3.1.2 PRESSURIZATION HEATUP AND COOLDOWN LIMITATIONS

Applicability

Applies to pressurization, heatup and cooldown of the reactor coolant system.

Objective

To assure that temperature and pressure changes in the reactor coolant system do not cause cyclic loads in excess of design for reactor coolant system components.

Specification

3.1.2.1 For operations until thirty-two (32) effective full power years, the reactor coolant pressure and the system heatup and cooldown rates (with the exception of the pressurizer) shall be limited in accordance with Figure 3.1-1 and Figure 3.1-2 and are as follows:

Heatup/Cooldown

Allowable combinations of pressure and temperature shall be to the right of and below the limit line in Figure 3.1-1. Heatup and cooldown rates shall not exceed those shown on Figure 3.1-1.

Inservice Leak and Hydrostatic Testing

Allowable combinations of pressure and temperature shall be to the right of and below the limit line in Figure 3.1-2. Heatup and cooldown rates shall not exceed those shown on Figure 3.1-2.

3.1.2.2 The secondary side of the steam generator shall not be pressurized above 200 psig if the temperature of the steam generator shell is below 100°F.

3.1.2.3 The pressurizer heatup and cooldown rates shall not exceed 100°F in any one hour. The spray shall not be used if the temperature difference between the pressurizer and the spray fluid is greater than 430°F.

3.1.2.4 Prior to exceeding thirty-two (32) effective full power years of operation, Figures 3.1-1 and 3.1-2 shall be updated for the next service period in accordance with 10 CFR 50, Appendix G. Section V.B. The highest predicted adjusted reference temperature of all the beltline materials shall be used to determine the adjusted reference temperature at the end of the service period. The basis for this prediction shall be submitted for NRC staff review in accordance with Specification 3.1.2.5.

3.1.2.5 The updated proposed technical specifications referred to in 3.1.2.4 shall be submitted for NRC review at least 90 days prior to the end of the service period. Appropriate additional NRC review time shall be allowed for proposed technical specifications submitted in accordance with 10 CFR 50, Appendix G. Section V.C.

BASES

All reactor coolant system components are designed to withstand the effects of cyclic loads due to system temperature and pressure changes (Reference 1). These cyclic loads are introduced by unit load transients, reactor trips and unit heatup and cooldown operations. The number of thermal and loading cycles used for design purposes are shown in Table 4.1-1 of the UFSAR. The maximum unit heatup and cooldown rates satisfy stress limits for cyclic operation (Reference 2). The 200 psig pressure limit for the secondary side of the steam generator at a temperature less than 100°F satisfies stress levels for temperatures below the Nil Ductility Transition Temperature (NDTT).

The heatup and cooldown rate limits in this specification are based on linear heatup and cooldown ramp rates which by analysis have been extended to accommodate 15°F step changes at any time with the appropriate soak (hold) times. Also, an additional 15°F step change has been included in the analysis with no additional soak time to accommodate decay heat initiation at approximately 252°F.

The unirradiated reference nil ductility temperature (RT_{NDT}) for the surveillance region materials were determined in accordance with 10 CFR 50, Appendixes G and H. For other beltline region materials and other reactor coolant pressure boundary materials, the unirradiated impact properties were estimated using the methods described in BAW-10046A, Rev. 2.

As a result of fast neutron irradiation in the beltline region of the core, there will be an increase in the RT_{NDT} with accumulated nuclear operations. The adjusted reference temperatures have been calculated as described in reference No. 6.

The predicted RT_{NDT} was calculated using the respective predicted neutron fluence at thirty-two (32) effective full power years of operation and the procedures defined in Regulatory Guide 1.99, Rev. 2, Section C.1.1 for the plate metals and the provisions of Section C.2.1 for the weld metals. The analysis of the reactor vessel material contained in the second Three Mile Island Nuclear Station Unit 1 surveillance capsule as well as B&WOG surveillance capsules confirmed that the current technique, as described in reference No. 6, for predicting the change in impact properties due to irradiation are conservative.

Analyses of the activation detectors in the TMI-1 surveillance capsules have provided estimates of reactor vessel wall fast neutron fluxes for cycles 1 through 4. Extrapolation of reactor vessel fluxes (average of cycles 8 & 9), and corresponding fluence accumulations, based on predicted future fuel cycle design conditions during thirty-two (32) effective full power years of operation are described in References 5 & 6.

Based on the predicted RT_{NDT} with thirty-two (32) effective full power years of operation, the pressure/temperature limits of Figure 3.1-1 and 3.1-2 have been established in accordance with the requirements of 10 CFR 50, Appendix G. Also, see Reference 4. The methods and criteria employed to establish the operating pressure and temperature limits are as described in BAW-10046A, Rev. 2. The protection against nonductile failure is provided by maintaining the coolant pressure below the upper limits of these pressure temperature limit curves.

The pressure limit lines on Figures 3.1-1 and 3.1-2 have been established considering the following:

a. A 25 psi error in measured pressure.

b. A 12°F error in measured temperature.

c. System pressure is measured in either loop.

d. Maximum differential pressure between the point of system pressure measurement and the limiting reactor vessel region for the allowable operating pump combinations.

The spray temperature difference restriction, based on a stress analysis of spray line nozzle is imposed to maintain the thermal stresses at the pressurizer spray line nozzle below the design limit. Temperature requirements for the steam generator correspond with the measured NDTT for the shell.

REFERENCES

(1) UFSAR, Section 4.1.2.4 - "Cyclic Loads"

(2) ASME Boiler and Pressure Code, Section III, N-415

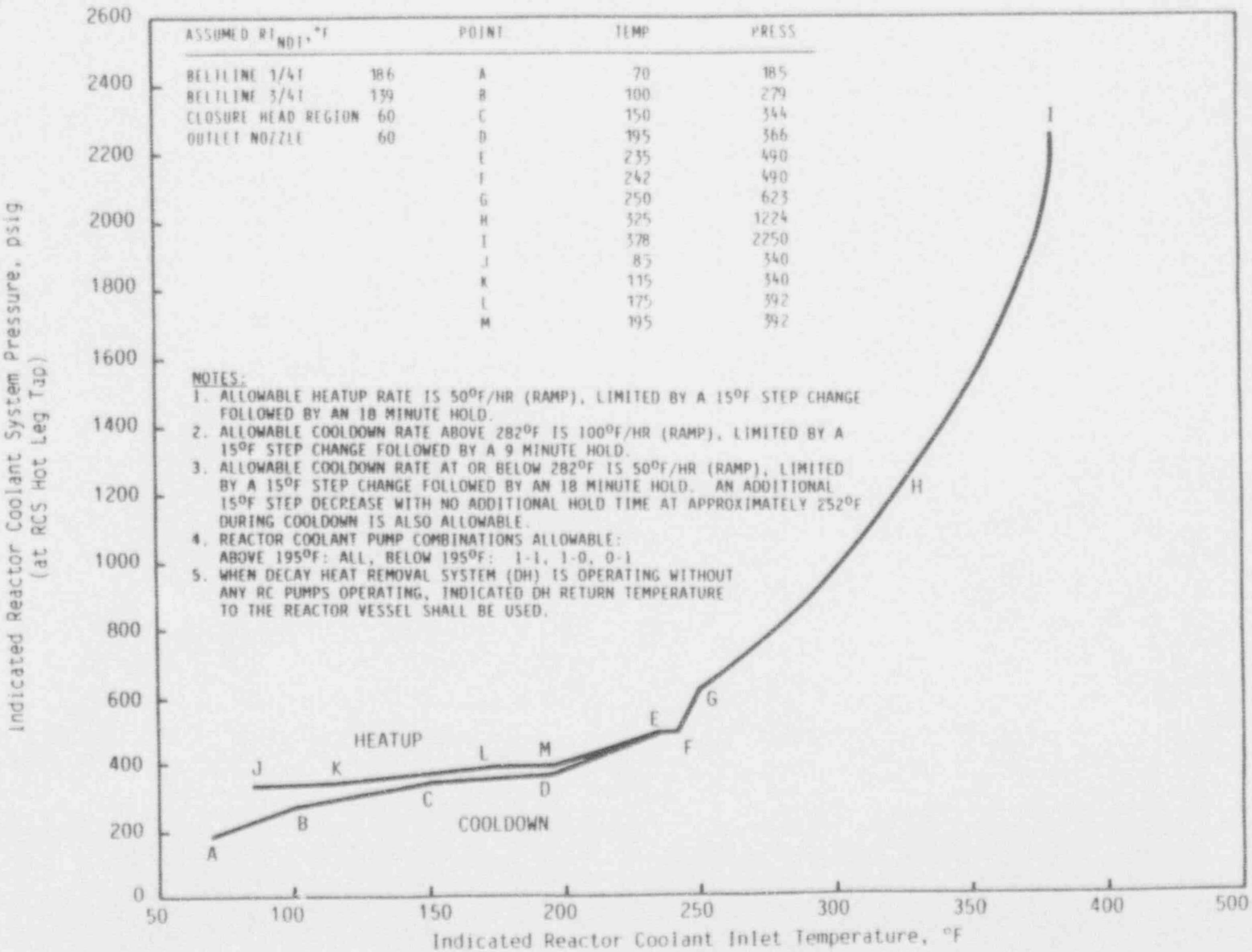
(3) BAW-1901, Analysis of Capsule TMI-1C, GPU Nuclear, Three Mile Island Nuclear Station - Unit 1, Reactor Vessel Materials Surveillance Program

(4) BAW-1901, Supplement 1, Analysis of Capsule TMI-1C, GPU Nuclear, Three Mile Island Nuclear Station - Unit 1, Reactor Vessel Materials Surveillance Program, Supplement 1 Pressure - Temperature Limits

(5) BAW 2108, Rev. 1, B&WOG Materials Committee report, "Fluence Tracking System"

(6) GPUN Topical Report 095, Rev. 0, "Summary Evaluation of the TMI-1 Reactor Vessel Embrittlement for Operating Pressure/Temperature Limits"

Figure 3.1-1 Reactor Coolant System Heatup/Cooldown Limitations
(Applicable thru 32 EFPY)



ASSUMED RT_{NDT}, °F		POINT	TEMP	PRESS
BELTLINE 1/4T	186	A	70	185
BELTLINE 3/4T	139	B	100	279
CLOSURE HEAD REGION	60	C	150	344
OUTLET NOZZLE	60	D	195	366
		E	235	490
		F	242	490
		G	250	623
		H	325	1224
		I	378	2250
		J	85	340
		K	115	340
		L	175	392
		M	195	392

NOTES:

1. ALLOWABLE HEATUP RATE IS 50°F/HR (RAMP), LIMITED BY A 15°F STEP CHANGE FOLLOWED BY AN 18 MINUTE HOLD.
2. ALLOWABLE COOLDOWN RATE ABOVE 282°F IS 100°F/HR (RAMP), LIMITED BY A 15°F STEP CHANGE FOLLOWED BY A 9 MINUTE HOLD.
3. ALLOWABLE COOLDOWN RATE AT OR BELOW 282°F IS 50°F/HR (RAMP), LIMITED BY A 15°F STEP CHANGE FOLLOWED BY AN 18 MINUTE HOLD. AN ADDITIONAL 15°F STEP DECREASE WITH NO ADDITIONAL HOLD TIME AT APPROXIMATELY 252°F DURING COOLDOWN IS ALSO ALLOWABLE.
4. REACTOR COOLANT PUMP COMBINATIONS ALLOWABLE:
   ABOVE 195°F: ALL, BELOW 195°F: 1-1, 1-0, 0-1
5. WHEN DECAY HEAT REMOVAL SYSTEM (DH) IS OPERATING WITHOUT ANY RC PUMPS OPERATING, INDICATED DH RETURN TEMPERATURE TO THE REACTOR VESSEL SHALL BE USED.

Figure 3.1-2 Reactor Coolant Inservice Leak and Hydrostatic Test
(Applicable thru 32EFPY)