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

ENCLOSURE 1

EVALUATION OF PROPOSED TECHNICAL SPECIFICATION CHANGE

FOR SAN ONOFRE NUCLEAR GENERATING STATION-UNIT 1

(PROPOSED CHANGE NO. 137)

OFFICE OF NUCLEAR REACTOR REGULATION

1.0 INTRODUCTION

In a letter from K. P. Baskin to H. R. Denton dated May 17, 1984 the Southern California Edison Company requested a revision to the San Onofre-1 Appendix A Technical Specification 3.1.3, "Combined Heatup, Cooldown and Pressure Limitation." To support this change the licensee referenced Westinghouse WCAP-9520, "Analysis of Capsule F from the Southern California Edison Company, San Onofre Reactor Vessel Radiation Surveillance Program." This analysis was submitted for staff review in a letter from K. P. Basin to H. R. Denton dated April 18, 1980.

2.0 EVALUATION

The purpose of the reactor vessel surveillance program is to monitor the effect that neutron irradiation and the thermal environment will have on the beltline materials' reference temperature (RT_{NDT}). The method recommended by the staff for predicting the effect of neutron irradiation and the thermal environment on beltline materials' RT_{NDT} is documented in Regulatory Guide 1.99, "Effects of Residual Elements on Predicted Radiation Damage to Reactor Vessel Materials." In revision 1 to this Regulatory Guide, the increase in RT_{NDT} is dependent upon the amount of copper, phosphorus and neutron fluence. In draft revision 2 dated July 23, 1984 (Ref. 1) the increase in RT_{NDT} is dependent upon the amount of copper, nickel and neutron fluence. Draft revision 2 was prepared from analysis of commercial reactor vessel material surveillance data generated during the staff's review of the issue of "Pressurized Thermal Shock" and is being prepared for review by the Committee for Review of Generic Requirements (CRGR). Table I compares the actual increase in RT_{NDT} for the Capsule F material surveillance data to the amounts predicted by Regulatory Guide 1.99 Rev. 1 and 2. This comparison indicates that the actual increase in RT_{NDT} for the surveillance materials was less than that predicated by Regulatory Guide 1.99, Rev. 1 or Rev. 2. However, the increase predicted by revision 2 was much closer to that measured on the surveillance materials. Hence, the staff believes that revision 2 would be more appropriate for predicting the increase in RT_{NDT} for the San Onofre-1 beltline materials.

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Pressure-temperature limits must be calculated in accordance with the requirements of Appendix G to 10 CFR Part 50, which became effective on July 26, 1983. Pressure-temperature limits that are calculated in accordance with the requirements of Appendix G, 10 CFR Part 50, are dependent upon the initial RT_{NDT} for the limiting materials in the beltline and closure flange regions of the reactor vessel and the increase in RT_{NDT} resulting from neutron irradiation damage to the limiting beltline material.

The initial RT_{NDT} for the limiting materials in the San Onofre-1 reactor vessel beltline and closure flange regions were 82°F and 60°F, respectively. These values were estimated by the licensee using the method recommended by the staff in Branch Technical Position MTEB 5-2, Standard Review Plan 5.3.2 (NUREG-0800, Rev. 1 dated July 1981). The licensee indicates that the increase in RT_{NDT} would result in an adjusted RT_{NDT} at the 1/4 thickness location of 217°F after 16 effective full power years (EFPY) of operation. This adjusted RT_{NDT} was used to calculate the licensee's pressure-temperature curves.

As previously discussed, the increase in RT_{NDT} predicted using Draft Regulatory Guide 1.99, Rev. 2 is dependent upon the neutron fluence and the amount of copper and nickel in the limiting beltline material. Using the method recommended in this draft Regulatory Guide, the adjusted RT_{NDT} for the limiting beltline material in the San Onofre-1 beltline at the 1/4 thickness location after 16 EFPY of operation is 208°F.

The limiting beltline material is plate W7601-5, which the licensee indicates has 0.14 percent copper. The staff has used the licensee's estimates of the amount of copper and neutron fluence and an assumed nickel of 0.20 percent to calculate the adjusted RT_{NDT} for the limiting beltline material. Since the adjusted reference temperature predicted using Draft Regulatory Guide 1.99, Rev. 2 is less than that used to calculate the San Onofre 1 pressure-temperature limits, the staff concludes that the pressure-temperature limits provide sufficient margin to account for neutron irradiation damage for 16 EFPY of operation.

Appendix G to 10 CFR Part 50, also requires that the temperature of the flange regions that are highly stressed by bolt preload must exceed the RT_{NDT} of the materials in those regions by at least 120°F for normal operation and 90°F for hydrostatic pressure and leak tests whenever the pressure exceeds 20 percent of the preservice system hydrostatic test pressure unless a lower temperature can be justified by showing that the margins of safety for those regions are equivalent to those required for the beltline, when it is controlling. The licensee indicates in its May 17, 1984 letter that the curves were revised based on the requirements in Appendix G to 10 CFR Part 50. The staff has not completed its evaluation of this statement and whether the beltline region in San Onofre 1 reactor vessel is more limiting than the closure flange region. However, since the surveillance capsule data indicate that the licensee's pressure-temperature curves have adequate safety margins, the staff concludes that it is acceptable for the licensee to utilize the

pressure-temperature curves until the completion of Cycle 8. We request that the licensee clarify the method of determining that the beltline region is more limiting than the closure flange region. The requested information should be sent for staff review and approval at least six months prior to the start of Cycle 9.

3.0 REFERENCE

Draft Regulatory Guide 1.99, Revision 2, "Radiation Damage to Reactor Vessel Materials," Working Paper C, July 23, 1984, Office of Nuclear Regulatory Research.

4.0 ACKNOWLEDGEMENT

B. Elliot prepared this evaluation.

TABLE I

COMPARISON OF ACTUAL INCREASE IN REFERENCE TEMPERATURE FOR MATERIAL IN CAPSULE F TO THAT PREDICTED BY REGULATORY GUIDE 1.99, REV. 1 AND REV. 2

Surveillance Materials	Neutron Fluence $F \times 10^{19}$ n/cm^2	Increase in Ref. Temp. (°F), R.G. 1.99 Actual	Rev. 1	Rev.2*
Corr. Monitor	1.20	120	192	139
	2.36	150	269	159
	5.14	130	385	174
Plate W7601-8	2.36	110	246	149
	5.14	120	363	174
Weld Metal	1.20	80	213	161
	5.14	145	385	196

*R.G. 1.99 Rev. 2 is mean increase in reference temperature plus 2 sigma uncertainty in prediction method.

ENCLOSURE 2

SOUTHERN CALIFORNIA EDISON COMPANY

SAN ONOFRE NUCLEAR UNIT 1

DOCKET NO. 50-206

REQUEST FOR ADDITIONAL INFORMATION

1. Provide the nickel composition for all plate materials in the reactor beltline.

2. Provide pressure temperature limit curves that comply with explicit closure flange material temperature requirements of the revised (May 27, 1983), Appendix G, 10 CFR 50, or provide responses to Items 3 and 4 below:

3. Provide the analysis that shows that the closure flange region is less limiting than the beltline region. Include as minimum the following information:

   a. A description of the finite element analysis used to determine the stresses within the closure flange region.

   b. Indicate the peak bolt-up, pressure, and thermal stresses determined by the finite element analysis at the inside and outside surface locations of the flange to head and flange to shell junctions.

   c. Indicate how the bolt-up, pressure, and thermal stresses were combined to determine the maximum applied stress intensity factors.

   d. Indicate the flaw geometry used to calculate the maximum applied stress intensity factors.

   e. Indicate the maximum applied stress intensity factors for the flange-to-head and flange-to-shell junctions.

   f. Indicate the non-destructive examination methods that will be used during inservice examination to determine that the critical flaw size, which was used in determining the maximum applied stress intensity factors, is not within the flange-to-head and flange-to-shell junctions.

   g. Indicate whether the non-destructive examination methods identified in (f) have been evaluated to demonstrate that the examination methods are capable of locating and sizing flaws of the geometry used for calculating the maximum applied stress intensity factors. Indicate the results of the evaluation.

4. In order to demonstrate that the beltline region is more limiting than the flange region, indicate the minimum metal temperature at the flange and beltline regions which result from the fracture analysis. During a heatup and cooldown what are the required minimum water temperatures to ensure that the limiting flange locations will be equal to or greater than the required minimum metal temperatures?