

Heatup and cooldown limit curves are calculated using the most limiting value of the nil-ductility reference temperature, RT_{NDT} , at the end of 47.6 Effective Full Power Years (EFPY) and 48.1 EFPY for Units 1 and 2, respectively. The most limiting value of RT_{NDT} (238.2°F) occurs at the 1/4-T, 0° azimuthal location in the Unit 1 intermediate-to-lower shell circumferential weld. The limiting RT_{NDT} at the 1/4-T location in the core region is greater than the RT_{NDT} of the limiting unirradiated material. This ensures that all components in the Reactor Coolant System will be operated conservatively in accordance with applicable Code requirements.

The reactor vessel materials have been tested to determine their initial RT_{NDT} ; the results are presented in UFSAR Section 4.1. Reactor operation and resultant fast neutron (E greater than 1 MEV) irradiation can cause an increase in the RT_{NDT} . Therefore, an adjusted reference temperature, based upon the copper and nickel content of the material and the fluence was calculated in accordance with the recommendations of Regulatory Guide 1.99, Revision 2 "Effects of Residual Elements on Predicted Radiation Damage to Reactor Vessel Materials." The heatup and cooldown limit curves of Figures 3.1-1 and 3.1-2 include predicted adjustments for this shift in RT_{NDT} at the end of 47.6 EFPY and 48.1 EFPY for Units 1 and 2, respectively (as well as adjustments for location of the pressure sensing instrument).

Surveillance capsules will be removed in accordance with the requirements of ASTM E185-82 and 10 CFR 50, Appendix H. The surveillance specimen withdrawal schedule is shown in the UFSAR. The heatup and cooldown curves must be recalculated when the ΔRT_{NDT} determined from the surveillance capsule exceeds the calculated ΔRT_{NDT} for the equivalent capsule radiation exposure, or when the service period exceeds 47.6 EFPY or 48.1 EFPY for Units 1 and 2, respectively, prior to a scheduled refueling outage.

Allowable pressure-temperature relationships for various heatup and cooldown rates are calculated using methods derived from Appendix G in Section III of the ASME Boiler and Pressure Vessel Code as required by Appendix G to 10 CFR Part 50.

The general method for calculating heatup and cooldown limit curves is based upon the principles of the linear elastic fracture mechanics (LEFM) technology. In the calculation procedures a semi-elliptical surface defect with a depth of one-quarter of the wall thickness, T , and a length of one and one half T is assumed to exist at the inside of the vessel wall as well as at the outside of the vessel wall. The dimensions of this postulated crack, referred to in Appendix G of ASME Section III as the reference flaw, amply exceed the current capabilities of inservice inspection techniques. Therefore, the reactor operation limit curves developed for this reference crack are conservative and provide sufficient safety margins for protection against non-ductile failure. To assure that the radiation embrittlement effects are accounted for in the calculation of the limit curves, the most limiting value of the nil ductility reference temperature, RT_{NDT} , is used and this includes the radiation-induced shift, ΔRT_{NDT} , corresponding to the end of the period for which heatup and cooldown curves are generated.

The ASME approach for calculating the allowable limit curves for various heatup and cooldown rates specifies that the total stress intensity factor, K_I , for the combined thermal and pressure stresses at any time during heatup or cooldown cannot be greater than the reference stress intensity factor, K_{IC} , for the metal temperature at that time. K_{IC} is obtained from the reference fracture toughness curve, defined in Section XI to the ASME Code. The K_{IC} curve is given by the equation:

$$K_{IC} = 33.2 + 20.734 \exp [0.02(T - RT_{NDT})] \quad (1)$$

where K_{IC} is the reference stress intensity factor as a function of the metal temperature T and the metal nil ductility reference temperature RT_{NDT} . Thus, the governing equation for the heatup-cooldown analysis is defined in Appendix G of the ASME Code as follows:

$$C K_{IM} + K_{II} \leq K_{IC} \quad (2)$$

where, K_{IM} is the stress intensity factor caused by membrane (pressure) stress.

K_{It} is the stress intensity factor caused by the thermal gradients

K_{IC} is provided by the code as a function of temperature relative to the RT_{NDT} of the material.

$C = 2.0$ for level A and B service limits, and

$C = 1.5$ for inservice hydrostatic and leak test operations.

At any time during the heatup or cooldown transient, K_{IC} is determined by the metal temperature at the tip of the postulated flaw, the appropriate value for RT_{NDT} , and the reference fracture toughness curve. The thermal stresses resulting from temperature gradients through the vessel wall are calculated and then the corresponding thermal stress intensity factor, K_{It} , for the reference flaw is computed. From Equation (2) the pressure stress intensity factors are obtained and, from these, the allowable pressures are calculated.

The heatup limit curve, Figure 3.1-1, is a composite curve which was prepared by determining the most conservative case, with either the inside or outside wall controlling, for any heatup rate up to 60°F per hour. The cooldown limit curves of Figure 3.1-2 are composite curves which were prepared based upon the same type analysis with the exception that the controlling location is always the inside wall where the cooldown thermal gradients tend to produce tensile stresses while producing compressive stresses at the outside wall. The cooldown limit curves are valid for cooldown rates up to 100°F/hr. The heatup and cooldown curves were prepared based upon the most limiting value of the predicted adjusted reference temperature at the end of 47.6 EFPY and 48.1 EFPY for Units 1 and 2, respectively. The adjusted reference temperature was calculated using materials properties data from the B&W Owners Group Master Integrated Reactor Vessel Surveillance Program (MIRVSP) documented in the most recent revision to BAW-1543 and reactor vessel neutron fluence data obtained from plant-specific analyses.

- (3) During the initial 72 hours, maintain a bubble in the pressurizer with a maximum narrow range level of 33%,

or
- (4) Maintain two Power Operated Relief Valves (PORV) OPERABLE with a lift setting of ≤ 395 psig and verify each PORV block valve is open at least once per 72 hours,

or
- (5) The RCS shall be vented through one open PORV or an equivalent size opening as specified below:
 - (a) with the RCS vented through an unlocked open vent path, verify the path is open at least once per 12 hours, or
 - (b) with the RCS vented through a locked open vent path verify the path is open at least once per 31 days.

2. The requirements of Specification 3.1.G.1.c.(4) may be modified as follows:

- a One PORV may be inoperable in INTERMEDIATE SHUTDOWN with the RCS average temperature $> 200^{\circ}\text{F}$ but $< 350^{\circ}\text{F}$ for a period not to exceed 7 days. If the inoperable PORV is not restored to OPERABLE status within 7 days, then completely depressurize the RCS and vent through one open PORV or an equivalent size opening within the next 8 hours.
- b One PORV may be inoperable in COLD SHUTDOWN or REFUELING SHUTDOWN with the reactor vessel head bolted for a period not to exceed 24 hours. If the inoperable PORV is not restored to OPERABLE status within 24 hours then completely depressurize the RCS and vent through one open PORV or an equivalent size opening within 8 hours.

**Surry Units 1 and 2
Reactor Coolant System Cooldown Limitations**

Material Property Basis

Limiting ART:

1/4-T, 238.2 deg. F

3/4-T, 183.9 deg. F

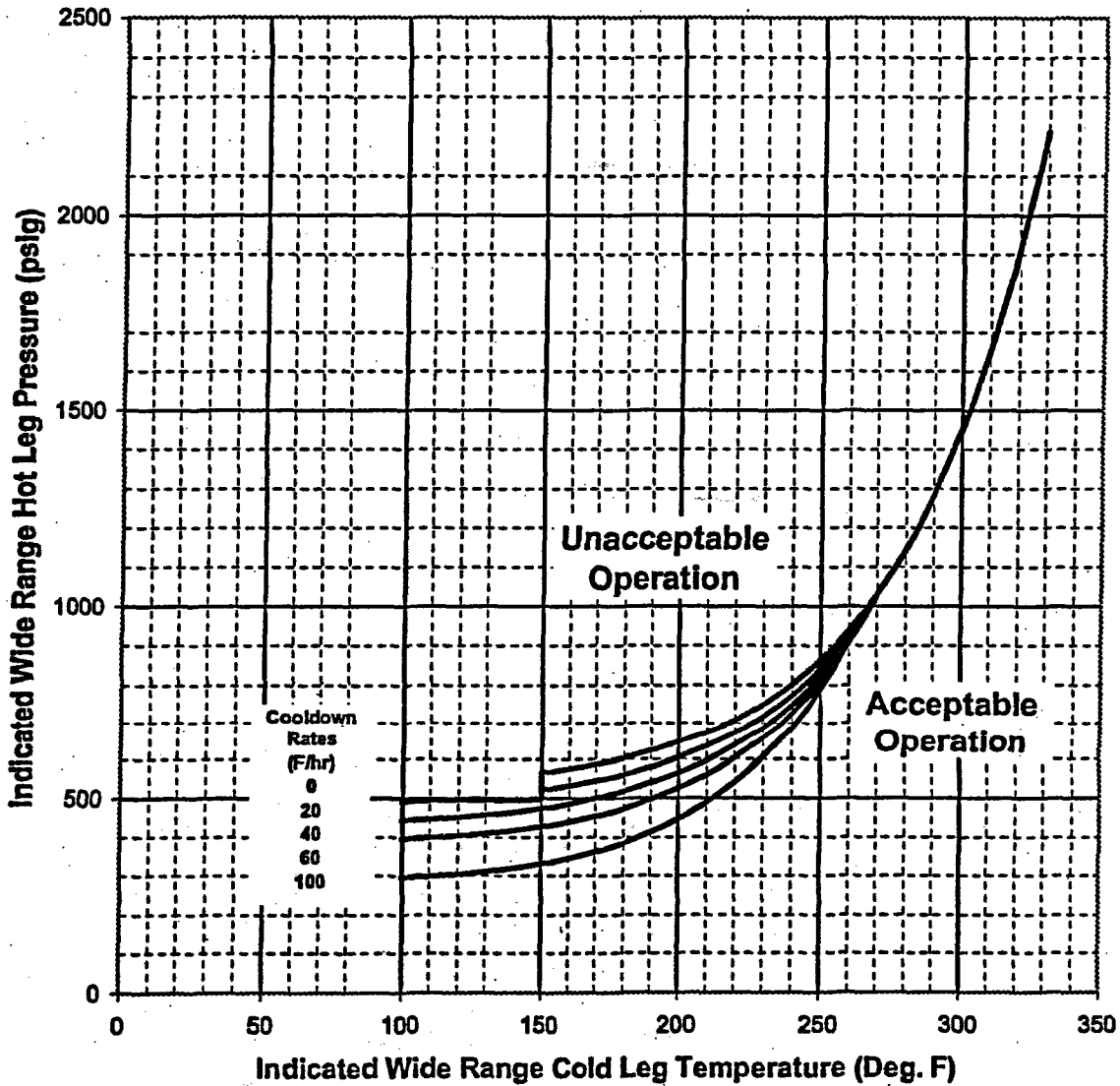


Figure 3.1-2 : Surry Units 1 and 2 Reactor Coolant System Cooldown Limitations (Cooldown Rates up to 100°F/hr) Applicable for the first 47.6 EFPY for Unit 1 and 48.1 EFPY for Unit 2 (Including Margins for Instrumentation Errors)