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

July 16, 2014

Mr. Mark A. Satorius
Executive Director for Operations
U.S. Nuclear Regulatory Commission
Washington, DC 20555-0001

SUBJECT: STANDARD REVIEW PLAN CHAPTER 19 AND SECTION 17.4

Dear Mr. Satorius:

During the 616th meeting of the Advisory Committee on Reactor Safeguards, July 9-11, 2014, we met with representatives of the NRC staff to review Chapter 19 and Section 17.4 of NUREG-0800, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants: LWR Edition." Our Reliability and PRA Subcommittee reviewed this material during a meeting on March 20, 2014. We also had the benefit of the documents referenced.

RECOMMENDATIONS

1. Section 17.4 and Section 19.3 of the Standard Review Plan (SRP) should be combined to provide consistent guidance for reviews of risk-significant non-safety-related structures, systems, and components (SSCs). The guidance should not distinguish between plant designs that employ "passive" safety features or "active" safety features. The guidance should consolidate expectations for regulatory and licensee programs that provide assurance of adequate availability and reliability for risk-significant non-safety-related SSCs that are not covered by the plant Technical Specifications.

2. The staff should re-evaluate the criteria that are used to determine risk significance in a manner that is consistent for a broad spectrum of designs and absolute levels of overall plant risk. Additionally, risk importance measures should be applied at the component level, and not at the level of specific failure modes.

3. The staff should consider revised guidance that endorses probabilistic risk assessment (PRA) conformance with ASME/ANS Capability Category II requirements to the greatest extent achievable at the design certification and combined license stages of the licensing reviews. Staff reviewers should assess the adequacy of peer reviews that are performed for the PRA and justifications why specific elements of Capability Category II cannot be achieved.

BACKGROUND

The SRP is updated periodically to incorporate evolving knowledge and to provide guidance for reviews of new licensing applications. Individual sections of the SRP are revised on an as-needed basis, depending on their particular scope of review and applicability. Our review covered the following sections of Chapter 17 and Chapter 19:

- Section 17.4, "Reliability Assurance Program," Revision 1, May 2014 (released for use)
- Section 19.0, "Probabilistic Risk Assessment and Severe Accident Evaluation for New Reactors," Draft Revision 3, September 2012 (issued for public comment)
- Section 19.1, "Determining the Technical Adequacy of Probabilistic Risk Assessment for Risk-Informed License Amendment Requests after Initial Fuel Load," Revision 3, September 2012 (released for use)
- Section 19.2, "Review of Risk Information used to Support Permanent Plant-Specific Changes to the Licensing Basis: General Guidance," Revision 0, June 2007 (released for use)
- Section 19.3, "Regulatory Treatment of Non-Safety Systems for Passive Advanced Light Water Reactors," Draft Revision 0, June 2013 (issued for public comment)
- Section 19.4, "Strategies and Guidance to Address Loss of Large Areas of the Plant due to Explosions and Fires," Draft Revision 0, May 2013 (issued for public comment)
- Section 19.5, "Adequacy of Design Features and Functional Capabilities Identified and Described for Withstanding Aircraft Impacts," Revision 0, April 2013 (released for use)

The staff currently intends to issue a new draft revision of Section 19.0 for public comments, including additional guidance for new reactor applicants. Our comments and recommendations in this report apply to Draft Revision 3 of that section, issued in September 2012.

DISCUSSION

We were briefed on each SRP section, its technical bases, and the reasons why each revision or newly issued section is needed. We reviewed the topics covered by each section, and we concur with the guidance in Sections 19.1, 19.2, 19.4, and 19.5.

The following sections of this letter report summarize our review of the material in Sections 17.4, 19.0, and 19.3.

<u>Consolidation of Sections 17.4 and 19.3</u>

Section 19.3 contains guidance for reviews of Regulatory Treatment of Non-Safety Systems (RTNSS) controls that are applied to non-safety-related SSCs which satisfy specific criteria or are identified as "risk-significant" in the design certification or combined license application for a plant that employs "passive" safety features. Section 17.4 contains guidance for reviews of the Reliability Assurance Program (RAP) that is applied to SSCs which are identified as "risk-significant" in a design certification or combined license application. Section 17.4 notes that all RTNSS SSCs are included in the RAP.

These sections of the SRP perpetuate an artificial distinction between non-safety-related SSCs that are designated for RTNSS treatment in a "passive" plant design, and non-safety-related SSCs that are designated for RAP treatment in an "active" plant design. This distinction has evolved from the historical progression of new reactor design certifications. It has no fundamental technical basis. The same process should be used to determine the risk significance of safety-related and non-safety-related SSCs, without regard to a particular design's designation as "passive" or "active", and the same regulatory oversight and licensee controls should apply to those SSCs.

These assessments are further complicated by specific guidance that applies to the identification of RTNSS SSCs for "passive" plants, but not to similar SSCs in "active" plants. For example, Section 19.3 delineates specific criteria for selection of RTNSS SSCs (e.g., RTNSS A through E) which are based on historical assessments of perceived safety importance for currently operating reactors without the benefit of integrated risk evaluations for a particular new plant design. As well as their inclusion in the RAP, selected RTNSS SSCs are subject to additional reliability and availability controls in the form of conditions that are specified in an Availability Controls Manual (ACM). The ACM is functionally similar to the plant Technical Specifications, but less restrictive. The ACM controls currently apply only to RTNSS SSCs. They do not apply to non-RTNSS SSCs that are included in the RAP for a "passive" plant design, or to any risk-significant non-safety-related SSCs for an "active" plant design.

This distinction has created somewhat differing perspectives about the identification of risk-significant non-safety-related SSCs for each plant design. It has also resulted in different guidance for reviews of the regulatory and licensee programs that are applied to those SSCs. Because sections of the SRP are updated individually on an as-needed basis, it is possible that the guidance in these sections will diverge further in the future.

Section 17.4 and Section 19.3 should be combined to provide consistent guidance for reviews of risk-significant non-safety-related SSCs. The guidance should not distinguish between the risk significance determinations for plant designs that employ "passive" safety features or "active" safety features. The guidance should also consolidate expectations for regulatory and licensee programs that provide assurance of adequate availability and reliability for risk-significant non-safety-related SSCs that are not covered by the plant Technical Specifications.

Determination of Risk Significance

Section 17.4 notes that the RAP and its review are focused on "dominant failure modes" for each risk-significant SSC. Section 19.0 notes that risk significance is determined at the level of individual basic events in the PRA models, which are typically equivalent to component failure modes.

The PRA models that are developed to support a design certification or combined license application are often quite simplified, compared to a full-scope PRA for an operating reactor. Identification of risk-significant SSCs often requires qualitative input from expert panels to supplement the quantitative measures from limited-scope PRA evaluations. It is premature to identify "dominant failure modes" at this stage of the risk assessment process. Staff acceptance of specific failure modes in NRC-approved programs at the issuance of a design certification or a combined license may cause reluctance to make changes as the PRA scope and detail are enhanced and plant operating experience is obtained. Therefore, risk importance measures that are used to identify the SSCs in availability and reliability assurance programs for a certified design or combined license should be applied at the component level, and not at the level of specific failure modes.

Section 19.0 also indicates that risk significance is determined according to the guidance in Regulatory Guide 1.200, which states:

> *Significant basic event / contributor: The basic events (i.e., equipment unavailabilities and human failure events) that have a Fussell-Vesely importance greater than 0.005 or a risk-achievement worth greater than 2.*

Even if these criteria are applied at the component level, rather than the basic event level, a large number of SSCs may be identified as "risk-significant". This is especially true for new plant designs that have very low estimated frequencies of core damage and large releases. Universal application of these specific numerical criteria may produce an inappropriately large population of SSCs that are subject to enhanced availability and reliability controls, with commensurate undue burden for both the licensee and regulatory staff. The staff should re-evaluate the criteria that are used to determine risk significance in a manner that is consistent for a broad spectrum of designs and absolute levels of overall plant risk.

PRA Capability

Section 19.0 states:

> *PRAs that meet the applicable supporting requirements for Capability Category I and meet the high-level requirements as defined in the ASME PRA Standard (ASME/ANS RA-S-2008) and addenda ASME/ANS RA-Sa–2009) should generally be acceptable for DC and COL applications.*

These requirements are often cited as a primary reason why PRA models that are developed for design certification and combined license applications are very simplified in both scope and level of detail. They have resulted in broad variations of technical quality among PRAs that have been developed for completed design certifications and for designs that are currently in the certification review process.

The staff should consider revised guidance that endorses PRA conformance with ASME/ANS Capability Category II requirements to the greatest extent achievable at the design certification and combined license stages of the review process. Staff reviewers should assess the adequacy of peer reviews that are performed for the PRA and justifications why specific elements of Capability Category II cannot be achieved.

We have a separate concern regarding the capability of PRAs that support the identification of SSCs that are subject to RTNSS and RAP treatment. The design certification and combined license PRAs have not fully addressed the probability of degradation of the sometimes-delicate thermal-hydraulic balances that drive passive system response. None have examined the uncertainties in environmental conditions, maintenance-induced damages, corrosion deposit effects, or age-related degradation that could reduce the effectiveness of the safety-related passive system performance. Therefore, the risk significance of non-safety-related SSCs may not be fully recognized or may be masked.

We understand that some of our recommendations involve issues that extend beyond revisions to the SRP and associated regulatory guidance. We look forward to working with the staff to achieve their resolution.

Sincerely,

/RA/

John W. Stetkar
Chairman

REFERENCES

1. NUREG-0800, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants: LWR Edition," Section 17.4, "Reliability Assurance Program," Revision 1, May 2014 (ML13296A435)

2. NUREG-0800, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants: LWR Edition," Section 19.0, "Probabilistic Risk Assessment and Severe Accident Evaluation for New Reactors," Draft Revision 3, September 2012 (ML12132A481)

3. NUREG-0800, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants: LWR Edition," Section 19.1, "Determining the Technical Adequacy of Probabilistic Risk Assessment for Risk-Informed License Amendment Requests after Initial Fuel Load," Revision 3, September 2012 (ML12193A107)

4. NUREG-0800, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants: LWR Edition," Section 19.2, "Review of Risk Information used to Support Permanent Plant-Specific Changes to the Licensing Basis: General Guidance," Revision 0, June 2007 (ML071700658)

5. NUREG-0800, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants: LWR Edition," Section 19.3, "Regulatory Treatment of Non-Safety Systems for Passive Advanced Light Water Reactors," Draft Revision 0, June 2013 (ML13081A756)

6. NUREG-0800, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants: LWR Edition," Section 19.4, "Strategies and Guidance to Address Loss of Large Areas of the Plant due to Explosions and Fires," Draft Revision 0, May 2013 (ML121110138)

7. NUREG-0800, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants: LWR Edition," Section 19.5, "Adequacy of Design Features and Functional Capabilities Identified and Described for Withstanding Aircraft Impacts," Revision 0, April 2013 (ML12276AA112)

8. DC-COL-ISG-003, "Interim Staff Guidance: Probabilistic Risk Assessment Information to Support Design Certification and Combined License Applications," August 11, 2008(ML080570048)

9. DC-COL-ISG-016, "Interim Staff Guidance: Compliance with 10 CFR 50.54(hh)(2) and 10 CFR 52.80(d) Loss of Large Areas of the Plant due to Explosions or Fires from a Beyond-Design Basis Event," September 18, 2010(ML100431200)

10. DC-COL-ISG-018, "Interim Staff Guidance on Standard Review Plan, Section 17.4, 'Reliability Assurance Program'," July 25, 2011(ML103010113)

11. DC-COL-ISG-020, "Interim Staff Guidance on Implementation of a Probabilistic Risk Assessment-Based Seismic Margin Analysis for New Reactors," September 18, 2010(ML100491233)

12. DI&C-ISG-03, "Interim Staff Guidance: Task Working Group #3: Review of New Reactor Digital Instrumentation and Control Probabilistic Risk Assessments," Revision 0, August 11, 2008 (ML080570048)

13. Regulatory Guide 1.200, "An Approach for Determining the Technical Adequacy of Probabilistic Risk Assessment Results for Risk-Informed Activities," Revision 2, March 2009 (ML090410014).

6. NUREG-0800, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants: LWR Edition," Section 19.4, "Strategies and Guidance to Address Loss of Large Areas of the Plant due to Explosions and Fires," Draft Revision 0, May 2013 (ML121110138)

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