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SUBJECT: Forwards retyped Tech Spec pages,per 900910 application for amend to License NPF-63 re RCS pressure-temp limits.

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SHEARON HARRIS NUCLEAR POWER PLANT
DOCKET NO. 50-400/LICENSE NO. NPF-63
RETYPED TECHNICAL SPECIFICATION PAGES
REACTOR COOLANT SYSTEM PRESSURE-TEMPERATURE LIMITS

Gentlemen:

On September 10, 1990, Carolina Power & Light Company requested a License Amendment for the Shearon Harris Nuclear Power Plant, Unit 1, Technical Specifications concerning the Reactor Coolant System Pressure-Temperature Limits. This change revises the heatup and cooldown limitations of Specification 3/4.4.9. Enclosure 1 provides the retyped, amendment bar pages for your convenience in issuance of the proposed Technical Specification change.

Please refer any questions regarding this submittal to Mr. Steve Chaplin at (919) 546-6623.

Yours very truly,

RW Prunty

for Leonard I. Loflin
Manager
Nuclear Licensing Section

LIL/lmr (rtspage)

Enclosure

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A001
1/1

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ENCLOSURE 1

SHEARON HARRIS NUCLEAR POWER PLANT
DOCKET NO. 50-400/LICENSE NO. NPF-63
RETYPED TECHNICAL SPECIFICATION PAGES

REACTOR COOLANT SYSTEM PRESSURE-TEMPERATURE LIMITS

INDEX

LIMITING CONDITIONS FOR OPERATION AND SURVEILLANCE REQUIREMENTS

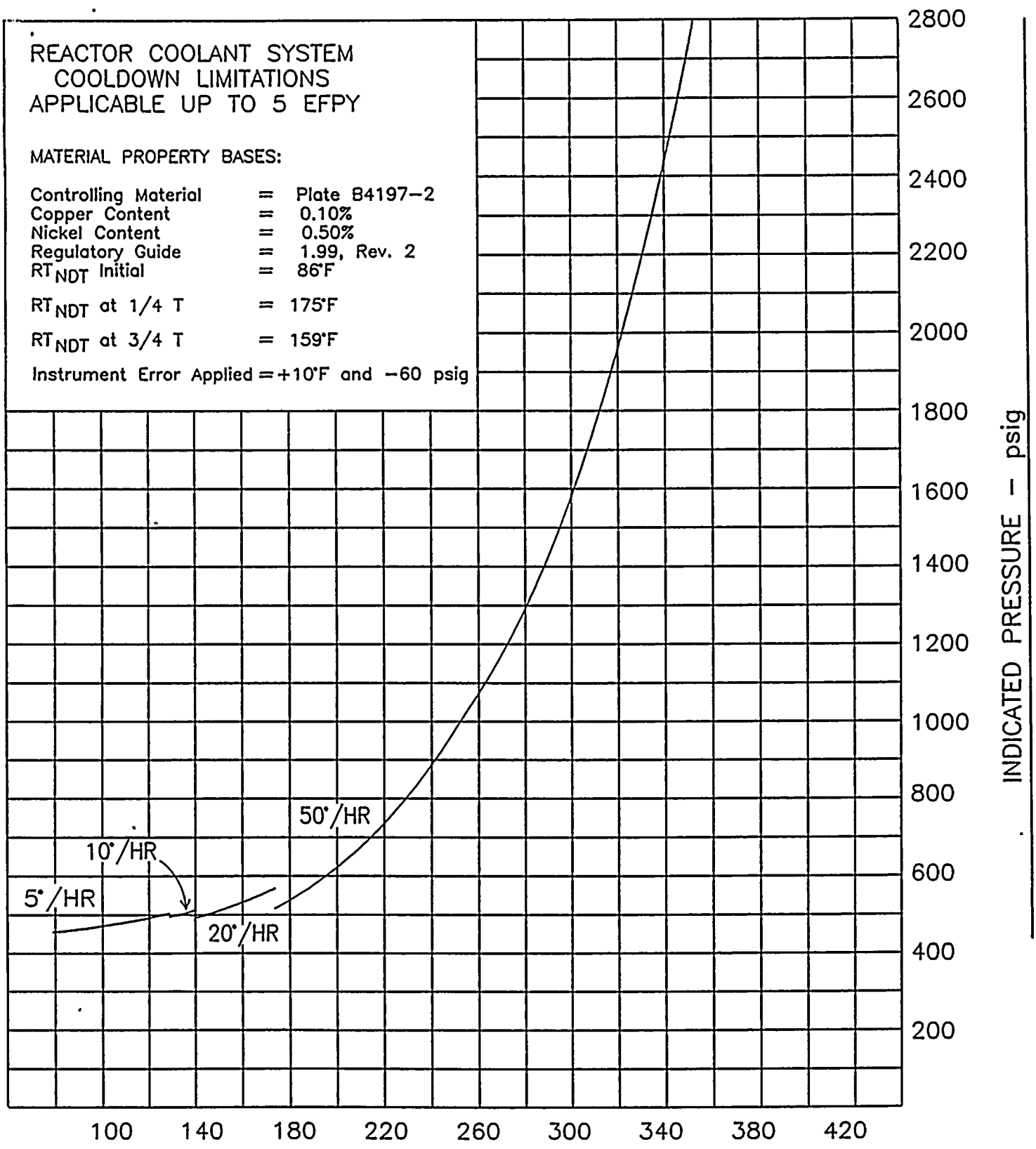


FIGURE 3.4-2

REACTOR COOLANT SYSTEM

COOLDOWN LIMITATIONS - APPLICABLE UP TO 5 EFPY

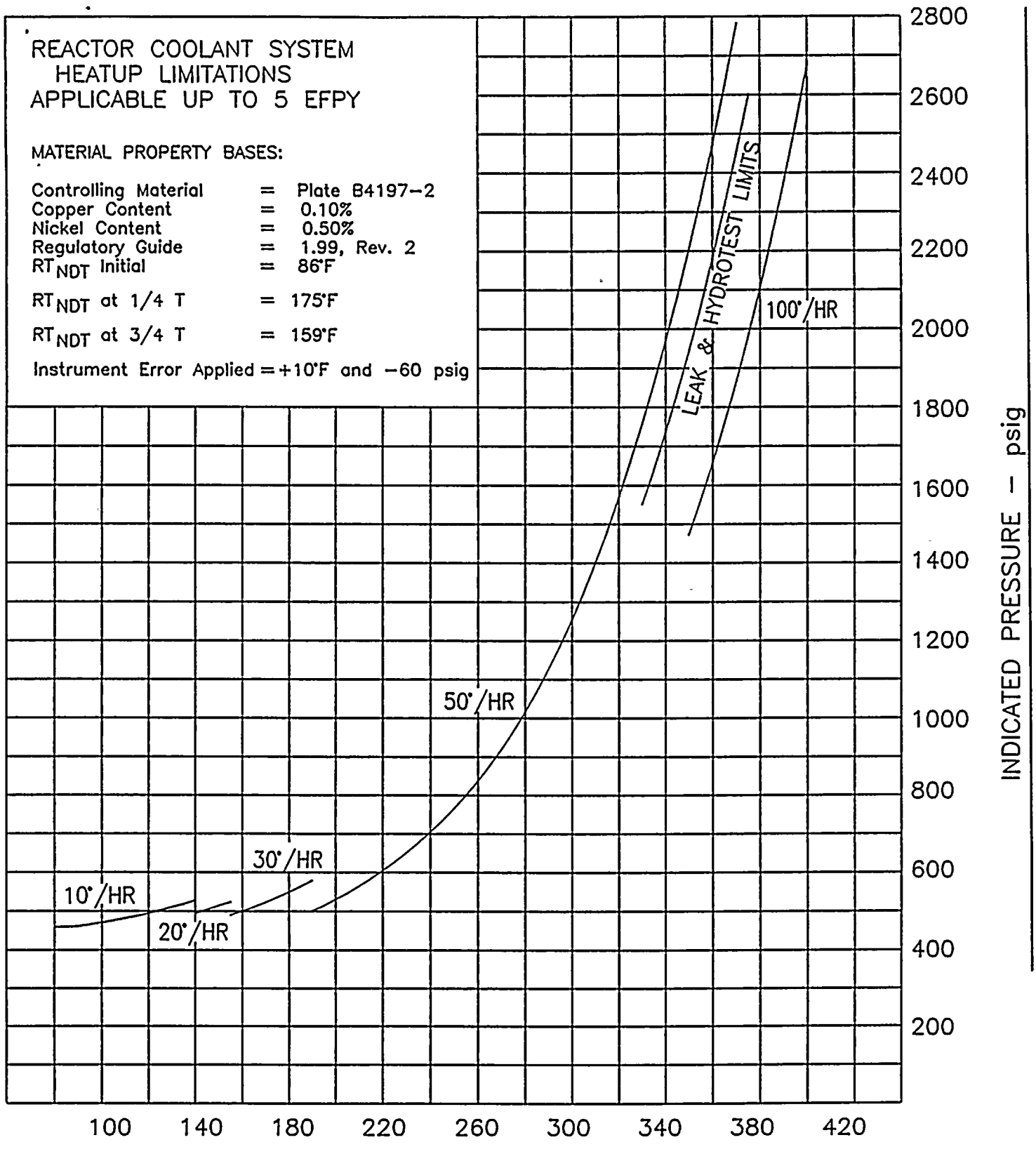


FIGURE 3.4-3

REACTOR COOLANT SYSTEM

HEATUP LIMITATIONS - APPLICABLE UP TO 5 EFPY

TABLE 4.4-6

MAXIMUM COOLDOWN AND HEATUP RATES
FOR MODES 4, 5, AND 6 (WITH REACTOR VESSEL HEAD ON)

COOLDOWN RATES

| TEMPERATURE* | COOLDOWN IN ANY 1 HOUR PERIOD** |
|---|---|
| 350-175°F | 50°F |
| 175-140°F | 20°F |
| 140-130°F | 10°F |
| < 130°F | 5°F |

HEATUP RATES

| TEMPERATURE* | HEATUP IN ANY 1 HOUR PERIOD** |
|---|---|
| < 140°F | 10°F |
| 140-155°F | 20°F |
| 155-190°F | 30°F |
| 190-350°F | 50°F |

*Temperature range used should be based on the lowest RCS cold leg value.

**Temperature used should be based on lowest RCS cold leg value except when no RCP is in operation; then use an operating RHR heat exchanger outlet temperature.

REACTOR COOLANT SYSTEM

OVERPRESSURE PROTECTION SYSTEMS

LIMITING CONDITION FOR OPERATION

3.4.9.4 At least one of the following Overpressure Protection Systems shall be OPERABLE:

a. The Reactor Coolant System (RCS) depressurized with an RCS vent of greater than or equal to 2.9 square inches, or

* b. Two power-operated relief valves (PORVs) with setpoints which do not exceed the limits established in Figure 3.4-4.

APPLICABILITY: MODE 4 when the temperature of any RCS cold leg is less than or equal to 325°F, MODE 5 and MODE 6 with the reactor vessel head on.

ACTION:

a. With one PORV inoperable, restore the inoperable PORV to OPERABLE status within 7 days or depressurize and vent the RCS through at least a 2.9 square inch vent within the next 8 hours.

b. With both PORVs inoperable, depressurize and vent the RCS through at least a 2.9 square inch vent within 8 hours.

c. In the event either the PORVs or the RCS vent(s) are used to mitigate an RCS pressure transient, a Special Report shall be prepared and submitted to the Commission pursuant to Specification 6.9.2 within 30 days. The report shall describe the circumstances initiating the transient, the effect of the PORVs or RCS vent(s) on the transient, and any corrective action necessary to prevent recurrence.

d. The provisions of Specification 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.4.9.4.1 Each PORV shall be demonstrated OPERABLE by:

a. Performance of an ANALOG CHANNEL OPERATIONAL TEST on the PORV actuation channel, but excluding valve operation, within 31 days prior to entering a condition in which the PORV is required OPERABLE and at least once per 31 days thereafter when the PORV is required OPERABLE;

b. Performance of a CHANNEL CALIBRATION on the PORV actuation channel at least once per 18 months; and

c. Verifying the PORV isolation valve is open at least once per 72 hours when the PORV is being used for overpressure protection.

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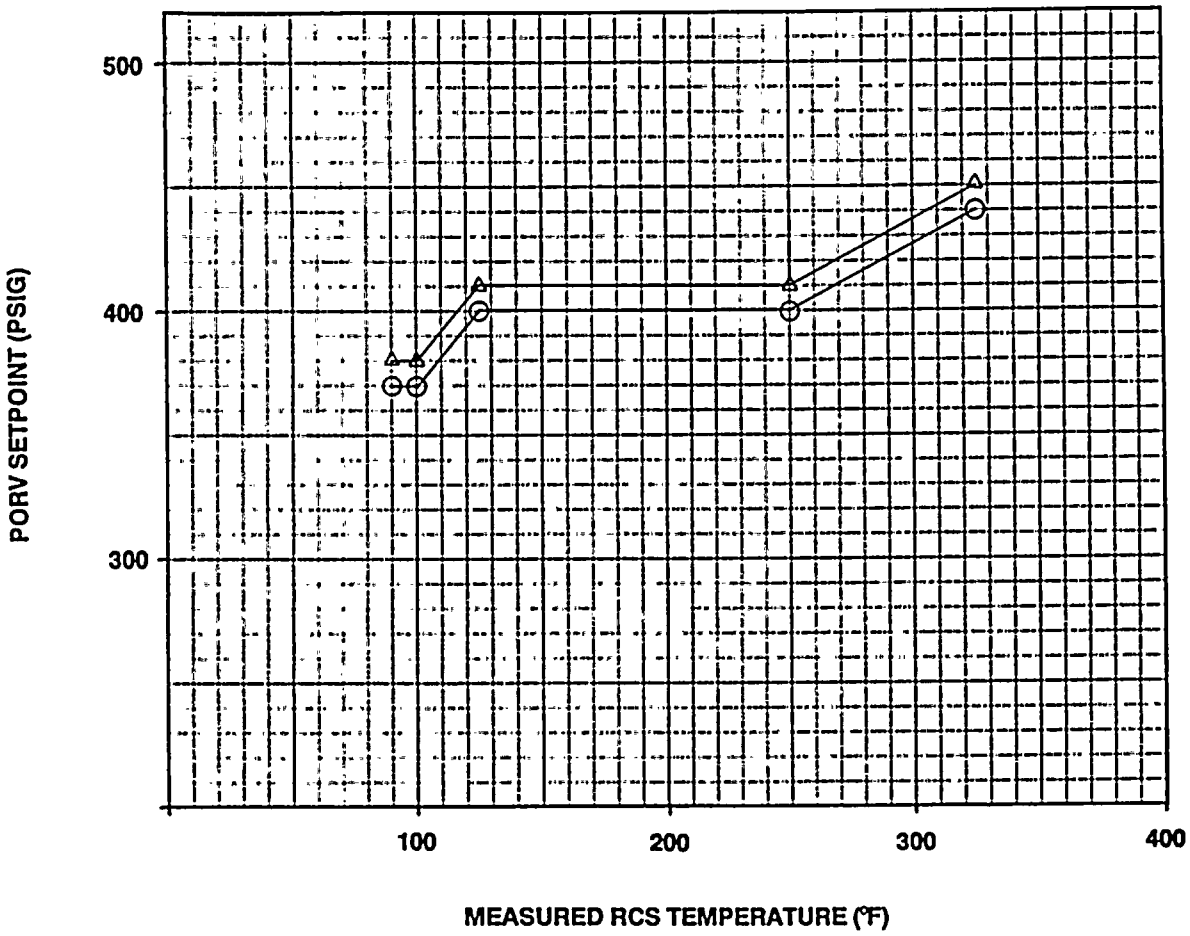
* Credit may only be taken for the setpoints when the RCS cold leg temperature $\geq$ 90°F.




| RCS TEMP °F | LOW PORV * PSIG ○ | HIGH PORV * PSIG △ |
|---|---|---|
| 90 | 370 | 380 |
| 100 | 370 | 380 |
| 125 | 400 | 410 |
| 250 | 400 | 410 |
| 300 | 427 | 437 |
| 325 | 440 | 450 |

* VALUES BASED ON 5 EFPY REACTOR VESSEL DATA AND CONTAINS MARGINS OF -10°F AND +60 PSIG FOR POSSIBLE INSTRUMENT ERROR

FIGURE 3.4-4

MAXIMUM ALLOWED PORV SETPOINT FOR THE LOW TEMPERATURE OVERPRESSURE SYSTEM

PRESSURE/TEMPERATURE LIMITS (Continued)

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 625°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 testing of the ferritic materials in the reactor vessel was performed in accordance with the 1971 Winter Addenda to Section III of the ASME Boiler and Pressure Vessel Code. These properties are then evaluated in accordance with the NRC Standard Review Plan.

Heatup and cooldown limit curves are calculated using the most limiting value of the nil-ductility reference temperature, $RT_{NDT}$, at the end of 5 effective full power years (EFPY) of service life. The 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 $RT_{NDT}$; the results of these tests are shown in Table B 3/4.4-1. 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 fluence, copper content, and nickel content of the material in question, can be predicted using Figure B 3/4.4-1 and the value of $\Delta RT_{NDT}$ computed by Regulatory Guide 1.99, Revision 2, "Radiation Embrittlement of Reactor Vessel Materials."