OFFICE OF THE SECRETARY
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To: Dyer, NRR

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AUTHOR:	Robert Leyse
AFFILIATION:	ID
ADDRESSEE:	Nils Diaz
SUBJECT:	PRM-50-78 comments on the petition for rulemaking
ACTION:	Appropriate
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DATE DUE: **DATE SIGNED:**

From: <Bobleyse@aol.com>
To: <Chairman@nrc.gov>
Date: Sun, Aug 8, 2004 6:37 PM
Subject: PRM-50-78 Posted 10/31/02

Dear Mr. Chairman:

Response to the above PRM is long overdue. Now, it turns out that the 2004 RELAP5/ATHENA International Users Seminar is being held in Sun Valley, ID during August 25-27, 2004, and I will be there. Please send some of your NRC staff to this meeting (partially funded by NRC) and have them look me up so that we can discuss the technology of fouling as well as the NRC delays in responding to PRM-50-78.

The attachment consists of items copied from the NRC web site. The first item is an abstract of PRM-50-78. The second item consists of excerpts from an ACRS letter to Travers. ACRS calls for better modeling of steam generator heat transfer. You should insist that fouling be included in those evaluations.

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CC: <Laurel.Hall@mail.house.gov>

PRM-50-78 Posted 10/31/02

About the Petition for Rulemaking

The Nuclear Regulatory Commission (NRC) has received and requests public comment on a petition for rulemaking filed by Robert H. Leyse. The petition has been docketed by the NRC and has been assigned Docket No. PRM-50-78. The petitioner is requesting that the NRC regulations that govern domestic licensing of production and utilization facilities be amended to address the impact of fouling on the performance of heat transfer surfaces throughout licensed nuclear power plants. The petitioner believes that the fouling of heat transfer surfaces is not adequately considered in the licensing and compliance inspections, testing programs, and computer codes for nuclear power facilities.

May 21, 2004

Dr. William D. Travers
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 Dr. Travers:

During the 512 meeting of the Advisory Committee on Reactor Safeguards, May 5-8, 2004, we completed our review of the progress made by the staff in resolving various steam generator (SG) tube integrity issues highlighted in our document NUREG-1740, "Voltage-Based Alternative Repair Criteria."

BACKGROUND

Item 3.4: Develop a better understanding of SG tube behavior under severe accident conditions. The staff proposes to close subtasks 3.4 a, b.1, c, e, and g.

CONCLUSIONS AND RECOMMENDATIONS

5. The studies of bypass scenarios due to thermally induced SG tube failures are still in progress. The staff should improve the thermal-hydraulic analyses needed for these studies to enable a realistic prediction for the fraction of heat that is transferred to the SG, rather than estimating a value for this fraction based on the 1/7 scale test results. The staff also needs to document the technical basis for its conclusion that the likelihood of cold-leg loop seal clearing is sufficiently low that the countercurrent flow situation is the appropriate model for these scenarios.

DISCUSSION

It is apparent that the amount of heat which goes to the SG depends on the convective flow and heat transfer processes between the reactor core and the SG and it may not be appropriate to assume it as an input variable based on the 1/7scale test, as was done in the SCDAP/RELAP analysis. Several features of the 1/7 scale tests (such as the method of cooling the SG tubes) may be atypical of the full-scale plants. The CFD work should be extended to include sufficient parts of the upper plenum and core flow process to permit a calculation of the hot-leg entrance conditions. This would permit the mixing process surrounding the cold plume emerging from the hot leg and descending into the reactor pressure vessel to be modeled in much the same way as the hot plume was modeled by the CFD calculation in the SG inlet plenum.

Despite the large uncertainties in the predicted failure times of the SG tubes and other reactor coolant system (RCS) components, the staff assumed that it will be able to determine the relative failure times with sufficient accuracy to permit conclusions about whether operation with flawed SG tubes will have a significant impact on the likelihood of SG bypass accidents. The uncertainty in the calculated thermal response of the primary system components under severe accident conditions may be too large to use such results to determine whether the primary system component or the SG tubing fails first. The primary system conditions (i.e., high-temperature natural circulation) involve phenomena beyond the prediction capability of a one-dimensional thermal-hydraulics system code. In addition, the uncertainties in the associated heat transfer and friction correlations under these conditions are significant. Even when the calculation is augmented with CFD modeling, the uncertainty of the predicted outcome may still be very large. The staff should continue its efforts to conduct uncertainty analyses to determine the probability of containment bypass. The staff also needs to document the technical basis for its conclusion that the likelihood of cold-leg loop seal clearing is sufficiently low that the countercurrent flow situation is the appropriate model for these scenarios.