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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 103 TO FACILITY OPERATING LICENSE NO. DPR-6

CONSUMERS POWER COMPANY

BIG ROCK POINT PLANT

DOCKET NO. 50-155

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 Consumers Power Company (the licensee) requested permission to revise the pressure/temperature (P/T) limits in the Big Rock Point Plant Technical Specifications, Section 4. The request was documented in letters from the licensee dated January 10, 1990 and January 23, 1990, with a supplement dated August 22, 1990. This revision changes the P/T limits of 18 effective full power years (EFPY). The proposed P/T limits were developed based on Regulatory Guide (RG) 1.99, Revision 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 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 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 Big Rock Point 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 18 EFPY was the beltline plates with 0.1% copper (Cu), 0.18% nickel (Ni), and an initial RT_{NDT} of 30°F. All the beltline plates--S-5503-1, -2, -3, and -4--come from the same heat and have the same chemical composition.

The licensee has removed five surveillance capsules from Big Rock Point. The results from four of the capsules were published in Nuclear Engineering and Design. The results from capsule 125 were published in Westinghouse Report WCAP-9794. All surveillance capsules contained Charpy impact specimens and tensile specimens made from base metal, weld metal, and HAZ metal.

For the limiting beltline material, the beltline plates, the staff calculated the ART to be 174.6°F at 1/4T (T = reactor vessel beltline thickness) and 161.6°F for 3/4T at 18 EFPY. The staff used a neutron fluence of 3.4E19 n/cm^2 at 1/4T and 1.79E19 n/cm^2 at 3/4T. The ART was determined by the least squares extrapolation method using the surveillance data. The least squares method is described in Section 2.1 of RG 1.99, Rev. 2.

The licensee used the method in Section 2 of RG 1.99, Rev. 2, to calculate an ART of 175°F at 18 EFPY at 1/4T for the beltline plates (Ref. 9). The staff judges that a difference of 0.4°F between the licensee's value of 175°F and the staff's ART of 174.6°F is acceptable. Substituting the ART of 174.6°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. Based on the flange reference temperature of 40°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 a surveillance capsule withdrawn after being irradiated with 10.7E19 n/cm² neutrons, the measured Charpy USE is 64 ft-lb for the beltline weld metal. The EOL fluence at the inside diameter is expected to be 4.7E19 n/cm². Since this fluence is smaller than that for the irradiated weld which resulted in a 64 ft-lb USE, the staff considers that the 50 ft-lb EOL USE requirement is satisfied.

3.0 ENVIRONMENTAL CONSIDERATION

This amendment involves a change in the pressure/temperature limits for the operation of the reactor coolant system during heatup, cooldown, criticality and hydrotest. The staff 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 offsite, 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.

4.0 CONCLUSION

The staff concludes that the proposed P/T limits for the reactor coolant system for heatup, cooldown, leak test, and criticality are valid through 18 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 Big Rock Point Technical Specifications.

We have 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, and (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.

5.0 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 10, 1990, Letter from K. W. Berry (CPCo) to USNRC Document Control Desk, Subject: Big Rock Point Plant Technical Specification Change Request--Reactor Temperature Limits

4. January 23, 1990, Letter from J. D. Eddy (CPCo) to USNRC Document Control Desk, Subject: Corrections to Technical Specifications Change Request--Reactor Vessel Pressure/Temperature Limits

5. August 7, 1990, Letter from R. J. Alexander (CPCo) to J. B. Toskey (CPCo), Subject: Big Rock Point Plant, Generic Letter 88-11, Pressure-Temperature Limits

6. June 12, 1978, Letter from W. S. Skibitsky (CPCo) to D. L. Ziemann (USNRC), Subject: Big Rock Plant--Reactor Surveillance Program

7. C. Z. Serpan and H. E. Watson, "Mechanical Property and Neutron Spectral Analyses of the Big Rock Point Reactor Pressure Vessel," Nuclear Engineering and Design, Volume II, No. 3, April 1970

8. S. E. Yanichko et al, Analysis of Capsule 125 from the Consumers Power Company Big Rock Point Nuclear Plant Reactor Vessel Radiation Surveillance Program, WCAP-9794, September 1980

9. August 22, 1990, Letter from J. D. Eddy (CPCo) to USNRC Document Control Desk, Subject: Supplemental Information Regarding Technical Specification Change Request-Reactor Temperature Limits

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