the individual cancer risk. The large amounts of cesium and strontium, which have long half-lives, coupled with the higher plume energy and the larger values of the dispersion parameters recommended by experts, increase the relative importance of societal risk (total deaths), injuries, and land contamination. These may become more important consequences than individual prompt and latent fatalities if compared on the basis of equivalent cost or other appropriate metrics. For decisions concerning safety at decommissioning plants, all of the projected consequences calculated by MACCS should be included in the technical study report. These results, however, may underestimate these consequences because they are calculated only out to 100 miles. It should be possible to derive acceptance limits for these consequences from the Safety Goals based on concepts like equivalent cost.

The technical study used the LLNL and EPRI seismic hazard curves in assessing the seismic risks. The staff had no basis for excluding either of these sets of curves. Given this situation, future regulatory decisions involving seismic issues will have to be based on the most conservative of these whenever it matters to the decisionmaking process. This amounts to an inappropriate exclusion of the EPRI curves and a built-in bias toward use of the LLNL curves. The NRC cosponsored a study that proposes a process for the development of a single set of hazard curves (Reference 3). The agency should proceed with the development of such a combined set of curves for its future decision making needs.

Sincerely,

Dana A. Powers

Dana A. Powers
Chairman

References:

1. Letter dated October 12, 2000, from G. M. Holahan, NRR, to John T. Larkins, ACRS, Subject: Final Technical Study of Spent Fuel Pool Accident Risk at Decommissioning Nuclear Power Plants.
2. Report dated April 13, 2000, from Dana A. Powers, Chairman, ACRS, to Richard A. Meserve, Chairman, NRC, Subject: Draft Final Technical Study of Spent Fuel Pool Accident Risk at Decommissioning Nuclear Power Plants.
3. U.S. Nuclear Regulatory Commission, NUREG/CR-6372, "Recommendations for Probabilistic Seismic Hazard Analysis: Guidance on Uncertainty and Use of Experts," prepared by R.J. Budnitz, G. Apostolakis, D.M. Boore, L.S. Cluff, K.J. Coppersmith, C.A. Cornell, and P.A. Morris, April 1997.

UNITED STATES
NUCLEAR REGULATORY COMMISSION
ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
WASHINGTON, D.C. 20555-0001

November 13, 2000

MEMORANDUM TO: William D. Travers
Executive Director for Operations

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

J. E. Lyons

SUBJECT: DRAFT SAFETY EVALUATION REPORT RELATED TO "JUSTIFICATION FOR NOT INCLUDING POSTULATED BREAKS IN LARGE-BORE REACTOR COOLANT SYSTEM PIPING IN THE LICENSING BASIS FOR EXISTING AND REPLACEMENT ONCE-THROUGH STEAM GENERATORS" (BAW-2374, REV. 0, JULY 2000)

During the 477th meeting of the Advisory Committee on Reactor Safeguards, November 2-4, 2000, the Committee considered the draft safety evaluation report and decided not to review it.

References:
U.S. Nuclear Regulatory Commission Draft Safety Evaluation Report Related to "Justification for Not Including Postulated Breaks in Large-Bore Reactor Coolant System Piping in the Licensing Basis for Existing and Replacement Once-through Steam Generators" (BAW-2374, Rev. 0, July 2000), dated October 27, 2000.

cc: A. Vietti-Cook, SECY
J. Craig, OEDO
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S. Collins, NRR
W. Bateman, NRR
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S. Bailey, NRR

UNITED STATES
NUCLEAR REGULATORY COMMISSION
ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
WASHINGTON, D.C. 20555-0001

November 15, 2000

The Honorable Richard A. Meserve
Chairman
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555-0001

SUBJECT: LICENSE RENEWAL GUIDANCE DOCUMENTS

Dear Chairman Meserve:

During the 477th meeting of the Advisory Committee on Reactor Safeguards, November 2-4, 2000, we completed our review of the draft Standard Review Plan (SRP), Generic Aging Lessons Learned (GALL) report, Draft Regulatory Guide DG-1104, and NEI 95-10, Revision 2 that provide guidance for preparing and reviewing license renewal applications. Our Subcommittee on Plant License Renewal met on October 19-20, 2000, to review these documents. During our reviews, we had the benefit of discussions with representatives of the staff and the Nuclear Energy Institute (NEI). We also had the benefit of the documents referenced.

Conclusions and Recommendations

1. The draft guidance documents developed by the staff and the industry provide a consistent and understandable process to support the preparation and review of license renewal applications. The staff and the industry have made a commendable effort to effectively integrate these guidance documents.

2. The staff should update the GALL report as lessons are learned from reviewing future license renewal applications and as new editions of codes and standards are approved by the staff.

3. The staff should validate that the artificially aged cables used in the studies conducted to address GSI-168 issues are representative of 30-40 year old cables.

4. The staff and the industry should provide consistent guidance on the use of emergency operating procedures and severe accident management guidelines as possible information sources to verify that equipment important to safety has not been inadvertently left out by the license renewal rule scoping process.

Discussion

The staff and NEI have developed a set of draft guidance documents to establish consistency and stability in the license renewal application process. The Draft Regulatory Guide endorses NEI 95-10 that provides guidance on developing license renewal applications. The SRP provides guidance for reviewing the scoping and screening processes implemented by licensees to identify long-lived passive structures and components. The SRP also provides guidance on how to use the GALL report to identify applicable aging effects and acceptable aging management programs. The SRP and supporting documents have properly considered the issues and concerns raised by stakeholders, and have incorporated the resolutions of the generic license renewal issues. The staff and the industry have made a commendable effort to effectively integrate these guidance documents. This interaction of the staff with the industry has significantly improved the GALL report. The guidance documents provide a consistent and understandable process.

The GALL report is a remarkable compendium of current knowledge regarding aging effects and acceptable aging management programs. The report provides a technical basis to support license renewal decisions by describing where existing programs are sufficient and where "further evaluation" is required. Where further evaluation is required, the report generally explains what is expected. In many cases, the lessons learned from the review of the Oconee Nuclear Station and Calvert Cliffs Nuclear Power Plant license renewal applications have provided examples of the programmatic enhancements required. In some instances, the report identifies the need for further evaluation, but provides no guidance on what is expected or what criteria will be used to judge adequacy. The staff indicated that this lack of guidance is due to the limited experience with license renewal, and that programmatic enhancements developed for new license renewal applications will provide additional examples and alternate approaches. Since the preparation and review of future applications are likely to result in a significant number of new lessons learned, the staff should update the GALL report to incorporate the lessons learned.

The provisions of the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code have been codified in 10 CFR 50.55a. The staff has been amending 10 CFR 50.55a periodically to incorporate later editions of the ASME code. During periodic revision of 10 CFR 50.55a, the staff plans to evaluate the adequacy of these later editions for license renewal using the criteria described in the SRP. We believe this process is appropriate for the period of extended operation. The staff should update the GALL report to incorporate new editions of codes and standards for which a similar process does not exist.

Until GSI-168, which deals with environmental qualification of low-voltage instrumentation and control cables, is resolved, aging management of such cables will continue to be addressed through plant-specific programs. It does not appear that condition monitoring is a reliable predictor of future performance of cables under accident conditions. Testing of cables, which have undergone accelerated aging, identified severe degradation. The staff should validate that the artificially aged cables used in the accelerated aging studies conducted to address the issues of GSI-168 are representative of 30-40 year old cables. We plan to review this issue during our review of the proposed resolution of GSI-168.

The SRP provides guidance to review the adequacy of the scoping and screening processes used by the licensees to identify structures and components that are subject to an aging management review. As the first two applications demonstrated, the scoping process for older plants is a challenging task that does not lend itself to a standard procedure. Systems and components in scope are identified based on a review of accident analyses that are part of the current licensing basis (CLB) of the plant. The accident analyses, especially those of older plants, provide abbreviated descriptions of events and seldom identify all of the equipment required to achieve safe shutdown. More detailed information is contained in the emergency operating procedures (EOPs) that are referenced in the Final Safety Analysis Report and, thus, are part of the CLB of the plant. However, the scoping process defined by the license renewal rule does not explicitly include the EOPs as a source of information to identify equipment in scope. In contrast, the maintenance rule explicitly includes the EOPs as a source of information to identify equipment in scope. As a result, there may be equipment whose active components are within the scope of the maintenance rule but its passive long-lived components are not within the scope of the license renewal rule.

We recognize that most of the equipment used in the EOPs will be identified by the license renewal rule scoping process. The EOPs are already listed in Table 2.1-1 of the SRP as a possible information source. However, they are not listed as a possible information source in the corresponding Table 3.1-1 of NEI 95-10. We recognize that the EOPs are not within the scope of the license renewal rule. However, we believe that it would be prudent for the industry and the staff to include the EOPs in the guidance documents as a possible information source. This would confirm that equipment important to safety has not been omitted inadvertently in the scoping process, rather than leaving it to the individual reviewers to deal with this issue.

Severe Accident Management (SAM) guidelines are currently implemented at all plants, are part of the CLB, and are tied to the EOPs. Operators are routinely trained on their use. However, SAM guidelines were developed as a voluntary industry initiative. The equipment used to support these guidelines is not necessarily within the scope of the license renewal rule. The SAM guidelines should be identified as a potential source of information in Table 2.1-1 of the SRP and Table 3.1-1 of NEI 95-10 to confirm that equipment important to safety has not been omitted inadvertently in the scoping process.

Dr. William J. Shack did not participate in the Committee's deliberations regarding the GALL report.

Sincerely,

Dana A. Powers

Dana A. Powers
Chairman

References

1. U. S. Nuclear Regulatory Commission, Draft for Public Comment, "Standard Review Plan for the Review of License Renewal Applications for Nuclear Power Plants," August 2000.

2. U. S. Nuclear Regulatory Commission, NUREG-xxxx, Volume 1, Summary (Draft for Public Comment), "Generic Aging Lessons Learned (GALL) Report," August 2000.
3. U. S. Nuclear Regulatory Commission, NUREG-xxxx, Volume 2, Tabulation of Results (Draft for Public Comment), "Generic Aging Lessons Learned (GALL) Report," August 2000.
4. U. S. Nuclear Regulatory Commission, Draft Regulatory Guide DG-1104, "Standard Format and Content for Applications to Renew Nuclear Power Plant Operating Licenses," August 2000.
5. NEI 95-10, Revision 2, "Industry Guideline for Implementing the Requirements of 10 CFR 54 - the License Renewal Rule," August 2000.
6. S. P. Carfagno, ACRS Consultant, "Review of Adequacy of Staff Guidance for Reviewing License Renewal Applications," October 12, 2000.
7. C. Chen, Apollo Consulting, Inc., ACRS Consultant, "Report to USNRC ACRS on the Independent Review of SRP-LR and GALL Report for Containment Structures," October 8, 2000.

UNITED STATES
NUCLEAR REGULATORY COMMISSION
ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
WASHINGTON, D.C. 20555-0001

November 20, 2000

Dr. William D. Travers
Executive Director for Operations
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555-0001

Dear Dr. Travers:

SUBJECT: BWROG PROPOSAL TO USE SAFETY RELIEF VALVES AND LOW PRESSURE SYSTEMS AS A REDUNDANT SAFE SHUTDOWN PATH TO SATISFY THE REQUIREMENTS OF 10 CFR 50, APPENDIX R

During the 477th meeting of the Advisory Committee on Reactor Safeguards, November 2-4, 2000, we reviewed a proposal by the BWR Owners Group (BWROG) which seeks staff's endorsement of using safety relief valves (SRVs) and low pressure systems (LPS) as a redundant method to achieve safe shutdown as required by 10 CFR 50, Appendix R. During our review, we had the benefit of discussions with representatives of the BWROG and the staff. We also had the benefit of the documents referenced.

We believe that the analysis being used by the staff to review the BWROG proposal is appropriate. We would like, however, to review the staff's safety evaluation report when it becomes available.

Section III.G.1 of Appendix R requires that one train of systems necessary to achieve and maintain hot shutdown conditions be free of fire damage. The use of SRVs for reactor pressure control coupled with the use of a low pressure system (i.e., core spray or residual heat removal/low pressure coolant injection) for inventory makeup meets this requirement. This is because the affected unit will achieve hot shutdown upon initial blowdown using SRVs. Following this initial blowdown, the bulk reactor coolant temperature will be around 212°F. At this point, hot shutdown conditions could be maintained for an extended period by closing the SRVs and allowing the reactor to repressurize to a level below the shutoff pressure of the available low pressure system. In this mode of operation, injection flow would be throttled to maintain reactor water level within a specified, safe range and an SRV would be used to control pressure within a specified band. An operator might elect to maintain hot shutdown in this manner, if repairs were required to systems necessary to achieve and maintain cold shutdown. An operator, however, would not normally elect to do this if the ability to proceed directly to cold shutdown were available.

In using SRVs/LPS, the potential does exist for the collapsed reactor water level to drop below the top of the core depending on reactor level conditions and the time at which the reactor depressurization is initiated and the rate at which the depressurization is performed. This could be interpreted to imply that the proposal does not meet the requirements of Section III.L of Appendix R. Because the uncollapsed reactor water level is above the top of the active core, adequate core cooling is maintained which is consistent with the technical objective of Section III.L.

The BWROG proposal states that use of SRVs and LPS, in support of Appendix R safe shutdown requirement, is consistent with the original design basis for the BWRs. The proposal specifies a technically acceptable and safe means of achieving and maintaining either hot or cold shutdown. The same method is specified in the emergency operating procedures as a means to achieve cold shutdown upon the occurrence of small break loss of coolant accidents in BWRs. Its use to achieve safe shutdown for fire initiators is consistent with the plant design and normal operator training and expected response. Since fire-initiated sequences that require this shutdown method are rare, there will not be frequent undue stress on safety systems. Using the SRVs/LPS in the manner described in the BWROG report will not damage fuel cladding or rupture the reactor coolant system pressure boundary or the primary containment.

Sincerely,

Dana A. Powers

Dana A. Powers
Chairman

References:

1. Letter dated September 1, 1999, from W. Glenn Warren, BWR Owners' Group, to U.S. Nuclear Regulatory Commission, Document Control Desk, Subject: BWR Owners' Group Appendix R Fire Protection Committee Position on SRVs + Low Pressure Systems Used as "Redundant" Shutdown Systems Under Appendix R.
2. Letter dated July 20, 2000, to U.S. Nuclear Regulatory Commission, Document Control Desk, from James M. Kenny, BWR Owners' Group, Subject: BWR Owners' Group Appendix R Fire Protection Committee, Use of Safety Relief Valves and Low Pressure Systems as Redundant Safe Shutdown Paths.
3. Memorandum dated June 23, 1999, from Michael J. Davis, NRC, to Cynthia A. Carpenter, NCR, Subject: Summary of May 19, 1999, Meeting between the NRC staff and the Boiling Water Reactor Owners Group (BWROG) Appendix R Committee.

UNITED STATES
NUCLEAR REGULATORY COMMISSION
ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
WASHINGTON, D.C. 20555-0001

November 20, 2000

Dr. William D. Travers
Executive Director for Operations
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555-0001

Dear Dr. Travers:

SUBJECT: PROPOSED FRAMEWORK FOR RISK-INFORMED CHANGES TO THE TECHNICAL REQUIREMENTS OF 10 CFR PART 50

During the 477th meeting of the Advisory Committee on Reactor Safeguards, November 2-4, 2000, we met with representatives of the NRC staff to discuss Attachment 1 to SECY-00-0198 entitled, "Framework for Risk-Informed Changes to the Technical Requirements of 10 CFR 50." Our Subcommittee on Reliability and Probabilistic Risk Assessment met on July 11, 2000, to discuss an earlier version of the proposed framework. We also had the benefit of the documents referenced.

The purpose of the framework is to provide guidance to the staff for the identification and development of risk-informed changes to the technical requirements of 10 CFR Part 50 (Option 3). The proposed framework is a work in progress. The staff has identified the elements that are important to the prioritization of candidate regulations to be risk informed. We agree with the staff that improvements will be made to the framework as experience is gained from evaluating its application to risk-informing candidate regulations such as 10 CFR 50.44 related to combustible gas control systems and 10 CFR 50.46 concerning emergency core cooling systems. We offer the following comments for consideration as the work progresses.

The structuralist approach to defense in depth has been applied to the top tiers of the framework by adopting the cornerstones of the revised reactor oversight process, i.e., limiting the frequency of accident initiating events and the conditional core damage probability (CCDP) given an initiating event, and limiting radionuclide releases and public health effects given core damage. The "tactics" for achieving these goals include safety margins and redundancy, diversity, and independence. We recommend that the tactics for implementing defense in depth be clarified. Will defense in depth be applied at all levels of the framework? Will it be invoked at lower tiers when it has already been applied to the top tiers?

In our May 19, 1999 report and the associated attachment, we offered a "preliminary proposal" to apply the structuralist approach at lower tiers only when there are significant uncertainties that have not been included in the probabilistic risk assessment (PRA) and could reduce confidence that the higher-level goals are met. For uncertainties that are included in the PRA,

we recommended that the rationalist approach be followed, i.e., appropriate safety margins and redundancy, and diversity would be developed by quantitative analyses. Even though the framework is consistent with this approach, an expanded discussion of these issues would be beneficial.

We are pleased that the proposed framework recommends the quantification of safety margins in terms of probabilities. While present PRA methods can provide estimates of the contribution of multiple barriers (defense-in-depth measures) to the risk metrics, the contribution from safety margins is not normally quantified. We believe that the quantification of safety margins would be an important step toward the wider use of the rationalist approach. It would also make the integrated decision-making process of Regulatory Guide 1.174 easier to implement.

The framework proposes goals for the frequency of three groups of initiating events: anticipated, infrequent, and rare initiators. Even though this is reasonable for the standard initiating events for light water reactor PRAs, there is a potential pitfall. The concept of an initiating event is not defined rigorously. For an infrequent initiating event, the framework requires that the CCDP be less than or equal to 10^{-2} per reactor-year. One could envision partitioning this initiating event into a number of more specific initiating events, each with a frequency less than or equal to 10^{-5} per reactor-year. These new initiating events would then belong to the group of rare initiators, and there would be no constraints imposed on the CCDP. Thus, creative definitions of initiating events could be used to inappropriately relax the CCDP goal.

The external events in a PRA, such as earthquakes and fires, affect all of the cornerstones. The treatment of events that affect more than one cornerstone extensively should be discussed.

We look forward to reviewing additional refinements to the framework as progress is made in its application to developing risk-informed alternative regulations.

Sincerely,

Dana A. Powers

Dana A. Powers
Chairman

<u>References</u>:

1. Memorandum dated September 14, 2000, from William D. Travers, Executive Director for Operations, for the Commissioners, Subject: SECY-00-0198, Status Report on Study of Risk-Informed Changes to the Technical Requirements of 10 CFR Part 50 (Option 3) and Recommendations on Risk-Informed Changes to 10 CFR 50.44 (Combustible Gas Control).
2. Report dated September 13, 2000, from Dana A. Powers, Chairman, Advisory Committee on Reactor Safeguards, to Richard A. Meserve, Chairman, NRC, Subject: Proposed Risk-Informed Revisions to 10 CFR 50.44, "Standards for Combustible Gas Control System in Light-Water-Cooled Power Reactors."

3. Report dated May 19, 1999, from Dana A. Powers, Chairman, Advisory Committee on Reactor Safeguards, to Shirley Ann Jackson, Chairman, NRC, Subject: The Role of Defense in Depth in a Risk-Informed Regulatory System.
4. Paper by J. N. Sorensen, G. E. Apostolakis, T. S. Kress, D. A. Powers, "On the Role of Defense in Depth in Risk-Informed Regulation," presented at the American Nuclear Society, International Topical Meeting on Probabilistic Safety Assessment, PSA '99, Washington, DC, August 22-26, 1999.
5. Memorandum dated February 3, 2000, from Annette Vietti-Cook, Secretary, NRC, to William D. Travers, Executive Director for Operations, NRC, Subject: Staff Requirements - SECY-99-264 - Proposed Staff Plan for Risk-Informing Technical Requirements in 10 CFR Part 50.
6. U. S. Nuclear Regulatory Commission, Regulatory Guide 1.174, "An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis," July 1998.

UNITED STATES
NUCLEAR REGULATORY COMMISSION
ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
WASHINGTON, D.C. 20555-0001

December 13, 2000

MEMORANDUM TO: William D. Travers
Executive Director for Operations

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

SUBJECT: SOUTH TEXAS PROJECT, UNITS 1 AND 2 - DRAFT SAFETY EVALUATION ON EXEMPTION REQUESTS FROM SPECIAL TREATMENT REQUIREMENTS OF 10 CFR PARTS 21, 50, AND 100

During the 478th meeting of the Advisory Committee on Reactor Safeguards, December 6-9, 2000, the ACRS discussed the staff's draft safety evaluation on the Exemption Requests from Special Treatment Requirements for the South Texas Project, Units 1 and 2. The draft safety evaluation is not complete. Because of the significance of the open items, an ACRS Subcommittee plans to discuss this issue during a meeting prior to completion of the staff's draft safety evaluation.

<u>References</u>:

1. Letter dated November 15, 2000, from J. Zwolinski, NRR, to W. Cottle, STP Nuclear Operating Company, Subject: South Texas Project, Units 1 and 2 - Draft Safety Evaluation on Exemption Requests From Special Treatment Requirements of 10 CFR Parts 21, 50, and 100.
2. Letter dated August 31, 2000, from J. J. Sheppard, STP Nuclear Operating Company, to NRC Document Control Desk, Subject: Revised Request for Exemption to Exclude Certain Components from the Scope of Special Treatment Requirements Required by Regulations.
3. Letter dated July 13, 1999, from J. J. Sheppard, STP Nuclear Operating Company, to NRC Document Control Desk, Subject: Request for Exemption to Exclude Certain Components from The Scope of Special Treatment Requirements Required by Regulations.

cc: A. Vietti-Cook, SECY
J. Craig, OEDO
I. Schoenfeld, OEDO

S. Collins, NRR
J. Zwolinski, NRR
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G. Holahan, NRR
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UNITED STATES
NUCLEAR REGULATORY COMMISSION
ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
WASHINGTON, D.C. 20555-0001

December 14, 2000

Dr. William D. Travers
Executive Director for Operations
U.S. Nuclear Regulatory Commission
Washington, D. C. 20555-0001

Dear Dr. Travers:

SUBJECT: NUCLEAR ENERGY INSTITUTE DRAFT REPORT, NEI 99-03, "CONTROL ROOM HABITABILITY ASSESSMENT GUIDANCE"

During the 478th meeting of the Advisory Committee on Reactor Safeguards, December 6-9, 2000, we reviewed a draft of the Nuclear Energy Institute (NEI) Report, NEI 99-03, "Control Room Habitability Assessment Guidance." Our Subcommittee on Severe Accident Management also reviewed this matter during its meeting of November 15, 2000. During our reviews, we had the benefit of discussions with representatives of NEI and the NRC staff. We also had the benefit of a presentation from Peter Lagus of Lagus Technology, Inc., and of the documents referenced.

BACKGROUND

Concern for the integrity of the control room envelope has been an issue for many years. The ACRS identified issues associated with this matter in the 1982-83 timeframe. In 1998, representatives of the NRC staff identified long-standing concerns associated with this issue during a public workshop. In response, NEI formed a Control Room Habitability Task Force and issued an initial draft of NEI 99-03. Subsequent work by a team of NEI and NRC staff representatives culminated in the development of the current draft version under review.

RECOMMENDATIONS AND OBSERVATIONS

1. The staff should continue with its development of a regulatory guide on control room habitability, making liberal and extensive use of NEI 99-03.

2. The staff should require that the results of component testing be validated by comparisons with those of tracer gas testing in several control room configurations prior to the staff agreeing to the exclusive use of component testing for pressurized control rooms.

3. If component testing is shown to correlate adequately with the tracer-gas testing, then component testing should be acceptable for the baseline testing for pressurized control rooms and for subsequent periodic testing.

4. The frequency for periodic assessment and testing should be placed on a performance basis much like the requirements of Appendix J to 10 CFR Part 50 for containment leakage testing.

5. A specific limit for allowed in-leakage should be made a part of a plant's licensing basis. This does not necessarily mean it has to be specified in the technical specifications.

6. The potential radiation doses from design basis accidents at adjacent or nearby nuclear power plants should be included in the control room habitability assessment whether or not this is part of the current licensing basis.

7. The approach in NEI 99-03 provides a sound basis for maintaining safe shutdown capability given a challenge from smoke generated external to the control room. The approach could be endorsed by the regulatory guide.

DISCUSSION

We believe that the current draft version of NEI 99-03 can provide excellent guidance to industry to deal with control room habitability issues and to ensure compliance with the applicable regulations. It is, however, a technical document that includes more detail than needed for a regulatory guide. In addition, there is a very significant open issue (i.e., the proposed option of using component testing for the initial baseline in-leakage assessment for pressurized control rooms). The staff will need to confirm that component testing can reliably establish the total unfiltered in-leakage. It is our view, therefore, that for component testing to be acceptable as the initial baseline determination of the in-leakage, it must be validated by use of tracer-gas testing for several control room configurations and types. By a limited number of comparisons with tracer gas testing, assurance can be provided that all of the vulnerable component leakage paths can be reliably identified. If the results from the two test methods are shown to correlate well, then component testing should be acceptable for subsequent baseline initial testing and for the subsequent required periodic testing to confirm that in-leakage control is being adequately maintained.

The frequency of required periodic assessments and testing needs to be put on a performance basis much like the requirements of Appendix J for containment leakage testing. The frequency is likely to be plant specific and will depend on the factors addressed in the Control Room Habitability Program described in NEI 99-03.

Even with the above approaches, maintaining an acceptable in-leakage rate at all times is not guaranteed. This appears to be an area that could use innovative research aimed at developing a continuous indication of the in-leakage potential.

It is important that the specific limit for in-leakage be made a part of the licensing basis. Rather than specifying the allowed in-leakage as a technical specification, however, NEI 99-03 proposes committing to a Control Room Habitability Program based on inspection, sealing, and

maintenance with periodic component testing. As long as this commitment provides appropriate regulatory control, we do not believe a technical specification commitment is necessary.

Control room habitability regulations are intended to ensure that the operators can occupy the control areas to mitigate any ongoing accident or to safely shut down the plant in the event of external threats. Thus, control room habitability assessment must include any potential radiological source from adjacent or nearby nuclear plants just as is done for toxic sources.

Although no current NRC regulations exist to establish smoke concentration limits or to define a design basis fire, NEI 99-03 contains a recommendation that licensees perform a qualitative evaluation of their ability to manage smoke infiltration into the control room. The industry is to be commended for taking this proactive approach and the staff should consider endorsing the proposed approach in its regulatory guide.

Sincerely,

Dana A. Powers

Dana A. Powers
Chairman

References:
1. Letter dated October 13, 2000, from D. Modeen, NEI, to R. Barrett, NRC, transmitting Draft NEI 99-03, Control Room Habitability Assessment Guidance.
2. Letter dated August 18, 1982, from P. Shewmon, ACRS, to Hon. Nunzio J. Palladino, ACRS, Subject: Control Room Habitability
3. Letter dated May 17, 1983, from J. C. Ebersole, ACRS, to W. J. Dircks, Executive Director for Operations, NRC, Subject: ACRS Subcommittee Report on Control Room Habitability

UNITED STATES
NUCLEAR REGULATORY COMMISSION
ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
WASHINGTON, D.C. 20555-0001

December 14, 2000

MEMORANDUM TO: William D. Travers
Executive Director for Operations

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

SUBJECT: PROPOSED REVISION TO THE COMMISSION'S SAFETY GOAL POLICY STATEMENT FOR REACTORS

During the 478th meeting of the Advisory Committee on Reactor Safeguards, December 6-9, 2000, the Committee considered the proposed revision to the Commission's Safety Goal Policy Statement for Reactors. Individual members suggested a number of changes that the staff agreed to consider. The Committee has no objection to the proposed revision to the Safety Goal Policy Statement.

Reference:
Memorandum dated November 14, 2000, from Ashok C. Thadani, Director, Office of Nuclear Regulatory Research, to John T. Larkins, Executive Director, Advisory Committee on Reactor Safeguards, Subject: Safety Goal Policy Statement.

cc: A. Vietti-Cook, SECY
J. Craig, OEDO
I. Schoenfeld, OEDO
A. Thadani, RES
J. Murphy, RES
S. Collins, NRR
W. Kane, NMSS

UNITED STATES
NUCLEAR REGULATORY COMMISSION
ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
WASHINGTON, D.C. 20555-0001

December 15, 2000

Dr. William D. Travers
Executive Director
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555-0001

SUBJECT: PROPOSED FINAL REGULATORY GUIDE DG-1053, "CALCULATIONAL AND DOSIMETRY METHODS FOR DETERMINING PRESSURE VESSEL NEUTRON FLUENCE"

Dear Dr. Travers:

During the 478th meeting of the Advisory Committee on Reactor Safeguards, December 6-9, 2000, we met with representatives of the NRC staff to discuss the proposed final Regulatory Guide DG-1053, "Calculational and Dosimetry Methods for Determining Pressure Vessel Neutron Fluence." Our Subcommittee on Materials and Metallurgy met on November 16, 2000, to discuss this matter. We had the benefit of the documents referenced.

Recommendation and Conclusion

- We recommend that the regulatory guide be issued for use by the industry.

- The NRC staff should be commended for the development of an excellent regulatory guide. The new guidance will result in more accurate calculations of the fluence and expedite review of licensee submittals.

Discussion

To ensure the integrity of reactor pressure vessels, the NRC has established fracture toughness requirements for normal operation and anticipated operational occurrences in 10 CFR Part 50, Appendix G, "Fracture Toughness Requirements," and for potential pressurized thermal shock events in pressurized water reactors in 10 CFR 50.61, "Fracture Toughness Requirements for Protection Against Pressurized Thermal Shock Events." Methods for the determination of the fast neutron fluence are needed to estimate the fracture toughness of reactor pressure vessel materials. Various methods implemented in numerous ways of varying reliability and accuracy have been used to determine the fluence. This wide variation in calculation methods has resulted in lengthy plant-specific reviews. To remedy this situation, the staff has developed a regulatory guide to provide standardized methods and procedures to simplify and expedite these reviews.

We reviewed draft Regulatory Guide DG-1025 on this subject in July 1993 and issued a letter dated July 15, 1993. Since then the staff has revised DG-1025 several times in response to

stakeholders comments and designated it as DG-1053. The latest revision includes guidance for using Monte Carlo methods as well as the more conventional discrete ordinates calculation methodology. In addition, the staff has issued NUREG/CR-6115, which provides PWR and BWR pressure vessel fluence calculation benchmark problems and solutions that can be used to validate licensee calculational methods. The revised regulatory guide and the associated NUREG/CR-6115 provide guidance on the selection of appropriate geometrical and material input data, fluence calculation methods, the qualification of the methodology and modeling assumptions, and the determination of the uncertainties associated with the analysis. The methodology provides a best estimate rather than a bounding or conservative estimate of the fluence.

The draft regulatory guide has received substantial input and comment from the industry and concerned citizens. The staff and its contractor have considered the comments and incorporated many of them to improve the clarity of the guidance. There appears to be a consensus between the staff and the industry that the new guidance will result in more accurate calculations of the fluence and more efficient review of licensee submittals. Indeed, the reduction in the uncertainties in the fluence calculation could lead eventually to reductions in dosimetry and the margin terms required in estimates of the fracture toughness.

We commend the staff for its efforts to develop improved guidance for the calculation of vessel neutron fluence. The regulatory guide should be issued for industry use.

Sincerely,

Dana A. Powers

Dana A. Powers
Chairman

References:

1. Memorandum dated November 29, 2000, from Michael E. Mayfield, Office of Nuclear Regulatory Research, to John T. Larkins, Advisory Committee on Reactor Safeguards, Subject: Request for Review and Concurrence to Issue Proposed Regulatory Guide, "Calculational and Domisetry Methods for Determining Pressure Vessel Neutron Fluence," including regulatory and backfit analysis.
2. Letter dated July 15, 1993, from J. Ernest Wilkins, Jr., Chairman, ACRS, to James M. Taylor, Executive Director for Operations, NRC, Subject: Proposed Draft Regulatory Guides DG-1023, "Evaluation of Reactor Pressure Vessels With Charpy Upper-Shelf Energy Less Than 50 FT-LB," and DG-1025, "Calculational and Dosimetry Methods for Determining Pressure Vessel Neutron Fluence."
3. U.S. Nuclear Regulatory Commission, NUREG/CR-6115, BNL-NUREG-52395, "PWR and BWR Pressure Vessel Fluence Calculation Benchmark Problems and Solutions," completed May 20, 1997.
4. Draft Summaries of Comments Received From Nuclear Energy Institute, Professor Alireza Haghighat, and Don't Waste Michigan concerning DG-1053, "Calculational and Dosimetry Methods for Determining Pressure Vessel Neutron Fluence," received October 23, 2000. [Predecisional].

5. Comments Received From Nuclear Energy Institute, Professor Alireza Haghighat, and Don't Waste Michigan concerning DG-1053, "Calculational and Dosimetry Methods for Determining Pressure Vessel Neutron Fluence," received October 23, 2000. [Predecisional].

NRC FORM 335
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U.S. NUCLEAR REGULATORY COMMISSION

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Same as above

10. SUPPLEMENTARY NOTES

11. ABSTRACT *(200 words or less)*

This compilation contains 56 ACRS reports submitted to the U. S. Nuclear Regulatory Commission (NRC), or to the NRC Executive Director for Operations, during calendar year 2000. In addition, a report to the Commission on the NRC Safety Research Program, NUREG-1635, Volume 3, is included by reference only. 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 organized in chronological order.

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

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