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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 177 TO FACILITY OPERATING LICENSE DPR-57

AND AMENDMENT NO. 118 TO FACILITY OPERATING LICENSE NPF-5

GEORGIA POWER COMPANY, ET AL.

EDWIN I. HATCH NUCLEAR PLANT, UNITS 1 AND 2

DOCKET NOS. 50-321 AND 50-366

1.0 INTRODUCTION

By letter dated July 15, 1991, the Georgia Power Company, et al. (the licensee), submitted a request for changes to the Edwin I. Hatch Nuclear Plant, Units 1 and 2, Technical Specifications (TS). The requested changes would revise the pressure/temperature (P/T) limits in the Hatch 2 TS 3/4.4.6, "Reactor Vessel Temperature and Pressure." In addition, the licensee requested: (1) to revise Unit 2 Bases Section 3/4.4.6 to reflect the changes in TS 3/4.4.6 and to include a brief description of the use of revised TS curves during inservice hydrostatic leakage testing, and (2) to add this brief description to Unit 1 Bases Section 3.6.B. The proposed P/T limits were requested for 32 effective full power years (EFPY). The proposed P/T limits were developed using Regulatory Guide (RG) 1.99, Revision 2, Generic Letter (GL) 88-11, "NRC Position on Radiation Embrittlement of Reactor Vessel Materials and Its Effect on Plant Operations," which recommends RG 1.99, Revision 2, be used in calculating P/T limits, unless the use of different methods can be justified. The P/T limits provide 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, Revision 2; Standard Review Plan (SRP) Section 5.3.2; and GL 88-11.

Each licensee authorized to operate a nuclear power reactor is required by 10 CFR 50.36 to provide TS for the operation of the plant. In particular, 10 CFR 50.36(c)(2) requires that limiting conditions of operation be included in the TS. The P/T limits are among the limiting conditions of operation in the TS for all commercial nuclear plants in the United States. 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 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 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.

The licensee also proposed to delete the withdrawal schedule of the RPV surveillance capsules from the Unit 2 TS in accordance with NRC GL 91-01. GL 91-01 allows the removal of the surveillance capsule withdrawal schedule from the TS but requires the schedule be incorporated in the Final Safety Analysis Report (FSAR). The licensee should incorporate the schedule in TS Table 4.4.6.1.3-1 into the updated Hatch Unit 2 FSAR, in accordance with GL 91-01. We conclude this proposal is acceptable.

2.0 EVALUATION

The NRC staff evaluated the effect of neutron irradiation embrittlement on each beltline material in the Hatch 2 reactor vessel. The amount of irradiation embrittlement was calculated in accordance with RG 1.99, Revision 2. The staff determined that the material with the highest ART at 32 EFPY at 1/4T (T = reactor vessel beltline thickness) was the lower longitudinal weld, 101-842, with 0.23% copper (Cu), 0.50% nickel (Ni), and an initial RT_{ndt} of -50°F. At 3/4T, the limiting material at 32 EFPY was plate C8553-1 with 0.08% copper (Cu), 0.58% nickel (Ni), and an initial RT_{ndt} of 24°F.

The licensee has removed surveillance capsule No. 3 from Hatch 2. The results from capsule No. 3 were published in General Electric Report SASR 90-104. All surveillance capsules contained Charpy impact specimens and tensile specimens made from base metal, weld metal, and HAZ metal.

For the limiting beltline material, weld 101-842, the staff calculated the ART to be 68.8°F at 1/4T. For plate C8553-1, the ART was 51.9°F for 3/4T. The staff used a neutron fluence of 1E18 n/cm² at 1/4T and 5.2E17 n/cm² at 3/4T. The ART was determined by Section 1 of RG 1.99, Revision 2, because the licensee has removed only one surveillance capsule from the Hatch 2 reactor vessel.

The licensee used the method in RG 1.99, Revision 2, to calculate an ART of 69°F at 32 EFPY at 1/4T for the same limiting weld metal. The licensee's ART 69°F is more conservative than the staff's ART of 68.8°F and, therefore, is acceptable. Substituting the ART of 68.8°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 pre-service 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 10°F, the staff has determined that the proposed P/T limits satisfy Section IV.2 of Appendix G.

Section IV.A of Appendix G requires that the predicted Charpy USE at end of life (EOL) be above 50 ft-lb. The material with the lowest initial USE is the lower intermediate shell plate C8579-2 with 70 ft-lb. Based on Figure 2 of RG 1.99, Revision 2, the staff predicted the USE at EOL to be 61.9 ft-lb. This is greater than 50 ft-lb. and, therefore, is acceptable.

The NRC staff concludes that the proposed P/T limits for the reactor coolant system for heatup, cooldown, leak test, and criticality are valid through 32 EFPY because the limits conform to the requirements of Appendices G and H of 10 CFR Part 50. The proposed P/T limits also satisfy GL 88-11 because the method in RG 1.99, Revision 2 was used to calculate the ART. Hence, the proposed P/T limits may be incorporated into the Hatch 2 TS.

3.0 STATE CONSULTATION

In accordance with the Commission's regulations, the Georgia State official was notified of the proposed issuance of the amendments. The State official had no comments.

4.0 ENVIRONMENTAL CONSIDERATION

The amendments change requirements with respect to installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20 and change surveillance requirements. The NRC staff has determined that the amendments involve no significant increase in the amounts, and no significant change in the types, of any effluents that may be released offsite, and that there is no significant increase in individual or cumulative occupational radiation exposure. The Commission has previously issued a proposed finding that the amendments involve no significant hazards consideration, and there has been no public comment on such finding (56 FR 60117). Accordingly, the amendments meet the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Pursuant to 10 CFR 51.22(b) no environmental impact statement or environmental assessment need be prepared in connection with the issuance of the amendments.

5.0 CONCLUSION

The Commission has concluded, based on the considerations discussed above, that: (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, (2) such activities will be conducted in compliance with the Commission's regulations, and (3) the issuance of the amendments will not be inimical to the common defense and security or to the health and safety of the public.

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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. July 15, 1991, Letter from J. T. Beckham (GP) to USNRC Document Control Desk, Subject: Plant Hatch - Units 1 and 2, NRC Dockets 50-321 and 50-366, Operating Licenses DPR-57 and NPF-5, Request to Revise Technical Specifications: Reactor Vessel Temperature and Pressure Limits

4. T. A. Caine, "E. I. Hatch Nuclear Power Station, Unit 2 Vessel Surveillance Materials Testing and Fracture Toughness Analysis,"SASR 90-104, General Electric Company, May 1991