3.1.2 Heatup and Cooldown Rates (Continued)

the plate has been 100% volumetrically inspected by ultrasonic test using both longitudinal and shear wave methods. The remaining material in the reactor vessel, and other primary coolant system components, meets the appropriate design code requirements and specific component function and has a maximum NDTT of +40°F.[5]

As a result of fast neutron irradiation in this region of the core, there will be an increase in the RT with operation. The techniques used to predict the integrated fast neutron ($E > 1$ MeV) fluxes of the reactor vessel are described in Section 3.3.2.6 of the FSAR and also in Amendment 13, Section II, to the FSAR.

Since the neutron spectra and the flux measured at the samples and reactor vessel inside radius should be nearly identical, the measured transition shift from a sample can be applied to the adjacent section of the reactor vessel for later stages in plant life equivalent to the difference in calculated flux magnitude. The maximum exposure of the reactor vessel will be obtained from the measured sample exposure by application of the calculated azimuthal neutron flux variation. The predicted RT_{NDT} shift for the base metal has been predicted based upon surveillance data and the US NRC Regulatory Guide.[10] To compensate for any increase in the RT caused by irradiation, limits on the pressure-temperature relationship are periodically changed to stay within the stress limits during heatup and cooldown.

Reference 7 provides a procedure for obtaining the allowable loadings for ferritic pressure-retaining materials in Class 1 components. This procedure is based on the principles of linear elastic fracture mechanics and involves a stress intensity factor prediction which is a lower bound of static, dynamic and crack arrest critical values. The stress intensity factor computed[7] is a function of RT_{NDT}, operating temperature, and vessel wall temperature gradients.

Pressure-temperature limit calculational procedures for the reactor coolant pressure boundary are defined in Reference 8 based upon Reference 7. The limit lines of Figures 3-1 through 3-3 consider a 54 psi pressure allowance to account for the fact that pressure is measured in the pressurizer rather than at the vessel beltline. In addition, for calculational purposes, 5°F and 30 psi were taken as measurement error allowances for temperature and pressure, respectively. By Reference 7, reactor vessel wall locations at 1/4 and 3/4 thickness are limiting. It is at these locations that the crack propagation associated with the hypothetical flaw must be arrested. At these locations, fluence attenuation and thermal gradients have been

3.1.2 Heatup and Cooldown Rates (Continued)

Basis (Continued)

The criticality temperature is determined per Reference 8 and the core operational curves adhere to the requirements of Reference 9. The inservice test curves incorporate allowances for the thermal gradients associated with the heatup curve used to attain inservice test pressure. These curves differ from heatup curves only with respect to margin for primary membrane stress.[7] Due to the shifts in RT_{NDT}, NDTT requirements associated with nonreactor vessel materials are, for all practical purposes, no longer limiting.

References

(1) FSAR, Section 4.2.2.
(2) ASME Boiler and Pressure Vessel Code, Section III, A-2000.
(3) Battelle Columbus Laboratories Report, "Palisades Pressure Vessel Irradiation Capsule Program: Unirradiated Mechanical Properties," August 25, 1977.
(4) Battelle Columbus Laboratories Report, "Palisades Nuclear Plant Reactor Vessel Surveillance Program: Capsule A-240," March 13, 1979, submitted to the NRC by Consumers Power Company letter dated July 2, 1979.
(5) FSAR, Section 4.2.4.
(6) US Nuclear Regulatory Commission, Regulator Guide 1.99, "Effects of Residual Elements on Predicted Radiation Damage to Reactor Vessel Materials," July, 1975.
(7) ASME Boiler and Pressure Vessel Code, Section III, Appendix G, "Protection Against Non-Ductile Failure," 1974 Edition.
(8) US Atomic Energy Commission Standard Review Plan, Directorate of Licensing, Section 5.3.2, "Pressure-Temperature Limits."
(9) 10 CFR Part 50, Appendix G, "Fracture Toughness Requirements," May 31, 1983.
(10) US Nuclear Regulatory Commission, Regulatory Guide 1.99, Draft Revision 2, April, 1984.
(11) Combustion Engineering Report CEN-189, December, 1981.
(12) "Analysis of Capsules T-330 and W-290 from the Consumers Power Company Palisades Reactor Vessel Radiation Surveillance Program," WCAP-10637, September, 1984.
(13) EA-PAL-85-101 "Calculation of PCS Pressure Increase from Adding 133 gpm (3 charging pumps) Before the PORVs Open," November 4, 1987.
(14) EA-PAL-LTOP-880119 - "Calculation of Required PORV Capacity to Maintain the PCS Below Appendix G," January 19, 1988.
(15) EA-PAL-LTOP-880120 Rev. A - PORV Flow Capacity at Expected LTOP Conditions" February 15, 1988.
(16) EA-PAL-LTOP-880121 - "Calculation of Time for Operator to Act for HPSI and Bubble" - January 20, 1988.
(17) EA-ESSR 88727-C/S-01 "Palisades Plant Primary Coolant System Pressure Temperature Limits Per Appendix G of the ASME Boiler and Pressure Vessel Code" Revision 0.