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

ENCLOSURE

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RESPONSE TO GENERIC LETTER 88-11

SYSTEM ENERGY RESOURCES, INC.

GRAND GULF NUCLEAR STATION, UNIT 1

DOCKET NO. 50-416

1.0 INTRODUCTION

In response to Generic Letter 88-11, "NRC Position on Radiation Embrittlement of Reactor Vessel Materials and Its Effect on Plant Operations," System Energy Resources, Inc. (the licensee), requested permission to revise the pressure/temperature (P/T) limits in the Grand Gulf Nuclear Station, Unit 1, Technical Specifications, Section 3/4.4.6. The request was documented in letters from the licensee dated January 9, 1989, February 28, 1989 and October 10, 1989. This revision also changes the effectiveness of the P/T limits from 32 to 10 effective full power years (EFPY). The proposed P/T limits were developed based on Section 1 of Regulatory Guide 1.99, Revision 2, (RG 1.99, Rev. 2). The proposed revision provides up-to-date P/T limits for the operation of the reactor coolant system during heatup, cooldown, criticality, and hydrotest.

To evaluate the P/T limits, the staff used 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 in Appendices G and H; 10 CFR 50.36(c)(2); RG 1.99, Rev. 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 of operation be included in the Technical Specifications. The P/T limits are among the limiting conditions of operation in the Technical Specifications for all commercial nuclear plants in the U.S. Appendices G and H of 10 CFR Part 50 describe specific requirements for fracture toughness and reactor vessel material surveillance that must be considered in setting P/T limits. An acceptable method for constructing the P/T limits is described in SRP Section 5.3.2.

Appendix G of 10 CFR Part 50 specifies fracture toughness and testing requirements for reactor vessel materials 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 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 vessel embrittlement by calculating the adjusted reference temperature (ART) and Charpy upper shelf energy (USE). Generic Letter 88-11 requested that licensees and permittees use the methods in RG 1.99, Rev. 2, to predict the effect of neutron irradiation on reactor vessel materials. This guide defines the ART as the sum of unirradiated reference temperature, the increase in reference temperature resulting from neutron irradiation, and a margin to account for uncertainties in the prediction method.

Appendix H of 10 CFR Part 50 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 capsules be installed in the vessel before startup and that they contain test specimens made from plate, weld, and heat-affected-zone (HAZ) materials of the reactor beltline.

2.0 EVALUATION

The staff evaluated the effect of neutron irradiation embrittlement on each beltline material in the Grand Gulf reactor vessel. The amount of irradiation embrittlement was calculated in accordance with RG 1.99, Rev. 2. The staff has determined that the material with the highest ART at 10 EFPY at 1/4T (T=reactor vessel beltline thickness) was the longitudinal weld 627260 with 0.06% copper (Cu), 1.08% nickel (Ni), and an initial RT_{ndt} of -30°F. The material with the highest ART at 3/4T, however, was plate C2594-2 with 0.04% Cu, 0.63% Ni, and an initial RT_{ndt} of 0°F.

The licensee has not removed any surveillance capsules from the Grand Gulf reactor vessel. The staff has ensured that all surveillance capsules contained Charpy impact specimens and tensile specimens made from base metal, weld metal, and HAZ metal.

For the limiting beltline material at 1/4T , longitudinal weld 627260, the staff calculated the ART to be 25.2°F at 10 EFPY. For the limiting beltline material at 3/4T, longitudinal plate C2594-2, the staff calculated the ART to be 11.3°F at 10 EFPY. The staff used a neutron fluence of 6.5E17 n/cm^2 at 1/4T and 2.9E17 n/cm^2 at 3/4T.

The licensee used the method in RG 1.99, Rev. 2, to calculate an ART of 25.76°F at 10 EFPY at 1/4T for the same limiting weld metal. The staff judges that a difference of 0.56°F between the licensee's ART of 25.76°F and the staff's ART of 25.2°F is acceptable. Substituting the ART of 25.76°F into equations in SRP 5.3.2, the staff verified that the proposed P/T limits for heatup, cooldown, and hydrotest meet the beltline material requirements in Appendix G of 10 CFR Part 50.

In addition to beltline materials, Appendix G of 10 CFR Part 50 also imposes P/T limits based on the reference temperature for the reactor vessel closure flange materials. Section IV.2 of Appendix G states that when the pressure exceeds 20% of the preservice system hydrostatic test pressure, the temperature of the closure flange regions highly stressed by the bolt preload must exceed the reference temperature of the material in those regions by at least 120°F for normal operation and by 90°F for hydrostatic pressure tests and leak tests. Paragraph IV.A.3 of Appendix G states "an exception may be made for boiling water reactor vessels when water level is within the normal range for power operation and the pressure is less than 20 percent of the pre-service system hydrostatic test pressure. In this case the minimum permissible temperature is 60°F (33°C) above the reference temperature of the closure flange regions that are highly stressed by the bolt preload." Based on the flange reference temperature of -30°F, the staff has determined that the proposed P/T limits satisfy Section IV.2 of Appendix G.

Section IV.B of Appendix G requires that the predicted Charpy USE at end of life be above 50 ft-lb. The material with the lowest initial USE is the girth weld with an initial USE of 79 ft-lb. Using the method in RG 1.99, Rev. 2, the predicted Charpy USE of the girth weld metal at the end of life will be approximately 68 ft-lb. This is greater than 50 ft-lb and, therefore, is acceptable.

3.0 CONCLUSION

The staff concludes that the P/T limits proposed in the February 28, 1989 submittal for the reactor coolant system for heatup, cooldown, leak test, and criticality are valid through 10 EFPY because the limits conform to the requirements of Appendices G and H of 10 CFR Part 50. The licensee's submittal also satisfies Generic Letter 88-11 because the licensee used the method in RG 1.99, Rev. 2 to calculate the ART. Hence, the proposed P/T limits may be incorporated into the Grand Gulf 1 Technical Specifications.

4.0 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. Grand Gulf Nuclear Station, Final Safety Analysis Report (NUREG 0831)

4. Grand Gulf Unit 1, Technical Specifications (NUREG 0934)

5. January 9, 1989, Letter from J. G. Cesare, Jr., (SER) to USNRC Document Control Desk, Subject: Response to Generic Letter 88-11

6. February 28, 1989, Letter from T. H. Cloninger (SER) to USNRC Document Control Desk, Subject: Response to Generic Letter 88-11, Updated Information

7. October 13, 1989, Letter from W. T. Cottle (SER) to USNRC Document Control Desk, Subject: Response to Request for Additional Information Regarding Generic Letter 88-11.

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