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U. S. Nuclear Regulatory Commission
Attn: Document Control Desk
Washington, DC 20555

Reference: Beaver Valley Power Station, Unit No. 1 and No. 2
BV-1 Docket No. 50-334, License No. DPR-66
BV-2 Docket No. 50-412, License No. NPF-73
NRC Bulletin 88-02

Gentlemen:

We have reviewed NRC Bulletin 88-02 "Rapidly Propagating Fatigue Cracks in Steam Generator Tubes" for applicability to Beaver Valley Power Station Units 1 and 2. Attached is our response to Items A, B and C of the Bulletin. As described in our response to Item C.2, we will submit an evaluation to address Items C.2.a and C.2.b for Beaver Valley Unit 1 by May 31, 1988.

If there are any questions concerning this response, please contact my office.

Very truly yours,

J. D. Sieber

J. D. Sieber
Vice President
Nuclear Group

Attachments

cc: Mr. J. Beall, Sr. Resident Inspector
Mr. W. T. Russell, NRC Region I Administrator
Mr. P. Tam, Project Manager
Director, Safety Evaluation & Control (VEPCO)

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COMMONWEALTH OF PENNSYLVANIA)
) SS:
COUNTY OF BEAVER)

On this 28th day of March, 1988, before me, Sheila M Fattore, a Notary Public in and for said Commonwealth and County, personally appeared J. D. Sieber, who being duly sworn, deposed, and said that (1) he is Vice President of Duquesne Light, (2) he is duly authorized to execute and file the foregoing Submittal on behalf of said Company, and (3) the statements set forth in the Submittal are true and correct to the best of his knowledge, information and belief.

Sheila M. Fattore

SHEILA M. FATTORE, NOTARY PUBLIC
SHIPPINGPORT BORO, BEAVER COUNTY
MY COMMISSION EXPIRES OCT. 23, 19[illegible]

DUQUESNE LIGHT COMPANY
Nuclear Group
Beaver Valley Units 1 and 2

Response to NRC Bulletin 88-02

Item A

The most recent steam generator inspection data should be reviewed for evidence of denting at the uppermost tube support plate. Inspection records may be considered adequate for this purpose if at least 3% of the total steam generator tube population was inspected at the uppermost support plate elevation during the last 40 calendar months. "Denting" should be considered to include evidence of upper support plate corrosion and the presence of magnetite in the tube-to-support plate crevices, regardless of whether there is detectable distortion of the tubes. The results of this review shall be included as part of the 45-day report. Where inspection records are not adequate for this purpose, inspections of at least 3% of the total steam generator tube population at the uppermost support plate elevation should be performed at the next refueling outage. The schedule for these inspections shall be included as part of the 45-day report and the results of the inspections shall be submitted within 45 days of their completion. Pending completion of these inspections, an enhanced primary-to-secondary leak rate monitoring program should be implemented in accordance with paragraph C.1. below.

Response

The Beaver Valley Unit #1 Fifth and Sixth Refueling Outages' eddy current data (1986 and 1987-88) were reviewed for evidence of denting and support plate corrosion at the uppermost tube support plate. This data represented 100% of the tubes in all three steam generators. Very limited denting and support plate corrosion was identified during this review. However, since denting and support plate corrosion was not absent from the Unit 1 steam generators, an evaluation which meets the requirements of Items C.2.(a) and C.2.(b) was performed. See response for Item C.2 below.

Beaver Valley Unit 2 has no post-commercial operation eddy current data and, therefore, service induced denting and/or support plate corrosion cannot be determined. Pre-operational eddy current data does not indicate any evidence of fabrication induced denting at the uppermost support plate.

Item B

For plants where no denting is found at the uppermost support plate, the results of future steam generator tube inspections should be reviewed for evidence of denting at the uppermost support plate. If denting is found in the future, the provisions of item C below should be implemented. Commitments to implement these actions shall be submitted when the results of A above are submitted.

Response

The pre-operational eddy current data for Beaver Valley Unit 2 steam generators did not indicate any evidence of denting at the uppermost support plate. Therefore, future Unit 2 steam generator tube inspections will be reviewed for evidence of denting at the uppermost support plate. If denting is found in the future, the provisions of Item C of the Bulletin will be implemented.

Item C.1

For plants where denting is found, the NRC staff requests that the following actions be taken:

1. Pending completion of the NRC staff review and approval of the program described in C.2 below or completion of inspections specified in item A above to confirm that denting does not exist, an enhanced primary-to-secondary leak rate monitoring program should be implemented as an interim compensatory measure within 45 days of the date of receipt of this bulletin. Implementation of this program shall be documented as part of the 45-day report. The enhanced monitoring program is intended to ensure that if a rapidly propagating fatigue crack occurs under flow-induced vibration, the plant power level would be reduced to 50% power or less at least 5 hours before a tube rupture was predicted to occur. The effectiveness of this program should be evaluated against the assumed time-dependent leakage curve given in Figure 1.

   This program should consider and provide the necessary leakage measurement and trending methods, time intervals between measurements, alarms and alarm setpoints, intermediate actions based on leak rates or receipt of alarms, administrative limits for commencing plant shutdown, and time limitations for (1) reducing power to less than 50% and (2) shutting down to cold shutdown. Appropriate allowances for instrument errors should be considered. Finally, the program should make provision for out of service radiation monitors, including action statements and compensatory measures.

Response

Beaver Valley Unit 1 has implemented Chemistry Manual Chapter 5P1 "Enhanced Primary to Secondary Leakrate Monitoring Program". The purpose of this program is to ensure that if a rapidly propagating fatigue crack occurs, then plant power level would be reduced to 50% power or less at least 5 hours before a tube rupture is predicted to occur.

The program provides methods and schedules for monitoring of primary to secondary leakage. Under normal circumstances (no primary to secondary leakage and radiation monitors RM-SV-100, RM-BD-101 and RM-SS-100 in service), each steam generator will be sampled for isotopic activity three (3) times per week. If radiation monitor RM-SV-100, RM-BD-101 or RM-SS-100 is placed out of service for greater than 8 hours, then the steam generators or monitor effluent will be sampled once per day for isotopic activity. If all three of the above radiation monitors are placed out of service, then sampling of the steam generators or monitor effluent once per shift for gross gamma activity will be initiated. Also, the sampling frequency of each steam generator for isotopic activity will be increased to once per day.

If there are indications of primary to secondary leakage, then samples will be taken to identify which steam generator is leaking. Any steam generator with a confirmed primary to secondary leak will be sampled at least once per day until the leak is repaired. Any leakrate data will be evaluated and trended against the assumed time dependent leakage curve given by Figure 1 of the Bulletin. Isotopic samples of the affected steam generator(s) will be taken as necessary to make this comparison.

The program provides administrative limits for commencing power reductions or plant shutdown based on the evaluation of leakage rate. If a primary to secondary leakrate reaches 250 gallons per day, a plant power reduction to 50% power or less will be commenced within one (1) hour. If the leakrate reaches 300 gallons per day, the plant will be placed in cold shutdown within 30 hours.

This program is an interim compensatory measure and will remain in effect until NRC review and approval of our actions to minimize the probability of a rapidly propagating fatigue failure of a steam generator tube as described in our response to Item C.2 below.

Item C.2

A program should be implemented to minimize the probability of a rapidly propagating fatigue failure such as occurred at North Anna Unit 1. The need for long-term corrective actions (e.g., preventive plugging and stabilization of potentially susceptible tubes, hardware and/or operational changes to reduce stability ratios) and/or long-term compensatory measures (e.g., enhanced leak rate monitoring program) should be assessed and implemented as necessary. An appropriate program would include detailed analyses, as described in subparagraphs (a) and (b) below, to assess the potential for such a failure. Alternative approaches and/or compensatory measures implemented in lieu of the actions in subparagraphs (a) or (b) below should be justified.

Although the 45-day report shall provide a clear indication of actions proposed by licensees, including their status and schedule, a detailed description of this program and the results of analyses shall be submitted subsequently, but early enough to permit NRC staff review and approval prior to the next scheduled restart from a refueling outage. Where the next such restart is scheduled to take place within 90 days, staff review and approval will not be necessary prior to restart from the current refueling outage. An acceptable schedule for submittal of the above information should be arranged with the NRC plant project manager by all licensees to ensure that the staff will have adequate time and resources to complete its review without adverse impact on the licensee's schedule for restart.

(a) The analysis would include an assessment of stability ratios (including flow peaking effects) for the most limiting tube locations to assess the potential for rapidly propagating fatigue cracks. This assessment would be conducted such that the stability ratios are directly comparable to that for the tube which ruptured at North Anna.

(b) The analysis would include an assessment of the depth of penetration of each AVB. The purpose of this assessment is twofold: (1) to establish which tubes are not effectively supported by AVBs and (2) to permit an assessment of flow peaking factors.

<u>Response</u>

Since Beaver Valley Unit 1 has evidence of dented tubes and support plate corrosion, Westinghouse was contracted to perform an evaluation for the susceptibility of the Unit 1 steam generator tube bundles to a North Anna type tube rupture event. This evaluation included three-dimensional flow analysis of the tube bundle, air tests performed to support the vibration analytical procedure, field measurements to establish AVB locations, structural and vibration analysis of selected tubes, and fatigue usage calculations to predict cumulative usage for critical tubes. The evaluation utilized operating conditions specific to Beaver Valley Unit 1 in order to account for plant specific features of the tube bundle loading response. The calculated stability ratios were directly compared to the stability ratio for the row 9, column 51 North Anna tube that ruptured. It was conservatively assumed in the analysis that all unsupported tubes in rows 8 through 12 were structurally "fixed" in the tube support plate holes, as if by denting or support plate corrosion. The results of this analysis concluded that five tubes in "C" steam generator and one tube in "A" steam generator exceeded the acceptance criteria of:

1. Stability ratio less than .9 x S.R. of North Anna row 9 Column 51.

2. Stress ratio less than 1.0

These tubes were preventively plugged with leak lim ing 'ugs. These plugs are designed to limit leakage through any c) of these tubes to approximately 300 gallons/day maximum. See Tab.: 1 for the location of leak limiting plugs. Recovery of the leak limiting plugs and the installation of stabilizers and conventional plugs in the susceptible locations will be explored for possible implementation during the 7th refueling outage.

We will submit our evaluation of the susceptibility of the Unit 1 steam generator tubes to rapidly propagating fatigue cracks as described above, for NRC review by May 31, 1988. This schedule has been discussed with our project manager, Mr. Peter S. Tam, and found acceptable.

TABLE 1

LOCATION OF LEAK LIMITING PLUGS

'A' STEAM GENERATOR

Row	Column
10	43

'B' STEAM GENERATOR

Row	Column
--- NONE	-----

'C' STEAM GENERATOR

Row	Column
10	60
11	41
11	51
11	52
12	51