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

November 17, 2004

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

SUBJECT: RESOLUTION OF CERTAIN ITEMS IDENTIFIED BY THE ACRS IN NUREG-1740, "VOLTAGE-BASED ALTERNATIVE REPAIR CRITERIA"

Dear Mr. Reyes:

Thank you for your letter of August 25, 2004, responding to our May 21, 2004, letter regarding the Resolution of Certain Items Identified by the ACRS in NUREG-1740, "Voltage-Based Alternative Repair Criteria."

We would like to contribute to the ongoing studies of steam generator tube behavior under severe accident conditions by offering further explanations of some of the points raised in our letter of May 21, 2004.

Countercurrent flow in the hot leg is an essential characteristic of severe accidents that may lead to thermally induced failure of steam generator tubes. Countercurrent flow is driven by buoyancy effects due to the difference in density between the steam in the upper plenum of the reactor vessel and the steam in the lower plenum of the steam generator. These densities are determined by the corresponding temperatures, which are outputs of an analysis of the behavior of the entire system. Therefore, the countercurrent flow rate must also be an output of this system analysis. We are uncertain how SCDAP/RELAP5 can model such phenomena and would like an explanation from the staff during a future meeting.

The staff has performed an excellent computational fluid dynamics (CFD) study of flows in the steam generator. To perform a similar study to guide development and validation of a model for countercurrent flow in the hot leg requires adequate modeling of the lower plenum of the steam generator and the upper plenum of the reactor vessel. These determine the boundary conditions at the ends of the hot leg where the buoyancy effects are created. It is not necessary to apply CFD to other parts of the system such as the vessel internals or the steam generator tubes. The problem is therefore simpler than is described in your response.

We agree that the $1/7^{th}$ scale tests provide an excellent basis for validating the modeling and theoretical representation of the phenomena which occur in various components and for developing a model of the entire system. However, since all of the thermal-hydraulic phenomena are not exactly scaled, the outputs from the $1/7^{th}$ scale tests, such as the ratio of heat transferred to the core and to the steam generator, are not directly transferable to full scale. They must be determined from a suitable model of the whole system, applied to the entire transient. The fraction of heat transferred to the steam generator is likely to be low at the start of the transient, however, if the steam generator is cooled at all and materials do not fail, the fraction will eventually tend asymptotically to unity, though by then the temperatures would probably be unrealistically high. How this ratio evolves during a transient must be predicted and not somehow used as an input to the analysis. Therefore, we reiterate the recommendation in our May 21, 2004, letter that the staff should improve the thermal-hydraulic analyses needed for these studies to enable a realistic prediction for the fraction of heat transferred to the steam generator, rather than estimating a value for this fraction based on the $1/7^{th}$ scale test results.

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

Sincerely,

/RA/

Mario V. Bonaca
Chairman

References:

1. Letter dated May 21, 2004, from Mario V. Bonaca, Chairman, ACRS, to William D. Travers, Executive Director for Operations, NRC, Subject: Resolution of Certain Items Identified by the Advisory Committee on Reactor Safeguards in NUREG-1740, "Voltage-Based Alternative Repair Criteria."

2. Letter dated August 25, 2004, from Luis A. Reyes, Executive Director for Operations, NRC, to Mario V. Bonaca, Chairman, ACRS, Subject: Resolution of Certain Items Identified by the Advisory Committee on Reactor Safeguards in NUREG-1740, "Voltage-Based Alternative Repair Criteria."