From: Jack Strosnider NRR
To: Bill Bateman NRR
Date: Wed, Jul 26, 2000 7:45 AM
Subject: QUICK LOOK LETTER

Bill, et. al.,

I have a few comments/suggestions on the IP-2 quick look letter, as indicated in the attached file. Do you or your staff have any comments? If so, please provide them to Dave L. Otherwise, I concur, with the comments noted.

Thx,

Jack

CC: Brian Holian, David Lew, Emmett Murphy, Stephanie Coffin

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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 for 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 may 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 this event 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 was deficient in several respects. As a result, Con Edison did not recognize and take appropriate corrective actions for significant conditions adverse to quality that affected eddy current data collection/analysis and tube integrity analyses, thereby increasing ~~that increased~~ the likelihood of tube degradation. This weakness in program quality 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. Apex cracking is more likely to tube failure ~~burst~~ 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 estimated to equate to a 70% - 100%**

through-wall defect. (What's the basis for this statement? How confident are we? This type of statement can lead to debate - is it necessary or do we only need to make the point that noise was a problem - as indicated by their own assessment, and they didn't deal with it) 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 was not assessed following the identification in 1997 of eddy current probe restrictions at the upper support plate.

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 June 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, Con Edison stated 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 that 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 was not in agreement with NRC Inspection Report 50-247/97007, dated July 1997. Many of these issues had been discussed during the inspection. The NRC will consider these points as part of the inspection report finalization process.

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,

Wayne D. Lanning, Director
Division of Reactor Safety

Docket No. 05000247
License No. DPR-26

cc w/encl:
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J. Baumstark, Vice President, Nuclear Power Engineering
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