March 14, 1996

Mr. James M. Taylor
Executive Director for Operations
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555-0001

Dear Mr. Taylor:

SUBJECT: RESOLUTION OF GENERIC SAFETY ISSUE 78, "MONITORING OF FATIGUE TRANSIENT LIMITS FOR THE REACTOR COOLANT SYSTEM"

During the 429th meeting of the Advisory Committee on Reactor Safeguards, March 7-9, 1996, we completed our deliberations on the resolution of the subject Generic Safety Issue that we started during our 424th meeting, September 7-8, 1995. We had the benefit of discussions with representatives of the NRC staff and of the documents referenced.

This Generic Safety Issue was originally developed to determine whether licensees need to perform transient monitoring to ensure compliance with requirements concerning fatigue failure. The transient monitoring concern was subsumed in the Fatigue Action Plan, which was reported as complete in SECY-95-245, "Completion of the Fatigue Action Plan."

The current scope of the Generic Safety Issue is focused on the evaluation of risk from fatigue failure. The staff completed a study that demonstrated that the risk from fatigue failure of the primary coolant pressure boundary components is very small. The analyses used in the study were based on the assumption that the probability of crack initiation by fatigue in a component subject to cyclic loads and the probability of crack propagation through the wall are independent. The product of these probabilities was used to calculate the change in core-damage frequency caused by fatigue failure of a component.

The analyses, as presented to us by the staff to demonstrate its conclusion, lacked sufficient detail to be convincing. Additional discussions with the staff demonstrated that more complete analyses using the PRAISE code have led to the same conclusion. The PRAISE analyses of the failure probability of primary system piping assumed that a distribution of cracks existed in a component and calculated the probabilities of crack propagation through the wall and failure. Parametric studies using the PRAISE code showed that the calculated probabilities of failure are small, even when very conservative loads and flaw-size distributions are assumed. The staff provided a careful quantification of uncertainty of fatigue crack initiation. We recommend such consideration of uncertainties in any future analyses regardless of the technical approach adopted.

We believe that the staff's conclusion concerning the risk significance of fatigue failure of reactor components is correct. Thus, we agree that this Generic Safety Issue is resolved.

Dr. Shack did not participate in the Committee's deliberations regarding this matter.

Sincerely,

/s/

T. S. Kress
Chairman

References:

1. Memorandum dated August 18, 1995, from Charles Serpan, Jr., NRC Office of Nuclear Regulatory Research, to John T. Larkins, ACRS Executive Director, Subject: Proposed Resolution of Generic Safety Issue 78, "Monitoring of Fatigue Transient Limits for the Reactor Coolant System"
2. SECY-95-245 dated September 25, 1995, from James M. Taylor, Executive Director for Operations, to the Commissioners, Subject: Completion of the Fatigue Action Plan
3. Memorandum dated October 27, 1995, from Jeff Keisler and Omesh Chopra, Argonne National Laboratory, to Craig Hrabal, NRC Office of Nuclear Regulatory Research, Subject: Uncertainty Estimates for the Probability of Fatigue Crack Initiation in Reactor Components, NUREG/CR-6335, ANL-95/15
4. U. S. Nuclear Regulatory Commission, NUREG/CR-6237, "Statistical Analysis of Fatigue Strain-Life Data for Carbon and Low-Alloy Steels," August 1994
5. U. S. Nuclear Regulatory Commission, NUREG/CR-6335, "Fatigue Strain-Life Behavior of Carbon and Low-Alloy Steels, Austenitic Stainless Steels, and Alloy 600 in LWR Environments," June 1995

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