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

October 14, 2011

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, UNITS 2 AND 3 – REQUEST FOR ADDITIONAL INFORMATION REGARDING USE OF AMERICAN CONCRETE INSTITUTE (ACI) REPORTS FOR RESTORATION OF UNIT 3 CONTAINMENT (TAC NOS. ME6179 AND ME6180)

Dear Mr. Dietrich:

On March 30, 2011, the U.S. Nuclear Regulatory Commission (NRC) completed a steam generator replacement inspection at San Onofre Nuclear Generating Station (SONGS), Unit 3. That inspection examined activities conducted under your license as they relate to safety and compliance with the Nuclear Regulatory Commission's (NRC's) rules and regulations and with the conditions of your license. The results of the inspection were documented in NRC inspection report 05000362/2010009, dated May 10, 2011 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML111300448).

During the inspection, the NRC inspectors identified an unresolved item regarding the engineering modeling inputs related to restoration of the Unit 2 and Unit 3 containment buildings (URI 05000362/2010009-01, "Adequacy of Model Inputs Used in Restoration of Nuclear Concrete Containment Structures"). As described in the inspection report, the inspectors reviewed three related engineering calculations and the screening required by 10 CFR 50.59 associated with the SONGS, Unit 3 containment restoration. The calculations reviewed were performed in conjunction with the inspection referenced models and equations from two contemporary reports; ACI 209R-92, "Prediction of Creep, Shrinkage, and Temperature Effects in Concrete Structures," and ACI 224.2R-92, "Cracking of Concrete Members in Direct Tension." The NRC determined that the licensee's Section 50.59, "Changes, tests, and experiments," of Title 10 of the *Code of Federal Regulations* (10 CFR 50.59) evaluation did not address the referenced models and equation inputs from these reports, and that these reports were not referenced in the SONGS final safety analysis report, nor in the licensee's concrete construction code of record. The licensee considered the use of the inputs from these newer industry standards to be an analytical refinement and not a methodology change. However, based on the available information, the inspectors could not determine if the licensee properly applied the 10 CFR 50.59 process in concluding that the new analysis did not involve a departure from approved methods of evaluation.

To enable the NRC staff to fully assess the regulatory and safety implications of this issue, we are transmitting the enclosed request for additional information (RAI). We request your response to the RAI within 30 days of receipt of this letter, so that the staff may resolve the unresolved item in a timely manner.

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

Sincerely,

for

James R. Hall, Senior Project Manager
Plant Licensing Branch IV
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket Nos. 50-361 and 50-362

Enclosure:
As stated

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REQUEST FOR ADDITIONAL INFORMATION

SAN ONOFRE NUCLEAR GENERATING STATION, UNITS 2 & 3

SOUTHERN CALIFORNIA EDISON COMPANY

DOCKET NOS. 50-361 AND 50-362

On March 30, 2011, the U.S. Nuclear Regulatory Commission (NRC) completed a steam generator replacement inspection at San Onofre Nuclear Generating Station (SONGS), Unit 3. The results of the inspection were documented in NRC inspection report 05000362/2010009, dated May 10, 2011 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML111300448).

During the inspection, the NRC inspectors identified an unresolved item regarding the licensee's engineering modeling inputs related to restoration of the Unit 2 and Unit 3 containment buildings (URI 05000362/2010009-01, "Adequacy of Model Inputs Used in Restoration of Nuclear Concrete Containment Structures"). The following information is requested in order to allow the NRC staff to complete its evaluation of the unresolved item.

RAI 1

In Section 5.1 of Calculation C-257-01.04.05 (Reference 1), the licensee stated:

> The new concrete mix for restoration of the containment opening will be tested to determine the compressive strength, the modulus of elasticity and the creep characteristics. However, the results will not be available at the time for the EOL [end of life] finite element analysis. As such, the methods described in ACI 209R-92 and ACI 318-05 are used to estimate the relevant concrete properties. The moduli of existing and new concrete as well as creep and shrinkage will be used in the containment analysis to investigate the stress distribution around the opening after restoration.

The licensee's position on the use of ACI 209R-92 is further summarized as follows:

> The ACI 209R Report is a widely recognized guidance document that provides a simple, yet reasonably accurate methodology for estimating creep and shrinkage design values. For the SONGS containment structure, the use of such estimated values has been further justified and validated by comparison to long term creep and shrinkage test results performed on the actual concrete mix used to restore the temporary construction opening.

Please provide the above stated comparison of the concrete properties (creep, shrinkage, elastic modulus) obtained from tests of the actual concrete mix used for the restoration of the steam generator replacement construction opening to those used in the analysis based on estimates using methods in ACI 209R-92, "Prediction of Creep, Shrinkage, and Temperature Effects in Concrete Structures," that would demonstrate that the properties used in the SONGS

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containment analysis are comparable or conservative relative to those obtained from the tests. In establishing values of creep and shrinkage, please indicate how any important differences in the environment between the test samples and the actual concrete in the structure, if any, were considered.

RAI 2

Please justify why it is acceptable to apply the methodology in Section 4-1 of the ACI 224.2R-92 report, concerning the axial stiffness of one-dimensional members due to cracking in reinforced concrete caused by direct tension, to account for cracking in: (a) prestressed concrete, and (b) more complex systems such as post-tensioned containments, for the end-of-life evaluation of the restored SONGS containments in Calculation No. C-257-01.04.06 (Reference 2).

RAI 3

Appendix H of Calculation No. C-257-01.04.06 (Reference 2) describes the methodology and criteria used, based on Equations 4.12 and 4.13 of the ACI 224.2R-92 report, to estimate a reduced concrete sectional stiffness to account for cracking in the restored containment opening area, in the ANSYS shell-element-based linear elastic finite element model of the SONGS containments.

With regard to the application of Equations 4.12 and 4.13 of ACI 224.2R to calculate the effective cross-sectional area, A_e, of a cracked member in the above calculation, please provide the following information:

(a) For both the hoop and vertical directions, was the cross-sectional area of prestressing tendon steel included in the calculation of A_g, A_s and A_{cr}? If not, please provide a supporting justification.

(b) For both hoop and vertical directions, was the gross cross-sectional area, A_g, replaced with the transformed area, $A_t = A_g + (n-1)A_s$, to include the contribution of bonded reinforcing steel and unbonded prestressing steel in the post-tensioned containment? If not, please provide a supporting justification.

(c) How was the cracking load, P_{cr}, calculated for the hoop and vertical directions? Please identify what values of P_{cr} were used for the hoop and vertical directions. Please indicate the material property threshold (such as tensile strength) that was used to determine the cracking load.

(d) Please provide a numerical example of all steps (with all inputs used) of a typical calculation (e.g., for the ratio, $(E_cA_e)_{ANSYS}/E_c A_g = 0.4$ or 0.6) that was performed to develop a data point (one in hoop direction and one in vertical direction) in Figure H.2 of Reference 2.

RAI 4

From Section 8.1.2.2, "Cracked Conditions," and Appendix H of Calculation C-257-01.04.06 (Reference 2), it appears that the same value of reduced concrete section stiffness (A_eE_c) of $0.4E_cA_g$ was used in the model for each of the load combinations III, IV, and VI.

Please confirm if this is true. If so, please provide the justification for using the same value for all the load combinations, considering the fact that the axial strains and the extent of concrete cracking, and therefore the sectional stiffness, is a function of the magnitude of the forces due to the applied loads.

RAI 5

The methodology used in the parametric study in Appendix H of Calculation C-257-01.04.06 (page 84 of Reference 2) and the ANSYS containment analyses accounting for cracking is based on the assumption that the reduced effective axial stiffness (A_eE_c) for the hoop and vertical directions are equal. Please justify this assumption considering the fact that the degree of cracking is likely to not be the same in the two directions.

RAI 6

(a) The ANSYS parametric analyses in Appendix H of Reference 2 used the same effective axial stiffness [$(E_cA_e)_{ANSYS}$] values for the hoop and vertical directions (see assumption described in RAI 5). However, the criterion used in Appendix H (page 84 of Reference 2) to determine the convergence of the effective sectional stiffness values between the parametric ANSYS analyses and the ACI 224.2R-estimated values [for the two directions] does not seek to satisfy nor does it satisfy the assumption that the effective stiffness in the two directions are considered equal. Instead, it averages the ACI 224.2R-estimated vertical and horizontal effective stiffness (see Figure H.2 in Appendix H of Reference 2). The average curve so obtained intersects the line representing the ratio $(E_cA_e)_{ACI\,224}/(E_cA_e)_{ANSYS}=1$ at two points corresponding to the ratio, $(E_cA_e)_{ANSYS}/E_cA_g$, of 0.4 and 0.7. It can be noted from Figure H.2 that the ACI 224.2R-estimated effective stiffness are not equal in the two directions for both of these values. The smaller of the two values (with no explanation provided), $0.4\ E_cA_g$, was selected as the reduced effective stiffness and was used for the containment opening area in the concrete cracking analysis, even though the larger value would occur earlier when cracking occurs.

Please explain the basis for the criterion used to determine the effective stiffness value with regard to the SONGS containment analysis.

(b) Assuming that the data and assumptions in Figure H.2 are correct, it appears that the appropriate criterion to be used to determine the converged value of the effective axial stiffness between the ANSYS parametric study and the ACI 224.2R-estimated values should be the value of $(E_cA_e)_{ANSYS}$ for which:

$$[(E_cA_e)_{ACI\,224}/(E_cA_e)_{ANSYS}]_{Hoop} = [(E_cA_e)_{ACI\,224}/(E_cA_e)_{ANSYS}]_{Vertical} = \sim 1$$

This criterion also satisfies the assumption that the effective stiffness in the two directions are equal. These ratios for the two directions are not expected to converge exactly to 1 because of the approximations in the 1-dimensional ACI 224.2R method relative to the 3-dimensional ANSYS parametric analyses, but would likely be roughly close to 1.

Accordingly, from Figure H.2 on page 84 of Reference 2, the converged value of the reduced effective axial stiffness to be used in the SONGS containment analysis would be the value corresponding to the intersection of the vertical and hoop curves, which is $0.55E_c A_g$, with the ratio $(E_cA_e)_{ACI\ 224} / (E_cA_e)_{ANSYS}$ being approximately 0.9 (close to 1).

Regarding this approach, please address the impact of the noted difference in the effective stiffness value on the SONGS end-of-life containment analyses, while also considering the questions raised in all of the other RAIs. Alternatively, please justify why the value of effective stiffness used ($0.4\ E_cA_g$) by the licensee is appropriate, considering the issues raised in paragraph (a) above and in all of the other RAIs, as applicable.

RAI 7

Assuming that the forces and moments at the concrete sections expected to be cracked, obtained on the basis of the uncracked ANSYS analysis, are reacted entirely by the combination of unbonded prestressing tendons and bonded reinforcing steel, please provide the following information for each of the hoop and vertical directions for the critical load combinations in the SONGS containment EOL analysis:

(a) the maximum tensile stress in the prestressing tendons,

(b) the maximum tensile stress in the reinforcement for the primary forces in the load combination,

(c) the maximum tensile stress and the maximum strain in the bonded reinforcement for the combined primary and secondary forces in the load combination, and

(d) the maximum stress and strain, as appropriate, in the liner (please indicate if tensile or compressive).

RAI 8

Concrete cracking could also result in reduction in flexural stiffness and shear stiffness that could contribute to redistribution of moments and forces, which have not been considered in the SONGS analyses accounting for concrete cracking in Reference 2. Therefore, please provide the justification as to why the end-of-life evaluation of the SONGS containment following steam generator replacement in Reference 2 selectively considered only reduction in axial tensile stiffness, and resulting redistribution of tensile membrane forces, due to concrete cracking.

References:

1. Calculation No. C-257-01.04.05, ECP No. 061200409-6 R0, Evaluation of Restored Containment - Concrete Modulus Ratio and Tendon Retensioning Forces, SONGS, Unit 2 and Unit 3.

2. Calculation No. C-257-01.04.06, ECN/Prelim CCN No. D0020134, Evaluation of Restored Containment End-of-Life Analysis, SONGS, Unit 2 and Unit 3.

response to the RAI within 30 days of receipt of this letter, so that the staff may resolve the unresolved item in a timely manner.

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

Sincerely,

/RA by N. Kalyanam for/

James R. Hall, Senior Project Manager
Plant Licensing Branch IV
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket Nos. 50-361 and 50-362

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