August 25, 2004

Dr. Mario V. Bonaca, Chairman
Advisory Committee on Reactor Safeguards
U.S. Nuclear Regulatory Commission
Washington, DC 20555-0001

SUBJECT: RESOLUTION OF CERTAIN ITEMS IDENTIFIED BY THE ADVISORY COMMITTEE ON REACTOR SAFEGUARDS IN NUREG-1740, "VOLTAGE-BASED ALTERNATIVE REPAIR CRITERIA"

Dear Dr. Bonaca:

The staff of the U.S. Nuclear Regulatory Commission (NRC) sincerely appreciates the comprehensive reviews by the Advisory Committee on Reactor Safeguards (ACRS) and its Subcommittees on Materials and Metallurgy and Thermal-Hydraulic Phenomena concerning the staff's progress in resolving various steam generator tube integrity issues that the ACRS highlighted in NUREG-1740, "Voltage-Based Alternative Tube Repair Criteria," dated March 2001. As you know, the staff is currently addressing these issues through the NRC's Steam Generator Action Plan (SGAP).

Your letter dated May 21, 2004, indicated that the ACRS agrees with the staff's closure of SGAP items 3.1, 3.2, 3.6, and 3.7. You also indicated that the NRC should not close Item 3.8, which relates to developing a program to monitor the prediction of flaw growth, until progress has been made on developing a better understanding of the cracking process under SGAP Item 3.10. As indicated in the closeout memorandum for Item 3.8 [Accession #ML020070081 in the NRC's Agencywide Documents Access and Management System], the staff has implemented a program to continue monitoring operating experience as it relates to flaw growth. In addition, the staff will continue to monitor research associated with SGAP Item 3.10. Therefore, the staff considers Item 3.8 closed.

The ACRS comments concerning the staff's thermal-hydraulic work indicated that the work related to computational fluid dynamics (CFD) should be extended to include the vessel and hot leg flows to predict the fraction of heat that is transferred to the steam generator. However, given the availability of the one-seventh scale test data developed by the Electric Power Research Institute in collaboration with the NRC, the staff believes it is not reasonable to use CFD to estimate the fraction of heat that is transferred to the steam generator because the tests provide the best source of data concerning these phenomena.

The ACRS also recommended that the staff should develop a mechanistic understanding for the treatment of the iodine spiking issue. To address that recommendation, the staff has identified a new generic issue, Number 197, "Iodine Spiking Phenomena."

Finally, in addition to the ACRS recommendations related to closure of the various SGAP items, your letter offered additional ACRS conclusions and recommendations. The attachment to this letter discusses the staff's position concerning each of those conclusions and recommendations.

Again, the staff wishes to thank the ACRS for its comprehensive review of the staff's progress toward completing the SGAP items, its support for the closure of several SGAP items, and its further insights discussed in your letter of May 21, 2004. The staff will provide future briefings to ensure that the ACRS remains well-informed concerning research and other activities related to open SGAP items, as well as items that were not discussed in detail during the meetings in February 2004.

Sincerely,

/RA/

Luis A. Reyes
Executive Director
for Operations

Attachment: Staff Responses to ACRS
Conclusions and Recommendations

cc w/attachment:
Chairman Diaz
Commissioner McGaffigan, Jr.
Commissioner Merrifield
SECY

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STAFF RESPONSES
TO ACRS CONCLUSIONS AND RECOMMENDATIONS

This attachment discusses the staff position of the U.S. Nuclear Regulatory Commission (NRC) regarding the seven conclusions and recommendations (shown herein in italics) that the Advisory Committee on Reactor Safeguards (ACRS) listed in its letter, dated May 21, 2004, from Dr. Mario V. Bonaca to Dr. William D. Travers.

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."*

The NRC does not have an established regulatory requirement for the utilities to submit such data. However, the staff did obtain data through the NRC's research program on steam generator tube integrity. Specifically, the data relate to 12 tubes that were pulled through 4 tube support plate intersections of the 1 D steam generator, which was removed from service at the William B. McGuire Nuclear Station. The mean pullout force per intersection of the McGuire tubes was 2,725 lbs. (1,236 kg), which is comparable to the mean pullout force of 3,120 lbs. (1,415 kg) per intersection for the 23 tubes at the French Dampierre plant. In addition, the staff obtained recent data from U.S. plants, which showed that the related pullout forces are consistent with the McGuire and Dampierre data. Consequently, the staff's conclusions remain unchanged, as they relate to tube integrity under MSLB conditions.

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

We agree; the staff closed Item 3.2 on December 31, 2001 [See Accession #ML021910311 in the NRC's Agencywide Documents Access and Management System (ADAMS).]

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.*"

The staff specified a constant "probability of detection" of 0.6 for plants implementing Generic Letter (GL) 95-05, "Voltage-Based Repair Criteria for Westinghouse Steam Generator Tubes Affected by Outside Diameter Stress Corrosion Cracking," dated August 3, 1995. For other mechanisms and at other locations, the NRC does not specify the probability of detection to be used in assessing tube integrity.

For plants implementing the voltage-based tube repair criteria, if a licensee elects to use a different probability of detection, the staff will review it based on its technical merits. The information developed under Item 3.6 of the NRC's Steam Generator Action Plan (SGAP) provides one source of input for these reviews. It is important to note, however, that the constant "probability of detection" of 0.6 specified in GL 95-05 is intended to ensure conservative projections of the condition of the steam generator tubes (that is, the value is an adjustment to account for the initiation of new flaws and other potential non-conservatisms in the methodology, as well as the probability of detection). The staff recently approved a

one-cycle relaxation of the constant “probability of detection” used at Diablo Canyon Nuclear Power Plant, Unit 2, and is currently reviewing a permanent change to the constant “probability of detection” for Diablo Canyon Units 1 and 2.

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.*”

4.a. “*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.*”

We agree with the ACRS conclusion that the methodology used to assess the consequences of steam generator tube leakage is conservative, and the staff closed SGAP Item 3.7 on April 26, 2003. In closing this item, the staff reexamined the status of the leak rate correlation in the then current database for ⅞-inch (2.2-cm) tubes and attempted to determine reasons for the lack of correlation. The close-out memorandum dated April 25, 2003 (ADAMS Accession #ML031150674), provides details concerning the staff’s findings and discusses potential reasons for the lack of correlation, and why the methodology used to predict leak rates should yield a conservative estimate of leakage. Specifically, the staff concluded that the evaluation of leakage data does not lead to a simple, conclusive explanation for the poor correlation in the ⅞-inch (2.2-cm) tube database when compared to the database for ¾-inch (1.9-cm) tubes. Consistent with the close-out memorandum and the ACRS recommendation, the staff will continue to collect and analyze the data from additional pulled tubes; however, there is no assurance that this will result in an improved correlation. Regardless of the parameter to which the leak rate data are correlated (flaw length, voltage, etc.), such correlations have a large amount of scatter.

Moreover, the industry’s database of leak rates from flaws in ⅞-inch (2.2-cm) diameter tubes is derived from pressure tests of degraded tubes that have been removed from operating steam generators or have been cracked in the laboratory. The data from these specimens are real and valid. Nonetheless, the staff agrees with the ACRS that correlating the leak rate data for ⅞-inch (2.2-cm) tubes to voltage shows significant scatter. Because of this scatter and the statistical requirements that GL 95-05 specifies to determine when a correlation can be assumed to exist, licensees are currently performing their leakage assessments based on the assumption that a correlation of leak rate to voltage does ***not*** exist (for the fraction of simulations in which the statistical criteria are not satisfied). The amount of leakage from a steam generator is then taken as the upper 95th percentile value (at 95-percent confidence) rather than the mean value.

The staff is currently reviewing a licensee request to relax the existing 2.0-volt limit for Diablo Canyon Units 1 and 2. That request is based, in part, on locking the tube support plates in place. For the reasons discussed above and in the close-out memorandum for SGAP Item 3.7, the staff has not requested (and does not plan to request) that the licensee develop a “qualified 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.*"

The ACRS recommends using a computational fluid dynamics (CFD) simulation of the vessel, hot leg, and steam generator flows to estimate the hot leg flow and energy transfer to the steam generators. By contrast, the staff believes that the one-seventh scale experiments provide the best source of information concerning the amount of heat that is transferred to the steam generators and the hot leg flows. The staff does not have confidence in the ability of CFD to estimate these values because of the physical scale of the problem and the complexity of the vessel internals.

The staff is currently using the SCDAP/RELAP5 code to predict system response and the natural circulation flows during a severe accident scenario. These predictions provide the pressure, temperature, and heat transfer boundary conditions that are subsequently applied in thermal-structural analyses of the reactor system components and steam generator tubes. Flow coefficients are adjusted in the one-dimensional SCDAP/RELAP5 system model to ensure that the overall system behavior is consistent with a set of one-seventh scale experiments. The staff is also using CFD to develop detailed predictions of specific aspects of the flow phenomena, such as the inlet plenum mixing and heat transfer coefficients where the surge line meets the hot leg.

The staff agrees with the ACRS that the vessel, hot leg, and steam generator natural circulation flows are coupled. The vessel flows, however, are very difficult to predict in detail using CFD. A detailed model of the vessel would require approximately 10^9 computational cells, which is well beyond our current capacity. Consequently, a practical model of the vessel would rely on gross simplifications of the geometry and user-defined flow loss coefficients. Validation with data would be required, and any solution would be dependent on user inputs.

By contrast, the one-seventh scale experiments that the Electric Power Research Institute (EPRI) sponsored in collaboration with the NRC include the coupled vessel, hot leg, and steam generator flows. The test reports describe geometrical design and flow loss considerations for these components, which are designed to ensure that the natural circulation flows accurately represent the prototype behavior. The model core is represented by 104 fuel assemblies, and axial and lateral flow resistances are established with slots, tabs, and orifices designed for consistency with the prototype. Each channel has an electrically heated rod to simulate the decay power. Sulfur Hexafluoride (SF6), with its relatively high density and thermal expansion coefficient and low viscosity, is used as the working fluid. The resulting governing parameters have good similitude with the prototype reactor flow conditions. Core simulation design and scaling are described in the EPRI test report entitled "Natural Circulation Experiments for PWR Degraded Core Accidents" (EPRI NP-63224-L), dated July 1989, while the steam generator simulation design and scaling are described in the EPRI test report entitled "Natural Circulation Experiments for PWR High-Pressure Accidents" (EPRI TR-102815), dated August 1993.

The Idaho National Engineering Laboratory (INEL) reviewed the scaling of the one-seventh scale tests and concluded that the high-pressure SF6 tests provide the closest representation of the natural circulation behavior in the full-scale plant. (See Appendix B to NUREG/CR-6285, "Severe Accident Natural Circulation Studies at the INEL," dated February 1995.) In addition, Dr. Peter Griffith of the Massachusetts Institute of Technology (MIT) reviewed the one-seventh scale test report used to establish the SCDAP/RELAP5 model and commented that "this report provides all the modeling guidance that is needed to solve this phase of the severe accident problem." (letter from Dr. Griffith to David Bessette (NRC), dated March 29, 2002.)

The staff plans to continue use of the results from the transient one-seventh scale tests with a dry secondary side as a basis for developing the SCDAP/RELAP5 model. The impact of uncertainties in the amount of heat transferred from the vessel to the steam generator is being considered through sensitivity studies. The staff is using CFD to make direct predictions of the hot leg flow, but these models are not yet sufficiently mature to form a basis for discounting the experimental results.

The staff will also formally document the calculations that support the SCDAP/RELAP5 predictions that the loop seals do not clear.

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.*"

To address the ACRS concern regarding the conservative, empirical fashion in which iodine spiking is being treated in accident consequence analyses, the staff has identified a new generic issue, Number 197, "Iodine Spiking Phenomena". In accordance with Management Directive 6.4, "Generic Issues Program," the next step will be to screen the issue. During this step, the staff will complete a technical analysis of the issue for consideration by a panel of experts, who will determine whether the issue has any safety significance that warrants further work. The staff will send the ACRS a copy of its screening analysis of Generic Issue 197.

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.*"

As indicated in the closeout memorandum for Item 3.8 (ADAMS Accession #ML020070081), the staff has implemented a program to continue monitoring operating experience as it relates to flaw growth. In addition, the staff will continue to monitor the results from the NRC and industry research associated with SGAP Item 3.10. If operating experience and/or research indicate(s) that the approach used to predict flaw growth is not sufficient to ensure tube integrity, the staff will take corrective action. Nonetheless, because monitoring operating experience and research is part of the staff's normal procedures, we plan to close SGAP Item 3.8.