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

ENCLOSURE 4

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 190 TO FACILITY OPERATING LICENSE NO. DPR-33

AMENDMENT NO. 205 TO FACILITY OPERATING LICENSE NO. DPR-52

AMENDMENT NO. 162 TO FACILITY OPERATING LICENSE NO. DPR-68

TENNESSEE VALLEY AUTHORITY

BROWNS FERRY NUCLEAR PLANT, UNITS 1, 2 AND 3

DOCKET NOS. 50-259, 50-260, AND 50-296

1.0 INTRODUCTION

By letters dated July 19, 1991 and October 24, 1991, the Tennessee Valley Authority (TVA), the licensee, requested permission to revise the pressure/temperature (P/T) limits of the Browns Ferry Nuclear Plant (BFN) Technical Specifications (TS) for Units 1, 2, and 3. More specifically, TVA proposed to revise the P/T limits of TS Section 3.6 in accordance with the guidance of Generic Letter (GL) 88-11, "NRC Position on Radiation Embrittlement of Reactor Vessel Materials and Its Impact on Plant Operations," dated July 12, 1988. TVA's proposed P/T limits were developed using the methodology of Regulatory Guide (RG) 1.99, Revision 2, as recommended by GL 88-11. These P/T limits establish bounding conditions for the operation of the reactor coolant system during heatup, cooldown, criticality, and hydrotest for a cumulative neutron fluence corresponding to 12 effective full-power years (EFPY).

2.0 EVALUATION

In evaluating TVA's proposed P/T limits, the staff referred to 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 GL 88-11.

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). GL 88-11 requested that licensees 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.

2.1 Browns Ferry Nuclear Plant, Unit 1

The staff evaluated the effect of neutron irradiation embrittlement on each beltline material in the BFN, Unit 1 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 12 EFPY was circumferential weld WF-154 between Shell Courses 1 and 2 with 0.31% copper (Cu), 0.50% nickel (Ni), and an initial RT_{ndt} of 20°F.

For the limiting beltline material, circumferential weld WF-154, the staff calculated the ART to be 87.3°F at 1/4T (T = reactor vessel beltline thickness) and 61.6°F for 3/4T at 12 EFPY. The staff used a neutron fluence of 2.86E17 n/cm^2 at 1/4T and 1.98E17 n/cm^2 at 3/4T. The ART was determined per Section 1 of RG 1.99, Rev. 2, because no surveillance capsules have been removed from the Unit 1 reactor vessel. Substituting the ART of 87.3°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.A.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 10°F, the staff has determined that the proposed P/T limits satisfy Section IV.A.2 of Appendix G.

Section IV.A.1 of Appendix G requires that the predicted Charpy USE at end of life be no less than 50 ft-lb. Based on data from the licensee and using the method in RG 1.99, Rev. 2, the staff calculated that the beltline material with the lowest predicted end of life Charpy USE at the end of life at 1/4T is the circumferential weld metal WF-154 at 51.5 ft-lb. This is greater than 50 ft-lb and, therefore, is acceptable.

TVA has not removed any of the surveillance capsules from BFN, Unit 1. All surveillance capsules presently contain Charpy impact specimens and tensile specimens made from base metal, weld metal, and HAZ metal.

2.2 Browns Ferry Nuclear Plant, Unit 2

The staff evaluated the effect of neutron irradiation embrittlement on each belt-line material in the BFN, Unit 2 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 12 EFPY was longitudinal weld in Shell Course 1 with 0.25% copper (Cu), 0.35% nickel (Ni), and an initial RT_{ndt} of 10°F.

For the limiting beltline material, circumferential weld WF-154, the staff calculated the ART to be 70.4°F at 1/4T (T = reactor vessel beltline thickness) and 48.3°F for 3/4T at 12 EFPY. The staff used a neutron fluence of 4.08E17 n/cm^2 at 1/4T and 2.83E17 n/cm^2 at 3/4T. The ART was determined per Section 1 of RG 1.99, Rev. 2, because no surveillance capsules have been removed from the Unit 2 reactor vessel. Substituting the ART of 70.4°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.A.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 10°F, the staff has determined that the proposed P/T limits satisfy Section IV.A.2 of Appendix G.

Section IV.A.1 of Appendix G requires that the predicted Charpy USE at end of life be no less than 50 ft-lb. Based on data from the licensee and using the method in RG 1.99, Rev. 2, the staff calculated that the beltline material

with the lowest predicted end of life Charpy USE at the end of life at 1/4T is plate C2467-2 at 68.1 ft-lb. This is greater than 50 ft-lb and, therefore, is acceptable.

TVA has not removed any surveillance capsules from BFN, Unit 2. All surveillance capsules presently contain Charpy impact specimens and tensile specimens made from base metal, weld metal, and HAZ metal.

2.3 Browns Ferry Nuclear Plant, Unit 3

The staff evaluated the effect of neutron irradiation embrittlement on each beltline material in the BFN, Unit 3 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 12 EFPY was longitudinal weld in Shell Course 1 with 0.25% copper (Cu), 0.35% nickel (Ni), and an initial RT_{ndt} of 10°F.

For the limiting beltline material, circumferential weld WF-154, the staff calculated the ART to be 69.2°F at 1/4T (T = reactor vessel beltline thickness) and 47.4°F for 3/4T at 12 EFPY. The staff used a neutron fluence of 3.94E17 n/cm^2 at 1/4T and 2.73E17 n/cm^2 at 3/4T. The ART was determined per Section 1 of RG 1.99, Rev. 2, because no surveillance capsules have been removed from the Unit 3 reactor vessel. Substituting the ART of 69.2°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.A.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 10°F, the staff has determined that the proposed P/T limits satisfy Section IV.A.2 of Appendix G.

Section IV.A.1 of Appendix G requires that the predicted Charpy USE at end of life be no less than 50 ft-lb. Based on data from the licensee and using the method in RG 1.99, Rev. 2, the staff calculated that the beltline material with the lowest predicted end of life Charpy USE at the end of life at 1/4T is plate C3222-2 at 68.5 ft-lb. This is greater than 50 ft-lb and, therefore, is acceptable.

TVA has not removed any surveillance capsules from BFN, Unit 3. All surveillance capsules presently contain Charpy impact specimens and tensile specimens made from base metal, weld metal, and HAZ metal.

3.0 STATE CONSULTATION

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

4.0 ENVIRONMENTAL CONSIDERATION

The amendments change requirements with respect to installation or use of facility components located within the restricted area as defined in 10 CFR Part 20. The NRC staff has determined that these 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 64662). Accordingly, these 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 staff concludes that the proposed P/T limits for the reactor coolant system for heatup, cooldown, hydrotest, and criticality are valid through 12 EFPY because the proposed 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, Rev. 2 was used to calculate the ART. Therefore, TVA's proposed P/T limits, and associated changes, are acceptable for the BFN, Units 1, 2, and 3 Technical Specifications.

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 amendment will not be inimical to the common defense and security or to the health and safety of the public.

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