UNITED STATES
NUCLEAR REGULATORY COMMISSION
ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
WASHINGTON, DC 20555 - 0001

May 13, 2004

MEMORANDUM TO: ACRS Members

FROM: Michael Snodderly, Senior ACRS Staff Engineer,

SUBJECT: CERTIFICATION OF THE MINUTES OF THE MEETING OF THE ACRS SUBCOMMITTEE ON RELIABILITY AND PROBABILISTIC RISK ASSESSMENT, FEBRUARY 19, 2004 - ROCKVILLE, MARYLAND

The minutes of the subject meeting, issued May 6, 2004, have been certified as the official record of the proceedings of that meeting. A copy of the certified minutes is attached.

Attachment: As stated

electronic cc: J. Larkins
H. Larson
S. Duraiswamy

MEMORANDUM TO: M. R. Snodderly, Senior ACRS Staff Engineer

FROM: G. E. Apostolakis, Chairman
Reliability and Probabilistic Risk Assessment Subcommittee

SUBJECT: CERTIFICATION OF THE MINUTES OF THE MEETING OF THE ACRS SUBCOMMITTEE ON RELIABILITY AND PROBABILISTIC RISK ASSESSMENT, FEBRUARY 19, 2004 - ROCKVILLE, MARYLAND

I do hereby certify that, to the best of my knowledge and belief, the minutes of the subject meeting on February 19, 2004, are an accurate record of the proceedings for that meeting.

May 6, 2004

George E. Apostolakis, Subcommittee Chairman — Date

PRE-DECISIONAL

May 6, 2004

MEMORANDUM TO: G. E. Apostolakis, Chairman
Reliability and Probabilistic Risk Assessment Subcommittee

FROM: M. R. Snodderly, Senior ACRS Staff Engineer

SUBJECT: WORKING COPY OF THE MINUTES OF THE MEETING OF THE ACRS SUBCOMMITTEE ON RELIABILITY AND PROBABILISTIC RISK ASSESSMENT, FEBRUARY 19, 2004 - 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. If you are satisfied with these minutes please sign, date, and return the attached certification letter.

Mike
Please see comments.
I've signed it.
Send me a copy
GA

Attachment: Minutes (DRAFT)

cc: Reliability and Probabilistic Risk Assessment Subcommittee Members
S. Duraiswamy
J. Larkins
H. Larson

Issued:05/06/04
Certified:05/13/04

ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
MEETING OF THE ACRS SUBCOMMITTEE ON
RELIABILITY AND PROBABILISTIC RISK ASSESSMENT
MEETING MINUTES - FEBRUARY 19, 2004
ROCKVILLE, MARYLAND

INTRODUCTION

The ACRS Subcommittee on Reliability and Probabilistic Risk Assessment (PRA) held a meeting on February 19, 2004, in Room T-2B3, 11545 Rockville Pike, Rockville, MD. The purpose of this meeting was to discuss the resolution of public comments on the proposed 10 CFR 50.69, "Risk-Informed Categorization and Treatment of Structures, Systems, and Components," and the staff's draft Regulatory Guide endorsing NEI 00-04, "10 CFR 50.69 Structures, Systems, and Components Categorization Guideline." The meeting was open to public attendance. Mike Snodderly was the Designated Federal Official for this meeting. There were no written comments. Members of the public representing the Nuclear Energy Institute (NEI), the American Society of Mechanical Engineers (ASME), and the American Nuclear Society (ANS) made oral presentations. The meeting was convened by the Subcommittee Chairman at 8:31 a.m. and adjourned at 5:38 p.m. on February 19, 2004.

ATTENDEES

ACRS Members

G. Apostolakis, Subcommittee Chairman
M. Bonaca, Member
F. Ford, Member
T. Kress, Member
S. Rosen, Member
W. Shack, Member
M. Snodderly, Designated Federal Official

Principal NRC Speakers

J. Fair, NRR
D. Harrison, NRR
T. Reed, NRR
T. Scarbrough, NRR

Other Principal Speakers

K. Balkey,. ASME
K.Ennis, ASME
S. Levinson, ANS
J. Mallay, ANS
A. Pietrangelo, NEI
C. Rowley, ASME
C. Sellers, ASME
D. True, ERIN Engineering
G. Zigler, ASME

There were approximately eight other 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.

OPENING REMARKS BY CHAIRMAN APOSTOLAKIS

George Apostolakis, Chairman of the ACRS Subcommittee on Reliability and PRA convened the meeting at 8:31 a.m. Dr. Apostolakis stated that the purpose of this meeting was to discuss the resolution of public comments on the proposed 10 CFR 50.69, "Risk-Informed Categorization and Treatment of Structures, Systems and Components." He said the subcommittee will gather information, analyze relevant issues and facts, and formulate proposed positions and actions, as appropriate, for deliberation by the full Committee. The rules for participation in the meeting were announced as part of the notice of the meeting published in the Federal Register on January 30, 2004.

DISCUSSION OF AGENDA ITEMS

Briefing on Revision D to NEI 00-04, "10 CFR 50.69 SSC Categorization Guideline"

Tony Pietrangelo, NEI, began by reminding the Committee that they had previously commented on Revision B to NEI 00-04. Mr. Pietrangelo then introduced Doug True of ERIN Engineering who was to brief the Committee on Revision D to NEI 00-04. Mr. Pietrangelo said that Revision D attempts to address the Committee's comments. Mr. Pietrangelo mentioned a table NEI provided prior to the meeting that addressed each of the NRC staff's comments. Based on a public meeting with the staff two weeks ago, Mr. Pietrangelo felt that NEI had any major issues with the staff in the area of categorization.

Mr. True began by showing several flow charts outlining the categorization process in Revision D to NEI 00-04. Mr. True than provided an example application of importance measures. Mr. True presented the recommended sensitivity studies for internal events. Mr. True then discussed the EPRI study on uncertainties in the categorization process. Mr. True said the key conclusions from EPRI TR-1008905 were: PRA codes calculate importance measures based on point estimate models, the correlated means for the importance measures calculated from a Monte Carlo evaluation are higher than point estimates, the correlation effect may have an influence on the calculation of the mean F-V value, the parametric correlation effect does not change the safety significance assessment, the NEI 00-04 sensitivity studies encompass the correlation effect on the mean importance measures, and either a formal parametric uncertainty assessment or a series of sensitivity studies provides equivalent results for use in the safety significance determination process.

Mr. True then described a set of deterministic rules and questions that were designed to assess defense-in-depth relative to: core damage prevention, large early containment failure, and long-term containment integrity. If the assessment indicates that the SSC is necessary for defense-in-depth than it would be categorized as a RISC-1 component. This lead to a discussion of risk sensitivity studies. Mr. True said that these studies were intended to assess the bounding change in CDF and LERF assuming a simultaneous decrease in reliability of all SSCs. He said

that the factor of increase was set by determining the amount of simultaneous change detectable in corrective action programs. The results of the sensitivity studies are then compared to RG 1.174 CDF and LERF guidelines. Mr. True ended with a discussion of how the integrated decisionmaking panel works. Mr. True believes the NRC staff's major issues with the categorization process had been resolved.

General Comments and Observations From the Subcommittee Members

- Dr. Apostolakis commented that the diagram on Slide 3 can play an important role in summarizing the NEI 00-04 categorization process. Dr. Apostolakis questioned whether what was being performed was really a sensitivity study on non-PRA components. Mr. True agreed it was confusing. Dr. Apostolakis suggested that non-PRA components should be split off from the diagram and shown more clearly. Dr. Bonaca agreed. Mr. True acknowledged the comment.

- Dr. Shack questioned why the emergency operating procedures and the severe accident management guidelines were not plant-specific inputs to the system engineering evaluation as shown on Slide 3. Dr. Ford added that future issues such as materials degradation should also be considered as inputs. Mr. True responded that the NEI categorization process addresses active functions of the systems. Mr. True continued that passive failure mechanisms, such as materials degradation, are addressed by ASME code case N-660.

- Dr. Bonaca recalled that only five percent of the components were modeled in the PRA and 95 percent were not. Dr. Bonaca said that these non-modeled components are addressed deterministically yet there is no mention of this part of the process. Dr. Bonaca commented that Slide 3 would be more useful if it included the deterministic treatment of non-modeled components. Mr. Rosen added that it should be made clear that these components were not modeled because they did not contribute to the dominate sequences. Dr. Bonaca agreed. Mr. Pietrangelo summarized the point and the commenting members agreed with his summary.

- Dr. Apostolakis quoted the following sentence from the end of Section 5 to DG-1121, "It should be recognized that the degree of relief that can be expected will be commensurate with the assurance provided by the evaluation." Dr. Apostolakis felt this was an important statement that could be made more explicit by showing how categorization is performed when you have a PRA and when you do not. For example, defense-in-depth can be addressed when you have a PRA by importance measures but defense-in-depth characterization is much more important when you don't have the PRA.

- Dr. Kress disagreed by saying that we don't properly pose what defense-in-depth is in the PRA and that in some cases defense-in-depth is necessary to address uncertainties that are not quantified by the PRA. Dr. Kress gave long-term cooling as an example of something that does not show up as risk significant in the PRA because it has little if any impact on CDF and LERF but it is very important to long-term cooling and late containment failure.

- Dr. Apostolakis quoted the following sentence from page six of NEI 00-04, "In the event a FIVE analysis is used, the categorization process is necessarily more conservative." Dr.

Apostolakis asked if anybody has shown that FIVE or seismic margin analysis is conservative or is it something that is widely accepted. Mr. True responded that he was not aware of a direct comparison but that any SSC or function that you credit in mitigating those unscreened sequences are all deemed safety significant.

- Dr. Apostolakis questioned the common cause failure RAW acceptance criteria of 20. Dr. Apostolakis had difficulty accepting that a common cause failure event that increases the CDF by a factor of 10 is not of high safety significance. Mr. True responded that it's measuring the impact of a whole system failing rather than an individual component. It was Dr. Apostolakis' opinion that the multiple greek letter method should be used instead. Mr. True agreed but said that the acceptance criteria of 20 was acceptable because it's measuring the impact on the system and CDF rather than an individual component.

- Mr. True asked if the Subcommittee was proposing that he identify a more extensive set of modeling uncertainties. Dr. Apostolakis responded that Mr. True would discover in a short time frame after consulting with his colleagues who have done real PRAs that they will give you a list of three or four modeling uncertainties that dominate. Dr. Kress was concerned that would only address CDF and not LERF and other release sequences. Dr. Apostolakis acknowledged this would only address Level One uncertainty. For Level Two, Dr. Apostolakis suggested using NUREG-1150 and consultation with other experts to identify the major model uncertainties and then multiply by a factor of three to do your sensitivity study.

- Mr. Rosen cautioned categorizing components as low with RAWs of just less than two. Mr. True responded that it also had to have a Fussell-Vesely less than 0.005. He said, as an example, the reactor core isolation cooling system had a RAW of .95 but the Fussell-Vesely was over 0.005 so it remained high safety significant.

- Dr. Apostolakis said that although he has not had a chance to review the EPRI study on uncertainties it appears to address the ACRS concern because it provides a rationale why the approximate method bounds the uncertainty based on three other methodologies. Dr. Apostolakis recommended that the EPRI study on uncertainties be referenced in NEI 00-04. Dr. Kress cautioned that this is responsive to our request but this is one PRA for one plant with a low CDF. We don't know how generic the results are or how to generalize them, especially for PWRs with higher CDFs.

- Dr. Bonaca agreed that an incentive to develop the PRA scope would be to confirm the rigor of the process.

- Dr. Shack commented that NEI 00-04 doesn't seem to have addressed the staff's comment that defense-in-depth should deal with more than just design basis events. Mr. True responded that by definition you don't have defense-in-depth for beyond design basis events. He went on to say that assessing and making some decision about defense-in-depth can only be done in the context of the likelihood of that event occurring, which is what the PRA is very good at.

- Dr. Kress said there are some functions that are so important that defense-in-depth is needed regardless of the PRA. He would have included shutdown, ECCS, containment, and long-term cooling as the important functions. Mr. True challenged that would be

status quo. Dr. Kress said it would only apply to a small number of subsystems such as the ones he mentioned above.

- Dr. Bonaca asked about the treatment of safety-related low safety significant MOVs. Mr. Pietrangelo said that NEI 00-04 only addresses categorization. Dr. Apostolakis said that treatment would be discussed in the afternoon.

- Dr. Shack asked about a change in the definition of long term integrity. On Page 46 of the previous edition to NEI 00-04, the definition of long term integrity stated, "It could be beneficial to preserving long term integrity." The definition was changed to say, "It would be the only means to preserving long term integrity." Mr. True explained that the change was made because "could be" was considered to broad. Mr. True related this back to Dr. Shack's point about the need to consider EOPs and SAMGs. He said EOPs and SAMGs invoke a lot of systems that could be beneficial practically speaking whether they really provide any benefit or not is better sorted out through processes like the PRA. Because you want your SAMGs to address all available resources, but it doesn't mean that everyone of those has the same significance from the standpoint of safety.

- Dr. Apostolakis suggested that the two bullets on Slide 21 be changed and say utilizes the strengths of PRA, therefore eliminating some of the weaknesses of the deterministic approach and addresses limitations of PRA bringing back the strength of the deterministic approach. Mr. Pietrangelo said he would change the slide.

- Mr. Rosen pointed out that the sentence that begins with, "Here again," on page 5 of NEI 00-04 was incomplete. Mr. True acknowledged that it was incomplete and committed to fix it.

NRC Staff Briefing on Part 50 Special Treatment Requirements Proposed Section 50.69

Tim Reed, Tom Scarbrough, John Fair, and Don Harrison, of the Office of Nuclear Reactor Regulation were the presenters for this presentation. Tim Reed said the objective of the briefing was to discuss the staff's efforts to address and resolve the comments that were received on 50.69 and the staff's review of NEI 00-04 draft revision D. Mr. Reed said that goal was to have the final rule package to the Commission by June 30, 2004. The staff committed to providing the draft final rule package by the middle of May 2004 to support a briefing of the Full Committee at it's June Meeting. Mr. Reed said that the staff has received 26 sets of comments which comprised a total of approximately 250 comments. The staff was concerned by the divergent range of interpretations of what the proposed rule language meant. In general, the states and public interest groups wanted a lot more prior review of RISC-3 treatment programs. Mr. Reed reminded the Committee of the staff's position that the RISC-3 treatment program that licensees would apply to safety related but low safety significant SSCs would be implemented by the licensees without prior NRC review and approval. This is covered by the requirements in 50.69(d)(2). This is the opposite approach from categorization which the NRC staff is reviewing and approving in detail.

The staff then discussed the following specific issues: (1) RISC-3 design requirement for fracture toughness, (2) consistency of RISC-3 categorization and treatment, (3) application of voluntary consensus standards, vendor recommendations, and operational experience for RISC-3 SSCs, (4) RISC-3 design control attributes, (5) RISC-3 design capability for

environmental and seismic conditions, (6) RISC-3 Corrective Action to Preclude Repetition, (7) operating experience feedback, (8) use of seismic experience data, (9) NRC review of planned treatment and inspection of implementation, (10) PRA scope requirements, (11) Crediting SSCs as part of selective implementation, and (12) 50.46a(b) scoped into 50.69.

General Comments and Observations From the Subcommittee Members

- Mr. Rosen asked if the inspection implementation was going to be broader than just treatment. Mr. Reed responded the temporary instructions have not been written but he expected it would focus more towards categorization and it would recommend a sampling in the RISC-3 treatment area. Mr. Harrison added that since the categorization process will be reviewed and approved by the staff beforehand, more emphasis would be on confirming that they're following their process and on treatment of RISC 2 and 3 components. Mr. Rosen was concerned that over emphasis of treatment would send the wrong message to the inspectors and licensees. He reminded the staff that categorization is the heart of this program.

- Mr. Rosen commented that this is the first time he had heard that the design can be changed under this rule. Mr. Scarbrough used the ASME code requirements as an example. The ASME code is a design code that low safety significant components would no longer have to meet. So the design may change for those class two and three components as long as they meet their functional requirements. They're not required to meet the original design as long as they meet the functional requirements. Mr. Rosen reiterated that importance that they still must meet their functional requirements.

- Dr. Apostolakis asked about the difference between functional requirements and design requirements. Mr. Scarbrough replied that a functional requirement for a pump would be so much flow under design basis conditions or it would have to be able to withstand an earthquake of so many G's. A design requirement is a characteristic, such as type of material, to meet the functional requirement.

- Dr. Ford asked if degradation mechanisms need to be considered when making material design changes to low safety significant components. Mr. Scarbrough said they are supposed to evaluate whether or not they have a known degradation mechanism. And, if they have a known degradation mechanism, they have to deal with that.

- Dr. Bonaca asked about coherence between 50.69 and Part 52. Mr. Gillespie agreed with Dr. Bonaca about the importance of coherence between 50.69 and Part 52 but acknowledged they have not worked on it for a year because of higher priority items such as 50.46 and 50.69 for operating plants.

- NEI 00-04 states the degree of relief that can be expected will be commensurate with the assurance provided by the evaluation. Dr. Bonaca asked how this statement will be verified. Mr. Reed replied that this whole framework is really based on robust categorization and having a lot of confidence in the categorization process which will be reviewed and approved by the staff. He went on to say that if you have confidence in that, then you can let go of the treatment and allow the licensees to apply the requirements of 50.69(d)(2).

- Mr. Pietrangelo of NEI commented that it didn't make any sense to put back into the high level treatment requirement language into 50.69 that is specifically excluded within the scope of 50.69 for low safety significant components. He used the equipment qualification requirements of 50.49 as an example. He said that components categorized as low safety significant are exempt from the requirements of 50.49 yet some of the treatment requirements in 50.69 could be interpreted as requiring equipment qualification.

- Dr. Bonaca brought up the issue of how corrective action program findings will be feed back into 50.69. Dr. Bonaca used an example where a plant scrammed and nine component failures resulted from the scam. The plant's senior management acknowledged that all the components had been removed from their preventive maintenance program sometime before. They acknowledged it was a shortsighted decision. Mr. Reed referred the Committee to 50.69(e)(2) for RISC 3 components which requires monitoring and feedback of performance data and corrective actions into the process.

- Dr. Kress asked if you have a site where there's more than one plant and you calculate RAW and Fussell-Vesely for the LERF, will you add those up for the different plants. Mr. Harrison said no, the criteria are applied on a per plant basis.

Staff Perspectives on Revision D of NEI 00-04

Dr. Apostolakis reconvened the meeting at 2:50 p.m. Mr. Harrison, NRR, gave the staff's perspectives on Revision D of NEI 00-04. Mr. Harrison said that the focus of the presentation will be on the remaining issues or areas that differ from where the staff had made prior comments. Mr. Harrison mentioned that the staff had a productive meeting with industry on February 5, 2004 and he believed they were coming to closure on a number of the issues. Mr. Harrison then discussed the following issues: (1) quality attributes of Analyses, (2) the factor used to represent the reduction in treatment, (3) limitations of types of analyses used, (4) uncertainty considerations, integral assessment and sensitivity studies, (5) prevents or mitigates core damage interpretation, (6) relevant failure modes interpretation, (7) safety significant attributes interpretation, (8) primary shutdown safety system interpretation, (9) common cause failure and degradation mechanisms, and (10) regulatory commitments.

General comments and observations from the Subcommittee members

- Dr. Apostolakis asked who is assessing the sensitivity studies performed to support the categorization of SSCs using PRA models to address the major identified sources of uncertainty, such as human error probability, cross cutting failures, and items identified during the assessment of PRA adequacy. Mr. Harrison said it was the peer reviewers. When a peer review is done on a PRA, they may have identified areas of weaknesses within the PRA or identified something that was in error. A licensee may have dealt with that by performing a sensitivity study saying if I change that information, it would have this impact on the analyses. Mr. Harrison then pointed out the last bullet in the NEI guidance on recommended sensitivity studies which said, "Any applicable sensitivity studies identified in the characterization of PRA adequacy."

- Dr. Shack said he was more supportive of NEI's sensitivity guidance to address identified weaknesses as opposed to applying a factor of four or five to account for uncertainties.

- Dr. Ford inquired about unknown degradation mechanisms. Mr. Reed responded that the requirements in 50.69(d)(2) are intended to capture future degradation mechanisms that might come up. Dr. Ford agreed that the rule language was adequate but it was at too high a level. Dr. Ford asked how unknown degradation mechanisms were addressed in the implementing guidance of NEI 00-04. The attendants were unable to answer Dr.Ford.

- Dr. Apostolakis commented that Page 12 from DG-1121 should be incorporated into the guidance to the integrated decision making panel in NEI 00-04.

- Mr. Rosen was concerned that the staff will only be approving the process for choosing the members of the integrated decision making panel. Mr. Rosen felt that the success or failure of 50.69 will ultimately hinge on the quality of the people that are doing the categorization. He thought there should be some standards for selection, training and qualification of these people.

- Dr. Bonaca spoke about the importance of coherence. He specifically mentioned coherence between risk-informed CDF and LERF criteria and Part 100. Dr. Bonaca gave the example where one has a detailed PRA analysis of the RPS. One could argue that since I have four redundancies, each one of them is not safety significant. And then maybe at that point one would begin to question the treatment. Lowering the treatment for something for which they have expanded so much focus and effort for so long. Dr. Bonaca felt this was an imbalance. An example of incoherent regulation.

Status of Risk-Informed Initiatives in ASME Nuclear Code and Standards

Dr. Apostolakis reconvened the meeting at 4:43 p.m. Wes Rowley, Vice President, Nuclear Codes and Standards, ASME, began by saying that ASME Board on Nuclear Codes and Standards had been meeting in Washington D.C. the past two days to discuss their strategic plan to manage their risk initiatives. He said the Board planned to brief the Committee on four aspects of their strategic plan: (1) PRA standards development, (2) risk-informed code cases, (3) 10 CFR 50.69 initiative, and (4) initiatives to develop a framework for future reactors. At the end of the presentation they would discuss future actions. Mr. Rowley than introduced Pat O'Regan, EPRI, Stanley Levinson, committee on nuclear risk management, and Doug True. Mr. Rowley said it had been five or six years since the board briefed ACRS on its risk initiatives and the board had done a fair amount in those intervening years.

Mr. Ken Balkey said he remembered meeting with Dr. Kress, probably 15 years ago and having the first idea of using risk analysis for in-service inspection. That led to a number of codes and standards initiatives back in the early and mid '90s. Mr. Balkey said the Board on Nuclear Codes and Standards then began to develop the PRA standard. In order to manage bringing risk-information into its Codes and Standards, the Board made a decision that it had to have a plan that they could track both short term, and long term initiatives. Within that, they have the elements within the plan covering across all the applications as well as the PRA standards and also the needs of the future reactors. Mr. Balkey said it would be a challenge to develop standards by 2008 in support of the Commission's phase approach to PRA quality.

Jim Mallay, Chairman of the ANS Coordinating Committee said that their goal was to make sure ANS and ASME are consistent and compatible but more than that, our emphasis really is going to be on the user ability to apply these standards. Mr. Mallay said there has been talk about a single standard. He cautioned that will never happen because there will always be a large suite of standards for various applications. He said that the joint committee hoped to develop a standard that will provide a framework so that one would know when to use the various elements.

General Comments and Observations From the Subcommittee Members

- Dr. Apostolakis thanked the presenters for a very informative presentation. He appreciated them briefing the Committee while they were in Washington and wished them luck with their efforts.

STAFF AND INDUSTRY COMMITMENTS

The staff committed to providing the draft final rule package by the middle of May 2004 to support a briefing of the Full Committee at it's June Meeting.

SUBCOMMITTEE DECISIONS AND ACTIONS

Dr. Apostolakis recommended that the EPRI study on uncertainties be referenced in NEI 00-04. The Subcommittee will make a recommendation concerning the draft final rulemaking package for the proposed Section 50.69 on special treatment requirements at the June Full Committee meeting to support the staff's schedule to provide it to the Commission by June 30, 2004.

BACKGROUND MATERIALS PROVIDED TO THE SUBCOMMITTEE PRIOR TO THIS MEETING

1. Subcommittee status report, including agenda.
2. Staff Requirements Memorandum dated March 28, 2003, from Annette L. Vietti-Cook, Secretary, to William D. Travers, EDO, Subject: Staff Requirements - SECY-02-0176 - Proposed Rulemaking to Add New Section 10 CFR 50.69, "Risk-Informed Categorization and Treatment of Structures, Systems, and Components".
3. SECY-02-0176 - Proposed Rulemaking to Add New Section 10 CFR 50.69, "Risk-Informed Categorization and Treatment of Structures, Systems, and Components," dated September 30, 2002.
4. Letter dated March 19, 2002, from George E. Apostolakis, Chairman, ACRS, to William D. Travers, Executive Director for Operations, NRC, Subject: Proposed Rulemaking and Associated Guidance for Risk-Informing the Special Treatment Requirements of 10 CFR Part 50 (Option 2).
5. Report dated October 12, 1999, from Dana A. Powers, Chairman, ACRS, to Greta Joy Dicus, Chairman, NRC, Subject: Proposed Plans for Developing Risk-Informed Revisions to 10 CFR Part 50, "Domestic Licensing of Production and Utilization Facilities."
6. Nuclear Energy Institute (NEI) Report 00-04, "10 CFR 50.69 Structures, Systems, and Components Categorization Guideline," Revision D, October 2003.
7. NEI, Table-1, "NRC's General Comments on Revision C to NEI 00-04," (Includes industry disposition).

8. Memorandum dated February 11, 2004, from Timothy A. Reed, Office of Nuclear Reactor Regulation (NRR), to Catherine Haney, NRR, Subject: Summary of February 5, 2004, Meeting With Nuclear Energy Institute (NEI) and Other Stakeholders on the Implementation Guidance for 10 CFR 50.69 (DG-1121 and NEI 00-04).
9. Memorandum dated February 5, 2004, from Catherine Haney, NRR, to John T. Larkins, Executive Director, ACRS, Subject: ACRS Subcommittee Meeting on Section 50.69, "Risk-Informed Categorization and Treatment of Structures, Systems and Components for Nuclear Power Plants."
10. EPRI Report 1008905, "Parametric Uncertainty Impacts on Option 2 Safety Significance Categorization," June 2003.

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Note: Additional details of this meeting can be obtained from a transcript of this meeting available for downloading or viewing on the Internet at "http://www.nrc.gov/ACRSACNW' or can be purchased from Neal R. Gross and Co., Inc., (Court Reporters and Transcribers) 1323 Rhode Island Avenue, NW., Washington, DC 20005 (202) 234-4433.

MRS

of no significant impact for license application; correction.

SUMMARY: The U.S. Nuclear Regulatory Commission published a Finding of No Significant Impact (FONSI) in the **Federal Register** on January 27, 2004 (69 FR 3956), concerning the United States Enrichment Corporation Inc.'s (USEC Inc.'s) license application for its American Centrifuge Lead Cascade Facility (Lead Cascade) in Piketon, Ohio. The FONSI contained an incorrect number.

FOR FURTHER INFORMATION CONTACT: Yawar Faraz, NMSS/FCSS (301) 415–8113.

Correction

In the **Federal Register** of January 27, 2004, in volume 69, number 17, on page 3956, correct the 0.0001% value to 1%. The corrected sentence, which is the third sentence of the third full paragraph in the third column, should read as follows:

"For example, NRC staff finds that public exposure to radiation from the proposed action will be less than 1% of the limits in 10 CFR part 20."

Dated in Rockville, Maryland this 27th day of January, 2004.

For the Nuclear Regulatory Commission.

Michael T. Lesar,

Chief, Rules Review and Directives Branch, Division of Administrative Services, Office of Administration.

[FR Doc. 04–2018 Filed 1–29–04; 8:45 am]

BILLING CODE 7590–01–P

NUCLEAR REGULATORY COMMISSION

Advisory Committee on Nuclear Waste, Meeting on Planning and Procedures; Notice of Meeting

The ACNW will hold a planning and procedures meeting on February 26, 2004, Room T–2B1, 11545 Rockville Pike, Rockville, Maryland.

The entire meeting will be open to public attendance, with the exception of a portion that may be closed pursuant to 5 U.S.C. 552b(c)(2) and (6) to discuss organizational and personnel matters that relate solely to internal personnel rules and practices of ACNW, and information the release of which would constitute a clearly unwarranted invasion of personal privacy.

The agenda for the subject meeting shall be as follows:

Thursday, February 26, 2004—8 a.m.–11 a.m.

The Committee will discuss proposed ACNW activities and related matters. The purpose of this meeting is to gather information, analyze relevant issues and facts, and formulate proposed positions and actions, as appropriate, for deliberation by the full Committee.

Members of the public desiring to provide oral statements and/or written comments should notify the Designated Federal Official, Mr. Howard J. Larson (telephone: 301/415–6805) between 7:30 a.m. and 4:15 p.m. (e.t.) five days prior to the meeting, if possible, so that appropriate arrangements can be made. Electronic recordings will be permitted only during those portions of the meeting that are open to the public.

Further information regarding this meeting can be obtained by contacting the Designated Federal Official between 7:30 a.m. and 4:15 p.m. (e.t.). Persons planning to attend this meeting are urged to contact the above named individual at least two working days prior to the meeting to be advised of any potential changes in the agenda.

Dated: January 23, 2004.

Sher Bahadur,

Associate Director for Technical Support, ACRS/ACNW.

[FR Doc. 04–2015 Filed 1–29–04; 8:45 am]

BILLING CODE 7590–01–P

NUCLEAR REGULATORY COMMISSION

Advisory Committee on Reactor Safeguards, Meeting of the ACRS Subcommittee on Reliability and Probabilistic Risk Assessment; Notice of Meeting

The ACRS Subcommittee on Reliability and Probabilistic Risk Assessment will hold a meeting on February 19, 2004, Room T–2B3, 11545 Rockville Pike, Rockville, Maryland.

The entire meeting will be open to public attendance.

The agenda for the subject meeting shall be as follows:

Thursday, February 19, 2004—8:30 a.m. Until the Conclusion of Business

The purpose of this meeting is to review the ongoing resolution of public comments on the proposed 10 CFR 50.69, "Risk-Informed Categorization and Treatment of Structures, Systems, and Components," and the staff's draft Regulatory Guide endorsing Revision D of NEI 00–04, "10 CFR 50.69 Structures, Systems, and Components Categorization Guideline." The Subcommittee will hear presentations by and hold discussions with representatives of the NRC staff and NEI regarding this matter. The Subcommittee will gather information, analyze relevant issues and facts, and formulate proposed positions and actions, as appropriate, for deliberation by the full Committee.

Members of the public desiring to provide oral statements and/or written comments should notify the Designated Federal Official, Mr. Michael R. Snodderly (telephone: 301–415–6927) five days prior to the meeting, if possible, so that appropriate arrangements can be made. Electronic recordings will be permitted during the meeting.

Further information regarding this meeting can be obtained by contacting the Designated Federal Official between 7:30 a.m. and 4:15 p.m. (e.t.). Persons planning to attend this meeting are urged to contact the above named individual at least two working days prior to the meeting to be advised of any potential changes to the agenda.

Dated: January 23, 2004.

Sher Bahadur,

Associate Director for Technical Support, ACRS/ACNW.

[FR Doc. 04–2016 Filed 1–29–04; 8:45 am]

BILLING CODE 7590–01–P

SECURITIES AND EXCHANGE COMMISSION

Submission for OMB Review; Comment Request

Upon Written Request, Copies Available From: Securities and Exchange Commission, Office of Filings and Information Services, Washington, DC 20549.

Extension:

Rule 17Ad–11; SEC File No. 270–261; OMB Control No. 3235–0274.

Notice is hereby given that pursuant to the Paperwork Reduction Act of 1995 (44 U.S.C. 3501 *et seq.*), the Securities and Exchange Commission ("Commission") has submitted to the Office of Management and Budget a request for extension of the previously approved collection of information discussed below.

Rule 17Ad–11: Reports Regarding Aged Record Differences, Buy-Ins, and Failure To Post Certificate Detail to Master Securityholder Files

Rule 17Ad–11 requires all registered transfer agents to report to issuers and the appropriate regulatory agency in the event that aged record differences exceed certain dollar value thresholds. An aged record difference occurs when an issuer's records do not agree with those of securityowners as indicated, for instance, on certificates presented to the

Revision 2:01/08/2004
Mike Snodderly

ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
MEETING OF THE SUBCOMMITTEE ON
RELIABILITY AND PROBABILISTIC RISK ASSESSMENT
ROOM T-2B3, 11545 ROCKVILLE PIKE, ROCKVILLE MD
FEBRUARY 19, 2004

Contact: Michael Snodderly (301-415-6927, mrs1@nrc.gov)

-PROPOSED SCHEDULE-

	TOPICS	PRESENTERS	TIME
I.	**Opening Remarks**	G. Apostolakis, ACRS	8:30-8:35 a.m.
II.	**Briefing on Revision D to NEI 00-04, "10 CFR 50.69 SSC Categorization Guideline"**	A. Pietrangelo, NEI D. True, ERIN	8:35-10:15 a.m.
	BREAK		**10:15-10:30 a.m.**
II.	**Briefing on Revision D to NEI 00-04, "10 CFR 50.69 SSC Categorization Guideline" (Continued)**	A. Pietrangelo, NEI D. True, ERIN	10:30-12:00 p.m.
	LUNCH		**12:00-1:00 p.m.**
III.	**Summary of Public Comments on 10 CFR 50.69 and Status of Resolution**	T. Reed, NRR D. Harrison, NRR T. Scarbrough, NRR	1:00-2:45 p.m.
	BREAK		**2:45-3:00 p.m.**
IV.	**Staff's Views on NEI 00-04**	D. Harrison, NRR	3:00-4:45 p.m.
	BREAK		**4:45-5:00 p.m.**
V.	**Status of Risk-Informed Initiatives Within ASME Nuclear Codes and Standards**	K. Balkey, ASME	5:00-6:00 p.m.
VI.	**Subcommittee Discussion**	G. Apostolakis, ACRS	6:00-6:15 p.m.
	Adjourn	G. Apostolakis, ACRS	6:15 p.m.

NOTE:

- Presentation time should not exceed 50 percent of the total time allocated for specific item. The remaining 50 percent of the time is reserved for discussion.
- 35 copies of the presentation materials are to be provided to the Subcommittee.

ADVISORY COMMITTEE ON REACTOR SAFEGUARDS

SUBCOMMITTEE MEETING ON RELIABILITY AND PROBABILISTIC RISK ASSESSMENT

February 19, 2004
Date

PLEASE PRINT

NAME	AFFILIATION
Glen E. Schinzel	STPNOC
Tony Pietrangelo	NEI
Nancy Chapman	SERCH/Bechtel
JASON BROWN	WESTINGHOUSE
Doug True	ERIN
Bob Lutz	Westinghouse
Patricia Campbell	Winston & Strawn
~~GOUTAM BACA~~	
Getachew Tesfaye	Constellation
STANLEY LEWINSON	FRAMATOME
C Wesley Rowley	ASME
CRAIG D. SELLERS	ALION
Ray R. Weidler	ASME
Gilbert Zigler	ASME
Brian Erler	ASME
Patrick O'Regan	EPRI
Ken Bailey	ASME / WESTINGHOUSE
Kevin Ennis	ASME

ADVISORY COMMITTEE ON REACTOR SAFEGUARDS

SUBCOMMITTEE MEETING ON RELIABILITY AND PROBABILISTIC RISK ASSESSMENT

February 19, 2004
Date

NRC STAFF SIGN IN FOR ACRS MEETING

PLEASE PRINT

NAME	NRC ORGANIZATION
JIM REED	NRC/NRR/DRIP
JOHN FAIR	NRC/NRR/DE
David Terao	NRC/NRR/DE
Goutam Bagchi	NRC/NRR/DE
DAVID FISCHER	NRR/DE/EMEB
Donnie Harrison	NRR/DSSA/SPSB
Ken Heck	NRR/DIPM
Stephen Dinsmore	NRR/SPSB
STU MAGRUDER	NRR/DSSA/SPSB
Thomas Scarbrough	NRR/DE/EMEB
Frank Cherny	RES/DET
Eileen McKenna	NRR/DRIP
Jim Strosnider	NRR/DE/EMEB
HOSSEIN HAMZEHEE	RES/DRAA
PAUL SHEMANSKI	NRR/DE/EEIB
Frank Gillespie	NRR/DRIP
THOMAS KOSIAK	NRR/DE/EEIB
Matthew A. Mitchell	NRR/DE/EMCB
Gene Imbro	NRR/DE

Master

United States
Nuclear Regulatory Commission

RISK-INFORMED PART 50 SPECIAL TREATMENT REQUIREMENTS PROPOSED SECTION 50.69

ACRS SUBCOMMITTEE ON RELIABILITY AND PROBABILISTIC RISK ASSESSMENT

FEBRUARY 19, 2004

Timothy Reed, Thomas Scarbrough
John Fair, Donald Harrison
Office of Nuclear Reactor Regulation
US Nuclear Regulatory Commission

BRIEFING OBJECTIVE

- **To brief the Committee on the current status regarding the significant technical issues that must be addressed to publish a final 50.69 rule - specifically:**

  1) Staff's efforts to address comments received on proposed §50.69

  2) Staff's review of NEI 00-04 draft revision D

- **Focus of the discussion will be on the possible changes from proposed rule to final rule**

BACKGROUND

- SECY-98-300 (12/98) proposed high level approaches ("options")
- SECY-99-256 (10/99) provided rulemaking plan and ANPR
- SECY-00-194 (9/00) provided preliminary views on ANPR comments and thoughts on regulatory approach
- South Texas exemption (8/01) approved (proof of concept for §50.69)
- SECY-02-0176 (9/30/02) provided proposed 50.69 to Commission
- Commission SRM - 3/28/03
- Proposed 50.69 published for comment - 5/16/03
- Public comment period closed - 8/30/03

ONGOING TASKS TO ISSUE 50.69

- Review/resolution of public comments
- Review of Draft Revision D of NEI 00-04 (and revision to DG-1121)
- WOG pilot examining 50.69 submittal and staff review
- Revision to rule package per public comment resolution/review of implementation guidance
- Review/concurrence process for final rulemaking process (meet with ACRS on final rulemaking package)
- Schedule - final rulemaking package due to the Commission 6/30/04

PUBLIC COMMENTS

- **26 sets of comments comprising approximately 250 comments**
- **Comments received from NEI, numerous industry groups, licensees, public interest groups, states, and nuclear organizations**

OVERVIEW OF PUBLIC COMMENTS

- **Comments reflected a wide range of views on many of the major issues associated with 50.69:**
  - **Divergent interpretations of the rule language and SOC**
  - **States and public interest groups recommend prior NRC review of SSC treatment while industry recommends no prior NRC review**
  - **Stakeholders generally support NRC inspection of 10 CFR 50.69 implementation**
  - **Industry does not support full scope PRA requirements while States and public interest groups recommend full scope PRA**

STAFF PROPOSAL TO RESOLVE PUBLIC COMMENTS

- Clarify rule language
- Simplify SOC
- No prior NRC review of treatment
- At a sampling of plants inspect implementation of 10 CFR 50.69 per TI
- Conduct public workshop to discuss final rule

SPECIFIC ISSUES

1. **RISC-3 Design Requirement for Fracture Toughness**

   SOC noted that design requirements for fracture toughness continued to apply for replacement ASME components.

   Several industry commenters stated that SOC exceeded rule requirements. One commenter asserted that fracture toughness is not a design issue.

   Staff considers fracture toughness to be a design consideration.

   Intent of 10 CFR 50.69 is to remove special treatment requirements while maintaining design requirements.

   Staff plans to clarify in paragraph (b)(1) of the rule that fracture toughness requirements retained for RISC-3 SSCs.

2. Consistency of RISC-3 Categorization and Treatment

Industry comments indicate that licensees might not consider impact of treatment in categorization process.

South Texas asserts that sensitivity studies eliminate need to specifically consider SSC reliability changes due to treatment.

Westinghouse Owners Group states that cross-system common cause interactions are rarely modeled in PRAs.

Dominion Power indicates that degradation mechanisms resulting from treatment processes are typically not considered in PRAs.

Treatment practices must be consistent with categorization process assumptions and assessment of potential change in risk.

Staff plans to clarify in paragraph (d)(2) of the rule that RISC-3 treatment must be consistent with assumptions credited in categorization process.

3. Application of Voluntary Consensus Standards, Vendor Recommendations, and Operational Experience for RISC-3 SSCs

SOC references use of voluntary consensus standards as an effective means to establish treatment requirements. SECY-00-0194 noted an NRC-sponsored study found too much variation in industrial practices to conclude that such practices will provide reasonable confidence in SSC functionality.

Industry comments indicate that only industrial practices might be applied when implementing treatment requirements for RISC-3 SSCs.

ASME did not recommend adding a provision on voluntary consensus standards in rule because SOC provided adequate guidance for RISC-3 treatment.

Additional stakeholders raised concern that proposed rule was not adequate to maintain plant safety.

3. Application of Voluntary Consensus Standards cont'

Staff plans to clarify in SOC supporting paragraph (d)(2) that industrial practices might not satisfy rule requirements regarding implementation of processes that provide reasonable confidence in RISC-3 design basis capability.

4. RISC-3 Design Control Attributes

SOC listed several attributes that should be considered as part of design control process in light of only high-level requirements in rule.

Importance of design control reflected in South Texas exemption which maintains Appendix B design control.

NEI suggested a focused list of design control attributes be substituted in 10 CFR 50.69, including selection of suitable materials; verification of design adequacy, and control of design changes.

With simplification of SOC, it may be appropriate to clarify design control attributes in rule.

Staff plans to clarify design control attributes for RISC-3 SSCs in paragraph (d)(2) of the rule to include the NEI suggestion plus the control of installation.

5. RISC-3 Design Capability for Environmental and Seismic Conditions

RISC-3 SSCs will be exempt from special treatment requirements for qualification methods for environmental conditions and effects, and seismic conditions.

RISC-3 SSCs must be capable of performing their safety-related functions under applicable environmental conditions and effects, and seismic conditions.

Some licensees interpret rule as requiring no evaluations of environmental and seismic capability.

NEI states that environmental or seismic requirements for RISC-3 SSCs in 10 CFR 50.69 should be deleted.

NUGEQ states that 10 CFR 50.69 exempts RISC-3 electrical equipment from aging issues, and that rule does not require establishment of design life.

5. RISC-3 Design Capability Cont'

Staff plans to clarify in rule that licensees must develop and implement documented treatment processes. The staff is not planning to revise design basis language.

6. RISC-3 Corrective Action to Preclude Repetition

NEI recommended revision of rule to address significant conditions adverse to quality such that measures are taken to provide reasonable confidence that cause of condition is determined and corrective action taken to preclude repetition.

New Jersey and NIRS raised concerns regarding apparent lack of consideration of common-cause issues for RISC-3 SSCs.

Staff plans to accept the NEI comment and clarify in paragraph (d)(2) of the rule that measures must be taken for significant conditions adverse to quality for RISC-3 SSCs.

7. Operating Experience Feedback

Commission requested comments on how operational experience should be considered in the rulemaking.

UCS states that relevant operating experience suggests that regulatory oversight of equipment credited with lowering risk should be increased.

Industry commenters believe that ongoing opportunities for sharing experience from existing industry and regulatory programs provide substantial data source for licensees in categorizing SSCs, and recognizing impacts and performance changes.

Staff plans to clarify in paragraph (e)(1) of the rule that licensees must feed back plant operational experience (e.g., corrective action) into processes.

8. Use of Seismic Experience Data

Several industry commenters stated that SOC might create additional burden on plants licensed prior to implementation of Appendix A to 10 CFR Part 100.

SOC needs to clarify that 10 CFR 50.69 will not change seismic design basis for USI A-46 plants, or impose additional seismic requirements.

Industry commenters also raised concerns regarding SOC discussion on use of seismic experience data.

Rule does not change seismic design requirements for RISC-3 SSCs.

Part 100 licensees must comply with technical requirements of Part 100 and have adequate technical bases to conclude that SSCs will perform safety-related functions under seismic design-basis conditions, which includes number and magnitude of earthquake events specified for SSC design.

8. Use of Seismic Experience Data Cont'

Staff plans to clarify in SOC that 10 CFR 50.69 will not change seismic design basis for USI A-46 plants, or impose additional seismic requirements for those plants.

9. NRC Review of Planned Treatment and Inspection of Implementation

Commission requested comments on NRC review of RISC-3 treatment processes, and whether changes are needed in inspection program.

New Jersey recommends that NRC review planned 10 CFR 50.69 treatment programs.

UCS states that NRC should review treatment and also inspect its implementation.

BWROG asserts that licensees should develop 10 CFR 50.69 processes based on rule requirements with routine NRC inspection verifying acceptable compliance.

NEI states that existing NRC inspection and enforcement process addresses all affected functional areas.

9. NRC Review of Planned Treatment and Inspection of Implementation Cont'

A sampling of plants will be initially inspected per TI. The ROP is a performance-based and risk-informed program and overall will remain sensitive to conditions that could significantly increase risk.

10. PRA Scope Requirements

Industry commenters do not believe that 10 CFR 50.69 should be dependent on full scope PRA.

Illinois Emergency Management Agency recommends full scope PRA for 10 CFR 50.69 implementation.

New Jersey recommends that NRC review licensee PRAs in depth periodically.

UCS states that rulemaking should not proceed when PRAs require adjustments as indicated in its submittal.

Conference of Radiation Control Program Directors recommends that PRAs be updated and submitted for NRC review.

Staff plans to continue to require Level 1, full power, peer-reviewed PRA for application of 10 CFR 50.69 with prior NRC review of categorization process and concludes this is consistent with the Commission SRM on PRA quality.

11. Crediting SSCs as Part of Selective Implementation

When a licensee selects a system for categorization and categorizes SSCs as "RISC-3" it means other SSCs must be RISC-1 and RISC-2.

What must a licensee do for these "credited" SSCs?

What must the NRC staff review in the PRA to support approval of the categorization process?

The staff plans to clarify the SOC that licensees must maintain credited SSCs (per paragraph (e) and (d)(1)) and that the staff will need to perform a broad review to support categorization approval.

12. 50.46a(b) SCOPED INTO 50.69

Certain provisions within the old § 50.44 were previously identified as containing STRs

The proposed rule noted this situation and indicated that the final rule may "scope-in" these provisions

Head vent requirements from old 50.44 were simply relocated to 50.46a(b) as part of the effort to risk-inform 50.44

The requirements impose Appendix B requirements on reactor vessel head vents

The staff plans to add the Appendix B portion of 50.46a(b) to the list of special treatment requirements within the scope of 50.69.

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STAFF PERSPECTIVES ON REVISION D OF NEI 00-04, "10 CFR 50.69 SSC CATEGORIZATION GUIDELINE"

ACRS SUBCOMMITTEE ON RELIABILITY AND PROBABILISTIC RISK ASSESSMENT

FEBRUARY 19, 2004

Donald Harrison
Office of Nuclear Reactor Regulation
US Nuclear Regulatory Commission

BRIEFING OBJECTIVE

- **To brief the Committee on the current status regarding the technical issues that must be addressed to publish a regulatory guide to endorse NEI 00-04 in support of 10 CFR 50.69 rulemaking - specifically:**

  1) **Resolution of staff comments on draft Revision C**

  2) **Review of NEI 00-04 draft Revision D**

  3) **Remaining issues that need to be addressed/clarified**

- **Focus of discussion will be on remaining issues and areas where Revision D differs from the staff positions provided on Revision C**

Note that the staff met with NEI/Industry on February 5, 2004 to discuss Revision D and their resolution of the staff comments on Revision C

SPECIFIC ISSUES

1. **Quality Attributes of Analyses (Comments A, E6, & E11)**

   Staff recommended guidance be developed to address expected quality attributes of the external events PRA and non-PRA type analyses for this application

   Revision D provides some quality guidance (§ 3.3), but leaves the quality justification up to the licensee for their plant-specific application (i.e., no application-specific guidance for external events PRA and non-PRA type analyses)

   Staff accepts the Revision D approach, recognizing this will put the burden on the licensee to justify and the staff to verify the quality of the PRA analyses and other risk information used for this application

   The Revision D approach limits the scope of application if non-PRA type analyses are used

2. The Factor Used to Represent the Reduction in Treatment (Comments B, E18-1, & E18-2)

Staff recommended a method be developed to determine the factor to use in the risk sensitivity study and guidance be developed to ensure by corrective action program the risk sensitivity study remains valid

Staff also recommended a method be developed when non-PRA type analyses are used to demonstrate impacts are acceptably small

Revision D provides some guidance on performing the risk sensitivity study (§ 8), but the linkage to the corrective action program is not explicit, though NEI has stated the intent was that the factor would be within what could be detected within the corrective action program

---

Staff expects additional guidance to be provided in the subsequent revision of NEI 00-04 to describe how the factor used in the risk sensitivity study is derived to be within detectability of the corrective action program and how this program will collect and analyze the data

The Revision D approach limits the scope of application if non-PRA type analyses are used

3. Limitations of Types of Analyses Used (Comments C & E11)

Staff stated that current state-of-the-art PRA methods are available to quantitatively address the full spectrum of potential events and the full range of plant operating modes for this application

Staff also stated that the degree of relief that can be expected under 10 CFR 50.69 will be commensurate with the type of analyses used

Revision D recognizes the limitation in relief imposed if non-PRA type analyses are used (§ 3.2)

Staff accepts the Revision D approach, in that it limits the scope of application, and thus relief provided, if non-PRA type analyses are used

4. Uncertainty Considerations, Integral Assessment, & Sensitivity Studies (Comments D, E5-3, E8-2, E8-4, E17-1, & E17-2)

Staff noted potentially large differences in the levels of uncertainty in the modeling and data for the PRA models for various types of events (i.e., internal, fires, seismic, etc.) and thus, recommended that the most conservative categorization should be used, considering the various types of events and associated sensitivity studies individually

Revision D contains additional guidance (§ 7 and 9), but does not explicitly discuss uncertainty considerations, though a number of sensitivity studies are part of the categorization process

Revision D allows an integral assessment (§ 5.6 & 7) of the various types of events, with results of the individual sensitivity studies also provided to the IDP for consideration

---

Staff expects uncertainties to be addressed in the risk sensitivity assessment, consistent with RG 1.174 § 2.2.5

Staff accepts the Revision D approach to integral assessment since the use of non-PRA type analyses limits the scope of application

5. Prevents or Mitigates Core Damage Figure 5-1 Interpretation (Comment E8-1)

Staff stated that the phrase could be mis-interpreted in such a way to not include consideration of containment systems and suggested changing it to "prevents or mitigates severe accidents"

Revision D does not change the phrase in Figure 5-1, though the supporting text (§ 5) is broader as it refers to severe accidents and NEI has stated that the intent was to include containment systems

---

Staff accepts Revision D approach that allows screening out SSCs that have no role in prevention or mitigation of severe accidents, but expects the terminology in Figure 5-1 to be clarified in the subsequent revision of NEI 00-04

6. Relevant Failure Modes Interpretation (Comment E9-2)

Staff indicated that the phrase "relevant failure modes" was open to interpretation and thus, all failure modes of the SSC identified in the PRA should be considered in the importance evaluation

Revision D maintains the phrase (§ 5.1), and NEI has stated its intent was to allow exclusion of failure modes that might be in the PRA that are not directly related to the component's performance

NEI has indicated that they will clarify (or delete) this phrase in a subsequent revision

---

Staff expects the intent of this phrase to be clarified or deleted in the subsequent revision of NEI 00-04

7. Safety Significant Attributes Interpretation (Comment E9-3)

Staff was not sure of the intent and implications of the discussion of using the component failure mode or dominant failure mode in the identification of safety significant attributes

Revision D still discusses (§ 5.1) the identification of safety significant attributes, but does not explicitly how this identification is to be used, though NEI has stated the intent was to assure that the factors that influence the risk significance of RISC-1 and RISC-2 SSCs are identified and controlled

---

Staff expects the intent of this identification to be explicitly discussed in the subsequent revision of NEI 00-04

8. Primary Shutdown Safety System Interpretation (Comment E13)

Staff stated that it was not sure use of the NUMARC 91-06 guidance would result in conservative categorization results as described and thus, indicated that any (not just the primary) SSCs identified in the plant-specific Outage Risk Management Guideline should be considered safety significant

Revision D (§ 5.5) clarifies the attributes of the primary shutdown system and the conditions for designating SSCs safety significant for shutdown conditions

NEI further clarified that the Outage Risk Management Guideline may identify numerous means of meeting the guidelines for shutdown safety identified in NUMARC 91-06 and stated their intention for the phrase primary shutdown safety system to include the typical running shutdown safety means as well as an alternative means

---

Staff accepts the Revision D approach, but expects the identification of the primary shutdown safety system to be clarified in the subsequent revision of NEI 00-04

9. Common Cause Failure and Degradation Mechanisms (Comment E18-4, E18-5, & E18-6)

The risk sensitivity study could realistically only be invalidated if extensive cross-system common cause failures could occur without early detection/warning, thus the Staff included in 10 CFR 50.69 the need for licensees to identify potential common cause interaction susceptibility, including cross-system interactions and potential impacts from known degradation mechanisms

If not explicitly evaluated, Staff expects those aspects of treatment necessary to prevent SSC degradation or failure from known mechanisms would be identified and such treatment aspects retained

Revision D does not address this area directly beyond reference to ASME Code Case N-660 and RI-ISI code cases and topical reports and NEI has indicated that they may provide additional guidance on this area in future revisions of NEI 00-04

Staff expects in the subsequent revision of NEI 00-04 to address how their process identifies/tags SSCs susceptible to known degradation mechanisms

10. Regulatory Commitments (Comment E23-2)

In response to a statement in NEI 00-04 Revision C on addressing changes to NRC commitments, the Staff stated that a licensee needed to establish under what conditions they would notify NRC of changes in categorization and/or resulting treatment

Revision D (§ 11.1) does not reference NEI 99-04, but states that any regulatory commitments associated with the special treatment requirements for SSCs categorized as RISC-3 would no longer be applicable and may be dropped at the license's discretion

---

Staff expects licensees to evaluate the RISC-3 SSC regulatory commitments to ensure they can be eliminated since some commitments may relate to design issues, which if eliminated would result in noncompliance with 10 CFR 50.69

Staff expects this discussion to be corrected in the subsequent revision of NEI 00-04

11. Miscellaneous

Change manual suppression fire sensitivity study wording

Fire CDF must address contribution from screened fire areas

Clarify meaning of "safe shutdown path" for other external events

Change NUMARC 91-06 approach CDF/LERF terminology

CONCLUSIONS

- NEI 00-04 Revision D is an improvement
- Staff and NEI are converging on approach
  1) Relatively few technical issues remain
  2) Issues can be resolved by providing additional clarifications and/or more specific guidance
- Providing these additional clarifications/guidance to Revision D could result in no objections from the staff; with the Regulatory Guide providing only some general staff positions and staff interpretations

Master

NEI DRAFT 50.69 CATEGORIZATION PROCESS

Presented To:

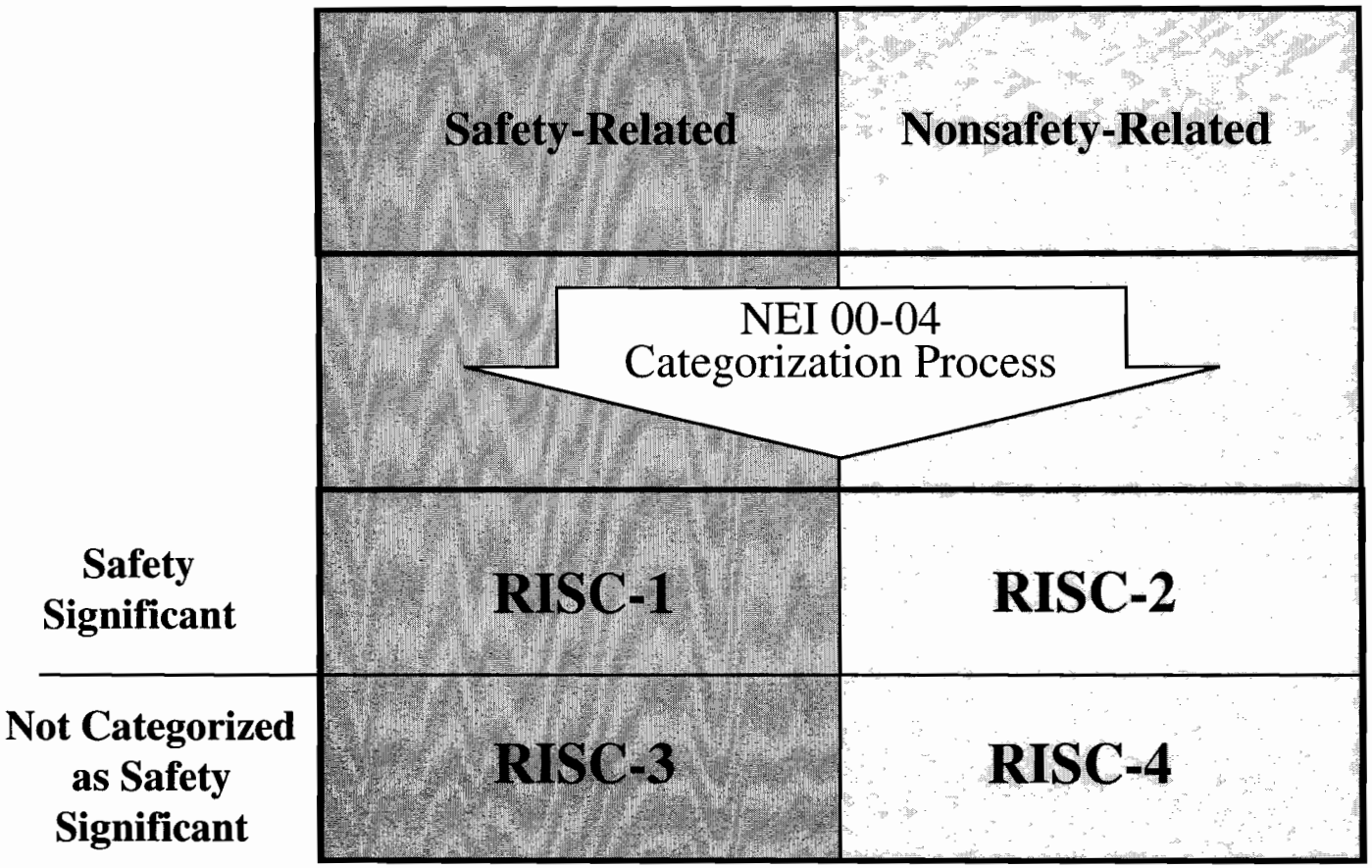
ACRS

NEI Option 2 Task Force

Doug True

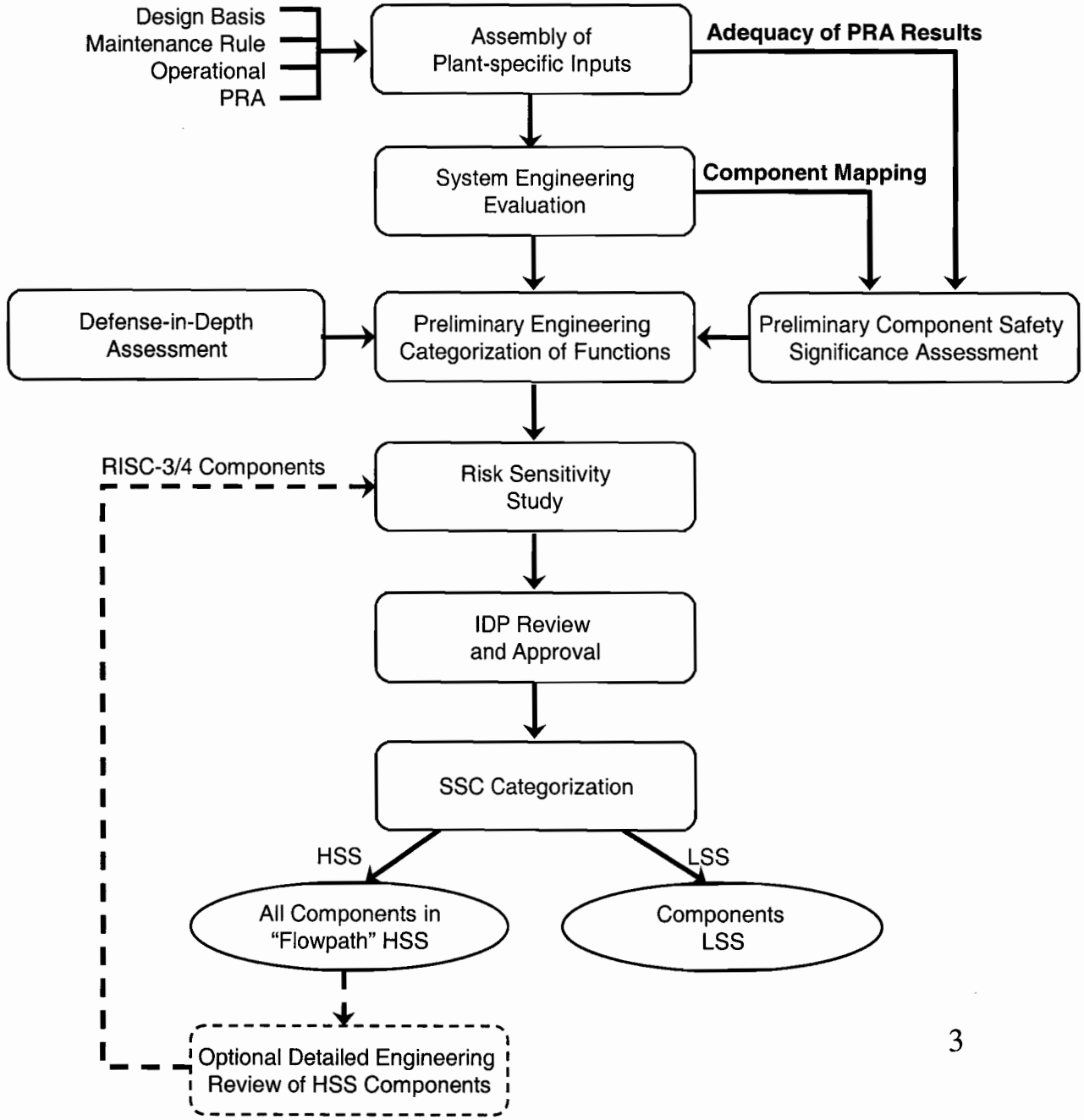
NEI

RISK-INFORMED SAFETY CATEGORIZATION



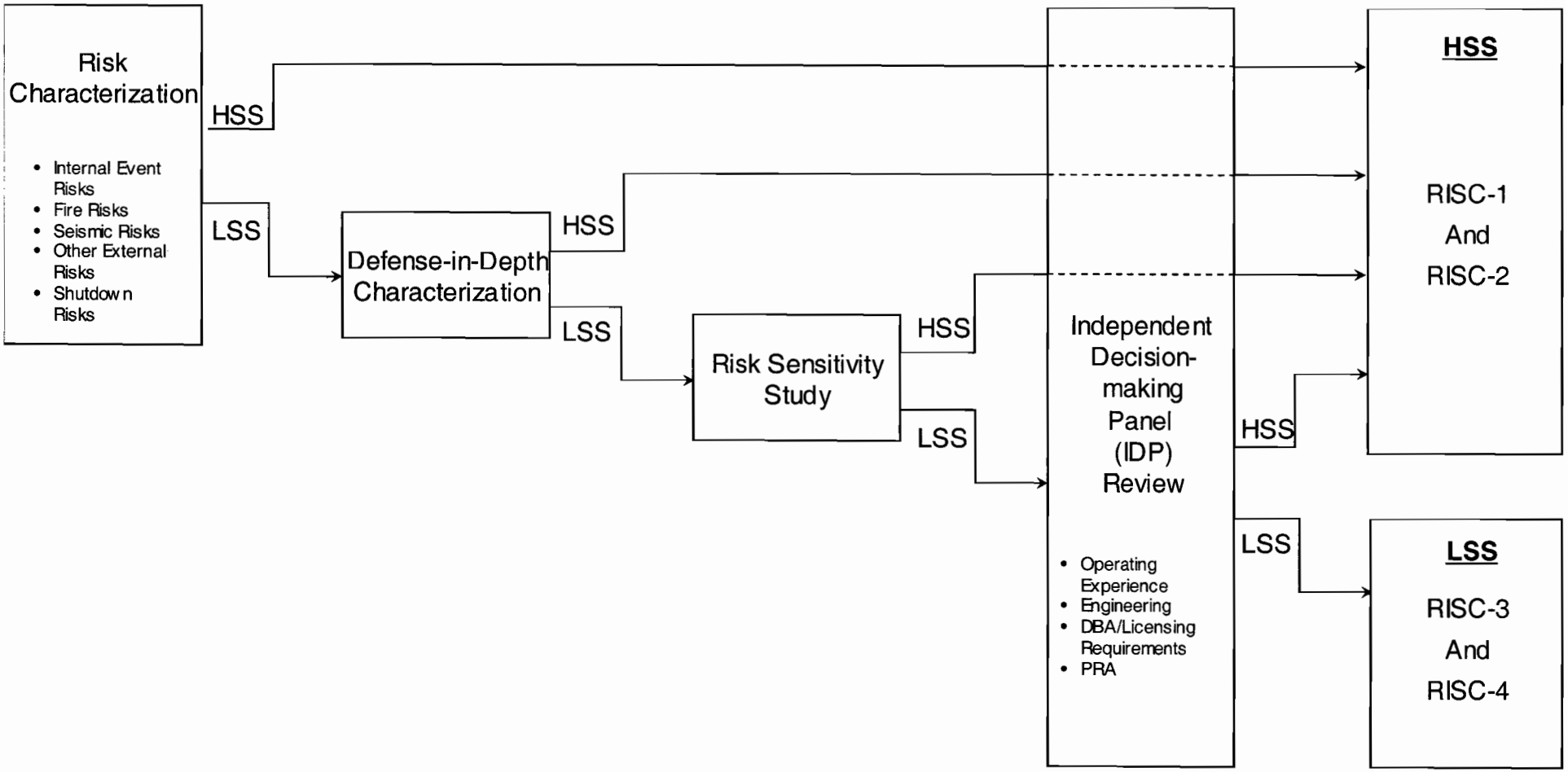
NEI

NEI 00-04 CATEGORIZATION PROCESS



NEI

OVERVIEW OF CATEGORIZATION APPROACH



NEI

RISK CHARACTERIZATION

Risk Source	Alternative Approaches	Scope of Safety Significant SSCs
Internal Events	PRA Required	Per PRA Risk Ranking
	Screening Approaches Not Allowed	n/a
Fire	Fire PRA	Per PRA Risk Ranking
	FIVE (Fire Induced Vulnerability Evaluation)	All SSCs Necessary to Maintain Low Risk
Seismic	Seismic PRA	Per PRA Risk Ranking
	SMA (Seismic Margins Analysis)	All SSCs Necessary to Maintain Low Risk
High Winds, External Floods, etc.	PRA	Per PRA Risk Ranking
	IPEEE Screening	All SSCs Necessary to Protect Against Hazard
Shutdown	Shutdown PRA	Per PRA Risk Ranking
	Shutdown Safety Plan	All SSCs Required to Support Shutdown Safety Plan

NEI

EXAMPLE APPLICATION OF IMPORTANCE MEASURES

COMPONENT FAILURE MODE	F-V	RAW	CCF RAW
1) Valve 'A' Fails to Open	0.002	1.7	n/a
2) Valve 'A' Fails to Remain Closed	0.00002	1.1	n/a
3) Valve 'A' In Maintenance (Closed)	0.0035	1.7	n/a
4) Common Cause Failure of Valves 'A', 'B' & 'C' to Open	0.004	n/a	54
5) Common Cause Failure of Valves 'A' & 'B' to Open	0.0007	n/a	5.6
6) Common Cause Failure of Valves 'A' & 'C' to Open	0.0006	n/a	4.9
Component Importance	0.01082 (sum)	1.7 (max)	54 (max)
Criteria	> 0.005	>2	>20
Candidate Safety Significant?	**Yes**	**No**	**Yes**

NEI

SENSITIVITY STUDIES

Recommended Sensitivity Studies - Internal Events -
• Increase all human error basic events to their 95th percentile value • Decrease all human error basic events to their 5th percentile value • Increase all component common cause events to their 95th percentile value • Decrease all component common cause events to their 5th percentile value • Set all maintenance unavailability terms to 0.0 • Any applicable sensitivity studies identified in the characterization of PRA adequacy

NEI

USE OF IMPORTANCE MEASURES

- **Importance measures are used to identify potentially significant system functions**
- **Importance measures useful as they measure a relative impact on CDF/LERF, thus focusing categorization on maintaining current level of safety**
- **Key limitations of importance measures are addressed in categorization process:**
  - Training of the IDP on interpretation of importance measures
  - Limitations of importance measures identified in Reg Guide 1.174 addressed in process

NEI

RG 1.174: LIMITATIONS OF IMPORTANCE MEASURES

RG 1.174 Issue	Manner Addressed
Truncation Limit	• Explicitly addressed in guidance
Risk Metric	• Importance measures for both CDF & LERF used • Separate consideration of hazards
Completeness	• Full scope of hazards addressed separately, then combined
Uncertainties	• Investigated in EPRI study
Impact of CCF	• F-V &/RAW of CCF events included in evaluation • Sensitivity studies required for CCF values
Recovery actions	• Sensitivity studies for HFEs
Not Good for Multiple Component	• Risk sensitivity study
Relationship to Risk Change	• Risk sensitivity study
SSCs not in solution (unmodeled)	• System function approach includes all SSCs that can impact function

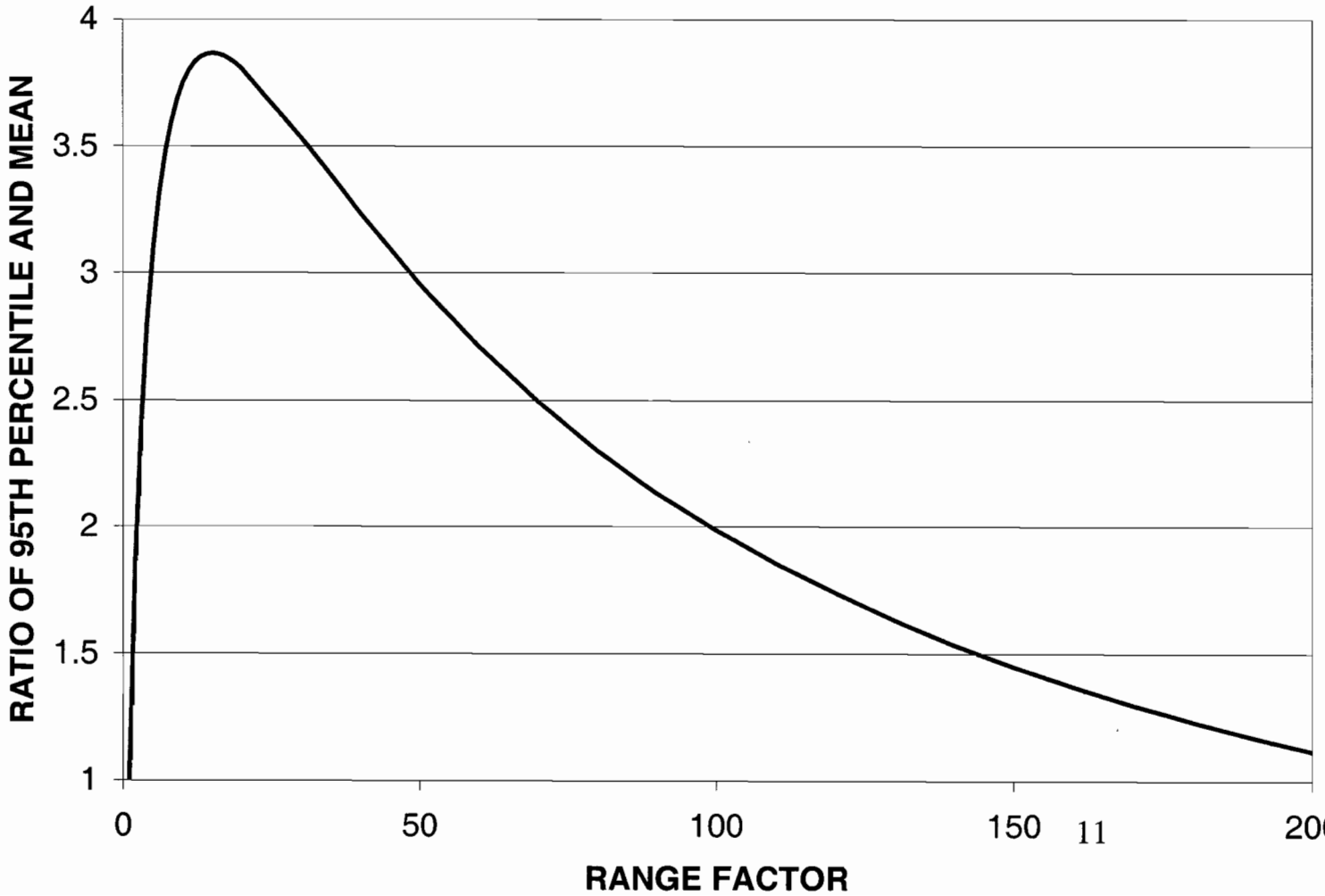
NEI

EPRI STUDY ON UNCERTAINTIES IN NEI 00-04 CATEGORIZATION PROCESS

- **Purpose:**
  Evaluate ACRS questions about the impact of uncertainties on the importance measures used in NEI 00-04 categorization process
- **Utilized same internal events PRA used in BWROG pilot**
- **Considered:**
  - General Evaluation Of Uncertainties
  - Point Estimate Results
  - Monte Carlo Results (Mean & Percentile)
  - Sensitivity Study Results

NEI

RATIO OF 95TH PERCENTILE TO MEAN VALUE FOR LOGNORMAL DISTRIBTUTION



SUMMARY OF CATEGORIZATION FINDINGS

SSC	FV or RAW	Safety Significance			
		Method #1	*Method #2*	*Method #3*	*Method #4*
		Point Estimate	True Mean	Uncertainty Distribution Propagation	Sensitivity Calculations(2)
Feedwater	RAW	LSS	LSS	LSS	LSS
	FV	SS	SS	SS	SS
	Integrated	SS	SS	SS	SS
RCIC	RAW	LSS	LSS	SS	SS
	FV	SS	SS	SS	SS
	Integrated	SS	SS	SS	SS
LPCS	RAW	LSS	LSS	LSS	LSS
	FV	LSS	LSS	LSS	LSS
	Integrated	LSS	LSS	LSS	LSS

NEI

KEY CONCLUSIONS FROM EPRI TR-1008905

- PRA codes calculate importance measures based on point estimate models
- The correlated means for the importance measures calculated from a Monte Carlo evaluation are higher than point estimates
- The correlation effect may have a influence on the calculation of the mean F-V value, especially for low F-V SSCs.
- However, the parametric correlation effect does not change the safety significance assessment
- The NEI 00-04 sensitivity studies encompass the correlation effect on the mean importance measures
- Either a formal parametric uncertainty assessment or a series of sensitivity studies provides equivalent results for use in the safety significance determination process

NEI

DEFENSE-IN-DEPTH ASSESSMENT

- **Addresses SSCs Categorized As RISC-3 in Risk Characterization**
- **Deterministic Rules/Questions Addresses Defense-in-Depth Relative to:**
  - Core Damage Prevention
  - Large Early Containment Failure
  - Long-term Containment Integrity
- **Indication That SSC is Necessary for Defense-in-Depth Categorizes it As RISC-1**

NEI

DEFENSE-IN-DEPTH MATRIX

Frequency	Design Basis Event	>3 diverse trains OR 2 redundant systems	1 train + 1 system with redundancy	2 diverse trains	1 redundant automatic system
>1 per 1-10 yr	Reactor Trip Loss of Condenser				
1 per 10-10^2 yr	Loss of Offsite Power Total loss of Main FW Stuck open SRV (BWR) MSLB (outside cntmt) Loss of 1 SR AC Bus Loss of Instr/Cntrl Air			POTENTIALLY SAFETY SIGNIFICANT	
1 per 10^2-10^3yr	SGTR Stuck Open PORV/SV RCP Seal LOCA MFLB MSLB Inside Loss of 1 SR DC bus	LOW SAFETY SIGNIFICANCE CONFIRMED			
<1 per 10^3 yr	LOCAs Other Design Basis Accidents				

DETERMINISTIC D-I-D QUESTIONS

Containment Bypass

- **Can the SSC initiate or isolate an ISLOCA event?**
- **Can the SSC isolate a faulted steam generator following a steam generator tube rupture event?**

Containment Isolation

- **Does the SSC support containment isolation for containment penetrations that are:**
  - >2" in diameter,
  - part of a system that is not considered closed as defined in GDC 57,
  - not normally closed or locked closed, and
  - not a part of a normally liquid filled system?

Early Hydrogen Burns

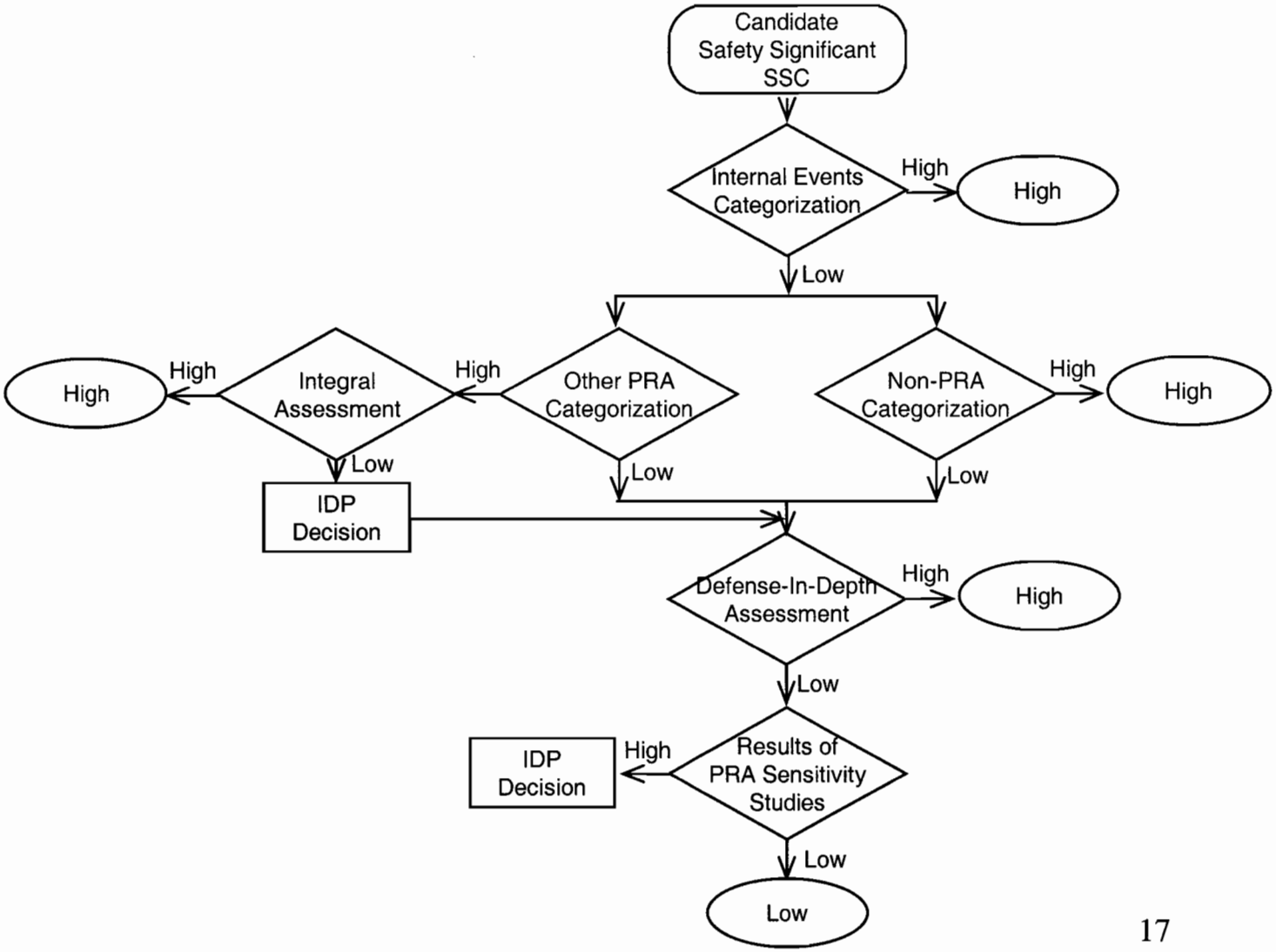
- **Does the SSC support operation of hydrogen igniters in ice condenser and Mark III containments?**

Long-term Containment Integrity

- **Does the SSC support a system function that is not considered in CDF and LERF, but would be the only means for preserving long-term containment integrity post-core damage (i.e., containment heat removal)?**

NEI

CATEGORIZATION FOR INPUT TO IDP



NEI

RISK SENSITIVITY STUDY

- Assesses Bounding Change in CDF/LERF Assuming a Simultaneous Decrease in Reliability of All LSS SSCs
- Factor of Increase Set By Determining Amount of Simultaneous Change Detectable in Corrective Action Programs
- Comparison to RG 1.174 CDF/LERF Guidelines

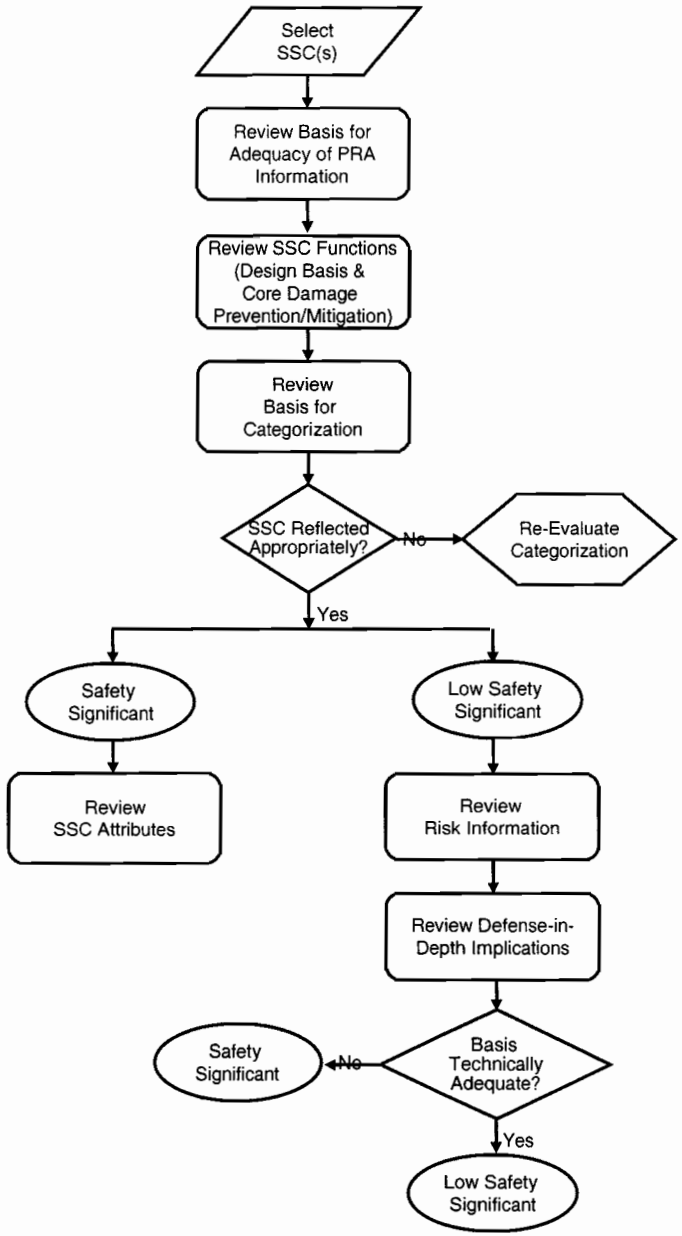
NEI

INTEGRATED DECISION-MAKING PANEL (IDP)

- **Confirms Technical Basis for Categorization (e.g., SSC functions, use, performance, etc.)**
  - The IDP may request re-assessment of categorization, if basis is found to be inadequate
- **Specific Review of LSS SSCs To Confirm Defense-in-Depth. May Move SSCs from LSS to HSS.**
- **Not Allowed to Move SSCs From HSS to LSS**

NEI

IDP REVIEW PROCESS



SUMMARY

- Rigorous Risk-informed Categorization Process
- Utilizes Strengths of PRA
- Addresses Limitations of PRA
- Allows Use of Non-PRA Analyses, But Standard for Safety Significance Applied Conservatively
- Believe NRC Staff's Major Issues on Categorization Have Been Resolved

NEI

Master

Status of Risk-Informed Initiatives in ASME Nuclear Code and Standards

Advisory Committee on Reactor Safeguards
Meeting of the Subcommittee on
Reliability and Probabilistic Risk Assessment

American Society of Mechanical Engineers

Rockville, Maryland - February 19, 2004

1

Status of Risk-Informed Initiatives in ASME Nuclear Code & Standards

AGENDA

- Introductions and Purpose of Briefing
- ASME Consensus Codes & Standards Organization
- ASME Board on Nuclear Codes and Standards Risk Management Strategic Plan
  - PRA Standards Development
  - Efforts to Support 10 CFR 50.69 Initiative
  - Initiatives to Develop Framework for New Reactors
  - Proposed Nuclear Risk Management Coordinating Committee
- Summary

2

Status of Risk-Informed Initiatives in ASME Nuclear Code & Standards

ASME
SETTING THE STANDARD

INTRODUCTION – PARTICIPATING ASME MEMBERS

- Wes Rowley – Vice-President, Nuclear Codes & Standards and Chair, Board on Nuclear Codes & Standards
- Ray Weidler – Vice-Chair, Board on Nuclear Codes & Standards
- Kevin Ennis – ASME Director, Nuclear Codes & Standards
- Ken Balkey – Chair, ASME BNCS Risk Management Task Group
- Bryan Erler – Chair, ASME BNCS Task Group Regulatory Endorsement
- Craig Sellers – Member, ASME Operations & Maintenance Committee
- Gil Zigler – Member BNCS, Vice-Chair, Committee on Nuclear Risk Management

3

Status of Risk-Informed Initiatives in ASME Nuclear Code & Standards

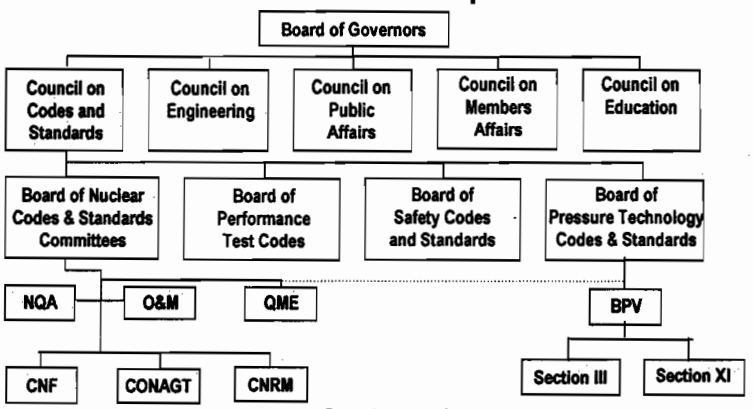
ASME
SETTING THE STANDARD

PURPOSE OF BRIEFING

- To Provide Status and Overview of Key Risk-Informed Initiatives Within ASME Nuclear Codes & Standards to Support Both Operating and New Reactors
- To Define Any Appropriate Actions or Impacts Related to Corresponding Regulatory Developments

4

ASME Organization Chart for Nuclear Codes and Standards Development



Over 3,000 volunteers
Participate in this process

5

ASME BNCS Risk Management Strategic Plan

MISSION

Factor risk, as well as performance, into all ASME Nuclear Codes and Standards, as appropriate, to further ensure, on a worldwide basis, that they protect public health and safety and meet the needs of users

ACTION PLAN

Identify and track short term (1-2 year) and long term (3+ year) risk-informed initiatives within the Committee on Board Operations and across all BNCS Committees

PRA STANDARDS DEVELOPMENT

Identify needs to enhance or develop PRA Standards to support applications defined in the action plan

UPDATES

Plan is updated and approved at each BNCS meeting (~4 months) and is then placed on the ASME website for public access

6

ASME PRA Standards Developments

PRA STANDARD DEVELOPMENTS

- ASME RA-S-2002, "*Standard For Probabilistic Risk Assessment For Nuclear Power Plant Applications,*" published April 2002
- Addendum A to ASME PRA Standard issued in Dec. 2003 to address NRC clarifications in RG-1.200
- ASME PRA Standard used in plant-specific PRA peer review at San Onofre in June 2003; Project Team drafting Addendum B to PRA Standard to address issues that arose in using the standard

NEW INITIATIVES

- Identify actions necessary to respond to the December 18, 2003 Commission paper on PRA Quality
- Work with proposed Nuclear Risk Management Coordinating Committee
- Investigate development of reliability data base to support the PRA Standard

7

ASME Section XI Risk-Informed Code Cases

ASME SECTION XI - INSERVICE INSPECTION (ISI) AND REPAIR/REPLACEMENT OF PRESSURE-RETAINING ITEMS			
CODE CASE	CONTENT	CLASSIFICATION	TREATMENT
N-577	Risk-Informed ISI for Class 1, 2, or 3 Piping - Method A	Yes	Yes
N-578	Risk-Informed ISI for Class 1, 2, or 3 Piping - Method B	Yes	Yes
N-660	Risk-Informed Safety Classification for Use in Risk-Informed Repair/Replacement	Yes	No
N-662	Alternative Repair/Replacement Requirements for Items Classified w/RI Processes	No	Yes

8

ASME O&M
Risk-Informed Code Cases

ASME OPERATIONS & MAINTENANCE - INSERVICE TESTING OF PUMPS, VALVES, & MECHANICAL EQUIPMENT			
CODE CASE	CONTENT	CLASSIFICATION	TREATMENT
OMN-3 Revision 1	Requirements for Safety Significance Categorization of Components Using Risk Insights for IST of LWR Power Plants	Yes	No
OMN-4	Requirements for Applying Risk Insights for IST of Check Valves of LWR Power Plants	No	Yes
OMN-7	Requirements for Applying Risk Insights for IST of Pumps of LWR Power Plants	No	Yes
OMN-10	Requirements for Safety Significance Categorization of Snubbers Using Risk Insights and Testing Strategies for IST	Yes	Yes
OMN-11	Risk-Informed Inservice Testing of Motor-Operated Pumps (Used in Conjunction with Code Case OMN-1)	No	Yes
OMN-12	Alternate Requirements for IST Using Risk Insights for Pneumatically and Hydraulically Operated Valve Assemblies	No	Yes

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ASME Efforts to Support 10 CFR 50.69 Initiative

- Regular interface with NRC and NEI
- ASME formal positions on §50.69 package
- On-going application activities
  - ASME Section XI Code Cases N-660 and N-662 for risk-informed repair / replacement
  - ASME O&M Committee proposed Standard for treatment of RISC-3 pumps and valves

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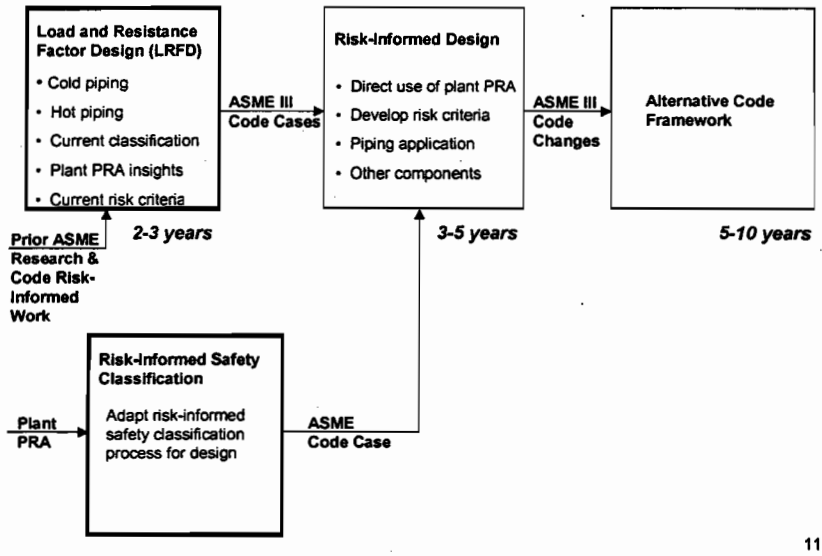
Initiatives to Develop Framework for Future Reactors



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Initiatives to Develop Code Requirements for New Reactors

- ASME is in the process of holding a series of work shops with new reactor suppliers to determine their needs
- New reactors use risk-informed technology to support design efforts
- Initial new reactor plants will use a combination of risk-informed system performance requirements and ASME design, fabrication and inspection rules

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Proposed Nuclear Risk Management Coordinating Committee

ASME
SETTING THE STANDARD

MEMBERS

- ASME, ANS and NRC have proposed to form committee
- Committee would also include DOE, NEI, and interested standards developing organizations, such as IEEE, ASTM, NFPA, and AISC

PURPOSE AND OBJECTIVES

- Coordinate codes and standards activities related to risk management for current and new nuclear power plants, nuclear facilities, and the transportation and storage of nuclear material
- Ensure that codes and standards associated with risk management, and their underlying principles, are consistent and compatible

CURRENT ACTIONS

- White paper drafted and issued for review; Defines need for Committee and potential short term and long term initiatives
- Inaugural meeting to be held Feb. 20, 2004 at ASME Offices in Wash, DC

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Summary

ASME
SETTING THE STANDARD

- ASME BNCS has developed a Risk Management Strategic Plan to bring risk technology into the development of consensus standards that are beneficial to its users
- ASME efforts to risk-inform nuclear Codes & Standards requirements, where appropriate, can be correlated with the NRC 10 CFR Part 50.69 Proposed Rule
- ASME is working with NRC, DOE, industry, and other Standards Development Organizations to coordinate codes & standards activities related to risk management such as PRA Standards, an appropriate framework to support new reactors, and other initiatives
- Interface with the ACRS on these developments in the future, as appropriate

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Master

ASME Nuclear Codes & Standards – Version: Feb. 2004

Risk Management Strategic Plan
by
BNCS Risk Management Task Group

1. Mission

Factor risk, as well as performance, into all ASME Nuclear Codes and Standards, as appropriate, to further ensure, on a worldwide basis, that they protect public health and safety and meet the needs of users.

2. Goal

To determine the need and provide risk-informed, performance-based ASME Nuclear Codes and Standards for the benefit of users.

3. Summary of Proposed Initiatives

The following short and long term initiatives summarize the specific actions identified in sections 5 through 8 of this Plan:

Short Term (2004-2005)

3.1 Convert Risk-Informed ISI Code Cases into a Non-Mandatory Appendix in ASME Section XI.

3.2 Integrate Risk-Informed IST Code initiatives relative to existing Code Cases and treatment requirements for low safety significant pumps and valves into proposed Subsection ISTE of ASME OM Code.

3.3 Ensure that ASME Risk-Informed Code Cases and Codes & Standards are properly integrated with Risk-Informed Regulation Initiatives.

3.4 Develop new Addendum to the Probabilistic Risk Assessment (PRA) Standard to clarify items defined from recent use of the Standard for plant-specific PRA review.

3.5 Integrate application of PRA Standard into other ASME Risk-Informed Codes and Standards, as appropriate.

3.6 Assist in formation of a Nuclear Risk Management Coordinating Committee to coordinate risk management development activities by SDOs, regulators, and industry, where appropriate.

3.7 Support ASME Research effort to Develop Reliability-Based Load and Resistance Factor Design (LRFD) Methods for Piping in ASME Section III.

3.8 Adapt the provisions of the Risk-Informed Safety Classification Code Case for Repair/Replacement Activities (Case N-660) for consideration of a new classification Case for Section III design.

3.9 Investigate use of LRFD and risk-informed methods for concrete components for nuclear service for Joint ACI-ASME III design.

3.10 Complete risk-informed initiatives on minimum exam coverage, pressure testing & leakage, and heat exchanger exams.

3.11 Using experience from risk-informed ISI applications and trial risk-informed repair/replacement applications, develop alternative piping risk-informed

classification and exam requirements, revise Code Case N-660 to address service water systems, and revise Code Case N-660 for all plant systems to support 10 CFR 50.69 initiative.

3.12 Develop an Appendix that risk-informs the requirements of NQA-1

3.13 Investigate approaches for the development of a Life-Cycle Risk-Informed Nuclear Code and make decision regarding Nuclear System Code development.

3.14 Investigate development of an O&M Standard on treatment of LSSC pumps and valves excluded from OM Code requirements through 10CFR50.69.

3.15 Identify actions necessary to respond to Commission paper COMNJD-03-0002 on PRA quality

Long Term (2006-Beyond)

3.16 Integrate risk-informed, performance-based approaches into new ASME Nuclear Codes & Standards to support next generation reactors.

3.17 Implement and complete scope additions to the PRA Standard.

3.18 Develop risk-informed safety classification Code Case(s) for use across all ASME Nuclear Codes & Standards.

4. Background

The consideration of risk has always been part of ASME's development of Codes and Standards. Traditional engineering methods, such as use of design factors, were incorporated into Codes and Standards to manage this risk. With the emergence of PRA technology, ASME has the opportunity to better focus its guidance on the most important risks. The mission of this Risk Management Strategic Plan is to assist BNCS in infusing PRA methods and perspectives into Codes and Standards development and implementation so that these Codes and Standards continue to effectively and efficiently provide for public health and safety.

For over a decade the nuclear industry has had initiatives in the utilization of risk for various regulatory requirements that reduce unnecessary burden while maintaining a high level of safety. Every plant has a PRA, primarily due to the industry response to NRC Generic Letter 88-20, and in meeting the requirements of the NRC Maintenance Rule, 10 CFR 50.65.

Over the last decade the ASME has developed risk-informed component applications for inservice testing (IST), inservice inspection (ISI), and repair/replacement activities in the form of Code Cases. Also, the ASME has developed the PRA Standard for commercial nuclear power plants.

5. Matrix of ASME Nuclear Codes & Standards Risk Management Developments

Table 1 has been developed and will be maintained to guide the development of risk management initiatives within the ASME BNCS. The table summarizes current and potential risk applications within the BNCS committees. In addition, the applications are related to the need for enhancement or development of PRA Standards.

ASME Nuclear Codes & Standards – Version: Feb. 2004

Table 1. Potential Applications within BNCS Committees / Relation to PRA Standards Development

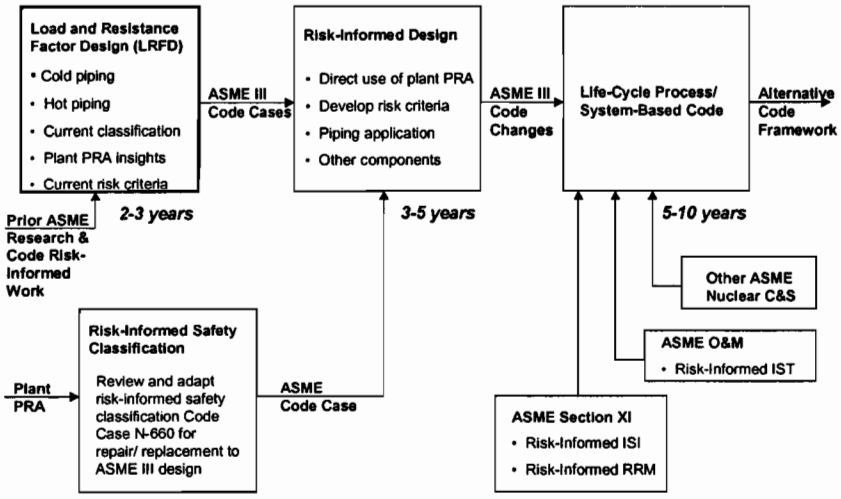
Nuclear Committee	Current Product	Can Use Existing PRA Standards?	Another PRA Standard Required?
Operation & Maintenance (O&M) – for pump & valve IST	Code Cases/ New ISTE	Yes, pumps & valves are modeled	No
O&M – for Snubber IST	Code Case	Yes, for supporting information; Snubbers are not explicitly modeled	May benefit from enhancement
Section XI – for piping ISI	Code Cases/ Non-Mandatory Appendix	Yes, for supporting information; Piping is not explicitly modeled	May benefit from enhancement
Section XI – for other component ISI	Code Cases	Yes, for supporting information; SSCs are not explicitly modeled	May benefit from enhancement
Section XI – for pressure-retaining item repair/replacement	Code Cases	Yes, for supporting information; SSCs are not explicitly modeled	May benefit from enhancement
Section III – Div 1 for risk-informed piping design	Under Development	Yes, for supporting information for existing reactor designs; Piping is not explicitly modeled	PRA standard may need to be expanded to address new generation reactor designs
Section III – Div 1 for other components	Under Review	Same as above	Same as above
Section III – Div 2 for concrete / metal structures	Under Review	Maybe to a limited degree; ANS Seismic and External Events Standard may have significant input	Existing standards may have to be modified
Section III – Div 3 for nuclear casks	None	Not likely	New standards would need to be developed
Nuclear Quality Assurance	Under Development	Yes with some possible modification	No. NQA should develop requirements compatible with risk categorization
Qualification of Mechanical Equipment – Valve Qualification	Previous Work on Hold	Yes with some possible modification	No
Nuclear Air & Gas Treatment Equipment	None	Yes, most SSCs are implicitly modeled; However passive components are not	May benefit from enhancement
Cranes For Nuclear Facilities	None	Not likely but impact of crane failures could be possibly evaluated	A new standard may be required to address crane structures / mechanical systems

		ANS Seismic and External Events Std may have significant input	
Nuclear Risk Management	PRA Standard/ Review of Potential New Standards	Not Applicable	New PRA Standards (or expanded scope) may need to be developed for the following needs: - Reliability data base standard to support PRAs - Treatment of items not modeled - Global risk-informed safety classification - Address needs of high temperature reactor designs - Independent Decision-Making Panel

6. Nuclear Systems Code

Current ASME nuclear codes and standards rely primarily on deterministic and mechanistic approaches to design of components, including piping systems. The design code is a separate volume from the code for ISI and both are separate from the code for IST. A decision needs to be made as to whether and to what extent an ASME Nuclear Systems Code should be developed, which would include a planned evolution that integrates the various nuclear codes and standards and adopts a risk-informed approach across a facility life-cycle – encompassing design, construction, operation, maintenance, and closure. Figure 1 offers a conceptual development.

Figure 1. Potential Evolution to Nuclear Systems Code



7. Plan for 2004 and 2005

Begin new initiatives and enhance the current series of ASME Risk-Informed products for the current generation of operating nuclear power plants as shown in Table 2.

Table 2. Short Term BNCS Risk Management Enhancement Actions and Status

Nuclear Committee	Enhancement Action	Status
Board Operations	Risk Management Task Group work with regulator & industry to insure that ASME risk-informed Codes & Standards are properly integrated in risk-informed regulation initiatives	ASME letters issued and discussed with NRC and industry; ASME comment letter issued and discussed with NRC in finalization of 10 CFR 50.69; ASME briefing with ACRS planned for 02-19-04
	Work with ANS and NRC to form Nuclear Risk Management Coordinating Committee to coordinate risk management development activities by SDOs, regulators, and industry.	White paper on purpose and potential activities prepared and reviewed with ANS and NRC; Telecon with members held 02-03-04; Inaugural meeting to be held 02-20-04 at ASME Washington, DC
	Form Task Group to recommend approaches for development of life-cycle process / systems-based Code using risk insights	Initial training presentation prepared; Dr. Asada to report at 02-17-04 CBO meeting
Operation & Maintenance of Nuclear Power Plants	Integrate RI-IST Code initiatives relative to existing Code Cases into proposed Subsection ISTE of ASME OM Code and	Revised Subsection ISTE resubmitted and out for pre-ballot review by Committee;
	Investigate the development of a standard describing acceptable treatment requirements for low safety significant pumps and valves excluded from OM Code requirements through 10CFR50.69.	Scope and need statement for treatment requirements for low safety significant pumps and valves out for ballot by Committee.
	Revise RI-IST snubber Code Case OMN-10 to address pressure boundary	Revision of OMN-10 underway
Section XI Subcommittee	Incorporate RI-ISI Code Cases into Non-Mandatory Appendix	Non-mandatory Appendix voted by ASME XI; addressing negatives dealing with role of Authorized Nuclear Inspector
	Evaluate extending the provisions of Case N-660 to also cover Section XI pressure testing and leakage	At Working Group on Implementation of Risk-Based Examination
	Develop Code Case addressing minimum exam coverage using risk insights	Code Case approved by WG on Risk; Being balloted by SG Water-Cooled Sys
	Develop Code Case on heat exchanger exams using risk insights	Under development within TG on Optimization and WG on Risk
	Develop Code Case providing alternative piping risk-informed classification and exams using experience from risk-informed ISI applications	At Working Group on Implementation of Risk-Based Examination
	Revise Code Case N-660 to address service water systems using trial application experience	At Working Group on Implementation of Risk-Based Examination
	Revise Code Case N-660 to address all plant systems using trial applications experience to support 10 CFR 50.69 initiative	Under development within Task Group on Risk-Informed Classification and to be presented to Working Group on Implementation of Risk-based Examination 2-23-04

ASME Nuclear Codes & Standards – Version: Feb. 2004

Section III – Div 1 Subcommittee	Begin research effort to develop Code Case(s) using Reliability-Based Load and Resistance Factor Design (LRFD) for piping	Agreement in place with International Institute of Universality of Japan to support; NRC support soon expected; Under review at DOE; Research effort soon to be initiated
	Coordinate with CNRM and Section XI on adapting Code Case N-660 for Risk-Informed Safety Classification for Design	Task group formed between Section III, XI, and CNRM; draft Code Case for risk-informed safety classification for design prepared and comments being addressed; Attributes matrix comparing technical and administrative requirements of Class 1, 2, 3 and B31.1 piping developed and being reviewed within ASME Section III
Section III – Div 2	Investigate use of LRFD and risk-informed methods for concrete components	Technical presentation given at Joint ACI-ASME Committee meeting 9-30-03; Task Group formed to investigate methods
Section III – Div 3	Consider using a non-PRA risk standard for Nuclear Cask applications	
Nuclear Quality Assurance	Develop an Appendix that risk-informs the requirements of NQA-1	Work is in progress
Qualification of Mechanical Equipment	Consider use of probabilistic methods in valve qualification standards	Research proposal prepared; Currently on-hold
Nuclear Air & Gas Treatment Equipment	Investigate use of risk-informed methods in testing of mechanical equipment	
Cranes For Nuclear Facilities	Consider using a non-PRA risk standard for nuclear cranes applications	
Nuclear Risk Management	Maintain PRA Standard (fold in or reference fire, external events, internal flooding, containment); Need to discuss low power and shutdown PRA	TBD
	Develop requirements and guidance for treating items not modeled in the PRA	Task assigned to Subcommittee on Applications for development
	Consider reliability data base standard for PRA	Research proposal approved by BNCS; Awaiting direction to proceed forward
	Work with other committees and SDOs to begin development of new risk-informed safety classification standard	See Section III Div 1 Subcommittee Item 2 for actions.
	Incorporate experience in first use of the ASME PRA Standard	Develop new Addendum to clarify items defined from recent use of the Standard for plant-specific PRA review; Project team is preparing draft document for April 2004 CNRM meeting
	Develop guidance on Independent Decision-Making Panels	Task assigned to Subcommittee on Applications for development
	Identify actions necessary to respond to Commission paper COMNJD-03-0002 on PRA quality	Task assigned to Subcommittee on Applications

8. Plan for 2006 and Beyond

Enhance ASME Risk-Informed Codes & Standards products, particularly to support next generation reactor needs. Develop new risk-informed safety classification process for use on all ASME Nuclear Codes & Standards. Begin efforts to move ASME Nuclear Codes & Standards toward a life-cycle process / system-based Code using risk insights, if decision made in 2004-2005 to begin this initiative. Table 3 summarizes the long term development needs.

Table 3. BNCS Long Term Risk Management Enhancement Actions

Nuclear Committee	Enhancement Action
Board Operations with All Committees	Implement recommendations of CBO Task Group to develop a life-cycle process / systems-based Code using risk insights, as required
Operations & Maintenance of Nuclear Power Plants	Revise Subsection ISTE of OM Code for risk-informed IST to support next generation reactor needs
	Revise Standard on treatment of LSSC Pumps and Valves to support next generation reactor needs.
Section XI Subcommittee	Review and revise, as appropriate, all risk-informed Code Cases and Non-Mandatory Appendices related to ISI, pressure testing, and repair/replacement activities to support next generation reactor needs
Section III – Div 1	Complete Code Case(s) on LRFD for piping, including merger of LRFD with new risk-informed safety classification process using plant PRA
	Begin research & Code Case development for use of LRFD for other components, including support of new generation reactor needs
Section III – Div 2	Begin development of use of LRFD / risk-informed methods for concrete components
Section III – Div 3	To be defined later
Nuclear Quality Assurance	Complete development of GQA Standard providing treatment in accordance with new risk-informed safety classification, including support of new generation reactor needs
Qualification of Mechanical Equipment	Develop standard for use of probabilistic methods in valve qualification requirements, including support of new generation reactor needs
Nuclear Air & Gas Treatment Equipment	Develop standard for use of risk-informed methods in testing of mechanical equipment, including support of new generation reactor needs
Cranes For Nuclear Facilities	To be defined later
Nuclear Risk Management	Develop PRA Standard suitable to support next generation reactor design needs
	Consider extension of PRA Standard to full Level 2 and Level 3 to support risk-informed regulation developments, including plant siting
	Develop new risk-informed safety classification standard in collaboration with other committees and SDOs

Master

NUCLEAR RISK MANAGEMENT

Proposed Standards Development Organization, Regulatory and Industry Risk Management Coordinating Committee

The American Society of Mechanical Engineers, the American Nuclear Society and the U.S. Nuclear Regulatory Commission, propose to form a Nuclear Risk Management Coordinating Committee. This committee would coordinate codes and standards activities associated with risk management for current and new nuclear power plants, nuclear facilities, and the transportation and storage of nuclear material. The committee would include representatives from the U.S. Nuclear Regulatory Commission, the U.S. Department of Energy (DOE), the Nuclear Energy Institute, interested standards developing organizations (SDOs), such as IEEE, ASTM, and NFPA, AISC, government agencies, and industry groups, as well as ASME and ANS.

The primary objective of this Committee is to ensure that codes and standards associated with risk management, and their underlying principles, are consistent and compatible. In addition, the committee would coordinate the development of such codes and standards so that affected stakeholders have documents that are coherent and can be applied in a uniform and ordered way. Also, as new topics emerge, the Committee will ensure that the proper organization assumes responsibility.

The benefit of this initiative would be the establishment of a common framework for the development and application of codes and standards as well as regulatory documents. While these initiatives are beneficial to current operating reactors and facilities, the coordination of such initiatives is crucial to support the development and construction of advanced and new reactors. The emergence of new challenges, such as the need to define effective measures to counter terrorism, is an additional motivation to gain closer coordination and collaboration in the use of risk technology.

Work is presently underway by several organizations to risk-inform codes, standards, and regulations for nuclear applications. Attachment 1 provides a list of current projects that illustrate the type of work that would be addressed by the committee.

Attachment 2 provides some preliminary ideas and examples of near-and-long term items that either have been assumed by an organization or need to be assigned to an SDO(s) by the committee to ensure proper coordination and consistency, and to meet user needs.

ATTACHMENT 1

Current Projects

- ASME Probabilistic Risk Assessment (PRA) Standard, Phase I, Level 1 Analysis of internal events at power, and limited Level 2 analysis to evaluate large, early release frequency for internal events at power. Standard was published April 2002 with an Addenda "a" being issued in 2003 to address NRC requested clarifications and an Addendum "b" is currently under development to address insights in use of Standard for plant peer review.
- ANSI/ANS-58.21-2003, External Events in PRA Methodology has been recently published. The standard covers both natural external events (e.g., earthquakes, high winds, and external flooding) and human-made external events (e.g., explosions at nearby industrial facilities, and impacts from nearby transportation activities).
- Internal Fire – ANS is preparing an internal fire PRA standard.
- Low Power Shutdown (LPSD) – ANS is preparing a low power shutdown (LPSD) PRA standard.
- NEI has issued NEI 00-02, "PRA Peer Review Process Guidance" including a "Self-Assessment Process" to address differences between the ASME Standard and NEI 00-02.
- NRC has issued a draft Regulatory Guide 1.200 For Trial Use to address PRA quality and regulatory positions on ASME PRA Standard, NEI Peer Review Process, and the ANS PRA Standard.
- NRC has prepared the Proposed Rule-Making Package for 10 CFR 50.69 ("Risk-Informed Categorization and Treatment of Structures, Systems and Components for Nuclear Power Reactors") as one of the initiatives to risk-inform regulations. Public comments have been received and are currently being addressed by the NRC.
- ASME Code Cases – Code Cases have been developed or are in the process of being incorporated into the main body of the respective Code for Pump and Valve inservice testing (IST), Snubber IST, Piping and Component ISI, and Categorization and Treatment for the Repair/ Replacement of pressure-retaining items.
- NEI has prepared a draft Option 2 Categorization Process Guideline, NEI 00-04 Rev. D, and is preparing a Supplemental Guidance Document to provide detailed treatment guidance for all risk-informed safety classification categories.

ATTACHMENT 2

Preliminary List of Potential Near-and-Long Term Projects

1. Coordination and Integration Projects:

- Develop an overall Risk-Informed Strategic Plan for use by SDO, industry and regulatory personnel.
- Develop a single document providing guidance for meeting PRA requirements rather than requiring the use of multiple documents (ASME, ANS, NEI, NRC, etc.).
- Integrate application of the ASME PRA Standard into other ASME codes and standards.

2. Development Projects:

- Support ASME Research effort to develop reliability-based load and resistance factor design methods for piping in ASME Section III.
- Support the Task Group formed to investigate approaches for the development of a Life Cycle, Risk-Informed Nuclear Code.
- Determine need for, and if appropriate, develop a Standard to develop a Risk-Informed Safety Classification scheme, particularly to assist advanced and new reactor designs.
- Determine need for, and, if appropriate, develop standards for Qualification of RISC-3 items (Safety-Related, Low Safety Significant SSCs).

3. New-Scope Projects:

- Extend plant PRA to full Level 2 and Level 3.
- Address PRA considerations for advanced/new reactor designs.
- Address PRA for other nuclear facilities, transportation and storage of nuclear materials, and related activities.
- Risk-informed treatment of structures, systems and components (SSCs) not modeled in the PRA.
- Develop risk methodology to address terrorism threats at nuclear power plants and facilities as well as the transport and storage of nuclear materials.