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Nuclear Power Plant
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New York Power Authority

William A. Josiger
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March 24, 1988
IP3-88-023

Docket No. 50-286
License No. DPR-64

William T. Russell
Regional Administrator
U.S. Nuclear Regulatory Commission
Region I
475 Allendale Road
King of Prussia, PA 19406

Reference:
1. Letter from Mr. J. C. Brons to the NRC, dated July 21, 1987 (IPN-87-038), entitled: "Steam Generator Inspection Results"

2. Letter from Mr. J. C. Brons to the NRC, dated December 2, 1987 (IPN-87-056), entitled: "Steam Generator Leakage Surveillance"

Subject: NRC Bulletin 88-02, "Rapidly Propagating Fatigue Cracks in Steam Generator Tubes"

Dear Mr. Russell:

The purpose of this letter is to provide the Authority's response to NRC Bulletin 88-02.

As a result of the steam generator tube failure, which occurred at North Anna Power Station Unit 1 on July 15, 1987, the NRC required some PWR Licensees to initiate investigatory and corrective actions that will minimize the potential for the North Anna mode of tube failure at their facilities. Further, the NRC requires the submittal of a written report identifying the status of and schedule for compliance with the required actions.

Attachment 1 provides the Authority's response to Bulletin 88-02. We have taken a proactive approach in dealing with this issue as discussed with the NRC staff in a meeting held on November 5, 1987, and in follow-up telephone conversations. Furthermore, we have committed to an enhanced program of steam generator primary-to-secondary leak rate monitoring as documented in References (1) and (2). This program will remain in effect until the planned replacement of the steam generators in February, 1989.

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The enhanced procedures we have instituted provide the operators with the ability to take actions necessary to assure continued safe operation of Indian Point 3.

Should you or your staff have any questions or comments regarding the attached information, please contact Mr. J. A. Schivera of my staff.

Sincerely,

W. A. Josiger
Resident Manager
Indian Point Unit 3 Nuclear Power Plant

Attachments

WAJ:JAS:lh

SUBSCRIBED AND SWORN TO BEFORE ME THIS 24th DAY OF March, 1988.

Ruthanne B. Bowman

RUTHANNE B. BOWMAN
Notary Public, State of New York
No. 4828871, Westchester County
Commission Expires Oct. 31, 1989

cc: Resident Inspector's Office
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Attachment 1
Response to NRC Bulletin No. 88-02
"Rapidly Propagating Fatigue Cracks in
Steam Generator Tubes"

New York Power Authority
Indian Point 3 Nuclear Power Plant
Docket No. 50-286

I. Purpose

The purpose of this report is to apprise the Nuclear Regulatory Commission (NRC) of the status of Indian Point 3's compliance with actions required by NRC Bulletin 88-02 for minimizing the potential for rapidly propagating fatigue failure, such as occurred at North Anna 1. The required actions include:

- Identification of those tubes that are dented at the uppermost tube support plate.
- Implementation of an enhanced program of primary-to-secondary leak rate monitoring as an interim compensatory measure until long-term actions are completed or proof of no denting is provided.
- Implementation of long-term corrective actions and/or compensatory measures as necessary to minimize the probability of this event occurring.

This report addresses the NRC's concerns and provides additional information on the Authority's efforts with regard to this event.

II. Background

On July 15, 1987, a steam generator tube rupture occurred at North Anna Unit 1. The Authority has closely followed the investigation of the North Anna rupture event through industry briefings conducted by Virginia Power, Westinghouse and EPRI. Initially, little was known about the cause of the rupture other than that it was classified as a fatigue failure, initiated by an unknown mechanism.

On October 5, 1987, Westinghouse notified the Authority that they identified the cause of the North Anna rupture as high cycle fatigue. Based on their preliminary assessment, Westinghouse further indicated that Indian Point 3 was potentially susceptible to this mode of tube failure. Their preliminary conclusion was based on information indicating that the Indian Point 3 steam generator tubes may be dented at the top support plate, and that the Indian Point 3 bundle flow parameters may be greater than 90% of the North Anna values prior to the rupture. Both of these conditions were identified as contributions to the North Anna rupture.

Based on the Westinghouse assessment, the Authority performed a thermal hydraulic evaluation of the Indian Point 3 steam generators in order to determine the plant's stability ratio with respect to North Anna. The evaluation used previous calculations performed in 1983 by Combustion Engineering (C-E) for the Authority, using the C-E version of the ATHOS thermal hydraulic code. This evaluation suggested that the Indian Point 3 stability ratio was less than 90% of the North Anna value, and therefore below the susceptibility criterion established by Westinghouse.

The Authority's evaluation results were formally brought to the attention of Westinghouse for resolution. Westinghouse was not able to confirm the Authority's determination without a detailed examination of Combustion Engineering's ATHOS analysis. This disparity remains unresolved at this time.

On November 5, 1987, the Power Authority met with the NRC to discuss the susceptibility of the Indian Point 3 steam generators to the North Anna tube failure mechanism. As a result of this meeting and subsequent discussions with the NRC, an enhanced primary-to-secondary leakage rate monitoring program was established and documented by Reference (2).

Further details of specific actions and of Indian Point 3's susceptibility to the event are described below.

III. Indian Point 3 Actions and Response to NRC Bulletin 88-02

The following paragraphs provide the status of actions required by Bulletin 88-02, as identified in Section I of this Attachment:

A. Denting

Virtually all of the Indian Point 3 steam generator tubes are dented to some degree at the top support plate intersection. The extent of denting has been determined through:

(a) Gauging (go/no-go testing) performed during or as part of eddy current testing (ECT) of the steam generator tubes.

(b) Profilometry during the 1982 refueling and 1984 maintenance outages.

In addition to gauging, the Authority has performed profilometry (approximately 200 tubes) in Steam Generator No. 31 during the 1982 and 1984 inspection programs. The data indicates that all of the tubes that were inspected were dented to some degree at the sixth (top) support plate. The profilometry programs included those areas of the tube bundle where denting has been traditonally known to occur, as well as a representative sample across the entire tube bundle.

Tube sheet maps identifying the available ECT probe restriction and profilometry data were made available to the NRC's Senior Resident Inspector during December, 1987. The Authority does not plan to acquire additional denting data for the existing steam generators, in view of the plan for corrective action that is described below.

B. Plants Without Denting

Item B of the NRC Bulletin is not applicable because Indian Point 3 is dented at the top tube support plate in all steam generators.

C. Leak Rate Monitoring and Long Term Actions

1. Enhanced Primary-to-Secondary Leak Rate Monitoring Program

Indian Point 3 uses the condenser air ejector exhaust radiation monitor (R-15) and the steam generator blowdown radiation monitor (R-19) as means of detecting primary-to-secondary leakage. Additionally, grab samples are taken as confirmation for each monitor.

The condenser air ejector exhaust radiation monitor (R-15) samples the discharge from the air ejector exhaust header of the condenser for gaseous radioactive effluents and displays the results on a strip chart recorder and two analog meters (in the Control Room and locally).

The methodology and administrative controls for calculating the alarm setpoint for the condenser air ejector monitor (R-15) are listed in procedure RE-CA-110, "Determination of Rate of Leakage of Primary System to the Secondary Side of the Steam Generator." The initial alarm setpoint for R-15 has been established at a value significantly less than .01 gpm. Upon receipt of an alarm, a grab sample is drawn to confirm the alarm, and the alarm setpoint is raised in accordance with RE-CCI-008 if the alarm was valid. By taking the grab sample from the air ejector exhaust, Indian Point 3 has been able to detect leakage as low as .00001 gpm.

The steam generator blowdown radiation monitor (R-19) monitors a composite sample from the four steam generators. The alarm setpoint methodology is set in accordance with the "Offsite Dose Calculation Manual." Upon receipt of an alarm, each steam generator is individually sampled to determine the source and confirm the alarm. Subsequent actions are in accordance with Appendix B of the Technical Specifications.

Both monitors can be trended in the control room. Additionally, radiation monitors R-62A, B, C and D are located on the four individual main steam lines, and provide an additional data source.

The following enhanced primary-to-secondary leak rate monitoring procedures are in effect at Indian Point 3:

a. In response to the NRC concerns with the degradation growth rate of certain tubes during the most recent eddy current inspection in May, 1987, Reference (1) provided for the following supplementary leakage surveillance requirements for Cycle 6 plant operation:

   1. Increase the frequency of condenser air ejector exhaust isotopic sampling from monthly (per Technical Specifications) to weekly.

   2. If the condenser air ejector exhaust isotopic sample indicates a primary-to-secondary leakage of 0.1 gpm or greater, the NRC will be promptly notified of the situation and the sampling frequency will be increased from weekly to daily.

   3. If the isotopic sample indicates a primary-to-secondary leakage rate of 0.2 gpm or greater, Indian Point 3 will be shut down (using normal operating procedures) and the steam generator tubes will be inspected in accordance with the Technical Specifications (Section 4.9). Further, we have agreed to perform this inspection if the plant is shutdown to repair primary-to-secondary leakage, regardless of whether or not the leakage limit is exceeded, unless technical justification is provided to the NRC Staff indicating that the inspection is not necessary.

b. In discussions with the NRC during November, 1987, two additional temporary administrative requirements have been implemented at Indian Point 3. These changes are documented in Reference (2). They are designed to provide for early detection of a potential problem and are to remain in effect until concerns and questions related to carbon steel support plates and indeterminate bundle flow conditions are resolved.

   1. During the periods of R-15 (air ejector radiation monitor) inoperability, samples of air ejector effluent will be drawn every four (4) hours and analyzed for steam generator primary-to-secondary leakage.

2. For primary-to-secondary leakage in excess of 0.03 gpm (43.2 gpd), leak rate data will be recorded and graphically trended every two (2) hours. If three (3) consecutive trend calculations indicate that the leakage will be equal to or exceed 0.2 gpm (288 gpd) within 24 hours, the unit load will be reduced to 50% power within two (2) hours.

Paragraph C.1. and Figure 1 of Bulletin 88-02 require a power reduction to 50% power or less at least 5 hours before a tube rupture is predicted to occur. The Authority's program described in References 1 and 2 provides for an operational response that is in compliance with the Bulletin requirements. Accordingly, the Authority will continue the enhanced monitoring program already in place.

2. Long Term Corrective Actions

The Authority has reviewed the NRC's requested actions for establishing an analytical program to assess the potential for the North Anna type of tube failure. Specific Authority actions corresponding to subparagraphs C.2(a) and C.2(b) of the Bulletin are described below:

(a) Section II of this report summarized the thermal hydraulic evaluation performed by the Authority in order to determine the Indian Point 3 stability ratio with respect to North Anna. The differing result from that determined by Westinghouse remains unresolved at this time. In addition, due to the lack of AVB penetration data, the Authority's evaluation could not consider flow peaking factors as requested by subparagraph C.2(b).

(b) The Authority has attempted to determine the as-built AVB penetration depths using the baseline ECT data that was acquired during pre-service inspection of the Indian Point 3 steam generators. However, this effort proved unsuccessful, due to the ECT technique (i.e., 400 kHz, single frequency) that was employed to collect the data. In addition, there is insufficient data available from subsequent in-service inspections in the areas and rows of interest to permit determination of AVB penetration. In absence of this data, no further analyses of flow peaking factors can be performed.

The Authority's plan for steam generator replacement provides an alternate approach to the analytical program that is suggested by Bulletin 88-02. It is the Authority's plan, as presented at a July 30, 1987 meeting with the NRC in Bethesda, MD, to replace the Indian Point 3 steam generators during the next (i.e, Cycle 6/7) refueling outage that is scheduled to start during February of 1989.

Replacement of the existing units with Westinghouse Model 44F steam generators will eliminate the potential susceptibility of Indian Point 3 to this tube failure mechanism. Specific features of the Authority's replacement steam generators that would preclude a North Anna type of failure include:

(a) The tube support plates are being constructed of stainless steel with quatrefoil holes that have proven effective in minimizing the potential for denting.

(b) The AVBs will penetrate the tube bundle through row 9.

(c) The current practices of AVB installation provide for uniform insertion depth, thus minimizing the potential for local flow peaking.

If for any reason the Authority's plans for steam generator replacement are revised such that replacement is not conducted during the Cycle 6/7 refueling outage, the NRC will be notified. The Authority will reevaluate its corrective action plan at that time and discuss a revised plan with the NRC.

In summary, the Authority's plans for steam generator replacement during the next refueling outage will eliminate the potential susceptibility of Indian Point 3 to the North Anna event. In the interim, the Authority's enhancements to the Indian Point 3 primary-to-secondary leak rate monitoring procedures are consistent with the actions requested in Bulletin 88-02 and are adequate to ensure continued safe operation of Indian Point 3. As previously discussed with the NRC, the Authority has terminated its efforts to analytically determine Indian Point 3 susceptability to this tube failure mechanism.