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

August 7, 2000

MEMORANDUM TO: ACRS Members

FROM: Michael T. Markley, Senior Staff Engineer
ACRS

SUBJECT: CERTIFICATION OF THE MINUTES OF THE MEETING OF THE ACRS SUBCOMMITTEE ON RELIABILITY AND PROBABILISTIC RISK ASSESSMENT - JUNE 28-29, 2000 - ROCKVILLE, MARYLAND

The minutes of the subject meeting, issued July 18, 2000, have been certified as the official record of the proceedings of that meeting. A copy of the certified minutes is attached.

Attachment: As stated

cc: via E-mail
J. Larkins
H. Larson
S. Duraiswamy
ACRS Staff Engineers
ACRS Fellows

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

MEMORANDUM TO: Michael T. Markley, Senior Staff Engineer

FROM: George Apostolakis, Chairman
Reliability and Probabilistic Risk Assessment Subcommittee

SUBJECT: CERTIFICATION OF THE SUMMARY/MINUTES OF THE MEETING OF THE ACRS SUBCOMMITTEE ON RELIABILITY AND PROBABILISTIC RISK ASSESSMENT, JUNE 28-29, 2000 - ROCKVILLE, MARYLAND

I do hereby certify that, to the best of my knowledge and belief, the minutes of the subject meeting on June 28-29, 1999, are an accurate record of the proceedings for that meeting.

July 24, 2000

George Apostolakis, Chairman Date
Reliability and PRA Subcommittee

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

PRE-DECISIONAL

July 18, 2000

MEMORANDUM TO: George Apostolakis, Chairman
Reliability and Probabilistic Risk Assessment Subcommittee

FROM: Michael T. Markley, Senior Staff Engineer

SUBJECT: WORKING COPY OF THE MINUTES OF THE MEETING OF THE ACRS SUBCOMMITTEE ON RELIABILITY AND PROBABILISTIC RISK ASSESSMENT, JUNE 28-29, 2000, ROCKVILLE, MARYLAND

A working copy of the minutes for the subject meeting is attached for your review. Please review and comment on them at your soonest convenience. Copies are being sent to each ACRS Member who attended the meeting for information and/or review.

Attachment:
As Stated

cc: ACRS Members
J. Larkins
H. Larson
S. Duraiswamy
ACRS Staff and Fellows

CERTIFIED

CERTIFIED BY:
G. Apostolakis - 7/24/00

Date:7/18/00

ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
MEETING OF THE SUBCOMMITTEE ON
RELIABILITY AND PROBABILISTIC RISK ASSESSMENT
MEETING MINUTES - JUNE 28-29, 2000
ROCKVILLE, MARYLAND

INTRODUCTION

The ACRS Subcommittee on Reliability and Probabilistic Risk Assessment met on June 28-29, 2000, at 11545 Rockville Pike, Rockville, MD, in Room T-2B3. The purpose of this meeting was to discuss the proposed final ASME standard for probabilistic risk assessment for nuclear power plant applications. The Subcommittee also discussed the status of risk-informed revisions to 10 CFR Part 50, including proposed revision to 10 CFR 50.44 concerning combustible gas control systems, issues in the Nuclear Energy Institute letter dated January 19, 2000 (Option 3), and public comments related to the Advance Notice of Proposed Rulemaking on 10 CFR 50.69 and Appendix T (Option 2).

The Subcommittee received no written comments from members of the public regarding the meeting. However, Mr. Bob Christie of Performance Technology, Inc. requested time to make a presentation during the June 29, 2000 session, concerning proposed revision to 10 CFR 50.44.

The entire meeting was open to public attendance. Mr. Michael T. Markley was the cognizant ACRS staff engineer for this meeting. The meeting was convened at 8:30 a.m. each day and recessed at 2:45 p.m on June 28 and adjourned at 3:05 p.m. on June 29, 2000, respectively

ATTENDEES

ACRS Members

G. Apostolakis, Co-Chairman
M. Bonaca, Member
T. Kress, Member
W. Shack, Member
J. Sieber, Member
R. Uhrig, Member
M. Markley, ACRS Staff
R. Savio, ACRS/ACNW Staff

Principal NRC Speakers

T. Bergman, NRR*
A. Camp, SNL*
C. Carpenter, NRR
M. Cheok, NRR
M. Drouin, RES*
T. King, RES
J. Lehner, BNL*
T. Pratt, BNL
T. Reed, NRR
M. Shuaibi, NRR
M. Snodderly, NRR
J. Williams, NRR

Principal Industry Speakers

S. Bernsen, ASME*
B. Budnitz, Future Resources
B. Christie, Performance Technology
G. Eisenberg, ASME
A. Heymer, NEI*
R. Hill, BWROG*
K. Fleming, ERIN Engineering
S. Floyd, NEI
B. Mrowca, BG&E*
F. Rahn, EPRI
R. Schneider, WOG*

NRR	Office of Nuclear Reactor Regulation
RES	Office of Nuclear Regulatory Research
BNL	Brookhaven National Laboratories
SNL	Sandia National Laboratories
ASME	American Society of Mechanical Engineers
BG&E	Baltimore gas and Electric Company
BWROG	Boiling Water Reactor Owners Group
WOG	Westinghouse Owners Group
EPRI	Electric Power Research Institute
ERIN	ERIN Engineering
NEI	Nuclear Energy Institute

There were approximately 12 members of the public in attendance at this meeting. A complete list of attendees is in the ACRS Office File, and will be made available upon request. The presentation slides and handouts used during the meeting are attached to the office copy of these minutes.

June 28, 2000

OPENING REMARKS BY THE SUBCOMMITTEE CHAIRMAN

Dr. George Apostolakis, Chairman of the ACRS Subcommittee on Reliability and Probabilistic Risk Assessment, convened the meeting at 8:30 a.m. He introduced the ACRS Members in attendance and stated that the purpose of this meeting was to discuss the proposed final ASME standard for probabilistic risk assessment for nuclear power plant applications. The Subcommittee also discussed the status of risk-informed revisions to 10 CFR Part 50, including proposed revision to 10 CFR 50.44 concerning combustible gas control systems, issues in the Nuclear Energy Institute letter dated January 19, 2000 (Option 3), and public comments related to the Advance Notice of Proposed Rulemaking on 10 CFR 50.69 and Appendix T (Option 2).

Dr. Apostolakis stated that the Committee previously provided its comments and recommendations to the EDO concerning the draft #10 version of the ASME Standard on PRA quality in a report dated March 25, 1999. He noted that the proposed ASME Standard, which focuses on internal events, has undergone several revisions in response to stakeholder comments. Dr. Apostolakis informed the Subcommittee that ASME held a public workshop on June 27, 2000 to discuss the proposed Standard with interested stakeholders and noted that he and Dr. Bonaca attended the workshop.

Dr. Apostolakis noted that the Subcommittee had received no written comments from members of the public regarding the meeting. However, Mr. Bob Christie of Performance Technology, Inc. requested time to make a presentation during the June 29, 2000 session, concerning proposed revision to 10 CFR 50.44.

DISCUSSION OF AGENDA ITEMS

ASME Presentation

Mr. Gerry Eisenberg, Director of ASME Nuclear Standards introduced the meeting participants. Mr. Sid Bernsen, Chairman of the ASME Committee on Nuclear Risk Management (CNRM) summarized the CNRM membership, development process, scope and purpose of the Standard development effort. Mr. Ron Simard, Chairman of the ASME Project Team discussed the major changes from the draft #10 version to the draft #12 version presently under consideration. Mr. Karl Fleming of the ASME Project Team summarized the comments from the June 27, 2000 public workshop including the relationship of PRA characteristics with associated categories in the proposed Standard. Mr. Robert Budnitz of Future Resources, Inc. participated via teleconference. Significant points made during the presentation include:

- Major public comments on draft #10 version were that the Standard (1) was too prescriptive, (2) needs to recognize that the primary use will be with existing PRAs, and (3) needs closer alignment to industry peer review and certification processes. CNRM removed most restrictive statements such as "shall" and "should" from the document. CNRM also modified the Standard to clarify applicability of existing PRA attributes and the linkage to industry certification processes.

- The ASME Standard uses published definitions rather than those customized for particular purposes. The proposed ASME Standard serves as an industrial guide which was not designed for the specific purpose of meeting NRC regulatory requirements.

- The proposed Standard utilizes three Categories of PRA for decision-making purposes:

  - Category 1: relies primarily on deterministic analysis supplemented with risk insights,
  - Category 2: relies on a "balanced" set of PRA insights and deterministic analyses, and
  - Category 3: relies primarily on PRA insights supplemented with little deterministic analyses.

- The PRA needed for the three Categories will be differentiated based on safety significance as measured by core damage frequency (CDF) and large, early release frequency (LERF). Safety significance will also be differentiated based on the dominant accident sequences and contributors, and prioritization and ranking of structures, systems, and components (SSCs).

- The proposed Standard is a consensus document which has not yet been approved by the at-large ASME membership. After considering additional feedback and insights from the ACRS and its members as well as other interested parties, CNRM will propose a revised version for consideration and final approval by ASME. Ultimately, ASME hopes to have the subject Standard endorsed by the American National Standards Institute (ANSI).

June 29, 2000

OPENING REMARKS BY THE SUBCOMMITTEE CHAIRMAN

Dr. George Apostolakis, Chairman of the ACRS Subcommittee on Reliability and Probabilistic Risk Assessment, convened the meeting at 8:30 a.m. He introduced the ACRS Members in attendance and stated that the purpose of this meeting was to discuss the status of risk-informed revisions to 10 CFR Part 50, including proposed revision to 10 CFR 50.44 concerning combustible gas control systems, issues in the Nuclear Energy Institute letter dated January 19, 2000 (Option 3), and public comments related to the Advance Notice of Proposed Rulemaking on 10 CFR 50.69 and Appendix T (Option 2).

Dr. Apostolakis noted that the Subcommittee had received no written comments from members of the public regarding the meeting. However, Mr. Bob Christie of Performance Technology, Inc. has requested time to make a presentation concerning proposed revision to 10 CFR 50.44.

DISCUSSION OF AGENDA ITEMS

NRC Staff Presentation - 10 CFR 50.69 and Appendix T (Option 2)

Mr. Thomas Bergman, NRR, led the discussions for the NRC staff. Messrs. Joseph Williams, Mohammed Shuaibi and Michael Cheok, NRR, provided supporting discussion. Ms. Cynthia

Carpenter and Timothy Reed, NRR, also participated. Significant points made during the presentation include:

- Most public comments on the Advanced Notice of Public Rulemaking (ANPR) were supportive of the proposed rule 10 CFR 50.69 and associated Appendix T concerning the categorization and special treatment of SSCs. Some public comments suggested that (1) a phased-approach should be applied, (2) it be performance-based, (3) the staff allow for selective implementation with limited prior NRC review approval, and (4) the backfit rule should be applied.

- The staff plans to review and comment formally on the proposed industry peer review process described in NEI 00-02, "Probabilistic Risk Assessment Peer Review Process Guidance." NEI 00-02 proposes the use of 4 categories of PRA qualification for risk-informed decisions. NEI proposes to use this peer review process as a means for certifying licensee PRAs for the purpose of Option 2 decisionmaking.

- The staff and industry positions are somewhat similar with respect to the categorization of SSCs. However, the major difference is in the treatment of SSCs that are risk-significant but are not currently safety-related, safety significance categories RISC-2. NEI proposes to manage SSCs via (a)(4) of the Maintenance Rule (10 CFR 50.65) in lieu of new regulatory requirements.

Performance Technology Presentation

Mr. Bob Christie of Performance Technology, Inc. provided a presentation on his petition for rulemaking on 10 CFR 50.44 concerning combustible gas control systems. He also discussed the staff's efforts related to develop a framework for Option 3 and the pilot associated with 10 CFR 50.44. Significant points made during the presentation include:

- The original issue of hydrogen control systems was brought to the attention of the NRC as a letter from Performance Technology, Inc. associated with initiatives being considered at the San Onofre Nuclear Power Plant. It was not submitted as a petition for rulemaking but was handled as such by the NRC. The initiative was integrated into the broader initiative known as the "NEI Whole Plant Study" related to risk-informing the requirements of 10 CFR Part 50.

- Mr. Christie stated that operator attention is unnecessarily distracted from more important safety activities, in order to address hydrogen control systems, during the early stages of an accident. While the NRC and the industry agree that there may be sufficient technical basis for removing hydrogen recombiners at most plants, there is a major disagreement regarding the need to maintain hydrogen ignitors and associated emergency electrical power supplies during a loss of offsite power event. Mr. Christie stated that containments are robust systems and that licensees should not be constrained by requirements associated with hydrogen control when the design can withstand a prompt hydrogen burn. He stated that licensee costs associated with maintaining hydrogen ignitors is a waste of resources when licensee attention should be focused on recovery of emergency power.

- The staff's proposed Option 3 framework is not needed to test 10 CFR 50.44. The staff should approve the petition for rulemaking under Option 2.

NRC Staff Presentation - 10 CFR 50.44 (Option 3)

Mr. Thomas King and Ms. Mary Drouin, RES, led the discussions for the NRC staff. Messrs. John Lehner and Trevor Pratt of Brookhaven National Laboratories, and Alan Camp of Sandia National Laboratories, provided supporting discussion. Messrs. Alan Kuritzky, RES, and Michael Snodderly, NRR, also participated. Significant points raised during the presentation include:

- The purpose of this briefing was to discuss the staff's efforts to revise 10 CFR 50.44. The staff is also revising its framework for risk-informing the technical requirements of 10 CFR Part 50 (SECY-00-0086), but the proposed revision to the framework will be the subject of a future briefing.

- The staff informed the Subcommittee that it had provided a response to NEI dated February 18, 2000, concerning issues and priorities in the NEI letter dated January 19, 2000. The staff stated that these issues were discussed during a public workshop on February 24-25, 2000, and that they were considering stakeholder input in proceeding on this matter. The staff plans to use the proposed Option 3 framework to evaluate the candidate regulations recommended for revision by NEI as well as the peer review certification process described in NEI 00-02.

- The staff plans to provide its revised framework and alternatives for revising 10 CFR 50.44 in a draft paper to the Commission in August 2000. The staff also plans to discuss the issue of "selective implementation." The staff stated that selective implementation would not be risk-informed and suggested that it may be necessary to increase regulatory requirements in certain areas. The staff requested to meet with the ACRS in September 2000 to discuss this matter.

NEI Presentation

Mr. Steven Floyd led the discussions for the Nuclear Energy Institute (NEI). Messrs. Biff Bradley and Adrian Heymer, NEI, provided supporting discussion. Significant points raised during the discussion include:

- NEI 00-02 is intended to serve as a means to qualify the use of PRAs for risk-informed decisionmaking under Option 2. NEI 00-02 was developed from the peer review certification process originally developed by the Boiling Water Reactor Owners Group (BWROG). The certification process does not provide an overall grade for PRAs. It can be used as a complement to or in lieu of industrial standards for PRA quality (e.g., ASME, ANS, etc.).

- NEI and the staff are in close agreement on the approach and issues related to Option 2. However, NEI is concerned that Option 3 is largely risk-based rather than risk-informed. NEI is concerned that risk criteria proposed for use in Option 3 will be used as a quantitative measure for adequate protection. NEI representatives stated that there continues to be too much emphasis on issues of low safety significance.

SUBCOMMITTEE COMMENTS, CONCERNS, AND RECOMMENDATIONS

Subcommittee members raised the following significant points during its discussion with ASME representatives:

Proposed ASME Standard

- Dr. Uhrig noted that this was a totally different kind of Standard than he would normally associate with ASME. He noted that OMB Circular A-119 directs federal agencies to endorse consensus industrial standards, where practicable.

- Mr. Sieber noted the presentations made by Karl Fleming and Gareth Parry during the ACRS retreat in January 2000 related to the simplicity and variability of current PRAs. He stated expressed the view that issue of PRA quality will probably remain and suggested that the industry certification process will be a major factor in validating PRAs.

- Dr. Kress stated that he was disappointed that the Standard was limited to the current definition of core damage frequency (CDF). He stated that, in risk-informing the regulations, more work is needed related to fission products and noted that the main issue is categories. He also questioned the definition of LERF and consideration of uncertainty. He noted specific problems with certain containment designs and suggested that ASME representatives look at Regulatory Guide 1.174. He stated that it would be a mistake to go away from the intended use of LERF as it relates to the timing of accidents and the modeling of fission product releases. He expressed the view that LERF should be site-independent.

- Drs. Shack and Apostolakis questioned the removal of detailed guidance in the transition from Revision 10 to Revision 12. Dr. Shack suggested that some of the detailed guidance that the ACRS reviewed in Revision 10 was useful. Dr. Apostolakis noted that a lot of useful information could also be provided in an expanded list of references.

- Drs. Bonaca and Shack expressed concern that, in some cases, the proposed ASME Standard allows or encourages limiting the problem/scope up-front before analysis is pursued. Drs. Bonaca and Shack noted that the proposed Standard was intended for general purposes and not to address specific regulatory concerns. Thus, they questioned why (a)(4) of the Maintenance Rule was considered "Category 1." Dr. Bonaca noted that his experience has shown that some issues cannot be bounded up-front (i.e., pre-judged) and are usually realized or revealed during the analysis. They also questioned the level of verification and validation provided by industry "peer reviews," including the qualification and independence of the review panel.

- Dr. Apostolakis suggested that "Category 1" may not be a useful category for evaluating the quality of PRAs. He stated that it provides a somewhat false sense of pedigree when it is largely based on traditional, deterministic analysis.

- Dr. Apostolakis reiterated a comment raised at the June 27 ASME workshop concerning the quality of NRC models and suggested that they (i.e., SPAR models) be subjected to similar quality evaluation/certification.

Risk-Informing 10 CFR Part 50 and Related Matters

Subcommittee members raised the following significant points during its discussion with the staff and industry representatives:

Option 2

- Dr. Apostolakis questioned the functional categorization provided in NEI-00-02, Probabilistic Risk assessment (PRA) Peer Review Process Guidance. In particular, he questioned the assessment of functions for various modes of plant operation. He also questioned the extent to which the NRC has considered the in which South Texas Project categorized more than 20, 000 structures, systems and components (SSCs) and how those results would compare to the process proposed by NEI. Dr. Apostolakis also questioned why other approaches such as the Palisades Top Event Prevent (TEP) were omitted from the NEI guideline.

NRR has submitted a User Needs Request for RES assistance in reviewing the NEI 00-02 peer review guideline. RES plans to complete the requested actions in tandem with current tasks related to PRA quality and industrial standards (ASME, ANS, NFPA, etc.).

Option 3 and 10 CFR 50.44

- Dr. Kress questioned the role hydrogen monitors have in the prevention and mitigation of accidents, with particular emphasis on the contribution to CDF. He expressed the view that the NRC should only be concerned with the dominant core damage sequences. He also questioned the role of ignitors in maintaining containment integrity and, thus, LERF. He suggested that it would be preferable to have the ignitors to provide for slow hydrogen burns rather than allowing hydrogen buildup to potentially explosive levels, even with sufficient design margin.

- Dr. Kress also questioned the possibility of selective implementation of proposed risk-informed revisions to 10 CFR 50.44. He agreed with the staff's view that the approach should not be selective and should "cut both ways," in requiring risk enhancements as well as allowing regulatory relaxation associated with burden reduction.

- Dr. Bonaca questioned the approach to containment capability provided in the petition for rulemaking associated with 10 CFR 50.44. In particular, he questioned the prudency of linking containment performance with severe accident phenomena. He stated that containment performance may not always be as conservative as previously estimated in IPEs due to aging, relaxation of containment tendons, variations in plant design feature (e.g., ice condenser and Mark III containments, etc.) performance.

NEI Letter

Despite its presence on the Subcommittee agenda, neither the staff nor NEI addressed the issues and priorities provided in the letter dated January 19, 2000. However, the staff offered to provide a letter received from NEI on April 18, 2000, concerning the draft NRC framework (Option 3) and draft report on 10 CFR 50.44.

STAFF AND INDUSTRY COMMITMENTS

During the discussion of the ASME Standard, Dr. Apostolakis suggested that ASME modify the Standard to refer to decision criteria provided in Regulatory Guide 1.174, "An Approach for Using Probabilistic Risk Assessment in Risk-Informing Decisions on Plant-Specific Changes to the Licensing Basis." ASME representatives agreed to consider this suggestion. At the conclusion of the Subcommittee meeting, Dr. Apostolakis requested ASME representatives to provide a brief overview of the Subcommittee presentation and requested that the presentation focus largely on the issues raised during the meeting.

The staff has not yet provided its draft Commission paper and associated revised report on 10 CFR 50.44. The staff expects to provide the subject documents by prior to the July 11, 2000 Subcommittee meeting.

SUBCOMMITTEE DECISIONS

At the conclusion of the meeting, Dr. Apostolakis recommended and the Subcommittee and staff agreed to hold another meeting to review the Option 3 framework document (revised SECY-00-0086). The subject meeting was scheduled for July 11, 2000.

FOLLOW-UP ACTIONS

During the discussion of risk-informed 10 CFR Part 50 (Option 2), Dr. Apostolakis requested copies of the South Texas Project risk-informed exemption request. The staff agreed to provide the subject documents following the meeting.

Dr. Apostolakis requested a copy the staff's viewgraphs from the October 1999 ACRS briefing on proposed risk-informed revisions to 10 CFR Part 50. The ACRS staff provided the subject documents prior to the conclusion of the meeting.

BACKGROUND MATERIALS PROVIDED TO THE SUBCOMMITTEE PRIOR TO THIS MEETING

1. Subcommittee agenda.
2. Subcommittee status report.
3. Letter dated June 12, 2000, from Cynthia Carpenter, NRC, to John T. Larkins, ACRS, Subject: Meeting of the Subcommittee on PRA June 28-29, 2000, and the full Committee on July 12-14, 2000, and attachments.
4. Staff Requirements Memoranda dated January 31, 2000 (SECY-99-256) on Option 2 and February 3, 2000 (SECY-99-264) on Option 3).
5. Staff Requirements Memoranda dated April 5, 2000 concerning the NEI letter dated January 19, 2000, and April 18, 2000, on staff plans to address the issue of PRA quality.
6. Letter dated January 19, 2000 from Joe Colvin, NEI, to Chairman Meserve, Chairman, NRC, Subject: Priorities for risk-informing 10 CFR Part 50 (Option 3).
7. Letter dated June 14, 2000, from G.M. Eisenberg, ASME, to Michael T. Markley, ACRS, Subject: Copies of Draft 12 of Proposed ASME Standard on Probabilistic Risk Assessment for Nuclear Power Plant Applications, and associated White Paper dated June 13, 2000.

8. Letter dated March 25, 1999, from Dana A. Powers, Chairman, ACRS, to William D. Travers, EDO, Subject: Proposed ASME Standard for PRA in NPPs.
9. 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.
10. Handouts and handouts from Commission meeting on June 20, 2000, concerning risk-informing 10 CFR Part 50 (Option 3), and associated meeting transcript.
11. Memorandum dated April 12, 2000, from William D. Travers, EDO, NRC, to The Commissioners, Subject: SECY-00-0086, Status Report on Risk-Informing Technical Requirements of 10 CFR Part 50 (Option 3).
12. Letter dated February 18, 2000, from Ashok C. Thadani, Director, RES, NRC, to Joe F. Colvin, NEI, Subject: Response to January 19, 2000 NEI letter, concerning risk-informing 10 CFR Part 50 (Option 3).
13. Letter dated June 7, 2000, from Stephen D. Floyd, NEI, to Scott F. Newberry, NRR, NRC, Subject: Comments on Advanced Notice of Proposed Rulemaking on 10 CFR 50.69 and Appendix T concerning risk-informed categorization and treatment of structures, systems and components.
14. Memorandum dated June 19, 2000, from Samuel J. Collins, Director, NRR, to Ashok C. Thadani, Director, RES, Subject: Request for Assistance in Review of NEI 00-02, "Probabilistic Risk Assessment Peer Review Process Guidance."
15. Draft report entitled, "A new importance measure for risk-informed decision making," by E. Borgonovo and G.E. Apostolakis to be presented at PSAM5, Osaka, Japan, November 27-30, 2000.

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Note: Additional details of this meeting can be obtained from a transcript of this meeting available in the NRC Public Document Room, 2120 L Street, N.W. Washington, D.C. 20006, (202) 634-3274, or can be purchased from Ann Riley & Associates, Ltd., (Court Reporters and Transcribers) 1250 I Street, NW, Suite 1014, Washington, D.C. Rhode Island Avenue, N.W. Washington, D.C. 20036 (202) 842-0034.

REVISED 6/26/00

ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
MEETING OF THE SUBCOMMITTEE ON PROBABILISTIC RISK ASSESSMENT
ROOM T-2B3, 11545 ROCKVILLE PIKE, ROCKVILLE, MD
JUNE 28-29, 2000

ACRS Contact: Michael T. Markley (301) 415-6885

- PROPOSED SCHEDULE -

June 28, 2000

	TOPIC	PRESENTER	TIME
1)	**Introduction**		8:30-8:35 am
•	Review goals and objectives for this meeting	G. Apostolakis, ACRS	
•	Review points raised in ACRS report dated March 25, 1999; ACRS member assignments for reviewing the proposed Standard	G. Apostolakis, ACRS	
2)	**ASME Presentation**		8:35-~~10:00~~ 9:45 am
•	Introductory remarks	G. Eisenberg, ASME	
•	Discussion of revised ASME document entitled, "Standard for Probabilistic Risk Assessment for Nuclear Power Plant Applications," including proposed use of industry certification programs.	S. Bernsen, Chairman ASME CNRM R. Simard, ASME Project Team Leader Others, TBD	
•	Reconciliation of comments (ACRS, NRC, industry, and public) on draft #10		
•	Public comments from the June 27, 2000 public workshop on the revised Standard.		
	** BREAK **		9:45-1000 ~~10:00-10:15~~ am
3)	**ASME Presentation - continued**		10:15-~~12:00 noon~~ 11:08 a.m.
•	Discussion of technical issues associated with the proposed Standard and its use, including the use of expert opinion, peer review, quantitative and qualitative aspects, methods and models.	ASME, TBD	
	** BREAK **		11:08 - 11:25 am

	** LUNCH **		12:00-1:00 pm
4)	**General Discussion and Recess**		1:00-~~2:30~~ 2:45 pm
•	General discussion and comments by Members of the Subcommittee; items for July 12-14, 2000 ACRS meeting	G. Apostolakis, ACRS	

June 29, 2000

	TOPIC	PRESENTER	TIME
5)	**Introduction**		8:30-8:35 am
•	Review goals and objectives for this meeting	G. Apostolakis, ACRS	
•	Review points raised during March 2000 ACRS meeting and issues noted in ACRS report dated October 12, 1999	G. Apostolakis, ACRS	
6)	**NRC Staff Presentation**		8:35-~~10:15~~ 9:35 am
•	Discussion of public comments on proposed 10 CFR 50.69 and associated Appendix T (Option 2)	C. Carpenter, NRR T. Bergman, NRR T. Reed, NRR	
•	NRC staff perspective on proposed industry peer certification process and draft NEI guideline on special treatment		
•	Plans to brief the Commission in September 2000 on proposed reconciliation of public comments.		
	** BREAK **		9:35 ~~10:15~~-10:30 am
7)	**Industry Presentation**		10:30-~~11:30~~ 12:45 am
•	Petition for rulemaking to 10 CFR 50.44 concerning combustible gas control systems	B. Christie, Performance Technology, Inc.	
	** LUNCH **		11:30-12:30 pm
8)	**NRC Staff Presentation**		12:30-~~2:00~~ 1:50 pm
•	Discussion of proposed revision to 10 CFR Part 50 (Option 3) and 10 CFR 50.44 concerning combustible gas control	T. King, RES M. Cunningham, RES M. Drouin, RES J. Lehner, BNL T. Pratt, BNL A. Camp, SNL	
	** BREAK **		1:50-2:00 pm

8) Staff Presentation Continued | T. King, RES et. al. | 2:00 - 2:20

systems

- Status of 10 CFR 50.44 rulemaking petition — C. Carpenter, NRR

~~** BREAK **~~ ~~2:00-2:15 pm~~

9) Industry Presentation ~~2:15~~ 2:20 - 2:45 pm

- Industry perspective on proposed revision to 10 CFR 50.69 and Appendix T — S. Floyd, NEI; A. Heymer, NEI
- Issues and priorities noted in the NEI letter dated January 19, 2000
- Status of industry guidance development

10) General Discussion and Adjournment 2:45-~~3:00~~ 3:05 pm

- General discussion and comments by Members of the Subcommittee; items for July 12-14, 2000 ACRS meeting — G. Apostolakis, ACRS

<u>Note</u>: Presentation time should not exceed 50% of the total time allocated for a specific item. Number of copies of presentation materials to be provided to the ACRS - 35.

INTRODUCTORY STATEMENT BY THE CHAIRMAN OF THE
SUBCOMMITTEE ON RELIABILITY AND PRA
11545 ROCKVILLE PIKE, ROOM T-2B3
ROCKVILLE, MARYLAND
JUNE 28-29, 2000

The meeting will now come to order. This is the first day of the meeting of the ACRS Subcommittee on Reliability and Probabilistic Risk Assessment. I am George Apostolakis Chairman of the Subcommittee.

ACRS Members in attendance are: Mario Bonaca, Thomas Kress, William Shack, Jack Sieber and Robert Uhrig.

The purpose of this meeting is to discuss the proposed final ASME standard for probabilistic risk assessment for nuclear power plant applications. Tomorrow, the Subcommittee will discuss the status of risk-informed revisions to 10 CFR Part 50, including proposed revision to 10 CFR 50.44 concerning combustible gas control systems, issues in the Nuclear Energy Institute letter dated January 19, 2000 (Option 3), and public comments related to the Advance Notice of Proposed Rulemaking on 10 CFR 50.69 and Appendix T (Option 2). The Subcommittee will gather information, analyze relevant issues and facts, and formulate proposed positions and actions, as appropriate, for deliberation by the full Committee. Michael T. Markley is the Cognizant ACRS Staff Engineer for this meeting.

The rules for participation in today's meeting have been announced as part of the notice of this meeting previously published in the *Federal Register* on May 16, 2000.

A transcript of the meeting is being kept and will be made available as stated in the Federal Register Notice. It is requested that speakers first identify themselves and speak with sufficient clarity and volume so that they can be readily heard.

We have received no written comments or requests for time to make oral statements from members of the public regarding today's meeting. However, Mr. Bob Christie of Performance Technology, Inc. has requested time to make a presentation during tomorrow's session concerning proposed revision to 10 CFR 50.44.

(Chairman's Comments-if any)

We will now proceed with the meeting and I call upon Mr. Gerry Eisenberg of ASME to begin.

INTRODUCTORY STATEMENT BY THE CHAIRMAN OF THE
SUBCOMMITTEE ON RELIABILITY AND PRA
11545 ROCKVILLE PIKE, ROOM T-2B3
ROCKVILLE, MARYLAND
JUNE 28-29, 2000

The meeting will now come to order. This is the second day of the meeting of the ACRS Subcommittee on Reliability and Probabilistic Risk Assessment. I am George Apostolakis Chairman of the Subcommittee.

ACRS Members in attendance are: Mario Bonaca, Thomas Kress, William Shack, Jack Sieber and Robert Uhrig.

The purpose of this meeting is to discuss the status of risk-informed revisions to 10 CFR Part 50, including proposed revision to 10 CFR 50.44 concerning combustible gas control systems, issues in the Nuclear Energy Institute letter dated January 19, 2000 (Option 3), and public comments related to the Advance Notice of Proposed Rulemaking on 10 CFR 50.69 and Appendix T (Option 2). The Subcommittee will gather information, analyze relevant issues and facts, and formulate proposed positions and actions, as appropriate, for deliberation by the full Committee. Michael T. Markley is the Cognizant ACRS Staff Engineer for this meeting.

The rules for participation in today's meeting have been announced as part of the notice of this meeting previously published in the *Federal Register* on May 16, 2000.

A transcript of the meeting is being kept and will be made available as stated in the Federal Register Notice. It is requested that speakers first identify themselves and speak with sufficient clarity and volume so that they can be readily heard.

We have received no written comments from members of the public. However, Mr. Bob Christie of Performance Technology, Inc. has requested time to make a presentation concerning proposed revision to 10 CFR 50.44.

(Chairman's Comments-if any)

We will now proceed with the meeting and I call upon Ms. Cynthia Carpenter, NRR, to begin.

ADVISORY COMMITTEE ON REACTOR SAFEGUARDS

PLANT OPERATIONS AND RELIABILITY AND PROBABILISTIC RISK ASSESSMENT

June 28-29, 2000
Today's Date

NRC STAFF SIGN IN FOR ACRS MEETING

PLEASE PRINT

NAME	BADGE #	NRC ORGANIZATION
Mohammed Shuaibi	B-8538	NRC/NRR/DSSA/SRXB
Tom Bergman	B-6074	NRC/NRR
TIM REED	B 6986	NRC/NRR
MICHAEL SNODDERLY	B 7109	NRC/NRR/SPSB
JOE WILLIAMS	B7257	NRC/NRR/DLPM
Mike Cheok	B7917	NRC/NRR/DSSA
Pete PRASSINOS	B8720	NRC/RES/DRAA
Steve West	B7258	NRC/NRR/DRIP
STU MAGRUDER	B-6721	NRC/NRR/DRIP
Alan Kuritzky	B-8696	RES/DRAA
CINDI CARPENTER	B-6464	NRR/DRIP/RGEB
GARETH PARRY	B-8060	NRR/DSSA
Eileen McKenna	B-8226	NRR/DRIP/RGEB
Nanette Gilles	B-4480	NRR/DRIP/RGEB
Charles Ader	B-8389	NRR/DRIP
Mark Rubin	B 7052	NRR/DSSA
ANTHONY MARKLEY	B 8559	NRR/DRIP

ADVISORY COMMITTEE ON REACTOR SAFEGUARDS

PLANT OPERATIONS AND RELIABILITY AND PROBABILISTIC RISK ASSESSMENT

June 28-29, 2000
Today's Date

ATTENDEES - PLEASE SIGN BELOW

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NAME	AFFILIATION
Richard Hill	GENE
Ray Schneider	W
BARRY SLOANE	WESTINGHOUSE
DEEPAK RAO	ENTERGY
Gerry Eisenberg	ASME
SID BERNSEN	ASME
Karl Fleming	ERIN Eng & Research
IAN WALL	Consultant - EPRI
FRANK RAHN	EPRI
Bruce Mrowca	BGE
Althela Wyche	SERCH Licensing / Bechtel
RON SIMARD	NEI

ADVISORY COMMITTEE ON REACTOR SAFEGUARDS

PLANT OPERATIONS AND RELIABILITY AND PROBABILISTIC RISK ASSESSMENT

June 28-29, 2000
Today's Date

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GARETH PARRY	B-8060	NRR/DSSA
Stephen Dinsmore	B-7898	NRR/DSSA
Nanette Gilles	B-8580	NRR/DRIP/RTSB
Steven West	B-7258	NRR/DRIP/RGEB
JF Costello	B-E385	NRC/RES/~~DET~~
Mike Cheok	B-7917	NRR/DSSA

ADVISORY COMMITTEE ON REACTOR SAFEGUARDS

PLANT OPERATIONS AND RELIABILITY AND PROBABILISTIC RISK ASSESSMENT

June 28-29, 2000
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ATTENDEES - PLEASE SIGN BELOW

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NAME	AFFILIATION
Bob Christie	Performance Technology
Vinod Mubayi	Brookhaven Lab
John Lehner	BNL
Eric Haskin	ERIConsulting
Allen Camp	SNL
Jeff LaChance	SNL
Nancy Chapman	SERCH/Bechtel
Parviz Moieni	SCE
Bob Youngblood	ISL
Biff Bradley	NEI
Adrian Heymer	NEI

White paper and guidance to reviewers of the draft ASME Standard for Probabilistic Risk Assessment for Nuclear Power Plant Applications

June 13, 2000

Background

A project team under the ASME Committee on Nuclear Risk Management (CNRM) is drafting a Standard on the use of PRAs to support applications of risk-informed decisionmaking at nuclear power plants. At this point in the development of the Standard, an earlier draft (called "Rev 10") has received broad public review and comment. The current draft of the Standard (called "Rev 12") addresses the comments that were received and is being released for one more round of review and comments before balloting by the CNRM.

The purpose of the current review is to seek feedback on the proposed revisions to the Standard in response to the earlier comments. To facilitate review and understanding of the project team rationale in developing this draft, a public workshop will be held to give reviewers an opportunity to meet with members of the project team on June 27. Information on this workshop is posted on the ASME website at *www.asme.org*

The comment period will end on August 14, 2000. Comments should be submitted to Gerry Eisenberg, ASME staff secretary to the CNRM at *eisenbergg@asme.org*

Scope and intended use of the Standard

Section 1 summarizes the scope of the Standard and the way in which it is to be used in support of risk-informed applications of PRA results. The Standard was developed to support the use of existing nuclear plant PRAs. It recognizes that existing PRAs vary in quality. It recognizes that the extent to which a PRA is relied upon in a risk-informed decision varies; that there is a broad spectrum of possible applications; that the level of PRA quality needed to support a particular application also varies. Therefore, the Standard is structured to:

- Approximate the range of possible applications by describing three broad categories of applications
- Provide a set of High Level Requirements that apply across all applications
- Provide Supporting Requirements whose scope of applicability varies across the three categories of applications
- Provide a process for identifying whether a PRA has the quality needed to support a specific application

- Require that a peer review process be used to establish the extent to which the PRA meets the requirements of the Standard

Response to comments on previous draft

An early draft of this standard (Rev 10) was released for broad review and comment in the Spring of 1999. A large number of comments were received, primarily through

- 46 sets of written comments, with over 2,000 observations and recommendations, submitted during the 90 day review period
- stakeholder observations at a March 16, 1999 public meeting on Rev 10
- several meetings of PRA users and utility representatives throughout 1999

These comments were collated and binned into three categories – observations, general comments, and specific comments on individual subsections of Rev 10. A large number of the observations and general comments fell into four areas, namely:

- the prescriptiveness, in terms of the number of requirements and perceived lack of flexibility in application of the standard
- the need to distinguish among grades of applications with a commensurate level of PRA quality
- the need to recognize that the primary current use of the standard will be for determining how existing PRAs can be used to support risk informed applications, and
- the related need for closer alignment of the standard with the peer review and certification process developed and being implemented by the U.S. nuclear industry (the "Industry Certification Process")

These comments represented widely held views from a broad spectrum of respondents: including the Nuclear Energy Institute (NEI), individual plant owners, NSSS Owners' Groups, and nuclear system suppliers and consultants, as well as NRC and other regulatory bodies. The project team carefully considered each comment and the majority of them were incorporated or resolved by the revisions made in Rev 12. Responses were drafted for most of the detailed comments received. However, as discussed below, resolution of the general comments involved a major restructuring of the standard. As a result, it was not practical to prepare detailed and consistent responses to all the comments received.

The following discusses the major changes that have been made and should help those who submitted comments on Rev 10 understand how their comments were resolved.

The current draft (Rev 12) has been restructured. In particular, the section on the Risk Assessment Application Process has been moved forward in the document and placed before the section identifying the Risk Assessment Technical Requirements. Also, the spectrum of risk informed applications at nuclear power plants has been approximated by three categories of application. The corresponding requirements for a PRA to be used with this standard to support these applications are presented in tables, where the table columns correspond to the three categories of application. To minimize redundancy and improve clarity, Section 1 of Rev 10 on General Requirements has been changed to an Introduction and essentially all the requirements of the previous Section 1 have been incorporated in other sections of Rev 12. Also, the requirements of Rev 10 Section 4 on Documentation have been incorporated into the requirements tables of Section 4 of Rev 12. Although the Generic Database contained in Appendix 1 of Rev 10 received several favorable comments, other comments questioned its adequacy and consistency with other data currently in use. As a result, that appendix has been removed and the Committee on Nuclear Risk Management will evaluate separately the potential for developing a suitable database for future standardization.

A number of comments were received questioning some of the administrative requirements for Quality Assurance, Owner's responsibility, etc. These have been deleted because this technical standard is not self-enforcing and is intended for use in conjunction with enabling regulatory documents or other Codes and Standards that include administrative requirements appropriate to their application

While this draft of the standard was being prepared, the industry certification process criteria were formalized and issued in an NEI document (NEI-00-02). Requirements for peer review of a PRA to be used with this standard, as well as the process for determining the ability of a PRA to support a specific application, are structured to incorporate results from this industry certification process. Also, the tables of PRA requirements have been referenced to the technical checklist items incorporated in the industry certification document, to facilitate the use of results from certification and peer reviews that have already been done.

In addition to comments in the above four areas, a fifth group of commentors felt the scope and technical contents of Rev 10 were appropriate. An effort was made to retain valid Rev 10 requirements in the new format. To help reviewers compare Rev 12 with Rev 10, the tables of PRA element requirements also show, where applicable, the corresponding subsection number where a requirement appeared in Rev 10.

The following summary description of selected sections of Rev 12 may be helpful in showing the evolution from the previous draft to this draft of the standard.

Subsection 1.5 describes the characteristics of the three categories of applications in terms of:

(a) the extent of the reliance of the risk informed decision on the PRA;
(b) the required level of resolution/specificity of the PRA results relative to the needs of the specified applications within a given category;
(c) the degree of accuracy required of the PRA results;
(d) the degree of confidence in the results; and
(e) the safety significance of the application

<u>Section 2 (Definitions)</u> has been modified to address a large number of comments. Terms not used in the current draft have been eliminated. In several cases, where more than one definition of a term may be used within the technical community, the standard conforms to the term used in existing ASME codes and standards.

<u>Section 3 (Risk Assessment Application Process)</u> describes a process to determine the capability of a PRA to support a particular application of risk-informed decisionmaking. The process is intended for use with PRAs that satisfy the peer-review requirements specified in Section 6. In response to public comments on Rev 10, this section has been expanded and moved from the back to the front of the standard.

<u>Section 4.2</u> has been added to explain the "top down" approach used to derive the requirements appropriate to the three categories of applications.

<u>Section 4.6</u> on the use of expert judgment has been rewritten and much of the detail previously contained in this section and the associated Appendix A has been recommended for separate publication that may suitable for future reference in this standard.

<u>Section 6 (Peer Review Requirements)</u> has been revised to incorporate the use of the High Level Requirements of Section 4 by the peer review team in assessing the completeness of a PRA Element. It also cites NEI-00-02 as containing an acceptable peer review methodology.

Markley
1
6/28 RPRA
Subcommittee

Workshop and ACRS subcommittee meeting on Rev 12 of the ASME standard June 27-28, 2000

INTRODUCTION

Sid Bernsen
Chair, ASME Committee on Nuclear Risk Management

DEVELOPMENT PROCESS

- Use ASME redesign process
- Project Team for development
- early opportunity for review & comment
- approval by balanced committee of stakeholders - CNRM
- oversight by ASME Board on Nuclear Codes & Standards
- recognition by ANSI

SCOPE AND PURPOSE

- Level 1 PRA analysis of internal events
  - at power - excluding fires
- Limited Level 2 - Sufficient for LERF evaluation
- Developed to support
  - risk informed applications
  - use of existing PRAs
- Process for determining PRA ability to support an application and provides options for augmentation

PURPOSE OF CURRENT REVIEW

- resolution of your specific comments on Draft 10.
- acceptability of other changes
- recommendations for future consideration
- comments should be supported with basis/justification
- include proposed word changes, additions or deletions

ROLE OF PARTICIPANTS

- individual experts
- comments do not necessarily represent position of CNRM or ASME
- seeking feedback and recommendations
- position still on several issues still needs definition
- **We welcome your interest and input**

Markley
2
6/28 PPRA
Subcommittee

Workshop and ACRS subcommittee meeting on Rev 12 of the ASME standard
June 27-28, 2000

Major changes from the previous draft in response to public comments

Ron Simard
Chair, ASME Project Team

Rev 10 approach

- Specify a single set of requirements for Elements of a PRA that provides a realistic estimate of CDF

- Specify requirements for documentation, configuration control, peer review

- Describe a process for
  - **determining the extent to which the PRA Elements are necessary and sufficient to support a particular application**
  - **comparing the plant PRA to the Standard PRA**
  - **evaluating the significance to that specific application of any differences between the plant and Standard PRAs**

Rev 10 comments

- Prescriptiveness and perceived difficulty in applying the process
- need to distinguish among grades of application with a commensurate level of PRA capability
- need to recognize primary use of standard will be with existing PRAs
- need for closer alignment with the industry peer review and certification process

Rev 12 approach

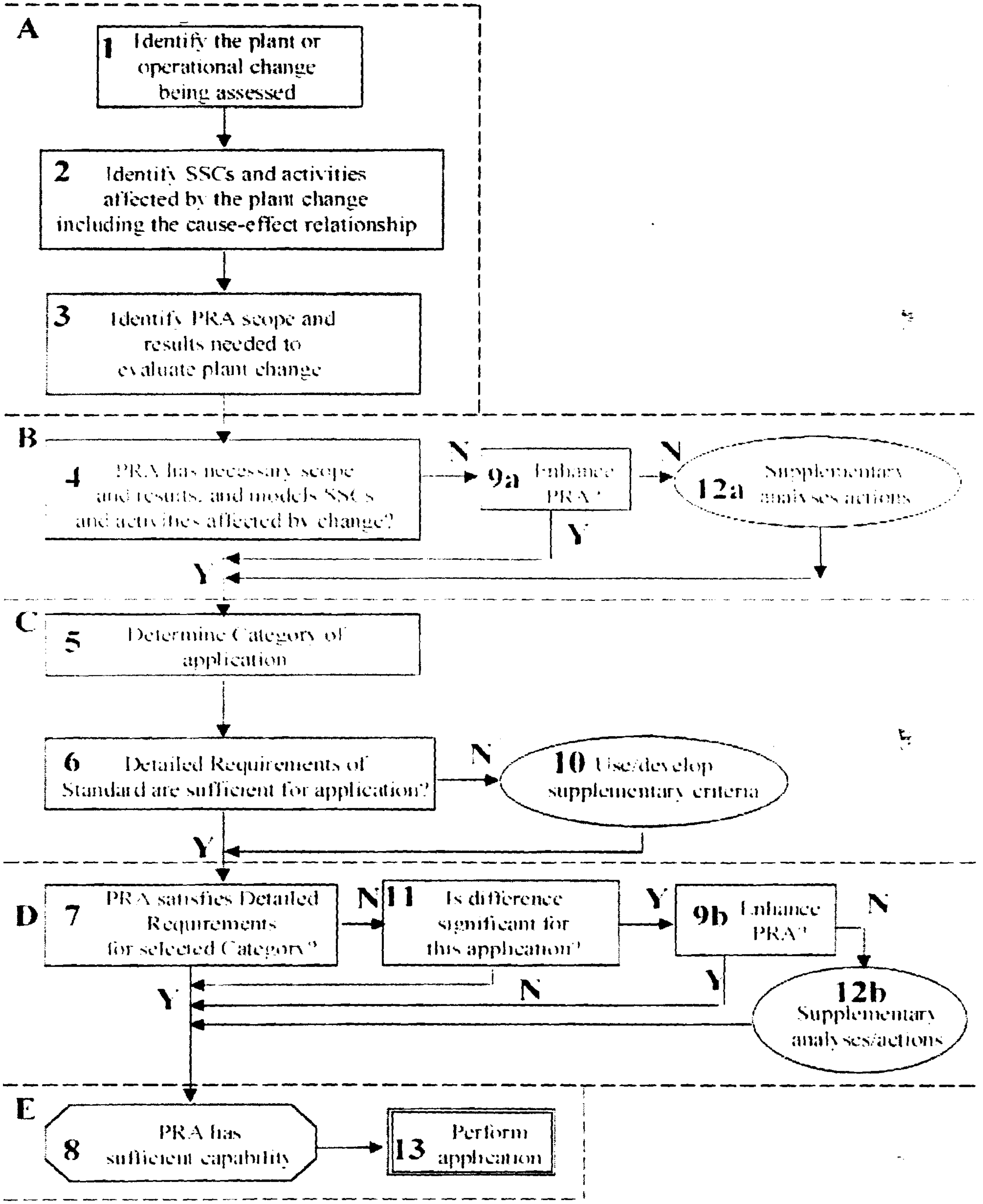
- **Significant restructuring, e.g.,**
  - process moved from back to front to emphasize intended use of the standard
  - mandatory appendix with generic data base removed
- **Range of possible risk informed applications approximated by three Categories**
- **Corresponding PRA capabilities presented in tables with three columns**
  - action statements whose scope of applicability varies across the three columns

Rev 12 approach (cont.)

- PRA Element requirements linked to industry certification process criteria, where possible
- peer review requirements reference the industry certification process methodology
- retention of Rev 10 requirements, where appropriate
- modification of the application process to make it easier to use

The application process (cont.)

- Define the application in terms of SSCs affected by the proposed change
- Determine if the scope and level of detail of the plant PRA is sufficient for the application (if not, enhance or supplement PRA)
- Determine the Category of the application and whether the level of detail in the standard is sufficient for the application (if not, use supplementary criteria)
- Compare the PRA to the appropriate requirements in the standard to determine whether the PRA has adequate capability to support the application
- If difference is significant, enhance or supplement PRA


A
1 Identify the plant or operational change being assessed
2 Identify SSCs and activities affected by the plant change including the cause-effect relationship
3 Identify PRA scope and results needed to evaluate plant change
B
4 PRA has necessary scope and results, and models SSCs and activities affected by change?
N
9a Enhance PRA?
N
12a Supplementary analyses/actions
Y
Y
C
5 Determine Category of application
6 Detailed Requirements of Standard are sufficient for application?
N
10 Use/develop supplementary criteria
Y
D
7 PRA satisfies Detailed Requirements for selected Category?
N
11 Is difference significant for this application?
Y
9b Enhance PRA?
N
Y
N
Y
12b Supplementary analyses/actions
E
8 PRA has sufficient capability
13 Perform application

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6/28 RPRA
Subcommittee

Workshop and ACRS subcommittee meeting on Rev 12 of the ASME standard June 27-28, 2000

Matching PRA Element capabilities and application characteristics

Karl Fleming
Member, ASME Project Team

Application Categories

- The standard is intended to be used in a wide range of applications
- Three broad Categories were used to develop and present the requirements of Section 4
- The plant PRA capabilities will not fall all into one Category
- For some applications ~~requirements in Section 4~~, the plant PRA may not have to meet any of the three Categories for a given set of elements

The Category of a given application depends on ...

1. Extent of the reliance of the risk informed decision on the PRA

Decisions are based ...

- Category I: primarily on deterministic analysis supplemented with risk insights
- Category II: ... on a balanced set of PRA insights and deterministic analyses Rev. 10
- Category III: ... primarily on PRA insights supplemented with little deterministic analyses

ation

RA

among broad
nitude CDF

k rank SSCs

k rank SSCs;
terminations;
/risk

The Category of a given application depends on ...

2. Required level of resolution of the PRA
   <u>results needed by applications</u>

- Category I: PRA products are used to differentiate among broad categories of safety significance using order of magnitude CDF and LERF estimates

- Category II: PRA products are used to prioritize/risk rank SSCs and to resolve risk contributors for risk significance determinations

- Category III: PRA products are used to prioritize/risk rank SSCs; to resolve risk contributions for risk significance determinations; and to achieve confidence in results when decision/risk acceptance criteria are approached

Scope of Coverage of High Level and Detailed Requirements

- Category I
  - Dominant accident sequences and contributors
  - Definition of dominant is to capture a major fraction that is sufficient to support intended applications
- Category II
  - Risk Significant accident sequences and contributors
  - Definition of risk significant is to capture sufficient fraction to support risk significant determinations in which PRA results are used supported by deterministic considerations
- Category III
  - Risk Significant accident sequences and contributors as well as non-risk significant sequences and contributors that are relevant to a Category III application
  - Definition of coverage of sequences and contributors is to capture sufficient fraction to support applications whose decisions are primarily based on PRA results are supported by deterministic considerations

The Category of a given application depends on ...

5. <u>Safety significance of the application</u>

- Category I: Typically do not impact safety related SSCs
- Category II: Expected to impact safety related SSCs
- Category III: Expected to impact safety related SSCs

4.3 PRA Elements and Attributes

Table 4.3-1 describes the attributes of PRA Elements appropriate to the three categories of applications described in Subsection 1.5.

TABLE 4.3-1 PRA ATTRIBUTES

ELEMENT		CATEGORY I	CATEGORY II	CATEGORY III
Initiating Events Analysis	IE	Identification and quantification of dominant accident initiating events	Identification and realistic quantification of risk significant accident initiating events	Identification and realistic quantification of initiating events
Accident Sequence Analysis	AS	Modeling of dominant core damage and large early release accident sequences	Modeling of risk significant core damage and large early release accident sequences	Modeling of core damage and large early release accident sequences
Success Criteria	SC	Bases and supporting analyses for establishing success or failure in dominant accident sequences	Realistic bases and supporting analyses for establishing success or failure in risk significant accident sequences	Realistic bases and supporting analyses for establishing success or failure for modeled accident sequences
Systems Analysis	SY	Modeling of key components and failure modes contributing to the function of systems expected to operate in dominant accident sequences	Realistic modeling of major components and failure modes contributing to the reliability and availability of systems expected to operate in risk significant sequences	Realistic modeling of components and failure modes contributing to the reliability and availability of systems expected to operate in modeled sequences
Human Reliability Analysis	HR	Modeling of major human actions (i.e., latent, response and recovery) with screening Human Error Probabilities (HEPs)	Realistic modeling of human actions (i.e., latent, response and recovery) with plant-specific HEPs in risk significant sequences	Realistic modeling of human actions (i.e., latent, response and recovery) with plant-specific HEPs
Data Analysis	DA	Quantification of point estimates for basic events, and associated parameters with generic data for dominant accident sequences	Realistic quantification of mean values for basic events, and associated parameters in a manner that accounts for relevant plant specific and generic data for risk significant sequences	Realistic quantification of risk significant basic events in a manner that quantifies impacts of uncertainties
Internal Flooding	IF	Modeling of dominant flood sequences	Realistic modeling of risk significant flood contributors	Realistic and thorough modeling of flooding contributors
Quantification	QU	Quantification of CDF and key contributors	Realistic quantification of CDF and key contributors supported by	Realistic quantification of CDF and risk significant contributors supported by a sound understanding and quantification of the impact of uncertainties

point estimates of CDF

mean values of CDF

full quantification of CDF

ELEMENT		CATEGORY I	CATEGORY II	CATEGORY III
		supported by an understanding of the impact of key uncertainties	a sound understanding of the impact of uncertainties	
Level 2 Analysis	L2	Quantification of LERF with an understanding of the impact of key uncertainties for the dominant LERF contributors	Realistic quantification of LERF with a sound understanding of the impact of uncertainties for risk significant accident sequences.	Realistic quantification of LERF supported by a sound understanding and quantification of the impact of uncertainties

Section 4 requirements

- High Level Requirements (HLRs) attempt to capture the important technical issues identified while drafting this standard
- HLRs apply to PRAs used with this standard for any application
- Supporting Requirements (SRs) are phrased as action statements that support the HLRs
- When an action statement extends to more than one Category, its scope of applicability varies as appropriate for applications in that Category

Table 4.4.2 HIGH LEVEL REQUIREMENTS FOR ACCIDENT SEQUENCE ANALYSIS (HLR-AS)

A	**Functional Sequence Categories** The *Accident Sequence Analysis* shall provide a reasonably complete set of scenarios that can lead to core damage following each initiating event or initiating event category defined in *Initiating Events Analysis*. These scenarios shall cover system responses and operator actions, including recovery actions, that support the key safety functions(2) necessary to prevent core damage, and shall be defined in a manner that supports the Level 1/Level 2 interface. (HLR-AS-A)
B	**Plant Specific CDF and LERF Quantification** The *Accident Sequence Analysis* shall provide a sequence definition structure that is capable of supporting plant specific quantification of the CDF, and LERF via the Level 1/Level 2 interface. (HLR-AS-B)
C	**Interface with Success Criteria** *Accident Sequence Analysis* shall provide an interface with the success criteria, mission times, and time windows needed to support each key safety function(2) represented in the modeled scenarios. (HLR-AS-C)
D	**Treatment Of Dependencies** Dependencies due to initiating events, human interface, functional dependencies, environmental and spatial impacts, and common cause failures shall be addressed. (HLR-AS-D)
E	**Documentation** The *Accident Sequence Analysis* shall be documented in a manner that facilitates PRA applications, updates, and peer review by describing the processes that were followed, with assumptions and bases stated. (HLR-AS-E)

(2) Key safety functions are the minimum set of safety functions that must be maintained to prevent core damage and large early release. These include, at a minimum, reactivity control, core heat removal, reactor coolant inventory control, reactor coolant heat removal, and containment bypass integrity in appropriate combinations to prevent core damage and large early release.

TABLE 4.4-2a SUPPORTING REQUIREMENTS FOR ACCIDENT SEQUENCE ANALYSIS HIGH LEVEL REQUIREMENT A
FUNCTIONAL SEQUENCE CATEGORIES: The *Accident Sequence Analysis* shall provide a reasonably complete set of scenarios that can lead to core damage following each initiating event or initiating event category defined in *Initiating Events Analysis*. These scenarios shall cover system responses and operator actions, including recovery actions, that support the key safety functions[2] necessary to prevent core damage, and shall be defined in a manner that supports the Level1/Level 2 interface. (HLR-AS-A)

Index No. AS	**CATEGORY I APPLICATIONS** Modeling of dominant core damage and large early release accident sequences	**CATEGORY II APPLICATIONS** Modeling of risk significant core damage and large early release accident sequences	**CATEGORY III APPLICATIONS** Modeling of core damage and large early release accident sequences
AS-A1 [AS-6] [3.3.2.2]	CHOOSE a method for *Accident Sequence Analysis* that explicitly models the appropriate combinations of system responses and operator actions that affect the key plant safety functions for each modeled initiating event. DEFINE and INCLUDE the critical safety functions that are assumed to be necessary to reach a safe stable state in the model.		
AS-A2 [AS-4] [3.3.2.2]	USE a method for *Accident Sequence Analysis* that : a) includes a reasonably complete set of event sequences involving core damage that could result from each modeled initiating event. b) considers the different plant responses and containment challenges that could result from each modeled initiating event; and c) provides a framework to support sequence quantification. d) reflects the initiating event categories defined in the *Initiating Events Analysis*	USE a method for *Accident Sequence Analysis* that : a) includes a reasonably complete set of event sequences involving core damage that could result from each modeled initiating event. b) models the different plant responses and addresses the containment challenges that could result from each modeled initiating event; and c) provides a framework to support sequence quantification. d) is explicitly traceable to the initiating event categories defined in the *Initiating Events Analysis*	
AS-A3 [AS-4]	DEFINE separate accident sequences as needed to address differences in timing, system success criteria, and operator actions.		
AS-A4 [AS-8]	ADDRESS a level of discrimination in the event tree structure that represents the key procedurally directed operator actions and delineates the differences in success criteria reflected in challenges to the critical safety functions.	DEVELOP a level of discrimination in the event tree structure that represents the key procedurally directed operator actions and delineates the differences in success criteria reflected in challenges to the critical safety functions.	
AS-A5 [AS-4] [3.3.2.2]	USE event trees or their equivalent to represent the accident sequence logic. JUSTIFY the use of alternatives to event trees (e.g., single top fault tree).		

(2) Key safety functions are the minimum set of safety functions that must be maintained to prevent core damage and large early release. These include, at a minimum, reactivity control, core heat removal, reactor coolant inventory control, reactor coolant heat removal, and containment bypass integrity in appropriate combinations to prevent core damage and large early release.

TABLE 4.4-2a SUPPORTING REQUIREMENTS FOR ACCIDENT SEQUENCE ANALYSIS HIGH LEVEL REQUIREMENT A

FUNCTIONAL SEQUENCE CATEGORIES: The *Accident Sequence Analysis* shall provide a reasonably complete set of scenarios that can lead to core damage following each initiating event or initiating event category defined in *Initiating Events Analysis*. These scenarios shall cover system responses and operator actions, including recovery actions, that support the key safety functions(2) necessary to prevent core damage, and shall be defined in a manner that supports the Level1/Level 2 interface. (HLR-AS-A)

Index No. AS	CATEGORY I APPLICATIONS Modeling of dominant core damage and large early release accident sequences	CATEGORY II APPLICATIONS Modeling of risk significant core damage and large early release accident sequences	CATEGORY III APPLICATIONS Modeling of core damage and large early release accident sequences
AS-A6 [AS-4] [3.3.2.2]	USE an acceptable event tree/fault tree method for interfacing the Accident Sequence Analysis with the Systems Analysis tasks. Acceptable approaches for event tree/fault tree modeling include. event trees with conditional split fractions(also referred to as event tree linking), and fault tree linking, both described in (Reference [4.4.2-1]). JUSTIFY the use of alternative approaches for this function.		
AS-A7 [3.3.2.4.1]	DEVELOP the event trees in sufficient detail to: a) determine which safety systems, functions, and operator actions have been challenged for each accident sequence b) determine whether core damage has occurred or core damage may be assumed initially in the PRA development c) identify the conditions needed to define the appropriate operator recovery actions and the necessary conditions for each sequence.		
AS-A8 [AS-4]	INCLUDE each necessary critical safety function in the quantitative model. JUSTIFY exceptions to the critical safety functions that are omitted from the model.		
AS-A9 [AS-7]	INCLUDE those relevant systems that support each critical safety function in the event sequence model in support of sequence quantification.		
AS-A10 [AS-8]	Transfers between event trees MAY be used to reduce the size and complexity of individual event trees. DEFINE any transfers that are used and the method that is used to implement them in the qualitative definition of accident sequences and in their quantification. USE a method for implementing an event tree transfer that preserves the dependencies that are part of the transferred sequence. These include functional, system, initiating event, operator, and spatial or environmental dependencies.		
AS-A11 [AS-8]	When event tree branching and event tree transfers are employed, DEVELOP the structure in a manner that maintains and unambiguously resolves the definition of success and failure paths.		
AS-A12 [3.3.2.4]	CONSIDER USING one or more accepted methods for developing and documenting the event sequence modeling process. Accepted methods include: a) functional and systemic event trees or both (as explained in Reference [4.4.2-1]) b) event sequence diagrams c) system dependency matrices	USE one or more accepted methods for developing and documenting the event sequence modeling process. Accepted methods include: a) functional and systemic event trees or both (as explained in Reference [4.4.2-1]) b) event sequence diagrams c) system dependency matrices	
AS-A13 [3.3.2.4]	INCLUDE a traceable interface between the event tree development process and the method or methods chosen from above or JUSTIFY use of alternative methods	INCLUDE a traceable interface between the event tree development process and the method or methods chosen from above.	

TABLE 4.4-2b SUPPORTING REQUIREMENTS ACCIDENT SEQUENCE ANALYSIS HIGH LEVEL REQUIREMENT B
PLANT SPECIFIC CDF AND LERF QUANTIFICATION: The accident sequence analysis shall provide a sequence definition structure that is capable of supporting plant specific quantification of the CDF and LERF via the Level 1/Level 2 interface. (HLR-AS-B)

Index No. AS	**CATEGORY I APPLICATIONS** Modeling of dominant core damage and large early release accident sequences	**CATEGORY II APPLICATIONS** Modeling of risk significant core damage and large early release accident sequences	**CATEGORY III APPLICATIONS** Modeling of core damage and large early release accident sequences
AS-B1 [AS-5]	INCLUDE models and analyses for *Accident Sequence Analysis* that are consistent with the as-built and as-operated plant. PERFORM realistic modeling of the as-built plant as supported by available information. Conservative modeling of the as-built plant MAY be performed to the extent that Category I applications are not distorted..	INCLUDE models and analysis for *Accident Sequence Analysis* that are consistent with the as-built and as-operated plant. PERFORM realistic modeling of the as-built plant as supported by available information.	
AS-B2 [AS-9]	DEFINE the success paths in the *Accident Sequence Analysis* that are logically consistent with the plant specific definition of core damage. Conservative treatment of success paths MAY be implemented only to the extent that Category I applications are not distorted by such conservative assumptions.	DEFINE the success paths in the *Accident Sequence Analysis* that are logically consistent with the definition of core damage and in a manner that supports a realistic and plant specific quantification of CDF.	
AS-B3 [AS-16]	INCLUDE models for repair and recovery that are based on data or accepted models applicable to the plant and that account for accident sequence dependencies such as time available, adverse environment, and lack of access, lighting, or room cooling. Conservative evaluations of repair and recovery MAY be incorporated only to extent that the relative risk significance of modeled SSCs is not distorted.	INCLUDE models for repair and recovery that are based on data or accepted models applicable to the plant and that account for accident sequence dependencies such as time available, adverse environment, and lack of access, lighting, or room cooling.	
AS-B4 [AS-19]	PROVIDE functions and structure of the event trees in a manner that is consistent with the plant specific EOPs and abnormal procedures.		

TABLE 4.4-2b SUPPORTING REQUIREMENTS ACCIDENT SEQUENCE ANALYSIS HIGH LEVEL REQUIREMENT B PLANT SPECIFIC CDF AND LERF QUANTIFICATION: The accident sequence analysis shall provide a sequence definition structure that is capable of supporting plant specific quantification of the CDF and LERF via the Level 1/Level 2 interface. (HLR-AS-B)

Index No. AS	CATEGORY I APPLICATIONS Modeling of dominant core damage and large early release accident sequences	CATEGORY II APPLICATIONS Modeling of risk significant core damage and large early release accident sequences	CATEGORY III APPLICATIONS Modeling of core damage and large early release accident sequences
AS-B5 [AS-19]	ACCOUNT FOR procedurally directed operator actions (both positive and negative impacts) that substantially influence the accident sequence progression or its probability in the accident sequence structure or the supporting fault tree analysis. INCORPORATE into the *Accident Sequence Analysis* the expected responses to an initiator as reflected in the plant emergency and abnormal operating procedures, training simulator exercises, and existing plant transient analysis. CHARACTERIZE the operator responses in a manner that is consistent with operator training and results of applicable simulator exercises. INCLUDE operator training input in the interpretation of proceduralized steps. INCLUDE operator actions that influence accident progression in the accident sequence model. Exceptions to this requirement MAY be taken only to the extent that Category I applications are not distorted.	ACCOUNT FOR procedurally directed operator actions (both positive and negative impacts) that substantially influence the accident sequence progression or its probability in the accident sequence structure or the supporting fault tree analysis. INCORPORATE into the *Accident Sequence Analysis* the expected responses to an initiator as reflected in the plant emergency and abnormal operating procedures, training simulator exercises, and existing plant transient analysis. CHARACTERIZE the operator responses in a manner that is consistent with operator training and results of applicable simulator exercises. INCLUDE operator training input in the interpretation of proceduralized steps. INCLUDE operator actions that influence accident progression in the accident sequence model.	
AS-B6 [AS-20, AS-22]	Clearly DEFINE the Level 1 end states as core damage or a safe stable state. USE a definition of core damage that is consistent with the requirements for *Success Criteria*		
AS-B7 [AS-20, AS-22]	RESOLVE other end states such as "core vulnerable" into core damage or safe stable states. ADDRESS the treatment of the impact of containment failure or vent on continued RPV makeup capability and basis for assumptions regarding ultimate end-state when such resolutions are made.		
AS-B8 [AS-20, AS-22]	Conservative definitions of core damage MAY be used only to the extent that Category I applications are not impacted.	DO NOT USE conservative definitions of core damage	

TABLE 4.4-2b SUPPORTING REQUIREMENTS ACCIDENT SEQUENCE ANALYSIS HIGH LEVEL REQUIREMENT B
PLANT SPECIFIC CDF AND LERF QUANTIFICATION: The accident sequence analysis shall provide a sequence definition structure that is capable of supporting plant specific quantification of the CDF and LERF via the Level 1/Level 2 interface. (HLR-AS-B)

Index No. AS	**CATEGORY I APPLICATIONS** Modeling of dominant core damage and large early release accident sequences	**CATEGORY II APPLICATIONS** Modeling of risk significant core damage and large early release accident sequences	**CATEGORY III APPLICATIONS** Modeling of core damage and large early release accident sequences
AS-B9 [AS-21]	USE a method for *Accident Sequence Analysis* that supports the development of an interface between Level 1 and Level 2 LERF analysis. To accomplish this, core damage sequences MAY be further developed by using accident sequence knowledge or information or consequence questions to unambiguously assign the modeled sequence to an appropriate plant damage state (PDS).		
AS-B10	USE Level 1 plant damage states that provide adequate information to support Level 2 analysis with minimal loss of information. If individual sequence cut sets are assigned to Plant Damage States (PDS), PROVIDE sufficient information to be able to remove ambiguities in mapping the basic event cutsets to unique PDS. Exceptions to this requirement MAY be made only to the extent that Category I applications are not distorted.	USE Level 1 plant damage states that provide adequate information to support Level 2 analysis with minimal loss of information. If individual sequence cut sets are assigned to Plant Damage States (PDS), PROVIDE sufficient information to be able to remove ambiguities in mapping the basic event cutsets to unique PDS.	
AS-B11 [AS-14]	Grouping of sequences into broader plant damage state categories MAY be performed only to the extent that Category I applications are not distorted. DO NOT GROUP sequences or plant damage states in a non-conservative manner (subsuming of sequences into broader categories not bounded by the worst case accident).	Grouping of sequences into broader plant damage state categories MAY be performed only to the extent that such grouping does not distort realistic CDF and LERF estimation. DO NOT GROUP sequences or plant damage states in a non-conservative manner (subsuming of sequences into broader categories not bounded by the worst case accident).	
AS-B12 [AS-15]	The Accident Sequence Analysis may be modeled using a single top event linked fault tree model. When this option is selected, DEVELOP such models in manner that meets all the technical requirements of this section. PROVIDE justification for any requirements that are not met or do not apply.		

TABLE 4.4-2c SUPPORTING REQUIREMENTS FOR ACCIDENT SEQUENCE ANALYSIS HIGH LEVEL REQUIREMENT C INTERFACE WITH SUCCESS CRITERIA: ***Accident Sequence Analysis*** **shall provide an interface with the success criteria, mission times, and time windows needed to support each key safety function [(2)] represented in the modeled scenarios. (HLR-AS-C)**

Index No. AS	CATEGORY I APPLICATIONS Modeling of dominant core damage and large early release accident sequences	CATEGORY II APPLICATIONS Modeling of risk significant core damage and large early release accident sequences	CATEGORY III APPLICATIONS Modeling of core damage and large early release accident sequences
AS-C1 [AS-17]	Based on the functional success criteria developed in *Success Criteria*, INCLUDE a reasonably accurate treatment of the functional requirements associated with the plant-specific safety functions, system capabilities and system interactions, procedural guidance to operators, and the timing of events within the *Accident Sequence Analysis* for each modeled initiating event category.		
AS-C2 [AS-18]	IDENTIFY the information sources used as the basis for the *Accident Sequence Analysis* including: (a) system analysis and system dependencies (b) success criteria, plant thermal hydraulics, and plant transient response (c) plant operating procedures and practices.		
AS-C3 [AS-18]	PROVIDE a sequence definition that is based on realistic thermal hydraulic analyses to support the success criteria used in the *Accident Sequence Analysis*. Conservative analyses MAY be used only to the extent that Category I applications are not distorted.	PROVIDE a sequence definition that is based on realistic thermal hydraulic analyses to support the success criteria used in the *Accident Sequence Analysis*. Conservative analyses MAY be used only to the extent that realistic estimates of CDF and LERF are not distorted.	
AS-C4	DEVELOP and SPECIFY the success criteria in a manner that shows an interface with the definition of core damage and PDS, definition of plant safety functions needed to prevent core damage or PDS, and the boundary conditions for the systems analysis. INCLUDE a definition of the success criteria and mission time for each event tree top event. If multiple success criteria and mission times are needed for the same event tree top event, PROVIDE this information for each case.		
AS-C5 [AS-23]	INCLUDE in the definition of success criteria for sequences terminating with no core damage, a mission of at least 24 hours with stable plant conditions or an appropriate representation for accident sequences with unstable conditions that is consistent with the sequence end-state. JUSTIFY and PROVIDE any mission times less than 24 hours for stable sequences and all assumed mission times for all unstable sequences.		

(2) Key safety functions are the minimum set of safety functions that must be maintained to prevent core damage and large early release. These include, at a minimum, reactivity control, core heat removal, reactor coolant inventory control, reactor coolant heat removal, and containment bypass integrity in appropriate combinations to prevent core damage and large early release.

TABLE 4.4-2d SUPPORTING REQUIREMENTS FOR ACCIDENT SEQUENCE ANALYSIS HIGH LEVEL REQUIREMENT D
TREATMENT OF DEPENDENCIES: Dependencies due to initiating events, human interface, functional dependencies, environmental and spatial impacts, and common cause failures shall be addressed. (HLR-AS-D)

Index No. AS	CATEGORY I APPLICATIONS Modeling of dominant core damage and large early release accident sequences	CATEGORY II APPLICATIONS Modeling of risk significant core damage and large early release accident sequences	CATEGORY III APPLICATIONS Modeling of core damage and large early release accident sequences
AS-D1 [AS-5] [3.3.2.4.1]	PROVIDE a sequence model development with a clear interface with the system analysis and dependency evaluation tasks of the PRA.		
AS-D2 [AS-10] [3.3.2.4.1]	INCLUDE a visible and a reasonably accurate treatment of dependencies and interfaces among the plant safety functions, system responses, and operator actions needed for accident mitigation in the *Accident Sequence Analysis*. These dependencies include functional, phenomenological, and operational dependencies and interfaces. IDENTIFY dependencies among all modeled event tree top events and INCLUDE these quantitatively in the model.		
AS-D3 [AS-11] [3.3.2.3]	PROVIDE a systematic evaluation of dependencies, such as that provided by dependency matrices. When using dependency matrices for this purpose INCLUDE a matrix or set of matrices that accounts for: a) initiating event to system dependencies b) dependencies among support systems c) dependencies between support and front line systems; d) dependencies among front line systems that support key safety functions (2) PROVIDE an event sequence model that realistically treats, and consistently applies, to capture the dependencies among event tree top events.		

(2) Key safety functions are the minimum set of safety functions that must be maintained to prevent core damage and large early release. These include, at a minimum, reactivity control, core heat removal, reactor coolant inventory control, reactor coolant heat removal, and containment bypass integrity in appropriate combinations to prevent core damage and large early release.

TABLE 4.4-2d SUPPORTING REQUIREMENTS FOR ACCIDENT SEQUENCE ANALYSIS HIGH LEVEL REQUIREMENT D
TREATMENT OF DEPENDENCIES: Dependencies due to initiating events, human interface, functional dependencies, environmental and spatial impacts, and common cause failures shall be addressed. (HLR-AS-D)

Index No. AS	CATEGORY I APPLICATIONS Modeling of dominant core damage and large early release accident sequences	CATEGORY II APPLICATIONS Modeling of risk significant core damage and large early release accident sequences	CATEGORY III APPLICATIONS Modeling of core damage and large early release accident sequences
AS-D4 [AS-10]	INCLUDE the following types of accident sequence dependencies: Functional: Functional failures, e.g.: a) LOCA initiator causes debris clogging of ECCS Suction b) turbine driven system dependency on SORV, depressurization, and containment heat removal (suppression pool cooling). c) low pressure system injection success dependent on need for RPV depressurization. Intra and Intersystem: Common cause failures and functional dependencies between systems. IDENTIFY system dependencies, dependency matrices, and/or linked fault trees. Human: Adverse environment or sequence timing influences on operator actions. Spatial/Environmental/Phenomenological: Spatial/Environmental dependencies that may result from initiating events and subsequent sequences. Example of Phenomenological dependencies: These dependencies manifest themselves when the environmental conditions generated during an accident sequence influence the operability of equipment or the capability of the operators to implement procedures and recovery actions. Examples of phenomenological impacts include generation of harsh environments that actuate protective trip circuits, loss of pump net positive suction head (NPSH), clogging of flow paths, and consequential effects of other failures.		
AS-D5 [AS-10]	INCLUDE dependencies between the initiating event and mitigating systems as well as dependencies between and among the mitigating systems and operator actions. ACCOUNT for dependencies between the initiating event and mitigating systems, including immediate (e.g. loss of electric power) and delayed responses (e.g., loss of room cooling) in the accident sequence model or reflected in the system logic models. Dependencies among mitigating systems and operator actions MAY also be modeled in the accident sequence model or the system logic models.		
AS-D6 [3.3.2.4.1]	When developing the event sequence structure, ORDER the event tree top events representing the response of systems and post initiator operator actions sequentially according to the timing of the events along the sequence to ensure proper treatment of time dependencies.		
AS-D7 [3.3.2.4.1]	When the event trees with conditional split fraction method is used, if the probability of Event B is dependent on the occurrence or non-occurrence of Event A, PLACE Event A to the left of Event B in the ordering of event tops.		

TABLE 4.4-2d SUPPORTING REQUIREMENTS FOR ACCIDENT SEQUENCE ANALYSIS HIGH LEVEL REQUIREMENT D
TREATMENT OF DEPENDENCIES: Dependencies due to initiating events, human interface, functional dependencies, environmental and spatial impacts, and common cause failures shall be addressed. (HLR-AS-D)

Index No. **AS**	**CATEGORY I APPLICATIONS** Modeling of dominant core damage and large early release accident sequences	**CATEGORY II APPLICATIONS** Modeling of risk significant core damage and large early release accident sequences	**CATEGORY III APPLICATIONS** Modeling of core damage and large early release accident sequences
AS-D8 [3.3.2.4.1]	For the event trees with conditional split fraction method, DEVELOP the event trees to a level of detail sufficient to identify intersystem dependencies and train level interfaces. For the fault tree linking method, DEVELOP fault trees and apply flag settings and mutually exclusive files or comparable method to resolve these same dependencies. If plant configurations and maintenance practices create dependencies among various system alignments, DEFINE and MODEL these configurations and alignments in a manner that reflects these dependencies. PROVIDE one event sequence model or set of event trees that accounts for each initiating event or initiating event category defined in the *Initiating Event Analysis* element so that initiating event dependencies can be properly modeled.		
AS-D9 [AS-12]	PROVIDE an explicit model of the Pump seal LOCA in the *Accident Sequence Analysis* when applicable. PROVIDE the basis for the model.		
AS-D10 [AS-13]	INCLUDE in the *Accident Sequence Analysis* and quantified model an explicit and realistic treatment of dependencies introduced by the time phasing of the event progression. A conservative treatment of time phasing MAY be used to the extent that Category I applications are not distorted.	INCLUDE in the *Accident Sequence Analysis* and quantified model an explicit and realistic treatment of dependencies introduced by the time phasing of the event progression. A conservative treatment of time phasing MAY be used to the extent that realistic estimates of CDF and LERF are not distorted.	

TABLE 4.4-2d SUPPORTING REQUIREMENTS FOR ACCIDENT SEQUENCE ANALYSIS HIGH LEVEL REQUIREMENT D
TREATMENT OF DEPENDENCIES: Dependencies due to initiating events, human interface, functional dependencies, environmental and spatial impacts, and common cause failures shall be addressed. (HLR-AS-D)

Index No. AS	CATEGORY I APPLICATIONS Modeling of dominant core damage and large early release accident sequences	CATEGORY II APPLICATIONS Modeling of risk significant core damage and large early release accident sequences	CATEGORY III APPLICATIONS Modeling of core damage and large early release accident sequences
AS-D11 [AS13]	INCLUDE events for which time phased dependencies could be introduced. For SBO/LOOP sequences , INCLUDE key time phased events such as: • AC power recovery • DC battery adequacy (time dependent discharge) • Environmental conditions (e.g., room cooling) for operating equipment and the control room For ATWS/failure to scram events, INCLUDE key time dependent actions such as: • SBLC initiation • RPV level control • ADS inhibit Other events that MAY be subject to explicit time dependent characterization include: • CRD as an adequate RPV injection source • Long term make-up to RWST		

TABLE 4.4-2d SUPPORTING REQUIREMENTS FOR ACCIDENT SEQUENCE ANALYSIS HIGH LEVEL REQUIREMENT D
TREATMENT OF DEPENDENCIES: Dependencies due to initiating events, human interface, functional dependencies, environmental and spatial impacts, and common cause failures shall be addressed. (HLR-AS-D)

Index No. AS	CATEGORY I APPLICATIONS Modeling of dominant core damage and large early release accident sequences	CATEGORY II APPLICATIONS Modeling of risk significant core damage and large early release accident sequences	CATEGORY III APPLICATIONS Modeling of core damage and large early release accident sequences
AS-D12 [AS-13]	As part of the time dependence assessment, ADDRESS the following: • Mission time of diesel generators • Mission time of RPT, ARI, scram system • Time to core uncovery		
AS-D13 [AS-15] [3.3.2.4.1]	To model the changing nature of certain sequences, ACCOUNT for operational dependencies. ACCOUNT for interfaces when sequences are modeled in multiple event trees with transfers. Example of event progression: In developing sequences for a transient initiating event in which the reactor coolant boundary is initially intact, event progression may lead to sequences in which reactor coolant system safety or relief valves open such that a transient induced LOCA condition is created.		
AS-D14 [AS-15]	When transfers are being employed, INCLUDE Transfers among event trees explicitly in the quantification except for cases that are noted in the documented descriptions of the sequences to address dependencies properly. PRESERVE the appropriate dependencies, both hardware and human related, from the original event sequence model across the transfer interfaces.		

TABLE 4.4-2e SUPPORTING REQUIREMENTS FOR ACCIDENT SEQUENCE ANALYSIS HIGH LEVEL REQUIREMENT E DOCUMENTATION: The accident sequence analysis shall be documented in a manner that facilitates PRA applications, updates, and peer review by describing the processes that were followed, with assumptions and bases stated. (HLR-AS-E)

Index No. AS	CATEGORY I APPLICATIONS Modeling of dominant core damage and large early release accident sequences	CATEGORY II APPLICATIONS Modeling of risk significant core damage and large early release accident sequences	CATEGORY III APPLICATIONS Modeling of core damage and large early release accident sequences
AS-E1 [AS-25]	DOCUMENT the results of the Accident Sequence Analysis consistent with the process that was used for its development. PROVIDE the basis for the accident sequence process.		
AS-E2 [AS-26]	DOCUMENT the results of independent reviews of the *Accident Sequence Analysis* and the qualifications of the reviewers.		
AS-E3 [AS-26]	DOCUMENT the treatment of each initiator and event tree to support reviews and applications.		
AS-E4	DOCUMENT interfaces between *Accident Sequence Analysis* and other PRA tasks. INCLUDE the following interfaces in the documentation: • a link between the definition of initiating event category in the Initiating Event Analysis Task and the event sequence model • the definition of core damage and associated success criteria that is consistent with that documented in the Success Criteria Task • key definitions of operator actions and sequence specific timing and dependencies reflected in the event trees that is traceable to the HRA for these actions • the basis for the sequence and cutset quantification in the Level 1 Quantification And Interpretation of Results Task • a framework for an integrated treatment of dependencies in the initiating events analysis, systems analysis, data analysis, human reliability analysis, Level 1 quantification, and Level 2 LERF quantification PRA elements.		

TABLE 4.4-2e SUPPORTING REQUIREMENTS FOR ACCIDENT SEQUENCE ANALYSIS HIGH LEVEL REQUIREMENT E

DOCUMENTATION: The accident sequence analysis shall be documented in a manner that facilitates PRA applications, updates, and peer review by describing the processes that were followed, with assumptions and bases stated. (HLR-AS-E)

Index No. AS	**CATEGORY I APPLICATIONS** Modeling of dominant core damage and large early release accident sequences	**CATEGORY II APPLICATIONS** Modeling of risk significant core damage and large early release accident sequences	**CATEGORY III APPLICATIONS** Modeling of core damage and large early release accident sequences
AS-E5	DOCUMENT a) a description of events and the end states included in the development of the models b) the success criteria for each modeled event c) the actual models.		
AS-E6	DOCUMENT: a) the success criteria established for each initiating event category including the bases for the criteria (i.e., the system capacities required to mitigate the accident and the necessary components required to achieve these capacities); b) the models used (including all sequences) for each initiating event category c) a description of the accident progression for each sequence or group of similar sequences (i.e., descriptions of the sequence timing, applicable procedural guidance, expected environmental or phenomenological impacts, dependencies between systems and operator actions, and other pertinent information required to fully establish the sequence of events); d) any assumptions that were made in developing the accident sequences, as well as the bases for the assumptions and their impact on the final results; e) existing analyses or plant-specific calculations performed to arrive at success criteria and expected sequence phenomena including necessary timing considerations; f) sufficient system operation information to support the modeled dependencies; g) calculations or other bases used to justify equipment operability beyond its "normal" design parameters and for which credit has been taken; and h) description of the interface of the accident sequence models with PDSs. i) how all requirements for *Accident Sequence Analysis* have been satisfied when sequences are modeled using a single top event linked fault tree.		

References

[4.4.2-1] NUREG/CR-4550, Vol. 1 Rev. 1, A Analysis of Core Damage Frequency: Internal Events Methodology, pp 4-1 to 4-22, January 1990

Markley
(1)

Risk-Informed Part 50 Option 2

Presentation for the ACRS Subcommittee on
Probabilistic Risk Assessment
June 29, 2000

Agenda

- ANPR comments
- Preliminary staff views on industry guideline and PRA peer certification process
- Status/Schedule

ANPR Comments

- 11 comment letters, over 200 comments
  - Licensees and industry groups (6)
  - Law firms (2)
  - Consulting firms (1)
  - Professional societies (1)
  - Public (1)

ANPR Comments - continued

Approach

- General agreement on the list of rules identified, with a proposal to risk-inform them in a phased approach
- Be performance-based, optional, and allow for selective implementation
- Limited NRC prior review and approval
- Backfit rule should be applied to Option 2

Categorization & Treatment Guideline - continued

Treatment - continued

- Staff is developing guidance for review of the STP exemption
- Staff to develop Option 2 treatment acceptance criteria
- Level of agreement between STP and NEI proposals

Status/Schedule

RIP50, Option 2

- August 2000 - ANPR comments & issues paper
- September 2000 - Commission briefing
- December 2000 - final acceptance criteria
- January 2001 - initiate pilot program
- August 2001 - proposed rulemaking to Commission
- December 2002 - final rulemaking to Commission

Markley
1A

June 19, 2000

MEMORANDUM TO: Ashok C. Thadani, Director
Office of Nuclear Regulatory Research

FROM: Samuel J. Collins, Director ***/RA Signed by S. Collins/***
Office of Nuclear Reactor Regulation

SUBJECT: REQUEST FOR ASSISTANCE IN REVIEW OF NEI 00-02, "PROBABILISTIC RISK ASSESSMENT PEER REVIEW PROCESS GUIDANCE" (TAC NO. MA8899)

We request the assistance of the Office of Nuclear Regulatory Research (RES) in the review of NEI 00-02, "Probabilistic Risk Assessment Peer Review Process Guidance," submitted by the Nuclear Energy Institute (NEI) on April 24, 2000. NEI has requested review of this document for applicability to the risk-informed categorization and treatment of nuclear plant equipment as described in SECY-99-256. Since the quality required of a probabilistic risk assessment (PRA) is directly related to the application for which the PRA results and insights are to be applied, NEI 00-02 will be reviewed in conjunction with NEI's *Industry Guideline for Risk-Informed Categorization and Treatment of Structures, Systems, and Components*, and with the staff's draft version of Appendix T to 10 CFR Part 50.

This memorandum documents our specific needs for your assistance. Review tasks are discussed below. Note that some of these tasks contain subtasks that may not be directly related to the review of NEI 00-02, but are related to establishing guidance on how the NRC staff is to use the results of the PRA peer review process. This review scope accommodates situations where there may be compensatory measures (or "tradeoffs") which can be used by a licensee when certain elements of the PRA do not fully conform to staff expectations.

REQUESTED ACTIONS

The outline of the overall staff review is described in the attachment to this memorandum. Based on discussions between the Office of Nuclear Reactor Regulation (NRR) and RES staff, we request that RES review the PRA technical elements and requirements given in NEI 00-02 to determine if they provide sufficient information for categorization of structures, systems, and components (SSCs) for application to the risk-informing of 10 CFR Part 50 (RIP 50) Option 2 effort. High-level characteristics and attributes required for an acceptable PRA should be used as the basis for this review. We also request that RES review the NEI 00-02 subtier criteria against typical industry and NRC good practices as reflected in various guidelines including the proposed ASME PRA standard. Review results should address discrepancies and their potential impact on Option 2 activities. This request corresponds to Task 2 of the attached outline. NRR staff will take the lead for Tasks 1, 3, and 4 which address the application of the PRA Certification process to RIP 50 Option 2.

Outline for Review of NEI 00-02

Probabilistic Risk Assessment Peer Review Process Guidance

Task 1: Process Review

a. Review the objectives, the mechanics of the peer review process, review team qualifications, required documentation, etc., to determine if the process is consistent with staff expectations of the characteristics and attributes of a peer review process.

b. Determine if the elements of the review process for determining "quality assurance" of the PRA are consistent with the requirements provided in Section 2.5 of Regulatory Guide 1.174.

Task 2: Review the technical elements and requirements for application to Option 2.

a. Determine if the technical requirements in NEI 00-02 are sufficient to provide assurance that the staff's high level expectations for the "characteristics and attributes of an acceptable PRA" can be satisfied.

b. Review the subtier criteria for "Grade 3" PRAs and compare to typical industry and NRC good practices as reflected in various guidelines including the ASME PRA standard. Document the differences. Provide relevance of the differences with respect to RIP 50 Option 2 applications.

c. Provide insights into other applications which a "Grade 3" PRA will support and the applications that it may not be good enough to support.

Task 3: Review the requirements for SSC categorization as required by RIP 50 Option 2. Determine the quality of PRA needed in light of the other requirements of the RIP 50 Option process.

a. Review the draft Appendix T requirements as well as NEI's categorization guidance document. From these documents:

i) define the decision to be made;

ii define the decision-making process, specifying the role of PRA results (what results are to be used, and how are they to be used); and

iii identify what is needed of the PRA to give confidence in the results in the context of the decision.

b. In conjunction with the findings of Tasks 2(b) and 3(a) above, determine if a PRA for which the peer review team has assigned a "Grade 3" for all its elements, can be used for the categorization of SSCs in the context of Option 2. Perform this review in light of: the risk exposure (e.g., backstops, controls, extent of change permitted, etc.); performance monitoring requirements (e.g., measures and criteria, timely detection and corrective action, margin to safety, etc.); use of traditional engineering analyses (e.g., defense-in-depth, safety margins, issue-specific engineering analyses, licensing basis calculations, etc.); and use of an integrated decision-making panel to appropriately utilize the PRA insights.

Note that, not all review elements have to be assigned a Grade 3 or higher for the PRA to be usable for Option 2. Some elements may be determined to be unimportant for Option 2 applications. Even if important elements (as defined by Task 2(b)) are non-conforming, there may be "tradeoffs" that a licensee may choose, e.g., when a PRA element does not meet a certain requirement, there could be different mechanisms to compensate for this non-conformance. Task 3(c) discusses the application-specific tradeoffs (i.e., tradeoffs that would apply for all applications in RIP 50 Option 2), and Task 3(d) discusses the decision-specific tradeoffs (i.e., tradeoffs that could result because of differences and variations in the plant-specific PRAs).

c. Define measures which could be used to compensate for cases when NEI 00-02 review elements are not consistent with staff expectations.

   i) Define sensitivity studies and other deterministic approaches that could be used in place of "consensus" PRA approaches (e.g., seal LOCA modeling, use of the MAAP code, etc.).

   ii) Determine if the sensitivity studies as currently specified in Appendix T and in NEI's categorization guidance document are sufficient to compensate for the non-use of consensus approaches in HRA modeling, CCF modeling and parameter estimation.

d. In the review of Option 2 applications, it is expected that the staff will have to address variations (on a plant-to-plant basis) in the level of conformance to the NEI 00-02 guidelines. For PRA elements that do not conform to "Grade 3" requirements and which are amenable to tradeoffs, define guidance for the staff review of these tradeoffs (e.g., use of conservatism, more reliance in defense-in-depth or margins, better monitoring, etc.).

Task 4: Review the documentation requirements (and define level of staff review)

a. Using the NEI 00-02 documentation requirements, determine the peer review documentation that should be included as part of the Option 2 submittal to t he NRC, and the documentation that should be available at the plant site and available for NRC audit. Suggest additional documentation requirements if necessary.

b. Relate the level of NRC review for Option 2 submittals to the results obtained from the peer review of the PRA supporting that submittal. Under what conditions is the " no-prior staff review and approval" option feasible?

Markley (1B)

Station Support Department

(10)

PECO ENERGY

DOCKETED
USNRC

'00 MAY 25 A8:01

PECO Energy Company
Nuclear Group Headquarters
965 Chesterbrook Boulevard
Wayne, PA 19087-5691

May 17, 2000

OF...
R...
ADJ...

Secretary
U.S. Nuclear Regulatory Commission
Attn: Rulemakings and Adjudications Staff
Washington, DC 20555-0001

DOCKET NUMBER
PROPOSED RULE PR 21, 50, 52, 54 & 100
(65FR11488)

Subject: Comments Concerning "Risk-Informing Special Treatment Requirements" (65FR11488, dated March 3, 2000)

Dear Sir/Madam:

This letter is being submitted in response to the Nuclear Regulatory Commission's (NRC) request for comments concerning "Risk-Informing Special Treatment Requirements," which was published in the Federal Register (i.e., 65FR11488, dated March 3, 2000). The NRC is considering new regulations that would provide an alternative risk-informed approach for special treatment requirements in the current regulations. This action is a result of the Commission's continuing efforts to risk-inform its regulations.

PECO Energy appreciates the opportunity to comment on the petition for rulemaking. PECO Energy supports the comments submitted on behalf of the nuclear energy industry, by the Nuclear Energy Institute.

If you have any questions, please do not hesitate to contact us.

Very truly yours,

James A. Hutton, Jr.
Director - Licensing

TEMPLATE = SECY-067

SECY-02

Markley
(2)

Advisory Committee on Reactor Safeguards
Subcommittee on Probabilistic Risk Assessment

Petition for Rulemaking
Combustible Gas Control

June 29, 2000
Two White Flint, Rockville, MD

Bob Christie

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Agenda

A. Letter from Bob Christie to Tom King, 5/30/00

B. Introduction/Background

C. San Onofre Task Zero Safety Evaluation Report

D. Other Exemption Requests

E. Key Points

F. Petition for Rulemaking
   1. 10CFR50, Appendix A, GDC 41
   2. 10CFR50.44

G. Summary

2/23

RJC
5/30/00

Attachment to Letter from Bob Christie, Performance Technology, to Dr. Tom King, Office of Research, dated 5/30/00

Slide 23

Agreement:

The hydrogen monitoring system can be commercial grade and not "safety-related."

Disagreement:

I believe that there should be no NRC requirements for hydrogen monitoring. The nuclear units may continue to have equipment for hydrogen monitoring for severe accident management but this equipment is not "safety significant" and should have no NRC requirements. Hydrogen concentration is not a primary indicator but rather only a confirmatory indicator. I do not believe that the hydrogen monitors have any significant impact of "reasonable assurance of adequate protection of public health and safety."

It appears that the NRC staff believes there should be NRC requirements for hydrogen monitoring in the long term and, while the monitors would not be "safety related" but rather commercial grade, the hydrogen monitors would still have to meet some "functional" requirements in the long term and be subject to NRC inspection and enforcement. As indicated above, I disagree with this position.

Slide 24

Agreement:

Containment air mixing should continue to be covered by other regulations with no changes. No changes should be made to containment air mixing systems.

Slide 25

Agreement:

Remove post LOCA hydrogen control from 10CFR50.44.

RJO
5/30/00

Slide 26

Agreement:

All nuclear reactors should continue to have high point vents as currently called for in the regulations.

Slide 27

Agreement:

Mark I and Mark II Boiling Water Reactors should continue to remain inerted as currently called for in the regulations.

Slide 28

I am unsure what agreement or disagreement exists because this slide was not clear as to what was being discussed. I have included some words in my petition for rulemaking regarding the capability of large dry containments during severe accidents. I do not know whether the NRC staff believes that my words are the wrong words and they want to change my words or add words to what I proposed, or exactly what is the concern of the NRC staff. This slide needs better definition as to what is being discussed.

To me it is not clear exactly what the NRC staff is concerned about with respect to Station Blackout at the ice condenser plants and Mark III Boiling Water Reactors. In any case, I believe any additional requirements on the igniters for Mark III Boiling Water Reactors and ice condenser plants should be addressed by the backfit process, 10CFR50.109.

Objective - Pilot Programs

The objective of the pilot programs will be to demonstrate a more objective an efficient way to maintain adequate protection of public health and safety, to promote the common defense and security, ant to protect the environment than the present detailed prescriptive regulatory process.

Integrated Approach

"Whole Plant"

Cost

Generation

Risk

BASIS

- The primary responsibility for the "public health and safety" of a nuclear unit lies with the people at the site who are running the nuclear unit.
- The regulatory process that oversees the nuclear unit must ensure "adequate protection of public health and safety."

PUBLIC HEALTH RISK

1. Is different for each nuclear unit.

2. Changes with time.

8/23

Dr. Thomas Pigford, Kemeny Report, October 1979, Separate views.

16. The Major Problems with NRC's Approach to Reactor Safety

The Commission (Kemeny) report has identified many mistakes by NRC personnel in their handling of the TMI-2 accident and deficiencies in NRC's regulatory practices. However, this criticism does not reach some essential elements of the problem. I believe that the following are some of the more important problems at NRC:

... Lack of quantified safety goals and objective. When a safety concern is postulated, there is no yardstick to judge the adequacy of mitigating measures.

... Inability to set priorities and to allocate resources in proportion to the estimated risk to the public. In my view, a disproportionate effort is being required for some issues which have only a marginal impact upon risk to the public.

... Lack of experienced staff. An undesirably large proportion of NRC staff and management have little or no practical experience in designing or operating the equipment which they regulate.

... Arbitrary requirements. Too many of the NRC requirements are mandated without valid technical back-up and value-impact analysis.

... A stifling adversary approach. The existing process inhibits the interchange of technical information between the NRC and industry. It discourages innovative engineering solutions.

... Ineffective evaluation of operations. NRC has no effective system for evaluating data from operating plants. Data should be analyzed systematically to identify trends and patterns.

... Lack of a comprehensive system approach to the whole plant. A large percentage of the NRC staff are specialists focusing upon narrow topics. There are relatively few systems engineers within NRC who can integrate individual safety features into an overall concept and who can place issues into perspective.

... An overwhelming emphasis on conservative models and assumptions. Realistic analyses are needed to identify the margins of safety and to aid competent decisions.

ISSUES FOR NUCLEAR PLANTS IN A DEREGULATED ELECTRIC UTILITY INDUSTRY

by

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AMERICAN NUCLEAR SOCIETY

INTERNATIONAL TOPICAL MEETING ON
SAFETY OF OPERATING REACTORS
SAN FRANCISCO, CALIFORNIA

OCTOBER 11-14, 1998

10/23

Excerpt from the San Onofre Task Zero Safety Evaluation Report:

"The overall public risk and radiological consequences from reactor accidents is dominated by the more severe core damage accidents that involved containment failure or bypass."

11/23

Excerpts from the San Onofre Task Zero Safety Evaluation Report:

"Subsequent risk studies have shown that the majority of risk to the public is from accident sequences that lead to containment failure or bypass, and that the contribution to risk from accident sequences involving hydrogen combustion is quite small."

"As mentioned in the previous section, the risk associated with hydrogen combustion is not from design-basis accidents but from severe accidents."

Excerpts from the San Onofre Task Zero Safety Evaluation Report:

"Although the recombiners are effective in maintaining the Regulatory Guide 1.7 hydrogen concentration below the lower flammability limit of 4 volume percent, they are overwhelmed by the larger quantities of hydrogen associated with severe accidents which are typically released over a much shorter time period (e.g., 2 hours)."

"From this information, the NRC staff concludes that the quantity of hydrogen, prescribed by 10CFR50.44(d) and Regulatory Guide 1.7, which necessitates the need for hydrogen recombiners and its backup the hydrogen purge system is bounded by the hydrogen generated during a severe accident. The NRC staff finds that the relative importance of hydrogen combustion for large, dry containments with respect to containment failure to be quite low. This finding supports the argument that the hydrogen recombiners are insignificant from a containment integrity perspective."

Excerpt from the San Onofre Task Zero Safety Evaluation Report:

"In a postulated Loss of Coolant Accident, the San Onofre Nuclear Generating Station Units 2 and 3 Emergency Operating Instructions direct the control room operators to monitor and control the hydrogen concentration inside the containment after they have carried out the steps to maintain and control the higher priority critical safety functions. The key operator actions in controlling the hydrogen concentration are to place the hydrogen recombiners or hydrogen purge system in operation which involves many procedural steps. These hydrogen control activities could distract operators from more important tasks in the early phases of accident mitigation and could have a negative impact on the higher priority critical operator actions."

Key Points - Combustible Gas Control

Public Health Risk

Severe Accidents - Not Design Basis Accidents

Containment integrity when fission products present

Existing hydrogen recombiners and purge ineffective

Existing procedures can distract operators

Combustible Gas Control Configurations

Unit	Monitors	Hydrogen % action level	Design pressure Failure pressure (psig)	Repressuriztion	Purge	Permanent Recombiners	Movable Recombiners
Unit 1	90 minutes	3.5%	59/153	NA	NA	primary inside containment	NA
Unit 2	90 minutes	3.5%	55/140	primary portable blowers 2 psig	primary 6" mini purge	NA	backup off site
Unit 3	30 minutes	3.0%	36/85	primary permanent dilution blowers 18 psig	primary 4"	NA	backup off site
Unit 4	30 minutes	3.0%	59/140	NA	NA	NA	primary on site
Unit 5	varies according to EOP	3.0%	55/137	backup portable blowers 1 psig	backup 48" butterfly	primary Intermediate Building	NA
Unit 6	90 minutes	3.0%	54/141	NA	NA	Primary inside containment	NA

16/23

Observations

(on six sites evaluated so far - all large dry containments)

Wide variation in implementation of 10CFR50.44.

Use of repressurization/purge and movable recombiners. Implementation of design basis LOCA requirements (FSAR) could result in significant detriment (public health risk and worker health risk) during severe accidents for some plants.

Containment capability more than adequate (IPE).

Hydrogen monitoring safety function only for repressurization/purge or recombiners.

Personal Belief

Personnel at the nuclear electric power units should not be in the position where implementation of design basis LOCA hydrogen requirements would be detrimental to public health risk and worker health risk during severe accidents especially with respect to repressurization/purge and movable recombiners. This impacts how personnel at the nuclear unit prepare accident procedures and emergency plans and might impact how personnel would respond in an actual severe accident.

In my opinion, immediate action to remedy this situation is warranted.

My proposed revised 10CFR50, Appendix A, General Design Criteria 41, Containment atmosphere cleanup, is as follows:.

As necessary, systems to control fission products, hydrogen, oxygen, and other substances which may be released into the reactor containment shall be provided, consistent with the functioning of other associated systems, to assure that reactor containment integrity is maintained for accidents where there is a high probability that fission products may be present in the reactor containment.

My proposed revised 10CFR50.44, Standards for combustible gas control system in light-water-cooled power reactors, is as follows:

a.) An inerted reactor containment atmosphere shall be provided for each boiling light-water nuclear power reactor with a Mark I or Mark II type containment.

b.) Each licensee with a boiling light-water nuclear power reactor with a Mark III type of containment and each licensee with an ice condenser type of containment shall provide its nuclear power reactor containment with a hydrogen control system. The hydrogen control system must be capable of handling (based on realistic calculations) the hydrogen equivalent to that generated from a metal-water reaction involving 75% of the fuel cladding surrounding the active fuel region (excluding the cladding surrounding the plenum volume).

My proposed revised 10CFR50.44, Standards for combustible gas control system in light-water-cooled power reactors, is as follows:

c.) All light water reactors with other types of containment than in (a) or (b), must demonstrate that the reactor containment (based on realistic calculations) can withstand, without any hydrogen control system, a hydrogen burn for accidents with a high probability of causing severe reactor core damage. If such an evaluation of reactor containment capability can not be demonstrated, then the licensee shall provide a hydrogen control system per the backfit process. This hydrogen control system must be capable of handling (based on realistic calculations) the hydrogen equivalent to that generated from a metal-water reaction involving 75% of the fuel cladding surrounding the active fuel region (excluding the cladding surrounding the plenum volume)

My proposed revised 10CFR50.44, Standards for combustible gas control system in light-water-cooled power reactors, is as follows:

d.) Each light-water nuclear power reactor shall be provided with high point vents for the reactor coolant system, for the reactor vessel head, and for other systems required to maintain adequate reactor core cooling if the generation of noncondensible gases in these systems would realistically lead to severe reactor core damage during an accident. High point vents are not required, however, for the tubes in U-tube steam generators.

SUMMARY

Sufficient knowledge exists to change the regulations for Combustible Gas Control.

Focus must be on severe accidents.

Petition for rulemaking is a combination of:

> Retain what is effective and efficient.
> Add where necessary.
> Delete what is not effective and efficient.

Implementation of the petition will be "risk positive."

Note: Rulemaking is a result of a letter I sent to the NRC Commissioners on October 7, 1999. The letter was changed to a petition for rulemaking with my agreement. Implementation does not depend on "Option 3."

Markley
(3)

United States
Nuclear Regulatory Commission

Risk-Informed 50.44
"Standards for Combustible Gas Control System in Light-Water-Cooled Power Reactors"

Mary Drouin, Alan Kuritzky
Office of Nuclear Regulatory Research
Mike Snodderly
Office of Nuclear Reactor Regulation

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Allen Camp
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Eric Haskin
ERI

Presentation to
ACRS Subcommittee

June 29, 2000

OUTLINE

- Stakeholder (NEI) input
- Approach
- Overview of 50.44
- Risk significance
- Risk-informed options
- Potential Issues
- Schedule

STAKEHOLDER INPUT

- Letter from J. Colvin (NEI) to Chairman Meserve, dated January 19, 2000
  - General support for NRC approach (SECY-99-264)
  - Need to complete risk-informed projects on fire protection, security and technical specifications
  - Option 3 focus should initially be on 50.46 and 50.44

- NRC response
  - Framework in SECY-00-0086 is Revision 0 and being updated to better clarify such items as defense-in-depth, safety margin, treatment of uncertainties
  - Top priority is 50.44 (trial implementation)
  - Work initiated on:
    - 50.46
    - Special treatment requirements
    - Prioritizing remaining regulations

APPROACH

- Balanced high-level defense-in-depth (prevention/mitigation)

- Quantitative guidelines
  - Prevention/mitigation
  - Four strategies

APPROACH

- Selection of regulation for risk-informing
- Development of risk-informed options
  - Based on current requirements
  - Based on defined objective of the regulation
- Evaluation of options and development of alternatives

FRAMEWORK IMPLEMENTATION

- Identify the concern
- Identify the strategy that addresses the concern
- Identify the relative importance of the concern against the quantitative guidelines for each strategy
- Develop options:
  - a single accident class does not contribute more than 10% (of the quantitative guidelines) and
  - accounts for both prevention and mitigation

50.44 REQUIREMENTS

- Analytical Requirements
  - postulated LOCA
  - degraded core accidents
  - H2 source term based on fuel cladding oxidation
  - H2 source term based on 5%/75% metal-water reaction

- Physical Requirements
  - measure H2 concentration in containment
  - insure mixed atmosphere in containment
  - control combustible gas concentrations (recombiners)
  - inert Mark I and II containments
  - install high point vents
  - install H2 control system (igniters) for Mark III and ice condensers

50.44: Licensee Compliance

Physical Requirement	Predominant Means of Compliance
Measure H2 concentration	Safety-grade continuous H2 monitors
Mixed containment atmosphere	Natural convective cooling, air return fans, or containment spray
Post-LOCA H2 control (recombiners)	Safety grade recombiners
Inert Mark I and II containments	Nitrogen inerting system
High point vents	Vents installed per 50.44
H2 control for Mark III and ice condenser containments (igniters)	Safety-grade AC powered igniters

50.44: Related Regulations and Implementing Documents (Examples)

- Appendix E to Part 50: *"Emergency Planning and Preparedness for Production and Utliization Facilities"*
  - Continuous H2 monitoring required for Emergency Response Data System

- 50.46(b): *"Acceptance Criteria for Emergency Core Cooling Systems for Light-Water Nuclear Power Reactors"*
  - Specifies maximum H2 generation in postulated LOCA for purpose of complying with ECCS acceptance criteria

- RG 1.97: *"Instrumentation for Light-Water-Cooled Nuclear Power Plants to Assess Plant and Environs Conditions During and Following an Accident"*
  - Establishes that hydrogen concentration in the containment and drywell is a Type C variable (i.e., safety grade)

RISK SIGNIFICANCE

- Each core damage/melt accident can potentially produce combustible gases (both H2 and CO) because of loss of coolant inventory
  - fuel cladding oxidation
  - core-concrete interaction

- WASH-1400
  - Accidents (e.g., transients) other than LOCAs contribute to CDF
  - Significant H2 generation
  - High conditional containment failure probability from H2 combustion

- Severe Accident Research Program (SARP)
  - Post TMI Accident - Confirmatory Research
  - Confirmed ignition limits for variety of H2/air/steam mixtures
  - Evaluated effectiveness of H2 mitigative systems
  - Established basis for detonability of H2
  - Studied H2 transport and mixing

RISK SIGNIFICANCE

- Severe Accident Risk Assessment (NUREG-1150)
  - Other accidents (e.g., SBO) also found to contribute to CDF
  - H2 combustion significant contributor to early containment failure for Mark III and ice condenser during SBO
  - H2 combustion not a challenge to large volume containments

- Insights derived from IPEs (NUREG-1560):
  - Wide range of accident initiators found to contribute to CDF
  - H2 combustion from SBO accident sequences a significant contributor to containment failure

- Research (DCH Issue Resolution)
  - Analysis of the challenge to containment integrity from DCH for large dry and ice condenser containments
  - H2 combustion found to be a challenge to containment integrity for ice condensers during SBO

- Internal fire and seismic CD sequences have the characteristics of SBO

PWR LARGE VOLUME AND SUBATMOSPHERIC CONTAINMENTS

- H2 combustion not a challenge to containment integrity in short term
  - NUREG-1150 found early failure probability of 0.01 for Surry and Zion
  - NUREG-1560, IPE results indicate early failure probabilities from all causes less than 0.15 for most plants (HPME with H2 combustion important challenge)
  - Recent DCH research indicates HPME not a viable challenge

- Combustible gas concentration may be sufficient to challenge containment in long term
  - NUREG-1560, IPE results identified combustion events (in conjunction with existing high pressure) as late failure mechanisms for some plants

BWR MARK I AND MARK II CONTAINMENTS

- H2 combustion not a challenge to containment integrity during early stages of core melt accident due to inerting

- H2 combustion may challenge containment during late stages
  - O2 generation from radiolysis can lead to combustible containment atmosphere

BWR MARK III

- If igniters are operating H_2 combustion is not a challenge to containment integrity early or late for most accidents
  - NUREG-1150 found early failure (before vessel breach) probability <0.1 for Grand Gulf
  - Exception is accidents with high pressure at the time of vessel breach (i.e.,failure probability in range of ~0.2-0.5)

- If igniters are not operating, large H_2 concentration can accumulate

- SBO a dominant contributor to core damage (NUREG-1150 and IPEs)
  - Conditional containment failure probability given a SBO
    ~0.4 for short-term
    ~0.8 for long term
    ⇛ NUREG-1150

PWR ICE CONDENSER CONTAINMENTS

- NUREG-1150 and IPE (NUREG-1560) results indicate early failure probabilities <0.1 with or without igniters
- NUREG/CR-6427 (i.e.,DCH Issue resolution report) results indicate early failure probabilities of ~0.2-to>0.9 given an SBO accident
- SBO a dominant contributor to CDF (NUREG-1150 and 1560)
  - Conditional containment failure probability given a SBO ~0.1 ⇛ NUREG-1150

A RISK-INFORMED 50.44

- Accident types
  ⇛ core melt accidents

- Combustible gases source term
  ⇛ realistic calculations
  ⇛ fuel cladding oxidation and core-concrete interaction

- Controlling combustible gases
  ⇛ both early and late

RISK-INFORMED 50.44

Analytical Requirements will:

- Account for core melt accidents
- Account for combustible gas generation from fuel cladding oxidation and core concrete interaction
- Specify the amount and rate of combustible gas generation based on realistic calculations

RISK-INFORMED 50.44

Physical Requirements:

- Alternative 1:Modify the individual requirements
  - Eliminate requirement for safety-grade, continuous monitors
  - Add capability to measure long-term H2 conc. under degraded core conditions
  - Insure mixed atmosphere for risk significant accidents (e.g., SBO)
  - Eliminate post-LOCA H2 control (recombiners)
  - Add long term H2 control for risk significant core melt accidents
  - Insure H2 control for risk-significant core melt accidents (e.g., SBO) for Mark III and ice condensers

- Alternative 2: Eliminate the individual requirements
  - Replace with performance-based requirement to control combustible gases for all light-water reactors for the risk significant accidents

RISK-INFORMED 50.44 (cont'd)

Physical Requirements:

- Alternative 3: Eliminate the individual requirements
  - Replace with performance-based framework strategies to control combustibles gases for all light-water reactors:
    - Demonstrate containment integrity not challenged from combustible gases by (in order of preference) limiting the radionuclide release, or core damage accidents or the initiating events, or ensuring emergency preparedness

- Require conforming changes in other regulations

Potential Implementation Issues

- Policy:
  - Selective implementation
  - Role of the backfit rule
  - Application of risk-informed guidelines
  - Current or future plants

- Technical:
  - Treatment of long term containment performance
  - Guidelines for:
    - Defense-in-depth
    - Safety Margins
    - Treatment of uncertainties

Future Plans

- Complete evaluation of 10 CFR 50.44 and provide recommendations to Commission in August 2000, including any policy issues

- Continue evaluation of 10 CFR 50.46 and special treatment requirements and conduct workshop (Sept. 2000)

- Report to Commission in December 2000

- Recommend priority and schedule for remaining evaluations

Markley
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Risk-Informed Regulation

Steve Floyd
NEI
June 29, 2000

NEI

Option 2 Issues

- **Industry PRA peer review process**
  - All US plants will be peer reviewed by end of 2001
  - Submitted for NRC review to support option 2 application
  - NRC review plans discussed in 6/28 meeting
- **Correlation with STP exemption request**
  - Processes are essentially similar
  - Industry reviewing comparison matrix developed by STP

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Option 2 Issues

- **Legal issues**
  - Differentiation of design basis from special treatment
  - Part 21 applicability to RISC-3
- **Commercial treatment for RISC-3**
  - Preservation of design function
  - Level of detail for regulatory control
- **Treatment of prior commitments**
  - Rulemaking alone will not explicitly address
  - Industry commitment management guidelines

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Option 3 NRC Framework

- **Thoughtful effort by NRC staff and contractors to quantify all elements of regulatory structure**
  - Approach is more risk-based than risk-informed
  - Would establish regulation to the safety goal subsidiary objectives on individual plant basis
  - Establishment of quantitative licensing basis is fundamental departure from current approach
  - Previously dispositioned technical issues are reintroduced

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Option 3 - Preferred approach

- **Pragmatic versus theoretical**
- **Use generic risk insights to improve ~~current req~~uirements**
  - Example: design basis accident assumptions
- **Preserve existing risk-informed philosophy**
  - Integrated consideration of risk insights, traditional engineering approaches, safety margin

Option 3 - Industry Priorities

- **Complete ongoing efforts**
  - Hydrogen control (§50.44)
  - Fire protection (§50.48, Appendix R)
- **Focus on areas of greatest potential benefit**
  - Codes and standards (§50.55a)
  - Large Break LOCA (§50.46)
- **Further activities based on demonstrated success with above**

Observations

- **RES and NRR approaches present fundamental differences**
  - Industry confidence and potential for success would be improved through a consistent agency approach
- **NRC discussions continue to focus on low safety significant functions, rather than those of high safety significance**

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Observations

- **Successful applications will create incentive for widespread use of risk-informed methods and improvements to models**
  - 10 CFR 50.44 rulemaking (Option 3)
  - STP exemption request (Option 2)

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