

December 23, 1999

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

Subject: **Docket No. 50-361**
Special Report: Inservice Inspection of Steam Generator Tubes
San Onofre Nuclear Generating Station, Unit 2

- Reference: 1 Southern California Edison (SCE) letter from A. E. Scherer to US NRC, "Special Report Inservice Inspection of Steam Generator Tubes," dated February 9, 1999.
- 2 Steam Generator Program Guidelines, Nuclear Energy Institute Document Number NEI 97-06 [Original], dated December 1997

Gentlemen:

In reference 1, Southern California Edison (SCE) submitted a special report satisfying the reporting requirements of Technical Specification 5.7.2.c. The enclosed report provides the supplemental information conforming to the content guidance of reference 2 and is provided for your information.

In addition to the enclosed report, because of the critical nature of these assessments to support the transition from mid-cycle examinations to full cycle operation, SCE elected to have two additional and independent analyses performed (discussed below). All three analyses support the same conclusion that the inspection interval can again be extended to support a full cycle of operation.

The enclosure is a fully probabilistic evaluation of the steam generator degradation consistent with the methodology used for our last two operational assessments and the probabilistic performance criteria of reference 2. The conclusion of this report is that the required structural and leak rate margins will be maintained for the 1.67 EFPY maximum planned Cycle 10 operating period.

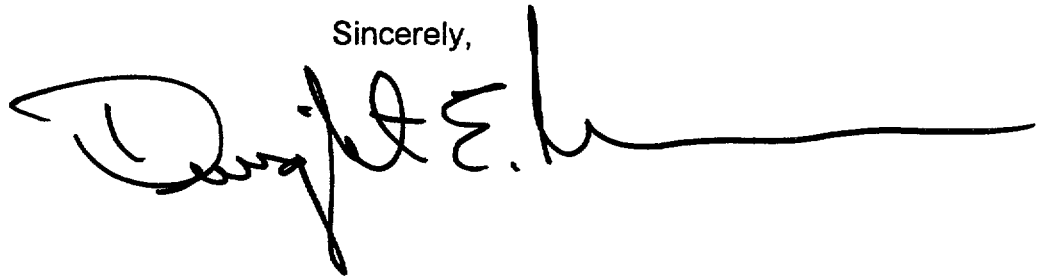
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For one supplemental evaluation, Edison commissioned a deterministic evaluation by Asea Brown Boveri Combustion Engineering (ABB/CE) of tube structural integrity and leakage consistent with the deterministic performance criteria of reference 2. The conclusion of that report is that the deterministic structural integrity and leakage performance criteria of reference 2 are satisfied until the end of the next planned operating cycle and is available on site.

Finally, the second supplemental evaluation SCE commissioned was a fully probabilistic analysis by E-Mech Technologies, Inc. consistent with the probabilistic performance criteria of reference 2. The analysis only addresses axial Primary Water Stress Corrosion Cracking (PWSCC) at tube support locations, which was the limiting consideration for partial cycle run times. That report concludes that axial PWSCC degradation at support locations is not a limiting consideration for the planned operating period and is available on site.

The enclosed report contains no new commitments. If you require any additional information, please feel free to contact us.

Sincerely,

A handwritten signature in black ink, appearing to read "Douglas E. L.", with a long horizontal flourish extending to the right.

Enclosure, Steam Generator Tube Integrity Operational Assessment

cc: E. W. Merschoff, Regional Administrator, NRC Region IV
L. Raghavan, NRC Project Manager, San Onofre Units 2 & 3
J. A. Sloan, NRC Senior Resident, San Onofre Units 2 & 3
Institute of Nuclear Power Operations (INPO)

Enclosure

**Steam Generator Tube Integrity
Operational Assessment**