NRC NEWS

UNITED STATES NUCLEAR REGULATORY COMMISSION

OFFICE OF PUBLIC AFFAIRS, REGION I

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NOTE TO EDITORS:

The Nuclear Regulatory Commission staff has issued a letter to Consolidated Edison Company of New York detailing the preliminary findings of a special inspection to review the cause of the February 15 steam generator tube failure at Con Ed's Indian Point 2 nuclear power plant in Buchanan, N.Y. The letter is attached.

Separately, the NRC today issued an amendment to the Indian Point 2 technical specifications. The amendment allows Con Ed, among other things, to operate with the containment recirculation filters and charcoal adsorbers removed. The request for the amendment - submitted by ConEd in November 1999 - was intended to take advantage of updated research findings on estimated public radiation doses from reactor accidents. Copies of this amendment are available from the NRC's electronic reading room at accession number ML003727500. Copies are also available from the NRC's Office of Public Affairs.

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July 27, 2000

Mr. A. Alan Blind
Vice President - Nuclear Power Consolidated Edison Company of New York, Inc.
Indian Point 2 Station
Broadway and Bleakley Avenue
Buchanan, NY 10511

SUBJECT: PRELIMINARY RESULTS OF NRC SPECIAL INSPECTION 50-247/2000010- STEAM GENERATOR TUBE FAILURE

Dear Mr. Blind:

This letter transmits the preliminary results of a special inspection conducted to review the cause of the

February 15, 2000, steam generator tube failure at your Indian Point 2 reactor facility. We are providing these preliminary results in advance of the full inspection report since the results have the potential to influence ongoing assessments of the most recent steam generator inspections and root cause analyses. These results are subject to NRC management final review. The overall significance determination for these findings remains under evaluation.

The NRC team members included personnel from the Office of Nuclear Reactor Regulation and Region I, as well as NRC-contracted specialists in steam generator eddy current testing. On July 20, 2000, the team leader discussed the preliminary results with you, Messrs. J. Groth and J. Baumstark, and other members of the Con Edison staff.

The team concluded that the overall technical direction and execution of the 1997 steam generator inspection program were deficient in several respects. Con Edison did not recognize and take appropriate corrective actions for significant conditions adverse to quality that affected eddy current data collection/analysis. This increased the likelihood that detectable flaws in low row U-bend tubes were not identified.

More specifically, Con Edison did not:

1. take appropriate corrective actions following identification of a new and significant tube degradation mechanism, i.e., inside diameter (ID) primary water stress corrosion cracking (PWSCC) at the apex of a low row U-bend tube. Operating experience indicates that apex cracking is more likely to result in tube failure than other U-bend cracks. The 1997 steam generator inspection program did not fully assess the implications of this new degradation mechanism and adjust, as appropriate, the inspection methods and analyses.

2. recognize the significance of, and fully evaluate, the flaw masking effects of the high noise encountered in the eddy current signal. In the case of the steam generator tube that failed, the magnitude of the noise was a problem that negatively impacted the probability of detection. The data analysis techniques were not adjusted to compensate for the noise to improve the identification of a flaw signal and ensure the appropriate probability of detection, particularly when conditions which increased susceptibility to tube degradation existed.

3. appropriately establish procedures and implement practices to address the potential for hour-glassing in the upper support plate flow slots. Hour-glassing in this location is indicative of increased stresses on the steam generator tubes, which increase the likelihood of tube cracks. Further, the potential existence and impact of upper support plate hour-glassing were not assessed following the identification in 1997 of eddy current probe restrictions at the upper support plate and the identification of a PWSCC indication at the apex of a steam generator tube.

4. ensure the use of properly qualified eddy current techniques. The U-bend plus-point eddy current probe was not set-up properly for use. Specifically, you did not use the proper calibration standard and phase rotation specified by the EPRI technique qualification standard. While this issue had a small effect on the probability of detection of low row U-bend indications, it was another example that reflected the deficiencies in the overall technical direction and execution of the 1997 steam generator program.

The team also concluded that Con Edison's root cause analysis for the tube failure, dated April 14, 2000, did not sufficiently address the above described deficiencies. While the root cause analysis attributed the tube failure to a flaw that was obscured by eddy current signal noise, it did not identify, nor address, deficiencies in the processes and practices that were implemented for the 1997 steam generator inspection.

At the exit meeting, Con Edison disagreed with the team's preliminary findings. Specifically, it is our understanding that Con Edison's position is that: 1) all 1997 steam generator inspection requirements were met; 2) the team had not identified any specific requirements, standards or guidelines that were not met; 3) no specific noise criteria existed relative to the probability of detection of flaws using eddy current examination; 4) the PWSCC indication was expected and no additional assessment was warranted after this discovery; 5) the root cause submitted was complete and accurate; and, 6) the NRC team's preliminary findings are not in agreement with NRC Inspection Report 50-247/97007, dated July 16, 1997. Many of these viewpoints had been discussed during the inspection. The NRC will continue to consider these points as part of our established regulatory process, which includes the significance determination process and inspection report finalization.

In accordance with 10 CFR 2.790 of the NRC's "Rules of Practice," a copy of this letter will be available electronically for public inspection in the NRC Public Document Room or from the Publicly Available Records (PARS) component of NRC's document system (ADAMS). ADAMS is accessible from the NRC Web site at http://www.nrc.gov/NRC/ADAMS/index.html (the Public Electronic Reading Room). Should you have any questions regarding this letter, please contact Mr. David C. Lew at 610-337-5120.

Sincerely,

/RA/

Wayne D. Lanning, Director
Division of Reactor Safety

Docket No. 05000247

License No. DPR-26

cc w/encl:

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