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

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 512th 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." We also heard presentations by and held discussions with the staff and its contractors regarding this matter during our 509th meeting, February 5-7, 2004. In addition, our Subcommittees on Materials and Metallurgy and on Thermal-Hydraulic Phenomena met with the representatives of the staff and their contractors on February 3-4, 2004, to review this matter in detail. We had the benefit of the documents referenced.

BACKGROUND

The Steam Generator Action Plan (SGAP) to resolve SG tube integrity issues includes the following items:

- Item 3.1: Investigate the effects of depressurization during a main steamline break (MSLB) on SG tube integrity. The staff proposes to close this item.
- Item 3.2: Complete investigation of jet penetration of adjacent tubes. The staff proposes to close this item.
- Item 3.3: Develop experimental information on source term attenuation on the secondary side of steam generators (ARTIST tests).
- 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.
- Item 3.5: Develop improved methods of assessing risk associated with SG tubes under accident conditions. The staff proposes to close subtasks 3.5a and b.
- Item 3.6: Assess the technical basis for improving the probability of crack detection in SG tubes. The staff proposes to close this item.
- Item 3.7: Assess the need for better leakage correlations as a function of voltage for 7/8" SG tubes. The staff proposes to close this item.

- Item 3.8: Monitor the predictions of flaw growth for systematic deviations from expectations. The staff proposes to close this item.
- Item 3.9: Assess the need for a more technically defensible treatment of radionuclide release to be used in safety analyses of design-basis events. The staff proposes to close this item.
- Item 3.10: Develop a better mechanistic understanding of tube cracking processes.
- Item 3.11: Resolve Generic Safety Issue 163, ‘Multiple Steam Generator Tube Leakage’.

CONCLUSIONS AND RECOMMENDATIONS

1. The analyses of the effects of depressurization during a MSLB on tube integrity have been completed and item 3.1 is appropriately closed out. However we recommend that, as a confirmatory measure, a review be performed of the U.S. industry SG tube pullout data and the associated extent of tube locking at tube support plates (TSPs) in degraded SGs.

2. The probability of jet impingement damaging adjacent tubes is negligibly small. Item 3.2 should be closed as proposed by the staff.

3. The staff has developed a technically defensible description of the probability of detection of a flawed tube as a function of flaw size. Item 3.6 should be closed. The continued use of the current constant flaw detection probability should be reexamined to have more realism in the evaluation.

4. The existing correlation of leakage with eddy current voltage for 7/8" diameter SG tubes is not accurate enough to be used. We agree with the staff that the choice of a 2 volt limit for the 7/8" diameter SG tubes is conservative with respect to the risk posed. Item 3.7 should be closed. We recommend that qualified data continue to be collected and analyzed in order to develop an improved correlation.

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 1/7th 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.

6. The staff continues to treat iodine spiking in a conservative, empirical fashion. We recommend that the staff develop a mechanistic understanding of iodine spiking so that analyses reflect current plant operations and the capabilities of modern fuel rods.

7. Item 3.8 (predictions of flaw growth) should not be closed until progress has been made on developing the cracking model under item 3.10.

DISCUSSION

We had extensive discussions with the staff on the differing professional opinion (DPO) concern that the blowdown forces and movement of TSP caused by the SG depressurization following a MSLB could cause cracks to form, grow, and unplug, leading to much higher primary-to-secondary side leakage than was initially assumed by the staff.

The staff presented the results of the tests and finite-element analyses which were performed to evaluate the stability of cracks in the SG tubes subjected to stresses due to the motion of the TSPs. This work supports the staff's contention that the bending stresses in the SG tubes are small compared to the axial stresses, and the effect of bending loads on the leakage from flawed SG tubes is also small.

The staff calculated hydraulic loadings on the internal components of a SG following a MSLB for use in the structural evaluation of the stability of cracks subjected to such loads. The loading calculations were performed using a TRACE three-dimensional model of the SG. The TRACE code's capability to predict an acoustically dominated thermal-hydraulic transient, such as that caused by a MSLB, was verified by comparing TRACE predictions to experimental data.

The tensile stresses in the SG tubes locked at the TSP junctions were also calculated for varying numbers of locked SG tubes. The staff concluded that even if a few percent of the SG tubes are locked at the TSP, the dynamic loads associated with a MSLB will have little impact on the integrity of the SG tubes, unless extensive circumferential cracking is present.

Results of an extensive study on the SG tube pullout forces at a French nuclear plant (Dampierre) strongly support the conclusion that SGs with drilled hole carbon steel TSPs will have enough SG tubes locked at the TSP and that the dynamic loads will be of little concern. The staff recognizes that its conclusions would be strengthened by a more extensive review of industry data, especially from the U.S. pressurized water reactor plants, on the expected number of locked SG tubes in degraded SGs and on the SG tube pullout stress data. However, the staff already requires that licensees who wish to implement an alternate repair criteria that go beyond the Generic Letter (GL) 95-05 limits expand tubes to provide assurance that TSP motion will not uncover flaws. The voltage limits in GL 95-05, the low likelihood of the uncovering of the TSP region at the lower TSPs where cracking is likely, and the low probability of circumferential cracking in the TSP region lead to the conclusion that the staff's present positions are reasonably conservative. However, since the staff's conclusions regarding MSLB-induced leakage is based heavily on the Dampierre data, we believe that a review of U.S. industry data should be considered as a confirmatory measure.

In addressing SGAP item 3.2, the staff developed analytical models and performed erosion tests to determine the effect of fluid jets on thinning of SG tubes. The staff performed tests that provide a conservative simulation of a steamline break to determine the susceptibility of steam generator tubes to erosive damage from impacting steam jets from adjacent flawed tubes. We agree with the staff's conclusion that the probability of damage progression via jet cutting of adjacent SG tubes is low and need not be considered in accident analysis.

Under SGAP item 3.6, the staff presented a detailed analysis of the probability of detection (POD) of SG tube cracks. The staff concluded that the continued use of a POD value of 0.6 was conservative for eddy current voltages in the 1-2 volts range where the observed POD value approaches 0.9. We believe that the analysis of this topic and the supporting data are sufficient to close out this item. However, the decision to continue the use of a POD value of 0.6, rather than using the developed analytical capability that recognizes the entire POD vs. crack depth correlation, should be reexamined with a view to making the evaluation more realistic.

It is apparent that some of the technical issues raised in NUREG-1740 may not be resolved. For instance, an improved correlation between eddy current voltage and leakage for 7/8" diameter SG tubes has not been achieved in SGAP item 3.7. Although there is a qualitative understanding of the reasons for the unacceptable scatter in the relationship, it is apparent that the accumulation of further unqualified data will not improve the correlation factor. We agree with the staff that the choice of a 2 volt limit for the 7/8" diameter SG tubes is conservative with respect to the risk posed. Item 3.7 should be closed. We also believe that in order to get relief on this 2 volt limit, a qualified correlation of burst pressure and leak rate vs. flaw size needs to be developed.

The staff has performed thermal-hydraulic analyses of the core, upper plenum, hot leg, SG plenum, and SG tubes during a severe accident scenario. One output of the thermal-hydraulic analyses is the component temperatures that result from the convective heating. Hot gases from the core rise into the upper plenum and flow toward the SG inlet plenum along the top portion of the hot leg. Cooler gases from the SG return by countercurrent flow along the bottom of the hot leg to the reactor vessel. This may lead to thermally induced failure of the hot leg or the SG tubes.

The analyses of this complex flow process have used an adjusted one-dimensional thermal-hydraulic system code. The code was calibrated with the 1/7th scale SG test data, to determine the flow to the SG and set the boundary conditions for a computational fluid dynamics (CFD) simulation of the mixing of hot and cold gases in the SG inlet plenum.

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 $1/7^{th}$ scale test, as was done in the SCDAP/RELAP analysis. Several features of the $1/7^{th}$ 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.

The staff has not accepted our recommendation to develop a mechanistic understanding of the iodine spiking issue. The staff continues to use a conservative, empirical estimate of iodine spiking for accident consequence analyses. This estimate is based on historical data that may not reflect current practices in plant operations or the capabilities of modern fuels to prevent coolant contamination. We again encourage the staff to take advantage of iodine studies available in the literature and develop a mechanistic understanding of the phenomenon.

In our meetings we did not discuss in detail the progress in certain SGAP tasks such as the modeling of time-dependent degradation (e.g., stress corrosion cracking) of tubes (item 3.10) and the development of source term information in the ARTIST program (item 3.3).

We look forward to continuing discussions with the staff on this important project.

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

Sincerely,

/RA/

Mario V. Bonaca
Chairman

References:

1. Memorandum from William D. Travers, Executive Director for Operations, NRC, to John T. Larkins, Executive Director, ACRS, Subject: Differing Professional Opinion on Steam Generator Tube Integrity Issues, July 20, 2000.
2. ACRS Report dated September 11, 2000, from Dana A. Powers, Chairman, ACRS Ad Hoc Subcommittee, to William D. Travers, Executive Director for Operations, NRC, Subject: Differing Professional Opinion on Steam Generator Tube Integrity.
3. ACRS Report dated February 1, 2001, from Dana A. Powers, Chairman, ACRS Ad Hoc Subcommittee, to William D. Travers, Executive Director for Operations, NRC, Subject: Differing Professional Opinion on Steam Generator Tube Integrity.
4. U.S. Nuclear Regulatory Commission, NUREG-1740,"Voltage-Based Alternative Repair Criteria," Advisory Committee on Reactor Safeguards, March 2001.
5. ACRS Report dated May 15, 1995, from T. S. Kress, Chairman, ACRS, to James M. Taylor, Executive Director for Operations, NRC, Subject: Proposed Final Generic Letter 95-05, "Voltage-Based repair Criteria for Westinghouse Steam Generator Tubes."
6. ACRS Report dated October 18, 2001, from George E. Apostolakis, Chairman, ACRS, to Richard A. Meserve, Chairman, NRC, Subject: NRC Action Plan to Address the Differing Professional Opinion Issues on Steam Generator Tube Integrity.
7. The Committee also reviewed the following documents that the staff provided to ACRS in support of the February 2004 Briefings:

SGAP Item No.	Document Title/Date	Public/Non-Public
3.1.a and b	**Memo to Mayfield from Eltawila, 12/30/02, "Calculation of Steam Generator Tube Support Plate and Tube Loads Following a MSLB or FWLB Using TRAC-M;"**	**Non-Public**
	Att: RES Report SMSAB-02-05, 9/02	**Non-Public**
3.1.d thru h	**Memo from Mayfield to Strosnider, 12/26/02, "Closure of Steam Generator Action Plan Items 3.1d) to h);"**	**Non-Public**
	Att: NUREG/CR-XXXX, "Sensitivity Studies of Failure of Steam Generator Tubes During Main Steam Line Break and Other Secondary Side Depressurization Events"	**Non-Public (Proprietary)**
3.1.i	**Memo from Mayfield to Strosnider, 7/14/03, "Closure of Steam Generator Action Plan Item 3.1i);**	**Non-Public**
	Att: Technical Letter Report: Tests and Analysis Of Failure of Degraded Tubes Under Internal Pressure and Bending Loading"	**Non-Public**
3.2	**Memo from Mayfield to Strosnider, 7/9/02, "Closure of Steam Generator Action Plan Items 3.2 and 3.6;"**	**Public**
	Att: NUREG/CR-6756, "Analysis of Potential for Jet-Impingement Erosion from Leaking Steam Generator Tubes during Severe Accidents;"	**Public**
	Att: NUREG/CR-6774, "Validation of Failure And Leak-Rate Correlation for Stress Corrosion Cracks in Steam Generator Tubes"	**Public**
3.4.a	**Memo from King to Zimmerman, 9/28/01, "Completion of Subtask Milestone in Steam Generator Action Plan;"**	**Public**
	Attached report: ISL-NSAD-NRC-01-004	**Non-Public**
3.4.c	**ISL-NSAD-TR-02-03, "Tube-to-Tube Temperature Variations During the Station Blackout Event" (Draft), 8/02**	**Non-Public**

SGAP Item No.	Document Title/Date	Public/Non-Public
3.4.e.1	Memo from Rosenthal to Barrett, Wermiel, Banerjee, A Completion of Preliminary Milestone from Steam Generator Action Plan, 8/31/01	Non-Public
	Att: NUREG-1781, "CFD Analysis of 1/7th Scale Steam Generator Inlet Plenum Mixing During a PWR Degraded Core Accident," 10/03	Public
3.4.e.2	Draft Report, "CFD Prediction of Full-Scale Steam Generator Inlet Plenum Mixing for the Evaluation of Scale Effect," 3/02	Non-Public
3.4.e.3	Memo from Eltawila to Holahan, "Preliminary Results from SGAP Item 3.4.e.3 Related to the CDF Evaluation of Inlet Plenum Mixing," 2/25/03	Non-Public
3.5.a	Memo from Cunningham to Chokshi and Rosenthal, "Transmission of a Proposed Framework for Analysis of Severe Accident Induced Steam Generator Tube Ruptures," 4/1/02	Non-Public
3.5.b	Memo from Newberry to Strosnider, "Closure of Steam Generator Action Plan Items 3.5(b) and 3.5(c), "Severe Accident Induced-Steam Generator Tube Rupture (SAI-SGTR) Methodology Report," 6/30/03 (Note: Contrary to this memo item 3.5.c is not yet closed)	Public
	Draft Report, "Methodology for Assessing Severe Accident-Induced Steam Generator Tube Rupture," 6/03	Public
3.6	Memo from Mayfield to Strosnider, 7/9/02 "Closure of Steam Generator Action Plan Items 3.2 and 3.6;"	Public
	Att: NUREG/CR-6785, "Evaluation of Eddy Current Reliability from Steam Generator Mock-Up Round-Robin," 9/02	Public
3.7	Memo from Barrett to Sheron and Borchardt, "Steam Generator Action Plan - Completion of Item Number 3.7 (TAC No. MB7216)," 4/25/03;	Public
	Letter from Bateman to Marion (NEI), "Exclusion of French Data from the Steam Generator Degradation Specific Management Database," 10/8/02	Public

SGAP Item No.	Document Title/Date	Public/Non-Public
3.8	**Memo from Strosnider to Sheron and Borchardt, “Steam Generator Action Plan - Completion of Item Number 3.8 (TAC No. MB0258),” 1/3/02**	**Public**
3.9	**Draft Writeup, “ASGAP Item 3.9 - Iodine Spiking”**	**Non-Public**