April 25, 2003

MEMORANDUM TO: Brian W. Sheron, Associate Director
for Project Licensing and Technical Analysis
Office of Nuclear Reactor Regulation

R. William Borchardt, Associate Director
for Inspection and Programs
Office of Nuclear Reactor Regulation

FROM: Richard J. Barrett, Director ***/RA/***
Division of Engineering
Office of Nuclear Reactor Regulation

SUBJECT: STEAM GENERATOR ACTION PLAN - COMPLETION OF ITEM NUMBER 3.7 (TAC NO. MB7216)

The purpose of this memorandum is to close out Item 3.7 in the Steam Generator Action Plan, entitled, "Assess the need for better leakage correlations as a function of voltage for 7/8-inch steam generator tubes."

In a letter dated July 20, 2000, the Executive Director for Operations requested the assistance of the Advisory Committee on Reactor Safeguards (ACRS) in reviewing issues raised in a differing professional opinion regarding steam generator tube integrity. In February 2001, an Ad Hoc Subcommittee of the ACRS published the results of its review of the voltage-based alternate repair criteria for steam generator tubes in NUREG-1740, "Voltage-Based Alternative Repair Criteria." In NUREG-1740, the Ad Hoc subcommittee concluded that the leakage correlation for the 7/8-inch diameter tubes was poor. On May 11, 2001, the NRC staff amended its steam generator action plan (ADAMS ML011300073) to incorporate the issues identified in NUREG-1740. The leak rate correlation issue was incorporated in the Steam Generator Action Plan as Item 3.7.

In a letter to the NRC Chairman dated October 18, 2001, the ACRS reiterated several of its concerns including the need to investigate the reason for the poor correlation between the leak rate and voltage for the 7/8-inch diameter steam generator tubes. In a letter dated January 18, 2002, the Executive Director for Operations provided the ACRS a preliminary response to the leakage correlation issue (ADAMS ML013300241).

As discussed in the attached, the Materials and Chemical Engineering Branch staff concludes that the evaluation of the leakage data does not lead to a simple conclusive explanation for the poor correlation in the 7/8-inch tube database when compared to the 3/4-inch tube database.

Attachment: As stated

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Nonetheless, the staff concludes that the methodology used by the industry and staff in assessing leakage from flaws at the tube support plates in the 7/8-inch diameter tubes is acceptable because: (1) NRC's Generic Letter 95-05 specifies necessary actions in the leak rate calculations when the correlation is weak and it specifies how to account for the uncertainty in the correlation; and (2) the overall methodology for assessing the consequences of steam generator tube leakage is conservative.

These conclusions are consistent with the conclusions that the staff provided to the ACRS in the January 18, 2002, letter. The staff considers that the leakage correlation issue as stated in Action Plan Item 3.7 is adequately addressed and is, therefore, closed. The staff will, however, continue to assess the leakage correlation as more data are added to the database.

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ASSESSMENT OF LEAKAGE VS. VOLTAGE CORRELATION OF 7/8-INCH STEAM GENERATOR TUBE DATABASE STEAM GENERATOR ACTION PLAN

1.0 INTRODUCTION

The purpose of this assessment is to close out Item 3.7 in the Steam Generator Action Plan, entitled, "Assess the need for better leakage correlations as a function of voltage for 7/8-inch steam generator tubes."

In a letter dated July 20, 2000, the Executive Director for Operations requested the assistance of the Advisory Committee on Reactor Safeguards (ACRS) in reviewing issues raised in a differing professional opinion regarding steam generator tube integrity. In February 2001, an Ad Hoc Subcommittee of the ACRS published the results of its review of the voltage-based alternate repair criteria for steam generator tubes in NUREG-1740, "Voltage-Based Alternative Repair Criteria." In NUREG-1740, the Ad Hoc subcommittee concluded that the leakage correlation for the 7/8-inch diameter tubes was poor. On May 11, 2001, the NRC staff amended its steam generator action plan (ADAMS ML011300073) to incorporate the issues identified in NUREG-1740. The leak rate correlation issue was incorporated in the Steam Generator Action Plan as Item 3.7.

In a letter to the NRC Chairman dated October 18, 2001, "NRC Action Plan to Address the Differing Professional Opinion Issues on Steam Generator Tube Integrity," the ACRS reiterated several of its concerns including the need to investigate the reason for the poor correlation between the leak rate and voltage for the 7/8-inch steam generator tubes. In a letter dated January 18, 2002, the Executive Director for Operations provided the ACRS preliminary responses regarding the leakage correlation issue (ADAMS ML013300241).

This assessment provides a detailed evaluation of the 7/8-inch leak rate correlation.

2.0 DISCUSSION

Steam generator tubes that are fabricated with mill-annealed Alloy 600 have experienced outside diameter stress corrosion cracking at drilled hole tube support plate intersections. To provide guidance on addressing tubes affected by this degradation mechanism, the staff issued Generic Letter (GL) 95-05, "Voltage-based Repair Criteria for Westinghouse Steam Generator Tubes Affected by Outside Diameter Stress Corrosion Cracking." The alternate repair criteria discussed in GL 95-05 allows predominantly axially oriented outside diameter stress corrosion cracking at the tube support plate intersections to remain inservice if the voltage response of the flaw detected by the bobbin coil probe is within specified limits. Many licensees have implemented voltage-based alternate repair criteria in their plant technical specifications in accordance with GL 95-05. Currently, there are nine plants (seven with 7/8-inch diameter tubes and two with 3/4-inch diameter tubes) using this repair criteria. As plants replace their steam generators, this number declines. No additional plants are expected to request the use of this repair criteria.

The leak rate correlation is used in assessing the leakage integrity of the steam generator tubing. Specifically, the leak rate correlation is used in determining a conservative estimate of

ATTACHMENT

the amount of primary-to-secondary leakage expected to occur during the most limiting postulated accident condition (typically a main steam line break). The calculated leakage value must be low enough to ensure that the applicable radiological dose limits in 10 CFR Part 100 and General Design Criterion 19 are not exceeded.

The leak rate correlation is based on an empirical relationship between the voltage amplitude of the flaw and the amount of leakage from that flaw under postulated steam line break conditions. Separate correlations between leak rate and bobbin voltage were developed for 3/4-inch and 7/8-inch diameter tubes. The database supporting these correlations is updated periodically as a result of the removal of tubes from inservice steam generators. Tubes are periodically removed from plants in accordance with the GL 95-05 repair criteria and an NRC approved tube pull program. The current leak rate database is documented in "Steam Generator Tubing Outside Diameter Stress Corrosion Cracking at Tube Support Plates Database for Alternate Repair Limits: Update 2002," Electric Power Research Institute (EPRI), 1007660, January 2003.

3.0 EVALUATION

The objective of the staff assessment is to: (1) examine the status of the leak rate correlation in the current database; (2) determine the reason(s) for the lack of correlation in the 7/8-inch database; and (3) provide an overview of the leak rate methodology. The staff reviewed the EPRI database reports from Addendum 1 (1996 update) to Addendum 5 (2002 update) to determine possible contributors to the lack of correlation in the 7/8-inch database.

3.1 Status of the Current Database

Two data points have been added to the 7/8-inch tube database since the ACRS review in 2001. Addition of these two data points significantly affected the leakage correlation because both flaws have low voltages with relatively high leak rates. The addition of the two data points did not improve the correlation between the leak rate and voltage.

In addition to adding these two data points, several data points were removed as discussed below. The leak rate correlation reviewed by the ACRS in 2001 consisted of data points (i.e., flaws) obtained from domestic nuclear plants, French nuclear plants, and from laboratory fabricated flaws. Subsequent to the ACRS review, the industry proposed to remove the French tube data from the leak rate database because there is a statistical difference between the French and domestic data (including the laboratory fabricated flaws) indicating that they are not from the same population. The industry stated that the statistical difference is supported by the physical differences in crack morphology. Physically, the French data with high bobbin voltages tend to have shallower depths than the corresponding domestic data. High bobbin voltage domestic data are almost always throughwall. As a result, the high voltage French data exhibit little or no leakage whereas all domestic high voltage indications exhibit leakage. In a letter dated October 8, 2002 (ML022810255), to the Nuclear Energy Institute, the staff concluded that the French data may be excluded from the ODSCC database. The removal of the French data improved the leak rate correlation in the 7/8-inch tube data.

The current 7/8-inch tube database (the 2002 Update) has a total of 134 data points (95 of which are from domestic pulled tubes and 39 are from the laboratory). Of the 134 data points, 29 leaked under simulated accident conditions. Of these 29 data points in the 7/8-inch leak rate correlation, 7 (24%) are from domestic pulled tubes and 22 (76%) are from the laboratory.

The 3/4-inch tube database has a total of 137 data points (90 of which are from domestic pulled tubes and 47 are from the laboratory). Of the 137 data points, 48 leaked under simulated accident conditions. Of these 48 data points in the 3/4-inch leak rate correlation, 25 (52%) are from domestic pulled tubes and 23 (48%) are from the laboratory.

During a meeting between the NRC and Pacific Gas and Electric Company (PG&E) on April 15, 2003, PG&E discussed the laboratory test results from two tubes that contain axial ODSCC indications at the tube support plate intersections that were recently pulled from Diablo Canyon Unit 2. PG&E indicated that the results of the laboratory leak rate tests strengthened the 7/8-inch tube leak rate correlation. These data points will be officially added to the 7/8-inch tube database as a part of the 2003 database update effort.

3.2 Potential Reasons for the Lack of Correlation

There are several potential reasons for the scatter in the leak rate correlation. One main reason is that for any voltage there is a range of crack morphologies. The morphology of the crack networks at the tube support plates is very complex, ranging from a small number of cracks to hundreds of cracks at tube support plate elevations. As such, a range of leak rates is possible for the same voltage. This has been observed for both the 3/4-inch and 7/8-inch leak rate database; however, the 7/8-inch database has more scatter than the 3/4-inch database.

Another reason for the scatter is that pre-pull bobbin voltages are used in the correlations instead of post-pull voltages. When a flawed tube is pulled from the steam generator for destructive examination, the crack tip may tear as a result of the tube pull operation. The voltage of the flaw will increase and the leakage in the leak test is expected to be higher than if the crack were not torn/opened. By correlating the leakage with the pre-pull voltages instead of post-pull voltages, the data are skewed in the conservative direction because the data show more leakage for a pre-pull voltage than if the post-pull voltage is used. The amount of crack tearing/opening during the pulling operation depends on plant-specific circumstances.

Because of the range of variables affecting the leakage and voltage, the “limited” number of data points, and the availability and current state of the specimens, a simple explanation for the differences in the correlations could not be established. Nonetheless, the methodology used in predicting leak rates should yield a conservative estimate of the leakage for the following reasons:

1. Pre-pull voltages are used in the correlations. If the crack tears as a result of the tube pull operation, the measured leakage is expected to be higher than if the tube were not damaged (i.e., there is more leakage for a given voltage).

2. The radiological dose consequences are determined from the leakage evaluated at the upper 95th percentile value at an upper 95 percent confidence bound. This results in a more conservative estimate of the leakage than if a mean value is used in the leak rate calculation.

3. If the crack is plugged by deposits as a result of the test, the leakage would be reduced, introducing more scatter in the data which usually results in more conservative predictions (at the 95/95 level).

4. If a statistical correlation cannot be demonstrated, GL 95-05 specifies that leakage should be treated as independent of voltage, which is conservative (since most indications left inservice are "low" voltage indications, which tend to leak less than the mean, assuming leakage does increase with voltage).

The staff finds that the correlation between leakage and bobbin voltage is weaker for the 7/8-inch diameter tubes than for the 3/4-inch diameter tubes. Future additional leakage data may result in a stronger correlation for the 7/8-inch tubes. Alternatively, additional 7/8-inch data may result in a weaker or even no correlation. A variety of parameters could affect the leakage and/or voltage data in the database; however, the leak rate data do not lead to a simple conclusive explanation for the poor correlation in the 7/8-inch tube leakage database.

4.0 CONCLUSIONS

The staff concludes that the evaluation of the leakage data does not lead to a simple conclusive explanation for the poor correlation in the 7/8-inch tube database when compared to the 3/4-inch tube database. Nonetheless, the staff concludes that the methodology used by the industry and staff in assessing leakage from flaws at the tube support plates in the 7/8-inch diameter tubes is acceptable because: (1) NRC's Generic Letter 95-05 specifies necessary actions in the leak rate calculations when the correlation is weak, and it specifies how to account for the uncertainty in the correlation; and (2) the overall methodology for assessing the consequences of steam generator tube leakage is conservative.

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