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

March 29, 2002

MEMORANDUM TO: ACRS Members

FROM: Michael T. Markley, Senior Staff Engineer
ACRS

SUBJECT: CERTIFICATION OF THE MINUTES OF THE MEETING OF THE ACRS SUBCOMMITTEE ON RELIABILITY AND PROBABILISTIC RISK ASSESSMENT - DECEMBER 4, 2001 - ROCKVILLE, MARYLAND

The minutes of the subject meeting, issued March 14, 2001, have been certified as the official record of the proceedings of that meeting. A copy of the certified minutes is attached.

Attachment: As stated

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

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

MEMORANDUM TO: Michael T. Markley, Senior Staff Engineer

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

SUBJECT: CERTIFICATION OF THE SUMMARY/MINUTES OF THE MEETING OF THE MEETING OF THE ACRS SUBCOMMITTEE ON RELIABILITY AND PROBABILISTIC RISK ASSESSMENT - DECEMBER 4, 2001 - ROCKVILLE, MARYLAND

I do hereby certify that, to the best of my knowledge and belief, the minutes of the subject meeting on December 4, 2001, are an accurate record of the proceedings for that meeting.

George E. Apostolakis, Chairman 3/18/02
Reliability and PRA Subcommittee Date

CERTIFIED BY: Date:3/14/02
G. Apostolakis - 3/18/02

ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
MEETING OF THE ACRS SUBCOMMITTEE ON
RELIABILITY AND PROBABILISTIC RISK ASSESSMENT
MEETING MINUTES - DECEMBER 4, 2001
ROCKVILLE, MARYLAND

INTRODUCTION

The ACRS Subcommittee on Reliability and Probabilistic Risk Assessment met on December 4, 2001, at 11545 Rockville Pike, Rockville, MD, in Room T-2B3. The purpose of this meeting was to discuss proposed revisions to the special treatment requirements of 10 CFR Part 50 (Option 2).

The Subcommittee received no written comments from members of the public regarding the meeting. The entire meeting was open to public attendance. Mr. Michael T. Markley was the cognizant ACRS staff engineer for this meeting. The meeting was convened at 8:30 a.m. and adjourned at 5:00 p.m.

ATTENDEES

ACRS Members

G. Apostolakis, Chairman
M. Bonaca, Member
P. Ford, Member
T. Kress, Member
S. Rosen, Member
W. Shack, Member
M. Markley, ACRS Staff

Principal NRC Speakers

C. Carpenter, NRR*
M. Cheok, NRR
G. Kelly, NRR
E. McKenna, NRR
T. Reed, NRR
S. West, NRR

Principal Industry Speakers

A. Heymer, NEI*
T. Pietrangelo NEI

NRR Office of Nuclear Reactor Regulation
NEI Nuclear Energy Institute

There were approximately 3 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 THE SUBCOMMITTEE CHAIRMAN

Dr. George Apostolakis, Chairman of the ACRS Subcommittee on Reliability and Probabilistic Risk Assessment convened the meeting at 8:30 a.m. He introduced the Subcommittee members in attendance and stated that the purpose of this meeting was to discuss proposed revisions to the special treatment requirements of 10 CFR Part 50 (Option 2). He stated that the ACRS previously issued a report concerning proposed 10 CFR 50.69 and Appendix T on October 12, 1999. He noted that the NRC staff is no longer pursing Appendix T and is now considering proposed industry guidance in NEI 00-04, "Option 2 Implementation Guideline." Dr. Apostolakis noted that the ACRS recently reviewed the license amendment requests from South Texas Project (STP) concerning special treatment of structures, systems, and components (SSCs) and issued a report dated July 23, 2001. He stated that the Subcommittee would consider Option 2 pilot plant initiatives during this meeting. Dr. Apostolakis noted that the Subcommittee had received no written comments from members of the public regarding the meeting.

DISCUSSION OF AGENDA ITEMS

NRC Staff Presentation

Mr. Timothy Reed and Ms. Eileen McKenna, NRR, led the discussions for the NRC staff. Ms. Cynthia Carpenter, Glenn Kelly, and Michael Cheok, NRR, provided supporting discussion. Mr. Steven West, NRR, also participated. The staff discussed the status of ongoing Option 2 tasks, draft rule language for 10 CFR 50.69, and the revised approach utilizing NEI 00-04 rather than the previously proposed Appendix T. Significant points raised during the presentation include:

- The requirements in the proposed 10 CFR 50.69(c) would partition SSC functions into risk-informed safety class (RISC) categories using the plant-specific probabilistic risk assessment (PRA). The evaluation model must include internal initiating events at full-power operations. External initiating events must be considered in the PRA or as part of the integrated decision-making panel (IDP). The IDP must consider PRA results and insights, the importance of SSC functions and operating modes not addressed in the PRA, defense in depth and safety margins. The output of the IDP should ensure that potential increases in core damage frequency (CDF) and large early release frequency (LERF) are small and that the design bases are maintained.

- The draft rule would require a means for monitoring the performance or condition of SSCs that can affect RISC categorization results and include provision for taking action to maintain the validity of the categorization. The draft rule also includes provisions for timely updates to the PRA and categorization process to reflect current plant configuration and operational data.

- For RISC-1 (safety related, safety significant) and RISC-2 (non-safety related, safety significant) SSCs, the existing regulatory requirements would continue and provisions would ensure that categorization assumptions and treatment are applied consistently.

- For RISC-3 (safety related, low safety significant) SSCs, special treatment requirements would be removed. Capability to perform safety functions would need to be maintained and processes controls would need to be implemented for design; procurement; installation; maintenance; inspection, test, and surveillance; corrective action; oversight; and configuration.

- For RISC-4 (non-safety related, low safety significant) SSCs, special treatment requirements would be removed and the management of SSCs would be treated as commercial.

- In the draft rule, the staff proposes to remove treatment or modify requirements for certain regulations including: 10 CFR Part 21(Reporting of Defects and Noncompliance); 10 CFR 50.44 (combustible gas control system); 10 CFR 50.49 (environmental qualification); 10 CFR 50.65 (maintenance rule) except (a)(4)(a)1 through (a)3; 10 CFR 50.72 and 50.73 (reporting requirements); 10 CFR Part 50, Appendix B (quality assurance) and Appendix J (Type B and C containment leakage); and 10 CFR Part 100 (reactor site criteria) except for seismic.

- Changes to the treatment requirements are not proposed for 10 CFR 50.55a (Codes and Standards) and 10 CFR Part 54 (license renewal).

- Industry pilot plants include Quad Cities, Wolf Creek, Surry, and Palo Verde.

Industry Presentation

Messrs. Adrian Heymer and Anthony Pietrangelo of the Nuclear Energy Institute (NEI) provided a brief overview of industry perspectives concerning the draft rule language, categorization and treatment of SSCs, and proposed industry guidance in NEI 00-04. Significant points raised during the presentation include:

- Option 2 principles would apply NRC special treatment requirements consistent with safety significance, maintain design bases unchanged, and replace NRC special treatment with licensee functional monitoring.

- A separate program is not needed for RISC-3 SSCs. Nuclear industrial balance-of-plant (BOP) controls should be supplemented with a simplified functional monitoring program (performance or condition) to provide confidence that the design bases functions are met.

- NEI agrees with the high-level requirements for categorization but has concerns regarding implementation via the license amendment process. NEI stated that a new treatment program needs to be developed to handle RISC-3 and RISC-4 in a similar manner.

- The risk-informed ASME Code Cases for 10 CFR 50.55a should apply to RISC-3 SSCs.

SUBCOMMITTEE COMMENTS, CONCERNS, AND RECOMMENDATIONS

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

- Dr. Apostolakis questioned whether any RISC-2 SSCs failed to function properly or adversely affected the progress of a transient. The staff stated that BWR feedwater could be considered to meet that criteria. Dr. Bonaca suggested that PWR power-operated relief valves (PORVs) might also fit the criteria and cited the accident at Three Mile Island Unit 2 as an example.

- Dr. Bonaca questioned whether a Final Safety Analysis Report (FSAR) could be based on frequency-consequence (F-C) curves. Dr. Kress stated that F-C curves only apply because you are using CDF and LERF as metrics. Mr. Rosen stated that a large number of plant components are not modeled and, thus, do not serve well in the measures of CDF and LERF. Mr. Rosen suggested that evaluations involving these types of components must be reconciled by the expert panel. Dr. Apostolakis suggested that NEI 00-04 (p. 58-60) be modified to better reflect the need for the expert panel to document the results of its IDP deliberations.

- Drs. Apostolakis, Shack, and Kress questioned the role of the industry peer review process, described in NEI 00-02, in ensuring the quality of PRAs. Dr. Apostolakis noted that there is a lot of burden on the reviewer to evaluate and certify licensee PRAs. Dr. Kress questioned the treatment of LERF for multiple-unit sites and suggested that this may be an issue for future reactors such as the pebble bed modular reactor. Dr. Shack questioned whether monitoring and performance could affect the categorization results. The staff stated that they have some concerns about the peer reviews and will be discussing these matters with NEI and industry representatives during future meetings.

- Dr. Apostolakis questioned whether NEI 00-04 would allow for a relaxation that affects a group of components across systems. In particular, he expressed concern that a collective relaxation may mask an increased likelihood of failure. The staff stated that it will be necessary to perform sensitivity studies across affected systems and that these evaluations should consider vulnerability for common-cause failures.

- Dr. Apostolakis questioned why NEI 00-04 categorization is less structured than the process implemented in support of the STP license exemption request. He also noted that NEI 00-04 does not ask a number of questions in the STP methodology. The staff noted that NEI 00-04 does not assign numeric values. NEI representatives stated that these questions are addressed in the Maintenance Rule.

- Dr. Apostolakis questioned the apparent preference in NEI 00-04 for the use of sensitivity studies and "point estimates." He stated that the use of point estimates would be a step backward and expressed the view that uncertainty analysis and "mean values" are more appropriate risk methods. Mr. Rosen noted that NEI 00-04 references a number of initiatives being pursued by industry organizations related to these matters.

STAFF AND INDUSTRY COMMITMENTS

None.

SUBCOMMITTEE DECISIONS

At the conclusion of the meeting, Dr. Apostolakis suggested that NEI 00-04 warrants further discussion and questioned whether the proposed rule could be issued without the implementing guidance being ready for use. The staff stated that it could be done. NEI representatives expressed support for going forward with the rule and completing the guidance later. The staff stated that they had a number of questions on the guidance in NEI 00-04 and suggested that a Subcommittee meeting be scheduled in the near future to possible staff positions on NEI 00-04. The Subcommittee agreed that a meeting specific to NEI 00-04 would be helpful.

FOLLOW-UP ACTIONS

None.

BACKGROUND MATERIALS PROVIDED TO THE SUBCOMMITTEE PRIOR TO THIS MEETING

1. Subcommittee agenda.
2. Subcommittee status report.
3. Memorandum dated November 19, 2001, from David B. Matthews, NRR, to John T. Larkins, Subject: ACRS Subcommittee and Full Committee Meetings on "RIP50" - Option 2, Risk-Informing the Scope of Special Treatment Requirements.
4. Memorandum dated August 2, 2001, from Annette-Vietti-Cook, Secretary, NRC, to William D. Travers, EDO, NRC, Subject: Staff Requirements - Briefing on Risk-informing Special Treatment Requirements.
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. Report dated July 23, 2001, from George E. Apostolakis, Chairman, ACRS, to Richard A. Meserve, Chairman, NRC, Subject: South Texas Project Nuclear Operating Company Requests for Exemption to Exclude Certain Components from the Scope of Special Treatment Requirements Required by Regulations (Option 2).

****

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.

REVISED 11/27/01

ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
MEETING OF THE SUBCOMMITTEE ON
RELIABILITY AND PROBABILISTIC RISK ASSESSMENT
ROOM T-2B3, 11545 ROCKVILLE PIKE, ROCKVILLE, MD
December 4, 2001

ACRS Contact: Michael T. Markley (301) 415-6885
E-mail: mtm@nrc.gov

- PROPOSED SCHEDULE -

	TOPIC	PRESENTER	TIME
1)	**Introduction**		1:00-1:05 pm
•	Review goals and objectives for this meeting; past ACRS deliberations on risk-informing the special treatment requirements of 10 CFR Part 50	George Apostolakis, ACRS	
2)	**NRC Staff Presentation**		1:05-3:00 pm
•	Overview	Cynthia Carpenter, NRR	
•	Items related to proposed rulemaking - Draft rule language for 10 CFR 50.69 - NEI 00-04 guidance for 10 CFR 50.69	Tim Reed, NRR Eileen McKenna, NRR Steve West, NRR David Diec, NRR	
	** BREAK **		3:10-3:30pm ~~3:00-3:15~~ pm
3)	**NRC Staff Presentation - continued**		3:15-4:00 pm
•	Items related to proposed rulemaking - Pilot trip reports (Quad Cities & Wolf Creek)	Tim Reed, NRR Eileen McKenna, NRR Steve West, NRR David Diec, NRR	
4)	**Industry Presentation**		4:00-4:30 pm
•	Guidance in NEI 00-04 and comments on 10 CFR 50.69, Appendix T, and pilots	Adrian Heymer, NEI	

5) General Discussion and Adjournment 4:30-5:00 pm

• General discussion and comments by Members of the Subcommittee; items for December 6-8, 2001 ACRS meeting	George Apostolakis, ACRS

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

ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
RELIABILITY AND PROBABILISTIC RISK ASSESSMENT
SUBCOMMITTEE MEETING

DECEMBER 4, 2001
Today's Date

ATTENDEES PLEASE SIGN IN FOR ACRS MEETING

PLEASE PRINT

NAME	AFFILIATION
MIKE KNAPIK	McGRAW-HILL
ADRIAN HEYMER	NEI.
Charles Brinkman	Westinghouse

ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
RELIABILITY AND PROBABILISTIC RISK ASSESSMENT
SUBCOMMITTEE MEETING

DECEMBER 4, 2001
Today's Date

NRC STAFF PLEASE SIGN IN FOR ACRS MEETING

PLEASE PRINT

NAME	NRC ORGANIZATION
John Fair	NRR/DE
TIM REED	NRR/DRIP
Eileen McKenna	NRR/DRIP
PAUL Shemanski	NRR/DE/EEIB
David Terao	NRR/DE/EMEB
Cindi Carpenter	NRR/DRIP/RGEB
Mike Cheok	
Goutam Bagchi	NRR/DE
Stephen Dinsmore	NRR/SPSB
Peter Balmain	NRR/DIPM
Steve West	NRR
Edmund Sullivan	NRR/DE
Tom Scarbrough	NRR/DE
DAVE FISCHER	NRR/DE/EMEB
Mark Rubin	NRR/DSSA/SPSB
Glenn Kelly	" " "

Markley
2

Preliminary Option 2 Rulemaking Proposals

ACRS Subcommittee Meeting
December 4, 2001

Tony Pietrangelo, NEI
Adrian Heymer, NEI

Initial Risk-Informed Applications

Safety-Related | **Nonsafety-Related**

Risk-Informed SSC C[illegible]rization Methodology

High Risk-Significant

Low Risk-Significant

2

Risk-Informed Categorization Option 2 -- SSC Scope

Safety-Related	Nonsafety-Related

Risk-Informed SSC C[illegible]rization Methodology

Safety Significant	RISC-1	RISC-2
Low Safety Significant	RISC-3	RISC-4

3

DRAFT

Option 2 Principles

- **Apply NRC special treatment requirements consistent with safety significance**
- **Design bases are not changed**
  - Monitoring program (performance or condition) plus treatment controls provide reasonable assurance that the (§50.2) safety-significant and design bases functions will be satisfied
- **For low safety-significant SSCs NRC special treatment requirements replaced by licensee controls plus a simplified functional monitoring program**
  - No need to develop separate treatment program for RISC-3

4

Main Issue
RISC-3 Treatment

- **NRC**
  - New program beyond BOP programs
- **Industry Position**
  - Nuclear Industrial (BOP) controls plus a simplified functional monitoring program (performance or condition) to provide adequate confidence that the design bases functions will be met
  - Elements to be listed in the rule
    - Guideline provides 4-page summary plus 22 pages of additional guidance with examples
    - Level of detail should not be the same as for Appendix B

5

NEI

Nuclear Industrial Treatment

- A set of high grade industrial practices that provide adequate confidence that the required functions will be satisfied under designed service conditions. Such practices are identified through applicable national, local and industry codes and standards, vendor recommendations, or operating experience. Implementation measures are applied commensurate with the relative importance and complexity of the activity, and the skill of the craft. These measures are accomplished through plant procedures, guidelines, guidelines, and work instructions. The scope of treatment includes: design control, procurement, inspection, testing, work processes, maintenance, assessment and corrective action.

6

NEI

Adequacy of Industrial Treatment for RISC-3 SSCs

Three principal bases:

- **No change to functional requirements**
- **Historical performance data**
- **Functional monitoring and corrective action**

7

NEI

Functional Requirements

- **50.69 does not change the design bases of any safety-related SSCs**
- **Engineering and procurement specifications and processes will preserve design bases requirements**
- **Alternative equipment designs can meet the specifications and thus preserve design bases functionality**

8

NEI

Historical Performance Data

- **Generic equipment performance data indicates robustness of industrial designs & equipment**
  - Industrial supplier test data
  - Reliability comparison of safety-related and nonsafety-related SSCs (STP report)
    - 33 component types investigated
    - No significant difference in reliability
- **90% industry average capacity factor**
- **Conclusion: Industrial treatment leads to comparable performance**

9

NEI

Monitoring/Corrective Action

- **No expected change in RISC-3 SSC performance**
- **Functional monitoring and corrective action assure SSC capability**
- **Aggregate impact sensitivity studies demonstrate adequate margin of safety**

10

NEI

Licensing Basis Regarding Treatment

- Rule should specify industrial treatment for RISC-3 SSCs including a list of attributes
- QA topical referenced in UFSAR should provide summary description of attributes (Use 50.54(a) to control)
- Licensee commitment to regulatory guide endorsing NEI 00-04 (Use NEI 99-04,CM guidance, to control)

Rationale for Licensing Basis

- Draft alternative proposal implies equivalent in level of detail to current Appendix B
  - Some elements more restrictive
- Low safety significance of RISC-3 SSCs does not warrant equivalent level of detail in rule as RISC-1 SSCs
- Other elements of licensing basis are consistent with current regulatory framework

RISC-3 Treatment Conclusions

- **Industrial controls provide adequate confidence that design bases will be maintained**
- **RISC-3 functional monitoring (performance/condition) assures equipment capability**
  - Maintenance rule reliability & availability monitoring not necessary
- **Special treatment requirements do not apply to RISC-3 & RISC-4**

NEI

NRC §50.69 Proposals Summary

Agree

- **High level requirements for categorization**
  - No Appendix T

Concerns - rationale

- **Implementation via a license amendment**
- **For RISC-1 and RISC-2 SSCs licensees shall ensure the assumptions in the categorization process are consistent with the treatment provisions**
  - Needs clarification

How PRA is currently affected

- **Develop new treatment program for RISC-3 & RISC-4 SSCs**

NEI

NRC §50.69 RISC-3 Proposals

- **NRC special treatment requirements applied to RISC-3 SSCs with the exception of**
  - Part 21
  - Appendix B, but requires development of new program
    - Corrective action more stringent than App. B
    - Maintain design inputs – more stringent than for RISC-1
  - §50.65, except (a)(4)(a) 1 thru (a) 3
  - Environmental qualification (needs clarification)
  - Reporting & Notification
  - Specific elements of Appendix J

15 NEI

NRC §50.69 RISC-3 Proposals

- **Use of specific ASME code cases for RISC-3 (§50.55a ~~continues to apply~~)**
- **No relief from seismic requirements**
  - Make consistent with §50.55a approach
  - Use alternative national consensus standards
- **No change in Part 54 scope**
- **Inconsistent approaches**

16 NEI

Conclusion

- **Need to re-establish an understanding of the guiding principles of risk-informed, performance-based regulation**
  - Reach an understanding on rule language
  - Reach an understanding on guidance
- **Reach a better understanding on the rulemaking proposals**
- **Finalize guidance incorporating pilot lessons learned**

17

NEI

Markley

3

Attachment 2

Draft Rule Language

DRAFT RULE LANGUAGE
as of November 19, 2001

The NRC staff has released the following draft rule language in response to guidance from the Commission dated August 2, 2001. The proposal would amend Title 10 of the *Code of Federal Regulations* (10 CFR) by adding Section 50.69, "Risk-Informed Treatment of Structures, Systems and Components." The proposal would permit power reactor licensees and applicants to implement an alternative regulatory framework with respect to treatment requirements currently imposed beyond practices for commercial grade equipment to add assurance of capability of structures, systems and components (SSCs) to perform their intended functions. Under this framework, licensees, using a risk-informed process for categorizing SSC according to their safety and risk significance, could remove SSCs of low safety significance from the scope of certain identified treatment requirements. The NRC has also provided additional information within the body of the draft rule language which is bracketed ("[ ]") to facilitate understanding of the NRC's intent on certain aspects of the proposed rule.

This draft rule language was released to inform stakeholders of the current status of the 10 CFR 50.69 risk-informed rulemaking and to provide stakeholders with an opportunity to comment on the draft revisions. The draft rule language is preliminary and may be incomplete in one or more respects.

§50.69 Risk-Informed Treatment of Structures, Systems and Components

§50.69(a) Definitions

RISC (risk-informed safety class)-1 functions are functions performed by safety-related SSCs that are safety-significant as determined by a categorization process that meets the requirements of paragraph (c) of this section.
RISC-2 functions are functions performed by nonsafety-related SSCs that are safety-significant as determined by a categorization process that meets the requirements of paragraph (c) of this section.
RISC-3 functions are functions performed by safety-related SSCs that are low safety-significant as determined by a categorization process that meets the requirements of paragraph (c) of this section.
RISC-4 functions are functions performed by nonsafety-related SSCs that are low safety-significant as determined by a categorization process that meets the requirements of paragraph (c) of this section.

For the purpose of this rule, SSCs performing RISC-1, -2, -3, and -4 functions are considered RISC-1, -2, -3, and -4 SSCs, respectively.

§50.69(b) Applicability. The requirements of this section are applicable to (1) holders of a license to operate a nuclear power plant under §50.21(b) or 50.22; (2) applicants for or holders of a combined license for a nuclear power reactor issued under part 52 of this chapter [applicability to and requirements for Part 52 certificates or combined licenses are still under staff review]; and (3) holders of renewed licenses under Part 54 of this chapter, who elect to adopt these requirements in lieu of other requirements (as specified below).

§50.69(c) Categorization Process Requirements. An applicant or licensee who elects to implement the alternative requirements of this section shall categorize SSC functions into one of the four RISC categories as defined in section 50.69(a) using a categorization process which has been approved by the NRC. The categorization process must:

(1) Use a plant-specific Probabilistic Risk Assessment (PRA) to determine the relative importance of modeled SSC functions in terms of core damage frequency and large early release frequency. This calculation must be performed with an evaluation model which includes internal initiating events at full power operations. External initiating events and low power and shutdown modes of operation must also be considered, either as part of this PRA or as part of the integrated decision-making process described in §50.69(c)(2). [The need to specify criteria on acceptability of the PRA is under staff review]

(2) Use an integrated decision-making process to determine the safety significance of functions performed by the SSCs. The categorization of these functions as either safety significant or low safety significant must include:

- (i) Results and insights from the PRA, including those from importance evaluations.
- (ii) Determination of SSC function importance using an acceptable process for addressing initiating events and plant operating modes not modeled in the PRA.
- (iii) Defense-in-depth.
- (iv) Maintenance of sufficient safety margins.
- (v) Sufficient supporting justification in terms of items (i) to (iv) above for SSC functions determined to be of low safety significance.

(2) Assure that the potential change in core damage frequency and large early release frequency is small including consideration of the change in risk resulting from categorizing SSCs and modification to special treatment.

(4) Include a means for monitoring the performance or condition of those SSCs that, when degraded, can affect the results of the categorization process and a means for taking actions as necessary such that the bases for an SSC's categorization continues to be satisfied.

(5) Include a provision for timely updates of the PRA and SSC categorization to assure that the actual design, construction, operational practices, and operational experience of the plant are realistically reflected in the bases for categorization.

§50.69(d) Requirements for Structures, systems, and components.

(1) SSCs that perform RISC-1 or RISC-2 functions are subject to the following:

- (i) Existing regulatory requirements continue to apply.

(ii) The licensee shall ensure that the assumptions in the categorization process and the treatment being applied to these SSCs are consistent.

(2) SSCs that perform RISC-3 functions are subject to the following:

(i) Existing regulatory requirements continue to apply except as allowed by §50.69(d)(3).

(ii) The licensee shall have processes to control the design; procurement; installation; maintenance; inspection, test, and surveillance; corrective action; oversight; and configuration, for RISC-3 SSCs. The pertinent requirements of the processes described below must be implemented to provide reasonable confidence in the capability of RISC-3 SSCs to perform their safety-related functions under design-basis conditions throughout their service life.

(A) Design Control Process.

Design control for RISC-3 SSCs must preserve functional requirements and bases; select suitable materials, methods, and standards; verify design adequacy; and control design changes to support the determination that RISC-3 SSCs remain capable of performing safety-related functions under design-basis conditions throughout their service life. As part of design control, design inputs related to the performance of design-basis functions of RISC-3 SSCs throughout their service life must be maintained and applied.

(B) Procurement Process.

Suitable methods must be used to support a documented determination that procured SSCs will be capable of performing their safety-related functions under design-basis conditions, including appropriate environmental conditions and combinations of normal and accident conditions with earthquake motions. Design inputs related to the performance of design-basis functions must be satisfied to support the determination that the procured RISC-3 SSCs remain capable of performing safety-related functions under design-basis conditions throughout their service life.

(C) Installation Process.

SSCs must be properly installed and tested to support the determination that RISC-3 SSCs are capable of performing their safety-related functions under design-basis conditions throughout their service life.

(D) Maintenance Process.

The scope, frequency, and detail of predictive, preventive, and corrective maintenance activities (including post-maintenance testing) must be established to support the determination that RISC-3 SSCs will remain capable of performing their safety-related functions under design-basis conditions throughout their service life.

(E) Inspection, Test, and Surveillance Process.

Data or information must be obtained to support the determination that these SSCs will remain capable of performing safety-related functions under design-basis conditions throughout their service life. The data or information for pumps, valves, and snubbers must allow evaluation of operating characteristics of these RISC-3 SSCs.

(F) Corrective Action Process.

Conditions that could prevent RISC-3 SSCs from performing their safety-related functions under design-basis conditions must be identified, documented, and corrected in a timely manner to preclude repetition.

(G) Oversight Process.

The implementation of the treatment processes for RISC-3 SSCs, and the assessment of the effectiveness of those processes, must be controlled and accomplished through documented procedures and guidelines (including the qualification, training, and certification of personnel) to support the determination that SSCs are capable of performing safety-related functions under design basis conditions throughout their service life.

(H) Configuration Control Process.

The configuration of RISC-3 SSCs and applicable plant documents must be controlled to reflect current plant status and design changes.

(3) SSCs that perform RISC-3 or RISC-4 functions are not subject to the following:

(i) 10 CFR Part 21

(ii) The requirements that high point vents must conform to Appendix B in §50.44c(3)(iii), the requirements to justify the hydrogen control system with a suitable program of experiment and analysis in §50.44c(3)(iv)(A); §50.44c(3)(iv)(B); §50.44c(3)(iv)(C); §50.44c(3)(iv)(D)(1); §50.44c(3)(iv)(D)(2); §50.44c(3)(iv)(D)(3); the requirements to qualify for the environment caused by inerting, systems and components required to establish and maintain safe shutdown and containment integrity in §50.44c(3)(iv)(E). [The NRC staff is working on a proposed revision to 10 CFR 50.44; this revision, if approved, would likely impact the specific citations noted above. As these rulemakings progress, appropriate changes to this item will be made]

(iii) The environmental qualification requirements except that the equipment must continue to satisfy the environmental conditions under which these SSC must perform as listed in 10 CFR 50.49(e)(1) through (7).

[Note that the staff intends to risk inform the special treatment requirements of 50.55a through the use of code cases.]

(iv) §50.55(e)

(v) § 50.65, except for paragraph (a)(4).

(vi) §50.72

(vii) §50.73

(viii) Appendix B to 10 CFR Part 50

(ix) The Type B and Type C leakage testing requirements in both Options A and B of Appendix J to 10 CFR Part 50, for SSCs meeting the following criteria:

(A) For containment penetrations that meet one or more of the following criteria:

(*1*) The penetration is 1-inch nominal size or less

(*2*) The penetration is continuously pressurized

(B) For containment isolation valves that meet one or more of the following criteria:

(*1*) The valve is required to be open under accident conditions to prevent or mitigate core damage events;

(*2*) The valve is normally closed and in a physically closed, water-filled system;

(*3*) The valve is in a physically closed system whose piping pressure rating exceeds the containment design pressure rating and that is not connected to the reactor coolant pressure boundary;

(*4*) The valve is in a closed system whose piping pressure rating exceeds the containment design pressure rating and is connected to the reactor coolant pressure boundary; and

(*5*) The valve is 1-inch nominal size or less.

[This paragraph does not include Appendix A to Part 100, Sections VI(a)(1) and VI(a)(2) requirements on qualification testing, because Part 100 states qualification testing or suitable dynamic analysis is required - thus, the staff's current view is that a rule change is not needed to eliminate the special treatment requirement to perform qualification tests for RISC-3 SSCs].

§50.69(e) Submittal and Approval Process.

(1) A licensee who wishes to implement section 50.69 shall submit a license amendment request pursuant to section 50.90 that contains the information in section (2) below

[The applicability to and requirements for Part 52 certificates or combined licenses are still under staff review]

(2)The submittal must contain the following information:

(i) A list of the regulations identified in §50.69 (d)(3) for which the requirements of §50.69 are being substituted.

(ii) A description of the categorization process and decision criteria used that meets the requirements of §50.69(c).

(iii) Description of the measures taken to assure that the quality of the PRA used in the categorization process is commensurate with the application.

(iv) A description of the scope of SSCs to which the requirements of §50.69 will be applied.

(v) A schedule for implementation of §50.69.

§50.69 (f) Program Description, Documentation, and Reporting.

(1) Licensees adopting the requirements of this section shall include in their FSAR in accordance with the provisions of §50.71(e), a summary description of processes and activities applied to SSCs that are the means of implementing the requirements of §50.69. Licensees shall update their FSAR to reflect status of implementation of section 50.69 at the system level.

(2)The licensee shall document, and maintain for the duration that an SSC is installed, the basis for categorization and treatment of SSCs made pursuant to the requirements of this section.

(3) The licensee shall submit a licensee event report to the NRC for any event or condition that could have prevented the satisfaction of a RISC-1 or RISC-2 safety significant function. The report shall be submitted consistent with the requirements of §50.73(b). [The staff is considering whether this requirement should be placed in §50.73 (but be applicable only to those who use 50.69) or be in this section]

(4) The licensee shall retain records required by this section until the license is terminated.

§50.69 (g) Change Control.

(1) When a licensee first implements section 50.69 for a structure or system, changes to the final safety analysis report for the implementation need not include a supporting §50.59 evaluation.

(2) Changes to the categorization process requirements contained in the submittal required by section 50.69(e) as approved by the NRC, may be made without prior NRC approval, unless the change would decrease the effectiveness of the process in identifying safety-significant SSCs.

(3) Changes to the procedures and processes for implementing §50.69(d), may be made if the requirements of this section continue to be met. The licensee (or applicant) shall prepare a written basis for this determination.

[7590-01-P]

NUCLEAR REGULATORY COMMISSION

10 CFR Part 50

Risk-Informed Treatment of Structures, Systems and Components

AGENCY: U.S. Nuclear Regulatory Commission.

ACTION: Availability of draft rule wording.

SUMMARY: The Nuclear Regulatory Commission (NRC) is making available the draft wording of a possible amendment of its regulations. The proposal would add 10 CFR 50.69, "Risk-Informed treatment of Structures, systems and components." The proposal would permit power reactor licensees and applicants to implement an alternative regulatory framework with respect to certain treatment requirements currently imposed beyond practices for commercial grade equipment to add assurance of capability of structures, systems and components (SSCs) to perform their intended functions. Under this framework, licensees, using a risk-informed process for categorizing SSC according to their safety and risk significance, could remove SSCs of low safety significance from the scope of certain identified treatment requirements. The availability of the draft wording is intended to inform stakeholders of the current status of the NRC's activities to adopt 10 CFR 50.69 and to provide stakeholders the opportunity to comment on the draft changes. The NRC has also provided additional ("[ ]") information within the body of the draft rule language which is bracketed ("[ ]") to facilitate understanding of the NRC's intent on certain aspects of the proposed rule.

DATES: Comments should be submitted within 30 days from the date of this notice. Any comments received after this date may not be considered during drafting of the proposed rule. Because of scheduling considerations in preparing a proposed rule, the NRC requests that stakeholders provide their comments at their earliest convenience before the end of the comment period, if practicable.

ADDRESSES: Submit written comments to: Secretary, U.S. Nuclear Regulatory Commission, Washington, DC 20555-0001, Attention: Rulemakings and Adjudications Staff, Mail Stop O-16C1 or deliver written comments to One White Flint North, 11555 Rockville Pike, Rockville, Maryland, between 7:30 a.m. and 4:15 p.m. on Federal workdays.

You may also provide comments via the NRC's interactive rulemaking Web site through the NRC's home page at http://www.ruleforum.llnl.gov. This site provides the capability to upload comments as files (any format), if your web browser supports that function. For information about the interactive rulemaking Web site, contact Ms. Carol Gallagher at (301) 415-5905 or by e-mail to cag@nrc.gov. Copies of any comments received and certain documents related to this rulemaking may be examined at the NRC Public Document Room, located at One White Flint North, 11555 Rockville Pike (first floor), Rockville, Maryland. The NRC maintains an Agencywide Documents Access and Management System (ADAMS), which provides text and image files of NRC's public documents. These documents may be accessed through the NRC's Public Electronic Reading Room on the Internet at http://www.nrc.gov/NRC/ADAMS/index.html. If you do not have access to ADAMS or if there are problems in accessing the documents located in ADAMS, contact the NRC Public Document Room (PDR) Reference staff at 1-800-397-4209, 301-415-4737 or by email to pdr@nrc.gov.

FOR FURTHER INFORMATION CONTACT: Eileen M. McKenna, Risk-Informed Initiatives, Environmental, Decommissioning, and Rulemaking Branch, Division of Regulatory Improvement Programs, Office of Nuclear Reactor Regulation, U.S. Nuclear Regulatory Commission, Washington, DC 20555-0001; Telephone: (301) 415-2189; Internet: emm@nrc.gov.

SUPPLEMENTARY INFORMATION: Since the Commission published a Policy Statement on the Use of Probabilistic Risk Assessment in 1995, the NRC's efforts to consider risk insights in the regulatory infrastructure have evolved over the years. In SECY-98-300, dated December 23, 1998, under Option 2, the NRC staff proposed to add provisions to Part 50 for risk-informed alternative regulations, revise existing requirements to reflect risk-informed considerations, and to remove unnecessary or ineffective regulations. In SECY-99-256, dated October 29,1999, the staff provided a rulemaking plan and an Advance Notice of Proposed Rulemaking (ANPR) for risk-informed changes using 10 CFR 50.69. In a Staff Requirements Memorandum dated January 31, 2000, the Commission directed the staff to proceed with the rulemaking and to publish the ANPR (65FR 11488, March 3, 2000). In SECY-00-0194, dated September 7, 2000, the NRC staff subsequently communicated to the Commission its preliminary analysis of public comments on the ANPR and discussed issues involving 10 CFR 50.69.

The NRC has now developed draft wording for the changes to its regulations and has made them available on the NRC's rulemaking Web site at http://ruleforum.llnl.gov. This draft rule language is preliminary and may be incomplete in one or more respects. This draft rule language was released to inform stakeholders of the current status of the 10 CFR 50.69 rulemaking and to provide stakeholders with an opportunity to comment on the draft revisions. Comments received prior to publishing the proposed rule will be considered in the development of the proposed rule. Comments may be provided through the rulemaking Web site at http://ruleforum.llnl.gov/ or by mail as indicated under the ADDRESSES heading. The NRC may post updates periodically on the rulemaking Web site that may be of interest to stakeholders.

Dated at Rockville, Maryland, this th day of November 2001.

FOR THE NUCLEAR REGULATORY COMMISSION

/R/ Signed by C. Carpenter

Cynthia A. Carpenter, Chief
Risk-Informed Initiatives, Environmental,
Decommissioning, and Rulemaking Branch
Division of Regulatory Improvement Programs
Office of Nuclear Reactor Regulation

INTRODUCTORY STATEMENT BY THE CHAIRMAN OF THE
MEETING OF THE ACRS SUBCOMMITTEE
ON RELIABILITY AND PROBABILISTIC RISK ASSESSMENT
11545 ROCKVILLE PIKE, ROOM T-2B3
ROCKVILLE, MARYLAND
DECEMBER 4, 2001

The meeting will now come to order. This is a meeting of the Advisory Committee on Reactor Safeguards (ACRS) Subcommittee on Reliability and Probabilistic Risk Assessment. I am George Apostolakis, Chairman of the Subcommittee.

Subcommittee Members in attendance are Mario Bonaca, Peter Ford, Thomas Kress, Steve Rosen, and William Shack.

The purpose of this meeting is to discuss proposed revisions to the special treatment requirements of 10 CFR Part 50 (Option 2). The Subcommittees will gather information, analyze relevant issues and facts, and formulate proposed positions and actions, as appropriate, for deliberation by the full Committee. Michael T. Markley is the Cognizant ACRS Staff Engineer for this meeting.

The rules for participation in today's meeting have been announced as part of the notice of this meeting previously published in the *Federal Register* on November 21, 2001.

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

We have received no written comments or requests for time to make oral statements from members of the public regarding today's meeting.

(Chairman's Comments-if any)

- ACRS last issued a report concerning proposed 10 CFR 50.69 and Appendix T dated October 12, 1999. The staff is no longer pursing Appendix T and is considering guidance in NEI 00-04, "Option 2 Implementation Guideline."

- ACRS reviewed the license amendment requests from South Texas Project concerning special treatment and issued a report dated July 23, 2001.

- Today, the Subcommittee will also consider pilot activities at Quad Cities and Wolf Creek nuclear power plants.

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

Markley
1

United States
Nuclear Regulatory Commission

RISK-INFORMED PART 50 SPECIAL TREATMENT REQUIREMENTS RIP50 OPTION 2

ACRS SUBCOMMITTEE DECEMBER 4, 2001

Timothy Reed
Division of Regulatory Improvement Programs
US Nuclear Regulatory Commission

BRIEFING OBJECTIVE

- Provide ACRS a status of ongoing Option 2 tasks
- Discuss draft rule language
- Get ACRS feedback on the current Option 2 direction

BACKGROUND

- SECY-99-256 (10/99) provided rulemaking plan and ANPR

- SECY-00-194 (9/00)
  - -- Provided preliminary views on ANPR comments
  - -- Provided further thoughts on regulatory approach

- South Texas exemption (8/01)
  - -- Proof of concept for Option 2

- Stakeholder interactions
  - -- Public workshops (4/00, 2/01, 11/01)
  - -- Commission briefings (9/00, STP/Option 2 brief --7/01)
  - -- Draft rule language(11/01)

United States
Nuclear Regulatory Commission

Risk-Informed

1 **"RISC-1" SSCs** **Safety-Related** **Safety Significant** *Current requirements continue* *Ensure categorization assumptions* *50.69 (d)(1) reqts*	2 **"RISC-2" SSCs** **Nonsafety-Related** **Safety Significant** *Current requirements continue* *Ensure categorization assumptions* *50.69(d)(1) reqts*
3 **"RISC-3" SSCs** **Safety-Related** **Low Safety Significant** *Maintain Design Basis Functions* *50.69(d)(2) reqts*	4 **"RISC-4" SSCs** **Nonsafety-Related** **Low Safety Significant**

Deterministic — GA — Traditional?

DRAFT RULE REQUIREMENTS
DEFINITIONS: 50.69 (a)
APPLICABILITY: 50.69 (b)

- **§50.69(a) Definitions**

  -- RISC-1 SSCs/functions are safety-related + safety-significant
  -- RISC-2 SSCs/functions are nonsafety-related + safety-significant
  -- RISC-3 SSCs/functions are safety-related + low safety-significant
  -- RISC-4 SSCs/functions are nonsafety-related + low safety-significant

- **§50.69(b) Applicability:**

  -- Current licensees
  -- Applicants for, or holders of, a Part 52/combined license [still under staff review]
  -- Holders of Part 54 renewed licenses

DRAFT RULE REQUIREMENTS
CATEGORIZATION REQUIREMENTS: 50.69 (c)

- **§50.69(c) Categorization Process Requirements:**
  - -- Categorize SSC functions into RISC categories using an NRC approved categorization process (App T not in draft rule)
  - -- Use a plant-specific PRA to determine the relative importance of modeled SSC functions in terms of CDF and LERF
  - -- Evaluation model must include:
    - o Internal initiating events at full power operations
    - o External initiating events, low power and shutdown modes of operation must also be considered, either as part of the PRA or as part of the integrated decision-making process (IDP) WJS

DRAFT RULE REQUIREMENTS
CATEGORIZATION REQUIREMENTS: 50.69 (c) CONT'

-- Use an IDP to determine the safety significance of functions

-- IDP must consider:
- PRA Results and insights (including importance evaluations)
- SSC function importance using an acceptable process for addressing initiating events and plant operating modes not modeled in the PRA
- Defense-in-depth
- Safety margins

-- IDP must justify SSCs as low safety significant in terms of above items

-- Assure that the potential increase in CDF and LERF is small

DRAFT RULE REQUIREMENTS CATEGORIZATION REQUIREMENTS: 50.69 (c) CONT'

-- Include a means for monitoring the performance or condition of those SSCs that can affect the categorization results

-- Include a means for taking actions to maintain the validity of an SSC's categorization

-- Include a provision for timely updates of the PRA and categorization process to reflect the current plant configuration and operational data

DRAFT RULE REQUIREMENTS
TREATMENT REQUIREMENTS: 50.69 (d)

- **§50.69(d) Requirements for structures, systems, and components:**

- **RISC-1 and RISC-2 SSCs :**

  -- Existing regulatory requirements continue

  -- Ensure that the categorization assumptions and the treatment applied to the associated SSCs are consistent

United States
Nuclear Regulatory Commission

DRAFT RULE REQUIREMENTS
TREATMENT REQUIREMENTS: 50.69 (d) CONT'

- **RISC-3 SSCs:**

  -- Special treatment requirements removed per §50.69(d)(3)

  -- Must have processes to control the design; procurement; installation; maintenance; inspection, test, and surveillance; corrective action; oversight; and configuration, for RISC-3 SSCs

  -- Draft rule contains programmatic, high level requirements for RISC-3

    - Must apply pertinent programmatic requirements to provide reasonable confidence in the capability of RISC-3 SSCs to perform their safety-related functions under design-basis conditions

- **RISC-4 SSCs:**

  -- Special treatment requirements removed per 50.69 (d)(3)

DRAFT RULE REQUIREMENTS
REQUIREMENTS REMOVED FROM RISC-3 AND RISC-4

- 10 CFR Part 21 Reporting Requirements
- Portions of 10 CFR 50.44 [10 CFR 50.44 Option 3 rulemaking would likely impact the specific citations]
- §50.49 -- EQ requirements
  -- Must continue to satisfy conditions listed in 50.49(e)(1) -- (7)
- §50.55a -- ASME Code Requirements is NOT on the list
  -- Staff and industry developing risk-informed code cases
- §50.55(e)
- §50.65 Maintenance rule -- except for paragraph (a)(4)
- §50.72 and 50.73 -- Reporting requirements

DRAFT RULE REQUIREMENTS
REQUIREMENTS REMOVED FROM RISC-3 AND RISC-4 CONT'

- 10 CFR 50 Appendix B Quality Assurance requirements
- 10 CFR 50 Appendix J -- Type B and Type C containment leakage testing requirements (Options A and B) that meet the criteria specified
- 10 CFR Part 100 seismic requirements are NOT on the list
  -- Part 100 requires suitable qualification testing, dynamic analysis, or equivalent static load method. Thus the rule provides flexibility for qualification of RISC-3 SSCs
- Part 54 License Renewal is NOT on the list

DRAFT RULE OTHER REQUIREMENTS

- **50.69 (e) Submittal and Approval Process**
  - -- License amendment (50.90)
  - -- Description of categorization process
  - -- Description of measures to achieve PRA quality
  - -- Information on scope and schedule

- **50.69 (f) Program Description, Documentation and Reporting**
  - -- Summary description of 50.69 processes in UFSAR
  - -- Document/maintain bases for SSC categorization and treatment
  - -- Report events that prevent safety significant functions

50.69(g) Change Control requirements

- -- 50.59 evaluations to support UFSAR changes not required for initial implementation
- -- Changes to categorization process that reduce effectiveness must have prior approval
- -- Changes to treatment procedures need a written basis

RIP50 OPTION 2 PILOT ACTIVITY

- **Objectives of pilot activity:**
  -- Acquire information to enable the development of 50.69, SOC, regulatory analysis, Option 2 Regulatory Guide
  -- Acquire cost/benefit information

- **Pilot activity is testing the draft NEI guidance (NEI 00-04) to:**
  -- Identify weaknesses in guidance
  -- Determine if the guidance can be implemented in predictable and repeatable manner at different facilities

- **BWR, Westinghouse, and CE Owners groups supporting pilot**
  -- Quad Cities (IDP in 8/01)
  -- Wolf Creek (IDP in 10/01) and Surry
  -- Palo Verde

RIP50 OPTION 2 PILOT ACTIVITY FEEDBACK

- **Observations to date:**
  -- IDP panels are knowledgeable -- excellent interaction
  -- IDP needs to consider treatment at a high level
  -- IDP decisions need to be well documented
  -- Safety margin guidance is needed
  -- Maintenance rule categorization -- excellent source of information
  -- System engineer support is very valuable
  -- No discussion of long term containment integrity
  -- Categorization of pressure boundary is different from the approach used for active components
  -- NEI 00-04 defense-in-depth guidance needs clarification

RIP50 OPTION 2 IMPLEMENTATION GUIDANCE

- **In support of Option 2 NEI has developed NEI 00-04**
  -- Provided initial draft version "A2" in January 2001
  -- Provided revised draft version "B" in June 2001

- **Two rounds of comments provided to NEI to date**
  -- Developing further comments

- **Open issues/comments:**
  -- Categorization
    - Most significant issue is consideration of long term containment integrity

  -- Treatment:
    - NEI 00-04 needs to be aligned to the draft rule
    - Need to agree on implementation of RISC-3 program
    - Need to incorporate pilot feedback

NEXT STEPS

- **Continue with ongoing tasks:**
  -- Reviewing draft NEI 00-04
  -- Reviewing NEI 00-02 (industry PRA peer guidance)
  -- Developing guidance for the review of a RIP50 Option 2 submittal
  -- Continuing to observe pilots

- **Issue revisions to draft rule language**

- **Regulatory analysis**

- **Develop proposed rule package**
  -- Developing detailed SOC outline
  -- Prepare/assemble package
  -- Include consideration of draft rule comments

NEXT STEPS CONT'

Issue proposed rule package for comment after:

-- RILP/ET review

-- ACRS and CRGR review

-- Office

-- Commission review