Basis

All components in the Reactor Coolant System are designed to withstand the effects of cyclic loads due to reactor system temperature and pressure changes.[1] These cyclic loads are introduced by normal unit load transients, reactor trips, and startup and shutdown operation. The number of thermal and loading cycles used for design purposes are shown in Table 4.1-8 of the FSAR. During unit startup and shutdown, the rates of temperature and pressure changes are limited. The maximum plant heatup and cooldown rate of 100°F per hour is consistent with the design number of cycles and satisfies stress limits for cyclic operation.

The ASME Code, Section III, Nonmandatory Appendix G, contains procedures for the development of heatup and cooldown curves for protection against nonductile failure. The ASME Code requires that a 1/4 wall thickness flaw, either on the inside or outside, depending upon the location of concern, be assumed to exist in the structure. As the Code of Federal Regulations, Title 10, Chapter 50, Appendix G, invokes the ASME Code, Appendix G, the ASME Code procedures are utilized in developing the heatup and cooldown limitation curves.

During heatup, the thermal gradients in the reactor vessel wall produce thermal stresses which vary from compressive at the inner wall to tensile at the outer wall. These thermal-induced compressive stresses tend to alleviate the tensile stresses induced by the internal pressure. Therefore, a pressure-temperature curve based on steady-state conditions (i.e., no thermal stresses) represents a lower bound of all similar curves for finite heatup rates when the inner wall of the vessel is treated as the governing location.

The heatup analysis also covers the determination of pressure-temperature limitations for the case in which the outer wall of the vessel becomes the controlling location. The thermal gradients established during heatup

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neutron exposure of the vessel is computed to be 3.5×10^{19} neutrons/cm^2 for forty years of operation at 1518 MWt and 80 percent laod factor.[2] This is the exposure expected at the inner reactor vessel wall. However, the neutron fluence used to predict the ΔRT_{NDT} shift is the one-quarter shell thickness neutron exposure. The relationship between fluence at the vessel ID wall and the fluence at the one-quarter and three-quarter shell thickness locations has been calculated and is presented in References 3 and 4 as a function of Effective Full Power Years. These curves are used to determine the fluence at the location of interest when the heatup and cooldown curves are to be revised.

Once the fluence is determined, the temperature shift used in revising the heatup and cooldown curves is obtained from the temperature versus fluence curves (the 0.25% copper base, 0.20% weld line for Unit 1 and the 0.30% copper base, 0.25% weld line for Unit 2) also contained in References 3 and 4. These curves are used because they are based upon a substantial amount of experimental data and represent the results of the chemical analysis of the weld metal in the reactor vessels.

The heatup and cooldown curves presented in Figure 15.3.1-1 and 15.3.1-2 (Unit 1) and 15.3.1-3 and 15.3.1-4 (Unit 2) were calculated based on the above information and the methods of ASME Code Section III (1974 Edition), Appendix G, "Protection Against Nonductile Failure", and are applicable up to the operational exposure indicated on the figures. Corrections for possible instrumentation inaccuracies have been incorporated into these curves. The temperature correction is made by adding the temperature error (24°F) to the required temperature and the pressure correction is made by subtracting the pressure error (64 psi) from the required pressure. These corrections adjust the curves in the conservative direction.

The actual temperature shift of the vessel material will be established periodically during operation by removing and evaluating reactor vessel material irradiation surveillance specimens installed near the inside wall of the reactor vessel in the core area. Since the neutron spectra at the irradiation samples and vessel inside radius are identified by a specified lead factor, the measured temperature shift for a sample is an excellent indicator of the effects of power operation on the adjacent section of the reactor vessel. If the experimental temperature shift (at the 30 ft-lb level) does not substantiate the predicted shift, new prediction curves and heatup and cooldown curves must be developed.

The pressure-temperature limit lines shown on Figures 15.3.1-1 (Unit 1) and 15.3.1-3 (Unit 2) for reactor criticality and for inservice leak and hydrostatic testing have been provided to assure compliance with the minimum temperature requirement of Appendix G to 10 CFR 50 for reactor criticality and for inservice leak and hydrostatic testing.

The spray should not be used if the temperature difference between the pressurizer and spray fluid is greater than 320°F. This limit is imposed to maintain the thermal streses at the pressurizer spray line nozzle below the design limit.

The temperature requirements for the steam generator correspond with the measured NDT for the shell.

The reactor vessel materials surveillance capsule removal schedules are presented in Table 15.3.1-1 for Unit 1 and Table 15.3.1-3 for Unit 2. These schedules have been developed based upon the requirements of the Code of Federal Regulations, Title 10, Chapter 50, Appendix H, and with consideration of ASTM Standard E-185-82. When the capsule lead factors are considered, the

scheduled removal dates will provide materials data representative of about 10%, 20%, 50%, 90% and 110% of the actual reactor vessel exposure anticipated during the vessel life.

References

(1) FSAR, Section 4.1.5

(2) Westinghouse Electric Corporation, WCAP-10638

(3) Westinghouse Electric Corporation, WCAP-8743

(4) Westinghouse Electric Corporation, WCAP-8738

TABLE 15.3.1-1

POINT BEACH NUCLEAR PLANT, UNIT NO. 1
REACTOR VESSEL SURVEILLANCE CAPSULE REMOVAL SCHEDULE

Capsule Letter	Approximate Removal Date
V	September 1972 (actual)
S	December 1975 (actual)
R	October 1977 (actual)
T	March 1984 (actual)
P	Spring 1994
N	Standby

*The actual removal dates will be adjusted to coincide with the closest scheduled plant refueling outage or major reactor plant shutdown.

TABLE 15.3.1-2

POINT BEACH NUCLEAR PLANT, UNIT NO. 2
REACTOR VESSEL SURVEILLANCE CAPSULE REMOVAL SCHEDULE

Capsule Letter	Approximate Removal Date*
V	November 1974 (actual)
T	March 1977 (actual)
R	April 1979 (actual)
P	Fall 1989
S	Fall 1995
N	Standby

*The actual removal dates will be adjusted to coincide with the closest scheduled plant refueling outage or major reactor plant shutdown.