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August 11, 2003

Chief, Rules and Directives Branch
Division of Administrative Services
Office of Administration, Mail Stop: T-6 D 59
U. S. Nuclear Regulatory Commission
Washington, DC 20555-0001

Subject: Comments on Proposed Rule 10 CFR 50, Risk-Informed Categorization and Treatment of Structures, Systems and Components for Nuclear Power Reactors, 68FR26511, Dated May 16, 2003

Duke Energy Corporation offers the attached comments relative to the above Federal Register notice of May 16, 2003.

Please address any questions to L. B. Jones at 704 382-4753.

Very truly yours,

W. R. McCollum, Jr.
Senior Vice President
Nuclear Support

Attachment

Template=SECY-067

SECY-02

Attachment

Proposed Rule on *"Risk-Informed Categorization and Treatment of Structures, Systems and Components for Nuclear Power Reactors (68FR26511 (May 16, 2003)) (RIN 3150-AG42)"*

Section V.5.2.1 of the Proposed Rule includes the following text ...

> The proposed rule would permit licensees to select a technically defensible method to show that RISC-3 SSCs will remain functional when subject to design earthquake loads. The level of confidence for the design basis capability of RISC-3 SSCs, including seismic capability, may be less than the confidence in the design basis capability of RISC-1 SSCs. *The use of earthquake experience data has been mentioned as a potential method to demonstrate SSCs will remain functional during earthquakes. However, it would be difficult to rely on earthquake experience alone to demonstrate functionality of SSCs if the design basis includes multiple earthquake events or combinations of loadings unless these specific conditions were enveloped by the experience data. Additionally, if the SSC is required to function during or after the earthquake, the experience data would need to contain explicit information that the SSC actually functioned during or after the design basis earthquake events as required by the SSC design basis. The successful performance of an SSC after the earthquake event does not demonstrate it would have functioned during the event. Qualification testing of an SSC would be necessary if no suitable alternative method is available for showing that the SSC will perform its design basis function during an earthquake.*[1]

The text shown in italics goes beyond the high-level requirements delineated in the proposed rule and must be removed from the Statement of Considerations ("SOC") for the final rule in order that it is not inappropriately construed as NRC guidance for implementing the rule. Furthermore, the prescriptive nature of this language is inconsistent with the discussion of other design basis functional capabilities contained in the Proposed Rule.

In addition, the language is inconsistent with the NRC's position regarding the use of an experience-based method for assuring the seismic capability and adequacy for RISC-1 SSCs in the majority of licensed nuclear plants, as described in Supplement 1 to Generic Letter 87-02[2]. In that document, the NRC recognized that the SQUG Generic Implementation Procedure (GIP) provides an acceptable methodology for (1) "ensuring that the purpose of the NRC regulations related to seismic design can be satisfied for those plants" (emphasis in original), and (2) evaluating the seismic adequacy of new, replacement, and modified equipment in A-46 plants. The NRC stated the following:

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[1] Emphasis added. 68FR26543.

[2] NRC to All Unresolved Safety Issue (USI) A-46 Plant Licensees Who Are Members of the Seismic Qualification Utility Group (SQUG), "Supplement 1 to Generic Letter (GL) 87-02 That Transmits Supplemental Safety Evaluation Report No. 2 (SSER No. 2) on SQUG Generic Implementation Procedure, Revision 2, as Corrected on February 14, 1992 (GIP)" (May 22, 1992).

> Therefore, the resolution as described in GL 87-02 and NUREG-1211, Regulatory Analysis for Resolution of Unresolved Safety Issue A-46, 'Seismic Qualification of Equipment in Operating Plants', was that the criteria and procedures described herein are determined to be an acceptable evaluation method for verifying the seismic adequacy of the equipment in USI A-46 plants including future modifications and replacement equipment in these plants.
>
> The backfit analysis described in NUREG-1211 did not specifically address the new equipment. However, the staff agrees that it is impractical and inconsistent with the USI A-46 philosophy to require that new equipment shall meet current seismic qualification requirements, whereas the seismic adequacy of all other safe shutdown equipment (which will presumably encompass the large majority of all safe shutdown equipment in the plant) is verified through the USI A-46 procedures. Therefore, the criteria and procedures described herein are determined to be an acceptable evaluation method for verifying the seismic adequacy of new equipment in USI A-46 plants.

More recently, the NRC accepted a pilot application of the SQUG earthquake experience-based method for seismic evaluation of new Safety Related equipment at Duke Power's Oconee Nuclear Station noting, "... the staff finds that Duke has reasonably addressed the appropriate design attributes and has provided sufficient information to indicate the seismic adequacy of the affected equipment."[3]

Retaining the language in the final rule SOC would effectively preclude the A-46 plants (which represent the majority of currently operating plants) from applying an alternative treatment for RISC-3 SSCs required to function in the event of an earthquake. This result is because the language suggests that the current treatment for the safety-related equipment at these plants would not be acceptable for even the low-risk significant SSCs if an A-46 plant elected to implement the proposed rule. Therefore, including the language in the SOC creates a dilemma for the majority of the operating plants as to whether they could implement <u>any</u> aspects of the proposed rule if they do not intend to implement its provisions regarding seismic alternative treatment. The suggestion of such a limitation undermines the purpose of the rule.

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[3] Safety Evaluation for License Amendment No. 230 License Condition No. 1 – Oconee Nuclear Station Unit 2 (TAC No. MA3639), (January 5, 1999).

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