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

December 17, 2004

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

SUBJECT: RISK-INFORMING 10 CFR 50.46, "ACCEPTANCE CRITERIA FOR EMERGENCY CORE COOLING SYSTEMS FOR LIGHT-WATER NUCLEAR POWER REACTORS"

Dear Mr. Reyes:

During the 518th meeting of the Advisory Committee on Reactor Safeguards on December 2-4, 2004, we reviewed a draft version of a proposed rule for a voluntary alternative to 10 CFR 50.46, "Acceptance Criteria for Emergency Core Cooling Systems (ECCS) for Light-Water Nuclear Power Reactors" (Reference 1). We also reviewed draft proposed rule language (Reference 2) during the 517th meeting on November 4-6, 2004. Our Subcommittee on Regulatory Policies and Practices reviewed this matter during a meeting on October 28-29, 2004. During these reviews, we had the benefit of discussions with the NRC staff, the Nuclear Energy Institute, Westinghouse Owners Group, and members of the public. We also had the benefit of the documents referenced. Although the proposed rule language has not been finalized, we present our views on some of the basic elements of a risk-informed 10 CFR 50.46.

CONCLUSIONS AND RECOMMENDATIONS

1. A risk-informed 10 CFR 50.46 should maintain defense in depth by including requirements intended to provide reasonable assurance of a coolable core geometry for breaks up to the double-ended guillotine break (DEGB) of the largest pipe in the reactor coolant system.

2. The results of the expert opinion elicitation need to be further reviewed and assessed by the staff before finalizing the selection of the transition break size. Nevertheless, it appears that a transition break size corresponding to the single-ended rupture of the largest pipe attached to the reactor coolant system bounds the range of break sizes corresponding to a frequency of 1 $\times 10^{-5}$/year.

3. A better quantitative understanding of the possible risk benefits of a smaller transition break size is needed to arrive at a final choice of the transition break size. If the defense-in-depth capability to mitigate breaks greater than the transition break size is maintained, a smaller choice of transition break size may be supportable.

DISCUSSION

Loss-of-coolant accidents (LOCAs) have been the focus of nuclear plant safety since the first commercial reactor designs. LOCAs can arise from many causes, and the current design basis requires the demonstration of the capability to mitigate a spectrum of break sizes up to the

DEGB of the largest pipe in the reactor coolant system. Since the Three-Mile Island accident and the earliest probabilistic risk assessments, it has been recognized that small-break LOCAs are more risk significant than large-break LOCAs (LBLOCAs). This has been reflected in operator training, procedures, etc., but it has not been fully reflected in the regulations.

Although the design-basis LBLOCA requirements have led to the development of robust safety systems, the burdens imposed by the design-basis requirement to deal with the DEGB of the largest pipe in the reactor coolant system are not commensurate with its risk importance, and the resulting requirements may have inhibited opportunities to optimize the system response for the entire range of challenges that must be met including those more likely to occur. For example, the current LBLOCA requirements result in rapid diesel start times. The testing necessary to demonstrate that these start times can be achieved increases wear on the diesel and reduces the reliability of the diesel in the case of more risk-important sequences that do not require such rapid start times.

A risk-informed 50.46 rule will be an enabling rule. It will not impose any specific changes that would be made in the design or operation of nuclear power plants. It will permit licensees to make changes that may decrease risk by optimizing system responses to accidents that are more likely to occur, and changes such as power uprates that will result in risk increases.

In a Staff Requirements Memorandum (SRM) dated July 1, 2004, the Commission approved the development of a proposed rule to risk-inform the requirements addressing LBLOCAs. The proposed rule was to use the initiating event frequencies from the expert elicitation process and other relevant information to guide the determination of an appropriate alternative break size. The staff was also to ensure that any changes to the plant or operating procedures would follow a change process consistent with Regulatory Guide (RG) 1.174. RG 1.174 permits only small increases in risk as long as it is reasonably assured that sufficient defense in depth and margins are maintained.

In our report, dated April 27, 2004, we concluded that the process and criteria in RG 1.174 are appropriate for evaluating the acceptability of changes proposed under a revised rule, but recommended explicit consideration of late release frequency (LRF) in addition to core damage frequency (CDF) and large early release frequency (LERF) to ensure that all of the safety objectives are addressed. The SRM and the proposed rule language posit a process, akin to the current 10 CFR 50.59 process, to permit licensees to make changes that result in "inconsequential" changes in risk without prior NRC review and approval. We agree that a process for making such changes is needed. The staff argues that the existing 10 CFR 50.59 process is not suitable, since it addresses design-basis issues, while the new process must address the acceptability of changes with respect to risk. Additional input on the need for a new change process can be obtained when a draft rule is issued for public comment.

In the proposed rule language, the staff introduces a transition break size (TBS). The TBS is chosen to ensure that the frequency of LOCAs corresponding to breaks larger than the TBS is less than $1x10^{-5}$/reactor-year. This frequency is consistent with the goal set in SECY-00-198, Attachment 1, "Framework for Risk-Informed Changes to the Technical Requirements of 10 CFR Part 50" for rare initiators and the criterion proposed for a vessel failure frequency due to pressurized thermal shock, when it is recognized that those are unmitigated events, and that a substantial mitigative capability will be maintained for LOCAs beyond the TBS.

For LOCAs corresponding to break sizes smaller than the TBS, the requirements are equivalent to those in the current 10 CFR 50.46. We agree that defense in depth should be maintained through the requirement that sufficient mitigating capability be available to prevent severe core damage (i.e., loss of coolable geometry) for breaks greater than the TBS up to the DEGB of largest pipe in the reactor coolant system. Because of the low frequency of such breaks, it should not be necessary to postulate a simultaneous loss of offsite power and single failure of the most critical component. Credit may be taken for operation of any equipment supported by appropriate availability data. Nominal operating conditions rather than technical specification limits, actual fuel burnup in decay heat predictions, and actual operating peaking factors can be used. Some increase in the degree of core damage beyond that implied in the current 10 CFR 50.46 should also be considered acceptable. The integrity of the reactor containment structure should be maintained using realistically calculated pressure, temperature, and containment capacity.

Because breaks with sizes greater than the TBS are not risk significant, and hence equipment needed to mitigate such breaks might be considered unimportant in 10 CFR 50.65(a)(4) assessments of acceptable configurations, the staff has included additional configuration control requirements to ensure the capability to mitigate such large breaks during all modes of operation when the reactor is critical. We agree that such configuration control requirements are appropriate.

The draft version of a proposed rule discussed with us proposes a TBS that is reactor specific and equivalent in area to a double-ended rupture of the largest pipe attached to the reactor coolant system. For a pressurized water reactor (PWR) this would correspond to surge, shutdown cooling, or safety injection lines that are typically 12-14 inches in diameter Schedule 160 pipe. For a boiling water reactor (BWR) these would be residual heat removal or feedwater lines, which are typically 20 inches in diameter Schedule 80 pipe.

The selection of the TBS requires estimates of LOCA frequencies as a function of break size. The most comprehensive assessment of this information is the expert opinion elicitation conducted by the Office of Nuclear Regulatory Research (RES). We believe that additional work needs to be done to complete the expert opinion elicitation and have issued a separate report on this matter, dated December 10, 2004. Hence, some of our judgments below on the implications of the elicitation must be considered preliminary.

The elicitation sought to develop LOCA frequency estimates for PWR and BWR piping and non-piping passive components. It focused on developing average values for the fleet of operating plants, and thus the uncertainty bounds represent bounds on these average values and not on LOCA frequency estimates for individual plants. Thus they are only applicable to plants that can demonstrate that they have no additional degradation mechanisms, no significant differences in the conditions that produce degradation, and no significant differences in their capability to detect degradation than is typical of most plants in the fleet.

The elicitation also did not consider the impact on the frequency of LBLOCAs of the power uprates that could likely result from a risk-informed 10 CFR 50.46. Such uprates could have substantial impacts on flow-assisted corrosion rates in secondary systems in PWRs. PWR power uprates are not likely to have a significant impact on primary system piping. The BWR feedwater piping is susceptible to flow-assisted corrosion. The potential impact of power uprates on LOCA frequency will have to be addressed as part of the licensing reviews of the uprates.

In its efforts to develop a new rule, the staff has considered other potential mechanisms that could cause pipe failure that were not explicitly considered in the expert elicitation process such as active system LOCAs, seismic loading, heavy load drops, and LOCA–induced waterhammer loading. No active system LOCAs were identified that would result in break sizes greater than about 4 inches. The staff concluded that heavy load drops would have little effect on the choice of the TBS. For seismic loads with magnitudes of occurrence of 1 $x10^{-5}$/yr, the staff has found that undegraded piping or piping with minor degradation has little likelihood of failure. More severely degraded piping could fail under such seismic loads, but the relatively low frequency of degradation in primary piping and the low frequency of the expected loading suggest that these will not have a significant impact on the choice of the TBS. RES is still performing some confirmatory research in this area.

Thus it appears that the expert elicitation has addressed the important potential contributors to the LBLOCA frequency. However, the choice of a TBS is strongly dependent on how the uncertainties in the elicitation are addressed.

For PWRs the break size (i.e., the equivalent diameter of the flow area) corresponding to a frequency of 1 x 10^{-5}/reactor-year from the expert opinion elicitation reported in Reference 3 ranges from 4-11 inches depending on the approaches used to aggregate and assess the expert opinions, whether the mean or 95th percentiles of the resulting distributions are used, and how the results are interpolated between the discrete break sizes in the elicitation.

The staff's choice of a break size corresponding to a double-ended break of the largest piping attached to the reactor coolant system appears to conservatively bound the range of values determined through the elicitation. The large disparity in size between the main reactor coolant system piping and the largest attached piping also provides an argument for the selection of the failure of the attached piping as the TBS. Although uncertainties in the elicitation could affect the choice of the TBS in the range of sizes up to the diameter of the attached piping, the physics of the failure processes give a very-high confidence in the low-failure probability of the main coolant piping. The staff notes that this choice for the TBS makes it very unlikely that any future reevaluations of the break frequency versus break size will result in the need for licensees to make any plant modifications as a result of implementing the revised 10 CFR 50.46 thus helping to ensure a more stable regulatory environment. It also bounds the flow areas associated with breaks of components such as bolted connections. Although these connections were considered in the elicitation, they are more likely to be affected by human errors and are thus perhaps subject to even greater uncertainty than the piping failure.

Based on our current understanding of the results of the expert opinion elicitation, it appears that the choice of the double ended rupture is overly conservative. Choosing the TBS as the diameter of the largest attached pipe (i.e., a single-ended rupture) would still bound the elicitation results and would be consistent with the argument that the failure of the main coolant piping is much more unlikely than the failure of the smaller attached piping. If the defense-in-depth capability to mitigate breaks greater than the TBS is maintained, a less conservative choice of TBS (e.g., one based on the mean value of the final "best estimate" distribution from the elicitation) may also be supportable.

A better quantitative understanding of the impact of the TBS on parameters, such as required diesel start time, is needed to help optimize the choice of a TBS to balance the defense in depth provided by the larger TBS in any new draft rule with the possible risk benefits of smaller break sizes. Since much of this may be plant specific and will require detailed plant information, it may have to be sought when a draft rule is issued for public comment. Any discussion of risk benefits should also include consideration of the impact of power uprates, which are the likely consequence of a risk-informed 10 CFR 50.46, on such risk benefits.

We would like to review any new draft rule before it is issued for public comment.

Sincerely,

/RA/

Mario V. Bonaca
Chairman

References:

1. Memorandum dated December 2, 2004, from Catherine Haney, Program Director, Policy and Rulemaking Program, NRR, to various members NRR, Subject: Office Concurrence on Proposed Rule - Risk Informed Changes to Loss-of-Coolant Accident Technical Requirements (Pre-Decisional For Internal ACRS Use Only).
2. Memorandum dated October 14, 2004, from Catherine Haney, Program Director, Policy and Rulemaking Program, NRR, to John T. Larkins, Executive Director, ACRS, Subject: Review of Risk-Informed 10 CFR 50.46 Proposed Rule Executive Summary and Draft Proposed Rule Language (Pre-Decisional For Internal ACRS Use Only).
3. Staff Requirement Memorandum dated July 1, 2004, from Annette L. Vietti-Cook, SECY, NRC to Luis A. Reyes, EDO, NRC Subject: 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.
4. Regulatory Guide 1.174, entitled, “An Approach for using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis,” Rev. 1, Office of Nuclear Regulatory Research, November 2002.
5. ACRS Report dated April 27, 2004, from Mario V. Bonaca, Chairman, ACRS to Nils J. Diaz, Chairman, NRC, Subject: 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.”
6. 10 CFR 50.59, “Changes, tests, and experiments.”
7. 10 CFR 50.65(a)(4), “Requirements for monitoring the effectiveness of maintenance at nuclear power plants.”
8. ACRS Report dated December 10, 2004, from Mario V. Bonaca, Chairman, ACRS to Nils J. Diaz, Chairman, NRC, Subject: Estimating Loss-of-Coolant Accident Frequencies through the Elicitation Process.
9. Memorandum dated November 4, 2004, from Michael E. Mayfield, Director, Division of Engineering Technology, RES, to John T. Larkins, Executive Director, ACRS, Subject: Transmittal of Draft NUREG on Passive System LOCA Frequency Development for use in Risk-Informed Revision of 10 CFR 50.46, Appendix K to Part 50, and GDC and Appendices (Pre-Decisional For Internal ACRS Use Only).