June 5, 2006

Mr. Richard M. Rosenblum
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 - EXEMPTION FROM THE REQUIREMENTS OF APPENDIX G TO 10 CFR PART 50 (TAC NOS. MC5773 AND MC5774)

Dear Mr. Rosenblum:

Pursuant to Title 10 of the *Code of Federal Regulations*, Part 50, Section 50.12 (10 CFR 50.12), the Commission has granted an exemption from specific requirements of 10 CFR Part 50, Appendix G, for the San Onofre Nuclear Generating Station, Units 2 and 3. This action is necessitated in response to your letter dated January 28, 2005, as supplemented by letter dated January 12, 2006, which, in part, requested to amend your facility licenses to use the methodology for calculating flaw stress intensity factors due to internal pressure loadings (K_{IM}) values as specified in Combustion Engineering Topical Report NPSD-683-A, Revision 6.

A copy of the exemption has been forwarded to the Office of the Federal Register for publication.

Your amendment request, which proposes to revise the Technical Specifications and relocate the reactor coolant system pressure-temperature curves and limits to a licensee-controlled document identified as the Pressure and Temperature Limit Report, is being reviewed and will be addressed separately from this exemption request, which, as noted above, is granted in the document included with this letter.

Sincerely,

/RA/

N. Kalyanam, Project Manager
Plant Licensing Branch IV
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket Nos. 50-361 and 50-362

Enclosure: Exemption

cc w/encl: See next page

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Docket Nos. 50-361 and 50-362

Enclosure: Exemption

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UNITED STATES NUCLEAR REGULATORY COMMISSION

SOUTHERN CALIFORNIA EDISON COMPANY

SAN DIEGO GAS AND ELECTRIC COMPANY

THE CITY OF RIVERSIDE, CALIFORNIA

THE CITY OF ANAHEIM, CALIFORNIA

DOCKET NOS. 50-361 AND 50-362

SAN ONOFRE NUCLEAR GENERATING STATION, UNITS 2 AND 3

EXEMPTION

1.0 BACKGROUND

Southern California Edison Company (the licensee) is the holder of Facility Operating License Nos. NPF-10 and NPF-15, which authorize operation of the San Onofre Nuclear Generating Station, Unit 2 and Unit 3 (SONGS 2 and 3), respectively. The licenses provide, among other things, that the facility is subject to all rules, regulations, and orders of the U.S. Nuclear Regulatory Commission (NRC, the Commission) now or hereafter in effect.

The facility consists of two pressurized-water reactors located in San Diego County, California.

2.0 REQUEST/ACTION

Title 10 of the *Code of Federal Regulations* (10 CFR), Part 50, Appendix G, which is invoked by 10 CFR 50.60, requires that pressure-temperature (P-T) limits be established for reactor pressure vessels (RPVs) during normal operating and hydrostatic or leak rate testing conditions. Specifically, 10 CFR Part 50, Appendix G, states that "[t]he appropriate requirements on both the pressure-temperature limits and the minimum permissible temperature must be met for all conditions," and "[t]he pressure-temperature limits identified as 'ASME [American Society for Mechanical Engineers] Appendix G limits' in Table 3 require that the limits must be at least as conservative as limits obtained by following the methods of analysis and the margins of safety of Appendix G of Section XI of the ASME Code

[Boiler and Pressure Vessel Code]." Part 50 of Title 10 of the *Code of Federal Regulations*, Appendix G, also specifies that the editions and addenda of the ASME Code, Section XI, which are incorporated by reference in 10 CFR 50.55a, apply to the requirements in 10 CFR Part 50, Appendix G. In the 2005 Edition of the *Code of Federal Regulations*, the 1977 Edition through the 2003 Addenda of the ASME Code, Section XI are incorporated by reference in 10 CFR 50.55a. Finally, 10 CFR 50.60(b) states that, "[p]roposed alternatives to the described requirements in Append[ix] G ... of this part or portions thereof may be used when an exemption is granted by the Commission under [10 CFR 50.12]."

In the licensee's January 28, 2005, license amendment request to implement a pressure-temperature limits report (PTLR) for SONGS 2 and 3, the licensee identified Combustion Engineering (CE) Owners Group Topical Report NPSD-683-A, "The Development of a RCS [Reactor Coolant System] Pressure and Temperature Limits Report for the Removal of P-T Limits and LTOP [low temperature overpressure protection] Setpoints from the Technical Specifications," as the PTLR methodology that would be cited in the administrative control section of the SONGS 2 and 3 Technical Specifications governing PTLR content. CE NPSD-683-A refers to an NRC-approved version of Topical Report CE NPSD-683. The NRC staff evaluated the specific PTLR methodology in CE NPSD-683, Revision 6. This evaluation was documented in the NRC safety evaluation (SE) of March 16, 2001, which specified additional licensee actions that are necessary to support a licensee's adoption of CE NPSD-683, Revision 6. The final approved version of this report was reissued as CE NPSD-683-A, Revision 6, which included the NRC SE and the required additional action items as an attachment to the report. One of the additional specified actions stated that if a licensee proposed to utilize the methodology in CE NPSD-683, Revision 6, for the calculation of flaw stress intensity factors due to membrane stress from pressure loading (K_{IM}), an exemption was required since the methodology for the calculation of K_{IM} values in CE NPSD-683, Revision 6, could not be shown to be conservative with respect to the methodology for the determination of K_{IM} provided in editions and addenda of the ASME Code, Section XI, Appendix G, through the 2003 Addenda. Therefore, in connection with the licensee's

January 28, 2005, license amendment request, as supplemented by its letter dated January 12, 2006, the licensee also submitted an exemption request, consistent with the requirements of 10 CFR 50.60, to apply the K_{IM} calculational methodology of CE NPSD-683-A, Revision 6, as part of the SONGS 2 and 3 PTLR methodology.

During the NRC staff's review of CE NPSD-683, Revision 6, the NRC staff evaluated the K_{IM} calculational methodology of CE NPSD-683, Revision 6, versus the methodologies for K_{IM} calculation given in the ASME Code, Section XI, Appendix G. In the staff's March 16, 2001 SE, the staff noted, "[t]he CE NSSS [nuclear steam supply system] methodology does not invoke the methods in the 1995 edition of Appendix G to the Code for calculating K_{IM} factors, and instead applies FEM [finite element modeling] methods for estimating the K_{IM} factors for the RPV shell ... the staff has determined that the K_{IM} calculation methods apply FEM modeling that is similar to that used for the determination of the K_{IT} factors [as codified in the ASME Code, Section XI, Appendix G]. The staff has also determined that there is only a slight non-conservative difference between the P-T limits generated from the 1989 edition of Appendix G to the Code and those generated from CE NSSS methodology as documented in Evaluation No. 063-PENG-ER-096, Revision 00. The staff considers that this difference is reasonable and that it will be consistent with the expected improvements in P-T generation methods that have been incorporated into the 1995 edition of Appendix G to the Code."

In summary, the staff concluded in its March 16, 2001, SE that the calculation of K_{IM} using the CE NPSD-683, Revision 6, methodology would lead to the development of P-T limit curves, which may be slightly non-conservative with respect to those which would be calculated using the ASME Code, Section XI, Appendix G, and that such a difference was to be expected with the development of more refined calculational techniques. Furthermore, the staff concluded in its March 16, 2001, SE that P-T limit curves that would be developed using the methodology of CE NPSD-683, Revision 6, would be adequate for protecting the RPV from brittle fracture under all normal operating and hydrostatic/leak test conditions.

3.0 DISCUSSION

Pursuant to 10 CFR 50.12, the Commission may, upon application by any interested person or upon its own initiative, grant exemptions from the requirements of 10 CFR Part 50 when (1) the exemptions are authorized by law, will not present an undue risk to public health or safety, and are consistent with the common defense and security; and (2) when special circumstances are present.

This exemption results in changes to the plant by allowing the use of an alternative methodology for calculating flaw stress intensity factors in the reactor pressure vessel due to membrane stress from pressure loadings in lieu of meeting the requirements in 10 CFR 50.60. As stated above, 10 CFR 50.12 allows NRC to grant exemptions from the requirements of 10 CFR Part 50. In addition, the granting of the exemption will not result in violation of the Atomic Energy Act of 1954, as amended, or the Commission's regulations. Therefore, the exemption is authorized by law.

The underlying purpose of 10 CFR 50.60 and 10 CFR Part 50, Appendix G, is to ensure that appropriate pressure-temperature limits and the minimum permissible temperature are established for the reactor pressure vessel under normal operating and hydrostatic or leak rate conditions. The licensee's alternative methodology for establishing the P-T limits and low-temperature overpressure protection setpoints are described in Combustion Engineering Owners' Topical Report NPSD-683-A, and has been approved by the NRC staff. Based on the above, no new accident precursors are created by using the alternative methodology, thus, the probability of postulated accidents is not increased. Also, based on the above, the consequences of postulated accidents are not increased. In addition, the licensee will use an NRC-approved methodology for establishing P-T limits and minimum permissible temperatures for the reactor vessel. Therefore, there is no undue risk to the public health and safety.

The exemption results in changes to the plant by allowing an alternative methodology for calculating flaw stress intensity factors in the reactor vessel. This change to the calculation of stresses in the reactor vessel material has no relation to security issues. Therefore, the common defense and security is not impacted by this exemption.

Special circumstances, pursuant to 10 CFR 50.12(a)(2)(ii), are present in that continued operation of SONGS 2 and 3 with P-T limit curves developed in accordance with the ASME Code, Section XI, Appendix G, without the authorization to utilize the alternative K_{IM} calculational methodology of CE NPSD-683-A, Revision 6, is not necessary to achieve the underlying purpose of 10 CFR Part 50, Appendix G. Application of the K_{IM} calculational methodology of CE NPSD-683-A, Revision 6, in lieu of the calculational methodology specified in the ASME Code, Section XI, Appendix G, provides an acceptable alternative evaluation procedure, which will continue to meet the underlying purpose of 10 CFR Part 50, Appendix G. The underlying purpose of the regulations in 10 CFR Part 50, Appendix G, is to provide an acceptable margin of safety against brittle failure of the RCS during any condition of normal operation to which the pressure boundary may be subjected over its service lifetime.

Based on the staff's March 16, 2001, SE regarding CE NPSD-683, Revision 6, and the licensee's rationale to support the exemption request, the staff accepts the licensee's determination that an exemption would be required to approve the use of the K_{IM} calculational methodology of CE NPSD-683-A, Revision 6. The staff concludes that the application of the technical provisions of the K_{IM} calculational methodology of CE NPSD-683-A, Revision 6, by SO*NGS 2 and 3 provides sufficient margin in the development of RPV P-T limit curves such that the underlying purpose of the regulations (10 CFR Part 50, Appendix G) continues to be met. Therefore, the NRC staff concludes that the exemption requested by the licensee is justified based on the special circumstances of 10 CFR 50.12(a)(2)(ii), "[a]pplication of the regulation in the particular circumstances would not serve the underlying purpose of the rule or is not necessary to achieve the underlying purpose of the rule."

Based upon a consideration of the conservatism that is explicitly incorporated into the methodologies of 10 CFR Part 50, Appendix G, and ASME Code, Section XI, Appendix G, the staff concludes that application of the K_{IM} calculational methodology of CE NPSD-683-A, Revision 6, as described, would provide an adequate margin of safety against brittle failure of the RPV. Therefore, the staff concludes that the exemption is appropriate under the special circumstances of 10 CFR

50.12(a)(2)(ii), and that the application of the technical provisions of the K_{IM} calculational methodology of CE NPSD-683-A, Revision 6, should be approved for use in the SONGS 2 and 3 PTLR methodology.

4.0 CONCLUSION

Accordingly, the Commission has determined that, pursuant to 10 CFR 50.12(a), the exemption is authorized by law, will not present an undue risk to the public health and safety, and is consistent with the common defense and security. Also, special circumstances are present. Therefore, the Commission hereby grants Southern California Edison Company an exemption from the requirements of 10 CFR Part 50, Appendix G, to allow application of the K_{IM} calculational methodology of CE NPSD-683-A, Revision 6, in establishing the PTLR methodology for SONGS 2 and 3.

Pursuant to 10 CFR 51.32, the Commission has determined that the granting of this exemption will not have a significant effect on the quality of the human environment (71 FR 19553; dated April 14, 2006).

This exemption is effective upon issuance.

Dated at Rockville, Maryland, this 5th day of June 2006.

FOR THE NUCLEAR REGULATORY COMMISSION

/RA/

Catherine Haney, Director
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