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

March 13, 2009

Mr. R. W. Borchardt
Executive Director for Operations
U. S. Nuclear Regulatory Commission
Washington, DC 20555-0001

SUBJECT: DRAFT FINAL RULE 10 CFR 50.61a, "ALTERNATE FRACTURE TOUGHNESS REQUIREMENTS FOR PROTECTION AGAINST PRESSURIZED THERMAL SHOCK EVENTS"

Dear Mr. Borchardt:

During the 560$^{th}$ meeting of the Advisory Committee on Reactor Safeguards (ACRS), March 5-7, 2009, we reviewed the Draft Final Rule 10 CFR 50.61a, "Alternate Fracture Toughness Requirements for Protection Against Pressurized Thermal Shock Events." Our Subcommittee on Materials, Metallurgy, and Reactor Fuels reviewed the technical basis for the rule on October 1, 2008, and March 4, 2009. During these reviews, we had the benefit of discussions with representatives of the NRC staff and its contractor Information Systems Laboratories, Inc. We also had the benefit of the documents referenced.

**CONCLUSIONS AND RECOMMENDATIONS**

1. The Draft Final Rule 10 CFR 50.61a, "Alternate Fracture Toughness Requirements for Protection Against Pressurized Thermal Shock Events," should be approved.

2. To aid in the implementation of the rule, the staff should undertake an effort to verify and document the capability of nondestructive examination (NDE) procedures that will be used to characterize the flaw distributions in reactor vessels.

3. An effort is needed to plan for the most effective use of surveillance samples to ensure that any deviations from the current understanding of embrittlement trends in reactor vessels will be identified in a timely manner.

**DISCUSSION**

Reactor pressure vessel steels undergo a transition from highly ductile behavior at elevated temperatures to brittle behavior at low temperatures. This change in behavior occurs abruptly over a narrow range of temperatures, and a transition temperature $RT_{NDT}$ can be defined to characterize the transition in fracture behavior. Under neutron irradiation, the transition temperature $RT_{NDT}$ increases, making the vessel susceptible to brittle fracture at higher temperatures. Some transients can lead to a rapid cooling of the inner surface of the reactor vessel and induce high thermal stresses. These stresses, together with the stresses due to internal pressure, can result in the initiation and growth of cracks through the wall of the vessel materials that have been embrittled by exposure to neutron irradiation. The pressurized thermal shock (PTS) rule, 10 CFR 50.61, was promulgated in 1985 to ensure the integrity of irradiation-embrittled reactor pressure vessels during such transients.

The PTS Reevaluation Project analyzed the susceptibility of reactor pressure vessels to PTS events. The results of this Project indicated that the current PTS rule is unduly conservative. We have reported on this Project in previous reports and concluded that it provides a sound technical basis for a revision of 10 CFR 50.61.

Based on the technical basis developed by the Office of Nuclear Regulatory Research (RES), the Office of Nuclear Reactor Regulation (NRR) developed a new, voluntary rule, 10 CFR 50.61a. Licensees may choose to use the new rule or continue to use the current 10 CFR 50.61. The new rule requires a plant-specific evaluation of the degree of vessel embrittlement and a demonstration that the flaw distribution is bounded by that used in the PTS Reevaluation Project. If the plant-specific flaw distribution is not bounded by this distribution, applicants must submit supplementary analyses showing that their vessel failure frequency is less than $1 \times 10^{-6}$ per reactor year. The rule contains correlations for the embrittlement of reactor vessel materials based on the best current understanding of the mechanisms of embrittlement and an extensive database. It also includes a requirement to assess surveillance data to determine if the embrittlement for a specific heat of material in a reactor vessel is consistent with that predicted by the correlation.

The findings of the PTS Reevaluation Project are based on a detailed study of the PTS challenges in three plants, a Westinghouse plant, a Combustion Engineering plant, and a Babcock & Wilcox plant. Despite the differences in the plants and in their operating procedures, the PTS challenges in the three plants are very similar. The study also concluded that medium- and large-break loss-of-coolant accidents (LOCAs) were the major contributors to the probability of vessel failures due to PTS events where the degree of embrittlement was high enough that the failure frequency began to approach $1 \times 10^{-6}$ per reactor year. A major issue in developing the rule is how to ensure that the results from the detailed study of three plants would be applicable to a broader range of plants.

The results for the three plants showed the severity of the challenge is largely independent of break size for breaks greater than about five inches. There is a physical explanation for these results that suggests this will be true in general. For breaks of this size, the fluid temperature in the downcomer decreases very rapidly to low temperatures. Because of the large thermal inertia of the vessel, the thermal stresses and, thus, the severity of the fracture challenge are controlled by the thermal diffusivity of the steel and the thickness of the vessel wall. Perturbations to the fluid cooldown rate controlled by break size, break location, system differences, and seasonal temperature variations do not significantly affect the severity of the challenge. Medium- and large-break LOCA frequencies are comparable for all pressurized water reactors (PWRs). Thus, both the likelihood and severity of the PTS challenges from these events at a specific plant should be comparable to those for the three plants in the detailed study.

To better understand the degree to which the results from the three-plant study could be generalized, the staff studied five additional plants to compare design and operational features of these plants potentially important for PTS challenges to the same features of the three plants in the detailed study. Based on these comparisons, judgments were made with regard to the appropriateness of treating the results from the detailed plant study as representative of the additional PWRs. The staff concluded that the results of the generalization study indicate that the likelihood and severity of the PTS challenges for the detailed analysis plants are representative of those for the five additional plants, and thus, by inference, the remainder of the PWRs.

The three-plant study did not address the effects of external events such as seismic events on the probability of vessel failure. Subsequently, the staff performed a bounding analysis of the effects of external events. Based on this analysis, the staff concluded that there is considerable assurance that the external event contribution to the probability of vessel failure due to PTS events is not significantly greater than the contribution from internal events, and it is more likely that the external event contribution to the probability of vessel failure is much less than the internal event contribution.

The study of the effects of external events and the generalization study are helpful in providing a technical basis to reach a conclusion that the results of the three-plant study can be extended to other operating plants. We concur with the staff's conclusion that its generalization study and recognition of the margins associated with expected operating lifetimes provide reasonable assurance that the results can be generalized to all plants without the need for a plant-specific evaluation of the frequency and severity of PTS challenges.

The flaw distribution used in the PTS Reevaluation Project is largely based on a detailed evaluation of the flaws in two pressure vessels from cancelled plants, one of them a boiling water reactor. The materials and fabrication processes for these vessels were representative, and the flaw distributions derived from these vessels should be broadly applicable. Nevertheless, we support the requirement of the rule that applicants perform inspections and analyses to verify that the flaw distributions in their vessels are consistent with those used in the detailed study of three plants.

The flaws of concern for PTS are smaller in size than those usually addressed in American Society of Mechanical Engineers Section XI inspections of the vessel. Based on discussions with the staff and industry NDE experts, the staff believes that current inspection capabilities are sufficient to characterize the flaw distributions of interest, but no documentation of this conclusion is available. This is not an obstacle to the issuance of the rule, but the availability of such documentation will be helpful in the implementation of the rule. We encourage the staff to pursue a study, either through RES or through a cooperative effort with industry, to verify and document this conclusion.

We support the staff's decision to include a requirement to assess surveillance data in the rule. Although there is an extensive database on embrittlement and an increased understanding of the mechanisms of embrittlement, surveillance specimens can provide an early warning that new and unexpected mechanisms of embrittlement are emerging. Based on the current interest in extending the life of operating reactors up to and beyond 60 years, it would seem prudent to review the availability and use of surveillance samples to ensure that best use is being made of this limited resource.

We commend the staff for outstanding technical work and thoroughness in the multidisciplinary effort that has lead to the development of the new rule, 10 CFR 50.61a. This rule will reduce unnecessary regulatory burden while still assuring adequate protection of the public health and safety and should be approved.

Sincerely,

***/RA/***

Mario V. Bonaca
Chairman

References:

1. Memorandum from Timothy A. Reed, Acting Branch Chief, Division of Policy and Rulemaking, NRR, to Edwin M. Hackett, Executive Director, ACRS, transmitting proposed rule, “Advisory Committee on Reactor Safeguards Review of Final Rule Regarding the Alternate Fracture Toughness Requirements for Protection Against Pressurized Thermal Shock Events (10 CFR 50.61a),” February 2, 2009 (ML090330688)

2. U. S. Nuclear Regulatory Commission, NUREG-1806, “Technical Basis for Revision of the Pressurized Thermal Shock (PTS) Screening Limit in the PTS Rule (10 CFR 50.61),” August 31, 2007 (ML072830076)

3. Letter from Graham B. Wallis, Chairman, ACRS, to Luis A. Reyes, Executive Director for Operations (EDO), “Pressurized Thermal Shock (PTS) Reevaluation Project: Technical Basis for Revision of the PTS Screening Criterion in the PTS Rule,” March 11, 2005 (ML050730177)

4. Letter from George E. Apostolakis, Chairman, ACRS, to William D. Travers, EDO, “Reevaluation of the Technical Basis for the Pressurized Thermal Shock Rule,” February 14, 2002 (ML020510265)

5. Letter from George E. Apostolakis, Chairman, ACRS, to William D. Travers, EDO, “Risk Metrics and Criteria for Reevaluating the Technical Basis of the Pressurized Thermal Shock Rule,” July 18, 2002 (ML022030522)

6. A. M. Kolaczkowski, D. Kelly, and D. W. Whitehead, Letter Report, “Estimate of External Events Contribution to Pressurized Thermal Shock (PTS) Risk,” December 14, 2004 (ML042880476)

7. D. Whitehead, A. Kolaczkowski, W. Arcieri, R. Beaton, M. Junge, M. Kirk, and T. Dickson, Letter Report, “Generalization of Plant-Specific Pressurized Thermal Shock (PTS) Risk Results to Additional Plants,” December 14, 2004 (ML042880482)

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Letter to R. W. Borchardt, Executive Director for Operations, NRC, from Mario V. Bonaca, Chairman, ACRS, dated March 13, 2009

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