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NRC Regional Administrator, Region III
U.S. NUCLEAR REGULATORY COMMISSION
799 Roosevelt Road
Glen Ellyn, Illinois 60137

Dear Sir:

DOCKETS 50-266 AND 50-301
NRC BULLETIN NO. 88-02, RAPIDLY PROPAGATING
FATIGUE CRACKS IN STEAM GENERATOR TUBES
POINT BEACH NUCLEAR PLANT UNITS 1 AND 2

This is in response to NRC Bulletin No. 88-02 regarding the potential susceptibility of steam generator tubes to rapidly propagating failure initiated by high cycle fatigue such as occurred at North Anna Unit 1 on July 15, 1987. Point Beach Nuclear Plant Units 1 and 2 are Westinghouse designed nuclear power reactors with Models 44F and 44 steam generators, respectively.

The Unit 1 steam generators are not considered to be susceptible to this phenomenon. The Unit 1 steam generator lower assemblies were replaced in 1983 with a design which included tube support plates designed to minimize the potential for denting of the tubes due to corrosion in the crevice between the tube and the tube support plate. The Unit 1 steam generator tube support plates are made of SA-240 Type 405 ferritic stainless steel and utilize quatrefoil shaped holes. Therefore, denting at the support plates and the resulting reduction in damping and increase in mean stress required for high cycle fatigue failure are not expected in Model 44F steam generators, and NRC Bulletin No. 88-02 is not applicable to Unit 1.

On September 22, 1987 a preliminary assessment by Westinghouse identified the Unit 2 steam generators as being potentially susceptible to tube failure due to high cycle fatigue. In order to more accurately assess the potential for high cycle fatigue failure of Unit 2 steam generator tubes, we performed evaluations with Westinghouse Electric Corporation during the fall 1987 refueling and maintenance outage. The details of the evaluations and our conclusions were presented at a meeting attended by repre-





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sentatives of the Nuclear Regulatory Commission staff, Westinghouse Electric Corporation and Wisconsin Electric on November 4, 1987 and are summarized in our November 11, 1987 letter to the Commission. Further information regarding our enhanced primary-to-secondary leakage monitoring program was provided in our November 20, 1987 letter. NRC staff concurrence with our enhanced leakage monitoring program was provided in a letter from Mr. D. H. Wagner dated November 20, 1987. Detailed reports containing our evaluations and the conclusions were submitted to the Commission in our January 11, 1988 letter which transmitted WCAP-11666, "Point Beach Unit 2 Evaluation for Tube Vibration Induced Fatigue (Proprietary)" and WCAP-11667, "Point Beach Unit 2 Evaluation for Tube Vibration Induced Fatigue (Non-Proprietary)." These evaluations confirmed that Unit 2 steam generator tubes are not susceptible to fatigue failure.

Our responses to specific actions requested in NRC Bulletin 88-02 are provided below. References are provided as appropriate to documents submitted previously and described above.

A. Steam Generator Tube Inspections

Eddy current testing was conducted during the fall 1987 refueling and maintenance outage on essentially 100% of the in-service tubes in rows 8 through 13 to identify those tubes which exhibit denting and to identify the antivibration bar (AVB) insertion depths for each column. As expected, the eddy current test data showed that virtually all of the tubes in these rows are dented at the sixth support plate (uppermost tube support plate for a Model 44 steam generator). The eddy current test data also showed that no tubes have wall thinning indications at the AVB locations. Therefore it is unlikely that those tubes have been unstable. As a conservative measure it was assumed for subsequent evaluations that all tubes of concern are dented at the sixth support plate.

B. Future Steam Generator Tube Inspections If No Denting Is Found

Not applicable to Point Beach Unit 2.

C.1. Enhanced Primary-to-Secondary Leakage Monitoring

Concurrent with the November 1987 evaluations to determine the susceptibility of the Unit 2 steam generator tubes to high cycle fatigue failure, we reviewed our existing primary-to-secondary leak rate monitoring capabilities and procedures. As a result of this review, the procedures were revised to enhance the ability to detect, assess, and take appropriate action during the progression of a tube failure

with leakage characteristics similar to that which occurred at North Anna Unit 1. Our enhanced primary-to-secondary leakage monitoring program is described in detail in our November 11 and November 20, 1987 letters and was implemented prior to return to power following the fall 1987 refueling and maintenance outage.

Since implementation of the enhanced leakage monitoring program, we have re-evaluated the precision and accuracy of leak rate determinations based upon radiochemical sample analysis. These evaluations indicate an uncertainty of approximately 20% with a reproducibility of about 1 to 2 gallons per day. We have also compared leak rate determinations from radiochemical sample analysis to those determined from Unit 2 air ejector monitor readings. These comparisons indicate that leak rate determinations from air ejector monitor readings are consistently and conservatively higher than those from radiochemical sample analysis.

Assuming a tube fatigue failure is in progress with time dependent leakage characteristics as shown in Figure 1 of Bulletin 88-02, the time between reaching 100 gallons per day (our administrative requirement for evaluating the need to reduce power) and total tube rupture is about 15 hours. Applying a 5 hour margin to rupture and a leak rate determination uncertainty of 20%, about 8 hours are available for further leak rate trend and stability evaluations, confirmation of leak rate magnitudes by other means, and evaluations of the need to reduce power or shut down.

A power reduction to 50% power could be accomplished in less than one hour, if necessary. Even if the entire 8 hours available were used for further evaluations, a 50% reduction in power could be accomplished prior to exceeding the existing technical specification limit of 500 gallons per day. Thus, in view of the conservatisms in our enhanced leak rate monitoring program and the evaluations of tube fatigue susceptibility discussed in items C.2.a and C.2.b below, it is not necessary to impose absolute leak rate limits for commencing plant shutdown or time limitations for reducing power or reaching cold shutdown. It should be noted that when leak rate determinations are from air ejector monitor readings, even more margin to tube rupture exists because of the higher leak rate estimates using this monitor.

C.2.a and C.2.b Detailed Analyses to Assess the Potential For Tube Failure

Evaluations to assess the potential for high cycle fatigue failure were performed with Westinghouse Electric Corporation during the fall 1987 refueling and maintenance outage and detailed reports were submitted in WCAP-11666 and WCAP-11667. These evaluations specifically address the requirements of C.2.a and C.2.b of Bulletin No. 88-02.

Subsequent to these submittals Westinghouse completed three refinements to the Unit 2 evaluations. These refinements include 1) the best estimate value of the stability ratio (without peaking factor) for the North Anna R9C51 tube using an ATHOS model of the North Anna steam generator with cartesian coordinates; 2) an updated value of flow peaking for North Anna R9C51 tube; and 3) the effect of operation of Unit 2 during coast-down periods at the end of fuel cycles. The overall effect of these refinements on the fatigue useage, stability ratios, and stress ratios for the Unit 2 steam generator tubes is a reduction in the corresponding values reported in WCAP-11666 and WCAP-11667. This demonstrates a greater margin to the lower limit for susceptability.

The initial three dimensional evaluations for North Anna were performed using an ATHOS model in cylindrical coordinates. These evaluations were repeated using an ATHOS model in cartesian coordinates, including a fine resolution cell mesh, for the peripheral columns of tubes in Rows 9 to 12. The cartesian coordinate model allows better simulation of the U-tube region including the AVB's. As a result, a more accurate solution is obtained using this model.

Modifications were made to the air test model used to determine flow peaking factors resulting from non-uniform AVB insertion. These modifications eliminated air leakage paths that occurred in the original tests near slots used to adjust AVB positions in the model. All AVB configurations were retested to obtain the final test values for the peaking factors. In addition, a detailed uncertainty evaluation was performed and the results were used conservatively to adjust the peaking factors used for the tube fatigue analyses. Included in the uncertainty evaluation were 1) a reassessment of the North Anna R9C51 AVB configuration using AVB extrapolation methods developed since the original North Anna evaluation; 2) test uncertainties from extrapolating the cantilever configuration in the air test model to an actual U-bend; 3) uncertainties from extrapolating the air test to a steam/water mixture; and 4) uncertainties in AVB position, particularly for low flow peaking configurations. The overall result of the air

test model modification and uncertainty evaluation was an increase of 9% in the North Anna R9C51 flow peaking factor. This increase leads to a reduction in the maximum relative stability and stress ratios for Unit 2 steam generator tubes and demonstrates a greater margin to the lower limit for susceptibility.

The following table presents the results of applying the revised values for flow peaking and stability ratio for North Anna R9C51. For the purpose of comparison, this list of tubes and associated revised relative stability ratios and stress ratios also contains the corresponding values as presented in Table 8-2 of WCAP-11666 and WCAP-11667. All stress ratios have decreased which demonstrates the conservatism of the original evaluation results. This is a direct result of the decrease in the relative stability ratio by including flow peaking effects. The largest relative stability ratio is now 0.82 as compared to the prior value of 0.88.

STEAM GENERATOR A

| | Relative Stability Ratio | | Stress Ratio | |
|---|---|---|---|---|
| Tube | Original | Revised | Original | Revised |
| R12C2 | 0.84 | 0.79 | 0 48 | 0.34 |
| R12C91 | 0.84 | 0.79 | 0.48 | 0.34 |
| R11C2 | 0.61 | 0.57 | 0.09 | 0.06 |
| R11C3 | 0.86 | 0.81 | 0.65 | 0.46 |
| R11C4 | 0.88 | 0.82 | 0.76 | 0.49 |
| R11C91 | 0.61 | 0.57 | 0.09 | 0.06 |
| R10C3 | 0.71 | 0.67 | 0.24 | 0.17 |
| R10C4 | 0.70 | 0.66 | 0.21 | 0.16 |
| R10C5 | 0.69 | 0.65 | 0.20 | 0.14 |

**STEAM GENERATOR B**

| Tube | Relative Stability Ratio | | Stress Ratio | |
|---|---|---|---|---|
| | Original | Revised | Original | Revised |
| R12C2 | 0.84 | 0.79 | 0.48 | 0.34 |
| R12C91 | 0.84 | 0.79 | 0.48 | 0.34 |
| R11C2 | 0.61 | 0.57 | 0.09 | 0.06 |
| R11C91 | 0.61 | 0.57 | 0.09 | 0.06 |
| R10C4 | 0.70 | 0.66 | 0.21 | 0.16 |
| R10C5 | 0.82 | 0.77 | 0.57 | 0.39 |
| R10C90 | 0.71 | 0.67 | 0.24 | 0.17 |

Note: Ratios are relative to North Anna Unit 1 R9C51.

Westinghouse has developed conservative acceptance criteria to determine susceptibility to high cycle fatigue failure of steam generator tubes as referenced to North Anna R9C51. Specifically, the acceptance criteria for a tube which is dented at the top tube support plate and without AVB support are 1) a relative stability ratio which is less than or equal to 0.9 and 2) a relative stress ratio less than or equal to 1.0. The results described above are well within these criteria.

In the tube fatigue evaluation of the Unit 2 steam generators presented in WCAP-11666 and WCAP-11667, the effect of reactor operation during coast-down periods at the end of fuel cycles was not evaluated. Coast-down periods involve plant operation at reduced primary coolant temperatures and possibly reduced reactor power at the end of the fuel cycle.

Operation at reduced primary temperatures results in lower steam pressures, lower U-bend density, higher U-bend velocities, and reduced damping as a result of higher void fractions in the U-bend. The analysis described below was performed to identify the effect on fluidelastic vibration and hence on tube fatigue resulting from the end of fuel cycle coast-down periods which have occurred for Unit 2.

Using plant data for each of eight coast-down periods the higher powered of the two steam generators was selected for evaluation. A one-dimensional analysis was performed for the beginning and the end of each period and a "Ratio of Stability Ratio, End/Start" was calculated. A value less than one indicates that the relative stability ratio became smaller and hence more favorable during the coast-down period. A value greater than one indicates that the

relative stability ratio became larger during the coast-down period. The results of these analyses are summarized in the following table.

**POINT BEACH UNIT 2**
**COAST-DOWN ANALYSIS RESULTS**

| Fuel Cycle Number | Ratio of Stability Ratio End/Start | Fuel Cycle Number | Ratio of Stability Ratio End/Start |
|---|---|---|---|
| 1 | 1.071 | 5 | 0.991 |
| 2 | 0.946 | 6 | 0.983 |
| 3 | 1.014 | 8 | 0.982 |
| 4 | 0.993 | 10 | 0.979 |

The values of Ratio of Stability Ratio End/Start show that during six of the coast-down periods the relative stability ratio decreased, and in two cases the relative stability ratio increased slightly. In the original analysis, fatigue usage was calculated by assuming that all fuel cycles operated at the highest stress level calculated for any fuel cycle since plant startup (3.04 ksi for Cycle 13). Since this results in a conservative estimate of usage and since the increase in stress caused by the two coastdown periods with small increases in stability ratios is small and less than the conservative case, it is concluded that the effects of a coastdown analysis are bounded by the original analysis.

The steam flow rates for the starting conditions of some of the coast-down periods exceeded the original analysis reference steam flow of $3.25 \times 10^6$ lb/hr. The differences between the 100% power conditions are believed to be the result of a combination of physical variability between steam generators and steam flow measurement uncertainties. The starting conditions for the coast-down periods listed in the above table can result in a 3.5% increase in the relative stability ratio as compared to the reference condition. However, as described above, the evaluation for the reference condition has been revised to update the relative stability ratios and stress ratios by incorporating a higher flow peaking factor for the North Anna, R9C51 tube. The general

effect of that revision is a 6.3% reduction in relative stability ratios. The effect of the higher than reference steam flow rates on relative stability ratios and stress ratios is therefore bounded by the original evaluation which incorporates the lower flow peaking factor for the North Anna, R9C51 tube.

Based on our evaluations, as originally presented, and as supplemented in accordance with the considerations discussed above, no tubes in the Unit 2 steam generators are susceptible to tube fatigue failure similar to that which occurred at North Anna Unit 1. Therefore, no modifications, preventive tube plugging, or other preventive actions to preclude such an event at Point Beach are required.

Please contact us if you have any questions regarding our response.

Very truly yours,

C. W. Fay

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Subscribed and sworn to before me
this 25th of March, 1988

Delores B Grzyzkowski
Notary Public, State of Wisconsin

My Commission expires 5/27/90.