Virginia Electric and Power Company
Richmond, Virginia 23261

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U.S. Nuclear Regulatory Commission
Attention: Document Control Desk
Washington, D.C. 20555

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Gentlemen:

VIRGINIA ELECTRIC AND POWER COMPANY
NORTH ANNA POWER STATION UNIT 1
STEAM GENERATOR OPERATING CYCLE EVALUATION

By letter dated March 7, 1991, the NRC provided formal approval for restart of North Anna Unit 1 from the 1991 refueling outage. As stated in the NRC's Safety Evaluation Report, several issues remained opened and are required to be resolved before full-cycle operation of the North Anna Unit 1 steam generators can be approved by the NRC. As a result, the NRC concluded that there was only adequate technical basis at that time to justify plant restart and operation of the North Anna Unit 1 steam generators for up to 10 calendar months. The purpose of this letter is to provide the results of our steam generator operating cycle evaluation.

The steam generator evaluation technically supports a full cycle of operation. Therefore, based on the results of the evaluation, we conclude that a mid-cycle steam generator inspection outage is not necessary and we request NRC approval to operate North Anna Unit 1 for a full cycle of operation.

Attachment 1 is a summary report of the North Anna Unit 1 steam generator operating cycle evaluation. Attachment 2 to this letter provides ten copies of Westinghouse report, NSD-TAP-1093, "North Anna Unit 1 Steam Generator Operating Cycle Evaluation." This report contains information which is proprietary to Westinghouse Electric Corporation. Accordingly, it is requested that this information be withheld from public disclosure. Westinghouse will comply with the requirements of 10 CFR 2.790 to provide proprietary and non-proprietary versions of the above material together with an affidavit as soon as the proprietary information contained in the submittal has been specifically identified and the proprietary and non-proprietary versions have been prepared. Copies of the proprietary and non-proprietary versions of the information and the required affidavit will be submitted within four weeks. Attachment 3 provides a Westinghouse letter requesting the Westinghouse report, NSD-TAP-1093 (Attachment 2), be withheld as proprietary information.

NOTE: NRC PDR
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Correspondence with respect to the proprietary aspects of this submittal should be addressed to R. P. DiPiazza, Manager of Operating Plant Licensing Support, Westinghouse Electric Corporation, P.O. Box 355, Pittsburgh, Pennsylvania 15230-0355.

Please note that based on the attached evaluation and our projection for future tube plugging, the schedule for replacement of the North Anna Unit 1 steam generators has been moved forward from 1995. As Mr. M. L. Bowling of my staff discussed with Mr. H. N. Berkow of your staff on July 23, 1991, the North Anna Unit 1 steam generators are now scheduled to be replaced during 1993.

Should you have any questions or require additional information, please contact us.

Very truly yours,

W L Stewart

W. L. Stewart
Senior Vice President - Nuclear

Attachments

1. Steam Generator Operating Cycle Evaluation Summary Report
2. Westinghouse Electric Corporation Report, NSD-TAP-1093, "North Anna Unit 1 Steam Generator Operating Cycle Evaluation" (10 Copies).
3. Westinghouse letter, Proprietary Information Withholding Request

cc: U.S. Nuclear Regulatory Commission
Region II
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Mr. M. S. Lesser
NRC Senior Resident Inspector
North Anna Power Station

ATTACHMENT 1

STEAM GENERATOR OPERATING CYCLE EVALUATION

SUMMARY REPORT

NORTH ANNA POWER STATION

UNIT 1

VIRGINIA ELECTRIC AND POWER COMPANY

ATTACHMENT 1

NORTH ANNA POWER STATION
UNIT 1

STEAM GENERATOR OPERATING CYCLE EVALUATION
SUMMARY REPORT

<u>Executive Summary</u>

The process for the North Anna Unit 1 operating cycle evaluation has been extremely detailed considering field NDE results, laboratory analysis, structural mechanics, specific characteristics of the degradation mechanisms, statistical methods, and flow induced vibration analyses. In order to establish the appropriate operating interval for the unit, the condition of the steam generators was estimated for the end of the proposed cycle of operation (September 1992) based on prior inspection results and then the capability of the steam generators was determined with respect to the guidelines of NRC Regulatory Guide 1.121.

The evaluation provides a technical basis for full-cycle operation of North Anna Unit 1. It presents a strong case relative to tube burst by incorporating the uncertainties relative to eddy current and crack growth. The most significantly degraded population of tubes considering each limiting degradation mechanism was analyzed. The evaluation provides a probabilistic assessment with respect to vibration induced propagation of circumferential cracks considering both turbulence and fluidelastic components and their effect relative to crack angle and depth. For tube vibration, the evaluation identifies the peripheral zone between the tubesheet and first tube support plate and a very limited population of tubes at the top support plate as the most limiting areas of the steam generator tube bundle.

North Anna Unit 1 has extremely low, administratively controlled, primary-to-secondary leakage rate limits and operating constraints. The N-16 monitors are the best equipment available to monitor such leakage. Should a tube in the peripheral zone between the tubesheet and first tube support plate develop a circumferential corrosion crack of sufficient size, the evaluation indicates that turbulence would be the predicted initial propagating mechanism. In the event that a crack were propagated by turbulence, the evaluation indicates that the increase in leakage and propagation rate would provide sufficient response time to take appropriate operator actions. For the top support plate, the small population of susceptible tubes and the lack of any circumferential degradation at elevations above the fifth support plate in Unit 1 make it extremely unlikely that tube leakage would occur.

Based on the results of our operating interval assessment, we find no technical reason to preclude a full cycle of operation and we conclude that a mid-cycle outage for North Anna Unit 1 is not necessary.

Background

During the 1991 refueling outage for North Anna Power Station Unit 1, an extensive eddy current inspection program was conducted on the tubes in each of the three steam generators. The results of the inspections were classified as Category C-3 for each of the unit's steam generators, i.e., greater than 1% of the inspected tubes were defective and required plugging. Based on these inspection results, NRC approval for resumption of power operation was required in accordance with North Anna Unit 1 Technical Specification Table 4.4-2. In addition, a tube segment was pulled from "B" steam generator to further examine circumferentially oriented indications at the tube to tube support plate intersections.

On February 26, 1991, the inspection results, our technical evaluation, and our basis for restart and operation of the unit were discussed in detail at a meeting with the NRC. Later that day, by letter dated February 26, 1991, Virginia Electric and Power Company requested NRC approval for restart of North Anna Unit 1. NRC approval for resumption of power operation was granted by letter dated March 7, 1991. However, the NRC's startup approval stipulates that we must either implement a mid-cycle inspection of the steam generator tubes (i.e., perform the next inspection of the steam generator tubes at a frequency not to exceed 10 months from startup of the unit) or provide the additional information necessary to justify a full cycle of operation.

As stated in our February 26, 1991 letter, we committed to provide the NRC with the results and assessments of the analysis on the tube segment pulled from the "B" steam generator. In addition, we identified that assessments must be made of the potential for crack propagation of circumferential cracks due to tube vibration, the potential for burst pressure reduction due to combined circumferential and axial cracks, and the potential leakage during a postulated steamline break for the projected end-of-cycle crack distribution.

Extent of Inspection Program

During the 1991 refueling outage, an inspection program was conducted which relied on diverse inspection methods and which focused on areas of the steam generator tubing which had previously experienced tube degradation (i.e., the tube to tube support plate intersections and the top of the tubesheet region). The scope of this program was conservative with respect to the requirements of the North Anna Unit 1 Technical Specifications and NRC Regulatory Guide 1.83.

One of the examination methods, the 8x1 probe, was used as a screening tool for the tube support plate areas on the hot leg side. The 8x1 probe was used because it proved useful in identifying volumetric indications previously seen in the tube support plate regions. The rotating pancake coil (RPC) was then used to disposition distorted indications (DIs) and possible indications (PIs) identified by the 8x1 probe examination of these areas. In addition, the RPC probe was used to examine other suspect areas such as the WEXTEX expansion zone at the top of the tubesheet (hot leg side) for circumferential indications.

Tube R11-C14 from the "B" steam generator was identified during the inspection as one of 92 tubes with potential circumferentially oriented degradation. This tube was cut below the second support plate on the hot leg side and the lower segment was removed for examination to characterize the corrosion cracking suspected to exist at the first support plate. The results of the examination of the pulled steam generator tube were discussed with the NRC in conference calls held on March 8, 1991 and April 26, 1991, by our letter to the NRC, Serial No. 91-267, dated May 7, 1991, and discussed in detail at our update meeting on May 30, 1991 at the NRC offices in White Flint.

Tube Burst Capability Assessment

In previous years, assessments of tube integrity for North Anna were performed for axial cracks. The results of these assessments have been previously presented to the NRC. Our current efforts have focused on circumferential cracking at the WEXTEX transition and at the tube support plate elevations.

As part of our evaluation, we have conducted an assessment of estimated tube integrity for the end of the operating cycle. Not only were the average tube conditions assessed, but the expected population and condition of the most significantly degraded tubes were determined. The assessment methodology employed to determine expected end-of-cycle tube degradation encompassed the estimated threshold of detection, estimated growth rate, and eddy current uncertainty as required by NRC Regulatory Guide 1.121.

Conservative models were developed for each circumferential degradation mechanism. A model for circumferential outside diameter stress corrosion cracking (ODSCC) at the support plates and for circumferential primary water stress corrosion cracking (PWSCC) at the WEXTEX transition was utilized, incorporating the unique characteristics of growth, eddy current uncertainty, and detection threshold for each. Results of recent work with the Westinghouse Owner's Group WEXTEX subgroup have been utilized in developing these models. As a result of this assessment, we concluded that all tubes (with projected end-of-cycle cracks using a 95% cumulative probability level) could withstand three times the normal operating condition pressure loadings.

We have also assessed the occurrence of combined cracks (axial and circumferential) at the support plate and the expansion transition. Based on the RPC inspection data as well as the pulled tube examination results, it is judged that the location of axial and circumferential cracking at the same elevation will be separated such that mixed mode cracking will not occur at the tube support plate elevations. Relative to the WEXTEX expansion transition region, RPC testing results indicated the presence of axial cracks on only four tubes. None of these tubes had both orientations of cracking in the WEXTEX transition region.

Tube Vibration Analyses

Vibration analyses were performed for postulated circumferential degradation that could occur in the unit. The most limiting areas that exist in the steam generator were determined to be the top of the tubesheet, the bottom of the first support plate, and a limited number of unsupported tubes at the top (seventh) support plate. For the tubesheet to first support plate area, the peripheral tubes are subject to the highest vibration loadings. Potential crack propagation by two mechanisms, turbulence and fluidelastic excitation, was reviewed for the lower region of the steam generator.

The first mechanism, turbulence, is characterized by substantially lower amplitudes and loadings. As a turbulence propagated crack is advanced, it has the potential to reach an extent where fluidelastic excitation may become the driving mechanism. Fluidelastic excitation is characterized by higher loadings and amplitudes, providing the driving force to rapidly propagate an existing crack as cited in NRC Bulletin 88-02. From our evaluation, turbulence is the expected initial activating mechanism, providing the initial leakage which permits timely shutdown of the unit with substantial margin. However, based on conservative projections for the end-of-cycle cracks and their associated distributions, no tubes are predicted to exceed the through-wall angular threshold for crack propagation by turbulence.

For the top support plate, the most limiting tubes in service from the NRC Bulletin 88-02 evaluation were identified (reference WCAP-12351 entitled, North Anna Unit 1 Evaluation for Tube Vibration Induced Fatigue, dated 1989). The majority of circumferential indications from the last inspection are located in the lower tube support plate areas. Only a few indications were observed at the fifth tube support plate, with none observed at the higher tube support plates. Based upon past experience, it is extremely unlikely that a circumferential indication would exist at the top support plate location. Even if this low probability event were to occur, North Anna has implemented a leakage monitoring program capable of timely shutdown of the unit.

Conservative Leakage Limits and Operating Constraints

We have implemented additional administrative operating limitations for action to be taken if significant primary-to-secondary leakage is detected. These guidelines involve close monitoring of primary-to-secondary leakage and prompt operator action.

By license amendment dated December 12, 1988, Specification 3.4.6.3 was added to the North Anna Units 1 and 2 Technical Specifications to implement more stringent primary-to-secondary leakage limits and Technical Specification 3.4.6.4 was added to establish surveillance instrumentation requirements necessary to assure compliance with those leakage limits. The applicability of these Specifications is MODE 1 above 50% power. The Specifications were added to ensure prompt operator action in response to a potential tube leak at the top tube support plate due to fatigue.

In 1989, more conservative primary-to-secondary leakage limits were administratively implemented. These administrative limits were implemented to address a concern for prompt operator action as a result of any tube leak indication. The administrative controls limit primary-to-secondary leakage to 150 GPD total from all steam generators or 50 GPD from an individual steam generator. Originally, the administrative controls required that if either of these limits were exceeded, power would be reduced to less than 50% power within 90 minutes.

In response to recent concerns of circumferentially oriented tube degradation at the lower tube support plates (and as discussed in our February 26, 1991 letter), these administrative controls and required actions have been made more conservative. The leakage limits previously established are maintained. However, the operating limitations are now applicable to operation at all power levels in MODE 1. If either limit is exceeded or will apparently be exceeded, then the operators are required to reduce power to less than 50% power within 90 minutes and below MODE 1 within two (2) hours from detection.

Leakage Detection and Response Capability

At North Anna, an N-16 radiation monitoring system was installed in 1987. This system, as currently configured, has a monitor on each main steam line and a monitor on the main steam header. This configuration provides a redundant means of determining the operating leakage from the steam generators. Additionally, the N-16 monitoring system provides real time leakage determination. The system readout permits small changes in the leakrate to be observed. Other backup or confirmatory leak detection methods include air ejector, steam generator blowdown, main steam radiation monitors, and grab sample chemistry and air ejector analyses.

Through increased frequency simulator training, control room operators have been made intimately aware and instructed to respond conservatively to indications of abnormal primary-to-secondary leakage indications. In addition, on-shift control room operators record and trend primary-to-secondary leakage from each N-16 radiation monitoring system and the condenser air ejector exhaust radiation monitor on a 4 hour surveillance interval.

Special Steamline Break Analysis

In addition to the general tube integrity assessments previously discussed, analyses were performed to assess the potential for crack propagation by vibration during a main steamline break event. For these analyses, no significant crack propagation is expected during such events.

An operational leak rate in excess of 50 GPD per steam generator will result in immediate action to shut down the unit, effectively ending the operating cycle. Thus, primary-to-secondary leakage in the event of a main steamline break is limited by the analysis that establishes the leakage rate of the most limiting crack. This potential leak rate is expected to be less than 9.5 GPM. Based on the operational leakage limit controls, it is expected that no other cracks would have propagated to the point that they would contribute more than a negligible amount of the overall leakage rate.

However, to further substantiate our conclusions, an overly conservative approach was also evaluated in that the cumulative potential leakage from a postulated main steamline break for the projected end-of-cycle (i.e., 18 months) crack distribution was assessed. This approach was evaluated to bound potential leakage in the unlikely event that a large population of tubes have through-wall or near through-wall cracks that do not leak during normal operating conditions, but may leak during a main steamline break event. Therefore, it is assumed that the 50 GPD limit is ignored and that the end-of-cycle crack distribution models are then applied. Further, a non-credible assumption is made that all cracks instantaneously become through-wall cracks. The integrated leakage rate from this overly conservative approach is calculated to be 49 GPM.

The radiological consequences of the potential 9.5 GPM post-accident primary-to-secondary leakage rate results were evaluated relative to the off-site dose assessment and control room habitability calculations. The increased leakage rate did not result in any increase in the dose consequences at the site boundary in excess of those previously evaluated and approved by the NRC for North Anna. It was concluded that no unreviewed safety question existed.

Consideration was also given to the control room operator dose consequences to the thyroid, skin and whole body. The dose consequence results to control room operators were calculated as follows:

Dose Type	1 GPM Leakrate Dose (Rem)	10 GPM Leakrate Dose (Rem)	GDC-19 Allowable Limit (Rem)
Thyroid	24.0	20.3 *	30.0
Skin	0.0278	0.0310 *	30.0
Whole Body	0.00141	0.0396 *	5.0

* Includes 20% conservatism

It was observed that the limiting dose to control room operators as a result of the revised leak rate remained the thyroid dose. Although the skin dose appeared to increase, additional conservatism in the new analysis accounted for this change. The whole body dose also increased. However, the whole body dose consequences were not significant (in absolute terms) and thyroid dose is clearly the limiting dose to control room operators when compared to the GDC-19 allowable limits. Therefore, it was determined that the increased leakage rate did not result in increasing the limiting dose consequences to control room operators in excess of that previously evaluated and approved by the NRC and it was concluded that no unreviewed safety question existed.

Conclusion

Based on the current status of our operating interval assessment, we believe that a mid-cycle outage is not necessary. Four tiers of conservatism exist in our technical approach and evaluation. They are:

- Conservative inspection

- Conservative analysis methodology

- Conservative, state-of-the-art leakage monitoring and detection capability

- Conservative leakage limits and operating restrictions

The first tier of conservatism is in the inspection program. A diverse inspection program was performed that utilized standard bobbin probe, 8x1 probe, and rotating pancake coil probe inspection methods. Based upon these inspections, corrective actions such as tube plugging or stabilizing have been performed. This re-established tube bundle integrity for beginning the current cycle of operation.

The second tier deals with the analysis methods used to determine tube integrity through the operating interval. The methodology employed is consistent with criteria contained in Regulatory Guide 1.121. An expected distribution of circumferential indications was developed for the next outage. Threshold of detection, uncertainty, and growth rates were incorporated to develop this distribution. To confirm the methods used, the 1989 inspection data was reviewed and an end-of-cycle (1991) distribution was predicted. This predicted distribution was consistent with the actual distribution encountered during the 1991 outage. For the 1992 refueling outage, it is projected (using a 95% cumulative probability level) that all tubes are expected to meet three times normal operating differential pressure loadings.

Potential tube vibration mechanisms were also investigated. The most limiting areas for potential vibration induced propagation of existing cracks were identified. The probability of having a tube sufficiently degraded to propagate by vibration is extremely low. In the event this were to occur, turbulence is the expected initiating vibration mechanism, providing adequate time to detect the leak and shut down the unit. For a few tubes at the top support plate, fluidelastic vibration is predicted to be the initiating mechanism. However, in accordance with the NRC Bulletin 88-02 analysis, only a small population of tubes are potentially susceptible (reference WCAP-12351). As no circumferential degradation has been observed at the top two support plates and with very little observed at the fifth support plate, it is extremely unlikely that this would occur in these locations during the current operating cycle.

The third tier of conservatism deals with leakage monitoring and detection. North Anna Technical Specifications require that certain primary-to-secondary leak detection systems be operable during operation. One of the key systems used is N-16. This system permits determination of real-time leak rates. The system has alert and alarm set-points and is located in the Control Room. As a result, plant operators are better able to trend leakage and take prompt corrective action should an adverse trend develop.

Finally, we have imposed conservative leakage limits and operating restrictions to ensure appropriate corrective actions are taken. The maximum primary-to-secondary leakage permissible has been established at 50 GPD, i.e., one-tenth the limit contained in most Technical Specifications. Additionally, projected rate of change limitations are contained in the Technical Specifications. In the event that either limit is exceeded, the unit is required to be off-line within two (2) hours of detection.

In conclusion, there are no technical or operating reasons to preclude a full cycle of operation for North Anna Unit 1.