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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

SUPPORTING AMENDMENT NO. 81 TO FACILITY OPERATING LICENSE NO. NPF-8

ALABAMA POWER COMPANY

JOSEPH M. FARLEY NUCLEAR PLANT, UNIT 2

DOCKET NO. 50-364

1.0 INTRODUCTION

By letter dated November 23, 1988, Alabama Power Company (the licensee) responded to Generic Letter 88-11, "NRC Position on Radiation Embrittlement of Reactor Vessel Materials and Its Effect on Plant Operations." In its response, the licensee stated that for Joseph M. Farley Nuclear Plant (Farley), Unit 2, the pressure/temperature (P/T) limits contained in the Technical Specifications required revision. By letter dated August 27, 1990, the licensee requested a license amendment to revise the P/T limits. The reqested amendment revises the P/T limits from 8 to 14 effective full power years (EFPY). The proposed P/T limits were developed based on the data from actual surveillance capsules. The proposed revision provides up-to-date P/T limits for the operations 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 American Society of Testing Materials (ASTM) Standards and the American Society of Mechnical Engineers Boiler and Pressure Vessel Code (ASME Code), which are referenced in Appendices G and H; 10 CFR 50.36(c)(2); Regulatory Guide (RG) 1.99, Revision 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. 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 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 the Appendix H of 10 CFR Part 50. Appendix H, in turn, refers to ASTM Standards for surveillance testing requirements. These surveillance 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 Farley, Unit 2, 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 1/4T (T = reactor vessel beltline thickness) at 14 EFPY was intermediate shell plate B7212-1 with 0.20% copper (Cu), 0.60% nickel (Ni), and an initial RT_{ndt} of -10°F. The material with the highest ART at 3/4T was intermediate shell plate B7203-1 with 0.14% copper (Cu), 0.60% nickel (Ni), and an initial RT_{ndt} of 15°F.

The licensee has removed three surveillance capsules from Farley, Unit 2. The results from capsules U, W, and X were published in Westinghouse Reports WCAP-10425, WCAP-11438, and WCAP-12471, respectively. All surveillance capsules contained Charpy impact specimens and tensile specimens made from base metal, weld metal, and HAZ metal.

For the limiting beltline materials, plates B7212-1 and B7203-1, the staff calculated the ART to be 149.5°F at 1/4T and 121.2°F for 3/4T at 14 EFPY. The staff used a neutron fluence of 9.43E18 n/cm^2 at 1/4T and 3.66E18 n/cm^2 at 3/4T. The ART for plate B7212-1 was determined by the least squares extrapolation method using the Farley, Unit 2, surveillance data. The least squares method is described in Section 2.1 of RG 1.99, Revision 2. The ART for plate B7203-1 was determined using Section 1 of RG 1.99, Revision 2, because plate B7203-1 was not in the surveillance capsules.

The licensee calculated an ART of 152°F and 124°F at 1/4T and 3/4T, respectively, for the limiting material. The licensee's ARTs are more conservative than the staff's ARTs; therefore, they are acceptable. Substituting the ART of 149.5°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 of 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.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. Based on the flange reference temperature of 60°F, the staff has determined that the proposed P/T limits satisfy Section IV.2 of Appendix G.

Section IV.A.1 of Appendix G requires that the predicted Charpy USE at end-of-life (EOL) be above 50 ft-lb. The material with the lowest predicted Charpy EOL USE was intermediate shell plate B7212-1 with an unirradiated USE of 99 ft-lb. Using Figure 2 of RG 1.99, Revision 2, the staff calculated that the EOL USE would be 60.4 ft-lb. This is greater than 50 ft-lb and, therefore, is acceptable.

3.0 SUMMARY

The staff concludes that the proposed P/T limits for the reactor coolant system for heatup, cooldown, leak test, and criticality are valid through 14 EFPY as 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 as the licensee used the method in RG 1.99, Revision 2, to calculate the ART. Hence, the proposed P/T limits may be incorporated into the Farley, Unit 2, Technical Specifications.

4.0 ENVIRONMENTAL CONSIDERATION

This amendment changes a requirement with respect to installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20. The staff has determined that the amendment involves no significant increase in the amounts, and no significant change in the types, of any effluents that may be released off site, and that there is no significant increase in individual or cumulative occupational radiation exposure. The Commission has previously issued a proposed finding that this amendment involves no significant hazards consideration, and there has been no public comment on such finding. Accordingly, this amendment meets 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 this amendment.

5.0 CONCLUSION

The Commission made a proposed determination that this amendment involves no significant hazards consideration which was published in the Federal Register (55 FR 40456) on October 3, 1990, and consulted with the State of Alabama. No public comments or requests for hearing were received, and the State of Alabama did not have any comments.

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

6.0 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. November 23, 1988, Letter from W. G. Hairston, III (APCo) to USNRC Document Control Desk, Subject: Generic Letter 88-11.

4. August 27, 1990, Letter from W. G. Hairston, III (APCo) to USNRC Document Control Desk, Subject: Farley 2 RCS Heatup and Cooldown Limit Curves.

5. M. K. Kunka, et al., "Analysis of Capsule U from the Alabama Power Company Joseph M. Farley Unit 2 Reactor Vessel Radiation Surveillance Program," WCAP-10425, Westinghouse Electric Company, October 1983.

6. R. P. Shogan, et al., "Analysis of Capsule W from the Alabama Power Company Joseph M. Farley Unit 2 Reactor Vessel Radiation Surveillance Program," WCAP-11438, Westinghouse Electric Company, April 1987.

7. E. Terek, et al., "Analysis of Capsule X from the Alabama Power Company Joseph M. Farley Unit 2 Reactor Vessel Radiation Surveillance Program," WCAP-12471, Westinghouse Electric Company, December 1989.

Dated: December 31, 1990

Principal Contributor: J. Tsao