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

September 28, 2000

MEMORANDUM TO: Annette L. Vietti-Cook
Secretary

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

SUBJECT: ACRS MEETING WITH THE NRC COMMISSIONERS - OCTOBER 6, 2000

The ACRS is scheduled to meet with the NRC Commissioners on October 6, 2000, between 9:30 and 11:30 a.m. to discuss the following items. Presentation materials related to these items are attached.

TOPICS	PRESENTER	PRESENTATION TIME
I. Introduction	Dr. R. Meserve NRC Chairman	9:30 - 9:35 a.m.
II. ACRS Presentation		
1. Overview: Topics and Near-Term Activities (pp. 1-4)	Dr. D. Powers, ACRS Chairman	9:35-9:45 a.m.
2. Risk-Informing 10 CFR Part 50 (pp. 5-16)	Dr. W. Shack	9:45-10:10 a.m.
- NEI letter		
- Proposed revision to 10 CFR 50.44 and ANPR for 10 CFR 50.69 and Appendix T		
3. Quality of PRAs (pp. 17-31)	Dr. G. Apostolakis	10:10-10:35 a.m.
- ASME PRA Standard		
- Assessment of the Quality of PRAs		

4. Spent Fuel Pool Fire Safety (pp. 32-47)	Dr. T. Kress	10:35-11:00 a.m.
5. More Realistic (Best Estimate) Thermal-Hydraulic Codes (pp. 48-65)	Dr. G. Wallis	11:00-11:20 a.m.
6. Status of ACRS Activities on License Renewal (pp. 66-84)	Dr. M. Bonaca	11:20-11:25 a.m.
7. Closing Remarks	Dr. D. Powers ACRS Chairman	11:25-11:30 a.m.
III. Adjournment	NRC Chairman	11:30

Attachments: As stated

cc: ACRS Members
ACRS Staff

MEETING OF THE
ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
WITH THE
U.S. NUCLEAR REGULATORY COMMISSION
OCTOBER 6, 2000

OVERVIEW
DR. DANA A. POWERS, ACRS CHAIRMAN

TOPICS

- Risk-Informing 10 CFR Part 50 — Shack
  - A first application: H_2 control and management
- Quality of PRAs — Apostolakis
  - An essential process for the practical use of risk information in plant-specific regulatory activities

TOPICS (continued)

- Spent Fuel Pool Fire Safety — Kress
    - A first step in the development of risk-informed requirements for decommissioning plants
- More Realistic (Best-Estimate) Thermal-Hydraulic Codes — Wallis
    - Assuming greater importance as industry moves toward more realistic analyses
- Status of ACRS Activities on License Renewal — Bonaca

SOME NEAR-TERM FUTURE ACTIVITIES OF ACRS

- Report to the Commission on NRC Safety Research

- DPO panel on steam generator tube integrity

- Synergisms among changes in nuclear power plants

    - Higher burnup fuel
    - Power uprates
    - Best-estimate or more realistic analyses
    - Plant life extension

Is there competition for the safety margins in plants?

UNITED STATES NUCLEAR REGULATORY COMMISSION

ACRS MEETING WITH THE NRC COMMISSIONERS

OCTOBER 6, 2000

RISK-INFORMING 10 CFR PART 50

DR. WILLIAM J. SHACK
ACRS

- ACRS met in July and September 2000.

- Subcommittee on Reliability and PRA met on June 29 and July 11, 2000, to discuss:

  - NEI letter dated January 19, 2000.

  - Proposed risk-informed revisions to 10 CFR 50.44 concerning combustible gas control systems and associated framework (Option 3) for evaluating the technical requirements of candidate regulations.

  - Public comments on Advance Notice of Proposed Rulemaking for 10 CFR 50.69 and Appendix T (Option 2) concerning special treatment requirements.

- ACRS report dated July 20, 2000, concerning NEI letter dated January 19, 2000.

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

    - Acknowledged that there may be benefits in reconsidering changes in the definition of challenges to the emergency core cooling systems (ECCS), i.e., replacement of the double ended guillotine break, with an alternative large-break loss-of-coolant accident. Noted that it is not clear that substantial changes can be made in terms of success criteria.

- ACRS report dated September 13, 2000, concerning proposed revision to 10 CFR 50.44 and related matters.

  - Agreed that there is little or no safety benefit associated with some of the requirements in the current 10 CFR 50.44 and these constitute unnecessary regulatory burden.

  - Recommended that the staff be directed to proceed with rulemaking on 10 CFR 50.44.

  - Discussion of how risk information was used to develop the results of conditional large release probability should be expanded.

- Results of the Option 3 study should assist the staff in the disposition of the petition for rulemaking.

- Plan to review revisions to the proposed framework document.

- Plan to review the proposed rulemaking associated with special treatment requirements for structures, systems, and components (SSCs).

* Plan to review proposed risk-informed revisions to 10 CFR 50.46 on ECCS in December 2000.

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

July 20, 2000

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

Dear Chairman Meserve:

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

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

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

Recommendations

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

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

Background

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

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

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

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

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

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

In the SRM of June 8, 1999, the Commission approved proceeding with the current rulemaking in Option 1, implementing Option 2, and proceeding with a study of Option 3. For Option 3, the Commission requested that the staff determine how best to proceed and provide a detailed plan outlining its recommendations regarding specific regulatory changes that should be pursued. 11
SECY-99-256 provides the staff's plans for implementing Option 2. SECY-99-264 provides the staff's plans with respect to the Commission request to proceed with a study of Option 3.

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

<u>Discussion</u>

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

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

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

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

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

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

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

Sincerely,

Dana A. Powers

Dana A. Powers
Chairman

References:

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

13

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

September 13, 2000

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

Dear Chairman Meserve:

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

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

<u>Background</u>

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

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

<u>Conclusions and Recommendations</u>

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

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

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

Discussion

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

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

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

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

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

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

Sincerely,

Dana A. Powers
Chairman

References:

1. Draft memorandum received August 18, 2000, from William D. Travers, Executive Director for Operations, NRC, to The Commissioners, Subject: Status Report on Risk-Informing the Technical Requirements of 10 CFR Part 50 (Option 3) and Recommendations on Risk-Informing 10 CFR 50.44 (Combustible Gas Control). 16
2. Memorandum dated February 3, 2000, from Annette Vietti-Cook, Secretary, NRC, to William D. Travers, Executive Director for Operations, NRC, Subject: Staff Requirements - SECY-99-264 - Proposed Staff Plan for Risk-Informing Technical Requirements of 10 CFR Part 50.
3. Letter dated April 18, 2000 from Steven D. Floyd, Nuclear Energy Institute, to Thomas L. King, Office of Nuclear Regulatory Research, NRC, Subject: Industry Comments on Draft NRC Framework for Risk-Informing NRC Technical Requirements, and Draft NRC Report on Risk-Informing 10 CFR 50.44, "Standards for Combustible Gas Control Systems in Light-Water-Cooled Power Reactors."
4. Report dated October 12, 1999, from Dana A. Powers, Chairman, ACRS, to Greta Joy Dicus, Chairman, NRC, Subject: Proposed Plans for Developing Risk-Informed Revisions to 10 CFR Part 50, "Domestic Licensing of Production and Utilization Facilities."
5. Report dated July 20, 2000, from Dana A. Powers, Chairman, ACRS, to Richard A. Meserve, Chairman, NRC, Subject: Nuclear Energy Institute Letter dated January 19, 2000, Addressing NRC Plans for Risk-Informing the Technical Requirements in 10 CFR Part 50.
6. U. S. Nuclear Regulatory Commission NUREG-1150, Vols. 1-3, "Severe Accident Risks: An Assessment for Five U.S. Nuclear Power Plants," Final Report, December 1990.
7. U. S. Nuclear Regulatory Commission NUREG-1560, Vols. 1-5, "Individual Plant Examination Program: Perspectives on Reactor Safety and Plant Performance," Final Report, December 1997.

ACRS MEETING WITH THE NRC COMMISSIONERS

OCTOBER 6, 2000

QUALITY OF PROBABILISTIC RISK ASSESSMENTS

DR. GEORGE E. APOSTOLAKIS
ACRS

BACKGROUND

- ACRS has reviewed the ASME PRA Standard (Report dated July 20, 2000).

- ACRS has reviewed SECY-00-0162 (Report dated September 7, 2000).

- On October 5, 2000, ACRS reviewed:

  - Industry certification process in NEI 00-02.

  - Staff's views on the ASME PRA Standard.

OBSERVATIONS

- PRA is ambitious. It models the whole plant including hardware failures, human performance, and relevant physical phenomena.

- The most complete PRA is Level 3, and includes all modes of operation as well as rigorous uncertainty and sensitivity analyses.

- Numerous regulatory decisions do not need a complete PRA, while others may require additional analyses.

OBSERVATIONS (continued)

- The integrated decision-making process of Regulatory Guide (RG) 1.174 supplements PRA limitations in the assessment of core damage frequency (CDF) and large, early release frequency (LERF) with engineering analyses and informed engineering judgment.

- Defining a "good-enough" PRA for a specific application *a priori* is a highly subjective and very difficult task, given the varied nature of potential risk-informed decisions.

CONCLUSIONS

- Regarding the ASME PRA Standard, ACRS concluded that:
  - The proposed Standard is not a traditional "design-to" engineering standard or procedures guide.
  - Staff will still need to make a case-by-case assessment of the adequacy of PRAs.
  - The differences among the categories should be delineated more clearly, especially the treatment of uncertainties.

CONCLUSIONS (continued)

- The discussion of the categories of requirements needed for particular regulatory applications can be misleading and should be deleted.

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

CONCLUSIONS (continued)

- Regarding SECY-00-0162, ACRS concluded that:
  - Staff should continue with the current process and the quality of PRA must be judged in the context of the regulatory decision the PRA supports.
  - Attachment 1, "PRA Scope and Tehnical Attributes," is a useful high-level tutorial exposition on PRA technical attributes. It is not intended to be a design-to standard.
  - The staff should augment its collection of examples of risk-informed decisions and the requisite PRA quality to include a more diverse set of examples and provide more details on how risk information was used (Attachment 2). Common themes and frequently asked questions should be identified.

MOVING FORWARD

- Focus on points of agreement, rather than disagreement. Arguing about categories of PRA applications delays the process of developing standards to facilitate risk-informed regulatory decisions.

- Recognize that reviews will be required, and focus on facilitating them.

- The staff's views, as expressed in Attachment 1 of SECY-00-0162, are consistent with the Category III (and to a large extent with Category II) requirements of the ASME Standard.

MOVING FORWARD (continued)

- Develop a standard that would be based on these agreed-upon requirements for a baseline PRA and would be consistent with RG 1.174 requirements.

- Continue with the collection of case studies ("bottom-up" approach) to develop guidance regarding the role and quality of the risk information utilized in specific applications.

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

July 20, 2000

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

Dear Dr. Travers:

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

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

26

<u>Conclusions and Recommendations</u>

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

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

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

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

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

Discussion

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

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

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

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

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

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

Sincerely,

Dana A. Powers

Dana A. Powers
Chairman

<u>References</u>:

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

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

September 7, 2000

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

Dear Chairman Meserve:

SUBJECT: ASSESSMENT OF THE QUALITY OF PROBABILISTIC RISK ASSESSMENTS

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

<u>Conclusions and Recommendations</u>

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

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

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

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

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

29

Discussion

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

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

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

30

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

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

Sincerely,

Dana A. Powers
Chairman

<u>References</u>:

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

UNITED STATES NUCLEAR REGULATORY COMMISSION

ACRS MEETING WITH THE NRC COMMISSIONERS

OCTOBER 6, 2000

SPENT FUEL POOL FIRE SAFETY

DR. THOMAS S. KRESS
ACRS

BACKGROUND

- Purpose: The staff's intention was to develop a rule instead of having to deal with a number of exemption requests.

- Issue: After how long a time is the risk of a spent fuel fire sufficiently low to be acceptable enough to relax requirements?

STAFF'S ACCEPTANCE CRITERIA

- Spent fuel fire equated to a LERF because pools are outside containment.

- LERF acceptance criterion in RG 1.174 was used (i.e., 10^{-5}/reactor-yr).

- After some elapsed time (years), decay heat insufficient to "ignite" the fire (lowest ignition temperature used was 1555°K)

TWO MAJOR ACRS CONCERNS

- The RG 1.174 LERF acceptance criterion may not be appropriate for spent fuel fires.
  - Derived from the prompt fatality safety goal using a source term appropriate for steam-oxidation driven accidents to cores.
  - Air-oxidation accidents are expected to produce significantly different source terms than do steam-oxidation driven accidents.

TWO MAJOR ACRS CONCERNS (continued)

- The "ignition" temperature does not recognize the significant hydridization of the clad.
  - Breakaway oxidation kinetics may be faster and more energetic.
  - Oxidation energetics significantly greater
  - Likely much lower "ignition" temperature

CONCLUSIONS

- A large release from a spent fuel fire is likely to be much more hazardous than one from steam-oxidation driven core accidents.

    - Equivalent release of cesium

    - Release of potent non-volatiles

    - Significant release of actinides

- The LERF acceptance criterion for spent fuel fires to correspond to the prompt fatality safety goal is believed to be much lower than the RG 1.174 value.

CONCLUSIONS (continued)

- Because of the different mix of fission products, prompt fatalities may no longer be the controlling consequences.
    - Land contamination
    - Latent cancer deaths
- The air-zirconium ignition temperature used to determine the decay time required to have acceptable frequency of a spent fuel fire needs to include the effects of hydridization of the clad.

BOTTOM LINE

- This is a serious issue. The safety measures we have in place should be considered as defense in depth to compensate for our inability to calculate the risks -- can only relax these measures when risk assessment is adequate.

- Needed are:

  - A good phenomenological understanding

  - A good risk assessment

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

April 13, 2000

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

Dear Chairman Meserve:

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

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

BACKGROUND

40

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

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

RECOMMENDATIONS

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

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

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

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

DISCUSSION

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

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

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

Our concern is not just with ruthenium. We are concerned with the appropriateness of the entire source term used in the study. There is a known tendency for uranium dioxide in air to decrepitate into fine particles. The decrepitation is caused by lattice strains produced as the dioxide reacts to form U_3O_8. This decrepitation is a bane of thermogravimetric studies of air oxidation of uranium dioxide since it can cause fine particles to be entrained in the flowing air of the apparatus. This suggests that decrepitating fuel would be readily entrained in vigorous natural convection flows produced in an accident at a spent fuel pool. The decrepitation process provides a low-temperature, mechanical, release mechanism for even very refractory radionuclides. The staff did consider the possibility that "fuel fines" could be released from fuel with ruptured cladding. It did not, however, believe these fuel fines could escape the plant site. Nevertheless, the staff considered the effect of a $6x10^{-6}$ release fraction of fines. This minuscule release fraction did not significantly affect the calculated findings. There is no reason to think that such a low release fraction would be encountered with decrepitating fuel.

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

The staff needs to review the air oxidation fission products release data from Oak Ridge National Laboratory and from Canada that found large releases of cesium, tellurium, and ruthenium at temperatures lower than 1000 °C. Based on these release values for ruthenium, and incorporating uncertainties in the MACCS plume dispersal models, the consequence analyses should be redone. 42

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

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

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

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

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

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

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

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

Because the accident analysis is dominated by sequences involving human errors and seismic events which involve large uncertainties, the absence of an uncertainty analysis of the frequencies of accidents is unacceptable. The study is inadequate until there is a defensible uncertainty analysis.

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

Sincerely

Dana A. Powers

Dana A. Powers
Chairman

References:

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

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

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.

45

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

47

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.

ACRS MEETING WITH THE NRC COMMISSIONERS

OCTOBER 6, 2000

MORE REALISTIC THERMAL-HYDRAULIC CODES

DR. GRAHAM B. WALLIS
ACRS

THERMAL-HYDRAULIC CODES

- Major tools for calculations of reactor system behavior during transients and accidents.

- Predictions are untested on a full-scale system.

- The original (Appendix K) regulations required conservative predictions and margins.

- The 1988 amendment to 10CFR 50.46 (ECCS Rule) introduced "realistic" (sometimes called "best-estimate") codes and emphasized computation of uncertainty.

THERMAL-HYDRAULIC CODES (continued)

- Risk estimation requires evaluation of model uncertainties and probability of meeting success criteria.

- Risk-informed regulation requires better understanding of what a realistic code is and the criteria for its acceptability.

THE PRESENT SITUATION

- Industry is submitting codes (RETRAN-3D, S-RELAP5, TRACG).

- A Regulatory Guide and a Standard Review Plan are under development to guide content of code submittals and agency review procedures.

- ACRS is reviewing all of the above.

- The quality of codes must be adequate to support regulatory decisions and public confidence.

THE NATURE OF CODES

- Many codes have the same ancestry, including a 30-year old foundation.
- Designed specifically for nuclear applications. Not commercial or academic.
- Contain many assumptions, idealizations, extrapolations, "best-shot" estimates, and user choices.
- Have evolved, but the development process is hard to trace.
- Codes may "work" but they are not based on a mature, secure, science.

THE NATURE OF CODES (continued)

- Pro: For several accidents (e.g. small-break LOCA) a few phenomena appear to dominate, and success criteria, such as peak clad temperature, appear insensitive to the details.

- Con: Some phenomena that could be important are poorly modeled (e.g. two-phase level in core boil-off).

- Code predictions have to be assessed for each application and extensive sensitivity checks performed.

- The staff's knowledge, experience, and thoroughness are key.

OBSERVATIONS

- Some code documentation is poor.
- The physical basis for analytical models is often incomplete and poorly explained.
- Assessment is unfocussed and insufficiently extensive.
- Methods for calculating uncertainties are primitive and not comprehensive.
- Documentation should be acceptable to knowledgeable, impartial observers.

OBSERVATIONS (continued)

- Risk-informed regulation will require more quantitative evaluation of model uncertainties and their consequences.

- The data base for assessment must be preserved and, in some cases, expanded.

- A base of "experts" needs to be maintained.

- Staff should run and evaluate vendor codes independently.

- Staff should maintain in-house code competence, including an NRC-developed code.

OBSERVATIONS (continued)

- Regulatory processes should encourage code improvements.
- NRC should be preparing for an eventual new generation of thermal-hydraulic codes.

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. 57

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

59

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

May 22, 2000

MEMORANDUM TO: William D. Travers
Executive Director for Operations

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

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

During the 472nd meeting of the Advisory Committee on Reactor Safeguards, May 11-13, 2000, the Committee met with the NRC staff to discuss the subject draft regulatory guide and standard review plan (SRP) Section. Following the NRC staff presentation, the Committee decided that additional review of these documents by the Committee prior to issuing them for public comment is not necessary. The Committee requested 60
that the staff provide any revisions of these documents to the ACRS prior to issuance for public comment.

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

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

cc: A. Vietti-Cook, SECY
J. Blaha, OEDO
G. Millman, OEDO
S. Collins, NRR
A. Thadani, RES
G. Holahan, NRR
E. Rossi, RES
J. Wermiel, NRR
F. Eltawila, RES

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 61
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.

62

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.

ACRS MEETING WITH THE NRC COMMISSIONERS

OCTOBER 6, 2000

STATUS OF ACRS ACTIVITIES ON LICENSE RENEWAL

DR. MARIO V. BONACA
ACRS

ACRS REVIEW OF LICENSE RENEWAL (LR) GUIDANCE DOCUMENTS

- ACRS is reviewing:
  - Standard Review Plan (SRP)
  - Regulatory Guide
  - Generic Aging Lessons Learned (GALL) report
  - NEI 95-10, Industry Implementation Document
- Consultants to assist ACRS in its review
- ACRS to prepare a report during its November 2000 meeting

ACRS EXPECTATIONS OF GUIDANCE DOCUMENTS

- Provide a consistent and understandable review process.
- Provide adequate technical bases to support decisions.
- Provide guidance to review staff to develop a comprehensive understanding of LR issues and related solutions.
- Adequately capture lessons learned from the review of Oconee and Calvert Cliffs applications.
- Provide adequate guidance for evaluating the effectiveness of unique programs dealing with plant-specific operating experience.
- Adequately incorporate LR generic issue resolutions.

ACRS PLAN FOR REVIEW OF LR APPLICATIONS

- ANO-1 LR safety evaluation report - March 2001.

- Hatch LR safety evaluation report and supporting generic boiling water reactor (BWR) technical bases - April 2001.

- Form two LR subcommittees to handle expected workload and divide review of LR applications starting in 2002.

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

March 13, 2000

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

Dear Chairman Meserve:

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

During the 470th meeting of the Advisory Committee on Reactor Safeguards, March 1-4, 2000, we completed our review of Duke Energy Corporation's application for license renewal of the Oconee Nuclear Station, Units 1, 2 and 3 and the related Final Safety Evaluation Report (FSER). Our review included a plant visit and four meetings, one of which was conducted in Clemson, South Carolina. We had the benefit of insights gained from two meetings concerning generic license renewal issues and the review of another license renewal application. We provided an interim letter dated September 13, 1999, concerning the Oconee license renewal application. During these reviews, we had the benefit of the documents referenced. 70

<u>Conclusion</u>

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

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

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

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

Background and Discussion

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

Sincerely,

Dana A. Powers

Dana A. Powers
Chairman

73

References:

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

8. U. S. Nuclear Regulatory Commission, NUREG/CR-5704, "Effects of LWR Coolant Environments on Fatigue Design Curves of Austenitic Stainless Steels," April 1999.

9. U. S. Nuclear Regulatory Commission, NUREG/CR-6260, "Application of NUREG/CR-5999 Interim Fatigue Curves to Selected Nuclear Power Plant Components," March 1995.

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

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. 75

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. 76

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. 77

Sincerely,

Dana A. Powers

Dana A. Powers
Chairman

<u>References:</u>

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

October 29, 1999

MEMORANDUM TO: William D. Travers
Executive Director for Operations

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

SUBJECT: PROPOSED STRATEGY FOR ACRS REVIEW OF LICENSE RENEWAL APPLICATIONS AND PROCESS

During the 466th meeting of the Advisory Committee on Reactor Safeguards, September 30 - October 2, 1999, the Committee formulated a proposed strategy for its review of license renewal applications and the process. The Committee requests that your staff review and comment on the attached document. 79

Attachment: Proposed Strategy for ACRS Review of License Renewal Applications and Process, updated October 26, 1999 **(Predecisional Draft)**

cc: A. Vietti-Cook, SECY
J. Blaha, OEDO
G. Tracy, OEDO
S. Collins, NRR
D. Matthews, NRR
C. Grimes, NRR

ATTACHMENT 1

PROPOSED STRATEGY FOR ACRS REVIEW OF LICENSE RENEWAL APPLICATIONS AND PROCESS

The NRC staff has a statutory requirement in 10 CFR 54.25 to refer each license renewal application to the ACRS for a review and report. An ACRS review is essential given the safety implications of extending power operation of a significant number of plants for 20 years beyond their current licensed life. ACRS involvement at this time is also important because congressional and industry interests have made license renewal a high-priority item for the Commission. This places significant pressure on the staff to expedite the review process and to reduce demonstration and documentation requirements (see the recent debate on credit for existing programs) at the very time that the rule interpretation and detailed requirements for future applications are being finalized. ACRS involvement will help in the ongoing development of a standardized license renewal process.

The ACRS can play a valuable role in several ways:

- By participating in the development of a standardized license renewal process to ensure that detailed requirements for license renewal applications are necessary and sufficient to provide reasonable assurance that plants will operate safely for up to 60 years.
- By identifying significant process issues and focusing attention on the way these issues are addressed in individual applications.
- By providing to the Commission independent views on contested interpretation of the rule, such as the recently debated issue of credit for existing programs.
- By identifying issues, as appropriate, that may be outside the narrow confines of the rule, for example, using risk information to further improve the license renewal process.

ACRS REVIEW OF LICENSE RENEWAL ACTIONS

During the initial application[*1] phase and the development of a standardized license renewal process, the ACRS will be involved in the following three main activities:

1. ***Review of the Applications and Associated Safety Evaluation Reports (SERs)***

   Selected license renewal topical reports will be reviewed as part of the license renewal application they support. In addition to evaluating the adequacy of each application and the related SER, the ACRS will develop a list of critical issues on which it intends to focus its attention for process improvement and review of future applications.

---

*1 The term "initial application" is intended to include those applications reviewed before the Final Standard Review Plan is issued and the first application reviewed for each vendor design (W, CE, B&W, GE).

2. ***Evaluation of the Adequacy of the License Renewal Process***

The ACRS will evaluate the effectiveness of the license renewal process in satisfying the intent of 10 CFR Part 54. This evaluation will generally take place during the review of the initial applications. Potential improvements will be communicated through the ACRS reports issued for initial applications.

During the ongoing development of standardized license renewal process, the ACRS Plant License Renewal Subcommittee will meet with the staff to review progress and provide its perspectives on license renewal issues. ACRS reports may be written as a result of these meetings.

3. ***Evaluation of Issues Raised During Application Reviews***

The ACRS may pursue issues that are identified during its review of initial applications, and this could lead to recommendations for changes to the license renewal rule. Issues of this nature identified to date by ACRS members for discussion are potential consideration of risk metrics in the license renewal process and process implications of life extension of plants licensed before current requirements were issued. The Committee plans to discuss and evaluate these issues, as appropriate.

The Committee plans to evaluate the improvements to the license renewal process. This evaluation, which is expected to be the most significant contribution of the ACRS to license renewal, will generally take place during the review of the initial applications and should end 81
with the issuance of the final License Renewal Standard Review Plan (SRP) around the first half of 2001. Therefore, after the final SRP is issued, the ACRS review of individual applications should be generally limited to activity 1 above.

FOCUS OF ACRS REVIEWS

As it becomes more conversant with license renewal issues through its review of initial applications and its interactions with the staff, the ACRS will identify a list of issues that it considers critical to the effectiveness of license renewal. This list is not intended to limit the scope of ACRS review but to focus it on key issues. Examples of critical issues identified to date for special attention in future applications are:

- Adequacy and completeness of the set of structures, systems, and components in the scope of the rule,
- Consideration of plant-specific operating experience in evaluating the adequacy of aging management programs,
- Demonstration of management of aging effects for reactor vessel, pressurizer, reactor coolant system piping and reactor vessel internals (void swelling of reactor vessel internals and aging of cast austenitic stainless steel components have been a focus of recent reviews),

- Resolution of applicable generic safety issues (GSIs),
- SER open issues and confirmatory items,
- New and modified aging management programs,
- Effectiveness of one-time inspections, and
- Maintenance of regulatory margins.

PROPOSED STRATEGY FOR ACRS REVIEW

The number of applications expected to be submitted in the next 3 to 4 years, combined with the Commission's expectation for a brief review period, poses a significant challenge to the ability of the ACRS to accommodate the expected number of applications without eliminating its involvement in other equally important issues. On the other hand, as the license renewal process is improved on the basis of reviewing the initial applications and the issuance of the final SRP, and as the ACRS gains experience, it is reasonable to expect that the ACRS review process will become less time intensive. Although the number of applications will increase with time, the use of a consistent staff review process and a more experienced ACRS should allow the same level of review of each application to be performed with fewer Subcommittee and Full Committee meetings than the four currently planned. Also with time, the number of ACRS reports for each application can be reduced.

82

Specifically, as described in the preceding section titled "ACRS Review of License Renewal Actions," the ACRS evaluation of the improvements to the license renewal process should come to an end after the final SRP is issued and the license renewal issues currently under evaluation are resolved. This is the phase in which the ACRS is expected to make its most significant contribution. To date, this contribution has come through generic recommendations presented in the interim reports. This makes the interim report essential for all initial applications, in that it allows the ACRS to contribute its perspectives as the license renewal process is being standardized. Closure of the open items that are generic to the process is also an important step that the ACRS must review. Therefore, Subcommittee meetings to review the revised SERs of initial applications are also necessary. This logic supports a four-step review of each initial application, which includes two Subcommittee and two Full Committee meetings, and the issuance of two reports.

Once the reviews of initial applications are completed and the staff review process is standardized, the focus of the subsequent ACRS reviews will shift to the adherence of the applications and related SERs to the requirements of the process. Because of this, the need for an interim report is greatly reduced. For example, all recommendations made by the ACRS in the interim report for the Oconee license renewal application were generic to the process. The interim report would have been unnecessary if the process had been standardized and the license renewal issues had been generically resolved. This consideration would support an ACRS review in which an interim report would only be written under special circumstances for which interim recommendations are necessary.

With these considerations in mind, following completion of the reviews of initial applications and issuance of the final SRP, the ACRS should be able to perform an effective review of each application with two meetings. The first would be a 2-day Subcommittee meeting to review the SER, and the second would be a Full Committee meeting session to review the revised SER and to write the ACRS report. The ACRS used this two-step review process when it reviewed operating licenses of new power plants.

As discussed above, in some instances the ACRS may have significant comments on the SER that may affect the SER and the resolution of open items. In that case, the ACRS may review the SER at a Full Committee meeting and write an interim report. Even in those instances, no more than three meetings would be required. Most applications should require only one report.

On the basis of the number of upcoming applications projected by the staff, it appears that the approach described above would lead to a reasonable ACRS resource commitment. The resource commitment associated with this approach is described in the next section.

<u>NUMBER OF EXPECTED APPLICATIONS, REVIEW ALTERNATIVES, AND RECOMMENDED APPROACH</u>

On the basis of the projected number of applications over the next 4 years, the ACRS estimated the number of meetings required to support different types of ACRS review alternatives. A four-step review of each application would require five Subcommittee and five Full Committee meetings in 2002, and a total of sixteen meetings in 2003. On the other hand and as discussed in the strategy section, as experience is gained and the license renewal 83
process is standardized, the ACRS can perform the same level of review with fewer Subcommittee and Full Committee meetings. Consequently, the following approach is proposed:

- **A four-step review** will be performed for all initial applications. This will allow the Committee to review the SERs and recommend improvements to the process in interim reports.

- **A two-step review** will be performed for subsequent applications as described in the previous section.

The implementation of this strategy would reduce the number of Subcommittee meetings to three in 2002 and four in 2003, and the number of Full Committee meetings to two in 2002 and to four in 2003.

<u>SUBCOMMITTEE REVIEW PROCESS</u>

The Committee identified a number of potential alternatives for Subcommittee review and report preparation. It is recommended that the Subcommittee continue to operate with each member reviewing introductory chapters and assigned chapters, and with the chairman having responsibility for report preparation. This approach appears most effective because it allows each member to become an expert on specific parts of the application and to gain a relative judgment of the quality of each application by comparing how different applicants deal with the same issue. Also, this approach will allow individual Subcommittee members to “own” ACRS issues across multiple applications.

g:\Dudley\LGM-LR_strategy.wpd
updated: 10\26\99

84