NUREG–1125
Volume 21

A Compilation of Reports of The Advisory Committee on Reactor Safeguards

1999 Annual

U. S. Nuclear Regulatory Commission

April 2000

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Volume 21

A Compilation of Reports of The Advisory Committee on Reactor Safeguards

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ABSTRACT

This compilation contains 74 ACRS reports submitted to the U. S. Nuclear Regulatory Commission (NRC), or to the NRC Executive Director for Operations, during calendar year 1999. In addition, a report to the Commission on the NRC Safety Research Program, NUREG-1625, Volume 2, 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 represent the recommendations and comments of the U. S. NRC's Advisory Committee on Reactor Safeguards during calendar year 1999. 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
13	Calendar Year 1991
14	Calendar Year 1992
15	Calendar Year 1993
16	Calendar Year 1994
17	Calendar Year 1995
18	Calendar Year 1996
19	Calendar Year 1997
20	Calendar Year 1998

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TABLE OF CONTENTS

TABLE OF CONTENTS

TABLE OF CONTENTS

TABLE OF CONTENTS

TABLE OF CONTENTS

TABLE OF CONTENTS

TABLE OF CONTENTS

Page

TABLE OF CONTENTS

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

February 11, 1999

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-1076, "SERVICE LEVEL I, II, AND III PROTECTIVE COATINGS APPLIED TO NUCLEAR POWER PLANTS"

During the 459th meeting of the Advisory Committee on Reactor Safeguards, February 3-6, 1999, the Committee considered the subject draft regulatory guide and decided not to review it. The Committee has no objection to the staff's proposal to issue this guide for public comment. The Committee would like the opportunity to review the proposed final version of DG-1076 after the staff has reconciled public comments.

Reference:
Memorandum dated January 22, 1999, from John W. Craig, Office of Nuclear Regulatory Research, to John T. Larkins, ACRS, Subject: DG-1076, "Service Level I, II, and III Protective Coatings Applied to Nuclear Power Plants," January 20, 1999.

cc: A. Vietta-Cook, SECY
J. Blaha, OEDO
J. Mitchell, OEDO
A. Thadani, RES
J. Craig, RES
A. Serkiz, RES

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

February 16, 1999

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-1080, "NUCLEAR POWER PLANT SIMULATION FACILITIES FOR USE IN OPERATOR TRAINING AND LICENSE EXAMINATIONS"

During the 459th meeting of the Advisory Committee on Reactor Safeguards, February 3-6, 1999, the Committee considered the subject draft regulatory guide and decided not to review it. The Committee has no objection to the staff's proposal to issue this guide for public comment.

Reference:
Memorandum dated December 11, 1998, from R. Lee Spessard, NRR, to Joseph R. Gray, OGC, James Lieberman, OE, Jesse L. Funches, CFO, and David L. Meyer, ADM, Subject: Draft Regulatory Guide DG-1080 - Nuclear Power Plant Simulation Facilities for Use in Operator Training and License Examinations.

cc: A. Vietti-Cook, SECY
J. Blaha, OEDO
J. Mitchell, OEDO
S. Collins, NRR
R. Gallo, NRR
F. Collins, NRR

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

February 18, 1999

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

Dear Chairman Jackson:

SUBJECT: LIST OF QUESTIONS TO BE ADDRESSED FOR POSSIBLE RESOLUTION OF KEY ISSUES ASSOCIATED WITH THE PROPOSED REVISION TO 10 CFR 50.59 (CHANGES, TESTS AND EXPERIMENTS)

During the February 3, 1999 meeting between the Commission and the Advisory Committee on Reactor Safeguards, the Commission requested that the ACRS provide a list of questions which, if answered, would aid in the resolution of key issues associated with the proposed near-term revision to 10 CFR 50.59. In our discussion of this request during our 459th meeting on February 3-6, 1999, we considered two approaches to the resolution of the issues associated with 10 CFR 50.59 and developed questions for each of these approaches.

In Approach 1, we propose a minimal set of questions that, if addressed, would preserve the desirable attributes of the 10 CFR 50.59 process that has been in place for over 30 years. In Approach 2, we propose another set of questions that, if addressed, would result in more profound changes to the 10 CFR 50.59 process. Both of these approaches are intended to address the proposed near-term revision to provide clarity and flexibility in the existing requirements, and not the long-term risk-informed revision to 10 CFR 50.59.

APPROACH 1: Reconciliation of the Differences Between 10 CFR 50.59 and NEI 96-07

There is general agreement that the 10 CFR 50.59 process has worked well for over 30 years. Licensee implementation of the current process has been based on the guidance provided by NSAC-125, which the industry has attempted to improve through the development of NEI 96-07. The NRC staff has never formally endorsed the guidance included in these documents, but the staff has acknowledged that the overwhelming majority of the safety evaluations performed by licensees using this guidance have been acceptable. We believe that answering the following questions would provide a near-term revision to 10 CFR 50.59 that could optimize the benefits of past practice and provide regulatory stability.

1. What are the specific elements of the guidance in NEI 96-07 that the staff finds unacceptable?

2. Are these elements unacceptable because the staff believes they contradict the legal requirements of the current 10 CFR 50.59, or because they are technically inadequate?

3. What are the minimum changes that must be made to 10 CFR 50.59 and NEI 96-07 so that the proposed rule and the guidance are consistent?

Observation on Approach 1

Answering the above questions could provide a near-term solution for 10 CFR 50.59 that would maintain a process that has worked successfully and provide regulatory stability by requiring only limited changes to the process currently implemented by licensees and the staff. Such a process would, however, still require safety evaluations for many changes of little or no risk significance.

<u>APPROACH 2</u>: Consideration of Margin of Safety and Definition of Change Associated with the Proposed Revision to 10 CFR 50.59

It is possible that, even in the short term, more profound changes to the 10 CFR 50.59 process can be developed by considering the fundamental goal and intent of the 10 CFR 50.59 process. To do this would require resolution of the following questions:

Margin of Safety

1. Do the current Technical Specification acceptance limits provide sufficient assurance of safety? If not, to what extent should the current Technical Specifications be modified to achieve the needed margin of safety?

2. Should the guiding principle be that cumulative changes do not result in exceeding the limits or is there a need for margin between a "best estimate" calculated value and the limits to provide confidence that the limits have not been exceeded? Should licensees be allowed to incrementally approach the limits?

3. Can the NRC accept a calculated value from a licensee based on the licensee's NRC approved methodology without prior NRC review? If not, what is needed to provide assurance that the Technical Specification limit has not been exceeded as a result of cumulative changes?

4. Can operational experience be used to quantify the "conservatism" in the licensee's methodology? If not, is the only alternative to perform an uncertainty analysis on the licensee's methodology?

5. If it is established that the licensee's methodology is conservative, is that sufficient to ensure that the cumulative effects (even when these are calculated not to exceed the

acceptance limits) still provide acceptable confidence that the limits have not been exceeded?

Definition of Change

The definition of "change" is central to the screening step that is implicit in the 10 CFR 50.59 process. The staff needs to define important structures, systems, and components (SSCs) as they relate to the facility, procedures, tests and experiments, malfunctions and accidents. In addressing the definition of change, we have developed the following questions:

1. Does the updated Final Safety Analysis Report (FSAR) constitute an adequate and complete description of the facility for the purpose of ensuring adequate protection of the health and safety of the public?

2. Does any change to the facility or procedures described in the updated FSAR, irrespective of its safety significance, require a safety evaluation?

3. Do proposed changes to SSCs not referenced in the updated FSAR, but affecting the safe performance of SSCs described in the updated FSAR, require safety evaluations?

4. What consequences, other than those having an effect on safety system performance, should be considered in a safety evaluation?

5. Can references to "probability" be deleted from the definitions of minimal changes?

Observation on Approach 2

It appears to us that many of the options for changes in the definition of "margin of safety" currently being considered greatly increase the importance of tracking the cumulative effect of such changes. Although the vast majority of changes introduced under the 10 CFR 50.59 process would still involve negligible changes in risk, the new definitions certainly could result in changes that, while acceptable, would not be negligible. This might require more frequent updating of the FSAR and a far more rigorous tracking of the changes. It is not clear to us that this might not result in more regulatory burden than a 10 CFR 50.59 process that is more restrictive on changes.

We plan to continue our review of the proposed revision to 10 CFR 50.59 during future meetings.

Sincerely,

Dana A. Powers

Dana A. Powers
Chairman

References:

1. Proposed rule dated October 14, 1998, from John C. Hoyle, Secretary, NRC, to the Federal Register, Subject: 10 CFR Parts 50, 52, and 72, RIN 3150-AF94, Changes, Tests and Experiments.
2. Electric Power Research Institute, Nuclear Safety Analysis Center, NSAC-125, "Guidelines for 10 CFR 50.59 Safety Evaluations," June 1989.
3. Nuclear Energy Institute, NEI 96-07, Revision 0, "Guidelines for 10 CFR 50.59 Safety Evaluations," September 1997.

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

February 18, 1999

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

Dear Chairman Jackson:

SUBJECT: NFPA 805, "PERFORMANCE-BASED STANDARD FOR FIRE PROTECTION FOR LIGHT WATER REACTOR ELECTRIC GENERATING PLANTS"

During the 459th meeting of the Advisory Committee on Reactor Safeguards, February 3-6, 1999, we reviewed a draft NFPA 805 Standard on fire protection developed by the National Fire Protection Association (NFPA). During our review, we had the benefit of discussions with representatives of the NRC staff, NFPA, and the Nuclear Energy Institute (NEI). We also had the benefit of the documents referenced.

CONCLUSIONS AND RECOMMENDATIONS

1. The draft version of the NFPA 805 Standard is intended to be an alternate method to meet the intent of existing fire protection requirements in 10 CFR 50.48, Appendix R, and General Design Criterion (GDC) 3. The draft Standard is not, however, a distinct, risk-informed, performance-based alternative to these existing fire protection requirements.

2. It may now be time for the NRC staff to revisit the strategy described in SECY-98-058 (Development of a Risk-Informed, Performance-Based Regulation for Fire Protection at Nuclear Power Plants) and initiate work on an alternate rule that makes good use of risk information and is distinctly performance based.

DISCUSSION

Existing fire protection requirements for nuclear power plants are quite prescriptive. Over the last several years, there has been interest on the parts of both the nuclear industry and the NRC in finding an alternative to these prescriptive requirements that would be more performance based. That is, outcomes, rather than methods and processes, would be specified in the performance-based requirements. More recently there also has been interest in using risk information to determine the performance standards for fire protection programs at nuclear power plants.

The NFPA has volunteered to help develop a performance-based fire protection standard for nuclear power plants. The draft version of the standard developed under the auspices of the NFPA was issued for public comment on November 25, 1998. Elements of the Standard are:

- basic, deterministic requirements imposed on all fire protection programs,
- additional requirement that the developers of fire protection programs must choose is either deterministic or performance-based, and
- a site-wide risk assessment to ascertain if more stringent or additional requirements are needed.

The risk assessment can be used to set performance criteria that are not yet defined. Risk assessment is not allowed by the standard to alter the basic fire protection requirements. Indeed, it is not readily apparent that risk analysis is considered as a means for justifying reductions in the additional performance-based fire protection requirements.

The objectives of the NFPA 805 Standard are to address nuclear safety, radiological release, life safety and property damage. The nuclear safety objectives include reactivity control and fuel cooling. These objectives are not the same as those set by the NRC, although the NRC is the only "agency having jurisdiction" over nuclear safety. Proliferation of goals and objectives does not contribute to the coherence and comprehension of safety regulation. Why should not fire safety objectives in the Standard be derived from NRC's safety objectives or from the cornerstones of safety adopted in revising the NRC inspection and assessment programs? The Standard could be a systematic, top-down derivation of fire protection objectives. This top-down process could be used to identify and even define performance criteria for individual pieces of equipment and elements of the fire protection program. Properly done, this process would make unnecessary a site-wide risk assessment to provide "additional assurance" of adequate fire protection. It would make possible the quantitative assessment of acceptable fire risk and acceptable levels of fire safety now called for in the Standard.

As formulated currently, it is difficult to distinguish some deterministic requirements and their performance counterparts. Consider, for example, the Standard's performance-based requirement: "Each fire pump and its driver and control shall be located in a room separated from the remaining fire pumps and from the remainder of the plant by barriers with fire resistance ratings as required by the Fire Hazards Analysis (FHA)."

This appears to be quite a deterministic requirement. We note that a performance requirement could be deterministic. We suspect that the concept of "performance" is interpreted differently by the NFPA and the nuclear safety community. The term as used by NFPA in the Standard should be clarified.

The NFPA 805 standard amounts to a rederivation of the fire protection requirements in Appendix R with minimal steps in the direction of using performance criteria and risk information. One could envision the NFPA 805 Standard, once completed, being endorsed in part in a regulatory guide as an acceptable, alternate way to meet the intent of existing fire protection requirements. The development of strategies for the NRC to inspect fire protection programs based on this new standard and to enforce requirements would require substantial effort.

It is clear that the draft NFPA 805 Standard is not a bold step in the direction of risk-informed, performance-based fire protection. It appears possible to make a far bolder step. There is an alignment of defense in depth for fire protection and risk analysis. Defense in depth for fire protection consists of steps to prevent fires from occurring, to detect and suppress fires, and to protect safety-related equipment from the effects of fires. Fire risk analyses attempt to quantify the effectiveness of these defense-in-depth steps. One can well imagine a rule calling for performance criteria based, perhaps, on risk analyses, for prevention of fires, detection and suppression of fires, and protection of equipment from the effects of fires. Performance indicators could be defined for each of these performance criteria.

Development of a risk-informed, performance-based alternative to existing fire protection rules should be done within the context of other ongoing activities within the NRC. The objectives and performance should be defined in a top-down fashion to yield results consistent with those of other elements of nuclear power plant safety strategies. There should be a systematic, transparent process that defines the pathway from the topmost objectives to individual performance criteria and performance indicators used for monitoring a plant fire protection program. Processes used to develop the NRC's improved plant inspection and assessment programs might well serve as a guide to develop a new fire protection rule.

We plan to follow the progress in the development of the NFPA 805 Fire Protection Standard, which is scheduled for completion in May 2000.

Sincerely,

Dana A. Powers

Dana A. Powers
Chairman

References:
1. NFPA 805, Draft 6.3, "Performance-Based Standard for Fire Protection for Light Water Reactor Electric Generating Plants," dated November 25, 1998.
2. Memorandum dated March 26, 1998, from L. Joseph Callan, Executive Director for Operations, NRC, for The Commissioners, SECY-98-058, Subject: Development of a Risk-Informed, Performance-Based Regulation for Fire Protection at Nuclear Power Plants.
3. Letter dated November 7, 1997, from George D. Miller, National Fire Protection Association, to Shirley Ann Jackson, Chairman, NRC, regarding NFPA development of a standard covering fire protection.
4. Memorandum dated September 11, 1997, from John C. Hoyle, Secretary of NRC, to L. Joseph Callan, Executive Director for Operations, NRC, Subject: Staff Requirements - SECY-97-127, Development of a Risk-Informed, Performance-Based Regulation for Fire Protection at Nuclear Power Plants.

5. Memorandum dated October 2, 1998, from L. Joseph Callan, Executive Director for Operations, NRC, for The Commissioners, SECY-98-230, Subject: Insights from NRC Research on Fire Protection and Related Issues.
6. Memorandum dated October 27, 1998, from William D. Travers, Executive Director for Operations, NRC, for The Commissioners, SECY-98-247, Subject: Risk-Informed, Performance-Based Fire Protection at Nuclear Power Plants.

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

February 19, 1999

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

Dear Dr. Travers:

SUBJECT: RESOLUTION OF GENERIC SAFETY ISSUE B-61, "ALLOWABLE ECCS EQUIPMENT OUTAGE PERIODS"

During the 459th meeting of the Advisory Committee on Reactor Safeguards, February 3-6, 1999, we reviewed the proposed resolution of Generic Safety Issue (GSI) B-61, "Allowable ECCS Equipment Outage Periods." During our review, we had the benefit of discussions with representatives of the NRC staff and the documents referenced.

CONCLUSION

The issues identified under GSI B-61 will be addressed through the implementation of the Maintenance Rule. Therefore, consideration of these issues under the aegis of GSI B-61 is not required, and GSI B-61 should be considered resolved.

DISCUSSION

GSI B-61, identified in June 1978 and prioritized in November 1983, was described in NUREG-0471, "Generic Task Problem Descriptions." It addresses the risk impact of surveillance test intervals and allowable equipment outage periods. These allowable outage periods, which are largely based on engineering judgment, are defined in Technical Specifications for safety-related systems. The allowable outages represent 20 to 80 percent of the total unavailability of emergency core cooling systems (ECCS).

The staff considered the need to implement a limit on cumulative outage time (COT) and conducted a limited regulatory analysis of the issues in GSI B-61 to evaluate the impacts of COT on systems during unscheduled or corrective maintenance. Results of this analysis revealed that implementation of COT did not meet the substantial added protection criterion specified in the regulatory analysis guidelines. The staff's analysis was limited to consideration of four representative plants. The staff did not compare the results of the analysis with those included in the Individual Plant Examination Insights report. Also, the staff did not perform an evaluation of uncertainties associated with its analysis. Although this analysis was inadequate for the resolution of GSI B-61, this issue should be considered resolved because the concerns identified under this GSI will be addressed through implementation of the Maintenance Rule.

Sincerely,

Dana A. Powers
Chairman

References

1. Memorandum dated January 12, 1999, from Thomas L. King, Division of Systems Technology, RES, to John T. Larkins, ACRS, Subject: Resolution of B-61, "Analytically Derived Allowable Equipment Outage Periods."
2. U. S. Nuclear Regulatory Commission, NUREG-0471, "Generic Task Problem Descriptions," June 1978.
3. U. S. Nuclear Regulatory Commission, NUREG-0933, "A Prioritization of Generic Safety Issues," Item B-61: Allowable ECCS Equipment Outage Periods, November 1983.
4. Memorandum dated June 17, 1983, from F. H. Rowsome, Division of Safety Technology, NRR, to G. C. Lainas, Division of Licensing, NRR, Subject: Safety Evaluation of the Licensees' Response to TMI Action Item II.K.3.17.
5. U. S. Nuclear Regulatory Commission, NUREG/BR-0058, Revision 2, "Regulatory Analysis Guidelines of the U.S. Nuclear Regulatory Commission," Final Report, November 1995.

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

February 19, 1999

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

Dear Dr. Travers:

SUBJECT: SECY-98-244, "NRC HUMAN PERFORMANCE PLAN"

During the 459th meeting of the Advisory Committee on Reactor Safeguards, February 3-6, 1999, we reviewed the current version of the NRC Human Performance Plan (HPP) contained in SECY-98-244 and the staff's strategy for completing the development of the HPP. Since February 1996, we have held several meetings with the staff to discuss various versions of the HPP and have issued three reports. During our most recent review, we had the benefit of discussions with representatives of the staff and of the documents referenced.

Observations and Recommendations

- We continue to believe that human performance is a major factor in the safe operation of nuclear power plants.

- We reiterate our previous recommendation that a well-planned research effort in human performance is needed to support both the present regulation of plant operations and the transition to risk-informed, performance-based regulation.

- The staff described a disciplined strategy for future development of a technically justified HPP. We believe that the following two elements of this strategy are valuable:

  - review of the Accident Sequence Precursor (ASP) data to identify the contribution of human performance to significant events, and

  - interaction with other organizations, such as the Institute of Nuclear Power Operations (INPO), that have a strong focus on human performance.

- Additional steps are needed to complete the development of the HPP, as discussed below.

Discussion

The staff has formulated an interim process for prioritizing human performance activities within the agency. This approach was based on the judgments of managers using information and knowledge available to them. The product of this "modified Delphi" process is a prioritized list of human performance activities with highest priorities assigned mostly to near-term activities.

Of more importance, the staff has formulated a disciplined strategy to develop a more technically defensible HPP. The future development of the HPP will begin with the identification of agency needs in the field of human performance. These identifications will be made quantitatively where possible. The ASP data for events, over the last five years, with conditional core damage probabilities greater than 10^{-5} will be reviewed to isolate the human performance contributions. Licensee event reports, insights from individual plant examinations, NRC inspection reports, and results of system studies performed by the then Office for Analysis and Evaluation of Operational Data will also be reviewed . The findings from these efforts will be augmented by human reliability analysis sensitivity studies. We believe that these findings should be compared to error classifications available in the literature. This strategy will lead to the formulation of a list of agency needs that can be justified by NRC line organizations and understood by stakeholders.

The list of human performance needs for NRC will be prioritized by a process now being developed within the Office of Nuclear Regulatory Research. Requirements and closure conditions for the priority activities will be defined, quantitatively where possible, using regulatory analysis guidelines and risk criteria such as those described in Regulatory Guide 1.174.

There are additional steps that will have to be defined to complete the process for the disciplined planning of technically justified work in human performance. Strategies to develop alternative candidate solutions to the prioritized needs will have to be developed. Testing and validation of solutions, as well as requirements for the interfaces among elements of the plan, will also have to be developed. We were pleased to see that the staff plans to interact with INPO in the search for agency needs and candidate solutions.

We are looking forward to the development and implementation of the proposed approach, and plan to hold future meetings to review progress in completing the development of the Human Performance Plan.

Sincerely,

Dana a. Powers

Dana A. Powers
Chairman

References:
1. Memorandum dated October 22, 1998, for The Commissioners, from William D. Travers, Executive Director for Operations, NRC, SECY-98-244, Subject: NRC Human Performance Plan.
2. Report dated June 12, 1998, from R. L. Seale, Chairman, ACRS, to Shirley Ann Jackson, Chairman, NRC, Subject: Proposed Final Draft of the NRC's Human Performance Plan.
3. Letter dated October 8, 1997, from R. L. Seale, Chairman, ACRS, to L. Joseph Callan, Executive Director for Operations, NRC, Subject: Human Performance and Human Reliability Implementation Plan.
4. Report dated February 13, 1997, from R. L. Seale, Chairman, ACRS, to Shirley Ann Jackson, Chairman, NRC, Subject: Human Performance Program Plan.
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 Licencing Basis," July 1998.

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

February 23, 1999

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

Dear Chairman Jackson:

SUBJECT: PROPOSED IMPROVEMENTS TO THE NRC INSPECTION AND ASSESSMENT PROGRAMS

During the 459th meeting of the Advisory Committee on Reactor Safeguards, February 3-6, 1999, we reviewed the proposed changes to the NRC Inspection and Assessment Programs, including initiatives related to the development of performance indicators and a risk-based inspection program, which are discussed in SECY-99-007. Our Subcommittees on Plant Operations and Reliability and Probabilistic Risk Assessment also reviewed this matter on January 26, 1999. During these reviews, we had the benefit of discussions with representatives of the NRC staff. We also had the benefit of the documents referenced. We provided an interim letter, dated December 16, 1998, to the Executive Director for Operations on this matter.

Conclusions and Recommendations

1. The process outlined in SECY-99-007 represents a substantial positive step in improving the NRC Inspection and Assessment Programs. The proposed improvements should lead to a risk-informed, efficient process and should improve the objectivity, consistency, and scrutability of these Programs.

2. The objectives of these Programs should be clearly formulated. In particular, the staff should state whether the objectives are to ensure that a specific licensee is maintaining its baseline performance level (related to its licensing basis), or to assess whether any individual plant is an outlier with respect to an expected population-wide performance level.

3. The choice of thresholds for increased NRC attention should be made consistent with the definition of objectives.

Discussion

In response to both the Commission and ACRS concerns, the staff has made substantial progress in improving the NRC Inspection and Assessment Programs for evaluating the performance of nuclear power plant licensees. Since our interim letter, the staff has issued SECY-99-007 which presents recommendations for improvement to the Inspection and Assessment Programs (now termed "Reactor Oversight Process Improvements") in a consolidated manner.

During our discussion of SECY-99-007, two different interpretations of the nature of the inspection program emerged. In one interpretation, inspections are viewed as quality control measures, i.e., a plant is viewed as having an acceptable baseline performance and the inspection program is intended to confirm that the performance remains acceptable. The other interpretation is that the program is intended to identify plants that become outliers with respect to an industry-wide acceptable performance level.

The difference between these two interpretations is whether the acceptable performance levels have different values for different plants. In SECY-99-007, the staff identifies a set of performance indicators (PIs) and sets thresholds for each PI at a level such that 95% of the plants have met this threshold of performance.

The use of this type of threshold on the PIs could imply that the second interpretation is the high-level objective of the Inspection and Assessment Programs. This approach could evolve to be a new, de-facto, regulatory requirement. Furthermore, if the 95% thresholds were to be periodically renormalized, this would constitute a process of continual ratcheting to ever more stringent performance expectations. During our meeting, we discussed the possibility that this could be avoided by developing plant-specific PI profiles and using trends to assess the performance status of the plant with respect to its specific acceptable performance level.

If, on the other hand, the 95% thresholds are one-time settings not subject to renormalization, the use of these thresholds will not lead to ratcheting and would serve the additional purpose of identifying potential outliers. In time, the process would evolve to the point that plant-specific considerations could be used to determine if these "outliers" actually have unacceptable performance.

We have also questioned the constraint of allowing only six months for the pilot program to assess the revised process. The concern is that a six-month pilot program could result in "cramming" (acceleration of both inspections and PI findings) a system intended to be exercised over a full year, such that the results may be distorted.

In addition, we believe that there is a need to use replicates in the pilot program to determine the effects of any uncontrolled variables such as the individuals performing the inspection. Clearly, it will be important to avoid confusing "inspector performance" with "licensee performance." As with any pilot program, there will be uncertainty associated with the results. The staff should include strategies for identifying and controlling such uncertainties in the interpretation of the results of the pilot program.

In the cover letter to SECY-99-007, the staff cites four policy issues that need to be addressed in conjunction with implementation of the revised Inspection and Assessment Programs. We have not heard the details of these policy issues, but expect to review them at a future meeting.

Sincerely,

Dana A. Powers

Dana A. Powers
Chairman

References:
1. Memorandum dated January 8, 1999, from William D. Travers, Executive Director for Operations, NRC, for the Commissioners, SECY-99-007, Subject: Recommendations for Reactor Oversight Process Improvements.
2. Report dated December 16, 1998, from R. L. Seale, Chairman, ACRS, to William D. Travers, Executive Director for Operations, NRC, Subject: Proposed Improvements to the NRC Inspection and Assessment Programs - Interim Report.
3. Memorandum dated November 19, 1998, from John C. Hoyle, Secretary of the NRC, to William D. Travers, Executive Director for Operations, NRC, Subject: Staff Requirements - Briefing on Reactor Oversight Process Improvements.
4. Memorandum dated June 30, 1998, from John C. Hoyle, Secretary of the NRC, to L. Joseph Callan, Executive Director for Operations, NRC, Subject: Staff Requirements, SECY-98-045, Status of the Integrated Review of the NRC Assessment Process for Operating Commercial Nuclear Reactors.

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

March 22, 1999

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

Dear Chairman Jackson:

SUBJECT: SECY-99-054, "PLANS FOR FINAL RULE - REVISIONS TO 10 CFR PARTS 50, 52, AND 72: REQUIREMENTS CONCERNING CHANGES, TESTS, AND EXPERIMENTS"

During the 460th meeting of the Advisory Committee on Reactor Safeguards, March 10-13, 1999, we met with representatives of the NRC staff and the Nuclear Energy Institute (NEI) to discuss SECY-99-054, "Plans for Final Rule - Revisions to 10 CFR Parts 50, 52, and 72: Requirements Concerning Changes, Tests, and Experiments," which includes the staff's proposed resolution of public comments and recommendations for revising 10 CFR 50.59. We also had the benefit of the documents referenced.

CONCLUSIONS AND RECOMMENDATIONS

1. We recommend that the term "minimal" rather than "negligible" be used in the final revision to 10 CFR 50.59. Although the staff and industry have not yet agreed on a definition for "minimal," they agree that "minimal" is greater than "negligible." We believe that the current guidance in NEI 96-07, "Guidelines for 10 CFR 50.59 Safety Evaluations," to determine whether increases in probability are "negligible" is acceptable. The staff's proposed revision removes the "zero risk" constraint in the current rule.

2. We agree with the staff's decision to adopt the industry approach for maintaining the design bases of fission product barriers. There are still some differences between the staff and industry positions relating to the scope of systems to be considered. We believe that these can be resolved in the ongoing discussions between the industry and staff.

3. We support the staff's proposed changes to align 10 CFR Part 71 for packaging and transportation of radioactive material and Part 72 for independent storage of spent nuclear fuel and high-level radioactive waste with 10 CFR 50.59. The staff should continue its work to extend these changes to an international level especially for the

transport of spent nuclear fuel. Experience has shown that having incompatible rules for domestic and international activities create a difficult situation.

4. At this time, there is no benefit from expanding the scope of 10 CFR 50.59. Redefinition of the scope should be considered as part of the risk-informed revision to the rule, which we believe should be pursued on an expedited basis.

5. We believe that the revised 10 CFR 50.59 can and should be implemented earlier than the schedule proposed by the staff.

6. The staff's proposed approach to resolve questions of scope and margin of safety appears to address our concerns.

DISCUSSION

There appears to be improved consistency between the staff's proposed approach and existing industry implementation guidance in NEI 96-07. We continue to believe that the 10 CFR 50.59 process has been implemented successfully for more than 30 years. The objective to simplify, clarify, and restore stability to the 10 CFR 50.59 process justifies the current initiative to revise the rule.

The proposed changes to 10 CFR 50.59 largely codify past practices. It seems, then, reasonable to expect that implementation can be accomplished in a shorter time than is being proposed.

In the March 5, 1999, Staff Requirements Memorandum, the Commission requested that the ACRS provide a list of key questions and issues which should be considered during the current 10 CFR 50.59 rulemaking effort along with any recommended answers or positions. We previously provided a list of questions to the Commission on this matter in our February 18, 1999 report. The staff and industry are making progress in resolving issues/questions associated with margin of safety. We believe that the ongoing dialogue between the industry and staff will resolve several of the key issues. Commission direction and guidance on proposed final rule language would expedite the current 10 CFR 50.59 effort.

We support completion of the proposed rulemaking to provide stability to the 10 CFR 50.59. process and look forward to reviewing the proposed final rule. The focus should soon shift to developing a risk-informed version of 10 CFR 50.59.

Sincerely,

Dana A. Powers

Dana A. Powers
Chairman

References:

1. Memorandum dated February 22, 1999, from William D. Travers, Executive Director for Operations, NRC, to the Commissioners, SECY-99-054, Subject: Plans for Final Rule - Revisions to 10 CFR Parts 50, 52, and 72: Requirements Concerning Changes, Tests, and Experiments.
2. Memorandum dated March 5, 1999, from Annette Viette-Cook, Secretary, NRC, to John T. Larkins, Executive Director, ACRS/ACNW, Subject: Staff Requirements - Meeting with Advisory Committee on Reactor Safeguards, February 3, 1999.
3. Report dated February 18, 1999, from Dana A. Powers, Chairman, ACRS, to Shirley Ann Jackson, Chairman, NRC, Subject: List of Questions to be Addressed for Possible Resolution of Key Issues Associated with the Proposed Revision to 10 CFR 50.59 (Changes, Tests and Experiments).
4. Nuclear Energy Institute, NEI-96-07, Revision 0, "Guidelines for 10 CFR 50.59 Safety Evaluations," September 1997.

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

March 22, 1999

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

Dear Chairman Jackson:

SUBJECT: CORE RESEARCH CAPABILITIES

During the 460th meeting of the Advisory Committee on Reactor Safeguards, March 10-13, 1999, we discussed the Office of Nuclear Regulatory Research (RES) proposal described in SECY-99-064, "Core Research Capabilities." We had the benefit of the documents referenced.

In SECY-99-064 RES has proposed to make no further revisions to its core research capability document, SECY-98-076. RES has concluded that competing priorities, such as the RES self-assessment, will be of greater benefit for RES than investing resources at this time to modify the core capability study.

We agree with the RES proposal to cease further work to modify the core capability assessment described in SECY-98-076.

Sincerely,

Dana A. Powers

Dana A. Powers
Chairman

References:
1. Report dated June 16, 1998, from R. L. Seale, Chairman, ACRS, to Shirley Ann Jackson, Chairman, NRC, Subject: Review of SECY-98-076, "Core Research Capabilities."
2. SECY-98-076, Memorandum dated April 9, 1998, from L. Joseph Callan, Executive Director for Operations, NRC, for the Commissioners, Subject: Core Research Capabilities.
3. SECY-99-064, Memorandum dated March 2, 1999, from William D. Travers, Executive Director for Operations, NRC, for the Commissioners, Subject: Core Research Capabilities.

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

March 22, 1999

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

Dear Dr. Travers:

SUBJECT: LESSONS LEARNED FROM THE ACRS REVIEW OF THE AP600 DESIGN

During the 460th meeting of the Advisory Committee on Reactor Safeguards, March 10-13, 1999, we completed deliberations regarding lessons learned from our review of the AP600 passive plant design. As noted in our July 23, 1998 report, issues on the safety aspects of the AP600 application were resolved to our satisfaction. In the course of our review, however, we identified some lessons learned that could affect reviews of future applications or that could be relevant to operating plants.

Recommendations

1. Guidelines on the acceptable quality of documentation submitted by the applicant and on the lead times necessary for staff reviews should be established and enforced.

2. Safety evaluation reports should include more of the technical rationale leading to the regulatory decision.

3. The NRC research program to improve and consolidate thermal-hydraulic codes should be continued.

4. Guidance for acceptable scaling methods, such as the Code Scaling, Applicability, and Uncertainty (CSAU) evaluation methodology, and for acceptable utilization of integral test data for the validation of computer codes should be developed.

5. The development of technical and policy guidelines for approving requests for reducing the main control room staffing levels below present regulatory limits should be considered.

6. More experiments and analyses will be required before in-vessel core debris retention can be credited as part of the licensing basis.

7. Better standards for qualification of catalytic hydrogen recombiners should be required before approving these recombiners for use as safety-related equipment in nuclear power plants.

Quality and Timeliness of Material Submitted

Our review was made particularly difficult because the associated documentation was submitted piecemeal, was sometimes of poor quality, and contained technical errors. For future applications, the staff should establish and enforce guidelines on the acceptable quality of documentation and on the lead times necessary for staff reviews.

The section of the safety evaluation report (SER) related to the AP600 test and analysis program lacked sufficient technical rationale for us to judge the quality of the staff's review. Our Thermal-Hydraulic Phenomena Subcommittee had to perform a much more exhaustive review than should have been necessary in order to become convinced of the adequacy of the staff review. Future SERs should include more of the technical rationale used to make regulatory decisions.

Thermal-Hydraulic Code Development

Our review identified deficiencies in the existing suite of NRC thermal-hydraulic codes and databases. In order to ensure that the staff has an acceptable thermal-hydraulic analysis capability for confirmatory review of license applications and amendments, the NRC research program to improve and consolidate thermal-hydraulic codes should be continued.

Code Validation Process

The scope of the Westinghouse test and analysis program in support of the AP600 certification was extensive. However, the test program was completed prior to both the scaling analyses and the phenomena identification and ranking process. Because of this, we had considerable difficulty in evaluating both the quality of the data used to validate the computer codes and the scaling of the test results to AP600 conditions. The staff should develop guidance for acceptable methods for scaling and uncertainty evaluation, such as the CSAU evaluation methodology, and for acceptable utilization of integral test data for the validation of computer codes. This is especially crucial as we make more use of best-estimate models for emergency core cooling system requirements.

Main Control Room Staffing Levels

The AP600 is designed to allow the reactor safety systems to remove decay heat without any required operator actions for up to 72 hours after the onset of a severe accident. In addition, the instrumentation and control systems and the human factors design of the main control room provide improved access to information on plant operating parameters. This facilitates and speeds the operator's ability to diagnose problems. Based on these developments and the results of current human factors research, the staff should consider developing technical and policy guidelines for reviewing and approving licensee and applicant requests for reducing the main control room staffing levels below present regulatory limits.

In-Vessel Retention of Core Debris

The AP600 design contains provisions to flood the reactor cavity to cover a significant portion of the reactor vessel. It was argued that this design provision could result in the removal of sufficient heat to prevent core debris from penetrating the vessel. Although this strategy was not part of the AP600 licensing basis, such a strategy might be included in future license amendment requests.

The staff identified weaknesses in the in-vessel core debris retention study used to support the AP600 application. The staff found that the results were quite sensitive to assumptions concerning the mass of metallic core debris in the vessel plenum and the magnitude of upward heat flux induced by vaporization of volatile constituents of core debris. In addition, analyses by the staff questioned assumptions made in the study concerning material properties. There are also unresolved questions about materials interactions, such as intermetallic reactions between molten Zircaloy cladding and the reactor vessel.

More experiments and analyses are needed before in-vessel core debris retention can be credited as part of the licensing basis. At this time, we believe in-vessel core debris retention should only be considered as a severe accident management strategy.

Catalytic Hydrogen Recombiners

The design of the AP600 utilizes hydrogen recombiners to control the accumulation of hydrogen in the reactor containment following a design-basis accident. The AP600 design also contains hydrogen igniters to prevent hydrogen accumulation in the event of more serious beyond-design-basis accidents. The possible use of catalytic processes to control hydrogen concentrations in reactor containments is gaining popularity throughout the world.

The catalytic recombiners that are proposed for use in the AP600 are based on palladium or platinum dispersed on alumina. There is lacking, however, a good understanding of the vulnerabilities of these devices to the environment expected to exist following either design basis or severe accidents. There is not yet a good understanding of what would constitute persuasive qualification of a catalytic recombiner. We believe that the staff should establish better standards for the qualification of these devices.

Dr. Thomas S. Kress did not participate in the Committee's deliberation regarding external reactor vessel cooling.

Dr. Dana A. Powers did not participate in the Committee's deliberation regarding the results of Sandia National Laboratories' tests on qualification of passive autocatalytic recombiners.

Dr. George Apostolakis did not participate in the Committee's deliberation regarding the analyses performed by the Idaho National Engineering and Environmental Lal concerning external reactor vessel cooling.

Sincerely,

Dana A. Powers

Dana A. Powers
Chairman

Reference:
Report dated July 23, 1998, from R. L. Seale, Chairman, ACRS, to Shirley Ann Jackson, Chairman, NRC, Subject: Report on the Safety Aspects of the Westinghouse Electric Company Application for Certification of the AP600 Passive Plant Design.

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

March 22, 1999

MEMORANDUM TO: William D. Travers
Executive Director for Operations

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

SUBJECT: DEGRADED SWITCHYARD VOLTAGE ISSUES AT PALO VERDE NUCLEAR GENERATING STATION UNITS 1, 2, AND 3

During the 460th meeting of the Advisory Committee on Reactor Safeguards, March 10-13, 1999, the Committee considered the February 16, 1999, letter from Arizona Public Service (APS) Company to you commenting on the statement made by the Committee in its letter dated November 23, 1998, regarding the Reprioritization and Proposed Resolution of GSI-171, "Engineered Safety Features Failure From Loss-of-Offsite-Power Subsequent to a Loss-of-Coolant Accident." In its letter, the Committee stated "NRR has raised concerns that degraded switchyard voltage events at Salem and Palo Verde nuclear plants indicate it is possible that plants have either not implemented under-voltage protection properly or conditions have changed that invalidate original design basis capability." This statement was quoted from the background information (Ref. 3) provided by the staff during the Committee's review of the proposed resolution of GSI-171.

The ACRS letter points out the NRR concerns stemming from the degraded switchyard voltage events and does not imply that Palo Verde has not implemented the under-voltage protection properly. Since ACRS does not plan to pursue this issue, additional information from, or a meeting with, APS is not necessary. The Committee suggests that the staff respond to the APS letter by clarifying the intent of the statement quoted in the ACRS letter.

The Committee would like to be kept informed of the staff's response to APS.

References:
1. Letter dated February 16, 1999, from James M. Levine, Arizona Public Service, to Nuclear Regulatory Commission, Subject: Palo Verde Nuclear Generating Station (PVNGS), Units 1, 2, and 3, Degraded Switchyard Voltage Issues at PVNGS.
2. Letter dated November 23, 1998, from R. L. Seale, Chairman, ACRS, to William D. Travers, Executive Director for Operations, NRC, Subject: Reprioritization and Proposed Resolution of Generic Safety Issue-171, "Engineered Safety Features Failure From Loss-Of-Offsite-Power Subsequent To A Loss-Of-Coolant Accident."
3. Memorandum dated August 18, 1998, from Charles E. Rossi, Office for Analysis and Evaluation of Operational Data, NRC, to John W. Craig, Office of Nuclear Regulatory Research, NRC, Subject: Request For Review of the Re-prioritization of GSI-171, "Engineering Safety Features Failure From A Loss-Of-Offsite Power Subsequent to a Loss-Of-Coolant Accident."

cc: A. Vietti-Cook, SECY
J. Blaha, OEDO
J. Mitchell, OEDO
S. Collins, NRR
A. Thadani, RES

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

March 23, 1999

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

Dear Dr. Travers:

SUBJECT: PROPOSED AMENDMENT TO 10 CFR 50.72, IMMEDIATE NOTIFICATION AND 50.73, LICENSEE EVENT REPORTING SYSTEM

During the 460th meeting of the Advisory Committee on Reactor Safeguards, March 10-13, 1999, we reviewed the proposed amendment to 10 CFR 50.72 and 50.73. During our review, we had the benefit of discussions with representatives of the NRC staff and the Nuclear Energy Institute (NEI), and of the document referenced.

CONCLUSIONS AND RECOMMENDATIONS

- The proposed amendment is a significant improvement over the current rule and should be issued for public comment.

- As noted by the staff, reports of equipment surveillance tests that are performed late are not needed provided that the equipment passes the test. The staff should amend the rule to this effect and not just revise the associated regulatory guide.

- We endorse the staff proposal to eliminate the requirement to report an unanalyzed condition that compromises plant safety because such a condition would be reported in accordance with other requirements.

- The staff should examine comprehensively the NRC reporting requirements to ensure that no unnecessary duplications or inconsistencies exist.

- We fully support the staff's position that licensees should report the actuation of risk-significant systems. Lists of such systems should be plant-specific and should be developed on the basis of probabilistic risk assessment (PRA) insights and individual plant designs. These lists should not be included in the rule.

DISCUSSION

While remaining consistent with the agency's reporting needs, the proposed amendment would reduce the reporting burden on licensees by modifying or eliminating requirements that do not provide needed data or that require data which are available through other reporting requirements. In the case of licensee event reports (LERs), extending the reporting due date from 30 to 60 days should enable licensees to complete a root-cause analysis and develop appropriate corrective actions. This change alone would reduce the number of supplemental LERs and thereby reduce the burden on both the NRC staff and licensees.

The staff has indicated that reports on events other than those classified as emergencies would be made within 8 hours. This class of reports would capture events where NRC actions may be required within the next 24 hours, such as initiating a special inspection or contacting a licensee to obtain a better understanding of the event. An advantage of this change is that it provides licensees the opportunity to submit a more detailed description of the event.

The staff has proposed eliminating the requirement to report an unanalyzed condition that significantly compromises plant safety because such a condition would be reported in accordance with other requirements. We agree that this requirement should be dropped.

The staff has proposed eliminating reports about equipment surveillance tests that are performed late, provided that the equipment passes the test when it is performed. This is an improvement to the rule because these reports are not significant since the equipment remains operable during the period of time involved. The NRC's responses to excessively late surveillance testing and to repeated instances of late surveillance testing are covered by other regulations. The staff should amend the rule to effect this proposed change instead of revising the associated regulatory guide.

Reporting requirements for safety system actuations would be changed. Instead of relying on the term "engineered safety feature," the rule would contain a list of specific risk-significant systems. The staff has developed such a list utilizing insights from a small sample of representative PRAs consisting of three pressurized water reactors and two boiling water reactors. NEI noted that the proposed list would result in new reporting requirements for some licensees. We fully support the staff's position that licensees should report the actuation of risk-significant systems. Plant-specific lists of such systems should be developed on the basis of PRA insights and individual plant designs. These lists should not be included in the rule. The stakeholders' workshop being planned by the NRC staff will provide an opportunity to discuss how to develop and document these lists.

The changes contained in the proposed amendment may affect reporting requirements in other regulations. The staff has not completed a systematic review of all the regulations that have reporting requirements and has not assessed whether the various requirements satisfy the needs of the agency. For example, the staff must resolve the difference between the proposed 8 hour reporting requirement and the existing 4 hour reporting requirement in 10 CFR Part 20 regarding radioactive releases.

We have no objection to issuing the proposed amendment for public comment and would like the opportunity to review the proposed final amendment after reconciliation of public comments.

Sincerely,

Dana A. Powers
Chairman

Reference:
Memorandum dated February 19, 1999, from David B. Matthews, Office of Nuclear Reactor Regulation, to NRC Office Directors and Regional Administrators, Subject: Office Review and Concurrence on a Proposed Rule to Modify the Event Reporting Requirements for Power Reactors in 10 CFR 50.72 and 50.73.

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

March 23, 1999

MEMORANDUM TO: William D. Travers
Executive Director for Operations

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

SUBJECT: SAFETY EVALUATION REGARDING COMBUSTIBLE GAS CONTROL IN CONTAINMENT AT INDIAN POINT UNIT 2

During the 460th meeting of the Advisory Committee on Reactor Safeguards, March 10-13, 1999, the ACRS considered the staff's safety evaluation regarding combustible gas control in containment at Indian Point Unit 2. The safety evaluation documented the staff's acceptance of the use of passive autocatalytic recombiners (PARs) for combustible gas control inside the containment. The ACRS decided not to review the safety evaluation associated with Indian Point Unit 2.

The ACRS previously reviewed and commented on the qualification of PARs as part of its review of the AP600 design. The ACRS position on the adequacy of the present qualification requirements for PARs is presented in its March 22, 1999 letter to you on lessons learned from ACRS review of the AP600 design.

References:

1. Memorandum dated February 22, 1999, from Carl H. Berlinger, Office of Nuclear Reactor Regulation, to Singh S. Bajwa, Office of Nuclear Reactor Regulation, Subject: Indian Point 2 Proposed Technical Specification for Hydrogen Recombiners.
2. Letter dated March 22, 1999, from D. A. Powers, ACRS Chairman, to William D. Travers, EDO, Subject: Lessons Learned from the ACRS Review of the AP600 Application.

cc: A. Vietti-Cook, SECY
J. Blaha, OEDO
J. Mitchell, OEDO
S. Collins, NRR
E. Adensam, NRR
G. Holahan, NRR

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

March 24, 1999

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

Dear Dr. Travers:

SUBJECT: GUIDANCE MEMORANDUM FOR IMPLEMENTATION OF THE REVISED ENFORCEMENT POLICY

During the 460th meeting of the Advisory Committee on Reactor Safeguards, March 10-13, 1999, we reviewed the guidance memorandum for implementing the revised Enforcement Policy and discussed proposals for making the Enforcement Policy risk informed. During our review, we had the benefit of discussions with representatives of the NRC staff and the Nuclear Energy Institute (NEI), and of the documents referenced.

Conclusions and Recommendations

- The guidance in the Enforcement Guidance Memorandum (EGM) is intended to aid inspectors in implementing the revised Enforcement Policy and is detailed and comprehensive.

- In coordination with the stakeholders, the staff has proposed additional revisions to the Enforcement Policy. The staff should explain how these proposed revisions will ensure that violations are corrected in a timely manner on the basis of relative risk.

- A process should be developed for entering information associated with violations into the assessment process.

- The staff should use the available risk-informed fire analysis methods as part of the enforcement decisionmaking process.

- The process for assessing the risk significance of all violations should be defined. Tools for quantifying risk associated with violations and for prioritizing the corrective action program should be developed.

- The staff and the stakeholders should continue to discuss concerns associated with the proposed revisions to the Enforcement Policy.

Discussion

During our November 1998 meeting, we reviewed a proposed revision to the Enforcement Policy and concurred with the staff's proposal. The major feature of the revision was the relaxation of the requirement that a notice of violation (NOV) be issued for all Level IV violations. In our report of November 17, 1998, we recommended that the staff develop specific guidance for implementing the revised policy and endorsed the plan to monitor and assess the implementation of this policy. We also recommended that the staff continue discussions with NEI on making other aspects of the Enforcement Policy more risk-informed and objective.

The revised policy became effective on March 11, 1999. An EGM was issued to aid inspectors in implementing this policy. The EGM contains a discussion of the limited conditions under which a Level IV violation would still result in the issuance of an NOV and specific examples of violations that would warrant an NOV. The EGM also includes guidance for documentation and management approval of Level IV violations, including circumstances for exercising enforcement discretion and the structure of the process under which non-cited violations (NCVs) may be appealed. The EGM guidance is detailed and comprehensive.

The staff initiated a pilot study to evaluate the implementation of the revised Enforcement Policy. This study is expected to be completed in 6 months. We would like the opportunity to review the results of the staff's assessment of the implementation of the revised policy.

The staff, in coordination with stakeholders, has developed additional proposed revisions to the Enforcement Policy. These revisions include integration of the enforcement and assessment programs to eliminate duplication of efforts. The staff, however, needs to explain the ways in which the proposed revisions will ensure that violations are corrected in a timely manner on the basis of relative risk. We have the following concerns related to the implementation of these proposed revisions:

- The method by which violations are provided as inputs to the assessment process is unclear. A process should be developed for entering information associated with violations into the plant issues matrix (PIM). For example, an appropriate performance indicator based on violation status could be provided as an input to the PIM.

- The treatment of repetitive violations is still unresolved. The proposal to issue an NOV for repetitive violations identified by the NRC is appropriate, provided that the licensee is given time to correct the initial NCV commensurate with its risk significance.

- The process used in assessing the risk significance of violations at any level is undefined. As with many problems of a nearly routine nature (Level IV violations and 10 CFR 50.59 screening decisions), the level of risk involved in a change in plant configuration or application of procedures falls below the threshold of the sensitivity of a probabilistic risk assessment. Yet, the staff has stated that risk significance will be the basis for prioritizing items in the corrective action program. Tools are needed to meet this expectation.

- Enforcement of fire protection requirements stands in sharp contrast to the intent of using a risk-informed decisionmaking process in evaluating plant status. In the risk-informed process, changes in plant status are evaluated in terms of the resulting change to overall risk. In contrast, the evaluation of the success for fire protection regulations is deterministic and binary. The staff should use the available risk-informed fire analysis methods as part of the enforcement decisionmaking process.

- The methods for quantifying risk associated with violations and the way in which risk judgments are made are not well defined. Development of quantitative risk assessment tools is a challenge to the NRC research program.

Since past discussions have proven to be so productive, the staff and the stakeholders should continue to discuss these concerns at future meetings.

Sincerely,

Dana A. Powers

Dana A. Powers
Chairman

References:
1. Draft Memorandum received February 22, 1999, from James Lieberman, Office of Enforcement, to Multiple Addressees, Subject: Enforcement Guidance Memorandum - Guidance to Implement Interim Power Reactor NCV Policy.
2. Letter dated November 17, 1998, from R. L. Seale, Chairman, ACRS, to Shirley Ann Jackson, Chairman, NRC, Subject: Proposed Revision to the Enforcement Policy.
3. Letter dated March 11, 1999, from Ellen C. Ginsberg, Nuclear Energy Institute, to David L. Meyer, Office of Administration, NRC, Subject: Nuclear Power Industry Comments on "Interim Policy for Severity Level IV Violations Involving Power Reactor Licensees."

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

March 24, 1999

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

Dear Dr. Travers:

SUBJECT: APPLICATION OF WESTINGHOUSE BEST-ESTIMATE LOSS-OF-COOLANT ACCIDENT ANALYSIS METHODOLOGY TO UPPER PLENUM INJECTION PLANTS

During the 460th meeting of the Advisory Committee on Reactor Safeguards, March 10-13, 1999, we reviewed the Westinghouse Electric Company's application of its best-estimate loss-of-coolant accident (LOCA) analysis methodology to plants with Upper Plenum Injection (UPI). Our Subcommittee on Thermal-Hydraulic Phenomena reviewed this matter on December 16, 1998, and February 23, 1999. We also had the benefit of the documents referenced.

The best-estimate LOCA analysis methodology, which utilizes the WCOBRA/TRAC code, has been approved for use in Westinghouse three- and four-loop pressurized water reactors (PWRs). Westinghouse is requesting NRC staff approval to apply this methodology to analysis of large-break (LB) LOCAs in its two-loop plants equipped with UPI of low-pressure emergency coolant. The staff intends to approve the request. This decision is based on the results of a contractor review and the staff's assessment of the methodology, which indicate that Westinghouse has followed the steps described in the Code Scaling, Applicability, and Uncertainty (CSAU) evaluation methodology, met the intent of Regulatory Guide 1.157, and satisfied the emergency core cooling system (ECCS) Rule criteria (10 CFR 50.46 and Appendix K). We note that Regulatory Guide 1.157 allows the staff considerable latitude in deciding on the acceptability and appropriateness of the supporting evidence and analyses.

Conclusions and Recommendations

1. We agree that the results of UPI tests and analyses, as presented by Westinghouse and the Office of Nuclear Regulatory Research, show that UPI plants as currently configured and operated are likely to keep the core cooled following a LBLOCA.

2. WCOBRA/TRAC UPI code predictions of peak cladding temperatures are either conservative or appear insensitive to details in the modeling. We have three concerns:

   - It is not clear that the code can be characterized fairly as "best-estimate" or "realistic" when applied to UPI plants.

- The CSAU evaluation methodology has been carried out in a way that marginally meets the intent of the process.

- Experimental data and sensitivity studies cover a limited range. In the Safety Evaluation Report (SER) the staff should caution that applications of the code be limited to conditions representative of those tested, such as the rates of steam flow in the Upper Plenum Test Facility (UPTF); otherwise, more extensive sensitivity studies and uncertainty calculations should be considered.

3. The NRC staff needs to develop a more proactive, comprehensive, and structured process to support the review of thermal-hydraulic codes.

Discussion

Evidence for the effectiveness of UPI is based on one larger-than-full-scale UPTF test, in which ECCS water penetrated to the simulated lower plenum for conditions representative of a LB LOCA, and two Cylindrical Core Test Facility scaled tests in which a simulated core was cooled at least as well as in corresponding cold-leg injection tests. Westinghouse was able to model these tests reasonably well with its code. Westinghouse also validated its modeling of the countercurrent flow limit (CCFL) against separate-effects tests of a General Electric (GE) fuel rod assembly and tie plate and against correlations based on results from small-scale, air-water tests of a perforated plate conducted at Northwestern University. Sensitivity studies showed that variation of the critical parameters in the code had no significant influence on predicted peak cladding temperature over the limited range explored.

WCOBRA/TRAC was constructed out of numerous models and correlations derived from limited tests at facilities that often differ greatly from full-scale PWRs (e.g., air-water tests in small, long, straight pipes at low pressure). Many of the correlations, formulae, and models are particularly suspect in the UPI context. For example:

- The physical models in the code are not particularly good for predicting two-phase flows in straight pipes. It is truly remarkable that these same models are able to come so close to representing CCFL data for GE tie plates modeled as an effective length of straight pipe.

- The nodalization used by Westinghouse results in modeling of favorable paths for water penetration to the lower plenum. This, however, is only an approximate treatment of the many parallel paths provided by the numerous holes in the tie plate. Such problems have been addressed more comprehensively in the chemical industry.

- No attempt is made to realistically model the thermal-hydraulic phenomena in the upper plenum. A jet with considerable momentum, directed at a forest of structures, is treated as either a slug of water with no momentum or as a dispersed fog of drops with no momentum. Both assumptions are unrealistic, and it is not conclusive that they bound the actual behavior, but they are used as the basis for sensitivity calculations.

- The model for de-entrainment in WCOBRA/TRAC is based on droplet diffusion, not on the inertial impaction that actually occurs.

- Condensation is empirically modeled by means of a coefficient, which Westinghouse varies over a limited range, that does not reflect basic technical uncertainty and is tuned to a small data set.

The staff has stated that the CSAU evaluation methodology was followed, but we recognize a number of shortcomings:

- CCFL modeling was verified from the GE tests but data from separate-effects tests performed at the University of Hanover and at the Idaho National Engineering and Environmental Laboratory, using PWR geometries that are more typical of Westinghouse plants, were ignored. CCFL is known to be significantly dependent on geometrical details.

- Results of the UPTF tests show that more condensation occurred in the upper plenum than was predicted by the code. Yet, the condensation coefficient was not ranged upwards to try to represent this. Had this been done, the predicted CCFL would probably have been more restrictive.

- There was little investigation of the possibility of compensating errors. For example, the underestimation of condensation mentioned above was probably balanced by underestimation of three-dimensional effects that allowed more penetration of water than was permitted by the limited noding in the code formulation.

- Although the calculated peak cladding temperature was insensitive to variations in the parameters that were ranged, it is clear that there are some values for these parameters, particularly interphase drag, that would significantly restrict water penetration. It would have been useful to extend the exploration of parameters into this region in order to know how much margin was available in the uncertainty range for coefficients that are known to be sensitive to conditions such as geometrical details.

We and our consultants raised these and other technical issues during our discussions, but Westinghouse and the staff regarded them as irrelevant to the overall conclusions. Although this may be broadly true in the present context, there is no assurance that it will always be so. Therefore, we believe that the staff needs to provide more explicit guidance regarding the quality of the application of the CSAU evaluation methodology and the code validation requirements. The lessons learned during this review are particularly timely because the staff is presently developing such guidance for future code evaluations. We believe that to carry out such evaluations the staff should:

- Have the capability to run the codes under review in a comprehensive, probing, critical, and objective manner so that a truly independent assessment is made.

- Maintain a thorough understanding of technical issues so that it is aware of when to question circumstances in which codes may be misleading or inadequate. One cannot rely on assurances from protagonists or on a routine following of steps in a process.

- Have its own code of sufficient quality that it can be used to assess the viability of other codes in situations where experimental evidence is not available or is inconclusive.

Throughout the coming year, we will be reviewing other codes intended for use in safety analyses. We look forward to working with the staff to develop the appropriate procedures.

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

Sincerely,

Dana A. Powers

Dana A. Powers
Chairman

References:

1. Letter dated August 6, 1998, from H. A. Sepp, Westinghouse, to Nuclear Regulatory Commission, transmitting Comparison of Best-Estimate LOCA Methodologies for Westinghouse PWRs With Upper Plenum Injection and Cold Leg Injection.

2. Westinghouse Topical Report, WCAP-14449-P, "Application of Best Estimate Large Break LOCA Methodology to Westinghouse PWRs With Upper Plenum Injection," August 1995, including an appendix of information provided to the NRC in response to requests for additional information on WCAP-14449-P **(contains proprietary information)**.

3. Letter dated December 2, 1998, from H. A. Sepp, Westinghouse, to Nuclear Regulatory Commission, Subject: Information Regarding the December 16, 1998, Meeting With the ACRS Thermal-Hydraulic Phenomena Subcommittee.

4. Excerpts from Westinghouse Topical Report, WCAP-12945-P-A, "Westinghouse Code Qualification Document for Best Estimate Loss of Coolant Accident Analysis," March 1998 **(contains proprietary information)**.

5. Letter dated July 12, 1995, from N. J. Liparulo, Westinghouse, to Nuclear Regulatory Commission, Subject: Summary of Westinghouse Best-Estimate LOCA Methodology.

6. E-Mail dated February 16, 1999, from G. Wallis, ACRS Member, to P. Boehnert, ACRS Staff, transmitting list of questions for Westinghouse response at the February 23, 1999 Thermal Hydraulic-Phenomena Subcommittee Meeting.

7. Response from Westinghouse to G. Wallis, ACRS Member, regarding List of Questions to be addressed at the February 23, 1999 Thermal-Hydraulic Phenomena Subcommittee Meeting.

8. Safety Evaluation by the Office of Nuclear Reactor Regulation Related to Acceptability of the Westinghouse Topical Report WCAP-14449(P), "Application of Best-Estimate Large Break LOCA Methodology to Westinghouse PWRs With Upper Plenum Injection."

9. U. S. Nuclear Regulatory Commission report prepared by Idaho National Engineering and Environmental Laboratory, INEEL/EXT-98-00802, Draft, Rev. 1, "Draft Technical Evaluation Report, Application of Best-Estimate Large Break LOCA Methodology to Westinghouse PWRs With Upper Plenum Injection, WCAP-14449-P," undated.

10. U.S. Nuclear Regulatory Commission, Regulatory Guide 1.157, "Best-Estimate Calculations of Emergency Core Cooling System Performance," dated May 1989.

11. U.S. Nuclear Regulatory Commission Report, NUREG/CR-5249, "Quantifying Reactor Safety Margins - Application of the Code Scaling, Applicability, and Uncertainty Evaluation Methodology to a Large-Break, Loss-of-Coolant Accident," December 1989.

12. ACRS Report dated February 23, 1996, from T. S. Kress, Chairman, ACRS, to Shirley Ann Jackson, Chairman, NRC, Subject: Westinghouse Best-Estimate Loss-of-Coolant Accident Analysis Methodology.

13. ACRS Report dated April 19, 1996, from T. S. Kress, Chairman, ACRS, to Shirley Ann Jackson, Chairman, NRC, Subject: Westinghouse Best-Estimate Loss-of-Coolant Accident Analysis Methodology.

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

March 24, 1999

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

Dear Dr. Travers:

SUBJECT: HIGH BURNUP FUEL PHENOMENA IDENTIFICATION AND RANKING

During the 460th meeting of the Advisory Committee on Reactor Safeguards, March 10-13, 1999, we reviewed the status of the NRC confirmatory research program on high burnup fuel. During this review, we had the benefit of discussions with representatives of the Offices of Nuclear Regulatory Research (RES) and Nuclear Reactor Regulation (NRR), the industry, and of the documents referenced.

CONCLUSIONS AND RECOMMENDATIONS

- Conducting an expert opinion elicitation to identify and rank important phenomena that affect high burnup fuel will provide a sound technical basis for refining the NRC's confirmatory research program. It would provide a technical basis for establishing the data and analyses needed to support applications for extending fuel burnup beyond current regulatory limits.

- We urge NRR to participate in the proposed elicitation.

- RES should develop the formalism for conducting and documenting the expert opinion elicitation. Consideration should be given to adapting for the high burnup fuel effort one of the several expert elicitation formalisms developed by NRC in other efforts.

- RES should augment the expert opinion elicitation to include accident source term issues for high burnup fuels.

DISCUSSION

In our report dated June 15, 1998, we discussed the NRC research to confirm the regulatory decision to limit the extent of fuel burnup. In that report, we suggested that the staff develop an understanding of what data and analyses would be required of licensees to support applications for extending fuel burnup beyond the current limit of 62 GWd/t. Development of such an understanding is a challenge since data on high burnup fuel behavior under accident conditions are sparse and scattered.

The RES staff will undertake an expert opinion elicitation to identify the physical and chemical phenomena that will affect fuel behavior, establish the state-of-knowledge concerning these phenomena, and rank them in terms of their importance to safety. The phenomena identification and ranking elicitation is to be done for accident scenarios found by RES to be risk important. These are loss-of-coolant accidents for both pressurized water reactors (PWRs) and boiling water reactors (BWRs), control rod ejection accidents for PWRs, and anticipated transients without scram (ATWS) for BWRs. RES plans to use this expert opinion elicitation to refine its own confirmatory research program, and is in the process of identifying and soliciting the participation of industry experts so that the phenomena identification and ranking can be extended to fuel burnup beyond current regulatory limits.

We are enthusiastic about the use of a disciplined, scrutable expert opinion elicitation to plan and refine the NRC research. We believe the elicitation to be an essential addition to the planning for extended fuel burnup being done by the Nuclear Energy Institute (NEI) and NRR staff. We encourage that both NEI and NRR participate in the effort being undertaken by RES.

Much remains to be done to complete the planning for the phenomena identification and ranking. RES will need to develop the formalism for conducting and documenting the expert opinion elicitation. RES can adapt one of several formalisms developed by the NRC in other efforts.

RES should expand the original scope of its phenomena identification and ranking elicitation to include the issues of accident source term. The revised accident source term (NUREG-1465) approved by the Commission was developed from analyses of fuel taken to burnups that are moderate in comparison to burnups being achieved in current plants. We are concerned that the accident source term will have to be modified to account for the effects of extended fuel burnup. Chemical forms and volatilities of radionuclides may be affected by burnup because of higher oxygen potentials in the fuel and the fuel-cladding gap. Releases of radionuclides from the fuel may be increased because of higher concentrations of interstitial oxygen, more extensive connection of intergrannular porosity, and smaller grain sizes in the so-called "rim" region.

RES has not established a technically defensible position on modifications of the accident source term to account for fuel burnup. Adequate data have not been marshaled. Analytical tools used to date do not appear to include adequate descriptions of pertinent phenomena and processes. Superior analytical tools may be available in the U.S. and in other countries. Additional data may be available from research done abroad. RES will gain substantial benefit for developing a position on modifying the accident source term by including elicitation of expert opinion on the effects of burnup on radionuclide behavior in fuel.

In October 1998, we met with representatives of France, Germany, and Japan to discuss technical issues of mutual interest and as a result formed a Quadripartite Working Group on High Burnup Fuel. We believe this group could contribute to the planned expert opinion elicitation for phenomena identification and ranking.

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

Sincerely,

Dana A. Powers
Chairman

References:
1. Report dated June 15, 1998, from R. L. Seale, Chairman, ACRS, to Shirley Ann Jackson, Chairman, NRC, Subject: NRC Reactor Fuels Research Program.
2. U. S. Nuclear Regulatory Commission, NUREG-1465, "Accident Source Term for Light-Water Nuclear Power Plant," February 1995.

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

March 25, 1999

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

Dear Dr. Travers:

SUBJECT: PROPOSED ASME STANDARD FOR PROBABILISTIC RISK ASSESSMENT FOR NUCLEAR POWER PLANT APPLICATIONS (PHASE 1)

During the 460th meeting of the Advisory Committee on Reactor Safeguards, March 10-13, 1999, we met with representatives of the American Society of Mechanical Engineers (ASME) Committee on Nuclear Risk Management (CNRM) to discuss the proposed Standard for Probabilistic Risk Assessment (PRA) for Nuclear Power Plant Applications (Phase 1). The purpose of this Standard is to provide a means to ensure that the technical quality of PRAs is sufficient to support the regulatory review and approval of licensee risk-informed applications. We also had the benefit of the documents referenced.

Conclusions and Recommendations

1. The proposed Standard has the potential of being very useful to both the industry and the NRC. Although additional work remains, the overall approach to defining necessary PRA requirements is good.

2. Subsection 3.5 on the use of expert judgment and the associated nonmandatory guidance in Appendix A are inconsistent with other parts of the Standard and should be revised. Subsection 3.5 should identify the major issues involving the use of expert opinion in a PRA and not focus on a particular approach.

3. We agree with the CNRM decision to move Section 7 to the beginning of the Standard to present the risk assessment application process early in the document.

4. Consideration should be given in the Standard to recommending participatory peer review throughout the development or application of the PRA in preference to *a posteriori* review.

Discussion

The move toward a risk-informed regulatory system has increased awareness of the need to examine the quality of PRA methodologies. Risk information used for regulatory decisions must be based on credible models and methods.

The lack of confidence in the quality of PRAs will impede their use in the regulatory process. For example, the Individual Plant Examination (IPE) Insights Report (NUREG-1560) showed that there is variability in PRA results that can be attributed to different analytical tools used by licensees. On the basis of its review of licensee IPEs, the staff determined that assumptions used by some licensees were unacceptable and requested those licensees to improve their analyses. The development of a Standard that defines the necessary and minimum requirements for acceptable PRA quality is, therefore, essential.

Developing this Standard is not a straightforward process. If the Standard is too prescriptive, it could impede the further development and refinement of PRA models. On the other hand, simply listing all the methods and models that analysts have used or proposed in the past is not helpful because it presents all such tools as being equally credible or useful when, in fact, experience has shown that they are not.

We believe that the CNRM, who developed the proposed Standard, has established an appropriate balance between specificity and flexibility. The proposed Standard provides requirements that the CNRM believes are necessary for a quality PRA. Although there are references to methods in which there is broad consensus on their appropriateness, the CNRM has wisely refrained from being overly prescriptive in areas where the choice of methods is less clear. Because the actual methods for satisfying the requirements are not prescribed, merely meeting the requirements does not guarantee that a PRA will be of acceptable quality. Thus, the Standard also requires a peer review process to ensure acceptable quality. We agree with the CNRM that a robust peer review process is at present the best way to assess quality. Consideration should be given in the Standard to recommending participatory peer review throughout the development or application of the PRA in preference to just a review after completion of the work.

An exception to the CNRM decision not to specify methods is the treatment of expert judgment. Expert judgment has proven to be a ubiquitous element of modern PRAs for nuclear power plants. Overall, the proposed treatment of expert judgment in the Standard and in the nonmandatory Appendix A touches on nearly all the points that are needed. It puts an unwarranted emphasis on a particular approach to expert judgment. Subsection 3.5 should be revised to be consistent with the remainder of the Standard. Also, since it is not common practice to employ formal expert judgment methods in Level 1 PRAs, a discussion of the conditions requiring such treatment, with examples, would be very useful.

Subsection 7.5 requires that the users determine whether the scope and level of detail of the Standard are sufficient for an application and to provide a technical basis for this determination. Additional guidance should be provided in the Standard to clarify what is expected of the users.

To date, the work done to develop the proposed Standard and associated guidance is commendable. The Standard, when integrated with other industry and NRC initiatives, should greatly enhance progress toward risk-informed nuclear operations and regulatory decisionmaking. We applaud the staff for initiating this effort and for actively participating in the working committees.

We offer detailed comments in the attachment to this letter for the benefit of the CNRM in developing the proposed final version of the Standard and the NRC staff in considering possible endorsement. We look forward to reviewing the proposed final Standard following the reconciliation of public comments.

Sincerely,

Dana A. Powers

Dana A. Powers
Chairman

References:
1. American Society of Mechanical Engineers, ASME RA-S-1999 Edition Draft #10, "Standard for Probabilistic Risk Assessment for Nuclear Power Plant Applications," draft released for public comment, dated February 1, 1999.
2. American Society of Mechanical Engineers, "White Paper and Guidance for Reviewers of the Draft ASME Standard for Probabilistic Risk Assessment for Nuclear Power Plant Applications," received February 8, 1999.
3. U.S. Nuclear Regulatory Commission, NUREG-1560, "Individual Plant Examination Program: Perspectives on Reactor Safety and Plant Performance," Vols. 1-3, December 1997.
4. U.S. Nuclear Regulatory Commission, NUREG-1150, "Severe Accident Risks - An Assessment for Five U.S. Nuclear Power Plants," December 1990.
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.

Attachment: As Stated

ATTACHMENT
Detailed Comments on Proposed ASME Standard for PRA for Nuclear Power Plant Applications (Phase1)

1.1 Scope

Subsection 1.1 states that the Standard sets forth criteria and methods for developing and applying PRA. It should be made clear that the emphasis is on criteria and that particular methods are not prescribed.

2. DEFINITIONS

A. Section 2 requires a thorough review. Considering the broad range of potential applications for this Standard, close scrutiny should be given to ensuring that the definitions are consistent with generally accepted reactor and risk terminology and that terminology used in each section of the Standard is appropriately addressed.

B. Many of the listed definitions are not needed. For example, there is no need to describe a mathematical method such as Monte Carlo simulation. Similarly, there is no need to define a "severe accident." The inclusion of the words "beyond design basis" in the definition is not appropriate.

C. Some of the listed definitions are not useful. For example, an "importance measure" is called a mathematical expression that defines a quantity of interest.

D. Several of the listed definitions are inaccurate or incorrect. Examples of the former are the definitions of "station blackout," "core damage frequency," "unavailability," and "cut sets." An example of the latter is the definition of the "failure rate."

E. Many terms in the text, which should be included in the definitions, are not defined in Section 2. Examples are: EOPs, I&C, ECCS, safety-related SSCs, aleatory and epistemic uncertainties, and single-failure criterion.

3.1 Scope

"Internal Flooding Analysis" is located in the wrong place in Fig. 3.1-1, "Technical Elements of a PRA Model."

3.2 Plant Familiarization

Page 18: An important example of the plant familiarization that should be made explicit is crew performance on simulators during known, generic, time-critical sequences. This provides an appropriate understanding of man-machine interaction.

3.3.1 Initiating Event Analysis

A list of the initiating events that have been used in PRAs should be included with appropriate guidance.

3.3.2 Sequence Development

The explicit description of conditional split fractions and of fault tree linking is appropriate because they are established and accepted approaches. Similarly, a portion of the discussion on event sequence diagrams and system dependency matrices should be removed from the nonmandatory Appendix A and relocated into the main body of the Standard.

3.3.3 Success Criteria

A. Page 23: The list of high-level functions should also include neutronic shutdown.

B. Page 23: Criteria resulting from neutronic analyses should be added to the list of requirements.

C. Page 23: The statement that bounding analyses can be used conflicts with Subparagraph 3.3.4.3, "Use of Realistic Success Criteria."

D. Page 23: Second column: specifies that "Bounding thermal-hydraulic analyses from the plant's SAR ... may be used when detailed analyses are not practical." This statement conflicts with the word "shall" used in Subparagraph 3.3.4.3 to ensure that realistic criteria are used.

3.3.4 Systems Analysis

A. The Standard should caution users that the calculation of the average unavailability of systems with redundant trains is not the product of the average unavailabilities of the individual trains. The time-averaging process introduces dependencies among train unavailabilities.

B. Page 32: The definition of the term "common-cause equipment failure" is not consistent with the definition provided in Section 2.

3.3.5 Data Analysis

A. Page 35: Although it is stated that the subjectivist approach to probability ought to be adopted, the Standard proceeds to discuss frequentist methods (Subparagraphs 3.3.5.1.4 and 3.3.5.3.5) that are inconsistent with this recommendation on the subjectivist approach.

B. Page 35: The Standard should be clarified to state when frequentist methods can be used and for what purpose. It should state that no PRA that has uncertainty analysis has considered these methods useful.

C. Page 40: The Standard should be clarified to state that the analysis of common-cause failures will require the use of generic data that are applicable to the specific plant under analysis.

3.3.6 Human Reliability Analysis

Page 45: The statement in Subparagraph 3.3.6.3.1 that recovery actions shall be limited to those actions for which some procedural guidance is provided or for which operators receive frequent training is inconsistent with the statement in 3.3.7.6 that extraordinary recovery actions that are not proceduralized shall be justified in the analysis.

3.3.8 Level 1 Quantification and Review of Results

A. Page 51: It is not clear what the CNRM means in Paragraph 3.3.8.1.2 by the exception stating, "If only point estimate quantification is completed, that point estimate shall be the mean." Does this mean that the "mean value" should be calculated using rigorous methods? What does the CNRM mean by "point estimates"?

B. Page 51: The requirement in Subparagraph 3.3.8.1.3 that model uncertainty be evaluated needs additional discussion. This evaluation can range from a quick estimate of uncertainty to the use of formal methods for expert opinion elicitation, as was done in NUREG-1150. Furthermore, additional guidance should be provided to clarify how the sensitivity studies should be done and how the results may be used.

3.3.9 Level 1 and Level 2 Interface

A. The determination of uncertainty should be given more discussion and a more prominent position in the Standard.

B. Page 55: The second example of accident sequence characteristics that should be considered refers to the "RCS pressure at core damage." This should be replaced with the "RCS pressure at the time of vessel penetration."

C. There should be a brief discussion on how to extract the Regulatory Guide 1.174 equivalent [large, early release frequency (LERF)] from the results of the detailed Level 2 PRA analysis.

3.4.2 Mapping of Level 1 Sequences

These risk assessments depend on the adequacy of the user's modeling of the physical response of the entire system to accident conditions. For example, whether or not a fan cooler fails due to internal waterhammer, or waterhammer in a piece of pipe to which it is connected, depends on many details of the piping geometry, ups and downs, water-storage tanks, starting transients of pumps when connected to the entire system of pipes, valves, tees and components, the rate of rise of containment temperature and humidity, etc. A technical analysis, including evaluation of uncertainties in modeling, plays the biggest role in assessing failure probability, rather than some characteristics of the device itself. The PRA is fragile if it is not based on the comprehensive analysis of system response. The Standard should reflect this dependence.

3.4.4 Radionuclide Release

A. Page 62: The last bullet calls for "the size distribution of radioactive material released in the form of an aerosol." Isn't this a time-dependent parameter? Is it to be specified as a function of time or an average?

B. Table 3.4.4-1 may be overkill with respect to the needs for determining LERF. Not all of the fission products are significant for LERF although they can be for a full Level 2 PRA analysis.

C. Page 64: Calls for including the release energy in the radionuclide source term. Is this the temperature, the enthalpy, the internal energy? Does it include radioactive energy?

D. Table 3.4.4-2 does not contain all of the key uncertainties. It should be expanded.

E. Page 65: Under the first example, the comment is made that "higher retention efficiencies were attributed to sequences involving low coolant system pressure than those involving high pressure." Is this correct? Was it not the inverse?

F. There is a need to discuss the release and effects of non-radioactive aerosols from the core.

3.5 Expert Judgment

A. What are the criteria for deciding when expert judgment must not be used in order to have a PRA of acceptable quality?

B. When are higher level treatments of expert judgment necessary to ensure that a PRA of acceptable levels of quality is produced? If there are not definable occasions when higher order treatment is needed to ensure adequate quality, why does not the Standard specify the minimum acceptable level of treatment and leave to guidance (i.e., in the Appendix) the discussion of higher levels of treatment that are not likely to ever be used?

C. The Standard requires that the problem to be addressed by the experts be specified in advance. Why is it not required that the experts be allowed to modify the problem? This is allowed in the nonmandatory guidance in Appendix A and would seem to be wise since the experts are very likely to know more about the issue than the PRA team.

D. The Standard requires that the degree of importance of the issue be determined, but provides no quantitative indication of the measure of importance. How can this be omitted if the goal is to have a PRA of adequate quality? The nonmandatory guidance provides some qualitative indications of importance that are sufficiently vague to ensure that all issues can be relegated either to the lowest or to the highest category of importance. Is it not possible to provide a specification of the measure of importance of an issue?

E. The Standard requires also that the complexity of the issue be determined. Here even the nonmandatory guidance is of no help. In the nonmandatory guidance, levels of complexity are described. In some cases these levels are described as "... levels of complexity of the issue under consideration..." (p.103-A-3.5.1[2.2]). But elsewhere these are described as "... levels of complexity in the use of experts..." (p.101-A) and it is apparent that this is the real meaning of the terms. What is the meaning of the "level of complexity of the issue" as specified in Paragraph 3.5.1(b)? What is the measure of complexity to be used?

F. Paragraph 3.5.3: The decision to use outside experts rather than relying on the collective wisdom of the PRA analysis team would seem to be a step in the direction of the quality of the PRA that may not be needed. The decision to do this is left completely to the judgment of the team. Surely, it must be known that there are issues that can be resolved properly for the purposes of producing a PRA of adequate quality only by using outside experts. Why are the characteristics of these issues not described?

G. Paragraph 3.5.4: A crucial step in the formulation of the expert judgment for the PRA is the aggregation of the various expert judgments. No requirements for this step are provided. How is this absence of any specification for such a crucial step consistent with the goal of having a PRA that has adequate quality?

H. Subparagraphs 3.5.4.1 and 3.5.4.2: Regarding Levels A, B, C, and D, there is no indication in the Standard of what these Levels are. The nonmandatory guidance provides some idea of what they are for those who choose to follow this guidance. What are the meanings of Levels A, B, C, and D for those who elect not to follow the nonmandatory guidance? People familiar with the formulation of standards should be added to the group preparing this Standard. Similar flaws arise throughout the discussion in these Subparagraphs. What are four levels of consensus? If the guidance in Appendix A is to be followed, the Standard should require it. Otherwise, revise the Standard so that it stands alone.

I. Why are requirements for documentation of the expert judgment process not mentioned by reference in Subsection 3.5?

4. Documentation

The CNRM provides a listing of specific documentation requirements for a PRA that reflects, one-for-one, the listing of Risk Assessment Technical Requirements provided in Section 3. Although this listing is redundant, a concise listing of these documentation requirements would be helpful in avoiding diverse assessments of the Section 3 requirements. A careful review of Section 4 should follow the rewrite of Section 3. Also, where documentation requirements are stated in Section 4, a more specific statement of the kind of assessments necessary to satisfy these requirements should be useful, e.g., in the evaluation of the consequences of a residual heat removal system train failure, an adequate thermal-hydraulics analysis of system response is needed.

6.2 Review Team Personnel Qualifications

A. Define or describe the requirements for "indoctrination on the PRA process."

B. How were the various experience requirements established? e.g., "The team, collectively, shall have 15 years of experience in performing the activities related to the technical elements of the nuclear power plant PRA identified in Section 3 of this Standard."

C. The last paragraph is a documentation requirement, which may not belong in Subsection 6.2.

6.5 Review of Technical Elements

Consider a generic approach to defining when detailed or limited review is required. Consider reducing the redundancy of review guidance.

7.6 Determination of Scope and Level of Detail of Standard are Sufficient for Application

We are perplexed by the suggestion in Subsection 7.5 that the users determine whether the Standard is sufficient. Subsection 7.5 should be expanded to provide detailed guidance regarding the determination that the Standard is not sufficient to support a particular application and why alternative methods are needed. Also, a new section should be added to provide guidance on how users may recommend improvements to the Standard and for ASME to maintain and update the Standard.

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

April 14, 1999

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

Dear Chairman Jackson:

SUBJECT: PROPOSED FINAL REVISION TO 10 CFR 50.65, "REQUIREMENTS FOR MONITORING THE EFFECTIVENESS OF MAINTENANCE AT NUCLEAR POWER PLANTS"

During the 461st meeting of the Advisory Committee on Reactor Safeguards, April 7-10, 1999, we completed our review of the proposed final revision to 10 CFR 50.65 and proposed revision 3 to Regulatory Guide 1.160, "Monitoring the Effectiveness of Maintenance at Nuclear Power Plants." During our review, we had the benefit of discussions with representatives of the NRC staff and the Nuclear Energy Institute (NEI). We also had the benefit of the documents referenced.

Conclusions and Recommendations:

1. We recommend that the staff proceed with the proposed revision to 10 CFR 50.65.

2. We recommend that the staff hold one or more workshops, as needed, for the licensees and regional staff to ensure consistency in implementing the requirements of the revised rule.

3. We support the staff's plan to issue the revised Regulatory Guide 1.160 for industry use before implementing the revised rule.

Discussion:

Both the staff and the industry agree that 10 CFR 50.65 needs to be revised to ensure that the safety assessments described in the current paragraph (a)(3) are recognized as requirements, that is, at a minimum "should" needs to be changed to "shall." The language in the new paragraph (a)(4) clarifies the obvious intent of the original rule.

We support the staff's position that the safety assessment should consider all components that are taken out of service at the same time. NEI has suggested that the scope of the revised rule be limited to high safety significant structures, systems, and components (SSCs), which are ranked using the guidance specified in NUMARC 93-01, "Industry Guideline for Monitoring the Effectiveness of Maintenance at Nuclear Power Plants." It is not apparent that components ranked as having low safety significance will continue to be of low safety significance under all the configurations that can occur when multiple components are simultaneously taken out of service. In the proposed revision 3 to Regulatory Guide 1.160, the staff provides guidance for assessing the safety significance of plant configurations that arise in the course of doing maintenance. The proposed rule and revision 3 to Regulatory Guide 1.160 are sufficiently flexible that the assessments can be performed without imposing excessive burden on the licensees.

The language in the revised rule expands the scope of the rule from monitoring or preventive maintenance activities to a wider range of maintenance activities. We support this change because there is no reason to require safety assessments for monitoring or preventive maintenance activities and not require such assessments for other types of planned maintenance activities. Expanding the scope of the rule to include such assessments is consistent with the original purpose of the rule.

The other substantive change to the rule is the addition of the introductory sentence clarifying that the rule applies during all conditions of plant operation, including normal shutdown operations. As we have stated on several occasions, we believe shutdown operations of nuclear power plants deserve increased regulatory attention.

The industry has requested guidance for implementing the requirements of the revised rule. It is essential that this guidance be developed with public input in advance of adopting the revised rule. We support the staff's proposal to issue this guidance 120 days before implementing the revised rule and to hold one or more workshops, as needed, for the licensees and regional staff to ensure consistency in the implementation of the revised rule.

The increasing use of on-line maintenance, if properly managed, can provide both cost reductions and improvements in safety. A better definition of the term "safety related" has been identified as a critical step in the development of a risk-informed 10 CFR Part 50. The potential multiplicity of configurations that result from on-line maintenance is one of the elements that must be considered in the development and use of such a definition.

Sincerely,

Dana A. Powers

Dana A. Powers
Chairman

References:

1. Memorandum dated March 9, 1999, from Bruce A. Boger, NRR, to Addressees, transmitting Final Revision to 10 CFR 50.65 to Require Licensees to Perform Pre-Maintenance Assessments.

2. U. S. Nuclear Regulatory Commission, Proposed Revision 3 to Regulatory Guide 1.160, "Monitoring the Effectiveness of Maintenance at Nuclear Power Plants," dated March 5, 1999.
3. Letter dated December 14, 1998, from Anthony R. Pietrangelo, Nuclear Energy Institute, to John C. Hoyle, Secretary of the Commission, Subject: Industry Comments on Proposed Rulemaking to 10 CFR 50.65, Requirements for Monitoring the Effectiveness of Maintenance at Nuclear Power Plants.
4. SECY-98-165, Memorandum dated July 2, 1998, from L. Joseph Callan, Executive Director for Operations, NRC, for the Commissioners, Subject: Proposed Revision to 10 CFR 50.65(a)(3) to Require Licensees to Perform Safety Assessments.
5. Memorandum dated September 3, 1998, from John C. Hoyle, Secretary of the Commission, to L. Joseph Callan, Executive Director for Operations, NRC, Subject: Staff Requirements - SECY-98-165 - Proposed Revision to 10 CFR 50.65(a)(3) to Require Licensees to Perform Safety Assessments.
6. Memorandum dated December 17, 1997, from John C. Hoyle, Secretary of the Commission, to L. Joseph Callan, Executive Director for Operations, NRC, and Karen D. Cyr, General Counsel, NRC, Subject: Staff Requirements: SECY-97-173 - Potential Revision to 10 CFR 50.65(a)(3) of the Maintenance Rule to Require Licensees to Perform Safety Assessments.
7. Letter dated January 22, 1999, from Ralph E. Beedle, Nuclear Energy Institute, to Shirley Ann Jackson, Chairman, NRC, regarding proposed revision to the maintenance rule and 10 CFR 50.59.
8. Nuclear Energy Institute, NUMARC 93-01, Revision 2, "Industry Guideline for Monitoring the Effectiveness of Maintenance at Nuclear Power Plants," April 1996.
9. Letter dated November 25, 1998, from Joe F. Colvin, Nuclear Energy Institute, to Shirley A. Jackson, Chairman, NRC, regarding Stakeholder meeting on November 13, 1998.
10. Letter dated March 23, 1999, from Winston & Strawn to U. S. NRC Commissioners, regarding Proposed Revision to Maintenance Rule.
11. Letter dated March 17, 1999, from R. E. Beedle, Nuclear Energy Institute, to Shirley Ann Jackson, Chairman, NRC, regarding concerns on proposed revision to 10 CFR 50.65, the maintenance rule.
12. 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."
13. Report dated April 23, 1996, from T. S. Kress, Chairman, ACRS, to Shirley Ann Jackson, Chairman, NRC, Subject: Probabilistic Risk Assessment Framework, Pilot Applications, and Next Steps to Expand the Use of PRA in the Regulatory Decision-Making Process.

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

April 14, 1999

MEMORANDUM TO: William D. Travers
Executive Director for Operations

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

SUBJECT: FINAL RULE: "RESPIRATORY PROTECTION AND CONTROLS TO RESTRICT INTERNAL EXPOSURES, 10 CFR PART 20"

During the 461st meeting of the Advisory Committee on Reactor Safeguards, April 7-10, 1999, the Committee considered the subject final rule and decided not to review it. The Committee has no objection to issuing the final rule for industry use.

Reference:
Memorandum dated March 16, 1999, from Frank Akstulewicz, Office of Nuclear Reactor Regulation, to John T. Larkins, Executive Director, ACRS, Subject: Final Rule: "Respiratory Protection and Controls to Restrict Internal Exposures, 10 CFR Part 20."

cc: A. Vieti-Cook, SECY
J. Blaha, OEDO
J. Mitchell, OEDO
S. Collins, NRR
C. Carpenter, NRR
F. Akstulewicz, NRR

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

April 15, 1999

MEMORANDUM TO: William D. Travers
Executive Director for Operations

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

SUBJECT: PROPOSED RULE ON AP600 DESIGN CERTIFICATION

During the 461st meeting of the Advisory Committee on Reactor Safeguards, April 7-10, 1999, the Committee considered the proposed rule on AP600 design certification and decided to review it following reconciliation of public comments.

Reference:
Memorandum dated March 17, 1999, from David B. Matthews, Office of Nuclear Reactor Regulation, to John T. Larkins, Executive Director, ACRS, Subject: Proposed Rule – AP600 Design Certification.

cc: A. Vietti-Cook, SECY
J. Blaha, OEDO
J. Mitchell, OEDO
S. Collins, NRR
D. Matthews, NRR
J. Wilson, NRR

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

April 19, 1999

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

Dear Chairman Jackson:

SUBJECT: STATUS OF EFFORTS ON REVISING THE COMMISSION'S SAFETY GOAL POLICY STATEMENT

During the 461st meeting of the Advisory Committee on Reactor Safeguards, April 7-10, 1999, we met with representatives of the NRC staff to discuss the status of efforts on revising the Commission's Safety Goal Policy Statement. Our Subcommittees on Reliability and Probabilistic Risk Assessment and on Regulatory Policies and Practices discussed this matter with the staff on April 7, 1999. We also had the benefit of the documents referenced.

> **Recommendation.** Revision of the Safety Goal Policy Statement for nuclear power reactors is needed and should be accomplished expeditiously.

In our report dated May 11, 1998, we recommended that the Safety Goal Policy Statement be revised to include: (a) a statement regarding the plant-specific use of the safety goals; (b) an expanded treatment of the role of uncertainties; (c) the removal of the general plant performance guideline; (d) a reconsideration of the set of fundamental goals and subsidiary objectives to ensure that they are consistent; and (e) a reconsideration of measures of societal risk such as environmental contamination and the total number of fatalities. We are pleased that the staff has been considering these issues and is proposing to complete its revision of the Safety Goal Policy Statement for reactors in one year. We agree with this proposal which we believe is necessary to develop a better foundation for making reactor regulation risk informed.

> **Observation.** We agree that it would be conceptually desirable to have an "overarching" Policy Statement for all NRC regulated activities. We do not, however, fully agree on the objectives, scope, utility, feasibility and schedule for developing this Policy Statement.

<u>Objectives</u>

The staff is proposing to develop a high-level overarching Policy Statement to include objectives, goals, and approaches that would apply to all NRC regulatory activities. We agree that such a Policy Statement would provide clarity and consistency to the diverse activities at the

NRC, thereby promoting regulatory stability and increased public confidence. Some ACRS members, however, believe that the primary objective at this time should be the implementation of a risk-informed regulatory system for nuclear reactors. If the staff focuses on developing high-level principles and can relate them to specific needs, then a much better case can be made that the staff is solving key problems which have been identified as impeding progress toward risk-informed regulation.

Scope

Some ACRS members would like to see progress that provides practical benefits before the scope of the Policy Statement is broadened to encompass all NRC regulated activities. This will involve first identifying high-priority needs that can feasibly be resolved by the clarification of high-level principles thereby demonstrating that there are practical benefits.

To this end, we prefer that the near-term effort focus primarily on revising the Safety Goal Policy Statement for nuclear power reactors. Some ACRS members believe that a parallel effort to investigate the issues associated with developing an overarching Policy Statement should be initiated at a more conceptual level. After a reasonable period of time, preferably less than a year, the staff should report its findings and conclusions. A better informed decision on the need to broaden this effort could then be made.

Utility

Several ACRS members expressed concern that the development of the proposed overarching Policy Statement would divert NRC resources from other more important activities, without sufficient likelihood of near-term results. For non-reactor activities, development of an overarching Policy Statement may be premature. Even if successful, such a Policy Statement might be a luxury for nuclear power reactors.

The same ACRS members point out that there is a need, at this time, to revise the existing Safety Goal Policy Statement for reactors to address the issues raised in our May 11, 1998 report, and, in particular, to recognize the practical reality that core damage frequency and large, early release frequency are more useful measures of safety for regulatory purposes than are the quantitative health objectives. This effort should not be encumbered by the requirement for consistency with safety measures yet to be defined for non-reactor activities. The development of analytical tools related to risk-informed regulation for nuclear reactors is more urgently needed than an overarching Policy Statement.

Although we do not agree on the staff's proposal to develop an overarching Policy Statement, we do agree that there are potential benefits for undertaking a feasibility study. Such a Policy Statement should provide practical benefits in terms of the efficiency and effectiveness of regulatory oversight of licensee activities.

Other ACRS members expressed the view that the resources being committed to this task are small and that there should be little concern regarding "diversion of resources." They consider this activity to be so important and essential for a proper, coherent, risk-informed regulatory system that the allocation of additional resources is justified. The potential benefits from developing an overarching Policy Statement applicable to all NRC regulated activities are worth the additional resources required.

The staff's proposal provides overly general assurances of utility of an overarching Policy Statement. We believe that the staff's proposal could be strengthened if, after preliminary exploration that need not be extensive, there is a clear definition of needs and identification of convincing practical use.

Feasibility

Presentations by the staff and industry have indicated that the risk-informed regulatory guides that were published in 1998 are working very well. Several ACRS members believe that this success is due to the fact that the general guidance provided in Regulatory Guide 1.174 starts with a statement of the principles that should govern risk-informed licensing decisions. Therefore, there is evidence to suggest that developing a good set of principles for the overarching Policy Statement is feasible and will lead to a successful outcome.

Several ACRS members believe that development of an overarching Policy Statement is not feasible within a year. These members have raised questions concerning the comparability of risks that have different characteristics. Examples are: (1) the risks from nuclear power plant accidents and high-level waste repository involve vastly different time scales; (2) the risks from nuclear power plants are largely involuntary, while the risks from medical use of radioactive materials can have a substantial voluntary component, and more generally (3) the risks from other industrial applications vary widely in potential accident initiators and frequencies, potential consequences, and populations at risk.

We, therefore, would prefer to see the staff make an early assessment of the feasibility of formulating an overarching Policy Statement through the development of principles. We believe that this will facilitate the development of a more limited Policy Statement that is sufficiently justified and well understood.

Conclusion

The staff has demonstrated great enthusiasm for this undertaking. We believe that this enthusiasm is essential for the vision of an overarching Policy Statement to be realized. This enthusiasm and objectives should be articulated in a short mission statement for the project. After preliminary evaluations have been made, we would like to review the staff's plans. We look forward to assisting the staff in this challenging initiative.

Sincerely,

Dana A. Powers

Dana A. Powers
Chairman

References:

1. Draft SECY paper dated April 2, 1999, from William D. Travers, Executive Director for Operations, NRC, for the Commissioners, Subject: Status of Efforts on Revising the Safety Goal Policy Statement.
2. Memorandum dated June 30, 1998, from John C. Hoyle, Secretary, NRC, to L. Joseph Callan, Executive Director for Operations, NRC, Subject: Staff Requirements - SECY-98-101 - Modifications to the Safety Goal Policy Statement.
3. Report dated May 11, 1998, from R. L. Seale, Chairman, ACRS, to Shirley Ann Jackson, Chairman, NRC, Subject: Elevation of CDF to a Fundamental Safety Goal and Possible Revision to the Commission's Safety Goal Policy Statement.

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

April 19, 1999

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

Dear Dr. Travers:

SUBJECT: SECY-99-017, "PROPOSED AMENDMENT TO 10 CFR 50.55a"

During the 461ST meeting of the Advisory Committee on Reactor Safeguards, April 7-10, 1999, we reviewed SECY-99-017. Also, our Materials and Metallurgy Subcommittee met on March 24-25, 1999, to review this matter. During these reviews, we had the benefit of discussions with representatives of the NRC staff and the Nuclear Energy Institute (NEI), and of the documents referenced.

Recommendation

We recommend against eliminating the 120-month update requirement for inservice inspection (ISI) and inservice testing (IST) programs from the proposed amendment to 10 CFR 50.55a.

Discussion

In May 1995, we decided not to review the proposed amendment to 10 CFR 50.55a until after the staff reconciled public comments. Since then, the proposed amendment has undergone numerous changes. The staff has reviewed the public comments and is preparing the proposed final amendment to 10 CFR 50.55a. Based on internal staff discussions and the public comments, the staff is considering eliminating the regulatory requirement that licensees update their ISI and IST programs to the latest American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code every 120 months. Before proceeding with the final amendment to 10 CFR 50.55a, the staff plans to request public comments specifically on the proposed elimination of the 120-month update requirement.

The staff originally endorsed the ASME Code in 1971. Recognizing that the ASME Code would be updated as experience was gained with its application, the staff also required licensees to update their ISI and IST programs every 120 months.

The primary justifications for the proposed elimination of the update requirement are the maturation of the currently applicable ASME Code and the reduction of the burden on licensees caused by the updating of ISI and IST programs.

We are perplexed by the argument that experience suggests that the current ASME Code requirements have reached such a level of maturity that further updating will provide little benefit. We have recently reviewed a staff safety evaluation report (SER) on a Westinghouse topical report concerning risk-informed inspections. The topical report demonstrated that current ASME Code inspections were not an effective use of resources, and that significant improvements in inspection efficiency could be achieved through the use of risk insights and operational experience. In addition, pilot efforts on risk-informed IST seem to promise similar benefits.

During the past decade, experience has shown that performance demonstrations are superior to prescriptive requirements for qualifying inspectors and inspection techniques. The experience of the past decade has also demonstrated that new modes of degradation can occur and may require changes in inspection procedures. Erosion/corrosion, boiling water reactor (BWR) vessel internals cracking, and circumferential stress corrosion cracking of steam generator tubes were not recognized as important degradation modes a decade ago and inspection procedures had to be updated to deal with such degradation modes. Inspection technologies have also matured. Indeed, in technologies that are heavily dependent on electronics and computer analysis of signals, a decade may represent four or five generations of technology.

This experience suggests that inspection technology is not so static and mature that 120-month updates are unnecessary. Rather, changes in technology and inspection requirements frequently require prompter action than can easily be accommodated by modifications of the ASME Code. The review of operational experience and technology changes through the ASME Code consensus process is important and worthwhile. The 120-month update provides a good baseline for inspection requirements.

In SECY-99-017, the staff recommends the elimination of the 120-month update requirement. Anecdotal information in SECY-99-017 suggests that a typical update may cost a licensee $200,000 to $300,000 every 10 years. An NEI representative cited an anecdotal number of $1 million. Even if this higher estimate is more realistic, the resultant burden does not seem excessive since the actual costs of inspections are far higher than the update costs. Updating would be expected to provide more cost-effective inspections and lower exposures.

In SECY-99-017 the staff states that if the 120-month update requirement is eliminated, licensees who voluntarily choose to update to a later ASME Code edition or addenda will be required to implement all provisions of that edition or addenda. We concur with this staff position on implementing all the provisions of an edition or addenda.

Sincerely,

Dana A. Powers

Dana A. Powers
Chairman

<u>References</u>:

1. SECY-99-017, memorandum dated January 13, 1999, from William D. Travers, Executive Director for Operations, NRC, for the Commissioners, Subject: Proposed Amendment to 10 CFR 50.55a.
2. U. S. Nuclear Regulatory Commission, Safety Evaluation Report Related to "Westinghouse Owners Group Application of Risk-Informed Methods to Piping Inservice Inspection" (Topical Report WCAP-14572, Revision 1), October 1998 (Predecisional).
3. Westinghouse Energy Systems, WCAP-14572, Revision 1, "Westinghouse Owners Group Application of Risk-Informed Methods to Piping Inservice Inspection Topical Report," October 1997.
4. Westinghouse Energy Systems, WCAP-14572, Revision 1, Supplement 1, "Westinghouse Structural Reliability and Risk Assessment (SRRA) Model for Piping Risk-Informed Inservice Inspections," October 1997.
5. Letter dated August 14, 1998, from John N. Hannon, Office of Nuclear Reactor Regulation, NRC, to C. Lance Terry, TU Electric, Subject: Approval of Risk-Informed Inservice Testing (RI-IST) Program for Comanche Peak Steam Electric Station, Units 1 and 2.

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

April 19, 1999

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

Dear Dr. Travers:

SUBJECT: REEVALUATION OF GENERIC SAFETY ISSUE PROCESS

During the 461st meeting of the Advisory Committee on Reactor Safeguards, April 7-10, 1999, we completed our review of the reevaluation of the generic safety issue (GSI) process. During our review, we had the benefit of discussions with representatives of the NRC staff and the documents referenced.

Conclusions and Recommendations

- The preliminary draft Management Directive (MD) 6.4, "Generic Issue Process," and the associated Handbook appear to provide an effective way to implement the revised GSI process.

- We recommend that the staff conduct a pilot study to evaluate the effectiveness of using the MD for implementing the revised GSI process prior to developing a final version of MD 6.4 and the associated Handbook.

- The staff proposes to use a risk-informed technical screening of new generic issues. The staff, however, needs to further develop the screening methodology for estimating the risk significance of generic issues. This methodology should include examination of results of the individual plant examination (IPE) and individual plant examination of external events (IPEEE) processes and should include an uncertainty analysis.

- We remain concerned about the technical resolution of the remaining GSIs. We plan to review the proposed resolution of these GSIs. The staff should provide a schedule for forwarding the resolution packages of these GSIs to us to facilitate our planning of the workload.

Discussion

We have had a long-standing interest in the GSI process. During 1998, we reviewed the mechanism for addressing GSIs and the proposed priority rankings of several GSIs, and identified a number of concerns in our letters of March 16 and October 16, 1998.

As a result of our concerns, the Office of Nuclear Regulatory Research (RES) reevaluated the GSI process to determine what changes were warranted to improve its effectiveness. Based on the reevaluation, RES has developed a revised GSI process to assess issues that are of generic interest but that may or may not be safety significant.

On the basis of the reevaluation, RES has proposed changes to the GSI process that provide for an expanded scope, programmatic purpose, a disciplined process, and the application of management tools to execute the revised process. RES proposes to implement the revised GSI process through the MD and an associated Handbook. We agree with this approach.

The use of risk insights in a screening process is a good practice. The staff described a proposed method for technical screening of generic issues, which would use risk insights related to changes in core damage frequency (CDF) or large, early release frequency (LERF). We have concerns about the applicability of the proposed screening method for generic issues. The problem with the proposed risk-informed screening process is the determination of changes in CDF and LERF due to particular generic issues for the set of affected plants. We believe it is impractical and not cost beneficial to exercise each plant-specific PRA for these determinations. A method for selecting a representative PRA for such determinations has not been defined.

Sincerely,

Dana A. Powers

Dana A. Powers
Chairman

References :

1. Memorandum dated February 8, 1999, from John W. Craig, Office of Nuclear Regulatory Research, NRC, to John T. Larkins, Executive Director, ACRS, Subject: Transmittal of Presentation Material for March 11, 1999 Presentation to ACRS.
2. U. S. Nuclear Regulatory Commission, Rough Draft, Management Directive 6.4, "Generic Issue Process," dated February 22, 1999, Revision 3 (Predecisional).
3. Letter dated March 16, 1998, from R. L. Seale, Chairman, ACRS, to L. Joseph Callan, Executive Director for Operations, NRC, Subject: SECY-98-001, Mechanism for Addressing Generic Safety Issues.
4. Letter dated May 28, 1998, from L. Joseph Callan, Executive Director for Operations, NRC, to Robert L. Seale, Chairman, ACRS, Subject: SECY-98-001, Mechanism for Addressing Generic Safety Issues.

5. Letter dated October 16, 1998, from R. L. Seale, Chairman, ACRS, to L. Joseph Callan, Executive Director for Operations, NRC, Subject: Proposed Priority Rankings of Generic Safety Issues: Tenth Group.
6. Letter dated November 27, 1998, from William D. Travers, Executive Director for Operations, NRC, to Robert L. Seale, Chairman, ACRS, Subject: Proposed Priority Rankings of Generic Safety Issues: Tenth Group.

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

April 22, 1999

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

Dear Dr. Travers:

SUBJECT: STATUS OF RESOLUTION OF STEAM GENERATOR TUBE INTEGRITY ISSUES

During the 461ST meeting of the Advisory Committee on Reactor Safeguards, April 7-10, 1999, we reviewed the staff and industry activities associated with steam generator tube integrity issues. During our review, we had the benefit of discussions with representatives of the NRC staff and the Nuclear Energy Institute (NEI), and of the documents referenced.

Conclusions and Recommendations

1. The NRC staff and the industry have several unresolved technical and regulatory differences associated with ensuring steam generator tube integrity. The staff and industry should continue to work toward the resolution of these differences.

2. The staff is in the process of resolving the differing professional opinion (DPO) issues associated with steam generator tube integrity. We plan to review the staff's resolution of the DPO issues after reconciliation of public comments.

Discussion

We last reviewed the issues associated with steam generator tube integrity during our 444th meeting on September 3-5, 1997, and provided a report to the Commission dated September 15, 1997. At that time, we agreed with the staff's decision not to proceed with rulemaking and recommended that a proposed generic letter and draft Regulatory Guide (DG) 1074, "Steam Generator Tube Integrity," be issued for public comment. We believed that the specifications contained in the proposed generic letter and DG-1074 would improve the integrity of steam generator tubes by having licensees do condition monitoring, operational assessment, and qualification of nondestructive examinations.

In December 1997, the industry committed to implement, on a voluntary basis, NEI 97-06, "Steam Generator Program Guidelines." The programmatic approach of NEI 97-06 is conceptually similar to that specified in DG-1074. Consistent with Direction Setting Issue 13, "The Role of Industry," the staff held discussions with the industry concerning resolution of the differences between NEI 97-06 and DG-1074, and the establishment of an appropriate regulatory framework for implementing NEI guidelines. Anticipating resolution of these differences through additional interactions with the industry, the staff postponed the issuance of the proposed generic letter and withdrew the associated advance notice of proposed rulemaking.

The staff and industry now plan to resolve the remaining technical differences, such as the definition of the differential pressure on a steam generator tube, the choice of an accident induced leakage limit, the definitions of tube burst and tube rupture, and the risk issues associated with probabilistic structural criteria. Following the resolution of these differences, NEI plans to propose to the staff performance-based technical specifications for ensuring steam generator tube integrity. The staff plans to review the NEI proposal and prepare a safety evaluation report, which would allow licensees to request license amendments to modify their technical specifications. We plan to review the results of these efforts.

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

Sincerely,

Dana A. Powers

Dana A. Powers
Chairman

References:
1. SECY-98-248, Memorandum dated October 28, 1998, from Willliam D. Travers, Executive Director for Operations, NRC, for the Commissioners, Subject: Proposed Generic Letter 98-XX, "Steam Generator Tube Integrity."
2. Letter dated December 16, 1997, from Ralph E. Beedle, Nuclear Energy Institute, to L. Joseph Callan, Executive Director for Operations, NRC, Subject: NEI 97-06, "Steam Generator Program Guidelines."
3. ACRS letter dated October 10, 1997, from R. L. Seale, Chairman, ACRS, to L. Joseph Callan, Executive Director for Operations, NRC, Subject: Resolution of the Differing Professional Opinion Related to Steam Generator Tube Integrity.
4. ACRS report dated September 15, 1997, from R. L. Seale, Chairman, ACRS, to Shirley Ann Jackson, Chairman, NRC, Subject: Proposed Generic Letter and Draft Regulatory Guide DG-1074 Concerning Steam Generator Tube Integrity.
5. ACRS report dated June 20, 1997, from R. L. Seale, Chairman, ACRS, to Shirley Ann Jackson, Chairman, NRC, Subject: Proposed Regulatory Approach Associated With Steam Generator Integrity.
6. ACRS letter dated November 20, 1996, from T. S. Kress, Chairman, ACRS, to James M. Taylor, Executive Director for Operations, NRC, Subject: Proposed Rule on Steam Generator Integrity.

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

April 23, 1999

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

Dear Dr. Travers:

SUBJECT: PROPOSED REVISIONS TO THE NRC GENERIC COMMUNICATIONS PROCESS

During the 461st meeting of the Advisory Committee on Reactor Safeguards, April 7-10, 1999, we reviewed the proposed revisions to the NRC generic communications process. During our review, we had the benefit of discussions with representatives of the NRC staff, Nuclear Energy Institute (NEI), and the documents referenced.

DISCUSSION

There are four basic types of generic communications currently in use: (1) bulletins (BLs); (2) generic letters (GLs); (3) information notices (INs); and (4) administrative letters (ALs). The industry and the members of the U.S. Senate have expressed concerns regarding the staff's use of BLs and GLs. The industry argued that the differences in regulatory requirements of these generic communications were not clearly differentiated, and although the NRC has adopted a policy that BLs and GLs be subject to the backfit rule, 10 CFR 50.109, the staff has often inappropriately invoked the compliance exemption of the rule in its requests for licensees' actions. Therefore, recipients of BLs and GLs feel obligated to respond and act on the actions requested. The industry expressed the need for the staff to clearly differentiate the differences between BLs and GLs, and to ensure appropriate consideration of the backfit rule requirements.

In addition, a number of BLs and GLs have invoked 10 CFR 50.54(f) to require licensees to submit information under oath or affirmation that is necessary to enable the Commission to determine whether to "modify, suspend, or revoke" a license. In fact, few of these generic communications have involved potential modification, suspension, or revocation of a license. The staff and the industry agree that the use of 10 CFR 50.54(f) should be restricted.

In responding to these concerns, the staff has proposed approaches to better define and specify requirements associated with BLs and GLs. The staff also has proposed to use the regulatory information letter (RIL) as a new generic communication tool.

CONCLUSIONS AND RECOMMENDATIONS

1. We agree with the staff's proposal for resolving concerns associated with the present use of generic communications. The benefits of this proposal include the following:

   - Reduction in the potential use of generic communications to impose regulatory requirements.

   - Assurance of appropriate consideration of the backfit rule and the associated compliance exemption.

   - Restriction of the use of 10 CFR 50.54(f) to cases in which the Commission is actually contemplating modification, suspension, or revocation of a license.

   - Implementation of a more uniform process across the agency for the use of generic communications.

2. The process for approving these generic communications is not clear from the description included in the draft Commission paper. Neither the generic communication development process discussed in the paper nor the flow chart presented by the staff at our meeting comprehensibly described the role of the Committee to Review Generic Requirements (CRGR) in the process.

3. Guidance for the decision to declare an issue "urgent" should be provided.

4. The staff stated that a limited cost-benefit analysis would be performed, even for cases in which the initial screening indicated that an exemption to the backfit rule was justified. An adequate justification for the limited cost-benefit analysis has not been provided. The staff should make clear that such cost-benefit considerations will only be used as guidance on the appropriate disposition of compliance issues.

5. In the draft Commission paper, the staff proposes that RILs be reviewed by CRGR "as appropriate." Because RILs can be used to announce the staff's technical or policy positions, we recommend that the paper be revised to require that all RILs be reviewed by the Office of Nuclear Reactor Regulation Executive Team and CRGR.

We commend the staff for its early interaction with the industry and its efforts to resolve the concerns associated with the generic communications process.

Sincerely,

Dana A. Powers

Dana A. Powers
Chairman

References:

1. The 105th Congress, U.S. Senate, Report 105-206 dated June 5, 1998, Subject: Energy and Water Development Appropriation Bill, 1999.
2. Letter dated August 11, 1998, from Joe F. Colvin, Nuclear Energy Institute, to Shirley A. Jackson, Chairman, NRC, regarding the July 17, 1998 NRC Public Meeting on Stakeholders' Concerns.
3. Memorandum dated March 3, 1999, from Robert L. Dennig, Office of Nuclear Reactor Regulation, to John Larkins, ACRS, Subject: ACRS Review of Draft Commission Paper on Generic Communication Process.

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

May 11, 1999

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

Dear Chairman Jackson:

SUBJECT: MODIFIED PROPOSED FINAL REVISION TO 10 CFR 50.65, "REQUIREMENTS FOR MONITORING THE EFFECTIVENESS OF MAINTENANCE AT NUCLEAR POWER PLANTS"

During the 462nd meeting of the Advisory Committee on Reactor Safeguards, May 5-8, 1999, we reviewed the modified proposed final revision to 10 CFR 50.65 and proposed revisions to Regulatory Guide 1.160, "Assessing and Managing Risk Before Maintenance Activities at Nuclear Power Plants." During our review, we had the benefit of discussions with representatives of the NRC staff and Nuclear Energy Institute (NEI). We also had the benefit of the documents referenced.

We reviewed a previous version of 10 CFR 50.65 during our 461ST meeting and issued a report dated April 14, 1999. In that report, we stated that both high safety significant structures, systems, and components (SSCs) and low safety significant SSCs need to be addressed by the Maintenance Rule. We note that the usual classification of SSCs as high or low safety significant is based on probabilistic risk assessments (PRAs) of typical configurations at power. A different configuration or a different mode of operation may change the relative rankings of the SSCs.

Since our April 14, 1999 report, the staff has proposed to add the following language to paragraph(a)(4) of 10 CFR 50.65: "Scope of the assessment may be limited to structures, systems, or components that a risk-informed evaluation process has shown to be significant to public health and safety." We recommend the following modification to the staff's proposed language:

> "Scope of the assessment may be limited to structures, systems, or components that a risk-informed evaluation process has shown to be significant to public health and safety for the proposed configuration."

The staff also stated that it is considering revising Regulatory Guide (RG) 1.160 to adopt the configuration risk management program (CRMP) in RG 1.177, "An Approach for Plant-Specific, Risk-Informed Decisionmaking: Technical Specifications." The program described under Key Component 1 of Section 2.3.7.2 of RG 1.177 requires an assessment of all SSCs modeled in the licensee's PRA in addition to all SSCs considered high safety significant that are not modeled in the PRA. This program, however, does not include a discussion of other SSCs. The CRMP was designed for extending outage time as allowed in the technical specifications and may not be appropriate for managing the risk of maintenance activities. Since the number of low safety significant SSCs modeled in licensees' PRAs may vary widely, we are concerned that there may be configurations of SSCs out of service for maintenance that would not have received an assessment. We recommend that the CRMP in RG 1.177 not be adopted.

We believe that licensees need to take responsibility for evaluating and managing the risk associated with taking multiple SSCs out of service. Plant operators should not be confronted with inadequately evaluated plant configurations. This can be avoided by appropriately evaluating the actual configuration. We note that currently operating plants have not been designed with the intent of performing on-line maintenance, but recognize that technology is now available to manage appropriately the risk associated with on-line maintenance. Therefore, we support the industry practice of performing on-line maintenance, as long as this is done safely.

Sincerely,

Dana A. Powers

Dana A. Powers
Chairman

References:

1. Modified proposed Final Revision to 10 CFR 50.65, "Requirements for Monitoring the Effectiveness of Maintenance at Nuclear Power Plants," received May 5, 1999.
2. U. S. Nuclear Regulatory Commission, Revision 3 to Regulatory Guide 1.160 (DG-1082), "Assessing and Managing Risk Before Maintenance Activities at Nuclear Power Plants," April 1999.
3. Report dated April 14, 1999, from Dana A. Powers, Chairman, ACRS, to Shirley Ann Jackson, Chairman, NRC, Subject: Proposed Final Revision to 10 CFR 50.65, "Requirements for Monitoring the Effectiveness of Maintenance at Nuclear Power Plants."
4. U. S. Nuclear Regulatory Commission, Regulatory Guide 1.177, "An Approach for Plant-Specific, Risk-Informed Decisionmaking: Technical Specifications," August 1998.
5. Letter dated December 14, 1998, from Anthony R. Pietrangelo, Nuclear Energy Institute, to John C. Hoyle, Secretary of the Commission, Subject: Industry Comments on Proposed Rulemaking to 10 CFR 50.65(a)(3), Requirements for Monitoring the Effectiveness of Maintenance at Nuclear Power Plants.
6. Letter dated January 22, 1999, from Ralph E. Beedle, Nuclear Energy Institute, to Shirley Ann Jackson, Chairman, NRC, regarding proposed revision to the maintenance rule and 10 CFR 50.59.

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

May 14, 1999

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

Dear Dr. Travers:

SUBJECT: PROPOSED RESOLUTION OF GENERIC SAFETY ISSUE-158, "PERFORMANCE OF SAFETY-RELATED POWER-OPERATED VALVES UNDER DESIGN BASIS CONDITIONS"

During the 462nd meeting of the Advisory Committee on Reactor Safeguards, May 5-8, 1999, we reviewed the proposed resolution of Generic Safety Issue-158 (GSI-158), "Performance of Safety-Related Power-Operated Valves Under Design Basis Conditions." During our review, we had the benefit of discussions with representatives of the NRC staff. We also had the benefit of the document referenced.

Recommendation

We recommend that GSI-158 not be considered resolved. The central issue, whether power-operated valves (POVs) are able to perform their intended functions under design basis dynamic conditions, has not been adequately addressed.

Discussion

The NRC staff recommended closure of GSI-158 based on the results of the analysis performed by the Idaho National Engineering and Environmental Laboratory (INEEL) that revealed that the potential reduction in risk from an improvement in the reliability of POVs will neither result in a substantial safety improvement nor be cost effective. The failure probabilities used in the analysis, however, were based on data reported by licensees for normal operating conditions. The staff did not demonstrate that there are sufficient data or analytical models to establish POV failure probabilities under design basis conditions. Therefore, the results of the INEEL analysis do not provide adequate justification for resolving GSI-158.

The central issue of whether POVs will perform their intended functions under design basis dynamic conditions has not been adequately addressed. Based on a review of POV testing at seven sites, the NRC staff concluded that most licensees were not performing dynamic testing or evaluating whether the static testing performed was indicative of POV performance under dynamic conditions. This indicates that current programs and existing requirements are not sufficient to ensure a systematic evaluation and resolution of GSI-158.

The NRC staff is relying on the Maintenance Rule to ensure that risk-significant valves are properly installed and maintained. The staff stated that there are industry initiatives to address issues associated with POVs. We are concerned that unless the staff undertakes a proactive effort to ensure resolution of this issue, the industry initiative will remain an optional, voluntary program that will not fully address the concerns of GSI-158. We plan to continue our discussion with the staff regarding the resolution of our concerns.

Sincerely,

Dana A. Powers

Dana A. Powers
Chairman

Reference:
Memorandum dated April 5, 1999, from John W. Craig, Office of Nuclear Regulatory Research, to John T. Larkins, ACRS, Subject: Resolution of Generic Safety Issue 158, "Performance of Safety-Related Power-Operated Valves Under Design Basis Conditions."

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

May 17, 1999

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

Dear Chairman Jackson:

SUBJECT: USE OF MIXED OXIDE FUEL IN COMMERCIAL NUCLEAR POWER PLANTS

During the 462nd meeting of the Advisory Committee on Reactor Safeguards, May 5-8, 1999, we completed our response to the Commission request, included in the March 5, 1999 Staff Requirements Memorandum, that the ACRS consider the impact on the revised source term if high burnup or mixed oxide fuel (MOX) were used in place of conventional uranium fuel in commercial nuclear power plants. We had the benefit of the documents referenced.

The U.S. Department of Energy is proposing to dispose of some fraction of the Nation's excess weapons-grade plutonium by converting this plutonium into MOX for use in commercial nuclear power plants. There is, however, rather limited operational or regulatory experience with the use of MOX in the U.S. Even the experience in other countries is not extensive.

We have not had the opportunity to review analyses by the U.S. Department of Energy on the safety of the use of MOX in commercial nuclear power plants, nor have we had the benefit of hearing NRC staff views on this subject. There are technical issues that will merit consideration in evaluating the safety of using MOX. We think there are policy issues that the Commission may want to consider in the evaluation of applications for the use of MOX.

Because current regulations are predicated on the use of low-enrichment uranium oxide fuel rather than MOX, applications for the use of MOX may be burdened by needs to propose amendments to numerous prescriptive regulations. To facilitate the evaluation of applications to use MOX, the Commission may want to encourage the use of the risk-informed approach delineated in Regulatory Guide 1.174 , "An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis," to amend licenses of currently operating nuclear plants. For similar reasons, the Commission may want to consider requiring that such applications adapt the revised accident source term described in NUREG-1465 for deterministic safety evaluations.

Technical issues that arise in the analysis of risk at plants using MOX focus on the vulnerability of fuel to neutronically induced core disruption and the different inventory of radionuclides available for release from the fuel during accidents. The differences in neutronics and coupling between neutronics and thermal hydraulics result in different responses of MOX and conventional fuel to reactivity transients. The differences in responses are consequences of changes in Doppler and moderator reactivity feedback, and decrease in delayed neutron fraction, which decreases the response time of MOX to reactivity transients. These dynamic characteristics of MOX pose both safety and control issues that will require the staff to conduct careful review of the neutronics analysis of reactor cores with MOX. Most experts believe now that the number of MOX fuel assemblies and the percentage of plutonium in MOX should be limited to reduce the vulnerability of the core to these neutronic effects. We are aware that the Office of Nuclear Regulatory Research (RES) is in the process of upgrading the tools available for the analysis of coupled neutronics and thermal hydraulics. As part of this work, RES is assessing uncertainties in the neutronics analyses, including uncertainties in the effective delayed neutron fraction for fuels rich in plutonium. We encourage this work so that improved analytic tools will be available to the staff when the time comes to evaluate an application to use MOX.

We are aware of experimental studies that show there to be enhanced release of fission gases to the fuel-cladding gap during reactor operations with MOX relative to conventional fuels. This may simply be an effect caused by fuel temperature. We are also aware of anecdotal accounts of the results of VERCOURS tests in France dealing with the release of volatile radionuclides such as cesium from MOX under severe accident conditions. Results of these tests revealed that during the early stages of core degradation, releases of volatile radionuclides from MOX are more extensive than from conventional fuels at similar levels of burnup. At higher temperatures at which extensive degradation and melting of fuel take place, integral releases of the volatile radionuclides are similar in the two types of fuel. The higher releases of volatile radionuclides at low temperatures (<2000 K) are consistent with the peculiar nature of porosity that develops in MOX during burnup and are, apparently, sensitive to the heterogeneity of the plutonium oxide distribution in the fuel. Whether these higher releases of volatile radionuclides are adequately estimated for safety analyses using the release prescriptions provided in NUREG-1465 will not be known until further data and analyses become available.

We are aware of a test of the vulnerability of MOX rods to reactivity insertion. The safety significance of the results of this test could be interpreted more confidently once results of the ongoing NRC research program on reactivity insertion in high burnup fuels become available.

Public attention has been drawn to the higher actinide inventories available for release from MOX than from conventional fuels. Significant releases of actinides during reactor accidents would dominate the accident consequences. Models of actinide release now available to the NRC staff indicate very small releases of actinides from conventional fuels under severe accident conditions. There is substantial uncertainty in these predictions. The staff is attempting to validate the predictions of actinide releases through its participation in the PHEBUS-FP program of experimental studies of radionuclide release and transport. There is some hope that the PHEBUS-FP program or a follow-on program will include tests of MOX degradation and fission product release. We encourage the NRC participation in this international collaborative research and hope that definitive results will be available for evaluating the applications to use MOX.

Comparisons are sometimes drawn between the inventories of actinides in MOX and the releases of actinides observed in the accident at the Chernobyl nuclear plant. Such comparisons are not valid in light of the peculiar nature of the accident at Chernobyl and the fact that radionuclide releases are strongly dependent on the details of accident phenomena. It is noteworthy that the releases of actinides during the Chernobyl accident were due almost entirely to fuel dispersal rather than vaporization. It will be important to ensure that fuel dispersal events such as steam explosions and high pressure melt ejection are of acceptably low probability at plants that propose to use MOX.

Our Subcommittee on Reactor Fuels will continue to follow progress in both the use of high burnup fuel and the use of MOX at commercial nuclear power plants. We are participating in a Quadripartite Working Group with our counterparts in France, Germany, and Japan that deals with these topics. We plan to report our observations and conclusions to you, as appropriate.

Sincerely,

Dana A. Powers

Dana A. Powers
Chairman

References:

1. Memorandum dated March 5, 1999, from Annette Vietti-Cook, Secretary of the Commission, to John T. Larkins, ACRS, Subject: Staff Requirements - Meeting with Advisory Committee on Reactor Safeguards.
2. Memorandum dated April 14, 1999, from William D. Travers, Executive Director for Operations, to the Commissioners, Subject: Mixed-Oxide Fuel Use in Commercial Light Water Reactors.
3. 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.
4. U. S. Nuclear Regulatory Commission, NUREG-1465, "Accident Source Terms for Light-Water Nuclear Power Plants," February 1995.
5. T. J. Downar and K.O. Ott, School of Nuclear Engineering, Purdue University, "Comparison of the Spatial Kinetics Codes PARCS and NESTLE and Other Related Issues," August 11, 1997.
6. Hj. Matzke, "Oxygen Potential in the Rim Region of High Burnup UO_2 Fuel," Journal of Nuclear Materials, 208 (1994) 18-26.
7. Oak Ridge National Laboratory, ORNL/TM-13424, R.T. Primm, III, J.C. Ryman, S.B. Ludwig, "Storage of Assemblies Containing Mixed Oxide Fuel." April 1997.
8. Oak Ridge National Laboratory, ORNL/TM-13170/V3, B.D. Murphy, "Characteristics of Spent Fuel from Plutonium Disposition Reactors Vol. 3: A Westinghouse Pressurized Water Reactor Design," July 1997.

9. K. Lassmann, C. O'Carroll, J. van de Laar, C.T. Walker, "The Radial Distribution of Plutonium in High Burnup UO_2 Fuels," Journal of Nuclear Materials, 208 (1994) 223-231.
10. C.T. Walker, M. Coquerelle, W. Goll, R. Manzel, "Irradiation Behavior of MOX fuel: Results of an EPMA Investigation," Nuclear Engineering and Design, 131 (1991) 1-16.
11. T. Fujino, N. Sato, T. Yamashita, K. Ouchi, "Calculation of Oxygen Potential Change of Irradiated UO_2 and UO_2-PuO_2 Mixed Oxide Fuels Using the Intra-cation Complex Model," Journal of Nuclear Materials, 201 (1993) 70-80.
12. M. Ishida, Y. Korei, "Modeling and Parametric Studies of the Effect of Pu-Mixing Heterogeneity on Fission Gas Release from Mixed Oxide Fuels of LWRs and FBRs," Journal of Nuclear Materials 210 (1994) 203-215.

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

May 17, 1999

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

Dear Chairman Jackson:

SUBJECT: PROPOSED FINAL RULE - REVISIONS TO 10 CFR PARTS 50 AND 72 CONCERNING CHANGES, TESTS, AND EXPERIMENTS

During the 462nd meeting of the Advisory Committee on Reactor Safeguards, May 5-8,1999, we met with representatives of the NRC staff and the Nuclear Energy Institute (NEI) to discuss the proposed final revisions to 10 CFR 50.59 and related requirements in 10 CFR Parts 50 and 72 concerning changes, tests, and experiments. We previously met with the staff and NEI in March 1999 to discuss SECY-99-054 and issued a report to the Commission on March 22, 1999. We also had the benefit of the documents referenced.

CONCLUSIONS AND RECOMMENDATIONS

1. We recommend issuance of the proposed final rule and conforming changes subject to resolution of our comments and concerns.

2. We recommend that criterion (vii) be modified to state "result in a fission product barrier being altered by a change in its design basis limit or a likely reduction in the margin between the design basis limit and the failure point."

3. We recommend that the "substantial review" criterion regarding escalated enforcement be deleted from the proposed final rule.

4. We are concerned that the current wording in criterion (viii) could result in a "zero increase" constraint for departure from a method of evaluation. We recommend that the rule language be changed to "a minimal departure from a method of evaluation."

DISCUSSION

The staff and industry are continuing discussions to simplify, clarify, and restore stability to the 10 CFR 50.59 process and its implementation. Progress is being made on resolving issues identified in our March 22, 1999 report.

Significant changes to 10 CFR 50.59 proposed by the staff since our meeting in March 1999 are the addition of the following two new criteria:

(vii) result in a design basis limit for a fission product barrier being exceeded or altered;

(viii) result in a departure from a method of evaluation described in the FSAR [Final Safety Analysis Report] (as updated) used in establishing the design bases or in the safety analyses.

The new criterion (vii) requires prior NRC review of any change that would result in a design basis limit related to the fission product barrier being exceeded or altered. We note that the margin provided by the fission product barrier is the margin between its design limit and its failure point. This margin can be reduced not only by a change in the design limit, but also by a change in the failure point. The installation of a hardened vent in a containment is an example of the containment design limit not being changed, but the containment barrier capability being reduced by introducing the ability to open the containment barrier before reaching its failure pressure. It is appropriate for the NRC staff to review such a possible reduction in capability before it is implemented by a licensee. Criterion (vii) should be revised to preclude such actions from being carried out under 10 CFR 50.59. To do this, criterion (vii) should be modified to state "result in a fission product barrier being altered by a change in its design basis limit or a likely reduction in the margin between the design basis limit and the failure point."

The new criterion (viii) requires prior NRC review of any change in a methodology or evaluation method that "results in a departure from a method of evaluation described in the FSAR (as updated) used in establishing the design bases or in the safety analyses." We agree with the staff that it is important to clearly define what is a method of evaluation and what are input parameters to the methods to ensure consistent implementation of new criterion (viii). To avoid introducing a "zero increase" constraint, this criterion should be revised to state "a minimal departure from a method of evaluation." It is important that the staff and the industry work closely to develop guidance on the specific elements and examples of the evaluation methods that would require prior NRC review. They should also work closely in developing guidance for input parameters.

In criterion (ii) of the Statement of Considerations, under "Guidance for likelihood of occurrence of malfunction," the staff states that "Changes that would invalidate requirements for redundancy, diversity, separation, and other such design characteristics, would be considered as 'more than a minimal increase in likelihood of malfunction,' and thus would require prior NRC approval." We agree that such changes should require prior NRC approval. We disagree that such changes are automatically more than a minimal increase in likelihood of malfunction. We are concerned about forcing the outcome of what should be a probability determination in order to fit the need for NRC review of design basis commitments. In our February 18, 1999 report, we questioned whether the reference to probability could be deleted from the definition of minimal changes.

In the discussion section on enforcement, the staff states that "a failure to submit an amendment as required would be considered a Severity Level III violation if either a) a substantial review is needed by the NRC before it could conclude that the licensee's actions were acceptable or b) NRC would not have found the licensee's actions acceptable...." We agree with the industry concern that it is unduly subjective to base the decision to issue a Severity Level III violation on whether a "substantial review" was needed to determine that the licensee had performed a proper evaluation. We also agree that the "substantial review" criterion is inherently subjective and that the extent of NRC review needed to verify the adequacy of a licensee's 10 CFR 50.59 safety evaluation is a function of the complexity of the change and the skill of the NRC reviewer.

In criteria (iii) and (iv) of the Statement of Considerations, the staff states "no more than a minimal increase in consequences if the increase is less than or equal to the more limiting of either 10 percent of the difference between the existing calculated value and the regulatory guideline value (10 CFR Part 100 or GDC [General Design Criteria]19 as applicable), or has reached the SRP [Standard Review Plan] guideline value for the particular design basis event." The rationale for the 10 percent incremental value lacks sufficient justification even though both the staff and industry agree to this approach. We believe there is a need to expand the discussion to clearly justify why 10 percent is the appropriate criterion and how the management of incremental changes will ensure that margins are not adversely reduced by frequent use of this criterion. Some ACRS members feel that because the increase in consequences for either an individual change or the cumulative changes is limited by the SRP guideline value, there is sufficient assurance that adequate margins are maintained. Some ACRS members feel that the concern over the particular choice of 10 percent is overwrought.

During our discussions of minimal increases in the likelihood of malfunction, the staff agreed to delete the words "for clarity" from the discussion of "likelihood" as substituted for the term "probability" in criterion (ii) of the Statement of Considerations. The staff also agreed to delete the words "frequency of" from the rule language in 10 CFR 72.48 to make it conform with the proposed rule language in 10 CFR 50.59.

The industry has begun the process of developing changes to the guidance provided in NEI 96-07, "Guidelines for 10 CFR 50.59 Safety Evaluations," and is expected to request NRC endorsement in a regulatory guide. We plan to review the proposed NRC regulatory guide.

Additional comments by ACRS member Graham B. Wallis are presented below.

Sincerely,

Dana A. Powers

Dana A. Powers
Chairman

Additional Comments by ACRS Member Graham B. Wallis

1. I am generally in favor of the objective of restoring 10 CFR 50.59 to the condition where it "worked well" in the past. However, once the revised rule is in place, licensees will adapt to it, so thought needs to be given to what the future might be. Neither the staff nor NEI had much to say about the consequences of implementing the revised rule, something that is surely an important part of the case that must be made for any rulemaking.

2. Likelihood and frequency have replaced probability in the rule. I don't see how this makes any difference. To conform to the criteria on page 118 of the proposed final rulemaking (Reference 1), the licensee has to assess these likelihoods and frequencies. The "qualitative standard" on page 31 appears to be asking for inconsistency in interpretation and I would expect that the regulatory guide will have to provide more specific guidance. A cautious licensee will probably choose to calculate the probabilities of occurrence of an accident or malfunction and evaluate consequences, just as it would now do for use in a PRA. The basic problem of introducing probabilistic language into a deterministic rule has not gone away.

3. "Minimal increase" occurs four times in the criteria on p.118 of Reference 1. For criteria (i) and (ii), there is little guidance on interpretation. The argument that "minimal" subsumes the NEI language of "negligible" does not help. Once "minimal" is in place, licensees will have greater freedom than they asked for with "negligible" in NEI 96-07. This is not hypothetical; in its April 30, 1999 letter (Reference 2), NEI expresses a desire to take advantage of this greater flexibility.

4. What is a "minimal increase" in criterion (ii)? The examples on pages 36-37 do not help because no measure of "likelihood of occurrence" is used. Discussion of items such as "redundant motive force, quality, and other requirements" avoids assessment of likelihood of malfunction, which is not determined by these parameters. The key criterion for evaluation in the rule is still remarkably vague, with no indication of the scale on which it is to be measured.

   Since minimal is no longer negligible, is it 1 percent or 10 percent of the existing likelihood of malfunction? Is it perhaps 1000 percent if the particular item has very little safety significance? Is it some percentage change or arithmetical value of the resulting change in a more universal measure of importance to safety such as core damage frequency (CDF) or large, early release frequency (LERF)? In the absence of a definition for minimal within the context of criterion (ii), one might turn to the discussion of criteria (iii) and (iv) on pages 37-41 of Reference 1 where minimal is defined as less than 10 percent of the margin between calculated values and acceptance values. It would seem that a similar definition should apply, for want of any other, to criterion (ii). Then, for example, a plant with a low CDF compared to the acceptable CDF might have a good argument for increasing its CDF by 10 percent of that margin, eventually working up to the level of CDF where regulatory action is warranted.

   This is not a hypothetical issue. In its April 30, 1999 letter, NEI proposes criteria for use in defining minimal, one of which is "The effect of the change on frequency of an accident can be calculated and would not cause more than a 10 percent increase in the estimated (pre-change) accident frequency."

5. A succession of 10 percent (or any percent) incremental reductions in margin eventually effectively reduces that margin to zero. Perhaps it should be stated straightforwardly that the purpose of this rule is to allow incremental approach to acceptance values at a

manageable rate. This may be the right policy, but it appears significantly different in philosophy from the idea of minimal change.

6. The rule sets a precedent for progressive reduction of margins by specified increments. Now, margins were originally established because of uncertainties in predictions. One stayed a prudent distance away from limits to avoid (qualitative) probability of exceeding them. Reduction in margin makes sense if uncertainty has been sufficiently reduced, so that approach to the limit does not increase the likelihood of stepping over it. I know of no arguments having been presented to show that this uncertainty has actually been reduced.

7. The rule has an impact, however minimal, on public safety. I realize that there has been ample opportunity for public comment, most of which has come from the nuclear industry, to which the NRC has responded. I suggest that it would help relations with the broader public if, when a rule such as this is finally issued, the Statement of Considerations contained a preamble informing an independent observer of what the rule is designed to accomplish and what the expected consequences are.

References:
1. Memorandum dated April 27, 1999, from David B. Matthews, Office of Nuclear Reactor Regulation, to John T. Larkins, ACRS, Subject: Request for Review and Endorsement of Final Rulemaking to Revise 10 CFR 50.59 and Related Provisions Concerning "Changes, Tests, and Experiments."
2. Letter dated April 30, 1999, from Anthony R. Pietrangelo, Nuclear Energy Institute, to David Matthews, NRC, Subject: Issues Concerning the Pending Revisions to 10 CFR 50.59.
3. Report dated March 22, 1999, from Dana A. Powers, Chairman, ACRS, to Shirley Ann Jackson, Chairman, NRC, Subject: SECY-99-054, "Plans for Final Rule - Revisions to 10 CFR Parts 50, 52, and 72: Requirements Concerning Changes, Tests, and Experiments."
4. Report dated February 18, 1999, from Dana A. Powers, Chairman, ACRS, to Shirley Ann Jackson, Chairman, NRC, Subject: List of Questions to be Addressed for Possible Resolution of Key Issues Associated with the Proposed Revision to 10 CFR 50.59 (Changes Tests and Experiments).
5. Nuclear Energy Institute, NEI-96-07, Revision 0, "Guidelines for 10 CFR 50.59 Safety Evaluations," September 1997.

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

May 19, 1999

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

Dear Chairman Jackson:

SUBJECT: THE ROLE OF DEFENSE IN DEPTH IN A RISK-INFORMED REGULATORY SYSTEM

During the 462nd and 461st meetings of the Advisory Committee on Reactor Safeguards, May 5-8 and April 7-10 1999, we discussed issues identified in the Staff Requirements Memorandum dated March 5, 1999, concerning the appropriate relationship and balance between probabilistic risk assessment (PRA) and defense in depth in the context of risk-informed regulation. We previously discussed this matter with the Commission during our meeting on February 3, 1999.

We are attempting to identify pitfalls that may exist along the path the Commission is taking toward risk-informed regulation so they may be addressed in a timely manner. We have communicated previously on the need for plant-specific safety goals that are practical for licensees to evaluate, the need for risk assessments for all modes of plant operation, and the need for research to support further use of risk information in regulatory activities. Several ACRS members, working with an ACRS Senior Fellow, have produced the attached paper in which two views of defense in depth are discussed along with a preliminary proposal regarding its role. Here, we further discuss the role that defense in depth should have in a risk-informed regulatory scheme.

Our motivation for this report has arisen because of instances in which seemingly arbitrary appeals to defense in depth have been used to avoid making changes in regulations or regulatory practices that seemed appropriate in the light of results of quantitative risk analyses. Certainly, we have seen defense in depth used as a basis for delaying changes in the existing regulatory practices:

- there has been reluctance to develop new, risk-informed limits on leakage from steam generator tubes because these are part of the defense-in-depth barriers,
- the development of extensions of the Regulatory Guide 1.174 process to define criteria for risk-informed revisions to 10 CFR 50.59 has been delayed because of defense in depth issues,

- the development of graded quality assurance measures has been overly conservative because of concerns about the imputed importance of quality assurance to defense in depth, and

- the development of regulatory requirements on software-based digital instrumentation and control systems was delayed because of concerns related to defense in depth.

We are concerned that arbitrary appeals to defense in depth could inhibit the effective use of risk information in the regulatory process. At the same time, we are mindful that risk analyses are not perfect. Defense in depth can be an effective means for compensating for any weaknesses in our ability to understand the risks posed by nuclear power plants.

As discussed in the attached paper, the defense-in-depth approach to safety arose in an earlier time when there was less capability to analyze a nuclear power plant as an integrated system. Subsystems were designed such that the necessity and sufficiency of defense in depth could be determined from experience and through exercising engineering judgment. Defense in depth was a design and operational philosophy that called for multiple layers of protection to prevent and mitigate accidents. Its practical implementation was most often associated with control of initiating event frequencies, redundancy and diversity in key safety functions, multiple physical barriers to fission-product release, and emergency response measures. This philosophy has been invoked primarily to compensate for uncertainty in our knowledge of the progression of accidents at nuclear power plants.

Improved capability to analyze nuclear power plants as integrated systems is leading us to reconsider the role of defense in depth. Defense in depth can still provide needed safety assurance in areas not treated or poorly treated by modern analyses or when results of the analyses are quite uncertain. To avoid conflict between the useful elements of defense in depth and the benefits that can be derived from quantitative risk assessment methods, constraints of necessity and sufficiency must be imposed on the application of defense in depth and these must somehow be related to the uncertainties associated with our ability to assess the risk.

We believe that two different perceptions of defense in depth are prominent. In one view (the "structuralist" view as described in the attached paper), defense in depth is considered to be the application of multiple and redundant measures to identify, prevent, or mitigate accidents to such a degree that the design meets the safety objectives. This is the general view taken by the plant designers. The other view (the "rationalist"), sees the proper role of defense in depth in a risk-informed regulatory scheme as compensation for inadequacies, incompleteness, and omissions of risk analyses. We choose here to refer to the inadequacies, incompleteness, and omissions collectively as uncertainties. Defense-in-depth measures are those that are applied to the design or operation of a plant in order to reduce the uncertainties in the determination of the overall regulatory objectives to acceptable levels. Ideally then, there would be an inverse correlation between the uncertainty in the results of risk assessments and the extent to which defense in depth is applied. For those uncertainties that can be directly evaluated, this inverse correlation between defense in depth and the uncertainty should be manifest in a sophisticated PRA uncertainty analysis.

When defense in depth is applied, a justification is needed that is as quantitative as possible of both the necessity and sufficiency of the defense-in-depth measures. Unless defense-in-depth measures are justified in terms of necessity and sufficiency, the full benefits of risk-informed regulation cannot be realized.

The use of quantitative risk-assessment methods and the proper imposition of defense-in-depth measures would be facilitated considerably by the availability of risk-acceptance criteria applicable at a greater level of detail than those we now have. Development of the additional risk-acceptance criteria would have to take into consideration safety objectives embodied in the existing regulations. For example, risk-acceptance criteria are needed to meet the Commission's safety objectives with respect to worker health and environmental contamination and to meet additional public health and safety objectives [e.g., total fatalities, land interdiction]. All of these may not be currently reflected in conventional risk assessments.

We believe that a key missing ingredient needed to place quantitative limits on defense-in-depth measures is acceptance values on the level of uncertainty for each safety objective. Setting such acceptance values is a policy role, very much like setting safety goal values. The uncertainties that are intended to be compensated for by defense in depth include all uncertainties (epistemic and aleatory). Not all of these are directly assessed in a normal PRA uncertainty analysis. Therefore, when acceptance values are placed on uncertainty, these would have to appropriately incorporate consideration of the additional uncertainties not subject to direct quantification by the PRA. These considerations would have to be determined by judgment and expert opinion. As a practical matter, we suggest that the acceptance values be placed on only those epistemic uncertainties quantifiable by the PRA but that these be set sufficiently low to accommodate the unquantified aleatory uncertainties.

When acceptance values have been chosen as policy for the regulatory objectives and their associated uncertainties, it would be possible to develop objective limits on the amount of defense in depth required for those design and operational elements that are subject to evaluation by PRA. To do this, it is necessary to incorporate the effects of the defense-in-depth measures into the PRA uncertainty analysis and the designer or regulator must be able to adjust the defense in depth until the acceptance levels for the regulatory objectives and the acceptance values for the associated uncertainties have both been achieved.

The balance between core damage frequency (CDF) and conditional containment failure probability (CCFP) can serve as an example of this defense-in-depth concept. We have previously recommended that CDF be elevated to a fundamental safety goal. Let us suppose, for example sake, that our acceptance value on this is 10^{-4} per reactor year. If that is the value actually achieved by the design, then a CCFP of about 0.5 has been shown (NUREG-1150) to be generally sufficient to meet the safety goal regulatory objective of individual risk of prompt fatality [which can be adequately represented by an acceptance value of 10^{-5} per reactor year on large, early release frequency (LERF) as noted in Regulatory Guide 1.174]. Does this CCFP provide <u>sufficient</u> defense in depth?

In our view, three acceptance criteria must be satisfied -- one each on CDF, LERF, and the epistemic uncertainty associated with LERF. The Safety Goal Policy Statement suggests candidate acceptance values on CDF and LERF. In addition to these, we must establish the acceptance value on the uncertainty associated with LERF. For the particular value of LERF achieved, let's say that the acceptance value has been set by policy to be on the epistemic uncertainty that can be directly developed from the PRA [but which properly reflects the unquantified aleatory uncertainties]. Now suppose our PRA uncertainty analysis tells us that the quantified uncertainty for this design is greater than the acceptance value. Employing our concept, the design with the 0.5 CCFP does not have sufficient defense in depth. The design must, then, include provisions for more defense in depth [e.g., a better containment perhaps] or reduction of the LERF to values for which the achieved uncertainty is acceptable. The acceptance value on uncertainty for any given regulatory objective could be a function of the absolute value achieved for the regulatory objective. That is, as the achieved mean value for LERF gets further below the acceptance value, the acceptable level of uncertainty on its determination can be greater.

We believe this concept of defense in depth can provide a rational way to develop sufficiency limits wherever the defense-in-depth measures can be directly evaluated by PRA. We acknowledge however, that considerable judgment will have to be exercised to set limits on uncertainty, especially uncertainties not quantified by the PRA. Our preceding example suggests one approach to managing these uncertainties.

For those regulatory functions that are not well suited for PRA or where the current capabilities of PRAs are not sufficient, we suggest that the limits on application of defense in depth be placed at levels lower than the top-level safety objectives (see Figure 1 of attached paper). We emphasize that, even under these circumstances, the PRA can still dictate when defense in depth is needed. Let us illustrate how we envision defense in depth to be applied under these circumstances with an example. Fire is one of the initiating events of interest. PRAs quantify the occurrence of fires in nuclear power plants and, among other things, their impact on control and power cables. The plant response to the loss of the relevant systems (due to the loss of these cables) is also analyzed.

The frequency of fires in specific critical locations, that is, locations in which cables of redundant systems may be damaged, is estimated in the PRA using experience-based rates of occurrence of fires, multiplied by subjective estimates of the fraction of fires that are large enough to have the potential to cause damage and the fraction of those fires that occur in the specified critical locations. This is a highly subjective part of the risk assessment (therefore, highly uncertain). It is, therefore, a suitable area to invoke defense in depth and to impose prescriptive requirements regarding the prevention of fires in those critical locations [e.g., strict administrative controls and periodic inspections]. Thus, the relative inadequacy of the PRA model suggests how defense in depth should be applied at levels lower than the top-level safety objectives.

We further realize that the fire risk assessment does not include the damaging effects of the smoke generated by a fire. This is a case of omission of a potentially significant effect. Therefore, we would, again, resort to defense in depth and may demand barriers to limit the spread of smoke and to protect sensitive equipment.

Since the impact on the risk metrics of these lower-level defense-in-depth measures cannot be quantified, nor can the uncertainties, the necessity and sufficiency of the defense-in-depth measures will have to be simply prescribed and that prescription would constitute the acceptance criteria.

We note that our first example dealing with CDF and CCFP addresses the top level of Figure 1 of the attached paper. If one adopts the structuralist viewpoint at that level, as the paper's preliminary proposal suggests, then the tradeoffs of our example between CDF and CCFP will have to be performed under the assumption that at least some level of defense in depth will be required. If, on the other hand, one adopts the rationalist view even at that level, it is conceivable that the LERF objectives could be satisfied without a containment. Our second example dealing with fires exemplified the rationalist view at lower levels, as the preliminary proposal recommends.

We acknowledge that these preliminary thoughts on the role of defense in depth in a risk-informed regulatory system identify a direction but fall short of closing the issue. We recommend that the Commission give further consideration to this matter.

Sincerely,

Dana A. Powers

Dana A. Powers
Chairman

References:
1. 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.
2. U. S. Nuclear Regulatory Commission, NUREG-1150, Vols. 1-3, "Severe Accident Risks: An Assessment for Five U. S. Nuclear Power Plants," December 1990.
3. Report dated August 15, 1996, from T. S. Kress, Chairman, ACRS, to Shirley A. Jackson, Chairman, NRC, Subject: Risk-Informed, Performance-Based Regulation and Related Matters.
4. Memorandum dated March 5, 1999, from Annette Vietti-Cook, Secretary of the NRC, to John T. Larkins, Executive Director, ACRS, Subject: Staff Requirements - Meeting with the Advisory Committee on Reactor Safeguards, February 3, 1999.

Attachment:
U. S. Nuclear Regulatory Commission, Advisory Committee on Reactor Safeguards, J. N. Sorensen, G. E. Apostolakis, T. S. Kress, D. A. Powers, "On the Role of Defense in Depth in Risk-Informed Regulation," to be presented at PSA 1999, August 22-25, 1999.

ON THE ROLE OF DEFENSE IN DEPTH IN RISK-INFORMED REGULATION

To be presented at PSA '99
Washington, D.C.
August 22-25, 1999

J. N. Sorensen, Senior Fellow
G. E. Apostolakis, Member
T. S. Kress, Member
D. A. Powers, Member
Advisory Committee on Reactor Safeguards
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ABSTRACT

The nascent implementation of risk informed regulation in the United States suggests a need for reexamination of the Nuclear Regulatory Commission's (NRC) defense in depth philosophy and its impact on the design, operation, and regulation of nuclear power plants. This reexamination is motivated by two opposing concerns: (1) that the benefits of risk informed regulation might be diminished by arbitrary appeals to defense in depth, and (2) that the implementation of risk informed regulation could undermine the defense in depth philosophy. From either perspective, two questions are suggested: (1) How is defense in depth defined? (2) How should the implementation of risk informed regulation alter our view of defense in depth? A preliminary proposal for the role of defense in depth in a risk-informed regulatory system is presented.

HISTORICAL DEVELOPMENT

Defense in depth is a nuclear industry safety strategy that began to develop in the 1950s. A review of the history of the term indicates that there is no official or preferred definition. Where the term is used, if a definition is needed, one is created consistent with the intended use of the term. Such definitions are often made by example.

In a 1967 statement[1] submitted to the Joint Committee on Atomic Energy by Clifford Beck, then Deputy Director of Regulation for the Atomic Energy Commission, three basic lines of defense for nuclear power reactor facilities were described. The first line was the prevention of accident initiators through superior quality of design, construction and operation. The second line was engineered safety systems designed to prevent mishaps from escalating into major accidents. The third line was consequence-limiting safety systems designed to confine or minimize the escape of fission products to the environment.

A 1969 paper[2] by an internal study group of the Atomic Energy Commission identified the issue of balance among accident prevention, protection, and mitigation, with the conclusion that the greatest emphasis should be put on prevention, the first line of defense.

A 1994 NRC document[3] identifies the elements of the defense in depth safety strategy as accident prevention, safety systems, containment, accident management, and siting and emergency plans. Other interpretations of defense in depth can be found in INSAG-3[4] and INSAG-10[5]

The historical record indicates an evolution of the term from a narrow application to the multiple barrier concept to an expansive application as an overall safety strategy. The term has increased in scope and gained stature over time. The history also indicates that defense in depth is considered to be a concept, an approach, a principle or a philosophy, as opposed to being a regulatory requirement per se.

Currently the term is commonly used in two different senses. The first is to denote the philosophy of high level lines of defense, such as prevent accident initiators from occurring, terminate accident sequences quickly, and mitigate accidents that are not successfully terminated. The second is to denote the multiple physical barrier approach, most often exemplified by the fuel cladding, primary system, and containment.

One of the essential properties of defense in depth is the concept of successive barriers or levels. This concept applies equally well to multiple physical barriers and to high level lines of defense. A closely related attribute would be requiring a reasonable balance among prevention, protection and mitigation.

EMERGING REGULATORY PRACTICE

The most recent NRC policy statement that deals with defense in depth is the Probabilistic Risk Assessment (PRA) Policy statement[6] published in 1995, which states, in part:

"The use of PRA technology should be increased in all regulatory matters to the extent supported by the state-of-the-art in PRA methods and data and in a manner that complements the NRC's deterministic approach and supports the NRC's traditional defense-in-depth philosophy."

The policy statement, thus, places PRA in a subsidiary role to defense in depth.

In 1998, the NRC published Regulatory Guide 1.174.[7] This guide establishes an approach to risk-informed decision making, acceptable to the NRC staff, which includes the provision that proposed changes to the current licensing basis must be consistent with the defense in depth philosophy. The RG 1.174

discussion states that, "The defense in depth philosophy . . . has been and continues to be an effective way to account for uncertainties in equipment and human performance." The discussion goes on to say that PRA can be used to help determine the appropriate extent of defense in depth, which, by example, is equated to balance among core damage prevention, containment failure prevention and consequence mitigation. The regulatory guide thus addresses the concern of preventing risk-informed regulation from undermining defense in depth. Defense in depth is primary, with PRA available to measure how well it has been achieved.

STRUCTURALIST MODEL

We have identified two different schools of thought (models) on the scope and nature of defense in depth. These models came to be labeled "structuralist" and "rationalist."

The structuralist model asserts that defense in depth is embodied in the structure of the regulations and in the design of the facilities built to comply with those regulations. The requirements for defense in depth are derived by repeated application of the question, "What if this barrier or safety feature fails?" The results of that process are documented in the regulations themselves, specifically in Title 10, Code of Federal Regulations. In this model, the necessary and sufficient conditions are those that can be derived from Title 10. It is also a characteristic of this model that balance must be preserved among the high-level lines of defense, e.g., preventing accident initiators, terminating accident sequences quickly, and mitigating accidents that are not successfully terminated. One result is that certain provisions for safety, for example reactor containment and emergency planning, must be made regardless of our assessment of the probability that they may be required. Accident prevention alone is not relied upon to achieve an adequate level of protection.

There does not appear to be any question that the implementation of defense in depth up to the present time reflects the structuralist model. While this philosophy has served the industry well from the safety perspective, it is now realized that, in some instances, it has led to excessive regulatory burden. Furthermore, the lack of an integrated view of the reactor systems has resulted in some significant accident sequences not being identified until PRA was developed, e.g., the interfacing-systems LOCA sequence.

The next issue, then, becomes how should the insights from PRA be integrated into this structure to reduce unnecessary burden and make it more rational? In the structuralist model, defense in depth is primary, with PRA available to measure how well it has been achieved.

THE RATIONALIST MODEL

The rationalist model asserts that defense in depth is the aggregate of provisions made to compensate for uncertainty and incompleteness in our knowledge of accident initiation and progression. This model is made practical by the development of the ability to quantify risk and estimate uncertainty using probabilistic risk assessment techniques. The process envisioned by the rationalist is: (1) establish quantitative acceptance criteria, such as the quantitative health objectives, core damage frequency and large early release frequency, (2) analyze the system using PRA methods to establish that the acceptance criteria are met, and (3) evaluate the uncertainties in the analysis, especially those due to model incompleteness, and determine what steps should be taken to compensate for those uncertainties. In this model, the purpose of defense in depth is to increase the degree of confidence in the results of the PRA or other analyses supporting the conclusion that adequate safety has been achieved.

The underlying philosophy here is that the probability of accidents must be acceptably low. Provisions made to achieve sufficiently low accident probabilities are defense in depth. It should be noted that defense in depth may be manifested in safety goals and acceptance criteria which are input to the design process. In choosing goals for core damage frequency and conditional containment failure probability, for example, a judgement is made on the balance between prevention and mitigation.

What distinguishes the rationalist model from the structural model is the degree to which it depends on establishing quantitative acceptance criteria, and then carrying formal analyses, including analysis of uncertainties, as far as the analytical methodology permits. The exercise of engineering judgement, to determine the kind and extent of defense in depth measures, occurs after the capabilities of the analyses have been exhausted.

A PRELIMINARY PROPOSAL

The structuralist and rationalist models are not generally in conflict. Both can be construed as a means of dealing with uncertainty. Neither incorporates any reliable means of determining when the degree of defense in depth achieved is sufficient. In the final analysis, they both depend on knowledgeable people discussing the risks and uncertainties and ultimately agreeing on the provisions that must be made in the name of defense in depth. The fundamental difference is that the structural model accepts defense in depth as the fundamental value, while the rationalist model would place defense in depth in a subsidiary role.

The remaining question is which model provides the better basis for moving forward with risk- informed regulation. How can capricious imposition of defense-in-depth be prevented from undermining the focus that can be provided by risk- informed methods of regulation? PRA methods have identified gaps in the regulations and in the safety profiles of individual plants. They have also identified regulations and plant systems that do not make a significant contribution to safety. Typically, however, regulatory reactions to findings that regulations or plant systems are superfluous to safety have been less aggressive than reactions to apparent safety deficiencies.

Two options can be identified:

(1) Recommend defense in depth as a supplement to risk analysis (the rationalist view)

(2) Recommend a high-level structural view and a low-level rationalist view.

Option (1) requires a significant change in the regulatory structure. The place of defense in depth in the regulatory hierarchy would have to change. The PRA policy statement could no longer relegate PRA to a position of supporting defense in depth. Defense in depth would become an element of the overall safety analysis.

Option (2) is to a large degree compatible with the current regulatory structure. The structuralist model of defense in depth would be retained as the high-level safety philosophy, but the rationalist model would be used at lower levels in the safety hierarchy. An example is shown in Figure 1.

The PRA uncertainties increase as we move from the initiating events to risk (from left to right). The structuralist view dictates that intermediate goals be set, such as core damage frequency (CDF), large early release frequency (LERF) or conditional containment failure probability (CCFP),or frequency-consequence (F-C) curves. This would satisfy the requirement of balance between prevention and mitigation. We note that the actual numerical value chosen for core damage frequency can express a preference for prevention, and such a preference is unrelated to defense in depth. One could proceed and set goals at the "cornerstone" level, i.e., one level below. This could include goals on initiating- event frequencies, safety-function or safety-system unavailabilities, and so on. How far down one would go would be a policy issue. The structuralist view would not be applied at lower levels.

The rationalist model would be applied at levels lower than the cornerstones of Figure 1. Defense in depth would be used only to address uncertainties in PRA at the lower levels, thus becoming an element of the overall safety analysis. For events or processes that are not modeled in PRA, defense in depth would play its traditional role. Such is the case with the impact of smoke from fires on plant safety. Current fire risk assessments do not account for the effects of smoke, therefore, prescriptive defense-in-depth based measures would be taken to limit this impact.

We view Option (2) as a pragmatic approach to reconciling defense in depth with risk-informed regulation. There can be little doubt, however, that the rationalist model, Option (1), will ultimately provide the strongest theoretical foundation for risk-informed regulation. When more experience has been gained with the application of PRA in the design and regulation of nuclear power plants, when PRA models can adequately treat most of the phenomena of interest, the role of defense in depth can and should be changed to one of supporting the risk analyses. This transition will need to be supported by the development of subsidiary principles from which necessary and sufficient conditions could be derived.

Note

The views expressed in this paper are the authors' and do not necessarily represent the views of the Advisory Committee on Reactor Safeguards

REFERENCES

1. C. Beck, "Basic Goals of Regulatory Review: Major Considerations Affecting Reactor Licensing," Statement submitted to the Joint Committee on Atomic Energy, Congress of the United States, Hearings on Licensing and Regulation of Nuclear Reactors, April 4,5,6,20, and May 3, 1967.

2. Internal Study Group, "Report to the Atomic Energy Commission on the Reactor Licensing Program," submitted to the Joint Committee on Atomic Energy, Congress of the United States, Hearings on AEC Licensing Procedure and Related Legislation, June 1969.

3. F. E. Haskin, and A. L. Camp,, "Perspectives on Reactor Safety," NUREG/CR-6042, Nuclear Regulatory Commission, Washington, DC, March 1994.

4. International Nuclear Safety Advisory Group, "Basic Safety Principles for Nuclear Power Plants," Safety Series No. 75-INSAG-3, International Atomic Energy Agency, Vienna, Austria, 1988

5. International Nuclear Safety Advisory Group, "Defense in Depth in Nuclear Safety," INSAG-10, International Atomic Energy Agency, Vienna, Austria, 1996

6. U. S. Nuclear Regulatory Commission, "Use of Probabilistic Risk Assessment in Nuclear Regulatory Activities; Final Policy Statement," Federal Register, 60 FR 42622

7. U. S. Nuclear Regulatory Commission, "An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Current Licensing Basis," Regulatory Guide 1.174, June 1998

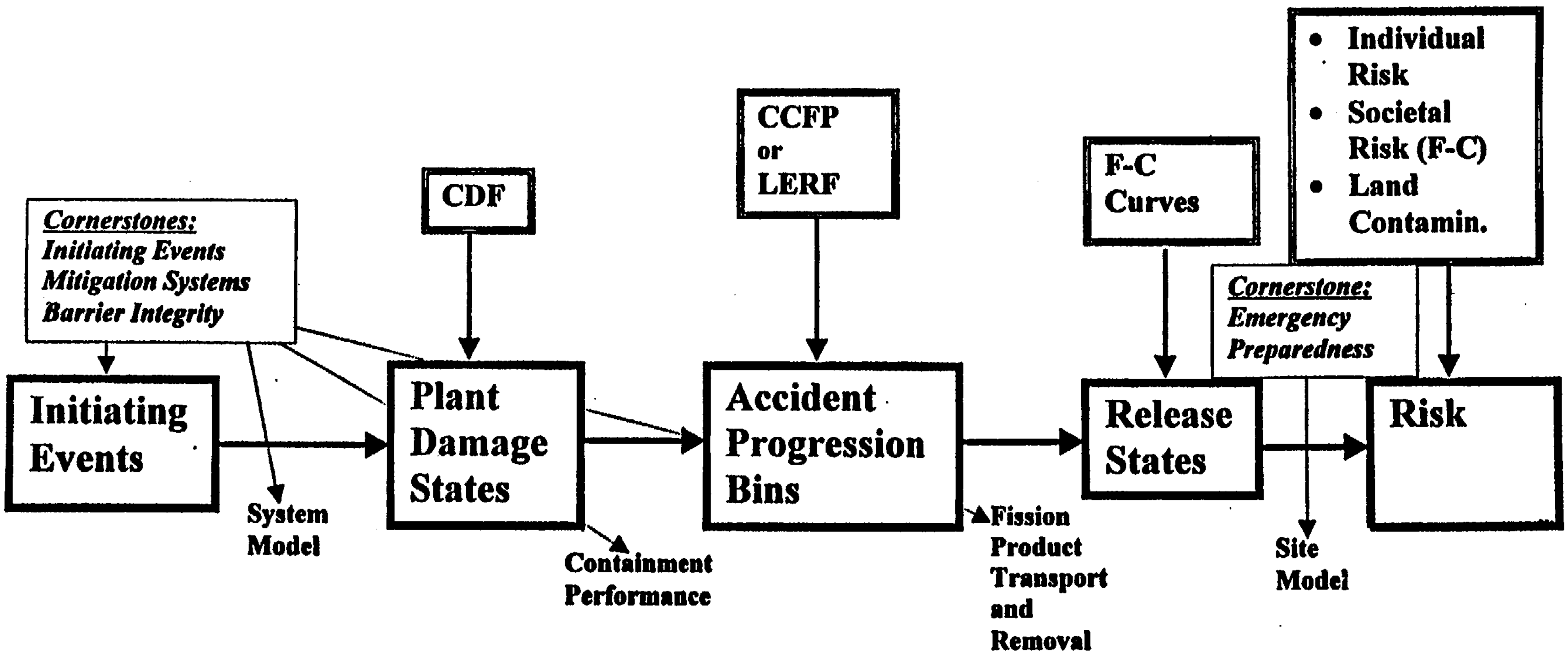

Figure 1. Possible implementation of the structural model at a high level.

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

May 19, 1999

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

Dear Dr. Travers:

SUBJECT: INTERIM LETTER ON THE SAFETY ASPECTS OF THE BALTIMORE GAS AND ELECTRIC COMPANY'S LICENSE RENEWAL APPLICATION FOR CALVERT CLIFFS NUCLEAR POWER PLANT, UNITS 1 AND 2

During the 462nd meeting of the Advisory Committee on Reactor Safeguards, May 5-8, 1999, we reviewed the NRC Staff's Safety Evaluation Report (SER) related to the license renewal application for the Calvert Cliffs Nuclear Power Plant, Units 1 and 2. Our Subcommittee on Plant License Renewal also reviewed this matter on April 28-29, 1999. During our review, we had the benefit of discussions with representatives of the NRC staff and the Baltimore Gas and Electric Company (BGE), and of the documents referenced.

Conclusions and Recommendations

1. The staff performed an extensive and thorough review of the Calvert Cliffs license renewal application. Although there are a number of open issues and confirmatory items that must be resolved, it appears that BGE has developed and implemented adequate processes to identify the structures, systems, and components that are subject to an aging management review and will be able to demonstrate that aging-induced degradation will be adequately managed during the period of extended operation.

2. Current regulatory requirements and existing BGE programs appear to be providing adequate management of aging-induced degradation for those components in the scope of the license renewal rule. BGE identified 446 programs that were needed to manage aging-induced degradation of which only 10 were new programs.

3. Although no new aging mechanisms have been identified, we believe that effective inspections are important to manage aging-induced degradation in order to avoid surprises. It is prudent, for example, to conduct periodic, enhanced visual inspections of reactor internals until data are available to indicate that stress corrosion cracking is not a

plausible degradation mechanism in pressurized water reactors. To date, no cracking has been observed in these components at the Calvert Cliffs units.

4. The issue of thermal aging of cast stainless steels has been resolved for the Calvert Cliffs license renewal application. We believe that the resolution proposed in the application is technically satisfactory and could be used by future applicants.

Discussion

By letter dated April 8, 1998, BGE submitted the license renewal application for Calvert Cliffs in accordance with 10 CFR Part 54, "Requirements for Renewal of Operating Licenses for Nuclear Power Plants." BGE requested renewal of the operating licenses for the Calvert Cliffs units for a period of 20 years beyond the current license expiration dates of July 31, 2014 for Unit 1, and August 13, 2016 for Unit 2.

The SER documents the results of the NRC staff's review of information submitted by BGE through March 5, 1999. The staff's review included the verification of the completeness of the identification and categorization of the structures, systems, and components 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. The staff also conducted on-site inspections to verify the implementation of the programs described in the application. The staff's review of the license renewal application for Calvert Cliffs was extensive and thorough.

Current regulatory requirements and licensee programs appear to be providing adequate management of aging-induced degradation for those components in the scope of the license renewal rule. BGE identified 446 activities that were needed to manage aging-induced degradation. Of these, 329 were existing programs, 107 were modifications or extensions of existing programs or analyses, and only 10 were new programs. BGE has the advantage that, even under the current pressurized thermal shock (PTS) screening criteria, the Calvert Cliffs reactor pressure vessels are not projected to reach the PTS screening limit until after 60 years of operation. BGE also has a robust reactor vessel surveillance program with sufficient surveillance materials for 60 years of operation and, thus, is well prepared to manage vessel embrittlement.

Among the new aging management programs are a number of one-time inspection programs. These are intended to verify the absence of aging-induced degradation that is currently thought unlikely to occur, but cannot be ruled out categorically. The staff stated that in some of these cases one-time inspections are not sufficient to provide assurance that degradation will not develop and that regular, periodic inspections are needed. This is one of the open items to be resolved. Although we have not reviewed the need for the particular inspections still being discussed by the staff and BGE, we believe that effective inspections are important to aging management in order to avoid surprises.

The determination of an aging management program for the embrittlement of cast stainless steels by thermal aging has been identified by both the staff and industry as an open technical issue for license renewal. Although the staff and industry have not yet defined an acceptable generic aging management program for the thermal aging of cast stainless steels, the issue has been resolved for the Calvert Cliffs license renewal application. The staff and BGE have agreed on metal compositions not susceptible to embrittlement. BGE has agreed to conduct inspections of components with metal compositions that could be susceptible to embrittlement. We believe that this resolution is technically satisfactory and could be used by future license renewal applicants.

__Considerations for Future Reviews__

The staff is exploring ways to improve the efficiency of the license renewal application and review processes. The Nuclear Energy Institute has submitted proposals concerning credit for existing programs. Because the review of the BGE application has confirmed that existing programs incorporate most of the aging management activities required for compliance with the license renewal rule, such credit does seem to offer the potential for greater efficiency. The staff is preparing a Commission paper concerning credit for existing programs. We plan to review this paper.

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

Sincerely,

Dana A. Powers

Dana A. Powers
Chairman

__References__

1. Letter dated March 21, 1999, from David B. Matthews, Office of Nuclear Reactor Regulation, to Charles H. Cruse, Baltimore Gas and Electric Company, Subject: Calvert Cliffs Nuclear Power Plant, Units 1 and 2, License Renewal Safety Evaluation Report.
2. Letter dated April 8, 1998, from Charles H. Cruse, Baltimore Gas and Electric Company, to U. S. Nuclear Regulatory Commission Document Control Desk, Subject: Calvert Cliffs Nuclear Power Plant Unit Nos. 1 & 2, Application for License Renewal.

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

May 19, 1999

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

Dear Dr. Travers:

SUBJECT: MODIFICATIONS PROPOSED BY THE WESTINGHOUSE OWNERS GROUP TO THE CORE DAMAGE ASSESSMENT GUIDELINES AND POST ACCIDENT SAMPLING SYSTEM REQUIREMENTS

During the 462nd meeting of the Advisory Committee on Reactor Safeguards, May 5-8, 1999, we reviewed the modifications proposed by the Westinghouse Owners Group (WOG) to the Core Damage Assessment Guidelines (CDAG) and the Post Accident Sampling System (PASS) requirements. Our Subcommittee on Severe Accident Management also reviewed this matter on April 30, 1999. During our review, we had the benefit of discussions with representatives of the NRC staff and WOG, and of the documents referenced.

Background

With the promulgation of the "Three Mile Island-2 Requirements," licensees developed the CDAG for assessing the extent of core damage to help guide offsite radiological protective action decisions. The specifications for the PASS are included in NUREG-0737, "Clarification of TMI Action Plan Requirements," and Regulatory Guide 1.97, Revision 3, "Instrumentation for Light-Water-Cooled Nuclear Power Plants to Assess Plant and Environs Conditions During and Following an Accident."

The specifications for the PASS are based substantially on guidelines developed around 1984 by the WOG for its member licensees. These guidelines relied primarily on sampling for radionuclide analysis and on confirming the results using indirect indicators including containment hydrogen concentration, core exit temperatures, reactor vessel level indication, and containment radiation monitoring. The regulatory requirements of the PASS for Westinghouse pressurized water reactors are to determine:

- from the reactor coolant system (RCS): dissolved gases, hydrogen, oxygen, pH, conductivity, chlorides, boron, and specific radionuclides,

- from the containment atmosphere: hydrogen, oxygen, and specific radionuclides, and
- from the containment sumps: pH, chlorides, boron, and specific radionuclides.

The licensees' experience with the PASS, derived from tests and emergency drills, has been that because of delays in acquiring and analyzing radionuclide samples the relevant information is not provided in a timely manner to guide short-term emergency response decisions. In practice, primary reliance is placed on the use of the indirect indicators to infer particular phases of core damage such as cladding damage, onset of significant hydrogen production, fuel overtemperature, and substantial core damage.

Based on this experience, the WOG has made a proposal outlined in its topical report (WCAP-14986-P) that broadly consists of:

1. Eliminating the PASS sampling requirements except for:
    - RCS boron concentration within 8 hours of obtaining a safe, stable state.
    - Containment hydrogen concentration within 30 minutes of core damage.
    - Containment sump pH only if all three of the following exist:
        - brackish water at the plant for cooling,
        - no passive pH control,
        - a single barrier only between the containment and the heat sink.

2. Retaining the capability to obtain PASS samples for long-term cleanup and recovery planning.

3. Relying primarily on core exit temperatures and containment high-range radiation monitoring as the primary indicators to be applied to the CDAG and using containment hydrogen concentration, reactor vessel level, source monitoring, and hot-leg temperature as secondary, confirmatory information.

<u>Discussion</u>

The WOG proposes to assess core damage based on information obtained from indirect measurements. This information and knowledge derived from calculations of accident progression, hydrogen generation, and fission product release and transport through the RCS and the containment will be used to make the core damage assessment.

We agree with the staff's preliminary review finding that the proposed modifications to the CDAG will provide information on a timely basis to support decisions regarding short-term emergency response.

With regard to the proposed modifications to the PASS requirements, it is our view that the intent of the regulations was to have direct information regarding the disposition of fission products and that this intent could have been easily met by a change to the sample measurements such as the addition of specific gamma monitors at the sampling station. Gamma monitors tuned to krypton and cesium, along with total gamma measurements, are all that is necessary to infer the full source term on a timely, accurate basis. There would be no need for removing the sample and subjecting it to chemical analysis.

In addition, without pH control, materials generated during a severe accident can lower containment sump water pH. Consequently, to assess the potential for fission-product iodine revolatilization from such sumps, we believe that the sump pH should continue to be measured at all plants.

Recommendations

We recommend that the Commission approve the WOG proposals to modify the CDAG and the PASS requirements, but with the qualification that pH measurements in the sump continue to be required.

The staff should revise the regulatory requirements to make clear that the PASS samples are to be used to assist long-term post-accident decisions and recovery actions.

Sincerely,

Dana A. Powers

Dana A. Powers
Chairman

References:

1. Westinghouse Electric Corporation Topical Report, WCAP-14696, "Westinghouse Owners Group Core Damage Assessment Guidance," July 1996.
2. Westinghouse Electric Corporation Topical Report: WCAP-14986-P, Revision 1, "Westinghouse Owners Group Post Accident Sampling System Requirements: A Technical Basis," August 1998 (Proprietary).
3. U. S. Nuclear Regulatory Commission, Office of Nuclear Reactor Regulation slides provided for ACRS Subcommittee meeting on April 30, 1999, "Background and NRR Staff Preliminary Evaluation of WCAP-14696, Westinghouse Owners Group Core Damage Assessment Guidance," April 19, 1999 (Predecisional).
4. U. S. Nuclear Regulatory Commission, Office of Nuclear Reactor Regulation slides provided for ACRS Subcommittee meeting on April 30, 1999, "Background and NRR Staff Preliminary Evaluation of WCAP-14986-P, Westinghouse Owners Group Post Accident Sampling System Requirements, A Technical Basis," April 21, 1999 (Predecisional).

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

June 3, 1999

MEMORANDUM TO: William D. Travers
Executive Director for Operations

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

SUBJECT: DRAFT GUIDE DG-1075, "EMERGENCY PLANNING AND PREPAREDNESS FOR NUCLEAR POWER REACTORS"

During the 462ND meeting of the Advisory Committee on Reactor Safeguards, May 5-8, 1999, the Committee considered the subject Draft Guide and decided not to review it. The Committee would like the opportunity to receive a briefing from the staff on the Nuclear Energy Institute document NEI 99-01 concerning emergency action levels, when it becomes available.

Reference:
Memorandum dated May 21, 1999, from Thomas H. Essig, Office of Nuclear Reactor Regulation, to John T. Larkins, ACRS, Subject: Draft Guide DG-1075, Endorsing Nuclear Energy Institute (NEI) Guidance on Development of Emergency Action Levels.

cc: A. Vietti-Cook, SECY
J. Blaha, OEDO
J. Mitchell, OEDO
S. Collins, NRR
D. Matthews, NRR
T. Essig, NRR
J. O'Brien, NRR

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

June 9, 1999

MEMORANDUM TO: William D. Travers
Executive Director for Operations

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

SUBJECT: EXEMPTION REQUEST TO THE HYDROGEN CONTROL REQUIREMENTS FOR THE SAN ONOFRE NUCLEAR GENERATING STATION, UNITS 2 AND 3

During the 463[RD] meeting of the Advisory Committee on Reactor Safeguards, June 2-4, 1999, the Committee reviewed the request by the Southern California Edison Company for a license exemption to the hydrogen control requirements for the San Onofre Nuclear Generating Station, Units 2 and 3. The Committee has no objection to the staff 's approving this license exemption request, as modified to maintain the requirements for containment hydrogen monitoring capability.

Reference:
Letter dated September 10, 1998, from D. Nunn, Southern California Edison, to U. S. Nuclear Regulatory Commission, Subject: Request for Exemption to 10 CFR 50.44, 10 CFR 50, Appendix A, General Design Criterion 41, and 10 CFR 50, Appendix E, Section VI, Proposed Technical Specification Change NPF-10/15-496, San Onofre Nuclear Generating Station, Units 2 and 3

cc: A. Vietti-Cook, SECY
J. Blaha, OEDO
J. Mitchell, OEDO
S. Collins, NRR
B. Sheron, NRR
G. Holahan, NRR
R. Barrett, NRR
M. Snodderly, NRR

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

June 9, 1999

MEMORANDUM TO: William D. Travers
Executive Director for Operations

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

SUBJECT: PROPOSED FINAL RULE AMENDING THE FITNESS-FOR-DUTY RULE

During the 463RD meeting of the Advisory Committee on Reactor Safeguards, June 2-4, 1999, the Committee considered the subject amendment and decided not to review it. The Committee has no objection to issuing the revised rule.

References:
1. Memorandum dated May 24, 1999, from William D. Travers, Executive Director for Operations, to the Commissioners, Subject: SECY-99-141, "Final Rule Amending the Fitness-For-Duty Rule."
2. Letter dated July 14, 1997, from Robert L. Seale, Chairman, ACRS, to L. Joseph Callan, Executive Director for Operations, NRC, Subject: Proposed Final Revisions to 10 CFR Part 26, Fitness-For-Duty Program Requirements.

cc: A. Vietti-Cook, SECY
J. Blaha, OEDO
J. Mitchell, OEDO
S. Collins, NRR
R. Gallo, NRR
R. Rosano, NRR
R. Albert, NRR

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

June 9, 1999

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

Dear Dr. Travers:

SUBJECT: PROPOSED RESOLUTION OF GENERIC SAFETY ISSUE-165, SPRING-ACTUATED SAFETY AND RELIEF VALVE RELIABILITY

During the 463[rd] meeting of the Advisory Committee on Reactor Safeguards, June 2-4, 1999, we reviewed the proposed resolution of Generic Safety Issue (GSI)-165, "Spring-Actuated Safety and Relief Valve Reliability." During our review, we had the benefit of discussions with representatives of the NRC staff. We also had the benefit of the document referenced.

Recommendation

We agree with the staff's proposal to resolve GSI-165 without any regulatory action.

Background

This Generic Issue was identified after licensees, on a number of occasions, reported that spring-actuated safety and relief valves (SRVs) failed to meet set point criteria within the desired tolerance. At the Shearon Harris plant, failure of an SRV had potentially degraded the high head safety injection system. This failure went undetected for a significant period. The primary concern of this GSI was that failure of SRVs in safety-related support systems could cause a significant diversion of flow from these systems and thus prevent the systems from performing their design function. The scope of GSI-165 was limited to small (< 4 inches) SRVs in safety-related support systems, for which no American Society of Mechanical Engineers (ASME) code requirements for testing existed at the time this concern was raised. GSI-165 was assigned high priority based on the results of a preliminary analysis, which showed that failure of SRVs could raise the core damage frequency (CDF) to a value as high as 5×10^{-2} per reactor year.

Discussion

To resolve this GSI, the NRC staff conducted a study with the technical assistance of the Idaho National Engineering and Environmental Laboratory (INEEL). In this study, piping and instrumentation diagrams (P&IDs) were evaluated along with other plant-specific information provided by licensees for a group of five light-water reactors (LWRs) representative of U.S. LWR designs.

None of these plants were found to contain the type of system cross-tying that contributed to the serious degradation of the high head safety injection system at the Shearon Harris plant. It was determined that many safety-related support systems do not have SRVs, or they have SRVs that cannot produce flow diversion sufficient to cause the failure of their train. Only a single oversized valve in one plant was identified as having the potential for failing its train. The analysis showed an increase in CDF of only $6x10^{-6}$ per reactor year even for this worst-case situation. This CDF is a conservative estimate of risk since the assumed SRV failure rate included all failure modes, most of which do not lead to significant flow diversion of the associated train.

To confirm the generic applicability of these findings to the other operating plants, the NRC staff reviewed the P&IDs of 19 additional plants. In order to review as many diverse configurations as possible, no sister plants were included in this set. This review confirmed the findings of the INEEL study. The number of configurations reviewed appears to be sufficiently large and diverse to justify generic applicability of the conclusions of the INEEL report.

Review of licensee event reports and the nuclear plant reliability data system database did not identify any other instances of valve spring failure besides the one at the Shearon Harris plant. Furthermore, the additional testing requirements originally contemplated as a possible resolution of this GSI were included in the 1986 Edition of the ASME code. That edition was endorsed in the 1992 update of 10 CFR 50.55a, and most plants are already performing this additional testing. This endorsement effectively resolved GSI-165 as early as 1992. As of now, more than 90 percent of all operating plants have included this testing in their inservice testing (IST) programs, and the remaining plants have committed to including this testing in their IST programs by the next refueling outage. We, therefore, agree with the proposed resolution of GSI-165.

Sincerely,

Dana A. Powers

Dana A. Powers
Chairman

Reference:
Memorandum dated April 2, 1999, from John W. Craig, Office of Nuclear Regulatory Research, to John T. Larkins, Executive Director, ACRS, Subject: Review of Generic Safety Issue 165, Spring-Actuated Safety and Relief Valve Reliability.

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

June 10, 1999

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

Dear Dr. Travers:

SUBJECT: PILOT APPLICATION OF THE REVISED INSPECTION AND ASSESSMENT PROGRAMS, RISK-BASED PERFORMANCE INDICATORS, AND PERFORMANCE-BASED REGULATORY INITIATIVES AND RELATED MATTERS

During the 463rd meeting of the Advisory Committee on Reactor Safeguards, June 2-4, 1999, we heard briefings by and held discussions with representatives of the NRC staff regarding the pilot applications of the revised inspection and assessment programs, risk-based performance indicators (PIs), and performance-based regulatory initiatives and related matters. Our Subcommittees on Reliability and Probabilistic Risk Assessment and on Regulatory Policies and Practices also met on April 21, 1999, to discuss performance-based regulatory initiatives. We had the benefit of the documents referenced.

In February 1999, we reviewed proposed revisions to the inspection and assessment programs, including the proposed use of PIs, and provided a report to the Commission dated February 23, 1999. We previously reviewed staff efforts to develop risk-based PIs as Program for Risk-Based Analysis of Reactor Operating Experience of the former Office for Analysis and Evaluation of Operational Data. In April 1998, we reviewed staff plans to increase the use of performance-based approaches in regulatory activities (SECY-98-132) and issued a report dated April 9, 1998.

Recommendations

1. The PI thresholds should be plant- or design-specific.

2. The staff should explain the technical basis for the choice of sampling intervals of PIs used to select a value for comparison with the thresholds.

3. Prior to implementation of the pilot applications of the revised inspection and assessment programs, the pilot applications should be reviewed to make explicit what information will be collected and what hypotheses will be tested.

4. The staff should examine domestic and international studies to determine whether it is possible to develop useful PIs for safety culture.

5. The action levels should be related explicitly to the risk metrics such as core damage frequency (CDF) and large, early release frequency (LERF), where possible.

6. The current performance-based initiatives program should document the lessons learned from current NRC activities in order to focus the diverse NRC activities related to performance-based regulation.

Discussion

A major lesson learned from probabilistic risk assessments (PRAs) is that the risk profile of each plant is unique. The major accident sequences and their contributions to the various risk metrics vary from plant to plant. A consequence of this lesson is that the importance of a PRA parameter, e.g., the unavailability of a system train, with respect to PIs can be assessed only in the context of the integrated risk profile that the PRA provides.

The intent of PIs is to provide objective measures for monitoring and assessing system, facility, and licensee performance. The performance metrics of the chosen set of PIs should assist in making better informed decisions regarding deviations in licensee performance from expectations. This information, combined with the PRA lesson noted above, leads us to the conclusion that the PI thresholds must be plant-specific or design-specific, where practicable. The staff has recognized this in at least one instance, the white-yellow threshold (substantially declining performance) for emergency diesel generator unavailability (SECY-99-007).

In the proposed reactor oversight process, however, most of the thresholds are based on generic industry averages. For example, the 95th percentile of the *plant-to-plant* variability curve for a given parameter, e.g., system unavailability, is defined as the green-white threshold (declining performance). There are two fundamental problems with this approach:

1. Selection of this criterion automatically results in about five plants being above the threshold. This creates an impetus for the licensee to bring the PI below the threshold simply because other plants are doing "better." This may, in effect, create the perception that new regulatory requirements are being imposed on licensees. We do not believe that the oversight process should ratchet expectations for plants which already meet the requirements for adequate protection. We note that this potential for ratcheting, whether actual or perceived, deviates from the intent of identifying declining plant performance.

2. Establishing generic thresholds would not account for plant-specific features that may compensate for the risk impact of any particular parameter. For example, setting the threshold for the unavailability of a system on a generic basis without looking at each plant to understand why a particular value is achieved is contrary to the PRA lesson mentioned above.

The staff has acknowledged that there are both epistemic and aleatory uncertainties in the PIs and that the threshold values must account for both. It is not clear how the staff intends to account for these uncertainties. How does the aleatory variability in an unavailability enter into an assessment? What is the sample that is used to calculate this unavailability? Is it calculated every month? Is the average value computed over a year? How does the sampling method affect the establishment of threshold values? We believe that the staff should prepare technical bases for these choices and develop alternative sampling methods to be tested in the pilot applications of the revised inspection and assessment programs.

This latter observation leads us to the issue of designing pilot applications. We would like to see a well-defined set of questions to be answered and hypotheses to be tested before the pilot applications of the revised inspection and assessment programs are implemented. For example, we would like to see in the pilot applications a staff evaluation of the administrative burden placed on inspectors. Although we agree that the proposed revisions to the assessment program are intended to enhance safety decisions and allocation of inspection resources, we are concerned that the proposed changes may adversely affect in-plant inspection time.

The staff has told us that it does not plan to develop PIs for the "cross-cutting" issue of safety conscious work environment (safety culture). The principal reason stated by the staff is that "if a licensee had a poor safety conscious work environment, problems and events would continue to occur at that facility to the point where either they would result in exceeding thresholds for various performance indicators, or they would be surfaced during NRC baseline inspection activities, or both." We believe that more justification is required for this argument. Safety culture has been recognized as an important determinant of good plant performance. For example, the International Atomic Energy Agency has developed an inspection manual that includes indicators of safety culture. Also, the Swedish Nuclear Power Inspectorate recently published a report describing a systematic procedure using elicitation of expert judgment to produce PIs for safety culture.

The values of the PIs that trigger regulatory action seem to be only qualitatively related to risk metrics (CDF and LERF). We believe that action levels should have a more quantitative relationship to risk metrics consistent with the guidelines in Regulatory Guide 1.174.

The NRC has several activities in the area of performance-based regulation that are either completed or ongoing. We believe that it would be useful to collect the lessons learned from these activities and develop a set of principles and recommendations for future programs. The staff should document these results. This should be the objective of the current program on performance-based approaches to regulation.

We commend the staff for its progress on these challenging matters.

Sincerely,

Dana A. Powers

Dana A. Powers
Chairman

References:

1. Memorandum dated March 22, 1999, SECY-99-007A, from William D. Travers, Executive Director for Operations, NRC, for the Commissioners, Subject: Recommendations for Reactor Oversight Process Improvements (Follow-up to SECY-99-007).
2. Memorandum dated January 8, 1999, SECY-99-007, from William D. Travers, Executive Director for Operations, NRC, for the Commissioners, Subject: Recommendations for Reactor Oversight Process Improvements.
3. Memorandum dated April 16, 1999, from Annette Vietti-Cook, Secretary of the Commission, to William D. Travers, Executive Director for Operations, NRC, Subject: Staff Requirements - SECY-99-086 - Recommendations Regarding the Senior Management Meeting Process and Ongoing Improvements to Existing Licensee Performance Assessment Processes.
4. 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.
5. Draft paper entitled, "Development of Risk-Based Performance Indicators," by Patrick W. Baranowsky, Steven E. Mays, and Thomas R. Wolf, NRC, received May 26, 1999 (Predecisional).
6. Draft memorandum, from William D. Travers, Executive Director for Operations, NRC, for the Commissioners, Subject: Plans for Pursuing Performance-Based Initiatives, received May 12, 1999 (Predecisional).
7. Memorandum dated February 11, 1999, from Annette L. Vietti-Cook, Secretary of the Commission, to William D. Travers, Executive Director for Operations, NRC, Subject: Staff Requirements - SECY-98-132 - Plans to Increase Performance-Based Approaches in Regulatory Activities.
8. Report dated April 9, 1998, from R. L. Seale, Chairman, ACRS, to L. Joseph Callan, Executive Director for Operations, NRC, Subject: Plans to Increase Performance-Based Approaches in Regulatory Activities.
9. 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.
10. International Atomic Energy Agency, IAEA-TECDOC-743, "ASCOT Guidelines, Guidelines for organizational assessment of safety culture and for reviews by the Assessment of Safety Culture in Organizations Team," March 1994.
11. Swedish Nuclear Power Inspectorate, SKI Report 99:19, "Research Project Implementation of a Risk-Based Performance Monitoring System for Nuclear Power Plants: Phase II - Type-D Indicators," February 1999.

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

June 11, 1999

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

Dear Chairman Jackson:

SUBJECT: DEVELOPMENT OF A LOW-POWER AND SHUTDOWN RISK ASSESSMENT PROGRAM

During the 463rd meeting of the Advisory Committee on Reactor Safeguards, June 2-4, 1999, we met with the staff to discuss its plans for developing a low-power and shutdown (LPSD) risk assessment program.

In a report dated April 18, 1997, we stated that it was essential to establish a more complete understanding of the full spectrum of risk if the Commission's efforts to adopt risk-informed, performance-based regulation were to be successful. This more complete understanding is now becoming urgent as pivotal decisions are being made on the implementation of risk-informed, performance-based regulation. LPSD operations are not included in most current probabilistic risk assessments (PRAs). Even when they are, the PRA methods are less mature than those for full power operations. We note that risk during LPSD operations has been estimated to be comparable to that of full power operations.

There are two distinct types of applications for LPSD risk assessments:

(1) risk management of outages, and

(2) risk-informing regulations and decisionmaking.

The risk management of outages focuses on specific outage configurations and the related current risk status. We believe that the LPSD risk assessment methodologies developed and used by the licensees are valuable tools for risk management during outages, and we are encouraged to see the increased use of such methodologies.

The needs for PRA development for supporting risk informing regulations are different and more difficult to satisfy than those for outage management. The LPSD risk assessment must determine the contribution to a plant's risk that results from all of its future shutdowns. Over a plant's lifetime of shutdowns, there may be hundreds of different plant configurations existing for short times during different modes of operation. Each of these configurations is sufficiently different to require a separate analysis, including configuration-specific initiating events and operator actions. The configurations in such future shutdowns cannot be known *a priori*, yet, their simulations in the PRA will be necessary. In essence, it appears that shutdown risk assessments will have to rely on representations of likely future shutdown configurations. It will be necessary to adapt PRA methodology to address the unique character of LPSD operations.

To simulate likely future shutdown configurations, we believe that LPSD PRAs will have to be internally capable of selecting the system/component/feature configuration on an industry-wide average time-out-of-service weighted basis. Thus, a substantial new industry-wide database will be needed on unavailability (or altered configuration) frequencies, durations, and correlations.

The development of the capability to make comprehensive, defensible, and quantitative shutdown risk assessments will require significant effort. To improve the PRA methodology, a better understanding of the unique phenomena that can occur during LPSD operations may be required. We recommend that the staff develop a research program along these lines and complete it on an expedited basis.

Sincerely,

Dana A. Powers

Dana A. Powers
Chairman

Reference:
Report dated April 18, 1997, from R. L. Seale, Chairman, ACRS, to Shirley Ann Jackson, Chairman, NRC, Subject: Establishing a Benchmark on Risk During Low-Power and Shutdown Operations.

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

June 11, 1999

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

Dear Dr. Travers:

SUBJECT: PROPOSED OPTIONS FOR USING AVERTED ONSITE COSTS AND VOLUNTARY INITIATIVES IN REGULATORY ANALYSES

During the 463rd meeting of the Advisory Committee on Reactor Safeguards, June 2-4, 1999, we reviewed the staff's proposed options for using averted onsite costs (AOSCs) and voluntary initiatives in regulatory analyses. During our review, we had the benefit of discussions with representatives of the NRC staff and the Nuclear Energy Institute, and of the documents referenced.

Background

The current NRC policy is to include AOSCs in regulatory analyses. In addition, sensitivity analyses are performed and results computed without including these costs. The staff proposes that there be no change to this current policy.

The current policy on the treatment of voluntary initiatives in regulatory analyses is that, for the "baseline" case calculation, no credit is to be given for voluntary initiatives. The guidelines specify that, for the purpose of sensitivity analyses, the costs and benefits should also be displayed with "full credit" for voluntary initiatives and that this information can be factored into the decision concerning the proposed regulatory action. In practice, however, no credit is given for voluntary initiatives in the regulatory analyses. Thus, the intent of the policy seems not to have been met in the implementation. Consequently, the staff is contemplating three options to the current policy. Our understanding of these options is as follows:

Option A: In addition to the "no credit" and "full credit" calculations, a "best estimate" calculation is performed based on specific guidance given to the analyst on the factors to be considered in assessing the extent to which the voluntary initiatives should be credited.

Option B: A preliminary screening is performed to see if the results of the "no credit" and "full credit" calculations lead to different decisions. If not, there would be no need to proceed further. Otherwise, proceed as in Option A.

Option C: A "full credit" calculation is performed and the results of the "no credit" calculation are displayed for the purpose of sensitivity analyses. This option would, in essence, give greater weight and importance to voluntary initiatives.

The staff recommends Option B as its preferred choice.

Recommendation and Comments

1. We agree with the staff's position on the treatment of AOSCs and recommend that these costs continue to be included in regulatory analyses.

2. We support the staff's preferred Option B for the treatment of voluntary initiatives in regulatory analyses because it would provide a more realistic estimate of the costs and benefits of a regulatory action. We expect that there will be a transition from giving only "some credit" to giving "full credit" as the Agency moves more toward a risk-informed regulatory system.

Discussion

The staff's reasons for including AOSCs in regulatory analyses are valid. These are societal benefits and all societal costs and benefits should be included in regulatory analyses. The AOSCs constitute economic benefits that are frequently referred to as private or internalized benefits. The inclusion of these benefits in cost-benefit analyses is standard practice recommended to all Federal agencies by the Office of Management and Budget.

The industry has been critical of the inclusion of AOSCs in the NRC regulatory analyses for a number of reasons. The industry has argued that AOSCs:

- constitute benefits that accrue solely to the licensee and should not be considered to be societal benefits of the regulations;
- are not a public health and safety issue and, therefore, their inclusion in regulatory analyses inappropriately involves the NRC in licensee internal management affairs; and
- are covered by insurance and their inclusion constitutes double counting.

We have discussed the staff's responses to these industry concerns and agree with the staff's positions.

Our review of AOSCs was not initiated because of any concern about whether AOSCs should be included in regulatory analyses. We have consistently supported the inclusion of AOSCs in regulatory analyses. We had a concern that AOSCs would be improperly co-mingled with present costs that are certain (probability = 1). Future costs are worth less than present costs of implementation and low probability costs may never be manifested. Our concerns in this respect have been allayed by a review of NUREG/BR-0184, "Regulatory Analysis Technical Evaluation Handbook." The processes outlined in this handbook appropriately include the probabilistic nature of future costs, as well as appropriate methods for discounting to present values. Consequently, we agree with the staff's proposal to continue the current policy on AOSCs.

The inclusion of voluntary initiatives in regulatory analyses is more problematic. Voluntary initiatives are discretionary, cannot be enforced by NRC, and could be eliminated by licensee action even without NRC knowledge. The question of how much credit to give for voluntary initiatives is broader than just the regulatory analysis application. This question arises whenever risk assessments are included in regulatory decisions.

For regulatory analyses, the staff's preferred Option B provides a means of giving graded credit to voluntary initiatives depending on the degree to which there is assurance that the requirements on continuation, scope, and effectiveness are satisfied by the characteristics of the initiative. We support Option B because it provides a realistic evaluation of the costs and benefits of regulatory actions.

Giving full credit for all risk-related issues will become more appropriate as the Agency moves closer to a fully risk-informed regulatory system. We anticipate that the regulatory attitude toward voluntary initiatives will change as the Agency moves away from the current deterministic system and more toward a risk-informed system.

In our previous reports we have noted that, in a risk-informed system, it will be necessary to have risk-acceptance criteria that are applied on a plant-specific basis. Such a regulatory system would focus on the actual risk status of individual plants. Therefore, in this kind of system, the concerns expressed by the staff about the likelihood of discontinuing voluntary actions and the plant-to-plant differences in scope and effectiveness disappear. If a voluntary initiative is discontinued, the risk status of the plant may increase. As long as the plant meets the risk-acceptance criteria, this increase should be acceptable. Similarly, since the focus would be on individual plants, the scope and effectiveness of any voluntary initiatives would be reflected in the plant-specific risk assessment. Thus, in this kind of risk-informed regulatory system, full credit should always be given for voluntary initiatives.

Sincerely,

Dana A. Powers

Dana A. Powers
Chairman

References:
1. Memorandum dated May 21, 1999, from Jack E. Rosenthal, Office of Nuclear Regulatory Research, to John T. Larkins, Executive Director, ACRS, Subject: Information Paper Concerning Treatment of Averted Onsite Costs in Regulatory Analyses.
2. Draft Commission Paper received May 21, 1999, from William D. Travers, Executive Director for Operations, NRC, for the Commissioners, Subject: Treatment of Voluntary Initiatives In Regulatory Analyses.

3. U.S. Nuclear Regulatory Commission, Office of Nuclear Regulatory Research, NUREG/BR-0184, "Regulatory Analysis Technical Evaluation Handbook," Final Report, issued January 1997.
4. Memorandum dated May 27, 1999, from Annette Vietti-Cook, Secretary of the Commission, to William D. Travers, Executive Director for Operations, NRC, Subject: Staff Requirements SECY-99-063 - The Use by Industry of Voluntary Initiatives in the Regulatory Process.
5. Report dated September 30, 1998, from R. L. Seale, Chairman, ACRS, to Shirley Ann Jackson, Chairman, NRC, Subject: Impact of Probabilistic Risk Assessment Results and Insights on the Regulatory System.
6. Report dated May 11, 1998, from R. L. Seale, Chairman, ACRS, to Shirley Ann Jackson, Chairman, NRC, Subject: Elevation of CDF to a Fundamental Safety Goal and Possible Revision of the Commission's Safety Goal Policy Statement.

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

July 19, 1999

The Honorable Greta Joy Dicus
Chairman
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555-0001

Dear Chairman Dicus:

SUBJECT: SECY-99-148, "CREDIT FOR EXISTING PROGRAMS FOR LICENSE RENEWAL"

During the 464th meeting of the Advisory Committee on Reactor Safeguards, July 14-16, 1999, we reviewed the staff's proposed options for crediting existing programs for license renewal that are included in SECY-99-148. Our Subcommittee on Plant License Renewal also reviewed this matter on July 1, 1999. During this review, we had the benefit of discussions with representatives of the staff and the Nuclear Energy Institute (NEI), and of the documents referenced.

Background

The license renewal rule requires a demonstration that the effects of aging will be adequately managed for the period of extended operation. The staff and the initial license renewal applicants (Baltimore Gas and Electric Company and Duke Energy Corporation) have found that most of the aging management programs relied upon for license renewal are existing programs. In a letter dated March 3, 1999, NEI provided its view on the level of demonstration required for existing programs under the license renewal rule. In a memorandum dated March 24, 1999, forwarding the NEI letter to the Commission, the staff stated that:

> The staff currently views Part 54 such that existing programs are not automatically adequate to manage aging effects for license renewal simply because they are part of the current licensing basis.

In SECY-99-148, the staff has proposed the following three options:

Option 1: Do not review the adequacy of existing programs.

Option 2: Amend 10 CFR Part 54 to exclude structures and components subject to existing programs.

Option 3: Focus staff review guidance in the Standard Review Plan on the areas where existing programs should be augmented.

The staff has recommended Option 3 because it provides an effective integrated review of programs being relied upon to manage aging for license renewal. The staff stated that Option 3 would reduce unnecessary burden by focusing the staff's review on augmented programs for license renewal. Option 3 could be implemented within the existing license renewal rule. We understand that Options 1 and 2 would require rule changes.

Recommendation

We endorse Option 3. In order to perform its review of license renewal applications, the staff must have a basis for deciding that existing programs are adequate or that the proposed modifications suffice.

Discussion

The extension of licenses for operating plants is predicated on the effectiveness of aging management programs specific to the various passive, long-lived structures and components in the plant, and on the inspection and test programs, such as those specified in the maintenance rule, specific to active, short-lived structures and components. The initial assessment of the current set of aging management programs, the identification of necessary modifications to existing programs, and the establishment of additional aging management programs are the responsibility of the licensee. Independent assessment of the conclusions of the licensee is the responsibility of the staff. The experience with both pilot applications confirms the importance of these roles. Both applicants prepared excellent documents and the staff was able to perform expeditious reviews and identify necessary improvements to the programs.

Additional documentation is currently being prepared by the various owners groups to guide the treatment of aging issues by future license renewal applicants. The guidance to the applicants in these documents and the review of aging management programs being performed by the staff in the Generic Aging Lessons Learned program create an opportunity to decide which programs may require detailed attention. The staff should still review the aging management programs that pertain to the unique features of individual plants.

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

Sincerely,

Dana A. Powers

Dana A. Powers
Chairman

References
1. Memorandum dated June 3, 1999, from William D. Travers, Executive Director for Operations, NRC, to the Commissioners, SECY-99-148, "Credit for Existing Programs for License Renewal."
2. Memorandum dated March 24, 1999, from William D. Travers, Executive Director for Operations, NRC, to the Commissioners, Subject: Credit for Existing Programs for License Renewal.

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

July 21, 1999

The Honorable Greta Joy Dicus
Chairman
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555-0001

Dear Chairman Dicus:

SUBJECT: PROPOSED REVISION 3 TO REGULATORY GUIDE 1.160 (DG-1082), "ASSESSING AND MANAGING RISK BEFORE MAINTENANCE ACTIVITIES AT NUCLEAR POWER PLANTS"

During the 464th meeting of the Advisory Committee on Reactor Safeguards, July 14-16, 1999, we reviewed the proposed Revision 3 to Regulatory Guide 1.160 (DG-1082), "Assessing and Managing Risk Before Maintenance Activities at Nuclear Power Plants." During our review, we had the benefit of discussions with representatives of the NRC staff and the Nuclear Energy Institute (NEI). We also had the benefit of the documents referenced.

<u>Recommendations and Conclusion</u>

1. Before issuing the proposed Revision 3 to Regulatory Guide 1.160 for public comment, the staff should revise it according to the following suggestions:

    - Section 5, "Assessment Scope," needs to be revised to clarify the meaning of "support systems with inter-system dependencies."

    - An introductory section is needed to clarify that the classification of systems, structures, and components (SSCs) as of high- or low-safety significance depends on the plant configuration and on how the measures of importance are determined.

    - The discussion on probabilistic risk assessment uncertainties should be deleted. Instead, the regulatory guide should state that the expert panel needs to consider the possible impact of these uncertainties on the importance rankings.

2. The staff should defer endorsing Section 11, "Assessment of Risk Resulting from Performance of Maintenance Activities," of NUMARC 93-01 until it has been revised by NEI and is made available for review.

3. The guidance provided by the staff to bound the scope of SSCs to be included in the assessment of maintenance activities is adequate to limit the number of analyses that must be performed.

Discussion

The staff has made revisions to Regulatory Guide 1.160 since our previous discussion and our report dated May 11, 1999. Although some of the revised language has improved this Guide, we believe that further revisions as noted in our recommendations are needed.

During our meeting, we were informed by NEI that a revision to Section 11 of the NUMARC 93-01 document would be forthcoming. Both we and the staff need to review the revised section to determine its acceptability for endorsement by Regulatory Guide 1.160, Revision 3.

Determining the risk significance of the plant configurations that may be encountered during maintenance and the large number of combinations of SSCs that may be out of service could require a large amount of resources. We believe that the four conditions set forth in Section 5 reasonably bound the number of configurations that must be considered. We encourage the staff to provide more guidance for determining the importance of multiple SSCs being out of service during maintenance. Such guidance is available in the literature (Reference 3).

We commend the staff for its efforts to revise the Maintenance Rule to better manage risk during maintenance activities and look forward to the resolution of our comments on the proposed Revision 3 to Regulatory Guide 1.160.

Sincerely,

Dana A. Powers

Dana A. Powers
Chairman

References

1. Memorandum dated June 28, 1999, from Theodore R. Quay, Office of Nuclear Reactor Regulation, NRC, to John T. Larkins, ACRS, Subject: Request for Review of Draft Regulatory Guide DG-1082, "Assessing and Managing Risk Before Maintenance Activities at Nuclear Power Plants."
2. Memorandum dated May 17, 1999, from William D. Travers, Executive Director for Operations, NRC, for the Commissioners, SECY-99-133, "Final Revision to 10 CFR 50.65 to Require Licensees to Perform Assessments Before Performing Maintenance."
3. *Reliability Engineering and System Safety 60* (1998) 213-226, M. C. Cheok, G. W. Parry, & R. R. Sherry, "Use of Importance Measures in Risk-Informed Regulatory Applications."
4. Nuclear Energy Institute, NUMARC 93-01, Revision 2, "Industry Guideline for Monitoring the Effectiveness of Maintenance at Nuclear Power Plants," April 1996.

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

July 21, 1999

The Honorable Greta Joy Dicus
Chairman
U.S. Nuclear Regulatory Commission
Washington, DC 20555

Dear Chairman Dicus:

SUBJECT: PROPOSED FINAL REGULATORY GUIDE 1.181, "CONTENT OF THE UPDATED FINAL SAFETY ANALYSIS REPORT IN ACCORDANCE WITH 10 CFR 50.71(e)"

During the 464th meeting of the Advisory Committee on Reactor Safeguards, July 14-16, 1999, we reviewed the proposed final Regulatory Guide 1.181, which endorses NEI 98-03, Revision 1, "Guidelines for Updating Final Safety Analysis Reports," without exception. During this review, we had the benefit of discussions with representatives of the NRC staff and the Nuclear Energy Institute (NEI). We also had the benefit of the documents referenced.

Recommendation

We recommend approval of proposed final Regulatory Guide 1.181 for use by the industry.

Discussion

As a result of industry experience and other initiatives related to updated Final Safety Analysis Reports (FSARs), the NRC has determined that additional guidance regarding compliance with 10 CFR 50.71(e) is necessary. This regulation requires licensees to periodically update their FSARs. Revisions must be filed either annually or 6 months following each refueling outage, provided that the interval between successive updates does not exceed 24 months.

Although 10 CFR 50.71(e) specifies the type of new information that must be evaluated to determine if the FSAR must be updated, experience has shown that additional guidance is necessary. The staff has worked with NEI and other stakeholders to develop this additional guidance. We believe that the guidance resulting from this joint effort, which is documented in NEI 98-03, Revision 1, is sufficient for licensees to demonstrate compliance with the requirements of 10 CFR 50.71(e).

The Commission is now encouraging an evolution in the regulations that will lead to changes in the assessment of safety system performance and the types of accidents that are considered.

The information in the FSAR (e.g., which describes the facility, presents the design bases and the limits on its operation, and presents a safety analysis of the structures, systems, and components) must reflect these changes. As part of its efforts to develop risk-informed regulation, the staff should anticipate how the safety analysis report will evolve. It may be necessary, for example, to include in the safety analysis report information crucial to the conduct of probabilistic risk assessments.

Sincerely,

Dana A. Powers

Dana A. Powers
Chairman

References:

1. Memorandum dated June 10, 1999, from David B. Matthews, Office of Nuclear Reactor Regulation, to Robert L. Seale, Chairman, ACRS, Subject: Request for Review and Endorsement of Regulatory Guide 1.181, "Content of the Updated Final Safety Analysis Report in Accordance with 10 CFR 50.71(e)."
2. 10 CFR 50.71, Maintenance of Records, Making of Reports.
3. 10 CFR 50.34(e), Final Safety Analysis Report.
4. Nuclear Energy Institute, NEI 98-03, Revision 1, "Guidelines for Updating Final Safety Analysis Reports," June 1999.
5. Memorandum dated February 16, 1999, from Annette Vietti-Cook, Secretary of the Commission, to William D. Travers, Executive Director for Operations, NRC, Subject: Staff Requirements - SECY-99-001 - Proposed Guidance for Updated Final Safety Analysis Reports in Accordance with 10 CFR 50.71(e).
6. Memorandum dated June 30, 1998, from John C. Hoyle, Secretary of the Commission, to L. Joseph Callan, Executive Director for Operations, NRC, Subject: Staff Requirements - SECY-98-087 - Proposed Generic Letter 98-XX: Interim Guidance for Updated Final Safety Analysis Reports in Accordance with 10 CFR 50.71(e).

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

July 21, 1999

MEMORANDUM TO: William D. Travers
Executive Director for Operations

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

SUBJECT: PROPOSED REVISION OF NUREG-0800, "STANDARD REVIEW PLAN FOR THE REVIEW OF SAFETY ANALYSIS REPORTS FOR NUCLEAR POWER PLANTS - - LWR EDITION," CHAPTER 13, "CONDUCT OF OPERATIONS"

During the 464th meeting of the Advisory Committee on Reactor Safeguards, July 14-16, 1999, the Committee considered the proposed revision of NUREG-0800 and decided not to review it.

Reference
Federal Register, June 3, 1999 (Volume 64, Number 106), pages 29922-29931 regarding Proposed Revision to Standard Review Plan (NUREG-0800), Chapter 13, "Conduct of Operations."

cc: A. Vietti-Cook, SECY
J. Blaha, OEDO
J. Mitchell, OEDO
S. Collins, NRR
B. Boger, NRR
R. Gallo, NRR
J. Bongarra, NRR

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

July 21, 1999

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-1086, "CRITERION FOR TRIGGERING A REVIEW UNDER 50.80 FOR NON-OWNER OPERATOR SERVICE COMPANIES"

During the 464th meeting of the Advisory Committee on Reactor Safeguards, July 14-16, 1999, the Committee considered the subject regulatory guide and decided not to review it. The Committee has no objection to issuing this draft regulatory guide for public comment.

Reference:
SECY-99-159, from William D. Travers, Executive Director for Operations, NRC, to the Commissioners, Subject: Response to Staff Requirements Memorandum SECY-97-304, February 5, 1998, "Response to SRM: SECY-97-144, 'Potential Policy Issues Raised by Non-Owner Operators'" attaching Draft Regulatory Guide DG-1086.

cc: A. Vietti-Cook, SECY
J. Blaha, OEDO
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S. Collins, NRR
B. Sheron, NRR
M. Davis, NRR

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

July 22, 1999

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

Dear Dr. Travers:

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

During the 464TH meeting of the Advisory Committee on Reactor Safeguards, July 14-16, 1999, we reviewed the proposed rule to revise Appendix K to 10 CFR Part 50. Our Subcommittee on Thermal-Hydraulic Phenomena reviewed this matter during its May 26, 1999 meeting. 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.

The proposed rule will permit a reduction in the conservatism of the reactor power level assumed for loss-of-coolant accident (LOCA) analysis. Specifically, the staff proposes to relax the requirement that the licensee use 1.02 times licensed power for the Appendix K Emergency Core Cooling System (ECCS) analysis. This rulemaking is in response to efforts of licensees to seek credit in safety analyses for reduction in uncertainties in measurement of reactor power by use of more accurate flow measurement systems. This rule change will avoid a large number of anticipated exemption requests and will reduce regulatory burden. Licensees granted this regulatory relief are likely to pursue small power uprates or cost-saving changes to plant operating parameters, which may have to be approved by the NRC.

Conclusion and Recommendation

- We agree with the intent of the proposed rule.

- The staff should evaluate the possible impact of the proposed rule on parts of the regulations other than Appendix K, such as limits on fuel performance.

Discussion

With this rule, the staff has embraced the principle that because margins have been incorporated into the regulations to account for uncertainties, appropriate reduction in these margins may be made when these uncertainties have been reduced. We support this principle.

In the current case, some simple arguments may suffice to justify relaxation of conservatism. In a more general situation, the connection between conservative assumptions and margins of safety is less obvious. One would have to be specific about the relationship between the allowable technical limits and more direct measures of safety, as well as the metric on which margins below those limits are measured. One would then need to evaluate the effects of assumptions and uncertainties in measurement, information (e.g., physical property data) and analysis of the probability of exceeding specified limits, given that the existence of certain margins was considered in making design decisions, perhaps on the basis of "best estimate" calculations. This is a major task. We expect that the staff will eventually need to develop a process, complete with clear definitions, methods of analysis, calculation procedures, and so on. In other words develop the entire technical structure to turn a good concept into a functioning methodology. As this structure is developed, words such as "conservative," "uncertainty," "risk," "margin," and "safety" should have more quantitative and rigorous interpretations.

We are concerned that the relaxation of the 102-percent power requirement is being considered only in the context of Appendix K. The modification of this requirement has margin implications that are not being addressed in the context of this rule change. Relaxation of the 102-percent power assumption in the ECCS rule will likely result in the same changes in initial condition assumptions in all Chapter 15 accident analyses. As noted above, it will likely result in requests to increase licensed reactor power levels. Although some plants are "LOCA-limited" such that the concern with margin reduction is addressed within the context of the rule, some other plants are "flow-limited." In these plants, this change will reduce existing margins to fuel performance limits under normal operation. Yet, the impact of such margin reduction is not being considered in the context of this rule change.

Sincerely,

Dana A. Powers

Dana A. Powers
Chairman

References:

1. Memorandum dated January 13, 1999, from William D. Travers, Executive Director for Operations, NRC, for the Commissioners, SECY-99-014, Subject: Rulemaking Plan: Revision of Appendix K to Title 10, Part 50, of the Code of Federal Regulations (10 CFR Part 50).
2. Memorandum (undated) from William D. Travers, Executive Director for Operations, NRC, for the Commissioners, Subject: Proposed Rule: Revision of Part 50, Appendix K, "ECCS Evaluation Models," received June 24, 1999.
3. Memorandum dated November 17, 1983, from William J. Dircks, Executive Director for Operations, NRC, for the Commissioners, SECY-83-472, Subject: Emergency Core Cooling System Analysis Methods.
4. Caldon, Inc., Engineering Report- 80P, Topical Report, ER-80P, "Improving Thermal Power Accuracy and Plant Safety While Increasing Operating Power Level Using the LEFM✓ (TM) System," Revision 0, March 1997.

5. Caldon, Inc., Responses to NRC Staff Questions ConcerningTopical Report: Improving Thermal Power Accuracy and Plant Safety While Increasing Operating Power Level Using the LEFM✓ (TM) System as Applied to Comanche Peak, dated September 29, 1998 (Proprietary Version).
6. Letter dated July 7, 1999, from C. R. Hastings, Caldon, Inc., to D. A. Powers, Chairman, ACRS, Subject, Proposed Revisions to 10 CFR Part 50, Appendix K to Allow Minor Power Level Increases

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

July 22, 1999

MEMORANDUM TO: William D. Travers
Executive Director for Operations

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

SUBJECT: LATEST DRAFT OF THE DIFFERING PROFESSIONAL OPINION CONSIDERATION DOCUMENT

During the 464th meeting of the Advisory Committee on Reactor Safeguards, July 14-16, 1999, the Committee considered the latest draft of the Differing Professional Opinion Consideration Document, which includes the staff's resolution of the differing professional opinion issues associated with steam generator tube integrity. In a letter dated October 10, 1997, the Committee commented on an earlier draft of the Differing Professional Opinion Consideration Document, which was subsequently issued for public comment. The Committee decided not to review the latest draft document since the Committee has no additional comments or concerns.

References:

1. Draft Differing Professional Opinion Consideration Document, received July 13, 1999.
2. Letter dated October 10, 1997, from R. L. Seale, Chairman, ACRS, to L. Joseph Callan, Executive Director for Operations, NRC, Subject: Resolution of the Differing Professional Opinion Related to Steam Generator Tube Integrity.

cc: A. Vietti-Cook, SECY
J. Blaha, OEDO
J. Mitchell, OEDO
S. Collins, NRR
J. Strosnider, NRR
W. Bateman, NRR
K. Thomas, NRR

July 23, 1999

The Honorable Greta Joy Dicus
Chairman
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555-0001

Dear Chairman Dicus:

SUBJECT: PROPOSED FINAL AMENDMENT TO 10 CFR 50.55a, "CODES AND STANDARDS"

During the 464th meeting of the Advisory Committee on Reactor Safeguards, July 14-16, 1999, we reviewed the proposed final amendment to 10 CFR 50.55a. Our Subcommittee on Materials and Metallurgy also reviewed this matter on March 25, 1999. During these reviews, we had the benefit of discussions with representatives of the NRC staff and of the documents referenced.

Conclusions and Recommendations

We recommend approval of the proposed final amendment to 10 CFR 50.55a. This amendment will: provide significant improvements in the effectiveness of inservice inspections through the expedited implementation of performance demonstration requirements for inspectors; update other requirements for inservice testing; and incorporate American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code Cases for assessment and temporary repair of Class 3 piping.

Discussion

The staff issued the proposed amendment to 10 CFR 50.55a for public comment in December 1997 and has reconciled the comments it received from 65 separate sources. The proposed final amendment to 10 CFR 50.55a will:

- revise the requirements for the construction, inservice inspections, and inservice testing of nuclear power plant components;
- update 10 CFR 50.55a to endorse the 1995 Edition of the ASME Boiler and Pressure Vessel Code and the 1996 Addenda thereto, with modifications and limitations;
- incorporate by reference for the first time the ASME Code for Operation and Maintenance;

- implement performance demonstrations for ultrasonic examination systems;
- supplement motor-operated valve stroke time testing with programs for demonstrating design-basis capabilities; and
- implement check valve condition monitoring on a voluntary basis.

The proposed final amendment to 10CFR 50.55a includes a number of changes in response to public comments. We believe that the staff has adequately addressed these comments. The proposed rule also contains a number of modifications and limitations on the use of the ASME Code by licensees. We believe that the staff has provided sound arguments for the imposition of these restrictions. The number of restrictions is not excessive. These restrictions do not undermine the intent of the requirement to utilize consensus industry standards.

On April 27, 1999, the staff published a supplement to the proposed amendment to 10 CFR 50.55a requesting public comment on a proposal by the staff to eliminate the current requirement for licensees to update their inservice inspection and inservice testing programs to the latest approved edition of the ASME Code every 120 months. Subsequently, as directed by the Commission, the staff is addressing this proposal in a separate rulemaking. The staff is in the process of resolving public comments on this proposal and we expect to have further discussions on this matter after the staff has reconciled public comments.

Sincerely,

Dana A. Powers

Dana A. Powers
Chairman

References

1. U. S. Nuclear Regulatory Commission, Draft Final Rule, 10 CFR Part 50, RIN 3150-AE26, "Industry Codes and Standards; Amended Requirements," received July 1, 1999.
2. Memorandum dated June 24, 1999, from Annette L. Vietti-Cook, Secretary of the Commission, to NRC Commissioner McGaffigan, Subject: COMEXM-99-001 - Reconsideration of SECY-99-017 (Proposed Amendment to 10 CFR 50.55a).
3. 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 - COMEXM-99-001 - Reconsideration of SECY-99-017 (Proposed Amendment to 10 CFR 50.55a).
4. Letter dated October 30, 1998, from Gus C. Lainas, Office of Nuclear Reactor Regulation, to Robert L. Seale, Chairman, ACRS, Subject: Final Amendment to 10 CFR 50.55a, "Codes and standards."

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

September 10, 1999

MEMORANDUM TO: William D. Travers
Executive Director for Operations

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

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

During the 465th meeting of the Advisory Committee on Reactor Safeguards, September 1-3, 1999, the Committee considered the proposed final Revision 4 to Regulatory Guide 1.101, which endorses NEI 99-01, "Methodology for Development of Emergency Action Levels," and decided not to review it. The Committee congratulates the staff and industry on their efforts over the last decade that have resulted in the issuance of this guidance. The Committee believes that the addition of guidance for developing emergency action levels for shutdown and refueling modes of plant operations is especially valuable.

Reference:
Memorandum dated September 1, 1999, from Thomas H. Essig, NRR, to John Larkins, Executive Director, ACRS, Subject: Draft Guide DG-1075 Endorsing Nuclear Energy Institute (NEI) Guidance on Development of Emergency Action Levels.

cc: A. Vietti-Cook, SECY
J. Blaha, OEDO
J. Mitchell, OEDO
S. Collins, NRR
T. Essig, NRR
J. O'Brien, NRR

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

September 13, 1999

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

Dear Dr. Travers:

SUBJECT: INTERIM LETTER RELATED TO THE LICENSE RENEWAL OF OCONEE NUCLEAR STATION

During the 465th meeting of the Advisory Committee on Reactor Safeguards, September 1-3, 1999, we reviewed the NRC staff's Safety Evaluation Report (SER) Related to the License Renewal of Oconee Nuclear Station, Units 1, 2 and 3. Our Subcommittee on Plant License Renewal also reviewed this matter on June 30 - July 1, 1999. During our reviews, we had the benefit of discussions with representatives of the NRC staff and the Duke Energy Corporation (Duke) and of the documents referenced.

Here we make a number of recommendations that are generic to the license renewal process. These recommendations are listed separately from the conclusions that are specific to the Oconee application.

Conclusions

1. The staff performed an extensive and thorough review of the Oconee license renewal application. Notwithstanding a number of open issues and confirmatory items yet to be resolved, Duke has developed and implemented adequate processes to identify structures, systems, and components (SSCs) at Oconee, Units 1, 2 and 3 that are subject to an aging management review and will be able to demonstrate that aging-induced degradation will be adequately managed during the period of extended operation.

2. We concur with the staff assessment that the Babcock and Wilcox Owners Group's topical report BAW-2251, "Demonstration of the Management of Aging Effects for Reactor Vessel," provides both an acceptable demonstration that aging effects will be adequately managed and an acceptable evaluation of time-limited aging analyses.

Recommendations Generic to License Renewal Process

1. We believe that determination of the design-basis accidents and other accidents that define SSCs within the scope of 10 CFR Part 54 is a generic issue for older plants licensed before NUREG-75/087, "Standard Review Plan for the Review of Safety

Analysis Reports for Nuclear Power Plants" (SRP), was issued in September 1975. Additional guidance needs to be developed for this determination.

2. We agree with the staff and industry that additional research and experience are needed to determine the significance of void swelling as a potential mode of degradation for pressurized water reactor internals. Because of the uncertainties, we believe that a focused inspection program as suggested by the staff is a prudent approach for this aging management issue.

3. One-time inspections for evidence of additional plausible modes of degradation for which there is no current experience will be most useful if performed late in the current licensing period. We agree with this strategy and recommend that the staff develop relevant guidance for future applicants.

4. Although updating the supplement to the Final Safety Analysis Report (FSAR) prior to approving the license renewal application is not required by Part 54, we believe that this should be done and recommend that a requirement for updating the supplement to the FSAR be considered in any future revision to Part 54.

5. Active components such as fuses, which are replaced easily, should not be included in the scope of Part 54.

<u>**Discussion**</u>

On July 6, 1998, Duke submitted the license renewal application for Oconee in accordance with Part 54. Duke requested renewal of the operating licenses for the three Oconee units for a period of 20 years beyond the current license expiration dates of February 6, 2013, for Unit 1; October 6, 2013, for Unit 2; and July 19, 2014, for Unit 3.

The SER documents the results of the staff's review of information submitted to the NRC through May 10, 1999. 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. The staff also conducted onsite inspections to verify the adequacy of the implementation of the programs described in the application. The staff's review of the license renewal application for Oconee was extensive and thorough.

The Oconee license renewal application incorporated by reference several Babcock and Wilcox Owners Group topical reports. We have reviewed the staff's safety evaluation of topical report BAW-2251. The staff's safety evaluation was thorough. We concur with the conclusion that BAW-2251 provides both an adequate demonstration that aging effects will be managed and an acceptable evaluation of time-limited aging analyses.

Duke has a robust reactor vessel surveillance program with surveillance materials sufficient for 60 years of operation and, thus, is well prepared to manage vessel embrittlement. Based on the best current data from the compositions of the limiting welds, Duke projected that Oconee, Units 1, 2 and 3 reactor pressure vessels will reach the pressurized thermal shock (PTS) screening limit after 60 years of operation. Duke has also updated the time-limiting aging analysis for flaw growth for 60 years of operation.

In the process of identifying plant SSCs within the scope of Part 54, Duke recognized, as with other plants licensed prior to the staff's issuance of the SRP, that 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, which are relied upon to perform safety-related functions, are within the scope of Part 54. In order to properly scope these SSCs, Duke considered 58 possible design basis events that included the 20 design basis events from the Oconee FSAR, but determined that only 26 events in total were needed for the purpose of scoping SSCs within Part 54.

Based on the limited number of initiating events considered, the staff identified the scoping process as an open item. This scoping issue is not unique to Oconee and must be addressed for all plants licensed before the issuance of the SRP. Additional guidance for identifying the complete set of events that define SSCs within the scope of Part 54 should be developed as part of revising the draft Standard Review Plan for the Review of License Renewal Applications for Nuclear Power Plants.

Even for plants licensed after the issuance of the SRP, additional guidance is needed to address the issues of adequacy and completeness of the set of SSCs within the scope of Part 54. Risk informing the scope of Part 54 may add risk-significant SSCs that are not identified by the current deterministic process. It may also make the implementation of Part 54 more efficient by removing SSCs that are not risk significant.

To address the concern with void swelling of baffle/former assembly components of the Oconee units, Duke has endorsed the industry position developed in the Electric Power Research Institute (EPRI) Technical Report TR-107521, "Generic License Renewal Technical Issues Summary." It is concluded in this report that void swelling of austenitic stainless steel is insignificant for the license renewal term, especially for plants using the low-leakage fuel loading pattern. It is also stated in this report that given the uncertainties involved in predicting the eventual amount of void swelling for the most affected internal components, it would be prudent for industry to follow or participate in research activities associated with this issue. We agree that additional research is needed.

We believe that it is premature to conclude that void swelling will not be a significant issue during the license renewal period. We agree with the staff position on this issue, that either the applicant needs to provide more convincing justification that void swelling will not be an issue or develop an aging management program perhaps based on focused inspections of some critical components.

As in the case of the Calvert Cliffs Nuclear Power Plant, current regulatory requirements and licensee programs appear to provide adequate management of aging-induced degradation for most components in the scope of Part 54. Duke identified 11 new programs and modified 11 existing programs that are needed for Oconee license renewal. Several of these new or modified programs consist of one-time inspections for possible modes of degradation for which there is no current experience. These inspections are to be performed before completion of the current license term. Such inspections will be most useful if they are done as late in the current licensing period as possible. The staff should develop guidance on this issue for future applicants.

The staff is responsible for verifying that the licensee incorporates commitments made in the license renewal application into the licensing basis. As part of its license renewal application, Duke prepared a proposed supplement to the FSAR that identified changes to the FSAR, including the addition of a new chapter concerning license renewal commitments. Duke may update this supplement prior to NRC approval of the license renewal application. We agree with this approach. Although updating of the FSAR supplement prior to the approval of the license renewal application is not required by Part 54, we believe that this should be done and that a requirement for this should be considered in any future revision of Part 54.

In its review of license renewal issue No. 98-0016, "Aging Management Review of Fuses," the staff considered potential aging mechanisms that may prevent fuses from performing their safety-related fault protection function. The staff agreed with the Nuclear Energy Institute position that fuses should be treated as active components and thus should be excluded from the scope of Part 54. We also agree that fuses should be excluded from the scope of Part 54.

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

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

Sincerely,

Dana A. Powers

Dana A. Powers
Chairman

References:
1. Letter dated July 6, 1998, from M. S. Tuckman, Duke Energy Corporation, to U. S. Nuclear Regulatory Commission Document Control Desk, Subject: Oconee Nuclear Station, Units 1, 2, and 3 - Application for Renewed Operating Licenses.
2. Letter dated June 16, 1999, from David B. Matthews, Office of Nuclear Reactor Regulation, NRC, to William R. McCollum, Jr., Duke Energy Corporation, Subject: Oconee Nuclear Station, Units 1, 2, and 3, License Renewal Safety Evaluation Report.
3. 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.
4. Letter dated April 26, 1999, from Christopher I. Grimes, Office of Nuclear Reactor Regulation, to David J. Firth, The 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. Office of Nuclear Reactor Regulation, Office Letter Transmittal, to All NRR Employees, Subject: NRR Office Letter No. 805, "License Renewal Application Review Process," approved June 19, 1998.
6. Argonne National Laboratory, M. C. Billone, Preliminary Assessment and List of Queries for Task Order No. 13 (JCN J-2076), "Review of Void Swelling of Reactor Internals for License Renewal," received July 23, 1999 (Predecisional).
7. U. S. Nuclear Regulatory Commission, "Standard Review Plan for the Review of License Renewal Applications for Nuclear Power Plants," Working Draft, September 1997.

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

September 13, 1999

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

Dear Dr. Travers:

SUBJECT: PROPOSED FINAL REVISION 3 TO REGULATORY GUIDE 1.105, "SETPOINTS FOR SAFETY-RELATED INSTRUMENTATION"

During the 465th meeting of the Advisory Committee on Reactor Safeguards, September 1-3, 1999, we reviewed the proposed Final Revision 3 to Regulatory Guide 1.105, "Setpoints for Safety-Related Instrumentation." This guide endorses ANSI/ISA S67.04-Part 1-1994 Standard with certain exceptions and clarifications. During our review, we had the benefit of discussions with representatives of the NRC staff and Westinghouse Electric Company. We also had the benefit of the documents referenced.

Conclusions and Recommendations

1. We recommend that the final Revision 3 to Regulatory Guide 1.105 be issued for industry use.
2. We agree that a graded approach to using setpoint methodology is appropriate and consistent with the use of risk-informed regulation. We encourage the development of guidance for such an approach.

Discussion

Operating experience indicates that improper setpoints for safety-related instrumentation may allow plants to operate outside the limiting conditions of operation specified in their Technical Specifications. Setpoint problems arose because of varying setpoint methodologies, a lack of a consistent definition of allowable value in different setpoint calculations, and improper understanding of the relationship of the allowable value to earlier setpoint terminology, procedures, and operability criteria.

To resolve these problems, the Instrument Society of America (ISA), with the participation of NRC, undertook development of a standard in the mid-1970s, and subsequently issued ANSI/ISA Standard S67.04 in 1982. Regulatory Guide 1.105, which endorsed the 1982 version of the Standard is being revised to endorse the 1994 version with some clarifications and exceptions.

The limiting safety system setting (LSSS) establishes the threshold for protective system action to prevent acceptable limits being exceeded during design basis accidents. The LSSS, therefore, ensures that the automatic protective action will correct abnormal situations before safety limits are exceeded. Section 4.3 of the 1994 Standard states that the LSSS may be the trip setpoint, an allowable value, or both. This arrangement allows the utilities and vendors more flexibility in developing their trip setpoint setting methodologies. Although all parties agree that this is not a safety issue, there are strongly held views by some that only the trip setpoint is the appropriate value for the LSSS. For the Standard Technical Specifications, the staff designated the allowable value as the LSSS.

Westinghouse argues that only the trip setpoint is appropriate for the LSSS. It maintains that acceptability for continued operation is always based on the premise that the *as left* condition of the instrument channel must be within the uncertainty calibration tolerance about the nominal trip setpoint. In addition, Westinghouse maintains that the allowable value is defined as an uncontrolled *as found* parameter in contrast to the trip setpoint which is a controlled *as left* parameter. Hence, Westinghouse concludes that the trip setpoint is the appropriate value for the LSSS. The staff does not preclude Westinghouse's conclusion in Revision 3 to Regulatory Guide 1.105.

The Regulatory Guide 1.105, Revision 3 endorses a graded approach to using setpoint methodology but gives little guidance on implementation. The staff should develop specific guidance on the use of the graded approach in all appropriate aspects of setpoint methodology. Regulatory Guide 1.176, which provides guidance on the graded approach to quality assurance, should be applicable to this situation.

Sincerely,

Dana A. Powers

Dana A. Powers
Chairman

References:

1. Memorandum dated August 5, 1999, from Ashok C. Thadani, Office of Nuclear Regulatory Research, NRC, to John T. Larkins, Executive Director, ACRS and Joseph A. Murphy, Committee to Review Generic Requirements, Subject: Revision 3 to Regulatory Guide 1.105, "Setpoints for Safety-Related Instrumentation."
2. The International Society for Measurement and Control, ANSI/ISA-S67.04-Part I-1994, "Setpoints for Nuclear Safety-Related Instrumentation," August 1995.
3. U. S. Nuclear Regulatory Commission, Regulatory Guide 1.176, "An Approach for Plant-Specific, Risk-Informed Decisionmaking: Graded Quality Assurance," August 1998.

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

September 13, 1999

MEMORANDUM TO: William D. Travers
Executive Director for Operations

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

SUBJECT: PROPOSED FINAL NRC GENERIC LETTER 88-18, SUPPLEMENT 1, "GUIDANCE ON MANAGING QUALITY ASSURANCE RECORDS IN ELECTRONIC MEDIA"

During the 465th meeting of the Advisory Committee on Reactor Safeguards, September 1-3, 1999, the Committee considered the subject Generic Letter and decided not to review it. The Committee has no objection to issuing the proposed final Generic Letter.

Reference:
Memorandum dated August 6, 1999, from Roy P. Zimmerman, Office of Nuclear Reactor Regulation, to Joseph A. Murphy, Committee to Review Generic Requirements, Subject: Request for Review and Endorsement of the Proposed Generic Letter Titled, "Guidance on Managing Quality Assurance Records in Electronic Media," as Revised to Reflect Public Comments.

cc: A. Vietti-Cook, SECY
J. Blaha, OEDO
J. Mitchell, OEDO
S. Collins, NRR
B. Boger, NRR
M. Bugg, NRR

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

September 14, 1999

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 8.15, "ACCEPTABLE PROGRAMS FOR RESPIRATORY PROTECTION"

During the 465th meeting of the Advisory Committee on Reactor Safeguards, September 1-3, 1999, the Committee considered the subject Regulatory Guide and decided not to review it. The Committee has no objection to issuing the proposed final Revision 1 to Regulatory Guide 8.15 for industry use.

Reference:
Note dated August 3, 1999, from Alan Roecklein, Office of Nuclear Reactor Regulation, to John T. Larkins, Executive Director, ACRS, Subject: Revision 1 to Regulatory Guide 8.15, "Acceptable Programs for Respiratory Protection."

cc: A. Vietti-Cook, SECY
J. Blaha, EDO
J. Mitchell, EDO
S. Collins, NRR
D. Matthews, NRR
A. Roecklein, NRR

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

September 15, 1999

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

Dear Dr. Travers:

SUBJECT: SAFETY EVALUATION REPORT RELATED TO ELECTRIC POWER RESEARCH INSTITUTE RISK-INFORMED METHODS TO INSERVICE INSPECTION OF PIPING (EPRI TR-112657, REVISION B, JULY 1999)

During the 465th meeting of the Advisory Committee on Reactor Safeguards, September 1-3, 1999, we met with representatives of the NRC staff, Electric Power Research Institute (EPRI), and Nuclear Energy Institute to discuss the staff's Safety Evaluation Report (SER) on the topical report (EPRI TR-112657, Revision B) regarding application of EPRI risk-informed methods to inservice inspection (ISI) of piping. Our Subcommittees on Materials and Metallurgy and on Reliability and Probabilistic Risk Assessment met on May 5, 1999, to discuss this matter. We also had the benefit of the documents referenced.

Conclusions

1. We agree with the staff's conclusion that the methodology described in EPRI TR-112657, Revision B, can be used to develop risk-informed ISI programs that will provide an acceptable alternative to the requirements of paragraphs (a)(3) and (g) of 10 CFR 50.55a and is consistent with the guidance in Regulatory Guides 1.174 (General Guidance) and 1.178 (ISI). The EPRI methodology is also consistent with the requirements of American Society of Mechanical Engineers (ASME) Code Cases N-560 (Class 1 piping systems) and N-578 (Class 1, 2, and 3 piping systems).

2. The EPRI methods will better focus inspections on piping with active degradation mechanisms and relatively high risk significance than the current ASME Section XI ISI programs and will lead to significant reductions in occupational radiation

exposure to personnel and associated inspection costs. In almost all cases, use of EPRI methods will also result in a reduction in risk. In those cases in which some increase in risk could occur, we believe it will be very small and well within the guidelines in Regulatory Guide 1.174..

3. Although the Westinghouse Owners Group (WOG) and EPRI risk-informed ISI methods will result in significant improvements in piping inspection programs, it may be possible to further reduce the number and frequency of inspections in the future with little or no increase in risk. Inspections are prioritized by relative risk ranking regardless of the absolute level of the risk involved which, in most cases, is very small. Consequently, excessive inspection resources may still be expended on systems like PWR primary piping which has no known active modes of degradation. In many cases, it can be shown that PWR primary piping has leak-before-break behavior. In contrast, fewer inspection resources are devoted currently to systems with less relative risk importance but with active modes of degradation such as flow-assisted corrosion or thermal fatigue and a much higher probability of failure.

<u>Discussion</u>

Although piping constitutes a significant portion of the reactor coolant system boundary, because of its robust design and the protection afforded by other engineered safety systems, piping failures generally make relatively small contributions to core damage frequency (CDF) or large, early release frequency (LERF). Therefore, even "perfect" piping ISI programs would lead to only small risk reductions.

Some ACRS members believe that, because of the low risk significance associated with piping failures, the current approach to risk-informed ISI as expressed in the EPRI and WOG methods and the current ASME Code Cases is overly timid and that it would be appropriate to make more drastic changes in ISI programs. The number and frequency of inspections could be further reduced without having a significant impact on risk. Instead of prioritizing ISI in terms of relative risk and frequency of failure, the inspections could be prioritized to reduce the total number of piping failures and forego the attempts to distinguish between piping segments virtually all of which have low risk significance. This could, for example, lead to a reduction of inspection resources expended on systems like PWR primary piping which has no known active modes of degradation. In many cases, it can be shown that PWR primary piping exhibits leak-before-break behavior. On the other hand, fewer inspection resources are devoted to systems with less relative risk importance but with active modes of degradation such as flow-assisted corrosion or thermal fatigue and a much higher (several orders of magnitude) probability of failure.

Other ACRS members believe that it is prudent to retain relative risk significance as an important element in design of the ISI program, and that the EPRI and WOG methods for the development and implementation of risk-informed ISI programs are reasonable.

Continued refinement of risk-informed ISI programs is possible so that, for example, the augmented inspection requirements which are currently excluded from both programs could be included in a single integrated program. We believe, however, that such activities should not impede the timely implementation of programs resulting from the use of EPRI and WOG methods by licensees.

Sincerely,

Dana A. Powers

Dana A. Powers
Chairman

References
1. Memorandum dated August 12, 1999, from William H. Bateman and Richard J. Barrett, Office of Nuclear Reactor Regulation, to John T. Larkins, Executive Director, Advisory Committee on Reactor Safeguards, Subject: Safety Evaluation Report Related to EPRI Risk-Informed Inservice Inspection Evaluation Procedure (EPRI TR-112657, Revision B, July 1999).
2. Electric Power Research Institute, EPRI TR-112657, Revision B, WO3230, Final Report, "Revised Risk-Informed Inservice Inspection Evaluation Procedure," July 1999.
3. Letter dated July 13, 1999, from Jeff Mitman, Electric Power Research Institute, to Mike Markley, Advisory Committee on Reactor Safeguards, Subject: EPRI Risk-Informed In-Service Inspection Procedure Discussion.
4. 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.
5. U.S. Nuclear Regulatory Commission, Regulatory Guide 1.178, "An Approach for Plant-Specific Risk-Informed Decisionmaking Inservice Inspection of Piping," issued for trial use September 1998.

6. American Society of Mechanical Engineers, "Case N-560, Alternative Examination Requirements for Class 1, B-J Piping Welds, Section XI, Division 1," August 9, 1996.
7. American Society of Mechanical Engineers, "Case N-578, Risk Informed Requirements for Class 1, 2, and 3 Piping, Method B, Section XI, Division 1," September 2, 1997.
8. Westinghouse Energy Systems, WCAP-14572, Revision 1, "Westinghouse Owners Group Application of Risk-Informed Methods to Piping Inservice Inspection Topical Report," October 1997.
9. Westinghouse Energy Systems, WCAP-14572, Revision 1, Supplement 1, "Westinghouse Structural Reliability and Risk Assessment (SRRA) Model for Piping Risk-Informed Inservice Inspection," October 1997.

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

September 16, 1999

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

Dear Dr. Travers:

SUBJECT: PROPOSED REVISION 1 TO REGULATORY GUIDE 1.78 (DG-1087), EVALUATING THE HABITABILITY OF A NUCLEAR POWER PLANT CONTROL ROOM DURING A POSTULATED HAZARDOUS CHEMICAL RELEASE

During the 465th meeting of the Advisory Committee on Reactor Safeguards, September 1-3, 1999, we reviewed the draft Regulatory Guide DG-1087, which revises Regulatory Guide (RG) 1.78, "Evaluating the Habitability of a Nuclear Power Plant Control Room During a Postulated Hazardous Chemical Release." Our Subcommittee on Severe Accident Management reviewed this matter during its August 9-10, 1999 meeting. During these meetings, we had the benefit of discussions with representatives of the NRC staff and of the documents referenced.

Recommendations

- The proposed Regulatory Guide should be redrafted to facilitate risk-informed license amendment requests to eliminate technical specification requirements for toxic gas monitoring systems.

- The staff should consider providing performance-based guidance to licensees rather than prescriptive guidance in the proposed Regulatory Guide.

- The staff should document evidence of the validity and the capability of computer codes endorsed in regulatory guides such as the HABIT code endorsed in this proposed Regulatory Guide.

Background

The staff proposes to revise RG 1.78 and to include into the revised Regulatory Guide information contained in RG 1.95, "Protection of Nuclear Power Plant Control Room Operators Against an Accidental Chlorine Release." The staff has undertaken this revision to:

- provide improved, consistent limits on toxic chemical concentrations that require actions to protect control room operators,

- provide clarification of screening criteria to be used by licensees to assess the threat of operator incapacitation posed by toxic chemical releases, and

- recommend an improved dispersion model for evaluating the atmospheric dispersal of toxic materials.

The staff anticipates that the revised Regulatory Guide will:

- reduce plant shutdowns caused by spurious alarms of the toxic gas monitoring system,

- reduce administrative burden on licensees arising from compliance with two similar regulatory guides, and

- reduce in some cases the estimated threat of core damage posed by toxic gas release and operator incapacitation.

Discussion

The contribution to a typical plant core damage frequency that is attributable to toxic gas release and operator incapacitation is quite small (about $4x10^{-7}$/yr). It is evident that this threat should not be a focus of safety efforts by the staff or by licensees that do not have peculiar vulnerabilities to toxic gas releases. Licensees may therefore be expected to use the risk-informed mechanisms described in RG 1.174 to seek license amendments to remove Technical Specification requirements for toxic gas monitoring systems. Indeed, the spirit of risk-informed regulation should motivate the staff to encourage such license amendments by revising the proposed Regulatory Guide in a manner that would facilitate review in the terms provided in RG 1.174.

The proposed Regulatory Guide specifies toxic chemical concentrations that prompt protective actions for the control room operators. The concentration limits are the concentrations "immediately hazardous to life and health" defined by the National Institute for Occupational Safety and Health. Such concentrations ". . . will cause death or immediate or permanent adverse health effects if no protection is afforded within 30 minutes." The staff has assumed operators will be able to don protective apparel within two minutes after concentrations this high are reached in the control room ventilation inlets. We support the limiting concentrations selected by the staff. They provide a consistent basis for evaluating threats posed by the diverse chemical releases that could occur.

The proposed Regulatory Guide recommends the use of the HABIT code by licensees to predict atmospheric dispersal of toxic materials. The HABIT model does have more sophisticated modeling than was available for previous versions of the Regulatory Guide. Superior physics alone does not guarantee validity of a computer code. The staff does not appear to have documented evidence of formal peer review, verification, and validation of the HABIT code. Such evidence and a defensible basis for staff confidence should be in hand before the staff endorses a computer code for regulated activities. We encourage the staff to consider review of the HABIT code following processes developed by the Office of Nuclear Regulatory Research for review of codes such as SCDAP-RELAP.

The proposed Regulatory Guide includes guidance that may be interpreted as being requirements:

> "Breathing apparatus should be provided and be readily accessible throughout the plant..."

> "A control room exit leading directly to the outside of the building should have two low-leakage doors in series."

These examples and others may pressure licensees to undertake activities with costs out of proportion to the risks associated with toxic gas releases. Protection against toxic gas releases appears to be an ideal opportunity for the staff to provide performance-based guidance rather than prescriptive guidance to licensees. The staff should consider revising the proposed Regulatory Guide in a performance-based format.

Sincerely,

Dana A. Powers

Dana A. Powers
Chairman

References:

1. Memorandum dated July 28, 1999, from Charles E. Rossi, Office of Nuclear Regulatory Research, NRC, to John T. Larkins, ACRS, Joseph A. Murphy, Committee to Review Generic Requirements, Stuart A. Treby, Office of the General Counsel, transmitting Draft Regulatory Guide DG-1087 (Proposed Revision 1 to Regulatory Guide 1.78), "Evaluating the Habitability of a Nuclear Power Plant Control Room During a Postulated Hazardous Chemical Release."
2. U. S. Nuclear Regulatory Commission, Regulatory Guide 1.95, Revision 1, "Protection of Nuclear Power Plant Control Room Operators Against an Accidental Chlorine Release," January 1977.
3. 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.
4. ACRS letter dated July 20, 1995, from T. S. Kress, ACRS Chairman, to James M. Taylor, Executive Director for Operations, NRC, Subject: Resolution of Generic Safety Issue 83, "Control Room Habitability."

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

September 17, 1999

The Honorable Greta Joy Dicus
Chairman
U.S. Nuclear Regulatory Commission
Washington, DC 20555-0001

Dear Chairman Dicus:

SUBJECT: PROPOSED FINAL RULE ON USE OF ALTERNATIVE SOURCE TERM AT OPERATING REACTORS, DRAFT REGULATORY GUIDE, AND STANDARD REVIEW PLAN

During the 465th meeting of the Advisory Committee on Reactor Safeguards, September 1-3, 1999, we reviewed the proposed final rule on Use of Alternative Source Term at Operating Reactors, the associated draft Regulatory Guide (DG-1081), and the associated Standard Review Plan Section (SRP) (15.0.1). Our Subcommittee on Severe Accident Management reviewed this matter during its August 9-10, 1999 meeting. During these meetings, we had the benefit of discussions with the NRC staff and of the documents referenced. We previously reviewed a proposed version of the source term rule and provided a report to the Commission dated November 19, 1998.

BACKGROUND

Because of the regulatory significance of source term usage, we have had a long-standing interest in the subject. For example, we previously endorsed the efforts to update and define a more realistic source term for future plants, as described in NUREG-1465, and to require the use of total effective dose equivalent (TEDE) and the "worst" two-hour release period.

Although the revised source term was intended for use by future plant licensees, the current rulemaking effort is aimed at allowing the use of the NUREG-1465 alternative source term by licensees of currently operating plants. In our November 19, 1998 report, we noted that the staff had done a commendable job of addressing the issues associated with allowing licensees of currently operating plants the option to make plant changes based on the NUREG-1465 alternative source term. Also, we supported the use of the alternative source term at operating plants on a selective and voluntary basis. Public comments have been received on the proposed rule, and the staff intends to seek Commission approval both to issue the final version of the rule and to publish DG-1081 and SRP Section 15.0.1 for public comment. Our comments are offered for consideration prior to publication of these documents.

OBSERVATIONS AND RECOMMENDATIONS

1. The staff has done an excellent job in developing a workable rule, regulatory guide, and SRP Section.

2. The staff should modify the proposed redefinition of the source term to eliminate the connotation that the release is necessarily to the containment but should retain the wording ". . . release from the RCS . . . ."

3. The staff should reassess the requirement for evaluating the effects of changes on core damage frequency (CDF) and large, early release frequency (LERF) and determine if this requirement could be relegated to the 10 CFR 50.59 change process.

4. The requirement to have prior NRC approval for "changes . . . that result in a reduction in safety margins" should be reevaluated for removal in light of both the analytical assessments done by the Office of Nuclear Regulatory Research and the results of the pilot applications of the alternative source term.

DISCUSSION

Redefinition of the Source Term

The staff has proposed to change the wording in the definition of the source term from " . . . released from the reactor core to the containment . . ." to " . . . released from the reactor fuel . . . ." The purpose of this proposed change is to avoid the implication that the alternative source term could not be used for the entire range of design basis accidents, including those that bypass containment.

We believe this proposed change would misrepresent the NUREG-1465 basis for the alternative source term in two respects: (1) the chemical forms in the source term become "stabilized" only after some distance of transport downstream from the point of release from the fuel, and (2) the intent of the NUREG-1465 alternative source term was that deposition within the reactor coolant system (RCS) is accounted for and that any implementation should not consider additional attenuation due to passage through the RCS. To avoid any potential misunderstanding, we believe the desired objective could be achieved more appropriately by eliminating the words "to the containment" but retaining the words, " . . . release from the RCS . . . ." We do not support use of the words ". . . release from the fuel."

Risk Issues

The draft Regulatory Guide and the SRP Section call for "identifying whether the application should be considered risk informed," and ". . . ensuring that any associated plant modification that may have an impact on CDF and LERF is reviewed by risk analysts . . . ."

While these are seemingly innocuous statements, we believe that they are not needed. There is ample evidence from both the Office of Nuclear Regulatory Research assessments and the pilot plant results that the risk metrics (CDF and LERF) are sufficiently insensitive to any plant modifications that can result from use of the alternative source term that there appears to be no need to continue to evaluate them for each plant modification. We believe that the staff should consider the approach of viewing such changes in the same light as the 10 CFR 50.59 change process.

Safety Margins

The draft Guide defines safety margins as "the difference between calculated parameters (e.g., postulated offsite or control room dose) and the associated limits . . . ." It goes on to state that ". . . changes, or the net effects of multiple changes, that result in a reduction in safety margins may require prior NRC approval." These statements in the draft Guide are troublesome to us. The changes resulting from adopting the NUREG-1465 alternative source term are likely to result in a reduction of the safety margins as they are defined above. The assessments made by the Office of Nuclear Regulatory Research have demonstrated that these reductions in margins are acceptable. Since there are no regulatory requirements that specify the magnitude of these safety margins and no guidance on how to determine them, there is little need for the stipulation for prior NRC approval. As noted above, it is conceivable that the changes resulting from application of the alternative source term could be considered minimal changes as discussed in 10 CFR 50.59.

The staff has done an excellent job overall. We plan to review the proposed final Regulatory Guide and SRP section following the reconciliation of public comments.

Sincerely,

Dana A. Powers

Dana A. Powers,
Chairman

References:

1. Memorandum dated July 13, 1999, from Gary M. Holahan, Office of Nuclear Reactor Regulation, NRC, to John T. Larkins, ACRS, Subject: Transmittal of the Final Amendments to 10 CFR Parts 21, 50, and 54: Draft Regulatory Guide; Draft Standard Review Plan Section; Regarding Use of an Alternative Source Term at Operating Reactors.
2. Compilation of Public Comments Received on Proposed Rule, "Use of Alternative Source Terms at Operating Reactors":
   - Letter dated May 25, 1999, from David J. Modeen, Nuclear Energy Institute
   - Letter dated May 25, 1999, from Daniel F. Stenger, Counsel to the Nuclear Utility Backfitting and Reform Group
   - Letter dated May 20, 1999, from M. S. Tuckman, Duke Energy
   - Letter dated May 24, 1999, from H. L. Sumner, Jr., Southern Nuclear Operating Company, Inc.
   - Letter dated May 20, 1999, from James M. Levine, APS, Palo Verde Nuclear

Generating Station
- Letter dated June 7, 1999, from Kent Tosch, State of New Jersey
- Letter dated May 10, 1999 from Ralph Cantral, State of Florida

3. ACRS report dated November 19, 1998, from R. L. Seale, Chairman, ACRS, to Shirley Ann Jackson, Chairman, NRC, Subject: Proposed Rule on Use of Alternative Source Term at Operating Reactors.
4. U.S. NRC Report, NUREG-1465, "Accident Source Terms for Light-Water Nuclear Power Plants" - Final Report, L. Soffer, et al., February 1995.

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

September 17, 1999

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

Dear Dr. Travers:

SUBJECT: PROPOSED RESOLUTION OF GENERIC SAFETY ISSUE-145, "ACTIONS TO REDUCE COMMON CAUSE FAILURES"

During the 465th meeting of the Advisory Committee on Reactor Safeguards, September 1-3, 1999, we reviewed the proposed resolution of Generic Safety Issue (GSI)-145, "Actions to Reduce Common Cause Failures." During our review, we had the benefit of discussions with representatives of the NRC staff. We also had the benefit of the documents referenced.

Recommendations

- We recommend that the staff issue an additional Administrative Letter summarizing the major insights derived from the common-cause failure (CCF) research project to make them more readily available to licensee management.

- After issuing the Administrative Letter, we recommend that GSI-145 be closed out without further regulatory action.

Discussion

Common-cause failures of redundant safety systems have been of concern ever since quantitative estimates of the availability and reliability of these systems were developed starting in the early 1970s. CCFs are intended to represent causes of dependent failures that are not modeled explicitly in the probabilistic risk assessment (PRA). The fact that a class of diverse failure causes must be modeled has created unusual challenges for the analyst. The difficulty is compounded by the realization that the operating experience contains a wealth of information on potential CCFs, i.e., partial failures that could have evolved into the complete failure of redundant components within a "small" period of time.

The efforts over the last 25 years to understand CCFs have been successful. The rate of occurrence of complete CCFs has been steadily decreasing (see attached Figure). Both the industry and the NRC staff have been sensitized to the significance of CCFs. A major contributor to this success has been the work sponsored by the former Office for Analysis and

Evaluation of Operational Data and continued by the Office of Nuclear Regulatory Research, to collect and analyze relevant operational experience, as well as disseminating this information.

On July 30, 1998, the staff issued NRC Administrative Letter 98-04 to inform the licensees about the availability of CCF database, CCF analysis software, and associated technical reports. Subsequently, the staff transmitted the multi-volume report NUREG/CR-6268 on CCF through a letter dated July 30, 1998. We are concerned that, although this report will eventually be used by PRA analysts, utility managers who could take specific actions to further reduce the potential for CCFs in the near term are unlikely to read this massive report. We, therefore, believe that before GSI-145 is declared as resolved, an additional Administrative Letter should be issued summarizing the insights from the CCF project in a way that will be useful to plant managers.

We are somewhat concerned that the staff does not plan to determine whether the licensees are implementing any actions based on the insights of NUREG/CR-6268 to reduce the potential for CCFs. However, given the general awareness of the CCF issue that we mentioned earlier, we do not believe that this is a sufficient reason to justify delaying the resolution of GSI-145. The staff should, of course, be vigilant to identify any signs that the downward trend in the CCF rate has reversed.

Sincerely,

Dana A. Powers

Dana A. Powers
Chairman

References :

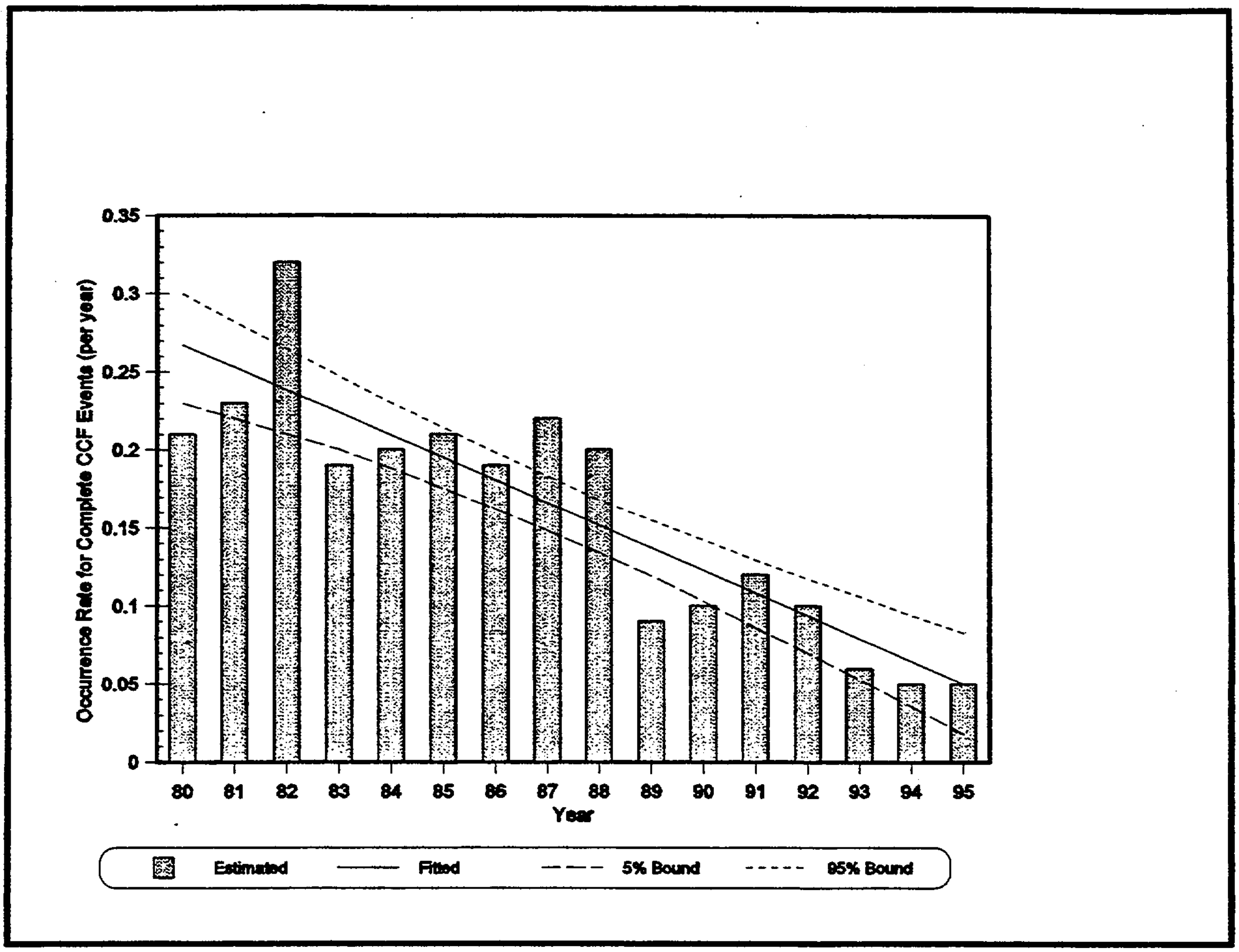
1. Memorandum dated July 30, 1999, from Charles E. Rossi, Office of Nuclear Regulatory Research, NRC, to John T. Larkins, Executive Director, ACRS, Subject: Proposed Resolution of Generic Safety Issue 145, "Actions to Reduce Common Cause Failures."
2. U. S. Nuclear Regulatory Commission Administrative Letter 98-04, "Availability of Common-Cause Failure Database," dated July 30, 1998.
3. Letter dated July 30, 1998, from Charles E. Rossi, Office for Analysis and Evaluation of Operational Data, NRC, to a list of Senior Licensee Officials, Subject: Common Cause Failure Database Distribution.
4. U. S. Nuclear Regulatory Commission, NUREG/CR-6268, June 1998, Vol. 1, "Common-Cause Failure Database and Analysis System: Overview;" Vol. 2, "Common-Cause Failure Database and Analysis System: Event Definition and Classification;" Vol. 3, "Common-Cause Failure Database and Analysis System: Data Collection and Event Coding;" Vol. 4, "Common-Cause Failure Database and Analysis System: Software Reference Manual."
5. U. S. Nuclear Regulatory Commission, NUREG/CR-5485, "Guidelines on Modeling Common-Cause Failures in Probabilistic Risk Assessment," November 1998.
6. U. S. Nuclear Regulatory Commission, NUREG/CR-5497, "Common-Cause Failure Parameter Estimations," October 1998.

7. U. S. Nuclear Regulatory Commission, NUREG/CR-4780, Volumes 1 and 2, "Procedures for Treating Common Cause Failures in Safety and Reliability Studies," January 1988 and January 1989, respectively.

8. U. S. Nuclear Regulatory Commission, NUREG/CR-5460, "A Cause-Defense Approach to the Understanding and Analysis of Common Cause Failures," March 1990.

Attachment: Figure

ATTACHMENT



Yearly occurrence rate for complete CCF events, with 90 percent confidence band on the fitted trend.

FIGURE

(Figure Attached to Memorandum dated July 30,1999, from Charles E.Rossi, Office of Nuclear Regulatory Research, U.S. Nuclear Regulatory Commission, to John T. Larkins, Advisory Committee on Reactor Safeguards.)

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

September 17, 1999

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

Dear Dr. Travers:

SUBJECT: MODIFICATIONS PROPOSED BY THE WESTINGHOUSE OWNERS GROUP TO THE CORE DAMAGE ASSESSMENT GUIDELINES AND POST ACCIDENT SAMPLING SYSTEM (PASS) REQUIREMENTS

During the 464th and 465th meetings of the Advisory Committee on Reactor Safeguards, July 14-16 and September 1-3, 1999, respectively, we discussed your June 22, 1999 response to our May 19, 1999 letter on the subject matter. Your letter included the following comments:

(1) "The staff . . . intends to allow options other than PASS samples (such as the use of specific gamma monitors) to provide information regarding the disposition of fission products."

(2) ". . . the staff concludes that, for plants with passive pH control or that are not subject to contamination of the sump with brackish water, pH measurement is not needed because, in these plants, pH will either be maintained alkaline or could be estimated with a sufficient degree of accuracy."

Because we disagree with both of these positions, we are clarifying our original recommendations.

With respect to Comment (1) above, our view is that the Post Accident Sampling Systems implemented in the Westinghouse plants do not meet the intent of TMI Action Plan Requirement II.B.3, as specified in NUREG-0737, to have direct and timely information regarding "certain radionuclides in the reactor coolant and containment atmosphere that may be indicators of the degree of core damage. . . ." Such intent could be satisfied by the use of specific gamma monitors installed in containment that are tuned to the isotopic gamma emissions of cesium and krypton. If Requirement II.B.3 for timely and radionuclide-specific information is no longer necessary, it should be removed rather than circumvented. If this requirement is retained, then the staff should consider a compliance backfit for the installation of such gamma monitors.

With respect to Comment (2), we disagree with the assertions regarding assurance of maintenance of containment sump alkalinity by passive pH control. The sources of acidic materials during severe accidents are very uncertain and may not have all been identified. In addition, the evaluation of sump alkalinity would be complicated by the need to quantitatively assess complexation, adsorption, and precipitation of buffers by materials introduced into the sump water over the course of an accident. Passive pH control cannot be assessed with sufficient accuracy to assure that an adequate level of alkalinity is maintained over the desired period of time. A direct measurement is needed for appropriate post-accident decisionmaking. Therefore, we repeat our original recommendation that pH measurement continue to be required for all sumps.

Sincerely,

Dana A. Powers

Dana A. Powers
Chairman

References:

1. Letter dated June 22, 1999, from William D. Travers, Executive Director for Operations, NRC, to Dana A. Powers, Chairman, ACRS, Subject: Modifications Proposed by the Westinghouse Owners Group to the Core Damage Assessment Guidance and the Post Accident Sampling System Requirements.
2. Letter dated May 19, 1999, from Dana A. Powers, Chairman, ACRS, to William D. Travers, Executive Director for Operations, NRC, Subject: Modifications Proposed by the Westinghouse Owners Group to the Core Damage Assessment Guidelines and the Post Accident Sampling System Requirements.
3. U. S. Nuclear Regulatory Commission, NUREG-0737, "Clarification of TMI Action Plan Requirements," dated November 30, 1980.

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

October 5, 1999

MEMORANDUM TO: William D. Travers
Executive Director for Operations

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

SUBJECT: PROPOSED FINAL REVISION 3 TO REGULATORY GUIDE 1.8, "QUALIFICATION AND TRAINING OF PERSONNEL FOR NUCLEAR POWER PLANTS"

During the 466th meeting of the Advisory Committee on Reactor Safeguards, September 30 - October 2, 1999, the Committee considered the proposed final Revision 3 to Regulatory Guide 1.8 and decided not to review it. The Committee has no objection to issuing this Guide for industry use.

Reference:
Proposed Revision 3 to Regulatory Guide 1.8, "Qualification and Training of Personnel for Nuclear Power Plants," September 1999.

cc: A. Vietti-Cook, SECY
J. Blaha, OEDO
G. Tracy, OEDO
A. Thadani, RES
J. Rosenthal, RES
I. Schoenfeld, RES

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

October 8, 1999

The Honorable Greta Joy Dicus
Chairman
U.S. Nuclear Regulatory Commission
Washington, DC 20555-0001

Dear Chairman Dicus:

SUBJECT: DRAFT COMMISSION PAPER REGARDING PROPOSED GUIDELINES FOR APPLYING RISK-INFORMED DECISIONMAKING IN LICENSE AMENDMENT REVIEWS

During the 465th and 466th meetings of the Advisory Committee on Reactor Safeguards, September 1–3, 1999, and September 30–October 2, 1999, respectively, we reviewed the draft Commission paper, "Proposed Guidelines for Applying Risk-Informed Decisionmaking in License Amendment Reviews." During our review, we had the benefit of discussions with representatives of the NRC staff and of the documents referenced.

Background

The staff recently reviewed the Union Electric Company submittal for a license amendment for the Callaway Plant, Unit 1, that would allow the use of the new Electrosleeve process to repair steam generator tubes. The license amendment request conformed to the existing license amendment process, but was not a risk-informed submittal in accordance with Regulatory Guide 1.174. The staff determined that the submittal met all the "deterministic" requirements for approval of the amendment except for a demonstration of the inspectability of the Electrosleeve repair. The staff was concerned, however, that Electrosleeve repairs would be more likely to fail during certain severe accident sequences than would unflawed tubes. The possible increase in the likelihood of failure is due to the relatively low mechanical strength of Electrosleeves at the high temperatures expected to exist in the steam generator tubes under "high-dry" (high reactor coolant system pressure and dry steam generator secondary side) severe accident conditions. This could result in an increase in the large, early release frequency.

The licensee chose not to submit its license amendment request using the voluntary, risk-informed process described in Regulatory Guide 1.174. A question then arose regarding the authority of the staff to require licensees to submit risk information when the staff believes such information is necessary for an adequate evaluation of the submittals. The staff identified this issue previously in SECY-98-300. The staff concluded that it has the requisite regulatory authority when adequate protection is in question. The staff acknowledged, however, that there is a need for guidance on when, and to what extent, the staff can require quantitative risk information from licensees for license amendment requests that are not submitted using Regulatory Guide 1.174.

In the draft Commission paper, the staff outlined a proposed process for identifying "special circumstances" and for using risk information. The essential elements of the process are:

- Guidance for screening of license amendment requests to identify "special circumstances" which warrant evaluation from a risk perspective.
- A methodology for assessing the risk implications of potentially risk-significant license amendment requests.
- Guidelines for determining the acceptability of the licensing action which factor in risk considerations.

In the draft Commission paper, the staff does not provide specific guidance to identify the "special circumstances" beyond stating that they are circumstances "under which, if [the amendment is] approved, plant operation may pose an undue risk to public health and safety." The staff stated that it expected that the vast majority of licensing decisions would not activate the "special circumstances" screening trigger.

Observations and Recommendations

1. We agree with the staff's assessment that additional guidance is needed on how and under what circumstances the staff can request additional information to address issues associated with submittals not supported by quantitative risk arguments.

2. The process outlined by the staff for determining when additional information should be required is acceptable. The critical element in the process will be the selection of the criteria that define "special circumstances."

3. The staff should be mindful not to create a process that discourages the use of risk-informed submittals. The staff's review of all license amendment requests should include consideration of the principles in Regulatory Guide 1.174.

4. The staff should be sensitive to the potential that its guidelines for requesting quantitative risk information could be interpreted as constituting an implicit requirement for licensees to have probabilistic risk assessments associated with all licensing actions. The staff should also be careful not to inhibit adoption of new, innovative technologies.

5. The staff needs to improve its own risk and accident analysis tools in order to better judge proposed license amendments on a risk-informed basis.

DISCUSSION

The primary mission of the NRC is to ensure adequate protection of public health and safety. The staff's authority to require and use risk information to provide this assurance does not appear to be in doubt. As noted by the staff, Section 182.a of the Atomic Energy Act of 1954, as amended, provides the NRC with the authority to require the submission of information, including risk information, in connection with a license amendment request when NRC has reason to question adequate protection of public health and safety.

The approval of the use of the Electrosleeve process at the Callaway Plant was largely based on analyses of the behavior of repaired tubes under severe accident conditions. The results of these analyses showed that for most flawed tubes the reactor coolant system surge line would fail before the Electrosleeve tubes, thus depressurizing the system and preventing containment bypass. As we have noted previously, these severe accident scenarios involve complex, counter-current, stratified natural circulation flow in the hot leg and in the steam generator plenum — situations that are difficult to analyze with one-dimensional or lumped parameter codes. Tube failure predictions need to account for the relatively large uncertainties in the predicted temperatures as well as the uncertainties in the flaw distributions in the tubes.

The failure of the available technical tools to adequately deal with steam generator tube ruptures under such conditions forces conservative decisionmaking on repair and plugging criteria. Yet, support for essential research necessary to improve these tools continues to diminish due to budgetary constraints.

Sincerely,

Dana A. Powers

Dana A. Powers
Chairman

References:

1. Memorandum dated August 13, 1999, from Gary M. Holahan, Office of Nuclear Reactor Regulation, NRC, to John T. Larkins, ACRS, Subject: Draft SECY Regarding Proposed Guidelines for Applying Risk-Informed Decisionmaking in License Amendment Reviews.
2. Memorandum dated August 3, 1999, SECY-99-199, for the Commissioners, from William D. Travers, Executive Director for Operations, NRC, Subject: Electrosleeve Amendment Issued to Union Electric Company for Callaway Plant, Unit 1.
3. Letter dated May 21, 1999, from M. Gray, Office of Nuclear Reactor Regulation, NRC, to Garry L. Randolph, Union Electric Company, Subject: Amendment No. 132 to Facility Operating License No. NPF-30 - Callaway Plant, Unit 1
4. Memorandum dated May 12, 1999, from Richard J. Barrett, Office of Nuclear Reactor Regulation, NRC, to Stuart A. Richards, Office of Nuclear Reactor Regulation, NRC, Subject: Probabilistic Safety Assessment Branch Input to Safety Evaluation Report on the

Change to Technical Specifications at Callaway Plant to Allow Use of Framatome Electrosleeve Steam Generator Tube Repair Method.

5. Memorandum dated May 28, 1999, from Ashok C. Thadani, Office of Nuclear Regulatory Research, NRC, to Samuel J. Collins, Office of Nuclear Reactor Regulation, NRC, Subject: Electrosleeving Repair of Degraded Steam Generator Tubes.
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.
7. Memorandum dated December 23, 1998, SECY-98-300, from William D. Travers, Executive Director for Operations, NRC, for the Commissioners, Subject: Options for Risk-informed Revisions to 10 CFR Part 50, "Domestic Licensing of Production and Utilization Facilities," December 23, 1998.
8. ACRS Report dated October 22, 1996, from T. S. Kress, Chairman, ACRS, to Shirley Ann Jackson, Chairman, NRC, Subject: Capability of the NRC SCDAP/RELAP5 Code to Predict Temperatures and Flows in Steam Generators Under Severe-Accident Conditions.

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

October 8, 1999

The Honorable Greta Joy Dicus
Chairman
U.S. Nuclear Regulatory Commission
Washington D.C. 20555

Dear Chairman Dicus:

SUBJECT: COMBUSTION ENGINEERING OWNERS GROUP (CEOG) APPLICATION TO ELIMINATE THE POST-ACCIDENT SAMPLING SYSTEM FROM THE PLANT DESIGN BASES FOR CEOG UTILITIES

During the 466th meeting of the Advisory Committee on Reactor Safeguards, September 30-October 2, 1999, we reviewed the CEOG proposal to eliminate the Post-Accident Sampling System (PASS) from the plant design and licensing bases for CEOG plants. Our Subcommittee on Severe Accident Management reviewed this matter during its September 16-17, 1999 meeting. During these meetings, we had benefit of discussions with representatives of the NRC staff, the CEOG, and of the documents referenced.

RECOMMENDATIONS

- The staff should approve the CEOG proposal to eliminate the PASS from the plant design and licensing bases.
- The staff should evaluate the need for new generic requirements on post-accident measurement of in-containment fission products and sump water pH.

DISCUSSION

The PASS regulatory requirements were established after the Three Mile Island, Unit 2 (TMI-2) accident and were provided in Section II.B.3 of NUREG-0737, in 10 CFR 50.34(f)(2)(viii), and in various Generic Letters (Generic Letter (GL) 82-05; GL 83-36; GL 83-37). Regulatory Guide 1.97 describes an acceptable method for compliance.

In general, the requirements stipulate that the licensee shall establish an onsite radiological and chemical analysis capability to provide quantification of the following within a 3-hour period:

- specific radionuclides in the reactor coolant and containment atmospheres,
- hydrogen concentration in the containment atmosphere,

- dissolved gases (e.g., hydrogen), chloride, and boron concentrations in liquids,
- pH in the reactor coolant system (RCS), and
- boron, pH, chlorides, and radionuclides in the containment sump.

In 1993, the staff reviewed and approved the deletion of certain PASS requirements for CEOG plants: (1) pH measurement in the containment sump, (2) hydrogen sampling of the containment atmosphere, (3) sampling for iodine, and (4) oxygen analysis of the reactor coolant. The current proposal is to eliminate the PASS from the plant design and licensing bases for CEOG plants.

In general, the PASS measurements have been required to provide post-accident information to guide decisionmaking with respect to:

- Possible void production due to noncondensable gases in the RCS (the measurement of RCS dissolved gases).
- Achieving cold shutdown (the measurement of RCS boron concentration).
- The needs for emergency response actions – including an estimate of the extent of core damage and fission product release (the measurement of hydrogen and fission products in RCS and containment).
- Re-evolution of gaseous iodine from containment sumps (the measurement of sump water pH).
- Post-accident stress corrosion cracking in the RCS (the measurement of RCS oxygen, chloride, and pH).
- Hydrogen deflagration in containment (measurement of hydrogen and oxygen in containment).
- Stress corrosion cracking of recirculation systems (measurement of containment sump chlorides).
- Assurance of subcriticality should sump water be used in the recirculation mode to cool the core (measurement of sump water boron concentration).

The CEOG has made a persuasive case that the PASS measurements are not needed and can be eliminated without undue increase in risk because each of the requirements is being satisfied by other information sources. We concur with this assessment. It is also our view, however, that the current post-accident sampling systems are poorly designed and poorly configured to provide the information for the needs listed above. This is the primary reason that other information sources are used for accident management and emergency response purposes.

We believe that there would be significant post-accident management benefit in having timely measurement of sump pH and fission product concentrations in the containment. Information on concentrations of krypton and cesium in containment can provide direct indications of fission product release and core damage that are difficult to infer from total radiation, temperature, and hydrogen concentration measurements.

We also believe that sump radiochemistry under post-accident conditions cannot be predicted to a level of accuracy that would provide the required assurance that buffered sumps will inhibit the re-evolution of gaseous species of iodine. The actual measurement of pH will be necessary to assess the pH status of sumps and to guide post-accident decisions related to the need for additional emergency response, accident management, containment venting, or ingress into containment in the long term.

We believe, however, that the value of these measurements does not warrant continuation of the current methods for implementation of the PASS requirements through grab sampling in the containment atmosphere and from the containment sump. On the other hand, we believe there is technology available with which this information could be obtained on a continuous basis by the use of tuned gamma monitors in containment and pH instrumentation in the sump. Therefore, we recommend that the staff evaluate the need for generic requirements for timely post-accident measurements of sump pH and fission product concentrations in the containment.

Sincerely,

Dana A. Powers

Dana A. Powers,
Chairman

References:

1. U. S. Nuclear Regulatory Commission, "Safety Evaluation Report by the Office of Nuclear Reactor Regulation Related to the Technical Basis for Allowing Combustion Engineering Pressurized Water Reactors to Change Commitments Related to Post Accident Sampling," undated draft, received September 21, 1999.
2. Combustion Engineering Owners Group, CENPSD-1157, "Technical Justification for the Elimination of the Post-Accident Sampling System from the Plant Design and Licensing Bases for CEOG Utilities," dated May 1999.
3. U. S. Nuclear Regulatory Commission, "Safety Evaluation Report by the Office of Nuclear Reactor Regulation Related to the Technical Basis for Allowing Westinghouse Pressurized Water Reactors to Change Commitments Related to Post Accident Sampling," undated, draft.
4. U.S. Nuclear Regulatory Commission, NUREG-0737, "Clarification of TMI Action Plan Requirements," dated November 30, 1980.
5. U.S. Nuclear Regulatory Commission, Office of Nuclear Reactor Regulation, Subject: NUREG-0737 Technical Specifications (Generic Letter No. 83-36), to all Boiling Water Reactor Licensees, dated November 1, 1983.

6. U.S. Nuclear Regulatory Commission, Subject: NUREG-0737 Technical Specifications (Generic Letter 83-37), to all Pressurized Water Reactor Licensees, dated November 1, 1983.
7. U.S. Nuclear Regulatory Commission, Generic Letter 82-05, Subject: Post-TMI Requirements, dated March 17, 1982.
8. U.S. Nuclear Regulatory Commission, Regulatory Guide 1.97, "Instrumentation for Light Water Cooled Nuclear Power Plants to Assess Plant and Environs Conditions During and Following an Accident," Revision 3, dated May 1983.

October 8, 1999

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 SAFETY ISSUE 23 (GSI-23), "REACTOR COOLANT PUMP SEAL FAILURE"

During the 466th meeting of the Advisory Committee on Reactor Safeguards, September 30-October 2, 1999, we reviewed the NRC staff's proposed resolution of GSI-23, "Reactor Coolant Pump Seal Failure." Our Subcommittee on Thermal-Hydraulic Phenomena also reviewed this matter during its September 15-16, 1999, meeting. During these meetings, we had the benefit of discussions with representatives of the NRC staff, the Westinghouse Owners Group, and of the documents referenced.

Recommendations

- We agree with the staff's proposed approach to the resolution of GSI-23.
- In performing plant-specific evaluations, the staff should assess how relevant uncertainties, in particular those arising from the predictions of reactor coolant pump seal leak rate, affect conclusions about risk significance.
- Despite major improvements in seal materials for Westinghouse pumps, only 75 percent of the plants that have such pumps have installed this new seal material. The staff should evaluate on a plant-specific basis whether installation of improved seal material should be required.
- The staff has chosen to analyze all reactor coolant pump seals using the predictions of flow rates and probability models developed for Westinghouse pumps. We recommend that more realistic analysis of non-Westinghouse pumps be made.

Discussion

GSI-23 was identified in 1980 as a result of a large number of pump seal failures experienced at nuclear power plants during normal operation. Concern arose because the possible leak rates through these failed seals could amount to several hundred gallons per minute (gpm) per pump. The resulting small-break loss-of-coolant accident could lead to core uncovery under some circumstances.

Since 1980 improvements have been made in pump seal materials and the methods for cooling them, and there have been no pump seal failures that have resulted in a leakage of primary coolant exceeding 100 gpm.

The staff has determined that the accident sequences involving pump seal failures are potentially risk-significant for only a handful of plants. Therefore, this matter no longer qualifies as a GSI. The staff plans to resolve GSI-23 on this basis and to conduct plant-specific reviews to determine whether backfits are needed.

The Offices of Nuclear Reactor Regulation and Nuclear Regulatory Research are developing a Task Action Plan to determine the need for plant-specific backfits. We plan to review this Task Action Plan and would like to be informed of future actions by the staff on this issue.

Sincerely,

Dana A. Powers

Dana A. Powers
Chairman

References:
1. Memorandum (undated) from Ashok C. Thadani, Office of Nuclear Regulatory Research, NRC, to William D. Travers, Executive Director for Operations, NRC, Subject: Closeout of Generic Safety Issue 23, "Reactor Coolant Pump Seal Failure," received September 20, 1999.
2. U.S. Nuclear Regulatory Commission, NUREG/CR-4294, "Leak Rate Analysis of the Westinghouse Reactor Coolant Pump," prepared by Energy Technology Engineering Center, dated July 1985.
3. U.S. Nuclear Regulatory Commission, NUREG/CR-5167, "Cost/Benefit Analysis for Generic Issue 23: Reactor Coolant Pump Seal Failure," prepared by SCIENTECH, Inc., dated April 1991.
4. WCAP-10541, Revision 2, Excerpts from Westinghouse Owners Group Report, "Reactor Coolant Pump Seal Performance Following a Loss of All AC Power," dated November 1986 (**Proprietary**).

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

October 8, 1999

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

Dear Dr. Travers:

SUBJECT: PROPOSED RESOLUTION OF GENERIC SAFETY ISSUE B-55, "IMPROVED RELIABILITY OF TARGET ROCK SAFETY RELIEF VALVES"

During the 466th meeting of the Advisory Committee on Reactor Safeguards, September 30-October 2, 1999, we reviewed the proposed resolution of Generic Safety Issue (GSI) B-55, "Improved Reliability of Target Rock Safety Relief Valves." During our review, we had the benefit of discussions with representatives of the NRC staff. We also had the benefit of the document referenced.

Conclusion and Recommendation

- We agree with the staff's proposed resolution of GSI B-55.
- The staff should perform a statistical analysis to ensure that the apparent improvement in performance of the two-stage valves is significant and confirms its conclusion.

Background

The boiling water reactor (BWR) pressure relief system is designed to prevent over-pressurization of the reactor coolant pressure boundary. This protection is accomplished through the use of a plant-unique combination of safety valves (SVs), power actuated relief valves (PARVs), and dual function safety relief valves (SRVs) that have both a mechanical self-actuating setpoint function and a power-actuated function. The majority of the SRVs in older BWRs were manufactured by Target Rock.

Discussion

Some SRVs have exhibited anomalies, such as:

- Spurious actuation
- Upward setpoint drift
- Excessive blowdown

The BWR Owners Group and the individual BWR licensees have improved the performance of the SRVs by installing ion-beam implanted platinum disks or Stellite 21 disks to improve seating and installing additional pressure switches to actuate these valves using pneumatic power. Based on recent performance data, the staff has concluded that both the Stellite 21 and the ion-beam implanted platinum disks are performing better than the former Stellite 6B disks with a lower rate of occurrence of high setpoint drift beyond that allowed by plant Technical Specifications. The conclusion concerning the relative performance of the different disk materials would be more persuasive if it were supported by an appropriate statistical analysis of the data.

In addition, the staff stated that the affected BWR plants have sufficient margin to accommodate upward valve setpoint drift as high as 10 percent. In view of the improvement in valve performance, the margin available to accommodate upward setpoint drift and other options such as pressure switches, the staff does not plan to initiate any new regulatory actions.

We agree with the proposed resolution of GSI B-55. The activities being pursued by the licensees under existing regulatory requirements are sufficient. There is no need to impose any additional regulatory requirements.

Sincerely,

Dana A. Powers
Chairman

Reference:
E-mail to John T. Larkins, ACRS, from Charles Hammer, Office of Nuclear Reactor Regulation, NRC, Subject: ACRS Briefing of Generic Safety Issue B-55, "Improved Reliability of Target Rock Safety Relief Valves," dated September 2, 1999.

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

October 12, 1999

The Honorable Greta Joy Dicus
Chairman
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555-0001

Dear Chairman Dicus:

SUBJECT: PROPOSED PLANS FOR DEVELOPING RISK-INFORMED REVISIONS TO 10 CFR PART 50, "DOMESTIC LICENSING OF PRODUCTION AND UTILIZATION FACILITIES"

During the 466th meeting of the Advisory Committee on Reactor Safeguards, September 30-October 2, 1999, we met with representatives of the NRC staff and Nuclear Energy Institute to discuss proposed plans for developing risk-informed revisions to 10 CFR Part 50. We also met with a representative of Public Citizen, Critical Mass Energy Project, to discuss these matters and a recent report issued by Public Citizen. Our Subcommittees on Reliability and Probabilistic Risk Assessment and on Regulatory Policies and Practices met on July 13 and September 24, 1999, to discuss these matters. We had the benefit of the documents referenced.

Conclusions and Recommendations

1. We agree with the staff's proposal to develop a new regulatory section 10 CFR 50.69 and associated Appendix T to implement Option 2 (changing the special treatment rules in 10 CFR Part 50) of SECY-98-300.

2. We agree that the current terminology of safety-related structures, systems, and components (SSCs) should be preserved and that additional terminology referring to the safety significance of SSCs should be considered. We recommend that the staff explore the potential benefits of defining more than two categories of safety significance.

3. The determination of the safety significance of SSCs relies heavily on the use of importance measures. These measures are strongly affected by the scope and quality of the probabilistic risk assessment (PRA). For example, incomplete assessments of risk contributions from low-power and shutdown operations, fires, and human performance will distort the importance measures.

4. Even with a full-scope, high-quality PRA, the importance measures have limitations. The guidance to be provided in the proposed Appendix T for the categorization of SSCs should clarify the proper roles of (a) importance measures, (b) sensitivity and uncertainty analysis, (c) baseline core damage frequency (CDF) and large, early release frequency (LERF), and (d) the changes in CDF and LERF (i.e., ΔCDF and ΔLERF).

5. It is essential that the implementation of Option 2 be scrutable and auditable. The staff should have access to the risk assessments and technical bases documents (e.g., inputs to and deliberations of the expert panel) that licensees use to justify requests.

6. The guidance to be provided in the proposed Appendix T for the expert panel should include insights gained from the implementation of recommendation 4 above. The staff should include guidance for conducting expert panel sessions and training of the panel members on the use of importance measures.

7. We agree with the staff's plan for implementing Option 3 (changing specific requirements in the body of 10 CFR Part 50 and associated regulations) of SECY-98-300. Policy issues regarding the role of defense in depth in a risk-informed regulatory system should be resolved before the plan is fully implemented.

Discussion

In a Staff Requirements Memorandum dated June 8, 1999, the Commission directed the staff to make risk-informed changes to the scope of SSCs covered by regulations that provide special treatment requirements (e.g., quality assurance, environmental qualification, technical specifications, 10 CFR 50.59, ASME Code, 10 CFR 50.72, and 10 CFR 50.73). 10 CFR 50.2 defines safety-related SSCs as those SSCs that "are relied upon to remain functional during and following design basis events to assure: (1) The integrity of the reactor coolant boundary; (2) The capability to shut down the reactor and maintain it in a safe shutdown condition; or (3) The capability to prevent or mitigate the consequences of accidents which could result in potential offsite exposures...."

To date, the determination of whether an SSC is safety related has been based largely on deterministic analyses that include engineering judgment. Advances in PRAs have made it possible to quantify the degree to which SSCs are relied upon to ensure that the requirements in 10 CFR 50.2 are met. For example, using a combination of deterministic and PRA insights, the South Texas Project Nuclear Operating Company has concluded that many SSCs currently categorized as safety-related contribute very little to CDF and LERF, while a few SSCs currently categorized as nonsafety-related are significant from a risk perspective.

The staff proposes to develop a new rule, 10 CFR 50.69, and an associated Appendix T. The new rule will explicitly allow the use of a new risk-informed scope. Appendix T will provide the criteria for the new categorization process. We agree with this approach.

The current “safety-related” and “nonsafety-related” categories will be retained. Two new categories that consider risk information, i.e., high safety significance and low safety significance, will be developed. Appendix T will provide criteria for the new categorization process. The staff proposes to use a 2x2 matrix where SSCs are to be placed in one of the four categories according to safety significance and safety-related status. Introducing these new categories while preserving the safety-related and nonsafety-related terminology should help to avoid the confusion that could result from a redefinition of the safety-related concept. We agree that such an approach is preferable to redefining “safety-related” and “important to safety.”

At this early stage, the staff has not decided what special treatment the SSCs in each of the four categories of the 2x2 matrix will receive. The staff has indicated that this decision may require a finer treatment of safety significance than the two groups to be proposed in Appendix T. The South Texas Project Nuclear Operating Company has chosen to consider four groups for safety significance instead of the two that will be proposed for Appendix T. They are: 1) high safety/risk significant (HSS), 2) medium safety/risk significant (MSS), 3) low safety/risk significant (LSS), and 4) non-risk significant (NRS). LSS and NRS SSCs support ancillary functions (e.g., vents and drains) for safety-related systems, but do not affect the primary functions of these systems. LSS SSCs may be included in the PRA while NRS SSCs are not.

We believe that the staff should further evaluate the various options for partitioning the range of safety significance before it settles on a grouping that it considers optimum.

Appendix T will include requirements for categorizing SSCs using PRA. We offer the following comments and suggestions for inclusion in the development of Appendix T:

1. The screening criteria are based primarily on two importance measures: Fussell-Vesely (FV) and Risk Achievement Worth (RAW). The criteria are: FV > 0.005 and RAW > 2 based on either CDF or LERF. It is important to fully understand what information these measures convey as well as their limitations. Detailed discussions on these matters are available in References 9, 12, and 13.

   As an example, consider a very simple case in which the risk metric, e.g., the CDF due to internal events, is a function of a single accident sequence. We have

   $$CDF^{IE} = fq = 10^{-4} \text{ per reactor-year} \qquad (1)$$

   where
   f: frequency of the initiating event (say, 10^{-2} per reactor-year)
   q: unavailability of the protection system (say, 10^{-2} per demand)

   The importance measures for the system are

   $$FV = \frac{fq}{fq} = 1 \qquad (2)$$

$$RAW = \frac{CDF^{IE,+}}{CDF^{IE}} = \frac{f}{fq} = \frac{1}{q} = 100 \tag{3}$$

where $CDF^{IE,+}$ is the new value of CDF with the protection system assumed unavailable.

Suppose that several protection systems are added, each of unavailability q_j. The new importance measures for the system are

$$FV' = \frac{fq \prod q_j}{fq \prod q_j} = 1 \tag{4}$$

$$RAW' = \frac{f \prod q_j}{fq \prod q_j} = \frac{1}{q} = 100 \tag{5}$$

Even though several protection systems have been added thereby reducing reliance on the original system and reducing the overall risk, the importance measures have not changed. We believe that this insensitivity should be better understood and communicated to the expert panel and that insights from this discussion need to be incorporated into the rule or the associated guidance documents.

2. Suppose that the CDF estimate of Equation (1) is expanded to include the contribution from external events. We assume that this contribution is 10^{-3} per reactor-year, i.e., it dominates the risk due to internal events, as is often the case with the seismic contribution. The new CDF is

$CDF = CDF^{IE} + CDF^{EE} = 10^{-4} + 10^{-3} = 1.1x10^{-3}$ per reactor-year (6)

A calculation of the new importance measures provides:

$$FV'' = \frac{10^{-4}}{1.1x\,10^{-3}} = 0.09 \tag{7}$$

$$RAW'' = \frac{10^{-2} + 10^{-3}}{1.1x\,10^{-3}} = 10 \tag{8}$$

As expected, the importance measures of the protection system have been reduced drastically. The question is whether including the dominant seismic contribution results in meaningful importance measures, especially within the context of the proposed new reactor oversight process where the frequency of initiating events and the unavailability of the protection systems are cornerstones of the assessment process.

In a PRA, the additional terms in the equation may be the products of analyses that are not as rigorous as those for the terms in which a particular system appears. For example, some terms may contain probabilities of recovery actions or damage caused by "external" events, such as fires and tornadoes. The current assessment of risk contributions from low-power and shutdown operations, fires, and human performance is incomplete. Because the PRA technology for such assessments is not as well developed as that for "internal" events, the analyses may contain many overly conservative assumptions, thus artificially increasing these contributions. Inconsistencies in the analysis of the various contributions to risk distort the importance measures.

It is evident that the absolute value of the baseline risk metric is a critical element in these evaluations and that the importance measures contain only relative information with respect to a given risk metric.

The change in risk depends on this absolute value also, i.e., ΔCDF at two plants with different baseline CDFs, will be different for the same change in the unavailability of a component whose importance measures have the same value at these plants. Reference 9 states that "if we are interested in controlling the change in risk in an absolute sense, it does not make sense to have a universally fixed value of FV as a criterion for risk significance," and "it is clear that it does not make much sense to define a universal criterion based on RAW."

3. The calculation of RAW in Equation (3) requires the estimation of $CDF^{IE,+}$, i.e., the CDF assuming that the protection system is unavailable. This assessment may be much more involved than simply setting the unavailability of the system equal to unity. The assumption of a system being unavailable may affect several terms in the PRA. For example, in a two-train redundant system, the PRA contains terms representing the "random" independent failure of the two trains, the probability of a common-cause failure, and the probability that coupled human errors after test and maintenance may disable both trains. All of these terms are affected by the assumption of one train being unavailable. Recovery actions may also be affected (see Reference 11).

   We question whether these considerations are adequately taken into account when RAW is calculated for hundreds of components.

4. The current practice of calculating FV and RAW is to use the mean epistemic values of the parameters in the ratios appearing in Equations (2) and (3). The more rigorous way is to first find the ratios and then to average them over the epistemic distributions of the parameters (Reference 10). The current practice is an approximation that is usually reasonable, unless the epistemic uncertainties of the parameters are very large (Reference 9). The section on sensitivity analysis in the proposed Appendix T should reflect this observation.

The preceding paragraphs are not intended to discourage the use of importance measures. Although our example is a simple one, it does illustrate that FV and RAW values must be carefully calculated and interpreted. We do believe that a good understanding of the limitations of importance measures is essential to their proper use.

The issues discussed above, as well as the detailed investigations in the cited references, suggest that the members of the expert panel that determines the categorization of SSCs need to be aware of these limitations and constraints. We believe that there is a need to ensure that members of expert panels have formal training in the properties of importance measures. Similar training sessions are provided in other contexts, e.g., before quantitative judgments are elicited from engineers and scientists who are not familiar with the cognitive issues associated with the elicitation of expert opinion.

Option 3 of SECY-98-300 deals with changes in specific requirements in 10 CFR Part 50, including general design criteria. The staff's high-level plan for implementing this option and associated study is acceptable. We note, however, that defense in depth plays a critical role in this plan.

The PRA Policy Statement of 1995 and subsequent agency documents such as Regulatory Guide 1.174 for risk-informed changes to the licensing basis place defense in depth at the level of a principle whereby PRA should be used in "a manner that supports the NRC's traditional defense-in-depth philosophy." As noted in our May 19, 1999 report, this may create conflicts between risk-informed insights and defense in depth. Since the staff's plan includes defense-in-depth considerations in several key areas, e.g., the identification of candidate requirements to be revised and the determination of the revisions, it is very important for the Commission to clarify the proper role of defense in depth.

We look forward to working with the staff to resolve the significant technical issues associated with the implementation of Options 2 and 3 of SECY-98-300.

Sincerely,

Dana A. Powers

Dana A. Powers
Chairman

References:

1. Memorandum dated September 16, 1999, from David B. Mattews, Office of Nuclear Reactor Regulation, to John T. Larkins, Executive Director, Advisory Committee on Reactor Safeguards, Subject: ACRS Preliminary Review of the Draft SECY Paper for Risk-Informing Special Treatment Regulations.
2. Memorandum dated September 23, 1999, from Thomas L. King, Office of Nuclear Regulatory Research, to John T. Larkins, Advisory Committee on Reactor Safeguards, Subject: ACRS Review of Proposed Staff Plan for Risk-Informing Technical Requirements in 10 CFR Part 50

3. Memorandum dated June 8, 1999, from Annette Vietti-Cook, Secretary, NRC, to William D. Travers, Executive Director for Operations, Subject: Staff Requirements - SECY-98-300 - Options for Risk-Informed Revisions to 10 CFR Part 50 - "Domestic Licensing of Production and Utilization Facilities."
4. Letter dated December 14, 1998, from R. L. Seale, Chairman, ACRS, to William D. Travers, Executive Director for Operations, NRC, Subject: Proposed Commission Paper Concerning Options for Risk-Informed Revisions to 10 CFR Part 50 - "Domestic Licensing of Production and Utilization Facilities."
5. Report dated May 19, 1999, from Dana A. Powers, Chairman, Advisory Committee on Reactor Safeguards, to Shirley Ann Jackson, Chairman, U.S. Nuclear Regulatory Commission, Subject: The Role of Defense in Depth in a Risk-Informed Regulatory System.
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.
7. Letter dated July 13, 1999, from J.J. Sheppard, South Texas Nuclear Operating Company, to U.S. Nuclear Regulatory Commission, Subject: Request for Exemption to Exclude Certain Components from the Scope of Special Treatment Requirements Required by Regulations.
8. Title 10: Code of Federal Regulations, Part 50, Domestic Licensing of Production and Utilization Facilities, Section 50.2, Definitions.
9. 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.
10. G. Apostolakis, "The distinction between aleatory and epistemic uncertainties is important: An example from the inclusion of aging effects into PSA," American Nuclear Society Conference, PSA '99, Washington, DC, August 22-25, 1999.
11. C.L. Smith, "Calculating conditional core damage probabilities for nuclear plant operations," *Reliability Engineering and System Safety* 59, 299-307, 1998.
12. W.E. Vesely, "Reservations on 'ASME Risk-Based Inservice Inspection and Testing: An Outlook to the Future,'" *Risk Analysis* 18, 423-425, 1998.
13. 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.
14. Report entitled, "Amnesty Irrational -- How the Nuclear Regulatory Commission Fails to Hold Nuclear Reactors Accountable for Violations of Its Own Safety Regulations," by James P. Riccio, Public Citizen, August 1999.
15. U.S. Nuclear Regulatory Commission, "Use of Probabilistic Risk Assessment Methods in Nuclear Activities: Final Policy Statement," dated August 16, 1995.

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

November 12, 1999

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

Dear Chairman Meserve:

SUBJECT: PROPOSED FINAL DESIGN CERTIFICATION RULE AND CHANGES TO THE DESIGN CONTROL DOCUMENT ASSOCIATED WITH AP600 DESIGN

During the 467th meeting of the Advisory Committee on Reactor Safeguards, November 4-6, 1999, we reviewed the changes to the AP600 Design Control Document and the associated Supplement 1 to NUREG-1512, "Final Safety Evaluation Report Related to Certification of the AP600 Standard Design." We also considered the proposed final AP600 Design Certification Rule. During our review, we had the benefit of discussions with representatives of the NRC staff and Westinghouse Electric Company, and of the documents referenced.

Conclusion

- Our review of the changes to the AP600 Design Control Document and the associated Supplement 1 to NUREG-1512 did not change the conclusion in our report of July 23, 1998. In that report, we concluded that acceptable bases and requirements have been established to ensure that the AP600 design can be used to engineer and construct plants that, with reasonable assurance, can be operated without undue risk to the health and safety of the public.

- We decided not to review the proposed final AP600 Design Certification Rule since it is essentially the same as the rules for certification of evolutionary nuclear power plant designs (General Electric Advanced Boiling Water Reactor design and ABB-Combustion Engineering System 80+ design.).

Background and Discussion

We reviewed the AP600 standard design in accordance with 10 CFR Part 52, which requires the ACRS to report on those portions of the application that concern safety. In our present review, we considered changes to the AP600 Design Control Document, including changes to the design of the plate above a containment sump screen and an increase in the calculated concentrations of hydrogen in the containment following a loss-of-coolant accident (LOCA).

The area of the plate was reduced to avoid mechanical interference with a steam generator. The redesigned plate was also lowered closer to the top of the containment sump screen in order to reduce debris accumulation on the screen. The design change was judged to increase safety.

Results of the calculations using the final version of the WGOTHIC code demonstrated that the long-term containment temperatures following a LOCA are higher than originally predicted. The higher temperatures lead to a predicted increase in the hydrogen concentrations. However, the post-LOCA hydrogen concentrations remain well below flammability limits throughout the accident.

Sincerely,

Dana A. Powers

Dana A. Powers
Chairman

References

1. Memorandum dated September 28, 1999, from David B. Matthews, Office of Nuclear Reactor Regulation, NRC, to John T. Larkins, ACRS, Subject: Final Rule -- AP600 Design Certification.
2. Memorandum dated October 7, 1999, from David B. Matthews, Office of Nuclear Reactor Regulation, NRC, to John T. Larkins, ACRS, transmitting Supplement 1 to the AP600 Final Safety Evaluation Report.
3. Letter dated September 29, 1999, from Brian A. McIntyre, Westinghouse Electric Company, to Document Control Desk, NRC, transmitting AP600 Design Control Document, September 1999 Revision.
4. Letter dated September 15, 1999, from Jerry N. Wilson, Office of Nuclear Reactor Regulation, NRC, to Westinghouse Electric Company, Subject: Meeting Summary On Design Control Document Changes.
5. Report dated July 23, 1998, from R. L. Seale, Chairman, ACRS, to Shirley Ann Jackson, Chairman, NRC, Subject: Report on the Safety Aspects of the Westinghouse Electric Company Application for Certification of the AP600 Passive Plant Design.

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NUCLEAR REGULATORY COMMISSION
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WASHINGTON, D.C. 20555-0001

November 12, 1999

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

Dear Chairman Meserve:

SUBJECT: PROPOSED REVISION 3 TO REGULATORY GUIDE 1.160 (DG-1082), "ASSESSING AND MANAGING RISK BEFORE MAINTENANCE ACTIVITIES AT NUCLEAR POWER PLANTS"

During the 467th meeting of the Advisory Committee on Reactor Safeguards, November 4-6, 1999, we reviewed the proposed Revision 3 to Regulatory Guide 1.160 (DG-1082), "Assessing and Managing Risk Before Maintenance Activities at Nuclear Power Plants," and the revised draft of Section 11 of NUMARC 93-01, "Industry Guideline for Monitoring the Effectiveness of Maintenance at Nuclear Power Plants." During our review, we had the benefit of discussions with representatives of the NRC staff and the Nuclear Energy Institute (NEI). We previously commented on an earlier version of this guide in a report dated July 21, 1999.

Recommendations

1. The proposed Revision 3 to Regulatory Guide 1.160 should be issued for public comment.

2. We support the staff's endorsement of the NEI guidance for industry use when revised to incorporate the staff's comments and to provide a concise definition of unavailability.

Discussion

Both the staff and NEI agree that the proposed Revision 3 to Regulatory Guide 1.160 and NUMARC 93-01 provide an acceptable method for assessing and managing the increase in risk that may result from nuclear power plant maintenance activities, as required by new paragraph (a)(4) of 10 CFR 50.65. The guidance that the staff and NEI have developed resolves our concerns that we raised in our report of July 21, 1999.

There are three minor issues between the staff and NEI that we were assured would be resolved easily. In addition, the definition of unavailability in the draft Regulatory Guide needs to be clarified. The description of unavailability provided in Appendix B of the proposed modification to NUMARC 93-01 is not a definition. The commonly accepted definition of the unavailability of a system that is under periodic surveillance testing is simply the average fraction of time during which the system is incapable of performing its intended function. The equation in Appendix B is correct, if "required operational hours" is interpreted as the period of surveillance tests.

Sincerely,

Dana A. Powers

Dana A. Powers
Chairman

References:

1. Memorandum dated October 18, 1999, from Theodore R. Quay, Office of Nuclear Reactor Regulation, NRC, to John T. Larkins, ACRS, Subject: Request for Review of Draft Regulatory Guidance for 10 CFR 50.65, The Maintenance Rule.
2. Final Draft of Section 11, "Assessment of Risk Resulting from Performance of Maintenance Activities," of NUMARC 93-01, "Industry Guideline for Monitoring the Effectiveness of Maintenance at Nuclear Power Plants," dated October 8, 1999.
3. Report dated July 21, 1999, from Dana A. Powers, Chairman, ACRS, to Greta Joy Dicus, Chairman, NRC, Subject: Proposed Revision 3 to Regulatory Guide 1.160 (DG-1082), "Assessing and Managing Risk Before Maintenance Activities at Nuclear Power Plants."

UNITED STATES
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November 12, 1999

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

Dear Dr. Travers:

SUBJECT: SPENT FUEL FIRES ASSOCIATED WITH DECOMMISSIONING

During the 467th meeting of the Advisory Committee on Reactor Safeguards, November 4-6, 1999, we reviewed a draft report of a technical study prepared by the NRC staff on the spent fuel pool accident risk at decommissioning plants. During our review, we had the benefit of discussions with representatives of the NRC staff, the Nuclear Energy Institute (NEI), and two members of the public. We also had the benefit of the documents referenced.

Background

The staff discussed with us the status of its ongoing work on this issue. We appreciate the opportunity to provide our views on the direction of this effort at this interim stage.

The staff has formed a Technical Working Group with the objective of assessing the risks associated with spent fuel pools for decommissioning plants. The intent is to assist the Office of Nuclear Reactor Regulation in developing an integrated rule for decommissioning, to provide guidance for interim exemption requirements, and to identify areas where additional work is needed.

Fuel removed from a reactor must be covered with water for cooling until its decay heat generation rate falls below a critical value. Risks posed by fuel stored in a pool arise from the possibility that this water cooling may be lost. The staff has a two-fold approach to evaluating the issues of spent fuel storage: (1) develop estimates of the decay time required to avoid runaway oxidation of spent fuel clad in the event of accidental uncovery, and (2) develop a risk assessment using a broad set of initiating events and using the end-state consequence of uncovery to the top of the fuel.

NEI has interacted with the staff on this effort and has provided a review of the draft report entitled, "Technical Study of Spent Fuel Pool Accidents for Decommissioning Plants." NEI provided us with its assessments. Our understanding of the more substantive issues raised by NEI is:

1. Conservatism, especially in human error rates, has skewed the preliminary risk insights.

2. The choice of uncovery to the top of the fuel as the endpoint is difficult to relate to public risk. NEI believes that the analyses should be carried all the way to postulated runaway oxidation.

3. The cladding temperature used as the threshold for onset of runaway oxidation is too low.

We also had benefit of the remarks by a member of the public who expressed concern about the:

- Degree of public participation in this effort
- Acceptability to the public of PRA (probabilistic risk assessment) based regulations
- Lack of sufficient margins and defense-in-depth
- Severity of the consequences
- Vulnerability to terrorism
- Applicability of the database used for equipment failures
- Potential for recriticality

<u>Conclusions and Recommendations</u>

1. We agree with the general approach for determining the decay time beyond which runaway oxidation cannot occur. However, an uncertainty analysis related to the oxidation kinetics and the heat rejection mechanisms is needed. The present analysis is limited to relatively low-burnup levels and associated clad hydriding and oxidation. There are no experimental data on the behavior of realistic fuel and cladding under representative conditions. Either very conservative choices will have to be made for decay times or additional experimental research will have to be conducted.

2. We support the staff's approach to developing a decay heat critical temperature for the onset of runaway oxidation. Uncertainties in these analyses need to be quantified and factored into any decisions regarding the required decay time.

3. PRAs should be as realistic as possible. The staff should reevaluate the basis for its choices particularly for human error rates. We agree with the staff's proposal to use expert opinion to validate or modify the human reliability analyses to ensure that the analyses are not overly conservative.

4. Arguments about conservative versus realistic values are aggravated when point estimates are used for the input parameters to the risk assessments. As stated in our December 16, 1997 report, we believe that uncertainties can be best addressed by expressing the inputs as probability distributions rather than point estimates. Such distributions are easier to defend. In addition, the insights to be gained from the risk analysis would greatly benefit if the results were presented as distributions.

5. We agree with the choice of uncovery to the top of the fuel as being an appropriate end state for the PRA consequence analysis. The database on air oxidation kinetics for high-burnup fuel, subsequent fuel damage behavior, and fission product release is too sparse and the uncertainties too great to provide confidence in carrying the analyses any farther. The acceptable frequency of this end point can be based on consideration of the health consequences resulting from postulated fuel failures. Because prompt fatalities cannot be ruled out, we recommend that the acceptable frequency for this end point be the same as that for large, early release frequency in Regulatory Guide 1.174, which is a surrogate for the prompt fatality Safety Goal.

With the choice of uncovery as the end state of the analysis, the uncertainties due to model inadequacies associated with fire risk assessment are not large. We believe that the spent fuel fire issue would be a good candidate for testing the development of a rationalist regulatory approach, as discussed in our May 19, 1999 report.

We look forward to reviewing the staff's progress in this area.

Sincerely,

Dana A. Powers

Dana A. Powers
Chairman

References:

1. Draft report entitled, "Technical Study of Spent Fuel Pool Accidents for Decommissioning Plants," prepared by NRC Technical Working Group, June 1999.
2. A Review of Draft NRC Staff Report: "Draft Technical Study of Spent Fuel Pool Accidents for Decommissioning Plants," prepared by ERIN Engineering and Research, Inc., for Nuclear Energy Institute, dated August 27, 1999.
3. Draft (undated) EPRI Technical Report, "Evaluation of Spent Fuel Pool Seismic Failure Frequency in Support of Risk Informed Decommissioning Emergency Planning," prepared by Duke Engineering & Services.
4. Letter dated September 3, 1999, from Mr. David A. Lochbaum, Union of Concerned Scientists, to NRC Commissioners, Subject: Inadequately Monitored Spent Fuel Pool Temperature and Operator Response Times at Permanently Closed Plants.
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.
6. ACRS report dated December 16, 1997, from R. L. Seale, Chairman, ACRS, to Shirley Ann Jackson, Chairman, NRC, Subject: Treatment of Uncertainties Versus Point Values in the PRA-Related Decisionmaking Process."
7. ACRS report dated May 19, 1999, from Dana A. Powers, Chairman, ACRS to Shirley Ann Jackson, Chairman, NRC, Subject: The Role of Defense in Depth in a Risk-Informed Regulatory System.

UNITED STATES
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November 12, 1999

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

Dear Dr. Travers:

SUBJECT: PROPOSED RESOLUTION OF GENERIC SAFETY ISSUE (GSI)-148, "SMOKE CONTROL AND MANUAL FIRE-FIGHTING EFFECTIVENESS"

During the 467th meeting of the Advisory Committee on Reactor Safeguards, November 4-6, 1999, we completed our review of the proposed resolution of GSI-148, "Smoke Control and Manual Fire-Fighting Effectiveness." During our review, we had the benefit of discussions with representatives of the NRC staff. We also had the benefit of the documents referenced.

Conclusion and Recommendation

- We concur with the staff's proposal for resolving GSI-148.
- Section 3 of Reference 1 should be revised to include guidance on addressing the effects of smoke on manual fire fighting.

Discussion

Smoke has a major influence on fire brigade response times and can hamper the operators' ability to shut down the plant safely. GSI-148 has been classified as a "licensing issue." The staff proposed that plant-specific reviews be performed to evaluate the significance of this issue. Such reviews have been performed as part of the Individual Plant Examination of External Events (IPEEE) program.

On the basis of IPEEE submittals and fire brigade training programs, and observations made by resident inspectors, the staff believes that smoke control and manual fire-fighting effectiveness have been adequately addressed.

In Reference 1 the staff discusses how smoke can impact plant risk, however, the effects of smoke are not addressed in Section 3 of this document that discusses review guidance for the staff. This section should be revised to include guidance for use by the staff in evaluating the impact of smoke on manual fire-fighting effectiveness.

Licensee assessments have focused on the localized effects of smoke on manual fire fighting. Smoke can spread well beyond the area of generation and create immediate and delayed effects on instrumentation and control circuits. These effects of smoke are not being addressed in GSI-148. The Office of Nuclear Regulatory Research is studying the effects of smoke from cable fires on digital electronic circuits. The results of this study should help to assess the potential impact of these effects.

Based on the results of the staff review of IPEEE submittals to date, anticipated revision to Section 3 of Reference 1, and the research activities in the area of smoke propagation, we agree with the staff's proposal to resolve GSI-148.

Sincerely,

Dana A. Powers

Dana A. Powers
Chairman

References:

1. Memorandum dated July 22, 1999, from Thomas L. King, Office of Nuclear Regulatory Research, NRC, to Ashok C. Thadani, Office of Nuclear Regulatory Research, NRC, Subject: Staff Review Guidance for Generic Safety Issue (GSI) 148, "Smoke Control and Manual Fire-Fighting Effectiveness."
2. U. S. Nuclear Regulatory Commission, Generic Letter No. 88-20, Supplement 4, "Individual Plant Examination of External Events (IPEEE) for Severe Accident Vulnerabilities - 10 CFR 50.54(f)," dated June 28, 1991.
3. SECY-89-170, "Fire Risk Scoping Study: Summary of Results and Proposed Staff Actions," dated June 7, 1989.
4. U. S. Nuclear Regulatory Commission, NUREG-1407, "Procedural and Submittal Guidance for the Individual Plant Examination of External Events (IPEEE) for Severe Accident Vulnerabilities," June 1991.

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November 12, 1999

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

Dear Dr. Travers:

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

During the 466th and 467th meetings of the Advisory Committee on Reactor Safeguards, September 30-October 2 and November 4-6, 1999, we met with representatives of the NRC staff and the Nuclear Energy Institute (NEI) to discuss the status of resolution of issues associated with DG-1093 regarding design bases information. We also discussed the revised Appendix B of NEI 97-04, "Design Bases Program Guidelines," which the staff proposes to endorse in DG-1093 as an acceptable method to meet NRC requirements. We had the benefit of the documents referenced.

Although the staff has not yet updated DG-1093 to reflect agreements reached during meetings with NEI and other industry representatives, we believe that the technical issues have been resolved satisfactorily. We recommend that DG-1093 be issued for public comment. We plan to review the proposed final version of DG-1093 following the reconciliation of public comments.

We commend the staff and NEI for their efforts on this difficult task.

Sincerely,

Dana A. Powers

Dana A. Powers
Chairman

References:
1. Memorandum dated September 24, 1999, from David B. Matthews, Office of Nuclear Reactor Regulation, NRC, to John T. Larkins, Executive Director, Advisory Committee on Reactor Safeguards, transmitting Draft Regulatory DG-1093, "Guidance and Examples for Identifying 10 CFR 50.2 Design Bases."
2. Letter dated October 28, 1999, from Anthony R. Pietrangelo, Nuclear Energy Institute, to David B. Matthews, Office of Nuclear Reactor Regulation, NRC, Subject: Revised Appendix B of NEI 97-04, "Design Bases Program Guidelines."

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

November 17, 1999

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

Dear Chairman Meserve:

SUBJECT: IMPLEMENTING A FRAMEWORK FOR RISK-INFORMED REGULATION IN THE OFFICE OF NUCLEAR MATERIAL SAFETY AND SAFEGUARDS

During the 113th meeting of the Advisory Committee on Nuclear Waste (ACNW), October 12-13, 1999, and the 467th meeting of the Advisory Committee on Reactor Safeguards (ACRS), November 4-6, 1999, the Committees considered the staff's proposed framework for risk-informed and performance-based regulation in the Office of Nuclear Material Safety and Safeguards (NMSS), as articulated in SECY-99-100 and an associated Staff Requirements Memorandum dated June 28, 1999. A meeting of the ACRS/ACNW Joint Subcommittee was held on May 11, 1999, to discuss these matters. We had the benefit of the documents referenced.

Recommendations

1. NMSS should develop a set of principles and a safety goal approach for each of its regulated activities to guide its implementation of risk-informed and performance-based regulation.

2. NMSS should identify the analytical methods to be applied to implement risk-informed and performance-based regulation on an application-specific basis.

Discussion

The NMSS staff is examining the use of risk information in four major categories of regulated activities: (1) long-term commitment of a site to the presence of nuclear material (e.g., high-level waste disposal); (2) use of engineered casks to isolate nuclear material under a variety of conditions (e.g., transportation and storage); (3) physical and chemical processing and possession of nuclear material at a large-scale facility (e.g., fuel fabrication); and (4) use of sealed or unsealed byproduct material in industrial and medical applications. The objectives of this examination are to focus regulatory activities on matters that are important to safety and avoid unnecessary burdens on licensees and the NRC staff.

The diversity of the four categories of activities listed above indicates that the risk assessment methods for material licensees are likely to be different from those for nuclear power plants. While quantitative risk assessment is a well-developed and utilized tool for nuclear power plant licensees, it may be unnecessarily complex for the NMSS regulated activities. The performance assessments (PAs) done for waste repositories are conceptually similar to probabilistic risk assessments (PRAs) for reactors. Recently, there have been developments for simplified approaches to quantitative risk analysis, e.g., integrated safety assessments (ISAs), that are less rigorous than PRAs or PAs.

The staff must address two crucial issues as it considers risk methods in the regulation of material licensees:

1. What criteria should be used to decide whether the regulations for a specific nuclear materials activity should be changed to a risk-informed regulation? Can the current deterministic criteria, accounting methods, or proposed approaches such as ISA accomplish risk-informed objectives?

2. What risk analysis methods (and scope) and risk acceptance criteria should be applied to the operations that merit risk-informed regulation?

To address the first question, we believe that the staff will need to develop a set of principles for risk-informed regulation. Such a set of principles is important to guide the need for and change from a prescriptive form of regulation to a less prescriptive, but risk-informed, method of regulation. In developing these principles, the staff should take full advantage of the knowledge base unique to materials and waste disposal regulation, as well as the staff's experience in developing principles for other regulatory applications, such as Regulatory Guide 1.174.

Some of the characteristics of nuclear materials regulation that differ from reactor regulation include: (1) experience in regulating to radiation exposure standards, as opposed to surrogate measures such as facility damage, (2) diversity of types of licensee activities involving major differences in materials, facilities, and practices, (3) activities not dominated by a clear-cut feature such as core damage, and (4) activities where the operational risk, as opposed to the accident risk, may be the central issue of risk regulation. Although these characteristics distinguish materials regulation from reactor regulation, the Committees believe that the approach to regulatory decisionmaking for the NMSS activities should have a basis that is consistent with the approach for reactor regulation.

An important element introduced in Regulatory Guide 1.174 and that should be investigated in the present context of materials regulation is that regulatory decisionmaking should be based on an analytic and deliberative process. Analytical results from risk assessments and other engineering analyses are only part of the input to this process. Qualitative inputs, e.g., the preservation of the defense-in-depth philosophy, may be considered by an expert panel or other decisionmaking entity. In developing the new principles, the staff should consider this approach and its applicability to the various NMSS activities. If qualitative information is to be used in the decisionmaking process, then the reason(s) should be explained. If there is a need for an expert panel for some activities, its form and composition should be discussed.[1]

Consideration should be given to developing variations on the safety goal approach to risk acceptance. One variation may be to include uncertainty directly in the risk acceptance criteria via required confidence levels in their determination. Another may be to define acceptance criteria that are either met or not, i.e., the range of risk is partitioned into two regions, the acceptable and unacceptable regions. Another might be to adopt a three-region approach. In this concept, there is a range of acceptability with an upper and lower bound. The lower bound constitutes the level below which no further action is required. The upper bound constitutes a level above which definitive action to control the risk is required. The middle region is the region in which cost-benefit tradeoffs can be made. These are a few concepts that should be investigated by the staff for materials regulation. There may be others.

The Committees believe that, just as "guiding principles" are important to establishing a well-founded philosophy of risk-informed regulation, so are certain risk assessment concepts. The representation of risk as a triplet set is such a guiding concept. The triplet consists of accident scenarios (what can go wrong?), probabilities of these scenarios (how likely is each scenario?), and the consequences (what are the consequences?). We view the various risk (or safety) assessment methods that exist in the literature as dealing with these three elements of the risk triplet in different ways. PRAs for reactors and PAs for HLW repositories offer the most complete treatment of the triplet, and they require the most resources. We believe that the staff should clarify how any chosen method deals with the risk triplet (either quantitatively or qualitatively) and justify the appropriateness of the selected scopes as differentiated among the four major categories of NMSS licensees. If methods that are less rigorous than PRAs or PAs are judged to be appropriate for certain applications, their treatment of the triplet should be explicitly identified. The reasons for resorting to these less rigorous methods should be carefully justified. We are especially concerned about the completeness of the scenario list and the analysis of uncertainties.

We look forward to reviewing staff activities on these matters during future meetings.

Sincerely,

B. John Garrick
Chairman, ACNW

Dana A. Powers
Chairman, ACRS

[1] This concept of an expert panel refers to the discussion on integrated decisionmaking in Regulatory Guide 1.174. The purpose of such an expert panel is to evaluate multiple sources of information to make decisions in an integrated manner. This is different from the guidance in the "Branch Technical Position on the Use of Expert Elicitation in the High-Level Radioactive Waste Program," NUREG-1563, that refers to a specific formalized process for developing information and "data" to be used in a performance assessment.

References:

1. Memorandum dated June 28, 1999, from Annette Vietti-Cook, Secretary, NRC, to William D. Travers, Executive Director for Operations, NRC, and John T. Larkins, Advisory Committee on Reactor Safeguards, Subject: Staff Requirements - SECY-99-100 - Framework for Risk-Informed Regulation in the Office of Nuclear Material Safety and Safeguards.
2. 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 Regulation in the Office of Nuclear Material Safety and Safeguards.
3. Memorandum dated February 24, 1999, from Annette Vietti-Cook, Secretary, NRC, to William D. Travers, Executive Director for Operations, NRC, Subject: Staff Requirements - SECY-99-144 - White Paper on Risk-Informed and Performance-Based Regulation.
4. Report dated April 19, 1999, from Dana A. Powers, Chairman, Advisory Committee on Reactor Safeguards, to Shirley Ann Jackson, Chairman, NRC, Subject: Status of Efforts on Revising the Commission's Safety Goal Policy Statement.
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.
6. U.S. Nuclear Regulatory Commission, NUREG-1563, "Branch Technical Position on the Use of Expert Elicitation in the High-Level Radioactive Waste Program," November 1996.

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

December 7, 1999

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 50.47, "EMERGENCY PLANS," RELATING TO A REEVALUATION OF POLICY ON THE USE OF POTASSIUM IODIDE (KI) FOR THE GENERAL PUBLIC AFTER A SEVERE ACCIDENT AT A NUCLEAR POWER PLANT

During the 468th meeting of the Advisory Committee on Reactor Safeguards, December 2-4, 1999, the Committee considered the proposed final amendment to 10 CFR 50.47 and decided not to review it.

Reference:
Memorandum dated November 15, 1999, from Roy P. Zimmerman, Office of Nuclear Reactor Regulation, NRC, to John T. Larkins, Executive Director, ACRS, Subject: Final Rule Change to 10 CFR 50.47 Relating to the Use of Potassium Iodide (KI) for the General Public.

cc: A. Vietti-Cook, SECY
J. Blaha, OEDO
W. Ott, OEDO
S. Collins, NRR
M. Case, NRR
M. Jamgochian, NRR

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

December 8, 1999

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

Dear Chairman Meserve:

SUBJECT: DRAFT COMMISSION PAPER REGARDING THE 120-MONTH UPDATE REQUIREMENT FOR INSERVICE INSPECTION AND INSERVICE TESTING PROGRAMS

During the 468th meeting of the Advisory Committee on Reactor Safeguards, December 2-4, 1999, we reviewed the options proposed by the staff regarding the current requirement for licensees to update inservice inspection (ISI) and inservice testing (IST) programs every 120 months to the most recent Edition of the American Society of Mechanical Engineers (ASME) Code incorporated by reference in 10 CFR 50.55a, "Codes and Standards." Our Subcommittee on Materials and Metallurgy also reviewed this matter during its meeting on December 1, 1999. During this review, we had the benefit of discussions with representatives of the NRC staff, ASME, and the Nuclear Energy Institute (NEI). We also had the benefit of the documents referenced.

Recommendation

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

Background

The staff issued a proposed amendment to 10 CFR 50.55a on April 27, 1999, to solicit public comment on a proposal to eliminate the current requirement that licensees update their ISI and IST programs every 120 months to the most recent edition of the ASME Code incorporated by reference in 10 CFR 50.55a. In a letter dated April 19, 1999, we recommended against eliminating this requirement. The NRC staff held a public workshop on May 27, 1999, to discuss the update requirement. In a Staff Requirements Memorandum dated June 24, 1999, the Commission directed the staff to evaluate public comments on the update requirement and develop options and recommendations on the retention or elimination of this requirement. The Commission also directed the staff to discuss this issue further with the ACRS.

The staff has identified three options:

OPTION 1: Replace the 120-month ISI/IST update requirement with a baseline of ISI and IST requirements, and allow voluntary updating to subsequent NRC-approved Code editions and addenda unless the baseline is revised based on 10 CFR 50.109, where the initial baseline will consist of:

Option 1.A. the 1989 Edition of the ASME Code for ISI of Code Class 1, 2, and 3 components (including supports) and for IST of Code Class 1, 2, and 3 pumps and valves; the 1992 Edition with the 1992 Addenda of Subsections IWE and IWL of the ASME Code for ISI of Class MC and Class CC components and their integral attachments; the 1995 Edition with the 1996 Addenda of Appendix VIII of the ASME Code, Section XI, with limitations and modifications specified in 10 CFR 50.55a,

Option 1.B. the 1995 Edition with the 1996 Addenda of the ASME Code with the limitations and modifications specified in the NRC regulations, or

Option 1.C. a later version (e.g., the 1998 Edition) of the ASME Code with appropriate limitations and modifications.

OPTION 2: Retain the current 120-month ISI/IST update requirement.

OPTION 3: Authorize plant-specific alternatives to the 120-month ISI/IST update requirement.

Discussion

The staff evaluated the update options in terms of the strategic goals of the Commission: (1) maintaining safety, (2) increasing public confidence, (3) reducing unnecessary regulatory burden, and (4) making NRC activities and decisions more effective, efficient, and realistic. Although the staff concludes that no particular option has an overwhelming advantage over the other options, it recommends the adoption of Option 1B, which eliminates the mandatory 120-month update. We believe that the later version of the ASME Code would provide technically superior baselines for the ISI and IST programs than the 1989 Edition, which is now over ten years old.

We agree with the conclusion of the staff that any of the options will maintain an acceptable level of safety. Each option purports to include provisions to update ISI and IST programs, although the criteria to require updating differ among the options. Furthermore, the analyses performed in support of the development of risk-informed inspections for Class 1, 2, and 3 piping and those done to support resolution of Generic Safety Issue (GSI)-190 show that ISI has a relatively modest impact on core damage frequency (CDF). We have not reviewed the analyses done to support risk-informed IST programs, but we believe that they would probably also show relatively modest impacts on CDF. This is not surprising. Because failures of these components were anticipated in the design of nuclear power plants, effective mitigation systems and procedures have been developed. However, because assurance of the integrity the reactor coolant pressure boundary and the containment is one of the cornerstones of the NRC regulatory system, ISI and IST programs have been required to provide additional assurance, through application of the defense-in-depth philosophy, of the integrity of these barriers and to compensate for uncertainties.

NEI and the staff argue in support of Option 1 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 and that developments in technology and operating experience could lead to additional changes in the inspection programs. Changes are not introduced in the ASME Code requirements frivolously. The ASME Code represents the consensus of a broad-based group of experts that includes strong utility representation (approximately 30% of the Section XI membership) as well as representation from manufacturers, vendors, the NRC, and other engineering and consulting organizations.

Under Option 1, any mandated updates to the ISI and IST programs would have to pass the 10 CFR 50.109 backfit criteria. The 50.109 evaluation is not well suited to assess the appropriateness of defense-in-depth requirements, which are intended to address uncertainties that are difficult to quantify. In our May 19, 1999 report, we outlined an approach for developing a systematic methodology for the evaluation of defense in depth; however, lacking such a methodology at the present time, decisions on defense in depth will have to be based on judgment. The collective judgment of the broad-based group of experts represented by the ASME Code should be reflected in the inspection requirements.

Sincerely,

Dana A. Powers

Dana A. Powers
Chairman

References:

1. Memorandum dated November 18, 1999, from William D. Travers, Executive Director for Operations, NRC, for the Commissioners, SECY-99-XXX, Subject: 120-Month Update Requirement for Inservice Inspection and Inservice Testing Programs (Predecisional Draft).
2. ACRS letter dated May 19, 1999, from Dana A. Powers, Chairman, ACRS, to Honorable Shirley A. Jackson, Chairman, NRC, Subject: The Role of Defense In Depth in a Risk-Informed Regulatory System.
3. Memorandum dated June 24, 1999, from Annette L. Vietti-Cook, Secretary, NRC, to William D. Travers, Executive Director for Operations, NRC, Subject: Staff Requirements - Reconsideration of SECY-99-017 (Proposed Amendment to 10 CFR 50.55a).
4. 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."
5. 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, 11/1/99.
6. Memorandum dated November 12, 1999, from Ashok C. Thadani, Director, Office of Nuclear Regulatory Research, NRC, to John T. Larkins, Executive Director, Advisory Committee on Reactor Safeguards, 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

December 10, 1999

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 CALVERT CLIFFS NUCLEAR POWER PLANT, UNITS 1 AND 2

During the 468th meeting of the Advisory Committee on Reactor Safeguards, December 2-4, 1999, we completed our review of the Baltimore Gas and Electric Company's (BGE's) application for license renewal of the Calvert Cliffs Nuclear Power Plant (CCNPP), Units 1 and 2 and the related Final Safety Evaluation Report (FSER). Our review included four meetings with the staff and the applicant concerning the license renewal of CCNPP and two meetings with the staff and the Nuclear Energy Institute concerning generic license renewal issues. During this review, we had the benefit of discussions with representatives of the NRC staff and BGE. We also had the benefit of insights gained from our review of another license renewal application and of the documents referenced. We provided an interim letter, dated May 19, 1999, concerning the BGE application.

Conclusion

On the basis of our review of BGE's application, the FSER, and the resolution of the open and confirmatory items identified in the Safety Evaluation Report (SER), we conclude that BGE has properly identified the structures, systems, and components (SSCs) that are subject to aging management programs. Furthermore, we conclude that the programs instituted to manage aging-related degradation of the identified SSCs are appropriate and provide reasonable assurance that Calvert Cliffs Nuclear Power Plant, Units 1 and 2 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. BGE requested renewal of the operating licenses for the CCNPP, Units 1 and 2 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

BGE, 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. The staff also conducted onsite inspections to verify the implementation of these programs.

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

The commitments made by BGE are adequate to resolve the open and confirmatory items. Several of the open items such as the effects of the reactor coolant environment on fatigue life and the thermal fatigue of American Society of Mechanical Engineers (ASME) Class 1 small-bore piping may have generic implications for other applications for license renewal.

BGE committed to the implementation of a plant-specific monitoring program in which it will use correlations published in NUREG/CR-5704 to calculate the effects of the reactor coolant environment on 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 BGE's proposed program is an acceptable plant-specific approach for the resolution of GSI-190 concerns.

BGE resolved an open item concerning cracking of ASME Class 1 small-bore piping by including small-bore piping in the CCNPP's age-related degradation inspection (ARDI) program. Under the ARDI program, inspections of small-bore piping will be performed during the last five years of the current license term. The timing of these inspections is appropriately set late in the current licensing period so that they will be most useful for assessing the need for additional requirements. We concur with the resolution of this open item.

Another open item concerned the adequacy of the bases provided to justify the use of one-time inspections to resolve some potential aging issues. The staff has accepted one-time inspections prior to the end of the current license term, rather than regular, periodic inspections, in those cases in which age-related degradation is not expected to occur. In such cases, the one-time inspection is intended to confirm the expectation that age-related degradation is not occurring, or that its effects are insignificant. We agree that this is an appropriate approach to address such aging issues. We reviewed the basis for the staff's acceptance of one-time inspections in individual cases (SER open Item 3.1.6.3-1) and concur with the staff's determination.

During our meeting, BGE informed us that it expects to conduct most of the one-time inspections after 30 years of plant operation. We believe that it is important that these one-time inspections be performed late in the current license term (the last ten years).

After the SER was issued, the staff identified void swelling as a potential mode of degradation for pressurized water reactor vessel internals. BGE committed to participate in the industry programs to address the significance of void swelling and to develop an inspection program if needed.

As CCNPP, Units 1 and 2 age, inspection and operating experience may prompt significant adjustments to their aging management programs. BGE is required to document in its FSAR that the 10 CFR Part 50 Appendix B quality assurance program also applies to those nonsafety-related SSCs which are subject to an aging management review. Furthermore, the staff has required that BGE include in its FSAR the license renewal application commitments that the staff relied on to conclude that aging effects will be adequately managed for the period of extended operation. These steps ensure that future changes can be controlled under the 10 CFR 50.59 process. Future schedule changes will require license amendments if the schedules are delayed.

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

We believe that the applicant and the staff have identified possible aging mechanisms associated with passive long-lived components. Adequate programs have been established to manage the effects of aging so that CCNPP, Units 1 and 2 can be operated safely in accordance with their licensing basis for the period of the extended license.

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

Sincerely,

Dana A. Powers

Dana A. Powers
Chairman

References:

1. Letter dated November 16, 1999, from Christopher I. Grimes, Office of Nuclear Reactor Regulation, NRC, to Charles H. Cruse, Baltimore Gas and Electric Company, Subject: Final Safety Evaluation Report.
2. Letter dated May 19, 1999, from Dana A. Powers, Chairman, ACRS, to William D. Travers, Executive Director for Operations, NRC, Subject: Interim Letter on the Safety Aspects of the Baltimore Gas and Electric Company's License Renewal Application for Calvert Cliffs Nuclear Power Plant, Units 1 and 2.
3. U. S. Nuclear Regulatory Commission, NUREG/CR-5704, "Effects of LWR Coolant Environments on Fatigue Design Curves of Austenitic Stainless Steels," April 1999.
4. U. S. Nuclear Regulatory Commission, "Safety Evaluation Report Related to the License Renewal of Calvert Cliffs Nuclear Power Plant, Units 1 and 2," March 1999.
5. Letter dated April 8, 1998, from Charles H. Cruse, Baltimore Gas and Electric Company, to U. S. Nuclear Regulatory Commission Document Control Desk, Subject: Calvert Cliffs Nuclear Power Plant Unit Nos. 1 and 2, Application for License Renewal.

6. U. S. Nuclear Regulatory Commission, Code of Federal Regulations, 10 CFR Part 54, "Requirements for Renewal of Operating Licenses for Nuclear Power Plants."
7. U. S. Nuclear Regulatory Commission, Code of Federal Regulations, 10 CFR Part 50, Appendix B, "Quality Assurance Criteria for Nuclear Power Plants and Fuel Reprocessing Plants."

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

December 10, 1999

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

Dear Dr. Travers:

SUBJECT: PROPOSED RESOLUTION OF GENERIC SAFETY ISSUE-190, "FATIGUE EVALUATION OF METAL COMPONENTS FOR 60-YEAR PLANT LIFE"

During the 468th meeting of the Advisory Committee on Reactor Safeguards, December 2-4, 1999, we reviewed the proposed resolution of Generic Safety Issue (GSI)-190, "Fatigue Evaluation of Metal Components for 60-Year Plant Life." During our review, we had the benefit of discussions with representatives of the NRC staff and of the documents referenced.

RECOMMENDATIONS

- We agree with the staff's proposal that GSI-190 be resolved without any additional regulatory requirements.
- The staff should ensure that utilities requesting license renewal consider the management of environmentally assisted fatigue in their aging management programs.

BACKGROUND

The effects of fatigue for the 40-year initial reactor license period were studied and resolved under GSI-78, "Monitoring of Fatigue Transient Limits for Reactor Coolant System," and GSI-166, "Adequacy of Fatigue Life of Metal Components."

The staff concluded that risk from fatigue failure of components in the reactor coolant pressure boundary was very small for 40-year plant life. In our March 14, 1996 letter, we agreed with the staff's conclusion.

GSI-190 was established to address the residual concerns of GSI-78 and GSI-166 regarding the environmental effects of fatigue on pressure boundary components for 60-years of plant operation. The scope of GSI-190 included design-basis fatigue transients, studying the probability of fatigue failure and its effects on core damage frequency (CDF) of selected metal components for 60-year plant life.

DISCUSSION

Resolution of GSI-190 was based on the results of an NRC-sponsored study performed by the Pacific Northwest National Laboratory (PNNL). In that study, PNNL examined design-basis fatigue transients and the probability of fatigue failure of selected metal components for 60-year plant life and the resulting effects on CDF.

The PNNL study showed that some components have cumulative probabilities of crack initiation and through-wall growth that approach unity within the 40- to 60-year period. The maximum failure rate (through-wall cracks per year) was in the range of 10^{-2} per year, and those failures were associated with high cumulative usage factor locations and components with thinner walls, i.e., pipes more vulnerable to through-wall cracks. There was only a modest increase in the frequency of through-wall cracks in major reactor coolant system components having thicker walls. In most cases, the leakage from these through-wall cracks is small and not likely to lead to core damage. Therefore, the projected increased frequency in through-wall cracks between 40- and 60-years of plant life does not significantly increase CDF. Based on the low contributions to CDF, we agree with the proposed resolution of GSI-190.

Environmentally assisted fatigue degradation should be addressed in aging management programs developed for license renewal. Minimization of leakage is important for operational safety, occupational doses, and for continued economic viability of the plants.

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

Sincerely,

Dana A. Powers

Dana A. Powers
Chairman

References:

1. Memorandum dated November 12, 1999, from Ashok C. Thadani, Director, Office of Nuclear Regulatory Research, NRC, to John T. Larkins, Executive Director, Advisory Committee on Reactor Safeguards, Subject: Generic Safety Issue-190, "Fatigue Evaluation of Metal Components for 60-Year Plant Life."
2. Letter dated March 14, 1996, from T. S. Kress, Chairman, Advisory Committee on Reactor Safeguards, to James M. Taylor, Executive Director for Operations, NRC, Subject: Resolution of Generic Safety Issue-78, "Monitoring of Fatigue Transient Limits for the Reactor Coolant System."
3. Letter dated October 16, 1995, from T. S. Kress, Chairman, Advisory Committee on Reactor Safeguards, to Shirley Ann Jackson, Chairman, NRC, Subject: Fatigue Action Plan.

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

December 15, 1999

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

Dear Dr. Travers:

SUBJECT: NUREG-1624, REVISION 1, "TECHNICAL BASIS AND IMPLEMENTATION GUIDELINES FOR A TECHNIQUE FOR HUMAN EVENT ANALYSIS (ATHEANA)"

During the 468th meeting of the Advisory Committee on Reactor Safeguards, December 2–4, 1999, we reviewed Revision 1 of NUREG-1624, "Technical Basis and Implementation Guidelines for A Technique for Human Event Analysis (ATHEANA)." Our Subcommittee on Human Factors also reviewed this document on November 19, 1999. 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 objective of ATHEANA is to develop a methodology that: (a) allows a realistic, qualitative analysis of potential accident sequences and past incidents involving human actions and (b) allows a realistic evaluation of the probabilities of unsafe human actions for inclusion in probabilistic risk assessments (PRAs). The qualitative evaluation in NUREG-1624, Revision 1, is at an advanced stage of development and is achieving its purpose. The quantitative portion still needs significant development.

2. ATHEANA's focus on the context within which the operators must act as well as on the error mechanisms is an appropriate paradigm shift away from a focus on "human error."

3. ATHEANA deals with operator actions that take place after an abnormal event has occurred, e.g., a fire or an initiating event, as defined in PRAs. Its scope should be extended to include normal activities that may cause a plant event.

4. The term "error-forcing context" is not used consistently and is misleading in some situations. An alternative, more descriptive term must be defined.

5. The process of searching for error-forcing contexts is complex. Not all human actions require such a detailed treatment, and a screening process should be developed to identify the level of analysis that a given situation requires. The development of the screening process should be given priority.

6. In developing symptom-based procedures, the industry considered many deviations from expected plant behavior. The ATHEANA search process for deviations should take advantage of this experience.

7. The elements of a plant's safety culture that influence the operators when they are faced with a decisionmaking situation should be explicitly considered when evaluating the error-forcing contexts.

8. The application of ATHEANA to a fire-initiated accident scenario does not make clear its advantages over existing, less complex methods. More examples of applications need to be developed.

Discussion

Understanding human errors and evaluating their probability of occurrence have been active areas of research since the Three Mile Island accident. "First-generation" models, i.e., those developed in the 1970s and 1980s, varied in their depth of modeling human performance. No serious attempt was made to incorporate concepts from the behavioral and cognitive sciences into these models. The focus was on "human error" with its connotation of blame.

In the late 1980s, a need for "second-generation" models that would delve deeper into the causes of human error was recognized. Attention shifted toward an examination of contextual elements that could trigger cognitive error mechanisms which could lead to unsafe crew actions. ATHEANA is the first major effort to develop a model for human performance based on this new paradigm. We believe that this shift in paradigm is appropriate and commend the staff for carrying out this work.

ATHEANA focuses on the analysis of human performance after a plant event. This is natural, since this has been the main perceived need for improving human reliability analysis. Errors made during routine activities, such as maintenance and testing, are analyzed satisfactorily by using the methods of the human reliability handbook (NUREG/CR-1278, Revision 1). Normal plant activities that may lead to plant events, such as the reactor coolant drain-down event at the Wolf Creek Generating Station, Unit 1, on September 19, 1994, are not currently addressed by ATHEANA.

The principal premise of ATHEANA is that "plant conditions" and "performance-shaping factors" may produce an "error-forcing context" that could trigger an error mechanism such as the refusal to change an initial misdiagnosis when contradictory evidence is received. The performance-shaping factors reflect human-centered influences such as training and communications.

The search for error-forcing contexts is a major effort. A multidisciplinary team consisting of human-reliability experts, plant operators, PRA specialists, and possibly others is needed. Such an extensive effort is not appropriate for all potentially unsafe human acts. We are concerned that the amount of resources required may discourage practitioners from even attempting to use ATHEANA. We believe that a set of screening guidelines should be developed to define different levels of treatment for various unsafe human acts. The qualitative insights gained from the detailed ATHEANA investigations should form the basis for the development of simpler methods for use when appropriate.

We note that a similar situation arises when a decision must be made about the methodology to be used to elicit and utilize expert opinions in probabilistic seismic hazard analysis (NUREG/CR-6372). In some situations of great national interest in which the uncertainties are large, a very formal methodology that is implemented by a multidisciplinary team is required. In other situations, experience has shown that a single technical integrator using informal input from experts is sufficient.

The process of searching for error-forcing contexts starts with a base-case scenario that describes the expected plant and operator behavior for a given initiator. The error-forcing contexts are, then, identified by searching for deviations from the base-case scenario. A great deal of work along these lines was done when the industry developed symptom-based emergency operating procedures. We believe that ATHEANA should take advantage of this experience.

ATHEANA defines an error-forcing context as "the combined effect of PSFs [performance-shaping factors] and plant conditions that create a situation in which human error is likely." Yet, in Chapter 10 of NUREG-1624, Revision 1, it is stated that an error-forcing context may be "so noncompelling that there is no increased likelihood of the UA [unsafe act] compared with the routine PRA context." We believe that the use of clear, accurate terminology is essential, especially when concepts from the behavioral sciences are brought into the practice of engineering. We believe that an alternative terminology should be developed to replace "error-forcing context."

The error mechanisms are developed from a cognitive model that consists of detection, situation assessment, response planning, and response implementation. All of these activities involve decisions that the plant crew must make, especially in the response planning phase. Although the discussion of error mechanisms clearly assumes that decisions are being made, e.g., establishing wrong goals is identified as a possible error, no formal attempt is made to investigate either the decisionmaking process or the impact of time. The decisionmaking processes (as well as the error-forcing contexts) are expected to be different for event sequences that evolve in a relatively short time, e.g., in less than about 30 minutes, and for sequences taking place over longer periods. In addition, decisionmaking may involve balancing conflicting safety and economic objectives; therefore, the plant's safety culture is a critical element in these decisions. Safety culture should be explicitly considered when evaluating the error-forcing context.

The application of ATHEANA to a fire-initiated accident scenario failed to convince us that the results obtained were sufficiently better than those obtained through other, presumably less resource-intensive methods to justify the use of ATHEANA . There are some inconsistencies between this application and the theoretical development in NUREG-1624, Revision 1. For example, the error-forcing contexts that the methodology claims are its foundation were not identified explicitly. We believe that a number of applications are urgently needed to convince the human reliability community and the end users that ATHEANA is a practical model that represents an improvement over existing models. These applications will also serve to guide the development of the screening process that we mentioned above.

A major motivation for the development of ATHEANA is the need for adequate models to support risk-informed regulatory applications. The guidance provided currently for evaluating the probabilities of unsafe human acts is very general. The HEART model (NUREG-1624, Revision 1, Chapter 10), whose quantitative results are proposed as one way for assessing the probability of a given error-forcing context, was developed several years before the ATHEANA project started and there is no effort to adapt it to ATHEANA. If the HEART model is to form the basis for quantifying the error-forcing context in the ATHEANA process, then ATHEANA should include sufficient information to assess the appropriateness of using this model for such purpose.

We acknowledge that any attempt at quantifying probabilities of error-forcing contexts will necessarily involve expert judgment. However, the guidance given by ATHEANA does not build on the large amount of work that has been done on the elicitation and utilization of expert opinions, e.g., in NUREG-1150, NUREG/CR-6372, and NUREG/CR-3518.

A more serious effort on probability evaluation will also help in developing the screening process that we recommended above. We expect that a lot of the details that are now investigated in the analysis of plant conditions, performance-shaping factors, and error mechanisms will not affect the quantification process, thus suggesting ways for limiting the qualitative investigation. While we recognize that the likelihood of plant conditions can be estimated, we believe that the probabilities of performance-shaping factors are much more difficult to evaluate. Thus, ATHEANA must demonstrate the feasibility of evaluating probabilities of error-forcing contexts, of which the performance-shaping factors are an important component.

We believe that the development of the screening process and the application of ATHEANA to several realistic accident scenarios are critical to its success. We look forward to working with the staff on these matters.

Sincerely,

Dana A. Powers

Dana A. Powers
Chairman

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