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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO FACILITY OPERATING LICENSE NO. NPF-12

SOUTH CAROLINA ELECTRIC & GAS COMPANY

SOUTH CAROLINA PUBLIC SERVICE AUTHORITY

VIRGIL C. SUMMER NUCLEAR STATION, UNIT NO. 1

DOCKET NO. 50-395

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," the South Carolina Electric & Gas Company (the licensee) requested permission to keep the pressure/temperature (P/T) limits in the Virgil C. Summer Nuclear Station, Unit 1, (Summer) Technical Specifications, Section 3.4. The request was documented in a letter from the licensee dated January 12, 1989. This request does not change the effectiveness of the P/T limits of 8 effective full power years (EFPY). The original P/T limits were developed based on Regulatory Guide (RG) 1.99, Revision 1, and were compared to the data from surveillance capsules U and V. The request 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 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 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 for 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 to 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

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Section 5.3.2. Appendix G to 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 to 10 CFR Part 50. Appendix H, in turn, refers to the 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, Revision 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 to 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 Summer reactor vessel. The amount of irradiation embrittlement was calculated in accordance with RG 1.99, Revision 2. The staff has determined that the material with the highest ART at 8 EFPY was the lower shell plate (C9923-2) with 0.08% copper (Cu), 0.41% nickel (Ni), and an initial RT_{ndt} of 10°F.

The licensee has removed two surveillance capsules from the Summer reactor vessel. The results from capsules U and V were published in Westinghouse reports WCAP-10814 and WCAP-11726, respectively. Both surveillance capsules contained Charpy impact specimens and tensile specimens made from base metal, weld metal, and HAZ metal.

For the limiting beltline material, plate C9923-2, the staff calculated the ART to be 94°F at 1/4T (T = reactor vessel beltline thickness) for 8 EFPY and a fluence of 9.42E18 n/cm^2. The ART at 3/4T was calculated to be 81°F for 8 EFPY and 3.71E18 n/cm^2. The ART was determined by Section 1 of RG 1.99, Revision 2, because the limiting material was not in the surveillance capsules.

The licensee used the method in RG 1.99, Revision 2, to calculate an ART of 94.5°F at 1/4T and 81.3°F at 3/4T at 8 EFPY for the same limiting plate material. Substituting the ART of 94°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 to 10 CFR Part 50.

In addition to beltline materials, Appendix G to 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. Based on the flange reference temperature of 10°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. Based on data from the Summer Final Safety Analysis Report, the material with the lowest initial USE is intermediate shell plate (A9154-1) with 80.5 ft-lb. Using the method in RG 1.99, Revision 2, the predicted Charpy USE of the plate material at the end of life (6.6E19 n/cm^2) will be 57 ft-lb. This is above 50 ft-lb and, therefore, is acceptable.

3.0 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 8 EFPY because the limits conform to the requirements of Appendices G and H to 10 CFR Part 50. The licensee's submittal also satisfies Generic Letter 88-11 because the licensee used the method in RG 1.99, Revision 2 to calculate the ART. Since this SE only covers operation through 8 EFPY, the licensee will need to submit a revised curve to address operation past the 8 EFPY prior to reaching that level. As the licensee will be pulling a capsule at their next refueling outage (March 1990), the data from that capsule should be available to generate new P/T limits.

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. January 12, 1989, Letter from O. S. Bradham (SCE&G) to USNRC Document Control Desk; Subject: V. C. Summer Nuclear Station Response to Generic Letter 88-11

4. Final Safety Analysis Report for V. C. Summer Nuclear Station

5. R. S. Boggs, et al., "Analysis of Capsule U from the South Carolina Electric and Gas Company Virgil C. Summer Unit 1 Reactor Vessel Radiation Surveillance Program, WCAP-10814," Westinghouse Electric Corporation, June 1985

6. D. G. Colburn, et al., "Analysis of Capsule V from the South Carolina Electric and Gas Company Virgil C. Summer Unit 1 Reactor Vessel Radiation Surveillance Program, WCAP-11726," Westinghouse Electric Corporation, January 1988

Dated: November 15, 1989

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