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SOUTHERN COMPANY

April 29, 2011

Docket Nos.: 50-321 50-348 50-424
50-366 50-364 50-425

NL-11-0840

U. S. Nuclear Regulatory Commission
ATTN: Document Control Desk
Washington, D. C. 20555-0001

Joseph M. Farley Nuclear Plant
Edwin I. Hatch Nuclear Plant
Vogtle Electric Generating Plant
Comments on NRC Draft Inspection Procedure 37060 titled
"10 CFR 50.69 Risk-Informed Categorization and Treatment
of Structures, Systems, and Components Inspection"

Ladies and Gentlemen:

On the Nuclear Regulatory Commission (NRC) website Documents for Comment page, the NRC has requested comments on NRC Draft Inspection Procedure 37060 titled "10 CFR 50.69 Risk-Informed Categorization and Treatment of Structures, Systems, and Components Inspection."

This letter is to advise that Southern Nuclear Operating Company (SNC) endorses the comments submitted by NEI. SNC comments are provided in Enclosure 1.

By letter dated December 6, 2010, SNC informed the NRC of SNC's intent to submit a license amendment request for implementation of 10 CFR 50.69 for the Vogtle Electric Generating Plant (VEGP) and requested that NRC assign pilot plant status to VEGP for implementation of 10 CFR 50.69. SNC is pursuing risk-informed initiatives, such as implementation of 10 CFR 50.69, because SNC believes risk-informed initiatives can improve plant safety and allow enhanced allocation of company resources on the most important plant equipment.

The enclosed SNC comments focus on ensuring the subject inspection procedure provides adequate guidance on acceptable alternative treatment strategies for risk-informed safety class (RISC)-3 structures, systems, and components (SSCs). The intent of Regulatory Guide 1.201 and 10 CFR 50.69 is for alternative treatment strategies to be performance-based rather than program-based. The benefits of the 10 CFR 50.69 application to licensees will be reduced or eliminated if licensees are required to establish new or modified programs to ensure "reasonable confidence". Clear inspection guidance is essential to ensuring a stable regulatory environment.

SNC appreciates the opportunity to comment on the subject inspection procedure and welcomes the opportunity to be a pilot for this important initiative. During the pilot process, it will be possible for the NRC to review SNC's alternative treatment strategies and refine the subject inspection procedure.

This letter contains no NRC commitments. If you have any questions, please contact Jack Stringfellow at (205) 992-7037.

Respectfully submitted,

M. J. Ajluni
Nuclear Licensing Director

MJA/CLT/lac

Enclosures: 1. SNC Comments on Draft Inspection Procedure 37060

cc: Southern Nuclear Operating Company
Mr. J. T. Gasser, Executive Vice President
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Mr. T. E. Tynan, Vice President – Vogtle
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RType:<Farley=CFA04.054; Hatch=CHA02.004; Vogtle=CVC7000

U. S. Nuclear Regulatory Commission
Mr. V.M. McCree, Regional Administrator
Mr. R. E. Martin, NRR Project Manager – Farley, Hatch and Vogtle
Mr. P. G. Boyle, NRR Project Manager
Mr. E. L. Crowe, Senior Resident Inspector – Farley
Mr. E. D. Morris, Senior Resident Inspector – Hatch
Mr. M. Cain, Senior Resident Inspector – Vogtle
James Isom, Reactor Inspection Branch

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Enclosure 1

SNC Comments on Draft Inspection Procedure 37060

1.0 General Comments

1. The inspection procedure should provide sufficient detail to minimize the need for NRC inspectors to contact NRR personnel. This practice could result in too many unresolved issues (URI).

2. The Electric Power Research Institute (EPRI) has developed guidance documents on RISC-3 alternative treatments, including specific guidance for seismic and environmental qualification. If the preparer of the inspection procedure found the documents acceptable, the inspection procedure could reference EPRI documents or sections of the documents. This would provide necessary clarification for the NRC inspector in the field. The three reports below are publically available; the fourth can be made available to the NRC staff during the pilot project reviews at Vogtle.
   a. EPRI Report No.1011234, "Program on Technology Innovation: 10CFR50.69 Implementation Guidance for Treatment of Structures, Systems, and 'Components."
   b. EPRI Report No. 1009748, "Guidance for Accident Function Assessment for RISC-3 Applications" (Alternative Treatment to Environmental Qualification for RISC-3 Applications).
   c. EPRI Report No. 1009669, "RISC-3 Seismic Assessment Guidelines"
   d. EPRI Report No. 1015099, "Option 2, 10CFR50.69 Special Treatment Guidelines."

3. The inspection procedure could reference industry guidance document AP-913, "Equipment Reliability," as an acceptable industry program for preventative maintenance (PM). Component criticality for AP-913 is not determined based on safety-related or non-safety related but on functional importance and the ability to perform maintenance. Criteria are structured and fairly consistent with the domestic nuclear fleet. For example, an acceptable alternative treatment for an RISC-3 motor operated valve may be dropped from high critical to low critical or non-critical. In this lower AP-913 category, the component will continue to be maintained, but the PM may be performed less often or the scope may be reduced. However, the PM performed will continue to ensure the component's intended functions will be provided. Likewise, an RISC-2 component (non-safety, but important) may have its AP-913 criticality increased to be considered a critical component for PMs.

4. The inspection procedure focuses on RISC-3 components and what constitutes acceptable treatments. However, NRC inspectors should also review selected RISC-2 structures, systems, and components (SSCs). These non-safety components have now been found to be risk significant and alternative treatments may require increased maintenance or testing.

5. There is a significant amount of "soft" guidance rather than black and white inspection criteria. While this is inherent in a risk-informed performance-based process, additional clarification should be provided, as noted in the specific comments. Some examples of black and white criteria that could be addressed in an inspection include the following:
   - Are the qualifications for integrated decision-making panel (IDP) members complete, documented, and retained?

- Do the implementing procedures adequately cover the required areas?
- Are the PRA analyses that form the basis for the risk inputs to the IDP performed in accordance with the defined categorization process?
- Are deterministic hazard assessments (e.g., seismic margins assessment [SMA] or fire-induced vulnerability evaluation [FIVE]), if used as allowed in the categorization process, appropriately applied and based on information determined to be currently applicable to the plant?
- Are the functions for a system adequately defined and correct?
- Is the component-function mapping used for the categorization correct?
- Are all components in a system mapped to a function (completeness)?

2.0 Specific Comments

02.01 a. 1st Paragraph:

Clarify distinction between active functions and passive pressure boundary functions because the two will likely be addressed separately by licensee procedures and processes.

02.01 a. 3rd Paragraph:

Reference is made to "… the importance of the component to seismic, fire, and other initiating events that are modeled in the PRA." Since 10 CFR 50.69 does not require PRA models for other than internal events at power, suggest clarifying this statement to read "… the importance of the component to the results of the internal events at power PRA and to the results of other hazards (e.g., seismic, fire, other) that are modeled in the PRA."

02.01 c. 2nd Paragraph:

Risk impact is discussed in a paragraph focused on defense in depth and safety margin. Suggest deleting the reference to 10 CFR 50.69(c)(1)(iv).

02.01 d. 2nd Paragraph:

The last sentence confuses the risk-informed, performance-based process allowed under 10 CFR 50.69 for relaxation of special treatment requirements with maintenance of the plant licensing basis by the licensee. The 10 CFR 50.69 performance feedback process is separate from application of codes and standards or safety analysis acceptance criteria. This paragraph needs to be clarified or replaced.

02.01 g. 2nd paragraph:

Additional clarity could be provided by addressing treatment procedure adequacy separate from procedure implementation. As written, limited guidance is provided.

02.01 h. 2nd paragraph, 3rd sentence:

Alternative treatments for RISC-3 components may be based on voluntary consensus standards and are not required to use industry codes and standards. However, RISC-1 components will continue to apply these industry codes and standards and the licensee may elect to continue using industry standards for RISC-3 components. Specific provisions of the credited industry code, standard, or program may be less restrictive for an RISC-3 component than an RISC-1 component.

2.01 h. 2nd paragraph, 4th sentence:

Use of the term "proven level of reliability" is inappropriate in context of consensus standards or NRC guidance. Such guidance may establish reliability targets or criteria for determining an acceptable level of reliability (performance). However, the key performance-based objective is, as stated earlier in the paragraph, to provide reasonable confidence that SSCs perform their safety-related function(s) under design basis conditions, and in some cases, beyond design basis conditions consistent with categorization process assumptions.

02.01 h: 2nd paragraph, last sentence:

The implication of listing 10 CFR 50.55a and the indicated Regulatory Guides (RGs) as acceptable methods of establishing RISC-3 treatment is that NRC inspectors will not accept other treatment as acceptable. The wording should be reviewed, and revised as necessary, to ensure that NRC inspectors are able to consider alternative approaches.

02.01 k. 3rd sentence:

The requirement to confirm that the licensee is implementing reporting requirements NOT required by 10 CFR 50.69 seems to be misplaced in this paragraph. Further, RISC-1 SSCs would already be covered by the noted requirements, so if the concern is focused on RISC-2 SSCs, it would be clearer to address that category specifically in a separate paragraph.

02.02 a. 1st Paragraph:

This seems to be a broadly-focused requirement that encourages NRC inspectors to second-guess the results of the licensee's IDP process. How will an inspector accomplish the requirement stated to "resolve any aspects of the licensee's categorization results during the inspection"? What objective criteria will be used?

02.02 b. 2nd Paragraph, 2nd sentence:

The statement is made that the PRA must be maintained "as described in the ASME/ANS PRA Standard endorsed by the latest revision of Regulatory Guide 1.200." NRC should not require that the PRA capabilities and maintenance practices (and thereby the 10 CFR 50.69 categorization results) be subject to change any time the PRA Standard or RG 1.200 are changed. Further, there is no formal regulatory or legal requirement regarding the process for PRA maintenance.

02.02 d. 3rd Paragraph:

The criteria stated in 02.02 d. are central to the concept of relaxation of special treatment requirements. However, it is unclear how an NRC inspector will assess the potential for degradation of containment as a barrier (to release of radioactivity) due to categorization of associated SSCs to RISC-3. What objective criteria are to be used? How will this assessment be uniformly made from one inspection to the next?

02.02 e.:

The heading of this paragraph would more properly be “RISC-3 SSC sensitivity evaluations were properly performed”.

02.02 e.:

The last paragraph of this section addresses safety margins and appears to be unrelated to the rest of the section. It would be more appropriate in a different section.

02.02 g. 2nd Paragraph:

The need for IDP members to have familiarity with the plant, and to have broad knowledge of the design and operation of nuclear power plants in general, is important to effective decision-making by the panel. However, the specifications for IDP panel member years of experience in this paragraph are arbitrary and go beyond the requirements stated in the rule or in RG 1.201. What is the basis for requiring a particular number of members to have a particular number of years of experience or a particular number of years of work on the plant PRA? What violation would be imposed if the mix of member experience was substantial but different than the requirements stated here?

02.02 h. 1st and 2nd Paragraph:

Consider changing “design-related functions” to “design functions” and add a sentence in the first paragraph noting that the structure, system, or component design function relates to the functional requirements developed during the categorization process.

02.02 i. 3rd Paragraph:

This paragraph imposes numerous expectations for RISC-3 SSCs without apparent basis or reference to the rule. An SSC can only be RISC-3 if it is low safety significance, and the performance monitoring aspects of the program are designed to ensure that assumptions made in the categorization process remain applicable over time. It is unclear as to how an NRC inspector will apply these requirements.

02.02 i. 13th Paragraph:

This paragraph deals with collective safety significance and maintenance of design basis capabilities of RISC-3 SSCs. It requires that licensees “obtain data or information sufficient to make a technical judgment ...” regarding RISC-3 SSC design basis capabilities. It is unclear as to what the expectations of an inspector might be in this regard. The NEI-00-04 categorization process, as endorsed in RG 1.201, provides process steps to evaluate the cumulative impact of LSS SSCs on safety significance, and the performance monitoring requirements of the rule address the ongoing need to evaluate performance issues of LSS SSCs. How will an inspector determine that the requirements in this paragraph are being met?

02.02 i. next-to-last paragraph:

The entire discussion implies that NRC inspectors will expect a more rigorous reliability program for RISC-3 SSCs than for other SSCs. As RISC-3 SSCs are those that are not safety significant, the performance monitoring and corrective action requirements of the program provide an adequate process for detecting and correcting any degradation in RISC-3 SSC performance. Licensees will obtain statistically significant operating experience data, including potential effects of equipment aging, over time and will be able to apply that experience to improvement of RISC-3 SSC maintenance and performance. But it is unclear as to what the expectations are that are being stated in this paragraph.

02.02 j. 1st Paragraph:

The rule does not require audits and self-assessments for the 10 CFR 50.69 program. Periodic reviews are required, but these are not the same thing. While most licensee's nuclear oversight programs would likely audit 10 CFR 50.69 program performance, and there may be programmatic self-assessment processes, these will vary among licensees. Additional clarification and guidance should be provided here.

02.04:

Reference is made to the South Texas Project (STP) safety evaluation for exemption from certain special treatment requirements as background reading for inspectors. However, the STP exemption was not granted in accordance with the current 10 CFR 50.69 requirements and RG 1.201 criteria. Additional clarification in this regard should be provided for use by NRC inspectors.