Docket No. 50-336

Attachment 1

Millstone Nuclear Power Station, Unit No. 2

Proposed Revisions to Technical Specifications

Pressure-Temperature Limits

January, 1984

Revise pages:	3/4 4-17
	3/4 4-19b
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3/4.4.9 PRESSURE/TEMPERATURE LIMITS

REACTOR COOLANT SYSTEM

LIMITING CONDITION FOR OPERATION

3.4.9.1 The Reactor Coolant System (except the pressurizer) temperature and pressure shall be limited in accordance with the limit lines shown on Figure 3.4-2 during heatup, cooldown, criticality, and inservice leak and hydrostatic testing with:

a. A maximum heatup of 40°F in any one hour period with T_{avg} at or below 200°F, 50°F in any one hour period with T_{avg} at or below 300°F and above 200°F, and 100°F in any one hour period with T_{avg} above 300°F.

b. A maximum cooldown of 100°F in any one hour period with T_{avg} above 300°F and a maximum cooldown of 20°F in any one hour period with T_{avg} below 300°F.

c. A maximum temperature change of 5°F in any one hour period, during hydrostatic testing operations above system design pressure.

APPLICABILITY: MODES 1, 2*, 3, 4 and 5.

ACTION:

With any of the above limits exceeded, restore the temperature and/or pressure to within the limit within 30 minutes; perform an engineering evaluation to determine the effects of the out-of-limit condition on the fracture toughness properties of the Reactor Coolant System; determine that the Reactor Coolant System remains acceptable for continued operations or be in at least HOT STANDBY within the next 6 hours and reduce the RCS T_{avg} and pressure to less than 200°F and 500 psia, respectively, within the following 30 hours.

*See Special Test Exception 3.10.3.

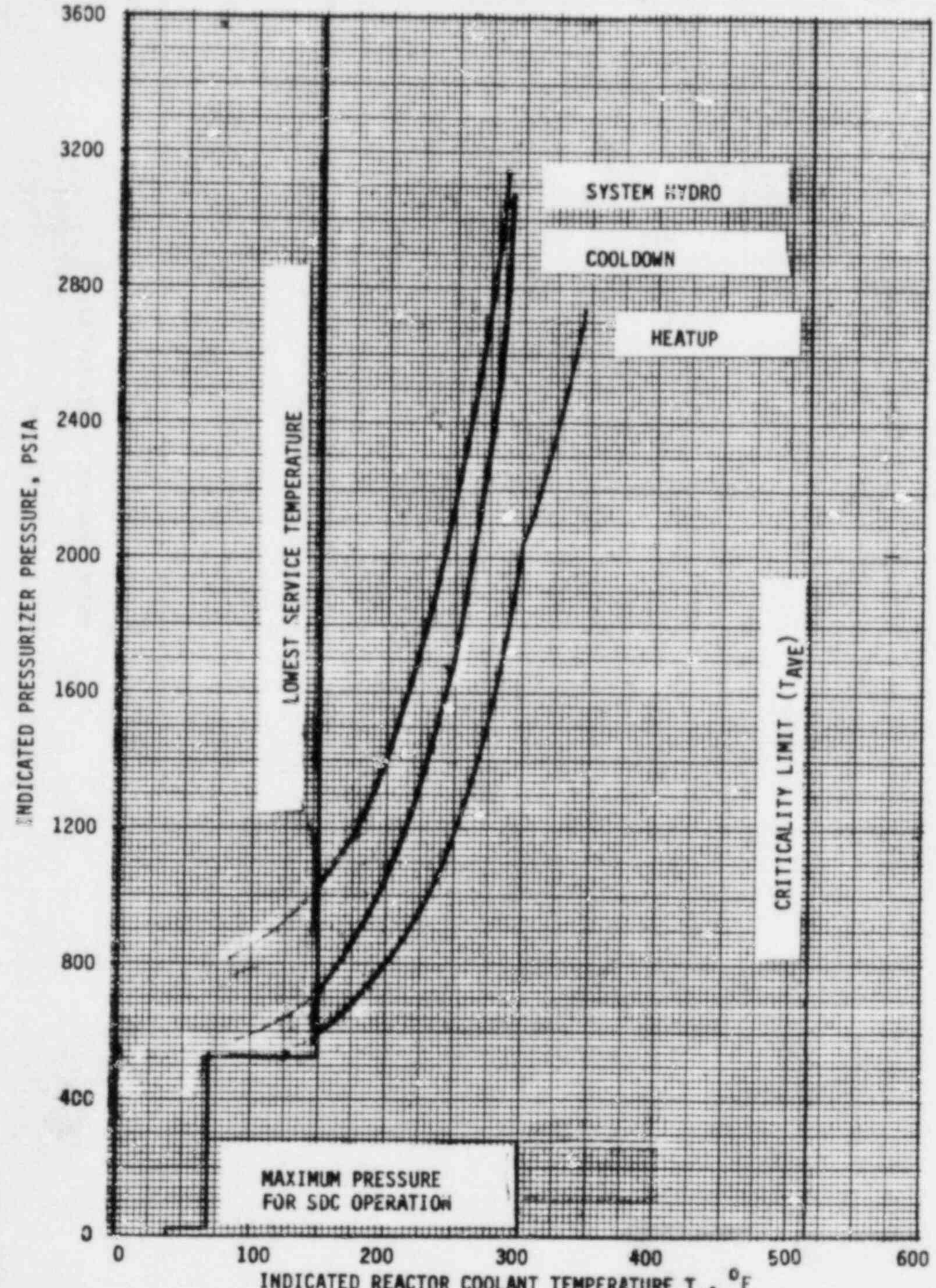

Figure 3.4-2
Reactor Coolant System Pressure Temperature Limitations for 7 Full Power Years

TABLE 4.4-3

Reactor Vessel Material Irradiation Surveillance Schedule

CAPSULE	SCHEDULE (EFPY)
W-97	3.0
W-104	10.0
W-284	17.0
W-263	24.0
W-277	32.0
W-83	Spare
W-97 (Flux Monitor)	10.0

The heatup and cooldown limit curves (Figure 3.4-2) are composite curves which were prepared by determining the most conservative case, with either the inside or outside wall controlling, for any heatup or cooldown rates of up to 100°F per hour. The heatup and cooldown curves were prepared based upon the most limiting value of the predicted adjusted reference temperature at the end of the service period indicated on Figure 3.4-2.

The reactor vessel materials have been tested to determine their initial RT_{NDT}; the results of these test are shown in Table 4.6-1 of the Final Safety Analysis Report. Reactor operation and resultant fast neutron (E>1 Mev) irradiation will cause an increase in the RT_{NDT}. Therefore, an adjusted reference temperature, based upon the fluence, can be predicted using the methods described in SECY-82-465 "NRC Staff Evaluation of Pressurized Thermal Shock", November, 1982.

The heatup and cooldown limit curves shown on Figure 3.4-2 include predicted adjustments for this shift in RT_{NDT} at the end of the applicable service period, as well as adjustments for possible errors in the pressure and temperature sensing instruments.

The actual shift in RT_{NDT} of the vessel material will be established periodically during operation by removing and evaluating, in accordance with ASTM E185-73, 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 essentially identical, the measured transition shift for a sample can be applied with confidence to the adjacent section of the reactor vessel. The heatup and cooldown curves must be recalculated when the RT_{NDT} determined from the surveillance capsule is different from the calculated RT_{NDT} for the equivalent capsule radiation exposure.

The pressure-temperature limit lines shown on Figure 3.4-2 for reactor criticality and for inservice leak and hydrostatic testing have been provided to assure compliance with the minimum temperature requirements of Appendix G to 10 CFR 50 for reactor criticality and for inservice leak and hydrostatic testing.

The maximum RT_{NDT} for all reactor coolant system pressure-retaining materials, with the exception of the reactor pressure vessel, has been determined to be 50°F. The Lowest Service Temperature limit line shown on Figure 3.4-2 is based upon this RT_{NDT} since Article NB-2332 (Summer Addenda of 1972) of Section III of the ASME Boiler and Pressure Vessel Code requires the Lowest Service Temperature to be RT_{NDT} + 100°F

for piping, pumps and valves. Below this temperature, the system pressure must be limited to a maximum of 20% of the system's hydrostatic test pressure of 3125 psia.

Included in this evaluation is consideration of flange protection in accordance with 10CFR50, Appendix G. The requirement makes the minimum temperature RT_{NDT} plus 90°F for hydrostatic test and RT_{NDT} plus 120° for normal operation when the pressure exceeds 20 percent of the preservice system hydrostatic test pressure.

The number of reactor vessel irradiation surveillance specimens and the frequencies for removing and testing these specimens are provided in Table 4.4-3 to assure compliance with the requirements of Appendix H to 10 CFR Part 50.

The limitations imposed on the pressurizer heatup and cooldown rates and spray water temperature differential are provided to assure that the pressurizer is operated within the design criteria assumed for the fatigue analysis performed in accordance with the ASME Code requirements.

The OPERABILITY of two PORVs or an RCS vent opening of greater than 1.3 square inches ensures that the RCS will be protected from pressure transients which could exceed the limits of Appendix G to 10 CFR Part 50 when one or more of the RCS cold legs are ≤ 275°F. Either PORV has adequate relieving capability to protect the RCS from overpressurization when the transient is limited to either (1) the start of an idle RCP with the secondary water temperature of the steam generator ≤ 43°F (31°F when measured by a surface contact instrument) above the coolant temperature in the reactor vessel or (2) the start of a HPSI pump and its injection into a water solid RCS.

3/4.4.10 STRUCTURAL INTEGRITY

The inservice inspection and testing programs for ASME Code Class 1, 2 and 3 components ensure that the structural integrity and operational readiness of these components will be maintained at an acceptable level throughout the life of the plant. These programs are in accordance with Section XI of the ASME Boiler and Pressure Vessel Code and applicable Addenda as required by 10 CFR Part 50.55a(g) except where specific written relief has been granted by the Commission pursuant to 10 CFR Part 50.55a(g)(6)(i).

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Attachment 2

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Surveillance Capsule W-97

Analysis Results

January, 1984