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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

SUPPORTING AMENDMENT NO. 114 TO FACILITY OPERATING LICENSE NO. DPR-36

MAINE YANKEE ATOMIC POWER COMPANY

MAINE YANKEE ATOMIC POWER STATION

DOCKET NO. 50-309

INTRODUCTION

In response to Generic Letter 88-11, "NRC Position on Radiation Embrittlement of Reactor Vessel Materials and its Effect on Plant Operations," the Maine Yankee Atomic Power Company (the licensee) requested permission to revise the pressure/temperature (P/T) limits in the Maine Yankee Atomic Power Plant Technical Specifications, Section 3.4. The request was documented in a letter from the licensee dated December 2, 1988. 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 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); 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. 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 Regulatory Guide 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.

2.0 EVALUATION

The staff has evaluated the effect of neutron irradiation embrittlement on each beltline material in the Maine Yankee reactor vessel. The amount of neutron irradiation embrittlement was calculated in accordance with RG 1.99, Rev. 2. The staff has determined that the material with the highest ART (most embrittled) at end of life (EOL) was intermediate-to-lower-shell girth weld 9-203 with 0.31% copper (Cu) and 0.74% nickel (Ni). The licensee reported that weld 9-203 had an initial RT_{ndt} of -30°F in tests run at Maine Yankee for the surveillance weld, but used a generic initial RT_{ndt} of -56°F obtained for submerged arc welds using Linde 1092 weld flux.

The licensee has removed three surveillance capsules from Maine Yankee. The results from capsule W-263 were published in Battelle-Columbus Report BCL-585-21, those from capsule A-25 were published in Effects Technology Report 75-317, and those from capsule A-35 in Westinghouse Report WCAP-9875. All surveillance capsules contained Charpy impact specimens and tensile specimens made from base metal, weld metal, and HAZ metal.

For the limiting beltline material, girth weld 9-203, the staff calculated the ART to be 196° F at 1/4 T (T=reactor vessel beltline thickness) for EOL. The neutron fluence used at 1/4T was 8.76E18 n/cm². The ART was determined by the least squares extrapolation method using the data from the three Maine Yankee surveillance capsules. 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 the same ART of 196°F for the limiting weld metal (9-203) in the beltline of the Maine Yankee reactor vessel. Substituting the ART value of 196°F into the 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 imposed 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 25°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 the end of life be above 50 ft-lb. Based on data from an accelerated surveillance capsule withdrawn at 4.56 EFPY, the measured Charpy USE is 50 ft-lb for the limiting girth weld (9-203). This is a 53.3% reduction from the unirradiated value of 107 ft-lb. The neutron fluence of this capsule was 8.84E19, which is far higher than the EOL fluence of 1.47E19. Therefore, the USE of the Maine Yankee beltline materials satisfy Section IV.B of Appendix G.

3.0 ENVIRONMENTAL CONSIDERATION

Notice of Consideration by the staff of issuance of the proposed amendment was published in the Federal Register on January 23, 1989 (54 FR 3167) and no comments or requests for hearing were received. The Commission also consulted with the State of Maine and no comments were received. An Environmental Assessment (EA) and Finding of No Significant Impact was published in the Federal Register on November 13, 1989 (54 FR 47277). Based upon the EA, the staff has determined not to prepare an environmental impact statement for the proposed license amendment, and has concluded that the proposed action will not have a significant adverse effect on the quality of the human environment.

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 for cumulative thermal generation no greater than 5.414E8 MWH(t), which is equivalent to 22.9 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 Maine Yankee Technical Specifications.

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. "Results of an Assessment of Reactor Pressure Vessel Beltline Materials Required by 10 CFR 50.61 (Pressurized Thermal Shock) for the Maine Yankee Atomic Power Plant," Maine Yankee Atomic Power Company, January 23, 1986 (accession number 8601280169)

4. Maine Yankee Final Safety Analysis Report, Section 4.3, "Component and System Design and Operation"

5. J. W. Sheckhard and R. A. Wullaert, "Evaluation of the First Maine Yankee Accelerated Surveillance Capsule," CR 75-317, Effects Technology, Inc., Santa Barbara, CA, August 15, 1975

6. J. S. Perrin et al, "Maine Yankee Nuclear Plant Reactor Pressure Vessel Surveillance Program: Capsule 263," BCL-585-21, Battelle-Columbus Laboratories, Columbus, OH, December 2, 1980

7. S. E. Yanichko et al, "Analysis of the Maine Yankee Reactor Vessel Second Accelerated Surveillance Capsule (WCAP-9875)," Westinghouse Electric Corporation, Pittsburgh, PA, March 1981

8. J. W. Sheckherd and R. A. Wullaert, "Unirradiated Mechanical Properties of Maine Yankee Nuclear Pressure Vessel Materials," Effects Technology, Inc., Santa Barbara, CA, February 1, 1975.

9. J. B. Randazza letter to USNRC, December 2, 1988 (accession number 8812150092)

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