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

ACRSR-2223

November 16, 2006

The Honorable Dale E. Klein
Chairman
U.S. Nuclear Regulatory Commission
Washington DC 20555-0001

SUBJECT: DRAFT FINAL RULE TO RISK-INFORM 10 CFR 50.46, "ACCEPTANCE CRITERIA FOR EMERGENCY CORE COOLING SYSTEMS FOR LIGHT-WATER NUCLEAR POWER REACTORS"

Dear Chairman Klein:

During the 537th meeting of the Advisory Committee on Reactor Safeguards, November 1-3, 2006, we met with representatives of the NRC staff and the Boiling Water Reactor (BWR) Owners' Group to discuss the draft final rule to risk-inform 10 CFR 50.46, "Acceptance Criteria for Emergency Core Cooling Systems for Light-Water Nuclear Power Reactors," (the Rule). We also had the benefit of the documents referenced.

**RECOMMENDATIONS**

1. The Rule to risk-inform 10 CFR 50.46 should not be issued in its current form. It should be revised to strengthen the assurance of defense in depth for breaks beyond the transition break size (TBS). Such assurance would reduce concerns about uncertainties in determining the TBS.

2. The revision of draft NUREG-1829, "Estimating Loss-of-Coolant Accident (LOCA) Frequencies Through the Elicitation Process," to include changes resulting from the resolution of public comments should be completed before the revised Rule is issued. This state-of-the-art review on the estimation of break size frequencies is an essential part of the technical basis for the Rule.

3. The interpretation that the Rule limits the total increase in core damage frequency (CDF) resulting from all changes in a plant that adopts the Rule to be "small" (i.e., $<10^{-5}$/yr) represents a significant departure from the current guidance for risk-informed regulation and should be reviewed for its implications.

**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. The Rule divides the spectrum of LOCA break sizes into two regions. The demarcation between the two regions is called a "transition break size." 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.

Because pipe breaks in the smaller break size region are considered more likely than pipe breaks in the larger break size region, each region would be subject to different ECCS requirements. Loss-of-coolant accidents in the smaller break size region would be analyzed using the methods, assumptions, and criteria currently used for LOCA analysis; accidents in the larger break size region would be analyzed using less stringent methods, assumptions, and criteria due to their lower likelihood of occurrence. Although LOCAs for break sizes larger than the TBS would become "beyond design-basis accidents," the Rule requires that licensees maintain the ability to mitigate all LOCAs up to and including the DEGB of the largest reactor coolant system pipe.

The fundamental principles of a risk-informed regulation should be to ensure that any increases in risk associated with a change are small, that changes are consistent with the defense-in-depth philosophy, and that adequate safety margins are maintained. Regulatory Guide 1.174 provides quantitative criteria for assessing changes in risk, but its guidance on ensuring consistency with the defense-in-depth philosophy and maintaining adequate safety margins is more subject to engineering judgment.

Probabilistic risk assessments of internal events typically show that large-break LOCAs (LBLOCAs) are relatively small contributors to CDF. The results in draft NUREG-1829 suggest that the contribution to CDF from breaks larger than the TBS proposed in the Rule is a small fraction of the already small contribution to CDF due to all LBLOCAs. Thus, the requirements for mitigation capabilities for breaks beyond the TBS should be based on defense-in-depth considerations to provide margin against unanticipated degradation phenomena, human errors, extremely large loads such as those associated with earthquakes beyond the safe shutdown earthquake, and other unanticipated events. The degree of defense in depth required can only be determined by judgment based on experience and best attempts to quantify uncertainties.

The Rule requires an analysis to demonstrate mitigation for breaks greater than the TBS, up to the DEGB of the largest pipe in the reactor coolant system. The requirements in the Rule provide a degree of assurance of this mitigation. It is our judgment, however, that the Rule should impose additional requirements to strengthen this assurance.

Because the Rule now defines pipe breaks greater than the TBS as “beyond design basis,” any equipment required solely to mitigate such breaks may no longer be safety-related and could be subject to less stringent maintenance and inspection requirements that could adversely affect its reliability. Such equipment could even be removed from technical specifications that control its availability. We agree that the low likelihood of breaks greater than the TBS justifies a relaxation in the requirements for mitigating such events, but this relaxation should instead result from the removal of additional requirements that make such events even more unlikely, such as the simultaneous loss-of-offsite-power (LOOP) and the assumption of the worst single failure. Confidence in the reliability and availability of the equipment needed to mitigate such breaks is important not only for defense in depth, but also for maintaining safety margins for breaks smaller than the TBS.

The Rule also provides restrictions on the unavailability of the non-safety-related equipment needed to mitigate breaks beyond the TBS, but it imposes no other requirements. We believe that the equipment needed to mitigate these breaks deserves some special treatment and control. The staff has dealt with the regulatory treatment of non-safety systems in other contexts, and similar approaches would be appropriate here.

The Rule should also increase confidence in the ability to mitigate breaks greater than the TBS by requiring licensees to submit the codes used for the analyses of breaks beyond the TBS to the NRC for review and approval.

The Rule is an enabling rule that will permit licensees to make changes that increase operational flexibility and reduce regulatory burden, which could result in increases or decreases in risk. The Rule contains a risk-informed change process that will control all changes in risk that occur after a licensee adopts the Rule. The risk-informed change process in the Rule uses the current 10 CFR 50.59 change process and the 10 CFR 50.65 maintenance rule categorization to screen changes that can impact risk. However, as currently envisioned by the staff, it allows the licensee in some cases to implement changes that have a ΔCDF greater than $10^{-6}$/yr but less than $10^{-5}$/yr without prior review by the staff. Regulatory Guide 1.174 would typically allow such changes only if the total CDF, including external events and low-power/shutdown events, is less than $10^{-4}$/yr. Licensees should submit such changes to the staff for prior review and approval. Licensees could still implement changes that result in a ΔCDF < $10^{-6}$/yr without prior review and should track the quantified changes in CDF in the 24 month report.

The Rule requires that the total increase in CDF resulting from all changes in a plant that adopts the Rule be “small” (i.e., < $10^{-5}$/yr). This “cap” on the increase in risk applies regardless of whether the changes in CDF result from changes related to 10 CFR 50.46. This represents a significant departure from the current guidance for risk-informed regulation and should be reviewed for its implications.

Maintaining sufficient safety margin is another important element of risk-informed regulation that is not treated quantitatively in Regulatory Guide 1.174. It is likely that, with this Rule, the NRC will find requests for additional power uprates at pressurized water reactors (PWRs) acceptable. However, the uprates will clearly decrease safety margins, even for breaks below the TBS. The Rule currently contains acceptance criteria for fuel cladding performance under LOCA conditions based on the current 10 CFR 50.46. The Office of Nuclear Regulatory Research is now completing an examination of the adequacy of these criteria for high-burnup fuel. The adequacy of the acceptance criteria for cladding performance is important to maintain adequate safety margins. The Rule should not be finalized until the fuel cladding acceptance criteria for LOCAs involving breaks at or below the TBS are reviewed and/or revised to assure their adequacy for the higher burnup fuel and more demanding conditions of current reactor operating conditions. Alternatively, the acceptance criteria in the Rule could be expressed in terms of general requirements, such as a high degree of confidence in maintaining a coolable geometry and retaining some ductility in the cladding. Specific cladding and core criteria could be placed in the associated regulatory guide.

An important element in the selection of the TBS is the state-of-the-art review of break size frequencies conducted by the Office of Nuclear Regulatory Research, documented in draft NUREG-1829. There is substantial uncertainty in the determination of these frequencies. If there is a high degree of assurance that breaks greater than the TBS can be mitigated, the impact of this uncertainty on the selection of the TBS is substantially reduced. The selection of the TBS could then include consideration of the benefits of small changes in the break size. For example, the current TBS for BWRs inhibits implementation of longer diesel start-up times, which are almost universally agreed to lead to improved emergency diesel generator operability. If the staff strengthens the defense in depth for breaks greater than the TBS, the TBS proposed by the BWR Owners’ Group could be acceptable and would not be inconsistent with the results in draft NUREG-1829.

Although the Rule defines TBSs for BWRs and PWRs, licensees should not presume that these automatically apply to all plants. As part of the adoption of the Rule, licensees should have to demonstrate that the results in draft NUREG-1829 are applicable to their plants. The staff should provide guidance for this demonstration in the associated regulatory guide. As part of this demonstration, licensees should demonstrate that the reactor coolant system piping of diameter corresponding to the TBS or larger meets the deterministic requirements currently used to credit leak-before-break for dynamic analysis of reactor coolant piping. Such demonstrations will provide additional assurance of the very low likelihood of failures greater than the TBS. Many plants should have already performed such analyses.

The staff is revising draft NUREG-1829 to incorporate, as appropriate, the changes resulting from the resolution of public comments. This revision should be completed prior to issuing the revised Rule.

For internal events, the occurrence of a LBLOCA and a LOOP can generally be considered as independent events, and thus the simultaneous occurrence of a break greater than the TBS and a LOOP is a very unlikely event. However, a LOOP is very likely for any seismic event that is large enough to induce failures in reactor piping systems. As part of its effort to establish the TBS, the staff performed a study of the likelihood of seismically induced failures in unflawed piping, flawed piping, and indirect failures of other components and component supports that could lead to piping failure. The study focused on piping systems in PWRs east of the Rocky Mountains. We have not yet completed our review of the staff's study in this area. However, the results of the study indicate that for these plants the likelihood of seismically induced failures in unflawed piping of size greater than the TBS is very low for earthquakes with $10^{-5}$ and $10^{-6}$ annual probabilities of exceedance. Even for pipes with long surface flaws, the depths of these flaws must be greater than 30-40% of the wall thickness for a high likelihood of failure during such earthquakes. Inspection programs, leak detection systems, and other measures taken to eliminate failure mechanisms such as stress corrosion cracking should make the likelihood of such cracks very low. Because seismic hazards are very plant specific, licensees adopting the Rule will have to demonstrate that the results developed by the staff bound the likelihood of seismically induced failure in their plants. For unflawed piping, the results of the individual plant examination of external events (IPEEE) program may provide the needed information. Licensees may have to perform additional calculations to demonstrate a comparable robustness of flawed piping.

Although substantial progress has been made in the development of a risk-informed 10 CFR 50.46, the Rule should not be issued in its current form. It would be significantly strengthened by addressing the issues raised in this report.

Additional comments by ACRS Member Graham B. Wallis and ACRS Member Sanjoy Banerjee are presented below.

Sincerely,

**/RA/**

Graham B. Wallis
Chairman

**Additional comments from ACRS Member Graham B. Wallis**

My colleagues have suggested some significant improvements to the draft final rule, which I support, if it should be issued as final.

However, I am not persuaded that an adequate case has been made for this rule or that its consequences have been sufficiently explored.

The probabilities for breaks of various sizes, as assessed in draft NUREG-1829, can be accommodated within the framework of the existing rule's "realistic (best estimate)" alternative without any new rulemaking. This can be done in numerous ways while preserving suitable caution and defense in depth. The details can be worked out between the staff and licensees through an evolutionary process that includes thorough consideration of practicality, enforcement, technical uncertainties, benefits, and risks.

**Additional comments from ACRS Member Sanjoy Banerjee**

I support the Recommendations in the ACRS letter regarding the draft final rule to risk inform 10 CFR 50.46, but would add the further Recommendation that the draft NUREG-1829 be externally peer reviewed before being issued.

I have arrived at this Recommendation after reviewing NUREG-1829 and transcripts of 5 meetings regarding the work contained in it, held by the ACRS Regulatory Policies and Practices Subcommittee from 11/21/03 to 11/16/04. Based on this, it is my opinion that the quality of the NUREG and the credibility of its conclusions, would be substantially enhanced by eliciting, and responding to, comments from external and independent peer reviewers. This point was also raised at several of the ACRS Subcommittee meetings, but no substantive external peer review appears to have been conducted.

Amongst the several issues which, in my opinion, may be elucidated by such a review are the wide divergence in the initial estimates for various LOCA frequencies, and the methods used to narrow the range of uncertainty in the final results from which the conclusions are drawn.

References:

1. Memorandum from Michael Marshall Jr., Acting Branch Chief, Financial, Policy, and Rulemaking Branch, Division of Policy and Rulemaking, Office of Nuclear Reactor Regulation, to Dr. Graham B. Wallis, Chairman, Advisory Committee on Reactor Safeguards, "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'," dated October 26, 2006.

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2. Report from Graham B. Wallis, Chairman, Advisory Committee on Reactor Safeguards, to Nils. J. Diaz, Chairman, U.S. Nuclear Regulatory Commission, “Proposed Rulemaking to Modify 10 CFR 50.46, ‘Risk-Informed Changes to Loss-of-Coolant Accident Technical Requirements’,” dated March 14, 2005.
3. Report from Mario V. Bonaca, Chairman, Advisory Committee on Reactor Safeguards, to Nils. J. Diaz, Chairman, U.S. Nuclear Regulatory Commission, “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’,” dated April 27, 2004.
4. Staff Requirement Memorandum from Annette L. Vietti-Cook, Secretary, U.S. Nuclear Regulatory Commission, to Luis A. Reyes, Executive Director for Operations, U.S. Nuclear Regulatory Commission, “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,” dated July 1, 2004.
5. U.S. Nuclear Regulatory Commission, NUREG-1829, “Estimating Loss-of-Coolant Accident (LOCA) Frequencies Through the Elicitation Process,” Draft Report for Comment, June 2005.
6. U.S. Nuclear Regulatory Commission, Regulatory Guide 1.174, “An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis,” November 2002.
7. U.S. Nuclear Regulatory Commission, “Seismic Considerations for the Transition Break Size,” December 2005, ADAMS ML053470439.
8. Letter from Randy C. Bunt, Chair, BWR Owners' Group, to Graham B. Wallis, Chairman, Advisory Committee on Reactor Safeguards, “Draft Final Rule Language, Risk-Informed Changes to Loss-of-Coolant Accident Technical Requirements, ADAMS Accession NO. ML062760146, dated October 3, 2006,” dated October 13, 2006.