UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

December 26, 2012

Mr. Peter T. Dietrich
Senior Vice President and Chief Nuclear Officer
Southern California Edison Company
San Onofre Nuclear Generating Station
P.O. Box 128
San Clemente, CA 92674-0128

SUBJECT: SAN ONOFRE NUCLEAR GENERATING STATION, UNIT 2 – REQUEST FOR ADDITIONAL INFORMATION REGARDING RESPONSE TO CONFIRMATORY ACTION LETTER (TAC NO. ME9727)

Dear Mr. Dietrich:

On March 27, 2012 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML12087A323), the U.S. Nuclear Regulatory Commission (NRC) issued a Confirmatory Action Letter (CAL) to Southern California Edison (SCE) regarding the San Onofre Nuclear Generating Station (SONGS), Units 2 and 3. The CAL confirms certain actions that SCE will take to address steam generator tube degradation issues at both units. The CAL also confirms that SCE will not resume power operation at either unit until the NRC completes its review of those actions and formally communicates its permission to restart in written correspondence.

By letter dated October 3, 2012 (ADAMS Accession No. ML12285A263), SCE submitted its response to the CAL for SONGS Unit 2. The NRC staff is conducting its detailed review of SCE's CAL response for SONGS Unit 2 and has determined that additional information is needed in order to complete our evaluation. The staff's questions are provided in the enclosed request for additional information (RAI). The staff previously issued these RAI questions in draft form, on November 30, 2012 (ADAMS Accession No. ML12338A110), on December 10, 2012 (ADAMS Accession No. ML12345A427), and on December 20, 2012 (ADAMS Accession No. ML12356A198). Based upon clarifying discussions on RAI questions 1-31 between NRC and SCE at a public meeting on December 18, 2012, the enclosed final version of these questions is unchanged from the previous draft versions. In that meeting, SCE stated that it expects to provide responses to RAI questions 1-31 by mid-January of 2013. Please provide an estimated date for your response to RAI question 32. The NRC staff expects to issue additional RAIs to SCE as our review continues.

If you have any further questions regarding this letter, please contact me at (301) 415-4032 or via e-mail at randy.hall@nrc.gov.

Sincerely,

BRIAN BENNEY FOR

James R. Hall, Senior Project Manager
San Onofre Special Projects Branch
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket No. 50-361

Enclosures:
Request for Additional Information

cc w/Encl Distribution via Listserv

OFFICE OF NUCLEAR REACTOR REGULATION
REQUEST FOR ADDITIONAL INFORMATION
SOUTHERN CALIFORNIA EDISON
SAN ONOFRE NUCLEAR GENERATING STATION, UNIT 2
RESPONSE TO MARCH 27, 2012, NRC CONFIRMATORY ACTION LETTER
DOCKET NO. 50-361
TAC NO. ME9727

By letter dated October 3, 2012, (Agencywide Documents Access and Management System (ADAMS) Accession No. ML12285A263), Southern California Edison (SCE) submitted its response to the NRC Confirmatory Action Letter (CAL) dated March 27, 2012, for San Onofre Nuclear Generating Station (SONGS), Unit 2. The details of SCE's response to the CAL are provided in Enclosure 2 to that October 3, 2012, letter (Reference 1). The NRC staff is conducting its detailed review of SCE's CAL response for SONGS Unit 2 and has determined that additional information is needed in order to complete our evaluation. The staff's additional questions are stated in this request for additional information (RAI) below. The staff previously issued these RAI questions in draft form, on November 30, 2012 (ADAMS Accession No. ML12338A110), on December 10, 2012 (ADAMS Accession No. ML12345A427), and on December 20, 2012 (ADAMS Accession No. ML12356A198). Based upon clarifying discussions between NRC and SCE at a public meeting on December 18, 2012, the final version of these questions is unchanged from the previous draft versions.

1. The Operational Assessment (OA) in Attachment 6, Appendix A (Reference 2), reports the 3 times normal operating pressure differential as being 4290 psi for 100% power conditions. This is the same value assumed in the Condition Monitoring Assessment provided in Attachment 2. This value is significantly higher than the values ranging from 3972-3975 psi for 100% power reported in Attachment 6, Appendices B, C, and D (References 3-5). Describe the reason for the differences.

2. The Operational Assessment in Attachment 6, Appendix C (Reference 4), pages 3-2 and 4-12, appears to state that tube-to-tube wear (TTW) growth rates are based on the maximum TTW depths observed in Unit 3 at EOC 16 divided by the first Unit 3 operating period (0.926 years at power). Provide justification for the conservatism of this assumption. This justification should address the following:

    a. Reference 4, page 3-2 defines "wear index" for a degraded tube and states that the existence of TTW and distribution of TTW depths are strongly correlated to the wear index. This is pictured in Figures 4-4 in terms of TTW initiation. This figure shows that TTW is not expected to have initiated until a threshold value of wear index is reached. This threshold value varies from tube to tube according to a cumulative probability distribution shown in the figure. This figure illustrates that TTW is not expected to have initiated until sometime after BOC 16. This suggests that the observed TTW depth at EOC 16 developed over a smaller time interval than the 0.926 years assumed in the analysis.

b. An independent analysis in Reference 3 also indicates an extremely low probability of instability onset at BOC 16 as illustrated in Figure 8-3. Reference 3, page 106 interprets this figure as indicating that the probability of instability only reaches 0.22 after 3 months and only becoming "high" after 4 months.

c. Reference 3 also considered a variety of different wear rate models to estimate how long it took to develop the observed TTW depths at Unit 3 after instability occurred. These analyses are documented in Appendix A of Reference 3 and produced estimates in the range of 2.5 to 11 months.

3. Regarding Reference 4, describe the sensitivity of the results in Figure 5-4 to the definition of "wear index." If alternate definitions significantly affect the results, what is the justification for the definition being used?

4. Regarding Reference 4, does the definition of "wear index" include summing the depths of 2-sided wear flaws at a given AVB intersection? If not, explain why SCE's approach is conservative.

5. Regarding Reference 4, third paragraph from the bottom of page 4-3, why is non-detected wear only assigned to no degradation detected (NDD) tubes and not to NDD tube/AVB intersections in tubes with detected wear at other intersections?

6. Regarding Reference 4, page 4-5, it seems that depths of undetected flaws are assumed to be associated with probability of detection (POD) $\leq$ 0.05. Why is this conservative? Is there a possibility that some undetected flaws may be associated with higher values of POD?

7. Regarding Reference 4, page 4-5, what is meant by the words, "each active wear location" in the 1350 NDD tubes? How are the "active wear" locations determined?

8. It is stated in Reference 4, page 4-6, second paragraph that, "It has been observed that the number of AVB supports that develop wear in the second cycle of operation can increase dependant on the number of worn AVB indications at the beginning of the second cycle. These data were used in the OA to add AVB locations at the start of Cycle 17 from a statistical representation of this data." Provide a more complete description of the model used to add AVB locations that will develop wear during the

second cycle. Confirm that this model applies to both the 560 tubes with existing tube support wear and the 1350 NDD tubes.

9. It is stated in Reference 4, at the top of page 4-9 that the simulation results of the bench marking process are shown in Figure 4-6. Provide additional detail on what Figure 4-6 is showing and how it relates to the benchmarking process. As part of this additional detail, explain the meaning of the ordinate label "number of observations" in the figure.

10. Technical Specification (TS) 3.4.13.d allows 150 gallons per day primary to secondary leakage. The Return to Service Report (Enclosure 2 of Reference 1), Section 9.4.1 states, "The plant operating procedure for responding to a reactor coolant leak has been modified to require plant Operators to commence a reactor shutdown upon a valid indication of a primary-to-secondary SG tube leak at a level less than allowed by the plant's TSs. This procedure change requires earlier initiation of operator actions in response to a potential SG tube leak." Does this mean that a reactor shutdown would be commenced upon <u>any</u> valid indication of primary to secondary leakage? Provide a description of the action levels in the procedure. Discuss any additional actions, planned or taken, such as simulator testing, operator training, and/or any evaluations to assess potential impacts of the revised procedure.

11. Please submit an operational impact assessment for operation at 70% power. The assessment should focus on the cycle safety analysis and establish whether operation at 70% power is within the scope of SCE's safety analysis methodology, and that analyses and evaluations have been performed to conclude operation at 70% power for an extended period of time is safe. The evaluation should also demonstrate that the existing Technical Specifications, including limiting conditions for operation and surveillance requirements, are applicable for extended operation at 70% power.

12. Operation at a lower power level could introduce additional uncertainty in measuring reactor coolant flow. Please provide a detailed evaluation of RCS flow uncertainty, identify how RCS flow uncertainty is affected by operation at 70% power, and discuss the overall treatment of the RCS flow uncertainty, actual and indicated, in the context of the remaining safety analyses. Provide similar information for secondary flow uncertainty, as well.

13. The installation of new steam generators involved changes to the steam generator heat transfer characteristics, which could affect the performance of the plant under postulated loss of coolant accident conditions. Please explain how the existing ECCS analysis accounts for these changes, and how considerable steam generator tube plugging has been addressed in the ECCS evaluation. Provide the ECCS evaluation that will apply to the planned operating cycle.

14. Provide a summary disposition of the U2C17 calculations relative to the planned reduced-power operation.

15. In Reference 1, Section 8.3.2, page 48 – How will the continued integrity of the non-stabilized, preventively-plugged tubes adjacent to the retainer bars be ensured? "Integrity" in this context refers to the tubes remaining intact and unable to cause damage to adjacent tubes.

16. Reference 1, Section 9.3, page 50 – Provide additional information concerning the "Operational Decision Making" process and describe how it would be applied if the proposed criterion is exceeded. Provide the procedural action statement.

17. Reference 1, Section 9.4.1, page 50 – Provide the procedural action levels/statements.

18. Reference 1, Section 11.1, page 52 – SCE proposes to upgrade the vibration and loose parts monitoring system (VLPMS) as a defense-in-depth measure to enhance plant monitoring capability to facilitate early detection of a steam generator tube leak and ensure immediate and appropriate plant operator and management response.

    Fluid Elastic Instability (FEI) was identified as a main cause of the tube wear for both the Unit 2 and 3 steam generators. The FEI experienced is due to a combination of the conditions of steam quality, secondary side fluid velocity in the vicinity of the tube bundle, and steam void fraction, and the degree of such fluid elastic instability is related to the damping provided by internal support structures. According to your report, "steam quality directly affects the fluid density outside the tube, affecting the level of hydrodynamic pressure that provides the motive force for tube vibration. When the energy imparted to the tube from hydrodynamic pressure (density times velocity squared, or pv^2) is greater than the energy dissipated through damping, FEI will occur." However, the proposed plant VLPMS enhancement does not appear to directly monitor steam quality, secondary side fluid velocity, or steam void fraction.

    Please provide the following information to address the effectiveness of the enhanced VLPMS:

    a. Describe the specific purpose of using the enhanced VLPMS equipment for monitoring steam generator performance. For example, is it to be used for monitoring acoustic noise indicative of flow velocity, steam quality, and void fraction, or for the measurement of metallic noise indicative of vibration of tubes against each other or against tube support structures? Exactly how will this be done? What is the theory of operation? If it will be used to monitor an increase in pv^2 leading to the onset of FEI, provide a description of the correlation of the velocity of steam voids through the secondary side of the steam generator and the relative changes in characteristics of the signal output from the various VLPMS accelerometers. If it is to be used for detecting actual tube vibration, provide a description of the process that will be used for discerning actual tube vibration noise from background noise, and the required threshold identification criteria that will be applied to reach the conclusion that tube vibration is occurring.

b. Identify the ranges of amplitudes and frequencies of the acoustic noise signals from each accelerometer that are indicative of an approach to the conditions leading to FEI or actual tube vibration, and the reasons for selection of the more sensitive accelerometers. Also, discuss the required response time of the signal processing equipment needed to detect and continuously monitor either fluid velocities within the steam generator or tube impact noise, depending on the intended use of the enhanced VLPMS, and the actual response time capabilities of the equipment, from sensor through processed signal output, that is being proposed for use.

c. Discuss the acceptance criteria (e.g., magnitude of signal, plant power level, etc.) that will be used to establish the setpoints for the alarms described in Section 11 of your report: "The signals from these sensors are compared with preset alarm setpoints." Provide a description of how the alarm setpoints were established, and at what point during the start-up of Unit 2 will these alarm setpoints be calibrated into the VLPMS. If the setpoints have not yet been determined, provide a description of your plan for determining and implementing these settings.

d. Describe the planned operator actions and any changes to the procedures for responding to alarms or signals potentially indicative of tube-to-tube contact, including time limits for analyzing the signals and taking any necessary action including plant shutdown. Describe the lessons learned that have been drawn from the signals of potential metal-to-metal contact experienced in Unit 3 and how these lessons have been factored into current procedures.

e. A description of how you determined that acoustic noise monitoring and predictive signal processing was the best method for monitoring either the onset of FEI or actual tube vibration, including a list of other methods (e.g., time domain reflectivity probes calibrated for steam void propagation monitoring) that had been considered for enhancing steam generator tube monitoring during start-up of Unit 2, and the reasons for their rejection.

19. Reference 1, Section 11.2, page 52 – Provide additional details on how the GE Smart Signal System will be used in the context of tube-to-tube wear and/or the circumstances associated with tube-to-tube wear. What information/data will the system be evaluating? For what purpose?

20. Reference 3, page 17 of 129, refers to tube-to-support design clearance of 2 mils diametral. Confirm that this is the nominal diametral clearance under ambient conditions, or clarify the statement otherwise.

21. Reference 3, page 44 of 129, states that the plugged tubes have an effect on local thermal/hydraulic conditions upon returning to power and have been included in the

stability ratio calculations. The staff interprets this to mean the effect of the plugged tubes on the calculated thermal/hydraulic conditions were considered in the stability ratio calculations and that the stability ratio calculations included the plugged (and stabilized) tubes. Is this correct? Clarify, if not.

22. Reference 3, page 57 of 129, first full paragraph beginning with the words "Figure 6-1" – The third sentence states, "... it is not practical to use an individual run of the quarter model as a single Monte Carlo trial for contact forces." However, the staff was unable to ascertain from the subsequent discussion exactly what was done as an alternative? Nor was the staff able to discern this from Reference 6, Appendix 9. Provide or cite by reference a more complete description of how the cumulative distributions of contact forces were determined. For example, what is a "run?" What does it mean to "combine runs?" How were zones employed in order to provide a more practical approach? Are all tubes in a given zone assumed to have the same initial clearances, final clearances, and contact forces? Do all AVB #5 in a zone have the same cumulative distribution of contact forces? Is a Monte Carlo performed for each zone?

23. Reference 3 – Provide figures similar to Figures 6-19 and 6-20 for Unit 3, SG E-088, and Unit 2, SG E-088.

24. Reference 3, page 59 of 129, last paragraph – The sentence, "AVBs 2, 3, 11 and 10 near row 27 have sporadic dents in the vicinity of the noses of AVBs 1, 4, 9 and 12" does not appear to make sense. Provide further clarification relative to the discussion of Figure 6-20.

25. Reference 3, page 59 of 129 – There is a statement in the last paragraph that reads, "Patterns of dents and associated high contact forces are in good agreement with the final quarter model calculations." Provide or show this comparison.

26. Reference 3, page 107 of 129, second to last paragraph – Provide additional details of the wear growth model at the tube supports. Were cumulative probability functions of observed wear rates constructed and randomly sampled when developing the contact force probability distributions at each intersection? Was total gap at each intersection (prior to applying temperature and allowing the model to settle, leading to the development of contact forces) assumed to be the sum of the manufacturing gap and the maximum wear depth?

27. Reference 6, Appendix 8, "SG Tube Flowering Analysis", page 8-2 (307 of 474) – MHI concludes, in part, that the tube-to-AVB gaps in the center columns increase due to hydrodynamic pressure by [ ] when the manufacturing tolerance dispersion is not taken into account. MHI also concludes that the gap increase due to hydrodynamic pressure is small when the manufacturing tolerance dispersion is taken into account. Discuss whether this latter finding may simply reflect the hydrodynamic pressures acting to

relieve the tube-to-AVB contact forces caused by the manufacturing tolerance dispersion, such that the gaps are relatively unchanged relative to the case were the hydrodynamic pressure is not considered. Reference 6, Appendix 9, "Simulation of Manufacturing Dispersion for Unit-2/3," does not seem to make specific mention of whether the calculated tube-to-AVB contact forces directly considered the effect of the hydrodynamic effect on tube-to-tube contact forces, but the staff understands that they did not. If the staff's understanding is correct, explain how the resulting contact forces are conservative.

28. Reference 5, Section 2.6.1 – What is the estimated growth rate of the tube-to-tube wear in steam generator 3E0-88, tube R106C78? Describe how it was determined.

29. Reference 5, Figures 2-12 and 2-13 – Provide similar figures for Case 78 (all AVBs missing).

30. Reference 1, Figure 8-2 – Provide similar figure for maximum interstitial velocities.

31. In References 7 and 8 (specifically, in Section 7.2 of Reference 7 and in Section 8.0 of Reference 8), AREVA used Revision 3 of the Electric Power Research Institute "Steam Generator Management Program: Steam Generator Integrity Assessment Guidelines," in part, to assess the most limiting structural integrity performance criteria (e.g., the more limiting structural limit determined from (a) the three times the normal operating differential pressure criterion or (b) the safety factor of 1.2 on combined primary loads and 1.0 on axial secondary load criterion). In some cases, it appears that the limits in the Integrity Assessment Guidelines may have been based on specific tests and plant data. Please discuss whether you have confirmed the applicability of the limits in the Integrity Assessment Guidelines (in particular, those related to when non-pressure loads need to be considered) to the SONGS replacement steam generators.

32. SONGS Unit 2 Technical Specification (TS) 3.4.17 requires that steam generator structural integrity be maintained in Modes 1, 2, 3, and 4 (Power Operation, Startup, Hot Standby, and Hot Shutdown, respectively). Limiting Condition for Operation (LCO) 3.4.17, "Steam Generator (SG) Tube Integrity," requires that steam generator tube integrity shall be maintained and all steam generator tubes satisfying the tube repair criteria shall be plugged in accordance with the Steam Generator Program in MODES 1, 2, 3, and 4. The steam generator tube rupture (SGTR) accident is the limiting design basis event for SG tubes and avoiding an SGTR is the basis for LCO 3.4.17. Surveillance Requirement (SR) 3.4.17.1 requires "Verify SG tube integrity in accordance with the Steam Generator Program."

    The structural integrity performance criterion is described in SONGS Unit 2 TS 5.5.2.11.b.1 as follows:

    > All in-service steam generator tubes shall retain structural integrity over the <u>full range of normal operating conditions</u> (including startup, operation in the power range, hot standby, cool down and all anticipated transients

included in the design specification) and design basis accidents. This includes retaining a safety factor of 3.0 against burst under normal steady state full power operation primary-to-secondary pressure differential and a safety factor of 1.4 against burst applied to the design basis accident primary-to-secondary pressure differentials. Apart from the above requirements, additional loading conditions associated with the design basis accidents, or combination of accidents in accordance with the design and licensing basis, shall also be evaluated to determine if the associated loads contribute significantly to burst or collapse. In the assessment of tube integrity, those loads that do significantly affect burst or collapse shall be determined and assessed in combination with the loads due to pressure with a safety factor of 1.2 on the combined primary loads and 1.0 on axial secondary loads. [emphasis added]

As described in the SONGS Unit 2 license, SCE "is authorized to operate the facility at reactor core power levels not in excess of full power (3438 megawatts thermal)," which is also defined as Rated Thermal Power (RTP).

In SCE's operational assessment (OA) that evaluated tube degradation caused by mechanisms other than tube-to-tube wear (Reference 2), on Page 30 of 32, SCE concluded that "there is reasonable assurance that the performance criteria for the non-[tube-to-tube wear] TTW degradation will be met if Unit 2 were to operate for a full fuel cycle of 1.577 EFPY [effective full power years] at 100% reactor power." Thus it appears that in Reference 2, SCE considered the requirements of TS 5.5.2.11.b.1 by addressing the licensed full power condition.

In contrast, SCE performed three other operational assessments that evaluated tube degradation due to tube-to-tube wear (References 3-5), but it appears that in these OAs, SCE addressed structural integrity requirements for TTW only at 70% reactor power, instead of at 100% reactor power. For example, in Reference 3, Section 10.0, "Conclusions," page 117 of 129, SCE states: "A 70% operating power level returns the Unit 2 steam generators to within the operational envelope of demonstrated successful operation...Operation at 70% power assures in-plane stability (SR<1) without dependence on any effective in-plane supports for U-bends."

Therefore, it appears that SCE has not provided an operational assessment that addresses compliance with TS 5.5.2.11.b. for tube-to-tube wear, without reliance on compensatory measures (e.g., limiting reactor power to 70% RTP).

Please clarify how the information submitted by SCE demonstrates that the structural integrity performance criterion in TS 5.5.2.11.b.1 is met for operation within current licensed limits up to the licensed RTP, or provide an operational assessment that includes an evaluation of steam generator TTW for operation up to the RTP.

REFERENCES

1. Letter from Peter T. Dietrich, SCE, to Elmo E. Collins, USNRC, "Docket No. 50-361, Confirmatory Action Letter – Actions to Address Steam Generator Tube Degradation,

San Onofre Nuclear Generating Station, Unit 2," October 3, 2012; **Enclosure 2**, "San Onofre Nuclear Generating Station Unit 2 Return to Service Report, Revision 0." (ADAMS Accession No. ML12285A263; ADAMS Package No. ML122850320)

2. Attachment 6 to Reference 1, "SONGS U2C17 Steam Generator Operational Assessment," **Appendix A**, Revision 2, "SONGS U2C17 Outage - Steam Generator Operational Assessment," prepared by Areva NP Inc. Document No. 51-9182833-002 (NP), Revision 2), October 2012. (ADAMS Accession No. ML12285A267)

3. Attachment 6 to Reference 1, "SONGS U2C17 Steam Generator Operational Assessment," **Appendix B**, Revision 0, "SONGS U2C17 Steam Generator Operational Assessment for Tube-to-Tube Wear," prepared by Areva NP Inc. Document No. 51-9187230-000 (NP), Revision 0), October 2012. (ADAMS Accession Nos. ML12285A267, ML12285A268, and ML12285A269)

4. Attachment 6 to Reference 1, "SONGS U2C17 Steam Generator Operational Assessment," **Appendix C**, "Operational Assessment for SONGS Unit 2 SG for Upper Bundle Tube-to-Tube Wear Degradation at End of Cycle 16," prepared by Intertek APTECH for Areva, Report No. AES 12068150-2Q-1, Revision 0, September 2012. (ADAMS Accession No. ML12285A269)

5. Attachment 6 to Reference 1, "SONGS U2C17 Steam Generator Operational Assessment," **Appendix D**, "Operational Assessment of Wear Indications In the U-Bend Region of San Onofre Unit 2 Replacement Steam Generators," prepared by Westinghouse Electric Company LLC, Report No. SG-SGMP-12-10, Revision 3, October 2012. (ADAMS Accession No. ML12285A269)

6. Attachment 4 to Reference 1, "MHI Document L5-04GA564, Tube Wear of Unit-3 RSG - Technical Evaluation Report," Revision 9, October 2012, prepared by Mitsubishi Heavy Industries, LTD. (ADAMS Accession Nos. ML12285A265, ML12285A266, and ML12285A267)

7. Attachment 2 to Reference 1, AREVA NP Inc., Engineering Information Record, Document No. 51-9182368 – 003 (NP), "SONGS 2C17 Steam Generator Condition Monitoring Report." (ADAMS Accession No. ML12285A263)

8. Attachment 3 to Reference 1, AREVA NP Inc., Engineering Information Record, Document No. 51-9180143 – 001 (NP), "SONGS Unit 3 February 2012 Leaker Outage - Steam Generator Condition Monitoring Report." (ADAMS Accession No. ML12285A264)

If you have any further questions regarding this letter, please contact me at (301) 415-4032 or via e-mail at randy.hall@nrc.gov.

Sincerely,

/RA by BBenney for/

James R. Hall, Senior Project Manager
San Onofre Special Projects Branch
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket No. 50-361

Enclosures:
Request for Additional Information

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