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

October 20, 2010

The Honorable Gregory B. Jaczko
Chairman
U.S. Nuclear Regulatory Commission
Washington, DC 20555-0001

SUBJECT: DRAFT FINAL RULE FOR RISK-INFORMED CHANGES TO LOSS-OF-COOLANT ACCIDENT TECHNICAL REQUIREMENTS (10 CFR 50.46a)

Dear Chairman Jaczko:

During the 576th meeting of the Advisory Committee on Reactor Safeguards, October 7-9, 2010, we reviewed the proposed Draft Final Rule for Risk-Informed Changes to Loss-of-Coolant Accident (LOCA) Technical Requirements (10 CFR 50.46a or the Rule). Our Regulatory Policies and Practices Subcommittee also reviewed this matter during its meeting on September 22, 2010. During these reviews, we had the benefit of discussions with representatives of the NRC staff. We also had the benefit of the documents referenced.

CONCLUSIONS AND RECOMMENDATIONS

1. The proposed draft final rule, 10 CFR 50.46a, "Alternative acceptance criteria for emergency core cooling systems for light-water nuclear power reactors," provides an acceptable risk-informed alternative to the current 10 CFR 50.46(a) for operating reactors.

2. It is premature to extend the proposed Rule to new reactors at this time. New reactors are expected to have risk profiles that are significantly different from the current operating fleet, and the development of appropriate risk metrics and risk acceptance criteria for these reactors is still at the conceptual stage.

3. If new reactors are included in the scope of the Rule, then we agree with the requirement that changes made possible by the adoption of the Rule should not result in a significant decrease in the level of safety. This requirement should be extended to the determination of the allowable time in configurations without a demonstrated capability to mitigate a beyond transition break size LOCA.

DISCUSSION

In response to a Staff Requirements Memorandum (SRM) dated July 1, 2004, the staff has developed an alternative set of risk-informed requirements for emergency core cooling systems (ECCS). Licensees may voluntarily choose to comply with these requirements in lieu of meeting the existing requirements in 10 CFR 50.46(a). The proposed Rule divides the spectrum of LOCA break sizes into two regions. The demarcation between the two regions is called a transition break size (TBS). The first region includes small breaks up to and including the TBS.

The second region includes breaks larger than the TBS up to and including the double-ended guillotine break (DEGB) of the largest reactor coolant system pipe.

In our report of November 16, 2006, on an earlier version of the proposed Rule, we recommended that it not be issued in its then current form. It needed to be revised to strengthen the assurance of defense in depth for breaks beyond the TBS, in particular, by requiring that licensees submit the codes used for the analyses of breaks beyond the TBS to the NRC for review and approval. The current version of the Rule includes this requirement.

The proposed Rule limits the allowable time of at-power operation in configurations without a demonstrated capability to mitigate a beyond-TBS LOCA to be a short time. A short time is defined to be 14 days or a risk-informed alternative proposed by the licensee and approved by the NRC. A minimal increase in core damage frequency of 10^{-7} per year for current reactors due to operation without the capability to mitigate would correspond to 4 days. The guidance in Regulatory Guide 1.177, “An Approach for Plant-Specific, Risk-Informed Decisionmaking: Technical Specifications,” for the increase in conditional core damage probability of 5×10^{-7} yields an allowed outage time of 18 days. Although neither of these conditions is exactly applicable to the situation contemplated by the proposed Rule, the results demonstrate that 14 days is reasonably consistent with risk-informed guidance for current reactors for breaks with a frequency of 10^{-5} per year. With 14 days as the allowable cumulative outage time, mitigation of beyond-TBS LOCAs will be available at least 96 percent of the time, consistent with Commission direction to maintain this capability. The staff also argues that a shorter time period could lead to significant burden to the industry with no clear safety benefits and, if maintenance activities were adversely affected, a possible reduction in safety. We agree that the proposed allowable outage time is acceptable for the current operating reactors.

In our November 16, 2006, report, we also expressed concerns about the magnitude of the increases in risk that could occur due to changes that did not require prior NRC approval. The process for changes that can be made without prior NRC approval has been revised and is now acceptable.

To help preserve safety margins, we also recommended that the proposed Rule not be finalized until fuel cladding acceptance criteria for LOCAs involving breaks at or below the TBS were reviewed and revised to assure their adequacy for the higher burnup fuel and more demanding conditions of current reactor operation. The current version of the Rule still relies on the current cladding acceptance criteria of 10 CFR 50.46(b). The staff recognizes that conforming changes will have to be made to the proposed Rule, if changes are made to 10 CFR 50.46(b). Because an advance notice of proposed rulemaking for revised cladding acceptance criteria has been issued and a substantial amount of relevant research completed, we find it acceptable at this time to proceed with the proposed 10 CFR 50.46a.

In NUREG-1903, “Seismic Considerations for the Transition Break Size,” the staff has assessed the potential for seismic effects on the failure of flawed and unflawed piping and the failure of other components and component supports that could lead to piping failures that affect the choice of the TBS. For the cases analyzed, even for very long circumferential cracks, the study showed that surface flaw depths must be larger than 40 percent of the wall thickness in order to become critical for the stresses resulting from a seismic event with a 10^{-5} annual probability of exceedance. The corresponding surface flaw depth is 30 percent of the wall thickness for seismic events with a 10^{-6} annual probability of exceedance. The study also confirmed that the seismic-induced failure probabilities of unflawed piping are very low compared to the frequency of 10^{-5} per year used as a basis to establish the TBS.

Two cases of indirectly induced piping failure attributable to major component support failure were also analyzed in NUREG-1903. The mean probability of the indirect failure was found to be on the order of 10^{-6} per year. More recently, the Electric Power Research Institute completed studies of indirect failures for three additional plants. The highest failure probability for breaks greater than the proposed TBS calculated for the three plants was 6×10^{-6} per year.

The proposed Rule requires that any changes made under the Rule will not increase the LOCA frequency (including the frequency of seismically induced LOCAs) by an amount that would invalidate the applicability of NUREG-1903 and NUREG-1829, "Estimating Loss-of-Coolant Accident (LOCA) Frequencies Through the Elicitation Process."

NUREG-1903 recognizes that direct and indirect failure evaluations are plant and site-specific. Details of plant layout, and component and pipe support designs vary significantly from plant to plant. In the rule language of the August 23, 2010, version reviewed by our Regulatory Policies and Practices Subcommittee, there was a requirement that licensees provide a plant-specific analysis demonstrating that the risk of seismically induced LOCAs larger than the TBS is comparable to or less than the seismically induced LOCA risk associated with the NUREG-1903 results. However, the Statement of Considerations implied, and the staff confirmed, that this was intended to apply only to direct failures. Members of the subcommittee recommended that indirect failures should also be evaluated on a plant-specific basis. In the current version of the Rule, the staff has amended the rule language and the Statement of Considerations to make clear that both direct and indirect seismically induced LOCAs must be shown to be consistent with the NUREG-1903 results. We agree that this is a necessary change. It is also consistent with the Commission direction in the SRM to SECY-04-0037 to use initiating event frequencies to determine an appropriate TBS.

The staff has prepared a draft regulatory guide, DG-1216, ''Plant-Specific Applicability of the Transition Break Size Specified in 10 CFR 50.46a," that provides acceptable methods and acceptance criteria for an evaluation of the applicability of NUREG-1829. It also provides an evaluation framework and acceptance criteria to demonstrate the applicability of the NUREG-1903 assessment of direct piping failures. The guide should be revised to include the assessment of the applicability of the NUREG-1903 results on indirect failures. The calculation of the likelihood of indirect failures can be difficult. The staff should explore methods to reduce the required effort.

The current design basis analysis for breaks up to the DEGB of the largest pipe in the reactor coolant system requires mitigation of the LOCA with a simultaneous loss of offsite power and a single failure of a safety system. For breaks beyond the TBS under the proposed Rule, it is not necessary to consider loss of offsite power or a single failure when demonstrating mitigation. This is acceptable for internal and many non-seismic external initiating events because the occurrence of a break greater than the TBS, the loss of offsite power, and the single failure are independent events. Therefore, the increase in risk due to the inability to cope with the loss of offsite power and a single failure for such breaks is very small.

Severe seismic accelerations that are strong enough to fail the primary system piping or robust structural supports are well beyond the design basis earthquakes for currently licensed plants. It is very likely that offsite power supplies and nonsafety-related structures, systems, and components (SSCs) will be damaged by these rare severe events. It is also likely that multiple safety-related SSCs will be damaged. These combined failures are inconsistent with current licensing basis assumptions regarding the availability of SSCs for large LOCA mitigation.

Based on these considerations, we conclude that use of the risk-informed criteria in 10 CFR 50.46a rather than the existing deterministic criteria in 10 CFR 50.46(a) will not result in a significant increase in the risk due to large LOCAs that may be caused by beyond design basis seismic events.

A major change in the current version of the proposed Rule over earlier versions is the applicability to new reactors. We agree that improved material selection, water chemistry, and design practices will further reduce the likelihood of large LOCAs in new plants. However, it is premature to extend the proposed Rule to new reactor designs at this time. Such reactors typically have risk profiles that are significantly different from the current operating fleet, and the development of appropriate risk metrics and risk acceptance criteria for these reactors is still at the conceptual stage. The staff has based the extension of the Rule to the new reactor designs on the concept from Option 2 in the recent SECY-10-0121 that the appropriate risk metrics for the new reactor designs should prevent a significant decrease in a new reactor's level of safety over its life. Language to this effect has been added to the current version of the Rule and the Statement of Considerations. However, the Commission has not yet endorsed Option 2, and the staff is still in the process of identifying the specific guidance needed to implement that concept. The Rule should be based on specific guidance rather than a concept not yet clearly defined.

For new reactors, the requirement that changes made possible by the adoption of the Rule should not result in a significant decrease in the level of safety could affect the allowable time in configurations without a demonstrated capability to mitigate a beyond-TBS LOCA. An effective way to do this would be to require that the allowable time, like the TBS, be determined on a plant-specific basis for new reactors.

The proposed draft final rule, 10 CFR 50.46a, is consistent with Commission direction and appropriately incorporates public and ACRS comments. It provides an acceptable risk-informed alternative to the current 10 CFR 50.46(a) for current operating reactors.

Sincerely,

/RA/

Said Abdel-Khalik
Chairman

References:

1. Memorandum to Edwin M. Hackett, transmitting "Advisory Committee on Reactor Safeguards Review of Final Rule on Risk-Informed Changes to Loss-of-Coolant Accident Technical Requirements – 10 CFR 50.46a (RIN 3150-AH29)," 09/27/2010 (ML102700493)

2. Memorandum to Edwin M. Hackett transmitting "Advisory Committee on Reactor Safeguards Review of Final Rule on Risk-Informed Changes to Loss-of-Coolant Accident Technical Requirements – 10 CFR 50.46a (RIN 3150-AH29)," 08/31/2010 (ML102420574)

3. Memorandum to Dr. Graham B. Wallis, transmitting “Advisory Committee on Reactor Safeguards Review of the Draft Final Rule to Amend 10 CFR 50.46, Risk-Informed Changes to Loss-of Coolant Accident Technical Requirements,” 10/26/2006 (ML081720066)

4. SECY-04-0037, “Issues Related to Proposed Rulemaking to Risk-Inform Requirements Related to Large Break Loss-of-Coolant Accident (LOCA) Break Size and Plans for Rulemaking on LOCA with Coincident Loss-of-Offsite Power,” 03/03/2004 (ML040490133)

5. “Staff Requirements - SECY-04-0037 - Issues Related to Proposed Rulemaking to Risk-Inform Requirements Related to Large Break Loss-of-Coolant Accident (LOCA) Break Size and Plans for Rulemaking on LOCA with Coincident Loss-of-Offsite Power,” 07/01/2004 (ML041830412)

6. SECY-10-0121, “Modifying the Risk-Informed Regulatory Guidance for New Reactors,” 09/14/2010 (ML102230076)

7. U.S. Nuclear Regulatory Commission, NUREG-1829, “Estimating Loss-of-Coolant Accident (LOCA) Frequencies Through the Elicitation Process,” April 2008 (Vol. 1-ML080630013; Vol. 2 –ML081060300)

8. U.S. Nuclear Regulatory Commission, NUREG-1903, “Seismic Considerations for the Transition Break Size,” February 2008 (ML080880140)

9. Regulatory Guide 1.174, Revision 1, “An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis,” November 2002 (ML023240437)

10. Regulatory Guide 1.177, “An Approach for Plant-Specific, Risk-Informed Decisionmaking: Technical Specifications,” August 1998 (ML003740176)

Based on these considerations, we conclude that use of the risk-informed criteria in 10 CFR 50.46a rather than the existing deterministic criteria in 10 CFR 50.46(a) will not result in a significant increase in the risk due to large LOCAs that may be caused by beyond design basis seismic events.

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Letter to the Honorable Gregory B Jaczko, Chairman, NRC, from Said Abdel-Khalik, Chairman, ACRS, dated October 20, 2010

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