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April 14, 1986

Director, Office of Nuclear Reactor Regulation
Attention: G. E. Lear, Director
PWR Project Directorate No. 1
U. S. Nuclear Regulatory Commission
Washington, D. C. 20555

Gentlemen:

Subject: Docket No. 50-206
Steam Generator Inspection Report
San Onofre Nuclear Generating Station
Unit 1

Reference: February 11, 1986 letter from H. E. Morgan (SCE) to J. B. Martin (NRC Region V) regarding the subject topic

In accordance with the requirements of Technical Specification 4.16, an Inservice Inspection of steam generator tubes was performed during December, 1985 as part of the San Onofre Unit 1 Cycle 9 refueling outage activities. The enclosed report entitled "1985 Steam Generator Inspection Results, San Onofre Unit 1" provides the information necessary to facilitate your review of the inspection results and approve the corrective actions. Also provided herein is a discussion of the existing Technical Specification reporting requirements and our plans to revise the Technical Specifications.

As part of the Inservice Inspection of steam generator tubes required by Technical Specification 4.16, a total of 2336 tubes (22 percent of the tubes in service) were inspected. The total number of tubes inspected, as indicated herein, is different than the total number of tubes that were reported to have been inspected in the referenced letter. Upon reevaluation of the inspection data, it was determined that the total number of tubes inspected, as reported in the referenced letter, reflected the number of times an eddy current probe was inserted in the tubes and not the actual number of tubes inspected. Accordingly, this number has been revised to reflect the actual number of tubes inspected. The results indicate limited degradation is occurring, but only in the cold leg near the top of the tubesheet. In general, the inspection demonstrated there has been no detectable progression of intergranular attack (IGA), denting, or antivibration bar (AVB) wear. Further there has been no sleeve degradation. Of the tubes inspected, 63 were removed from service by mechanical plugging. The breakdown of imperfection location is as follows; 39 tubes with imperfections near the top of the cold leg tubesheet, 1 tube with an imperfection near the top of the hot leg tubesheet, and 23 tubes with imperfections at the antivibration bars. More

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than half of these, however, showed no real growth since previous inspections and required plugging only as a result of improved inspection and analysis techniques.

During the 1982 steam generator inspection, a secondary side visual inspection discovered all but three wrapper support bars to be broken or missing. For this reason, a supplemental visual inspection was performed as part of the 1985 steam generator inspection to assess the intact wrapper support bars in steam generators "A" and "B". The results of the inspection showed the support bars are still intact and have not moved. To ensure continued integrity, these support bars will be reinspected during the next refueling outage as part of our secondary side inspection program.

Pursuant to Technical Specification 4.16.E.4, the referenced letter provided to the NRC Regional Administrator a summary of the results of the 1985 Steam Generator Inspection. Specification 4.16.E.4 establishes provisions to report to the Commission, in accordance with Technical Specification 6.6, in the event that more than 3 of the tubes inspected exceed the plugging limit. As stated in the referenced letter, Specification 6.6, Reportable Event Action, requires reporting in accordance with 10 CFR 50.73. Based on this determination, the referenced letter was provided to the NRC Regional Administrator. This rule, however, does not require reporting increased degradation or plugging of steam generator tubes. It has been determined that Specification 4.16.E.4 should require reporting in accordance with Technical Specification 6.9.2, Special Reports. It is our plan to submit an amendment application to revise Specification 4.16.E.4 to require reporting in accordance with Specification 6.9.2. It is expected that this revision will be submitted well in advance of the next steam generator inspection in order to avoid unnecessary confusion should the reporting provisions of Specification 4.16.E.4 become necessary.

In conclusion, the remedial actions taken are considered to be appropriate to resolve the steam generator tube degradation identified during the 1985 Steam Generator Inspection. Further, it is noted that the steam generator tube degradation will not impact plant operation or the health and safety of the public or plant personnel.

If you have any questions or require additional information regarding the above or enclosure, please let me know.

Very truly yours,

M.O. Medford

Enclosure

cc: Mr. J. B. Martin (Regional Administrator USNRC Region V)
Mr. F. R. Huey (USNRCResident Inspector, Units 1, 2 and 3)
Document Control Desk, NRC
Institute of Nuclear Power (INPO)