3/4.2.5 DNB PARAMETERS (Continued)

initial FSAR assumptions and have been analytically demonstrated adequate to maintain a minimum DNBR of greater than or equal to the design limit throughout each analyzed transient. The T_{avg} value of 598°F and the pressurizer pressure value of 2189 psig are analytical values. The readings from four channels will be averaged and then adjusted to account for measurement uncertainties before comparing with the required limit. The flow requirement (392,300 gpm) includes a measurement uncertainty of 2.8%.

The 12-hour periodic surveillance of these parameters through instrument readout is sufficient to ensure that the parameters are restored within their limits following load changes and other expected transient operation.

PRESSURE TEMPERATURE

can cause an increase in the F
based upon the fluence, coppe
computed using the method de:
Elements on Predicted Radiatio
limit curves of Figures 3.4-2 anc
end of 32 EFPY as well as adju
sensing instruments.

Values of ΔRT_{NDT} determ
surveillance program, evaluated a
removed in accordance with the r
and new values of ΔRT_{NDT} will be
Revision 2, "Radiation Embrittlem
surveillance specimens can be us
material by using the lead factor a
cooldown curves must be recalcul
exceeds the calculated ΔRT_{NDT} for

Allowable pressure-tempera
calculated using methods derived fr
Vessel Code as required by Appenc
detail in WCAP-7924-A.

The general method for calc
principles of the linear elastic fractur
procedures a semielliptical surface d
a length of 3/2T is assumed to exist
vessel wall. The dimensions of this
III as the reference flaw, amply excee
Therefore, the reactor operation limit
and provide sufficient safety margins
radiation embrittlement effects are acc
limiting value of the nil-ductility referen
radiation-induced shift, ΔRT_{NDT}, corresp
cooldown curves are generated.

The ASME approach for calcula
cooldown rates specifies that the total s
pressure stresses at any time during he
stress intensity factor, K_{IR}, for the metal
reference

PRESSURE TEMPERATURE LIMITS (Continued)

a. Allowable combinations of pressure and temperature for spec
change rates are below and to the right of the limit lines sh
cooldown rates between those presented may be obtained by

b. Figures 3.4-2 and 3.4-3 define limits to assure prevention of
For normal operation, other inherent plant characteristics, e.g
and pressurizer heater capacity, may limit the heatup and co
be achieved over certain pressure-temperature ranges.

2. These limit lines shall be calculated periodically using methods pro

3. The second ry side of the steam generator must not be pressuriz
temperature of the steam generator is below 70°F,

4 The pressurizer heatup and cooldown rates shall not exceed 100°F
respectively. The spray shall not be used if the temperature differ
pressurizer and the spray fluid is greater than 621°F, and

5. System preservice hydrotests and inservice leak and hydrotests
pressures in accordance with the requirements of ASME Boiler ar
Section XI.

The fracture toughness properties of the ferritic materials in the r
determined in accordance with the NRC Standard Review Plan, ASTM E
accordance with additional reactor vessel requirements. These propertie
accordance with Appendix G of the 1976 Summer Addenda to Section I
Pressure Vessel Code and the calculation methods described in WCAP
and Cooldown Limit Curves," April 1975.

Heatup and cooldown limit curves are calculated using the mos
nil-ductility reference temperature, RT_{NDT}, at the end of 32 effective full
service life. The 32 EFPY service life period is chosen such that the l
location in the core region is greater than the RT_{NDT} of the limiting unir
selection of such a limiting RT_{NDT} assures that all components in the R
be operated conservatively in accordance with applicable Code require

The reactor vessel materials have been tested to determine th
Reactor operation and resultant fast neutron (E greater than 1 MeV)

PRESSURE TEMPERATURE LIMITS (Continued)

a. Allowable combinations of pressure and temperature for specific temperature change rates are below and to the right of the limit lines shown. Limit lines for cooldown rates between those presented may be obtained by interpolation; and

b. Figures 3.4-2 and 3.4-3 define limits to assure prevention of non-ductile failure only. For normal operation, other inherent plant characteristics, e.g., pump heat addition and pressurizer heater capacity, may limit the heatup and cooldown rates that can be achieved over certain pressure-temperature ranges.

2. These limit lines shall be calculated periodically using methods provided below,

3. The secondary side of the steam generator must not be pressurized above 200 psig if the temperature of the steam generator is below 70°F,

4 The pressurizer heatup and cooldown rates shall not exceed 100°F/h and 200°F/h, respectively. The spray shall not be used if the temperature difference between the pressurizer and the spray fluid is greater than 621°F, and

5. System preservice hydrotests and inservice leak and hydrotests shall be performed at pressures in accordance with the requirements of ASME Boiler and Pressure Vessel Code, Section XI.

The fracture toughness properties of the ferritic materials in the reactor vessel are determined in accordance with the NRC Standard Review Plan, ASTM E185-73, and in accordance with additional reactor vessel requirements. These properties are then evaluated in accordance with Appendix G of the 1976 Summer Addenda to Section III of the ASME Boiler and Pressure Vessel Code and the calculation methods described in WCAP-7924-A, "Basis for Heatup and Cooldown Limit Curves," April 1975.

Heatup and cooldown limit curves are calculated using the most limiting value of the nil-ductility reference temperature, RT_{NDT}, at the end of 32 effective full power years (EFPY) of service life. The 32 EFPY service life period is chosen such that the limiting RT_{NDT} at the 1/4T location in the core region is greater than the RT_{NDT} of the limiting unirradiated material. The selection of such a limiting RT_{NDT} assures 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}. Reactor operation and resultant fast neutron (E greater than 1 MeV) irradiation

PRESSURE TEMPERATURE LIMITS (Continued)

can cause an increase in the RT_{NDT}. Therefore, ΔRT_{NDT} and an adjusted reference temperature, based upon the fluence, copper content, and nickel content of the materials in question were computed using the method described in Regulatory Guide 1.99, Revision 1, "Effects of Residual Elements on Predicted Radiation Damage to Reactor Vessel Materials." The heatup and cooldown limit curves of Figures 3.4-2 and 3.4-3 include predicted adjustments for this shift in RT_{NDT} at the end of 32 EFPY as well as adjustments for possible errors in the pressure and temperature sensing instruments.

Values of ΔRT_{NDT} determined in this manner may be used until the results from the material surveillance program, evaluated according to ASTM E185, are available. Capsules will be removed in accordance with the requirements of ASTM E185-73 and 10 CFR Part 50, Appendix H and new values of ΔRT_{NDT} will be computed using the method described in Regulatory Guide 1.99, Revision 2, "Radiation Embrittlement of Reactor Vessel Materials". The results obtained from the surveillance specimens can be used to predict future radiation damage to the reactor vessel material by using the lead factor and the withdrawal time of the capsule. 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.

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, and these methods are discussed in detail in WCAP-7924-A.

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 semielliptical surface defect with a depth of one-quarter of the wall thickness, T, and a length of 3/2T 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 nonductile 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_{IR}, for the metal temperature at that time. K_{IR} is obtained from the reference

TABLE B 3/4.4-1a

(This table number not used)

TABLE B 3/4.4-1b
(This table number not used)

FIGURE B 3/4.4-1
(This figure number not used)

3/4.7.1.2 AUXILIARY FEEDWATER SYSTEM

The OPERABILITY of the Auxiliary Feedwater System ensures that the Reactor Coolant System can be cooled down to less than 350°F from normal operating conditions in the event of a total loss-of-offsite power.

Each auxiliary feedwater pump is capable of delivering a total feedwater flow of 500 gpm at a pressure of 1363 psig to the entrance of the steam generators. This capacity is sufficient to ensure that adequate feedwater flow is available to remove decay heat and reduce the Reactor Coolant System temperature to less than 350°F when the Residual Heat Removal System may be placed into operation. The AFW pumps are tested using the test line back to the AFST and the AFW isolation valves closed to prevent injection of cold water into the steam generators. The STPEGS isolation valves are active valves required to open on an AFW actuation signal. Specification 4.7.1.2.1 requires these valves to be verified in the correct position.

3/4.7.1.3 AUXILIARY FEEDWATER STORAGE TANK (AFST)

The OPERABILITY of the auxiliary feedwater storage tank with the minimum water volume ensures that sufficient water is available to maintain the RCS at HOT STANDBY conditions for 4 hours with steam discharge to the atmosphere concurrent with total loss-of-offsite power and failure of the AFW automatic recirculation control (ARC) valve followed by a cooldown to 350°F at 25°F per hour. The contained water volume limit includes an allowance for water not usable because of tank discharge line location or other physical characteristics.

3/4.7.1.4 SPECIFIC ACTIVITY

The limitations on Secondary Coolant System specific activity ensure that the resultant offsite radiation dose will be limited to a small fraction of 10 CFR Part 100 dose guideline values in the event of a steam line rupture. This dose also includes the effects of a coincident 1 gpm primary-to-secondary tube leak in the steam generator of the affected steam line. These values are consistent with the assumptions used in the safety analyses.

3/4.7.1.5 MAIN STEAM LINE ISOLATION VALVES

The OPERABILITY of the main steam line isolation valves ensures that no more than one steam generator will blow down in the event of a steam line rupture. This restriction is required to: (1) minimize the positive reactivity effects of the Reactor Coolant System cooldown associated with the blowdown, and (2) limit the pressure rise within containment in the event the steam line rupture occurs within containment. The OPERABILITY of the main steam isolation valves within the closure times of the Surveillance Requirements are consistent with the assumptions used in the safety analyses.