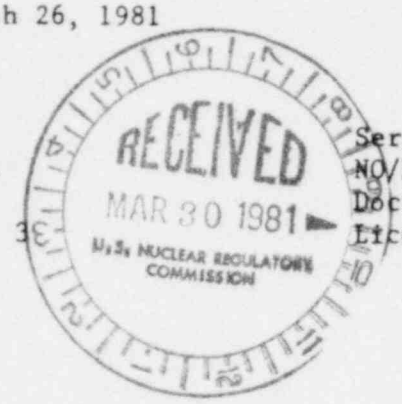
VIRGINIA ELECTRIC AND POWER COMPANY
RICHMOND, VIRGINIA 23261

March 26, 1981



Mr. Harold R. Denton, Director
Office of Inspection and Enforcement
Attn: Mr. Robert A. Clark, Chief
Operating Reactors Branch No. 3
Division of Licensing
U. S. Nuclear Regulatory Commission
Washington, D. C. 20555

Serial No. 194
NO/RMT:ms
Docket No. 50-338
License No. NPF-4

Gentlemen:

NORTH ANNA POWER STATION UNIT 1
STEAM GENERATOR INSERVICE INSPECTION
PRELIMINARY REPORT

Enclosed are the results of the second inservice eddy current examination of steam generators during the current Cycle II-III refueling outage.

A random sampling of unsaturated data from the first inservice inspection of each of the three (3) steam generators has been compared to coinciding data from this inspection period and it is apparent that denting of the steam generator tubing has not increased since the initial inservice examination.

A more detailed report, containing a listing of tubes and the condition of each is being compiled. This report, along with all other pertinent information and all collected data, will be on file at the station. This preliminary report is being submitted pursuant to North Anna Power Station Technical Specification 4.4.5.5.b.

Very truly yours,

B. R. Sylvia
Manager - Nuclear
Operations and Maintenance

Enclosure

cc: Mr. James P. O'Reilly

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NORTH ANNA POWER STATION
UNIT NO. 1
STEAM GENERATOR INSERVICE INSPECTION
PRELIMINARY REPORT

The second inservice eddy current examination of steam generators A, B, and C was performed during the current Cycle II-III refueling outage.

The tubes chosen for examination were the tubes that were previously inspected during the first inservice inspection. There were some additional tubes inspected to standardize programs in all three (3) generators. The examinations far exceeded the requirements of Regulatory Guide 1.83 in that approximately 13% of each tube bundle was examined.

The tubing was examined using a multifrequency eddy current system. Frequencies used for the examination were 400Khz and 100Khz differential to detect possible tube defects. A 100Khz absolute test was performed to supplement information gained from the 400 and 100Khz examination, and 10Khz was used to determine support plate corrosion or possible ligament cracking. A mix of the 400Khz and 100Khz data was also used as an aid to help eliminate tube support signals.

Regarding your review of tube denting dated November 21, 1980 in which it was concluded that denting was the result of acidic contamination caused by the decomposition of condensate polisher ion exchange resin which had been discharged into the Steam Generators in 1979, the following comments are provided.

Reports included in the review were provided by NWT Corporation (NWT-160) and Electric Power Research Institute (RP-623-1). These reports did give supportive evidence to the probability of corrosion/denting occurring in Steam Generators due to resin ingress. The extent to which denting would have occurred is dependent on the quantity and frequency of ingress as well as the total operating time under these conditions. The reports also evaluated other contributing areas of concern pertaining to steam generator corrosion such as resin throw, condenser inleakage and corrosion product transport. Another contributor is plant design.

Since North Anna Unit No. 1 has, at one time or another, experienced several of these problems, it would be difficult to pin point the cause of denting as resin ingress alone as was indicated in your safety evaluation. Based on the findings reported in these studies, we would have to take exception to the conclusion that resin ingress is the cause of the steam generator tube problems.

The results of each steam generator examination are as follows:

"A" STEAM GENERATOR

A total of 485 tubes were examined in the inlet side of "A" steam generator. These tubes were examined to the 7th support of the outlet side. There were no tubes with discontinuity indications greater than or equal to 20% thruwall. Three (3) tubes would not pass a .700 probe and were re-examined with a .650 probe.

All of the Row 2 tubes were examined except for six (6) tubes that were located under the fixture.

Of the 485 tubes examined, data was collected on 485 tubes for ligament defect detection. 202 tubes, or approximately 42% were observed to have possible ligament defects.

191 tubes were examined in the outlet side of "A" steam generator as per a Westinghouse recommendation to evaluate denting. These tubes were examined to the third tube support using a .720 O.D. probe. Of the 191 tubes examined, no significant denting was observed. No tubes were found to have discontinuity indications greater than or equal to 20% thru-wall.

"B" STEAM GENERATOR

A total of 476 tubes were examined in "B" steam generator. There were no discontinuity indications greater than or equal to 20% thru-wall observed. One (1) tube would not pass a .700 probe and was re-examined using a .650 probe. All of the Row 2 tubes were examined with the exception of twelve (12) tubes. Six (6) were under the fixture and six (6) tubes were not examined due to lack of probes at the time.

Of the 476 tubes examined, data was collected on 438 tubes for ligament defect detection. 159 tubes, or approximately 33%, were observed to have possible ligament defects.

"C" STEAM GENERATOR

A total of 478 tubes were examined in "C" steam generator. There were no discontinuity indications greater than or equal to 20% observed. One (1) tube would not pass a .700 probe and was re-examined using a .650 probe. All of the Row 2 tubes were examined with the exception of six (6) tubes that were beneath the fixture.

Of the 478 tubes examined, data was collected on 345 tubes for ligament defect detection. 185 tubes, or approximately 39%, were observed to have possible ligament defects.