Enclosure 2

SAFETY EVALUATION REPORT (Non-Proprietary)

EVALUATION OF WESTINGHOUSE METHODOLOGY
TO ADDRESS ITEM C.2 OF
NRC BULLETIN 88-02

MATERIALS ENGINEERING BRANCH
DIVISION OF ENGINEERING AND SYSTEMS TECHNOLOGY

1 INTRODUCTION

On February 5, 1988, the staff issued NRC Bulletin 88-02, "Rapidly Propagating Fatigue Cracks in Steam Generator Tubes." The bulletin requested that holders of operating licenses or construction permits for Westinghouse-designed plants with steam generators having carbon steel support plates implement actions specified in the bulletin to minimize the potential for steam generator tube ruptures caused by rapidly propagating fatigue cracks such as occurred at North Anna Unit 1 on July 15, 1987.

This Safety Evaluation Report (SER) addresses the program developed by Westinghouse to resolve item C.2 of the Bulletin. Item C.2 is applicable to Westinghouse-designed plants where denting is known or assumed to be present at the uppermost carbon steel support plate in one or more steam generators. Item C.2 of the Bulletin requests that a program be implemented to minimize the probability of a rapidly propagating fatigue crack such as occurred at North Anna Unit 1.

This evaluation specifically addresses generic aspects of the program described in WCAP-11799 (Proprietory Version) and WCAP-11800 (Non-Proprietary Version), "Beaver Valley Unit 1 - Evaluation for Tube Vibration Induced Fatigue," April 1988 (Reference 1). However, similar programs are being implemented at a number of other facilities besides Beaver Valley Unit 1. Thus, the conclusions of this SER are applicable to other facilities which have implemented similar programs to that for Beaver Valley Unit 1. These conclusions will be incorporated by reference in plant-specific SERs, where applicable.

Licensee programs which utilize alternate approaches to that developed by Westinghouse will be evaluated on a case basis.

2 BACKGROUND - CIRCUMSTANCES OF FAILURE AT NORTH ANNA UNIT 1

The circumstances of the North Anna failure were evaluated in detail by the staff in an SER dated December 3, 1987, supporting return of North Anna Unit 1 to full power operation (Reference 2). This Section provides a brief summary of these circumstances. Westinghouse findings discussed in this section were accepted by the staff in the above-mentioned SER for North Anna Unit 1.

The steam generator tube rupture event at North Anna Unit 1 occurred on July 15, 1987 shortly after the unit reached 100% power. For several days prior to the event, operators had observed erratic air ejector radiation monitor readings. Grab samples were taken prior to the tube rupture for purposes of performing environmental release calculations. Subsequent analysis of this data indicated that increasing primary to secondary leakage had occurred over a 24- to 36-hour period before the tube rupture event. This leakage had been below the limit given in the Technical Specifications. The ruptured tube was located in Row 9 Column 51 (R9-C51) in steam generator "C". The rupture location in this model 51 steam generator was at the top support plate on the cold leg side of the tube. The rupture extended circumferentially 360 degrees around the tube.

A portion of the ruptured tube extending from the cold leg inlet to the fracture location was removed from the steam generator for laboratory examination. The examination of the fracture surface clearly established high cycle fatigue as the failure mechanism as evidenced from the pattern of striation marks and other fracture surface features associated with fatigue. Multiple initiation sites were observed over approximately a 40 degree arc at the OD of the fracture surface. The geometric center of the initiation zone was oriented about 90 degrees from the plane of the U-bend.

The initial alternating stress associated with fatigue crack initiation was determined by Westinghouse to be in the range of 4 to 10 ksi. This estimate was based on fracture mechanics analyses employing stress intensity factors estimated from the observed striation spacings on the fracture surface. Large-amplitude flow-induced vibration, normal to the plane of the U-bend, was

established as the driving mechanism for this alternating stress in view of large number of cycles ($>10^5$ as inferred from the striation spacings observed on the fracture surface) required to propagate the crack from the point of initiation to complete fracture of the tube. This mechanism is consistent with the observed location of the crack initiation zone since the maximum bending stress produced by this mechanism also occurs 90 degrees from the plane of the U-bend.

For alternating stresses in the 4 to 10 ksi range, a high mean stress caused by denting at the uppermost support plate was established by Westinghouse as a requisite condition for fatigue crack initiation. [The staff concluded in Reference 2 that shallow intergranular attack (IGA) penetrations (1 or 2 grains) on the tube surface may also have contributed to fatigue crack initiation.] Denting also served to shift the maximum alternating (bending) stress caused by flow-induced vibration to the vicinity of the upper support plate. Denting is known to have been present in the North Anna Unit 1 steam generators since the first operating cycle.

Another requisite condition for the failure at North Anna was the absence of an AVB support at the U-bend region of the tube. For a row 9 tube at North Anna, a [ ] cantilever deflection of the U-bend (as measured at the apex) is necessary to develop the 4 to 10 ksi alternating stress which led to the fatigue failure. The presence of an AVB support will restrict tube motion and thus preclude the deflection amplitude required for fatigue. The original design configuration for North Anna required AVBs to be inserted down to as far as row 11. However, inspections show that some AVBs in the North Anna Unit 1 steam generators penetrate to row 8, exceeding the minimum AVB required depth. However, no AVB support was present for the R9-C51 tube that ruptured.

3 DISCUSSION

This section describes the criteria and supporting rationale developed by Westinghouse for identifying tubes which may become susceptible to fatigue cracks such as occurred at North Anna. Subsection 3.1 describes the flow-induced vibration mechanisms acting on steam generator tubes in the U-bend region. Subsection 3.2 provides an overview of the various factors which may lead to excessive vibration amplitudes, thus creating the potential for fatigue failure. Subsection 3.3 discusses the development of the Westinghouse methodology for identifying susceptable tubes.

3.1 FLOW INDUCED VIBRATION

Westinghouse considered the following flow induced vibration mechanisms as possible contributors to the fatigue failure at North Anna Unit 1: vortex shedding, cross flow turbulence, and fluid-elastic instability.

[

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Westinghouse analysis of linear turbulence and fluid-elastic excitation of the tubing was conducted with a computer program called FLOVIB. FLOVIB incorporates a finite element model of the tube and tube support system and evaluates the dynamic response of the tube based on models for modal vibration amplitude in the turbulent and fluidelastic regimes.

Thermal-hydraulic input to the FLOVIB program was provided by a computer program called ATHOS which used to model the Model 51 steam generator and operating conditions. The program provides velocities, densities, and other parameter distributions for any specific tube (such as R9-C51 which ruptured).

The above mentioned model for evaluating tube response from the turbulence mechanism has been qualified against several series of tests including prototypical two-phase tests. Turbulence induced vibration amplitudes for tube R9-C51 are predicted to be on the order of less than [ ] at the tube apex. This order of amplitude would cause maximum stresses in the top of the uppermost support plate (where the rupture occurred) with peak-to-peak amplitudes of less than 1000 psi. Based on these low stress levels, Westinghouse believes it to be highly improbable that the turbulence mechanism is primarily responsible for the North Anna tube rupture.

The fluidelastic mechanism will have a significant effect on the tube response in cases where the fluidelastic stability ratio (SR) equals or exceeds 1.0. The stability ratio is defined as

$$SR = \frac{V_{eff}}{V_c}$$

were V_{eff} is the effective crossflow velocity and V_c is the critical velocity beyond which the displacement response increases rapidly.

For R9-C51 at North Anna, the estimated stability ratio (from FLOVIB/ATHOS) utilizing nominal estimates of parameters such as damping ratio, stability constant, and natural frequency is [ ] indicating no fluidelastic instability.

As will be discussed later, these parameters are subject to significant uncertainties (particularly for damping ratio and local flow velocity) such that depending on the actual values of the parameters, the stability ratio may substantially exceed 1.0.

Motions and corresponding stresses developed by a tube in the fluid-elastically unstable mode are quite large in comparison to the other known mechanisms. For this reason, and because none of the other mechanisms discussed above appear to be a plausible mechanism for crack initiation, Westinghouse has concluded that the failed tube is most likely a result of its being fluid-elastically unstable.

Given fluid-elastic instability as the mechanism for fatigue crack initiation, the stability ratios for the failed tube can be inferred from [

]

To estimate the critical velocity and, hence, the stability ratios of the tube which failed, it is helpful to consider Figure 1. A reference value of effective velocity (termed V_{oper} in Figure 1) can be estimated from ATHOS/FLOVIB.

Additionally, FLOVIB provides the turbulence response amplitude associated with the reference operating velocity (i.e., point 1 in Figure 1). The displacement value for point 2 in Figure 1 is the above-mentioned [ ] displacement necessary to initiate a crack at the upper support plate.

FIGURE 1

Displacement Response Versus Velocity

From experimental results, Westinghouse states that it is known that the turbulence response curve (in log-log coordinates) has a slope of [ ]. Test results also show that the slope for the fluid-elastic response depends somewhat on the instability displacement amplitude. Westinghouse states that it has been shown by tests that a slope of [ ] is appropriate for displacement amplitudes in the range of [ ], whereas below [ ] are conservative (lower bound) values.

As can be seen from Figure 1, definition of points 1 and 2 and the slopes of the turbulence and fluid-elastic response curves are sufficient to solve for Vc and thus for stability ratio (i.e., V_{oper}/V_c). On this basis, tube R9-C51 is estimated to have been operating (prior to crack initiation) at a stability ratio in the range of 1.22 [ ] to 1.56 [ ], assuming the tube to be vibrating with a displacement amplitude of [ ] inches and a corresponding alternating stress level of 7 ksi.

It follows from Figure 1 that for a given reduction in stability ratio from SR_2 to SR_1, the corresponding reduction in displacement response (i.e., from D_2 to D_1) and alternating stress (S_2 to S_1) can be expressed as follows:

[ ]

[ ] (Equation 1)

It can be seen from this equation that a small percentage reduction in stability ratio will lead to a much larger percentage reduction in displacement response and alternating stress. With a known reduction in alternating stress, the corresponding reduced value of fatigue usage factor can be estimated.

Westinghouse has also proposed a more generalized form of Equation 1 to compare the displacement and stress responses for different tubes in the same or other steam generators, accounting for possible differences in tube geometry, as follows (Reference 1):

[ ]

[ ] (Equation 2)*

where: [ ]
[ ]

3.2 OFF-NOMINAL CONDITIONS POTENTIALLY LEADING TO HIGH STABILITY RATIOS

As discussed in Section 3.1, the estimated stability ratio for tube R9-C51 at North Anna is [ ] using nominal estimates of parameters such as stability constant, natural frequency, and damping ratio. Thus, the tube would have been expected to be stable with considerable margin. The following summarizes Westinghouse's assessment (References 1 and 3) of factors which may have caused the actual stability ratio to significantly exceed 1.

3.2.1 Natural Frequency, Stability Constant, and Average Flow Field Uncertainties

Westinghouse considers uncertainties associated with calculated tube natural frequencies (in FLOVIB) to be insignificant. This is based on good correlations between analytical estimates and data for real structures, particularly in instances where the tubes are [ ] at the support plates as a result of denting.

The stability constant (beta) value used in the stability ratio and critical velocity evaluations (utilizing FLOVIB) are based on Westinghouse data and other experimental results as documented in Reference 1. This includes data from [

*[

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] This value is used in all stability ratio evaluations addressed in Reference 1 for Beaver Valley and Reference 3 for North Anna 1 and presumably in similar Westinghouse evaluations for other plants.

The impact of uncertainties associated with ATHOS flow velocity and density distribution predictions on stability analyses were evaluated by using ATHOS to model high pressure steam-water (MB-3) tests conducted at Mitsubishi Heavy Industries (MHI) in Japan. The assumed stability constant of [ ] was found to be consistent with the ATHOS predicted flow conditions and the MB-3 measured critical velocity.

Based on the above evaluations, Westinghouse has concluded that natural frequency, stability constant, and average flow field uncertainties contribute about [ ] uncertainty to stability ratio and critical velocity estimates from FLOVIB. Thus, Westinghouse estimates that the maximum effect of these uncertainties would be to increase the nominal stability ratio estimate for R9-C51 at North Anna from [ ] to [ ] which is still well within the stable regime. Westinghouse further states that uncertainties associated with these parameters could not be large enough to lead to instabilities (i.e. SR ≥ 1.0). In making this statement, Westinghouse notes that uncertainties in the average flow field parameters, stability constant, and natural frequency are essentially the same for units with dented or non-dented top support plates. If the errors associated with these uncertainties were large, similar instabilities would be expected in the non-dented units with resulting wear at either the top support plate or inner row AVBs. Westinghouse states that significant tube wear has not been observed in inner row tubes in operating steam generators without denting.

3.2.2 Damping Ratio Uncertainties

Measurements of mechanical damping in air were performed using a U-bend shaker test facility. For [

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The above air tests do not consider the additional damping in a two-phase water/steam environment. Data cited by Westinghouse shows [

]

Based on the data for [ ], Westinghouse assumed a nominal value of damping ratio of [ ] which is associated with the above-mentioned nominal stability ratio estimate of [ ] The Westinghouse report for Beaver Valley (Reference 1) omits discussion of potential uncertainties in the damping ratio estimate. However, Westinghouse in the earlier North Anna report (Reference 3) estimated a [ ] uncertainty factor associated with the assumed [ ] damping ratio which would contribute approximately [ ] uncertainty to the nominal stability ratio estimate. The assumption of a [ ] would increase the nominal stability ratio estimate by about a factor of 2.

3.2.3 Local Flow Velocities

Eddy current test results at North Anna Unit 1 established that the AVB supports generally extended as far as row 10 with most extending as for as row 9 and many as far as row 8. These non-uniform AVB penetrations have been shown by Westing-

house to have channeled some of the flow to row 8 and row 9 tubes without adjacent AVB supports causing a "velocity peaking" effect for these tubes. Preliminary analysis by Westinghouse in Reference 3 indicated that flow peaking may have increased the nominal stability ratio for R9-C51 at North Anna Unit 1 by a factor of [ ]. As reported in Reference 1, however, Westinghouse now estimates on the basis of an air model tests that the AVB insertion configuration in the vicinity of R9-C51 may have increased the stability ratio for R9-C51 by from [ ] relative to the nominal stability ratio [ ] as determined from FLOVIB/ATHOS. It should be noted that the ATHOS code does not include the capability to model the presence of the AVBs in the U-bend region.

Flow peaking effects, including a description of the air model tests and analyses, are discussed in detail in Section 3.3.4.2.

3.2.4 Summary of Off-Nominal Conditions Potentially Associated with High Stability Ratios

The following summarizes off-nominal conditions potentially contributing to the high stability ratio for tube R9-C51 at North Anna Unit:

	Stability Ratio
° Nominal stability ratio estimate from FLOVIB/ATHOS	[ ]
° Including [ ] increase to account for potential uncertainties in natural frequency, stability constant, and average flow velocity	[ ]
° Also including [ ] increase to account for damping uncertainty	[ ]

- Also including [ ] increase to account for flow peaking associated with AVB configuration near tube R9-C51 at North Anna [ ]

Westinghouse now believes (Reference 3) that local flow peaking effects from non-uniform AVB penetrations was a major contributor to the fluid-elastic instability which led to the failure of R9-C51 at North Anna Unit 1. Although Westinghouse is somewhat non-committal in Reference 1 on the potential contribution from lower-than-nominal damping ratios, the staff believes that such off-nominal damping cannot be ruled out as a possibility and, thus, as a significant contributor to the onset of an instability. The staff notes that Westinghouse reported in Reference 3 that a few row 8 to row 10 tubes were found to contain wall thinning indications at AVB locations which may have occurred as a result of fluid-elastic exitation. Some of these tubes were located in regions of relatively uniform AVB penetrations where flow peaking effects would be minimal.

3.3 DEVELOPMENT OF METHODOLOGY TO IDENTIFY SUSCEPTIBLE TUBES

3.3.1 Fatigue Strength Considerations

"S/N curves" are curves relating the magnitude of alternating stress to the number of alternating stress cycles necessary to initiate a fatigue crack. The S/N curves used by Westinghouse to evaluate conditions for crack initiation were taken from data in [ ]. This data is based on fully reversed loading. The natural frequency of the row 9 tube U-bends at North Anna is about [ ]. Assuming that the fatigue crack initiated over the 9-year lifetime of the plant to date, then the number of loading cycles leading to fatigue failure (assuming a plant availability factor of 75%) was 1.3×10^{10} cycles.

The corresponding fatigue strength is estimated by the staff to be 22 ksi utilizing a best fit S/N curve and 16 ksi utilizing a 3 standard deviation (3 sigma) lower bound S/N curve from the data in Reference 4. These fatigue strength estimates exceed the 4 to 10 ksi alternating stress believed by Westinghouse to have caused fatigue crack initiation, but do not include an adjustment for mean stress at the fracture location induced by denting of the tube at the seventh support plate.

The mean stress at the fracture location was evaluated by Westinghouse with an elastic-plastic finite element analysis utilizing an axi-symmetric model. The assumed denting deflection profile from the bottom of the seventh support plate to the top was based on profilometry measurements conducted on the fractured tube and other adjacent tubes. The fractured tube had a maximum radial displacement of 2.5 mils. Steady state pressure and thermal loadings were also included in the model. The results of the analysis revealed a tensile mean stress level at the tube OD at the yield strength level.

The [ ] was used to make the mean stress adjustment to the S/N curves derived from the [ ] data. [ ]. In addition, the [ ] model provides the most reasonable reconciliation between the observed number of cycles to failure (approximately 1.3×10^{10} cycles) and the range of initiating stress amplitudes (4 to 10 ksi).

In Reference 2, the staff concluded that IGA could not be entirely discounted as a contributor to fatigue crack initiation in view of small (1 or 2 grains) IGA penetrations which were observed as close as 4 mils from the fracture surface. If IGA did play a role, IGA penetrations would have provided a stress raiser for the nominal 4 to 10 ksi initiating stress amplitude. Thus, a mean stress effect as severe as that assumed by Westinghouse would not be necessary to explain

crack initiation for this nominal stress range. Even if an IGA influence is assumed, however, the staff believes it should not detract from the effectiveness of actions proposed by Westinghouse to ensure that nominal stresses are sufficiently low to minimize the potential for fatigue crack initiation.

3.3.2 Effect of 10% Stability Ratio Reduction for R9-C51 at North Anna

Westinghouse has conservatively estimated that a 10% reduction in stability ratio relative to that which led to the failure of R9-C51 at North Anna would reduce fatigue usage sufficiently to preclude fatigue crack initiation over the 40 year lifetime of the unit. As previously discussed, Westinghouse estimates the alternating stress associated with fatigue crack initiation at North Anna to be in the 4 to 10 (actually 9.5*) ksi range. Assuming the upper bound value of 9.5 ksi, it follows from equation 1 that a 10% reduction in stability ratio will reduce this stress to 4 ksi. This reduction has conservatively been estimated assuming a [ ]. The corresponding fatigue usage factor is .021 per year if the lower-bound, 3 sigma [ ] adjusted S/N curve is used. This translates to a total fatigue usage factor of 0.84 over 40 years indicating that crack initation should not occur. It should be noted that for the 3 sigma [ ] adjusted S/N curve, the assumption of an initial stress level of 9.5 ksi is unrealistically high. This is because the number of cycles to failure for 9.5 ksi using this S/N curve is only about 10^9 cycles (less than one year) whereas the actual failure occurred after 1.3×10^{10} cycles.

*Westinghouse estimates this to be the maximum possible stress consistent with the observed striation spacings on the fracture surface without having to assume a physically impossible stress history during crack propagation (Reference 1).

Westinghouse has also performed a number of more realistic calculations to demonstrate that above-mentioned fatigue usage estimate of 0.021 per year conservatively upper-bounds the fatigue usage which would have occurred for R9-C51 at North Anna, given a 10% reduction is stability ratio for that tube. These calculations considered a number of different combinations of initial stress level and S/N curves which were consistent with the observation that failure of R9-C51 occurred over 1.3×10^{10} cycles. These calculations also explicitly considered the variations in steam generator flow parameters (e.g., mass flow rates, steam pressure) and, thus, fluid-elastic stability ratios and resulting alternating stresses which existed from cycle to cycle prior to the North Anna failure. The staff's review of these more realistic calculations indicates that fatigue usage associated with a 10% reduction in stability ratio for R9-C51 likely would not have exceeded 0.01 per year.

3.3.3 Stress Ratio Criteria

As discussed in Section 3.3.2, a 10% reduction in stability ratio was established by Westinghouse as sufficient to reduce the stress amplitude of R9-C51 at North Anna Unit 1 to a value (i.e., 4 ksi) which would not have initiated a crack over the lifetime of the plant. Assuming for the moment that 4 ksi is the maximum acceptable value for any tube in any steam generator, then the alternating stress for any such tube must satisfy the following criteria:

$$S(x)/4 \text{ ksi} \leq 1.0 \qquad \text{(stress ratio criteria)}$$

From Equation 2 (discussed in Section 3.1), the stress ratio can also be expressed as follows:

[ ] (Equation 3)

where: x refers to the specific tube being evaluated

9/51 refers to R9-C51 at North Anna Unit 1

The quantity, SR(x)/SR(9/51), is referred to as the normalized stability ratio for the tube under consideration. Westinghouse has chosen to work with normalized stability ratios in lieu of absolute stability ratios to further minimize the impact of uncertainties in the stability ratio estimates by essentially eliminating constant error factors. Westinghouse calculates the normalized stability ratio using the methodology described in Section 3.3.4. The corresponding stress ratio is then calculated from Equation 3.

The stress ratio criterion is used by Westinghouse as a screening criterion to ensure that S(x) is less than or equal to 4 ksi. However, 4 ksi will not necessarily ensure acceptable fatigue usage for tubes whose stiffness and/or section modulus differ from that for R9-C51 at North Anna. The fatigue usage can be conservatively estimated from the lower bound, 3 sigma [ ] S/N curve considering the natural frequency of the tube and the design basis service life. The alternating stress used in these estimates is given by the expression:

[ ] (Equation 4)

3.3.4 Normalized Stability Ratio Estimates

As discussed in subsection 3.3.3, equations 3 and 4 are used to calculate the maximum alternating stresses for tubes in any steam generator which are dented at the uppermost tube support plate and which are unsupported by AVBs. For a given tube, the [

] are known quantities (as determined from [

]. The remaining parameter which must be determined in order to solve for the maximum alternating stress is the normalized stability ratio (SR(x)/SR(9/51)) for the tube in question. The normalized stability ratio can be broken down as follows:

$$\frac{SR(x)}{SR(9/51)} = \left(\frac{SR_r(x)}{SR_r(9/51)}\right)\left(\frac{FP(x)}{FP(9/51)}\right) \qquad \text{(Equation 5)}$$

where: SR_r = Nominal stability ratio for uniform flow (no flow peaking) conditions

FP = local flow peaking factor

3.3.4.1 Nominal Stability Ratios

'nal stability ratios (SR_r) are now determined with the Westinghouse ·roprietory finite element code called FASTVIB which calculates the response of individual tubes exposed to tube location dependant fluid velocities, densities, and void fractions. [FASTVIB has been verified by Westinghouse to be equivalent to the FLOVIB program and an associated pre-processor program which were used for the analysis of R9-C51 at North Anna Unit 1.] The velocities, densities, and void fractures are determined using the 3D ATHOS code which models the steam generator and operating conditions (e.g., steam flow and pressure, circulation ratio).

The nominal stability ratio estimates are generally performed for a reference operating cycle for each plant. Actual stability ratios may vary from cycle to cycle due to differences in steam flow, steam pressure, and circulation ratio. Detailed FASTVIB/ATHOS analyses are very time consuming (and expensive) and are generally not performed for each cycle. However, approximate estimates of the cycle dependent stability ratios relative to the reference cycle are obtained from simplified, one-dimensional analyses considering relative values of the major operational parameters affecting stability ratio.

3.3.4.2 Flow Peaking Factors

The ATHOS code does not have the capability to assess local flow peaking effects associated with non-uniform AVB penetrations. Westinghouse has performed extensive air model (wind tunnel) testing to study the flow peaking phenomenon and its potential effect on fluid-elastic stability ratios.

The test procedure consisted of [

]. In rerunning a particular test, good repeatability of the results was demonstrated.

Some of the various AVB insertion configurations which were tested are shown in Figure 2. The uniform AVB insertion configuration was selected as the reference configuration. The other configurations tested were taken to be representative of the configurations actually occurring in steam generators in the field, including that associated with the failed North Anna tube (R9-C51). The flow peaking factor is defined as the ratio of the critical velocity associated with uniform AVB insertion (the reference configuration) divided by the critical velocity for the configuration of interest.

Westinghouse has performed as detailed assessment of potential uncertainties associated with the flow peaking factors derived from the air model tests, and has directly considered these uncertainties to ensure conservative estimates of normalized peaking factors (i.e, FP(x)/FP(R9-C51)). The major sources of uncertainty which have been considered and accounted for in the peaking factor estimates are:

1) velocity test measurements
2) test repeatability
3) geometry differences between the test model and the actual steam generators
4) use of air model tests to predict behavior of prototypical steam-water mixture.
5) uncertainties in AVB insertion depth estimates as derived from eddy current test

Flow peaking factors were determined to range to a maximum of [ ] for the various configurations tested. The maximum flow peaking factor was associated with an AVB configuration which matched that which existed local to R9-C51 of North Anna Unit 1. This is consistent with the observation that flow peaking was a primary contributor to the North Anna failure event. However, for purposes of calculating conservative normalized flow peaking factors for other tubes, Westinghouse has considered a lower bound flow peaking factor of [ ] for R9-C51 at North Anna Unit 1 based on the uncertainty assessment discussed above.

FIGURE 2

Typical AVB Insertion Configurations

3.3.5 AVB Insertion Depth Assessment

The AVB insertion depths are determined on the basis of interpretation of the eddy current test (ECT) signals produced by the intersection of AVBs with the U-bends and the known geometry of the AVBs (apex radius and included angle). Detailed criteria for assessment of these data have been developed by Westinghouse. The use of data from multiple tubes in the same column provide consistency checks for the data from individual tubes and also provide a means for estimating the location of AVB intersections for tubes which were previously plugged or for which ECT data could not be obtained. Although ECT is generally capable of detecting AVB intersections of a tube, it does not provide a direct indication of what side of the tube the detected AVB is located. This can be determined indirectly, however, by enforcing consistency of data between adjacent columns of tubes.

AVBs whose centerlines extend to at least the center-plane of a tube at the apex of the U-bend can be considered to provide an effective support to that tube. Tubes with an effective AVB support on only one side of the tube can be considered to be fully supported. This latter point has been confirmed by air-model tests performed by Westinghouse. These tests show that the amplitude of vibration for a tube with a single-sided support is limited to gap between the tube and the support.

3.3.6 Summary of Westinghouse Methodology to Identify Susceptible Tubes

The Westinghouse methodology for identifying tubes potentially susceptible to fatigue crack initiation is summarized below. The expression, "region of interest," is interpreted by the staff to include the region defined by the following boundaries. The outer boundary is a tube row located at or beyond the row of the shallowest AVB penetration depth. The inner boundary corresponds to a tube row where the net stability ratios (for dented, unsupported tubes) have attenuated sufficiently to be of no significance.

<u>Eddy Current Data Review</u>

- Identify all tubes in the region of interest which are dented at the uppermost tube support plate. Denting is a requisite condition for fat-

igue crack initiation. [As stated in Bulletin 88-02, the staff notes that denting should be considered to include evidence of support plate corrosion and the presence of magnitude in the tube-to-support plate crevices, regardless of of whether there is detectable distortion.]

- Identify location of AVB support signals for all tubes in the region of interest.

Evaluation Activities

- Evaluate AVB insertion depths to identify unsupported tubes in the region of interest and to assess flow peaking factors. The Westinghouse methodology for evaluating AVB insertion depths is briefly described in Section 3.3.5.

- Evaluate normalized flow peaking factors (relative to R9-C51 at North Anna Unit 1) for unsupported tubes in region of interest based on configuration of AVB insertion depths. The basis for the flow peaking factor estimates is described in Section 3.3.4.2.

- Perform ATHOS analysis for reference operating cycle to assess effective flow velocities, densities, void fractions, and other relevant thermal-hydrualic parameters for tubes within the region of interest. This analysis is briefly described in Section 3.3.4.1.

- Evaluate normalized stability ratios (without flow peaking effects) relative to R9-C51 at North Anna for tubes in the region of interest as discussed in Section 3.3.4.1.

- Determine net normalized stability ratio (including flow peaking effects) from Equation 5 (Section 3.3.4) for unsupported tubes in region of interest.

- Determine stress ratio for unsupported tubes utilizing equation 2 and corresponding maximum alternating stress from equation 3. Equation 2 and 3 were presented earlier in Section 3.3.3. Calculate corresponding fatigue usage factors utilizing lower bound, 3 sigma [ ] S/N curve. The number of stress cycles over the projected service life of the plant should be based on the natural frequency of the subject tube assuming clamped supports.

Corrective Actions

- Plug unsupported, dented tubes with projected usage factors exceeding 1.0 over the service life of the plant. Sentinel plugs should be employed to ensure that any fatigue crack initiation subsequent to plugging will produce a detectable leak before the tube completely severes and causes damage to adjacent tubes. Alternatively, the tube should be stabilized to preclude possible damage to adjacent tubes.

4. SUMMARY OF STAFF FINDINGS

1. For reasons cited in Reference 2, the staff concurs with the Westinghouse conclusions that (1) the failure of R9-C51 at North Anna Unit 1 was caused by fatigue, (2) the alternating stress associated with fatigue crack initiation was in the 4 to 10 ksi range, and (3) fluid-elastic instability has reasonably been established as the only credible mechanism for producing displacements of sufficient magnitude to cause a fatigue failure. Further, it has reasonably been established that such an instability is possible within the uncertainty bounds of key parameters such as stability constants, damping ratio, and local flow peaking factors.

2. The mean stress model developed by Westinghouse (described in Section 3.3.1) provides a reasonable reconciliation of the estimated stress level versus the observed number of cycles to failure for R9-C51 at North Anna. As discussed in Section 3.3.1, the staff believes that IGA cannot be entirely discounted as a contributor to fatigue crack initiation. If IGA did play a

role, a mean stress effect as severe as that assumed by Westinghouse is not necessary to explain fatigue crack initiation for tube R9-C51 at North Anna Unit 1. Nevertheless, the finds that the assumed 3 sigma, lower bound [ ] S/N curve stemming from the Westinghouse mean stress model should lead to conservative results.

3. The staff findings herein are based on the premise that stress ratio and fatigue estimates will be based on the assumption of a full mean stress effect (i.e., yield stress). Westinghouse analyses show that if a tube is simply clamped at the upper support (due to support plate corrosion) without actual distortion of the tubes, then the mean stress will be less than the yield stress, and thus only a partial mean stress correction of the S/N curve need be considered. However, the staff believes that a full mean stress adjustment is necessary to ensure a conservative analysis since the potential influence of small IGA penetrations in initiating fatigue cracks has not been directly considered in the Westinghouse model.

4. Westinghouse's stress ratio method for identifying susceptible tubes, as identified in Section 3.3.6, does not ensure that unsupported, dented tubes remaining in service will have stability ratios less than 1.0, even if all such tubes exhibit nominal damping. Lower-than-nominal damping would further increase the stability ratio. The stress ratio method is intended to ensure that tubes which exhibit a fluid-elastic instability and which still remain in service will exhibit displacement and stress responses which are sufficiently small to preclude a fatigue failure such as occurred at North Anna Unit 1.

5. The stress ratio equation, as expressed in Equations 2 and 3, is an approximate relationship since for it to be strictly valid, the instability response curves for the tubes being compared must be identical. For this to be case, all fluid and mechanical properties and, most importantly, the flow field must be identical. This will generally not be the case. The staff notes, however, that Westinghouse's application of this equation has utilized conservative slopes (for the instability response curves) and

fatigue usage has been calculated with conservative S/N curves. The staff concludes that the stress ratio approach developed by Westinghouse represents a sound engineering approach which, if properly implemented, will provide reasonable assurance against future failures of the kind which occurred at North Anna Unit 1.

6. Given the complexity of the problem (e.g., two phase flow, U-tubes, denting) and the associated uncertainties, the Westinghouse analyses conducted with ATHOS and FLOVIB provide only "ballpark estimates" of the response amplitudes and instability thresholds. Absolute values of predicted critical velocities and displacement amplitudes in the instability region incorporate significant uncertainty. However, the results of these analyses are appropriate for the use that Westinghouse has made of them; namely to develop relationships between relative stability ratios and the corresponding relative displacement and stress responses, thus permitting an assessment of the relative potential for failure (compared to R9-C51 at North Anna Unit 1).

7. The air model tests conducted by Westinghouse clearly demonstrate (1) the importance of local flow peaking effects (due to non-uniform AVB insertion) as a contributor to fluid-elastic instability and (2) that particular AVB configurations lead to more severe flow peaking than other configurations. It is especially noteworthy that the configuration associated with the failed North Anna tube was found to be the most susceptible of all the configurations tested. The earlier Westinghouse analysis in Reference 3, which considered lower-than-nominal damping to be the dominant contributor to the North Anna failure, did not fully explain why the failure affected tube R9-C51 rather than higher row tubes which were also unsupported (by AVBs) and which exhibited nominal stability ratios (flow peaking not considered) which were higher than that for R9-C51.

8. Although local flow peaking appears to have been a major contributor to the instability of R9-C51 at North Anna, low damping relative to the nominal values assumed by Westinghouse may also have been an important contributor. This is evidenced by the fact that a number of tubes at North Anna Unit 1

located in rows 8 to 10 exhibited wall thinning indications at AVB support locations which Westinghouse speculated in Reference 3 may have occurred as a result of fluid-elastic excitation. Some of these tubes were located in regions of relatively uniform AVB penetrations where flow peaking effects would be minimal.

9. For plants where indications of denting were found in accordance with item A of Bulletin 88-02, all tubes in the region of interest should be assumed to be dented (pursuant to the definition of denting provided in the bulletin) except for tubes for which the absence of denting on both the hot and cold leg side has been specifically verified by inspection. Some non-dented tubes in the region of interest may become dented at a later time. Should the licensee elect not to plug undented tubes which would otherwise be pluggable based on the estimated stress ratio and/or fatigue usage factor, the licensee should submit and commit to an appropriate inspection program for these tubes to ensure the timely detection of the onset of denting of these tubes in the future.

10. Calculated stress ratios and projected fatigue usage factors are based on reference steam generator operating parameters (e.g., steam flow and pressure, circulation ratio) which are assumed to exist over the remaining life of the plant. The licensees should commit to developing administrative controls to ensure updated stress ratio and fatigue usage calculations are performed in the event of any significant changes to these operating parameters relative to the assumed reference conditions.

11. The leakage holes in the sentinel plugs should be sized to permit primary-to-secondary leakage in excess of the Technical Specification leak rate limit in the event the plugged tube should fail subsequent to plugging. This is to ensure plant shutdown for appropriate corrective action before damage to adjacent tubes can potentially occur.

5 CONCLUSIONS

Based on the above evaluation, the staff concludes that the Westinghouse generic program identified in Reference 1 and summarized in Section 3.3.6 of this SER is

an acceptable approach for implementing Item C.2 of NRC Bulletin 88-07. The Westinghouse approach, if properly implemented, will provide reasonable assurance against future failures of the kind which occurred at North Anna Unit 1.

REFERENCES

1. Westinghouse Report WCAP-11799 (Proprietory Version) and WCAP 11800 (Non-Proprietary Version), "Beaver Valley Unit - Evaluation for Tube Vibration Induced Fatigue," April 1988. NRC Assession No. 8805160073.

2. NRC Letter dated December 11, 1987 to Mr. W. L. Stewart, Virginia Electric and Power Company, enclosing proprietary and non-proprietary versions of staff's Safety Evaluation authorizing 100% power operation of North Anna Unit 1 following steam generator tube rupture event on July 15, 1987.

3. Westinghouse Report WCAP-11601 (Proprietary Version) and WCAP-11602 (Non-Proprietary Version), "North Anna Unit 1 Steam Generator Tube Rupture and Remedial Actions Technical Evaluation," September 1987. NRC Accession Nos. 8710050087 and 8710050084.

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