UNITED STATES
NUCLEAR REGULATORY COMMISSION
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ENCLOSURE

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO REACTOR VESSEL EMBRITTLEMENT (GL 88-11)

SOUTHERN CALIFORNIA EDISON COMPANY

SAN ONOFRE NUCLEAR GENERATING STATION, UNIT NO. 1

DOCKET NO. 50-206

1.0 INTRODUCTION

Generic Letter (GL) 88-11, "NRC Position on Radiation Embrittlement of Reactor Vessel Materials and Its Impact on Plant Operations," dated July 12, 1988, identified new NRC staff positions regarding the calculation of radiation embrittlement of reactor vessel beltline materials. The generic letter referred licensees to Revision 2 of Regulatory Guide (RG) 1.99, "Radiation Embrittlement of Reactor Vessel Materials," dated May 1988, for addressing reactor vessel pressure-temperature (P-T) limits and for performing analyses, other than pressurized thermal shock (PTS) analyses, that require an estimate of reactor vessel beltline material embrittlement. Revision 2 of RG 1.99 provides revised procedures for calculating the reactor vessel adjusted reference temperature (ART). Procedures for calculating the decrease in Charpy upper-shelf energy (USE) were not changed by Revision 2 of the regulatory guide because preparatory work in this regard had not been completed.

Generic Letter 88-11 requested that licensees predict the effect of neutron radiation on reactor vessel materials as required by 10 CFR 50, Appendix G, using the methods prescribed by RG 1.99, Revision 2, or justify the use of other methods. The generic letter recognized that licensees may be required to make changes to existing Technical Specification (TS) P-T limits as a result of the calculations being performed.

Generic Letter 88-11 also indicated that the NRC staff was considering an amendment to 10 CFR 50.61 related to PTS, which would change the existing equations for calculating the PTS reference temperature. The generic letter suggested that licensees calculate the PTS reference temperature by using the methodology of RG 1.99, Revision 2, to determine if reactor vessel neutron fluences should be reduced over the life of the plant.

2.0 DISCUSSION

Southern California Edison Company (SCE or the licensee) provided its response to GL 88-11 by letter dated December 16, 1988, for San Onofre Nuclear Generating Station, Unit No. 1 (SONGS-1).

The current P-T limits for SONGS -1, which are specified by TS 3.1.3, were based on the results of Westinghouse analysis WCAP-9520, "Analysis of Capsule F from the Southern California Edison Company, San Onofre Reactor Vessel

Radiation Surveillance Program," dated May 1979, for plant operation through 16 effective full power years (EFPY). Based on that analysis, Westinghouse determined that the limiting adjusted reference temperatures were 217°F and 163°F, for 1/4 inch thickness and 3/4 inch thickness locations, respectively. The licensee's more recent analysis, which was done in accordance with RG 1.99, Revision 2, for 16 EFPY, determined that the limiting ARTs for the 1/4 inch and 3/4 inch thickness locations are 188.7°F and 162.7°F, respectively. This analysis credited continued use of the reactor vessel thermal shield following the Cycle 11 outage. The licensee concluded that these lower ARTs are bounded by the existing Technical Specification requirements and that 10 CFR 50, Appendix G, requirements were still satisfied.

Although not specifically required by GL 88-11, the licensee provided clarification with regard to its compliance with 10 CFR 50, Appendix G, requirements related to Charpy upper-shelf energy by letter dated January 3, 1990. SONGS-1 was previously evaluated relative to 10 CFR 50, Appendix G, requirements during the Systematic Evaluation Program (SEP). The NRC staff's review of this issue was documented in Appendix D of NUREG-0569, "Evaluation of the Integrity of SEP Reactor Vessels," dated December 1979, which stated the following conclusion:

> "...The combination of inservice inspections, conservative pressure-temperature operating limits, low vessel stresses, and the use of materials having acceptable fracture toughness properties provides that the vessel integrity will be maintained at acceptable levels throughout service life. The generic safety items applicable to San Onofre (low upper shelf toughness and overpressurization protection) have been successfully resolved and will not adversely affect the integrity of the vessel."

In its January 3 letter, the licensee further stated that, "...the materials used in the Unit 1 vessel exhibit saturation of material toughness with increasing fluence, and this behavior has been included in the projections of decreasing Charpy upper-shelf energy rather than the relation provided in Figure 2 of Regulatory Guide 1.99." Therefore, the licensee felt that the methods of RG 1.99 were not applicable to SONGS-1 for determining Charpy upper-shelf energy since saturation effects were not considered by the regulatory guide.

By letter dated March 13, 1990, the NRC requested that the licensee provide additional information to facilitate the staff's review of the licensee's response to GL 88-11. The licensee responded to the staff's request by letter dated June 18, 1990.

3.0 EVALUATION

In performing evaluations regarding reactor vessel embrittlement, the staff uses the following NRC regulations and guidance: Appendices G and H of 10 CFR Part 50; the ASTM Standards and the ASME Code, which are referenced by 10 CFR 50, Appendices G and H; 10 CFR 50.36(c)(2); RG 1.99, Revision 2; Standard Review Plan (SRP) Section 5.3.2; and Generic Letter 88-11.

Each licensee authorized to operate a nuclear power reactor is required by 10 CFR 50.36 to provide Technical Specifications for the operation of the plant. In particular, 10 CFR 50.36(c)(2) requires that limiting conditions for operation be included in the Technical Specifications. The P-T limits for the reactor vessel are among the limiting conditions for operation that are included in the Technical Specifications. Appendices G and H of 10 CFR Part 50 describe specific fracture toughness and reactor vessel material surveillance requirements that must be considered in setting the P-T limits. A method acceptable to the staff for constructing P-T limits is described in SRP Section 5.3.2.

Part 50 of Title 10 of the Code of Federal Regulations, Appendix G, requires that fracture toughness and other testing of reactor vessel materials be performed in accordance with the ASME Code and, in particular, that the beltline materials in the surveillance capsules be tested in accordance with Appendix H of 10 CFR Part 50. Appendix H, in turn, refers to ASTM Standards. These tests define the extent of reactor vessel embrittlement at the time of capsule withdrawal in terms of the increase in reference temperature. Appendix G also requires the licensee to predict the effects of neutron irradiation on reactor vessel embrittlement by calculating the adjusted reference temperature and Charpy upper-shelf energy.

Part 50 of Title 10 of the Code of Federal Regulations, Appendix H, requires the licensee to establish a surveillance program to periodically withdraw surveillance capsules from the reactor vessel. Appendix H refers to the ASTM Standards which, in turn, require that the surveillance capsules be installed in the reactor vessel before startup and that they contain test specimens made from plate, weld, and heat-affected-zone (HAZ) materials of the reactor beltline.

<u>Adjusted Reference Temperature</u>

The staff evaluated the effect of neutron irradiation embrittlement on each beltline material in the SONGS-1 reactor vessel. The amount of irradiation embrittlement was calculated in accordance with RG 1.99, Rev. 2. The staff determined that the material with the highest ART at 16 EFPY was intermediate shell plate W7601-9 with 0.18% copper (Cu), 0.20% nickel (Ni), and an initial reference temperature (RT_{ndt}) of 55°F.

The licensee has removed three of the SONGS-1 surveillance capsules. The results from capsules A and D were published by Southwest Research Institute. The results from capsule F were published in Westinghouse Report WCAP-9520. Surveillance capsule A contained Charpy impact specimens and tensile specimens made from base metal plate W7601-9, weld metal, and HAZ metal. Surveillance capsule D contained Charpy impact specimens and tensile specimens made from base metal plates W7601-1, W7601-8 and W7601-9. Surveillance capsule F contained Charpy impact specimens and tensile specimens made from base metal plate W7601-8, weld metal, and HAZ metal. Only specimens from the longitudinal direction were available in the irradiated condition, but specimens in both longitudinal and transverse directions were available in the unirradiated condition.

For the limiting beltline material (plate W7601-9), the staff calculated the

ART to be 199°F at 1/4T (T = reactor vessel beltline thickness) for 16 EFPY. The staff used a neutron fluence of 2.61E19 n/cm^2 at 1/4T. The ART was determined by the least squares extrapolation method described in Section 2.1 of RG 1.99, Rev. 2.

The current P-T limits for SONGS-1 are based on an ART of 217°F at 1/4T and an ART of 163°F at 3/4 T for plates W7601-9 and W7601-5, respectively. These ARTs were calculated in accordance with RG 1.99, Revision 1. In response to GL 88-11, the licensee calculated ARTs in accordance with RG 1.99, Revision 2, which yielded ART values that were less than the 217°F ART which forms the basis for the current SONGS-1 P-T limits. Since the lower ART values are more conservative and represent additional margin, they are bounded by the existing Technical Specification requirements for P-T limits. The staff verified that the licensee's current P-T limits for heatup, cooldown and hydrostatic testing remain valid by substituting an ART of 199°F into the equations in SRP 5.3.2. Therefore, the licensee's current Technical Specifications are acceptable in this regard.

Charpy Upper-Shelf Energy

The NRC staff reviewed the information submitted by the licensee in letters dated January 3 and June 18, 1990, regarding determination of Charpy upper-shelf energy for the SONGS-1 reactor vessel. This issue was reviewed previously during SEP, and the staff concluded at that time that the matter was satisfactorily resolved for SONGS-1. This position was stated in Appendix D of NUREG-0569, "Evaluation of the Integrity of SEP Reactor Vessels," dated December 1979. Revision 2 of RG 1.99 only changed the procedures for adjusting the reactor vessel reference temperature and did not change the existing procedures for calculating the decrease in Charpy upper-shelf energy. The staff is considering making a revision to existing procedures for calculating Charpy upper-shelf energy, however, and a future revision of RG 1.99 will reflect the staff's position when it has been fully developed. Any actions specifically required by the licensee as a result of the new staff position in this regard will be addressed by generic correspondence.

4.0 CONCLUSION

The staff concludes that the current P-T limits for the reactor coolant system for heatup, cooldown, leak test, and criticality are valid through 16 EFPY because the limits conform to the requirements of Appendices G and H of 10 CFR Part 50. Although the current P-T limits are based on calculations that were performed in accordance with RG 1.99, Rev.1, the limits are more conservative than those that would be required based on RG 1.99, Revision 2, calculations. Therefore, the current P-T limits specified by the SONGS-1 Technical Specifications remain valid and are acceptable.

Fracture toughness of the SONGS-1 reactor vessel was reviewed previously as a function of SEP and found to be acceptable by the NRC staff as documented in NUREG-0569. However, the staff believes that there is insufficient data to support the saturation effects on the Charpy upper-shelf energy of the reactor vessel materials and the licensee should continue to evaluate the Charpy upper-shelf energy as more surveillance data becomes available.

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Dated: November 5, 1990

5.0 REFERENCES

1. Regulatory Guide 1.99, Radiation Embrittlement of Reactor Vessel Materials, Revision 2, May 1988

2. NUREG-0800, Standard Review Plan, Section 5.3.2 Pressure-Temperature Limits

3. December 16, 1988, Letter from M. O. Medford (SCE) to USNRC Document Control Desk, Subject: Response to Generic Letter 88-11

4. November 10, 1977, Letter from K. P. Baskin (SCE) to USNRC Document Control Desk, Subject: Reactor Vessel Surveillance Program

5. San Onofre Nuclear Generating Station Updated Final Safety Analysis Report, Section 5.2

6. April 18, 1980, Letter from K. P. Baskin (SCE) to H. R. Denton (USNRC), Subject: Heatup and Cooldown Curves (also includes WCAP-9520, "Analysis of Capsule F from the Southern California Edison Company, San Onofre Reactor Vessel Radiation Surveillance Program")

7. "San Onofre Nuclear Generating Station Unit Analysis of Second Surveillance Material Capsule," prepared by Southwest Research Institute, July 1972

8. "Analysis of First Surveillance Material from San Onofre Unit 1," prepared by Southwest Research Institute, July 1971

9. January 3, 1990, letter from R.M Rosenblum (SCE) to NRC Document Control Desk, Subject: 10 CFR 50, Appendix G, Fracture Toughness SONGS Unit 1.

10. June 18, 1990, letter from F. R. Nandy (SCE) to NRC Document Control Desk, Subject: 10 CFR 50 Appendix G, Fracture Toughness (GL 88-11) SONGS Unit 1.