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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION RELATED TO MATERIAL PROPERTIES AND FAST NEUTRON FLUENCE FOR FRACTURE TOUGHNESS REQUIREMENTS FOR PROTECTION AGAINST PRESSURIZED THERMAL SHOCK EVENTS (10 CFR 50.61) MAINE YANKEE ATOMIC POWER COMPANY MAINE YANKEE ATOMIC POWER STATION DOCKET NO. 50-309

1.0 INTRODUCTION

By letters dated April 22, 1983 and January 21, 1986, the Maine Yankee Atomic Power Company, licensee for the Maine Yankee Atomic Power Station, submitted information on the material properties and the fast neutron fluence ($E > 1.0$ MeV) of the reactor pressure vessel in compliance with the requirements of 10 CFR 50.61.

2.0 EVALUATION

2.1 The following evaluation concerns the estimation of the fluence to the pressure vessel to the end of the current operating license on October 21, 2008 and the corresponding value of the RT_{PTS}.

The calculation on which the Maine Yankee Atomic Power Company based its January 21, 1986 submittal was performed by Westinghouse in connection with an earlier pressurized thermal shock (PTS) submittal which was submitted by letter dated April 22, 1983. The calculation was performed with the DOT code using pinwise source distribution but it utilized a P_1 scattering approximation and a cross section set which was not based on ENDF/B. However, Maine Yankee surveillance capsule 263 which was included in that calculation showed a measurement to calculation ratio of 1.51. The measurement of the capsule was performed by Battelle Columbus Laboratories (Reference 1). Because of the wide discrepancy between measurement and calculation, the capsule analysis was repeated by Hanford Engineering Development Laboratory (HEDL) and Brookhaven National Laboratory (BNL). These analyses established that the original discrepancy was due to the fact that the in-capsule water was omitted, affecting the calculated flux at the dosimeters and the neutron spectrum (References 2 and 3). The estimated measurement to calculation flux uncertainty was about 12% (Reference 3). The licensee normalized the fluence estimate on the original capsule measurement which is conservative. Therefore, with calculations independent of the original licensee sponsored calculation, it was established that the capsule 263 results were within the calculational uncertainty limits and normalized toward the upper limit.

In view of:

(a) the Pressure-Temperature updating requirements for the fracture toughness of the beltline material in 10 CFR 50 Appendix G, and

(b) the fact that the RT_{PTS} value is readily available from the calculations of the Pressure Temperature limits, and

(c) the staff desire to be informed on the current value of the RT_{PTS} for all PWRs

the licensee should submit a re-evaluation of the RT_{PTS} and a comparison to the prediction of that submitted in the January 21, 1986 letter along with the future Pressure-Temperature operating limits which are required by 10 CFR 50 Appendix G*. It should be noted that this re-evaluation is a requirement by 10 CFR 50.61, whenever core loadings, surveillance measurements, or other information indicate a significant change in projected values.

The licensee projections were then based on acceptable results. The low leakage loading for the Cycle 9 fuel load and subsequent cycle fuel loads were used to estimate the fluence to the end of the current operating license.

The equation specified in 10 CFR 50.61, as applicable for the Maine Yankee plant is:

$$RT_{PTS} = I+M+(-10+470xCu+350xCuxNi)xf^{0.27}$$

where:

I = Initial RT_{NDT}		= -56°F
M = Uncertainty Margin		= 59°F
Cu = w/o Cooper in circumferential weld 9-203		= 0.31
Ni = w/o Nickel in circumferential weld 9-203		= 0.74
f = peak azimuthal fluence to the end of license (E > 1.0 MeV) lower shell circumferential weld 9-203 in units of 10^{19} n/cm²		= 1.47

Therefore:

$$RT_{PTS} = -56+59+(-10+470x0.31+350x0.31x0.74)x1.47^{0.27}$$
$$= 3+216x1.109 = 242.7°F$$

which is lower than 300°F, the applicable PTS rule screening criteria and is acceptable.

2.2 The following evaluation concerns material properties for fracture toughness requirements for protection against thermal shock events.

*This request for information is covered under OMB Clearance No. 3150-0011.

The controlling beltline material from the standpoint of PTS susceptibility was identified to be the intermediate to lower shell circumferential weld 9-203, weld wire heat number IP3571.

The material properties of the controlling material and the associated margin and chemistry factor were reported to be:

	Utility Submittal	Staff Evaluation
Cu (copper content, %)	0.31	0.31
Ni (nickel content, %)	0.74	0.74
I (initial RT_{NDT}, °F)	-56	-56
M (Margin, °F)	59	59
CF (Chemistry Factor, °F)		216.0

3.0 CONCLUSIONS

The licensee projections concerning the estimation of the fluence to the pressure vessel to the end of the current operating license on October 21, 2008 were based on acceptable results and the corresponding value of RT_{PTS} (242.7°F) is lower than 300°F, the applicable PTS rule screening criteria, and is acceptable.

The controlling material has been properly identified. The justifications given for the copper and nickel contents and the initial RT_{NDT} are acceptable. The margin has been derived from consideration of the bases for these values, following the PTS Rule, Section 50.61 of 10 CFR Part 50. Assuming that the reported values of fluence are correct, Equation 1 of the PTS rule governs, and the chemistry factor is as shown above.

References

1. Perrin, J. S. et. al "Maine Yankee Nuclear Plant Reactor Pressure Vessel Surveillance Program: Capsule 263" BCL-585-21, dated December 12, 1980.

2. McElroy W. N. et. al "LWR Pressure Vessel Dosimetry Program; LWR Power Reactor Surveillance Physics Dosimetry Compedium" Hanford Engineering Development Laboratory, NUREG/CR-3319 dated August 1985.

3. Carew J. F., et. al "Pressure Vessel Fluence Benchmark Calculations" BNL-NUREG-34715, dated February 1984.

Date:

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