November 21, 2001

MEMORANDUM TO: Michael T. Lesar, Chief
Rules and Directives Branch
Division of Administrative Services
Office of Administration

FROM: Cynthia A. Carpenter, Chief**/RA/**
Risk Informed Initiatives, Environmental, Decommissioning,
and Rulemaking Branch
Division of Regulatory Improvement Programs
Office of Nuclear Reactor Regulation

SUBJECT: IMPLEMENTATION OF COMMISSION ACTION: DRAFT RULE WORDING: NEW SECTION OF 10 CFR PART 50, SECTION 50.69 RISK-INFORMED TREATMENT OF STRUCTURES, SYSTEMS AND COMPONENTS

By memorandum dated August 2, 2001, the Secretary of the Commission indicated that the Commission has directed the NRC staff to engage stakeholders early in the process of several rulemakings including 10 CFR 50.69. The Commission also stated that the staff may share draft language with all stakeholders in advance of the proposed rule.

Please implement the Commission's direction by arranging for publication of the attached *Federal Register* notice announcing the availability of the draft rule language and posting the draft rule language, provided in Attachment 2, on the NRC Web site at http://ruleforum.llnl.gov/ with a link entitled "Proposed Rulemaking - Risk-Informed Treatment of Structures, Systems and Components (10 CFR 50.69)."

Attachments:
1. *Federal Register* Notice
2. Draft rule language

CONTACT:
Eileen M. McKenna
NRR/DRIP/RGEB
301-415-2189

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Attachment 1

Federal Register Notice

[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

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.