



Consolidated Edison Company of New York, Inc.
Indian Point Station
Broadway & Bleakley Avenue
Buchanan, New York 10511-1099

December 23, 1987

Re: Indian Point Unit No. 2
Docket No. 50-247

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

SUBJECT: Susceptibility of Indian Point Unit No. 2 (IP-2) Steam
Generators to High Cycle Fatigue Tube Failure

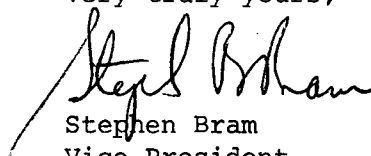
In our letters of November 23, 1987 and December 17, 1987 we submitted our evaluation of the susceptibility of steam generator U-bend tubing to fatigue failure of the type experienced at North Anna-1 on July 15, 1987. In those submittals we concluded that the tubes remaining in service are not expected to be susceptible to fatigue rupture at the top support plate in a manner similar to North Anna. In subsequent discussions with the NRC staff we indicated that our operating practices relative to primary-to-secondary leaks were more conservative than those permitted under the applicable Technical Specifications. We were asked to apprise the staff of the operating procedures which will be in place before unit startup from the current outage, including certain procedural enhancements intended to further confirm incipient indications of potential North Anna-type tube failures.

This letter describes our revised procedural enhancements to determine primary-to-secondary leak rate resulting from our review of new leak-rate versus time curves provided by the NRC staff. These enhancements were discussed with the NRC staff on December 18, 1987 and are consistent with the NRC provided leak-rate versus time curves. A description of these enhancements is contained in Attachment I to this letter. Changes to our procedures will be made before startup from the current refueling outage and provide reasonable assurance that timely corrective action would be taken before any fatigue-induced gross tube rupture similar to that at North Anna were to occur.

These procedure revisions will remain in effect pending resolution of concerns regarding the potential susceptibility of the unit's steam generators to high cycle fatigue tube failure. At such time, we will evaluate the need to continue these procedural enhancements.

If you have any questions on this matter, please contact us.

Very truly yours,



Stephen Bram
Vice President
Nuclear Power

attachment

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ATTACHMENT I

Temporary Procedural Enhancements
Steam Generator Leakage Surveillance and Corrective Action

Consolidated Edison Company of New York, Inc.
Indian Point Unit No. 2
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PROCEDURAL ENHANCEMENTS:

Chemistry procedure IPC-A-110 and operating procedure AOI 1.2 will be revised prior to startup from the current refueling outage to include the following provisions:

A. With Condenser Air Ejector Monitor R-15 Operable:

- o If the primary-to-secondary leak rate is equal to or greater than 0.05 gpm and less than 0.2 gpm, log R-15 readings and graphically plot leak rate once every two hours. If three consecutive projected leak rate calculations indicate that leakage will be equal to or exceed 0.2 gpm within 24 hours, reduce unit load to 50% within 2 hours.
- o If the primary-to-secondary leak rate is equal to or greater than 0.2 gpm, remove the unit from service and make appropriate notifications.

B. With Condenser Air Ejector Monitor R-15 Inoperable:

- o If the primary-to-secondary leak rate is less than 0.05 gpm, sample and calculate the leak rate once every four hours.
- o If the primary-to-secondary leak rate is equal to or greater than 0.05 gpm and less than 0.2 gpm, sample once every two hours and graphically plot the leak rate. If three consecutive projected leak rate calculations indicate that leakage will be equal to or exceed 0.2 gpm within 24 hours, reduce unit load to 50% within 2 hours.
- o If the primary-to-secondary leak rate is equal to or greater than 0.2 gpm, remove the unit from service and make appropriate notifications.