SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO GENERIC LETTER 88-11

WISCONSIN PUBLIC SERVICE CORPORATION

KEWAUNEE NUCLEAR POWER PLANT

DOCKET NO. 50-305

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 Wisconsin Public Service Corporation (the licensee) evaluated the present pressure/temperature (P/T) limits in the Kewaunee Nuclear Power Plant Technical Specifications, Section 3.1. The evaluation was documented in letters from the licensee dated November 1, 1988 and September 19, 1989. The present P/T limits were developed based on the data from actual surveillance capsules and they are valid for 15 effective full power year (EFPY). The P/T limits support 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 Kewaunee 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 15 EFPY was the circumferential weld between the intermediate and lower shells with 0.20% copper (Cu), 0.77% nickel (Ni), and an initial RT_{ndt} of -56°F.

The licensee has removed three surveillance capsules from Kewaunee. The results from capsules V, R, and P were published in Westinghouse report WCAP-8908, WCAP-9878, and WCAP-12020, respectively. All the surveillance capsules contained Charpy impact specimens and tensile specimens made from base metal, weld metal, and HAZ metal.

For the limiting beltline material, weld 1P3571/1092, the staff calculated the ART to be 204°F at 1/4T (T = reactor vessel beltline thickness) and 162.6°F for 3/4T at 15 EFPY. The staff used a neutron fluence of 1.1E19 n/cm^2 at 1/4T and 0.496E19 n/cm^2 at 3/4T. The ART was determined by the least squares extrapolation method using the surveillance data from capsules, V, R, and P. The least squares method is described in Section 2.1 of RG 1.99, Rev. 2.

The licensee used the method in RG 1.99, Rev. 2, to calculate an ART of 226°F at 15 EFPY at 1/4T for the same limiting weld metal. The licensee used the surveillance data from capsules V and R. The licensee did not use results from Capsule P because Capsule P was withdrawn after the current P/T limits were developed. The staff judges that the licensee's ART of 226°F is acceptable because it is more conservative than the staff's ART of 204°F. Substituting the ART of 226°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 60°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. Using the method in RG 1.99, Rev. 2, the predicted Charpy USE of the weld metal at the end of life will be 65.5 ft-lb. This is greater than 50 ft-lb and, therefore, is acceptable.

3.0 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 15 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 current P/T limits may be maintained in the Kewaunee Technical Specifications.

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. Kewaunee Technical Specifications, Section 3.1.

4. Kewaunee Final Safety Analysis Report, Section 4.

5. October 29, 1989, Letter from R. C. Jones (USNRC) to J. Hannon (USNRC), Subject: Kewaunee, Pressure Vessel Fast Neutron Exposure Evaluation for Pressured Thermal Shock (TAC No. 73332).

6. S. E. Yanichko et al, "Analysis of Capsule V from the Wisconsin Public Service Corporation Kewaunee Nuclear Plant Reactor Vessel Radiation Surveillance Program, WCAP-8908," Westinghouse Electric Corporation, January 1977.

7. S. E. Yanichko et al, "Analysis of Capsule R from the Wisconsin Public Service Corporation Kewaunee Nuclear Plant Reactor Vessel Radiation Surveillance Program, WCAP-9878," Westinghouse Electric Corporation, March 1981.

8. S. E. Yanichko et al, "Analysis of Capsule P from the Wisconsin Public Service Corporation Kewaunee Nuclear Plant Reactor Vessel Radiation Surveillance Program, WCAP-12020," Westinghouse Electric Corporation, November 1988.

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