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## REACTOR COOLANT SYSTEM

### COLD SHUTDOWN

#### LIMITING CONDITION FOR OPERATION

3.4.1.3 At least two of the loop(s)/train(s) listed below shall be OPERABLE and at least one Reactor Coolant and/or shutdown cooling loops shall be in operation.\*

- a. Reactor Coolant Loop 1 and its associated steam generator and at least one associated Reactor Coolant pump,\*\*
- b. Reactor Coolant Loop 2 and its associated steam generator and at least one associated Reactor Coolant pump,\*\*
- c. Shutdown Cooling Train A,
- d. Shutdown Cooling Train B.

APPLICABILITY: MODE 4

#### ACTION:

- a. With less than the above required Reactor Coolant loops and/or shutdown cooling trains OPERABLE, immediately initiate corrective action to return the required loops/trains to OPERABLE status as soon as possible; if the remaining OPERABLE loop is a shutdown cooling train, be in COLD SHUTDOWN within 24 hours.
- b. With no Reactor Coolant loop or shutdown cooling train in operation, suspend all operations involving a reduction in boron concentration of the Reactor Coolant System and immediately initiate corrective action to return the required coolant loop/ train to operation.

\* All Reactor Coolant pumps and shutdown cooling pumps may be de-energized for up to 1 hour provided (1) no operations are permitted that would cause dilution of the Reactor Coolant System boron concentration, and (2) core outlet temperature is maintained at least 10°F below saturation temperature.

\*\* A Reactor Coolant pump shall not be started with one or more of the Reactor Coolant System cold leg temperatures less than or equal to 235°F unless 1) the pressurizer water volume is less than 900 cubic feet or 2) the secondary water temperature of each steam generator is less than 100°F above each of the Reactor Coolant System cold leg temperatures.

## REACTOR COOLANT SYSTEM

### COLD SHUTDOWN - LOOPS FILLED

#### LIMITING CONDITION FOR OPERATION

- 3.4.1.4.1      a.    At least one of the following loop(s)/trains listed below shall be OPERABLE and in operation\*:
1.    Reactor Coolant Loop 1 and its associated steam generator and at least one associated Reactor Coolant Pump\*\*
  2.    Reactor Coolant Loop 2 and its associated steam generator and at least one associated Reactor Coolant Pump\*\*
  3.    Shutdown Cooling Train A
  4.    Shutdown Cooling Train B
- b.    One additional Reactor Coolant Loop/shutdown cooling train shall be OPERABLE, or
- c.    The secondary side water level of each steam generator shall be greater than 10% (wide range).

APPLICABILITY:    MODE 5, with Reactor Coolant loops filled.

#### ACTION:

- a.    With less than the above required shutdown trains/loops OPERABLE or with less than the required steam generator level, immediately initiate corrective action to return the required trains/loops to OPERABLE status or restore the required level as soon as possible.
- b.    With no loop/train in operation, suspend all operations involving a reduction in boron concentration of the Reactor Coolant System and immediately initiate corrective action to return the required loop/train to operation.

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\*All reactor coolant pumps and shutdown cooling pumps may be de-energized for up to 1 hour provided (1) no operations are permitted that would cause dilution of the Reactor Coolant System boron concentration, and (2) core outlet temperature is maintained at least 10°F below saturation temperature.

\*\*A Reactor Coolant pump shall not be started with one or more of the Reactor Coolant System cold leg temperatures less than or equal to 235°F unless (1) the pressurizer water volume is less than 900 cubic feet, or (2) the secondary water temperature of each steam generator is less than 100°F above each of the Reactor Coolant system cold leg temperatures.

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## REACTOR COOLANT SYSTEM

### 3/4.4.8 PRESSURE/TEMPERATURE LIMITS

## REACTOR COOLANT SYSTEM

### LIMITING CONDITION FOR OPERATION

3.4.8.1 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 and Figure 3.4-3 during heatup, cooldown, criticality, and inservice leak and hydrostatic testing with:

- a. A maximum heatup of 30°F in any one hour period with RC cold leg temperature less than 280°F. A maximum heatup of 60°F in any one hour period with RC cold leg temperature greater than 280°F.
- b. A maximum cooldown of 30°F in any one hour period with RC cold leg temperatures less than 280°F. A maximum cooldown of 100°F in any one hour period with RC temperature greater than 280°F.
- c. A maximum temperature change of less than or equal to 10°F in any one hour period during inservice hydrostatic and leak testing operations above the heatup and cooldown limit curves.

APPLICABILITY: At all times.

#### 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 structural integrity 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<sub>avg</sub> and pressure to less than 200°F and 500 psia, respectively, within the following 30 hours.

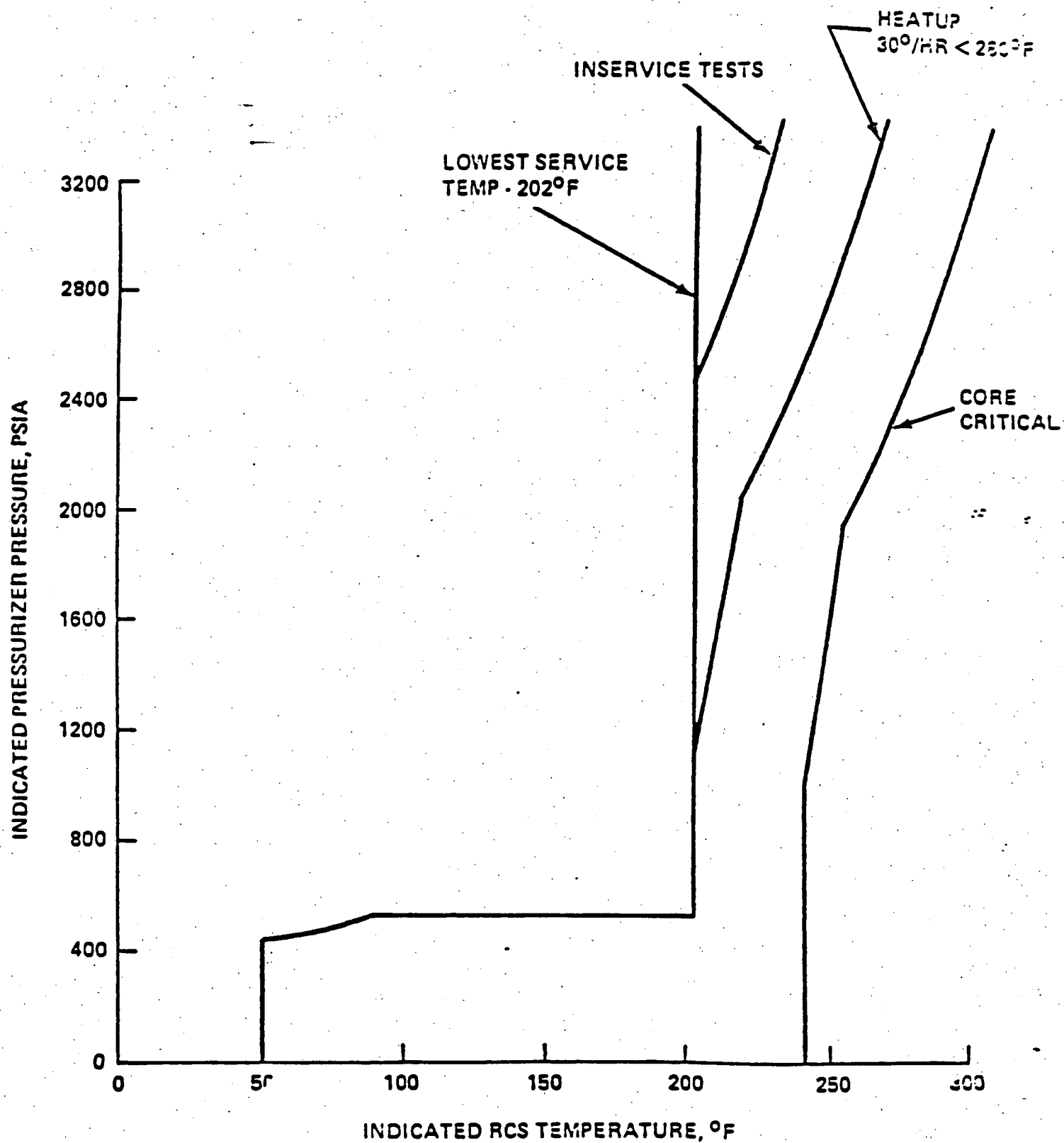
### SURVEILLANCE REQUIREMENTS

4.4.8.1.1 The Reactor Coolant System temperature and pressure shall be determined to be within the limits at least once per 30 minutes during system heatup, cooldown, and inservice leak and hydrostatic testing operations.

4.4.8.1.2 The reactor vessel material irradiation surveillance specimens shall be removed and examined, to determine changes in material properties, at the intervals required by 10 CFR 50 Appendix H in accordance with the schedule in Table 4.4-5. The results of these examinations shall be used to update Figures 3.4-2 and 3.4-3. Recalculate the Adjusted Reference Temperature based on the greater of the following:

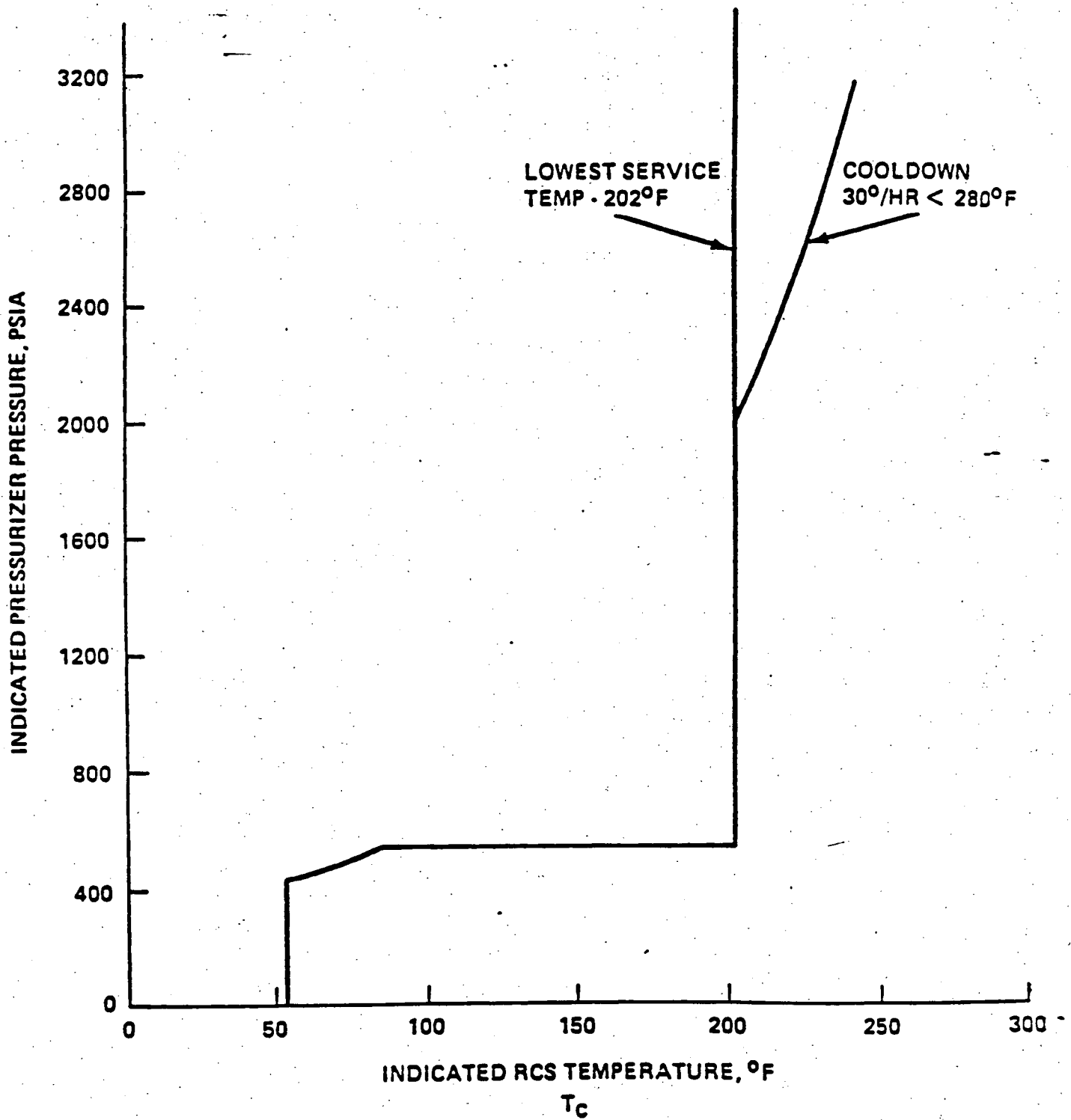
- a. The actual shift in reference temperature for plates C-6404-2 as determined by impact testing, or
- b. The predicted shift in reference temperature for weld seams 3-203A or 3-203B as determined by Regulatory Guide 1.99, Revision 1, April 1977, "Effects of Residual Elements on Predicted Radiation Damage to Reactor Vessel Materials."

Figure 3.4-2



HEATUP RCS PRESSURE/TEMPERATURE LIMITATIONS FOR 0-5 YEARS

Figure 3.4-3



COOLDOWN RCS PRESSURE/TEMPERATURE LIMITATIONS FOR 0-5 YEARS

## REACTOR COOLANT SYSTEM

### OVERPRESSURE PROTECTION SYSTEMS

RCS TEMPERATURE  $\leq 235^{\circ}\text{F}$

### LIMITING CONDITION FOR OPERATION

3.4.8.3.1 At least one of the following overpressure protection systems shall be OPERABLE:

- a. The Shutdown Cooling System Relief Valve (PSV9349) with:
  - 1) A lift setting of  $406 \pm 10$  psig\*, and
  - 2) Relief Valve isolation valves 2HV9337, 2HV9339, 2HV9377 and 2HV9378 open, or,
- b. The Reactor Coolant System depressurized with an RCS vent of greater than or equal to 5.6 square inches.

APPLICABILITY: MODE 4 when the temperature of any one RCS cold leg is less than or equal to  $235^{\circ}\text{F}$ ; MODE 5; MODE 6 with the reactor vessel head on.

### ACTION:

- a. With the SDCS Relief Valve inoperable, reduce  $T_{\text{avg}}$  to less than  $200^{\circ}\text{F}$ , depressurize and vent the RCS through a greater than or equal to 5.6 square inch vent within the next 8 hours.
- b. With one or both SDCS Relief Valve isolation valves in a single SDCS Relief Valve isolation valve pair (valve pair 2HV9337 and 2HV9339 or valve pair 2HV9377 and 2HV9378) closed, open the closed valve(s) within 7 days or reduce  $T_{\text{avg}}$  to less than  $200^{\circ}\text{F}$ , depressurize and vent the RCS through a greater than or equal to 5.6 inch vent within the next 8 hours.
- c. In the event either the SDCS Relief Valve or an RCS vent is 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 SDCS Relief Valve or RCS vent on the transient and any corrective action necessary to prevent recurrence.
- d. The provisions of Specification 3.0.4 are not applicable.

### SURVEILLANCE REQUIREMENTS

4.4.8.3.1.1 The SDCS Relief Valve shall be demonstrated OPERABLE by:

- a. Verifying at least once per 72 hours when the SDCS Relief Valve is being used for overpressure protection that SDCS Relief Valve isolation valves 2HV9337, 2HV9339, 2HV9377 and 2HV9378 are open.

\* For valve temperatures less than or equal to  $130^{\circ}\text{F}$ .



REACTOR COOLANT SYSTEM

OVERPRESSURE PROTECTION SYSTEMS

RCS TEMPERATURE > 235°F

LIMITING CONDITION FOR OPERATION

3.4.8.3.2 At least one of the following overpressure protection systems shall be OPERABLE:

- a. The Shutdown Cooling System Relief Valve (PSV9349) with:
  - 1) A lift setting of  $406 \pm 10$  psig\*, and
  - 2) Relief Valve isolation valves 2HV9337, 2HV9339, 2HV9377 and 2HV9378 open, or,
- b. A minimum of one pressurizer code safety valve with a lift setting of  $2500 \text{ psia} \pm 1\%^{**}$ .

APPLICABILITY: MODE 4 with RCS temperature above 235°F.

ACTION:

- a. With no safety or relief valve OPERABLE, be in COLD SHUTDOWN and vent the RCS through a greater than or equal to 5.6 square inch vent within the next 8 hours.
- b. In the event the SDCS Relief Valve or an RCS vent is 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 SDCS Relief Valve code safety valve or RCS vent on the transient and any corrective action necessary to prevent recurrence.

SURVEILLANCE REQUIREMENTS

4.4.8.3.2.1 The SDCS Relief Valve shall be demonstrated OPERABLE by:

- a. Verifying at least once per 72 hours that the SDCS Relief Valve isolation valves 2HV9337, 2HV9339, 2HV9377 and 2HV9378 are open when the SDCS Relief Valve is being used for overpressure protection.
- b. Verifying relief valve setpoint at least once per 30 months when tested pursuant to Specification 4.0.5.

4.4.8.3.2.2 The pressurizer code safety valve has no additional surveillance requirements other than those required by Specification 4.0.5.

4.4.8.3.2.3 The RCS vent shall be verified to be open at least once per 12 hours when the vent is being used for overpressure protection, except when the vent pathway is provided with a valve which is locked, sealed, or otherwise secured in the open position, then verify these valves open at least once per 31 days.

\*For valve temperatures less than or equal to 130°F.

\*\*The lift setting pressure shall correspond to ambient conditions of the valve at nominal operating temperature and pressure.

### 3/4.4 REACTOR COOLANT SYSTEM

#### BASES

#### 3/4.4.1 REACTOR COOLANT LOOPS AND COOLANT CIRCULATION

The plant is designed to operate with both reactor coolant loops and associated reactor coolant pumps in operation, and maintain DNBR greater than 1.31 during all normal operations and anticipated transients. As a result, in MODES 1 and 2 with one reactor coolant loop not in operation, this specification requires that the plant be in at least HOT STANDBY within 1 hour since no safety analysis has been conducted for operation with less than 4 reactor coolant pumps or less than two reactor coolant loops in operation.

In MODE 3, a single reactor coolant loop provides sufficient heat removal capability for removing decay heat; however, single failure considerations require that two loops be OPERABLE.

In MODE 4, and in MODE 5 with reactor coolant loops filled, a single reactor coolant loop or shutdown cooling train provides sufficient heat removal capability for removing decay heat; but single failure considerations require that at least two loops/trains (either RCS or shutdown cooling) be OPERABLE.

In MODE 5 with reactor coolant loops not filled, a single shutdown cooling train provides sufficient heat removal capability for removing decay heat; but single failure considerations, and the unavailability of the steam generators as a heat removing component, require that at least two shutdown cooling trains be OPERABLE.

The operation of one Reactor Coolant Pump or one shutdown cooling pump provides adequate flow to ensure mixing, prevent stratification and produce gradual reactivity changes during boron concentration reductions in the Reactor Coolant System. The reactivity change rate associated with boron reductions will, therefore, be within the capability of operator recognition and control.

The restrictions on starting a Reactor Coolant Pump in Modes 4 and 5 with one or more RCS cold legs less than or equal to 235°F are provided to prevent RCS pressure transients, caused by energy additions from the secondary system, which could exceed the limits of Appendix G to 10 CFR Part 50. The RCS will be protected against overpressure transients and will not exceed the limits of Appendix G by either (1) restricting the water volume in the pressurizer and thereby providing a volume for the primary coolant to expand into or (2) by restricting starting of the RCPs to when the secondary water temperature of each steam generator is less than 100°F above each of the RCS cold leg temperatures.

#### 3/4.4.2 SAFETY VALVES

The pressurizer code safety valves operate to prevent the RCS from being pressurized above its Safety Limit of 2750 psia. Each safety valve is designed to relieve  $4.6 \times 10^5$  lbs per hour of saturated steam at the valve setpoint plus 3% accumulation. The relief capacity of a single safety valve is adequate to relieve any overpressure condition which could occur during shutdown with RCS cold leg temperature greater than 235°F. In the event that no safety valves are OPERABLE and for RCS cold leg temperature less than or equal to 235°F, the operating shutdown cooling relief valve, connected to the RCS, provides overpressure relief capability and will prevent RCS overpressurization.

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## REACTOR COOLANT SYSTEM

### BASES

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#### SPECIFIC ACTIVITY (Continued)

Reducing  $T_{avg}$  to less than 500°F prevents the release of activity should a steam generator tube rupture since the saturation pressure of the primary coolant is below the lift pressure of the atmospheric steam relief valves. The surveillance requirements provide adequate assurance that excessive specific activity levels in the primary coolant will be detected in sufficient time to take corrective action. Information obtained on iodine spiking will be used to assess the parameters associated with spiking phenomena. A reduction in frequency of isotopic analyses following power changes may be permissible if justified by the data obtained.

#### 3/4.4.8 PRESSURE/TEMPERATURE LIMITS

All components in the Reactor Coolant System are designed to withstand the effects of cyclic loads due to system temperature and pressure changes. These cyclic loads are introduced by normal load transients, reactor trips, and startup and shutdown operations. The various categories of load cycles used for design purposes are provided in Section 3.9.1.1 of the FSAR. During startup and shutdown, the rates of temperature and pressure changes are limited so that the maximum specified heatup and cooldown rates are consistent with the design assumptions and satisfy the stress limits for cyclic operation.

During heatup, the thermal gradients in the reactor vessel wall produce thermal stresses which vary from compressive at the inner wall to tensile at the outer wall. These thermal induced compressive stresses tend to alleviate the tensile stresses induced by the internal pressure. Therefore, a pressure-temperature curve based on steady state conditions (i.e., no thermal stresses) represents a lower bound of all similar curves for finite heatup rates when the inner wall of the vessel is treated as the governing location.

The heatup analysis also covers the determination of pressure-temperature limitations for the case in which the outer wall of the vessel becomes the controlling location. The thermal gradients established during heatup produce tensile stresses at the outer wall of the vessel. These stresses are additive to the pressure induced tensile stresses which are already present. The thermal induced stresses at the outer wall of the vessel are tensile and are dependent on both the rate of heatup and the time along the heatup ramp; therefore, a lower bound curve similar to that described for the heatup of the inner wall cannot be defined. Consequently, for the cases in which the outer wall of the vessel becomes the stress controlling location, each heatup rate of interest must be analyzed on an individual basis.

# REACTOR COOLANT SYSTEM

## BASES

### PRESSURE/TEMPERATURE LIMITS (Continued)

The heatup and cooldown limit curves (Figures 3.4-2 and 3.4-3) are composite curves which were prepared by determining the most conservative case, with either the inside or outside wall controlling, for any heatup rate of up to 60°F/hr or cooldown rate of 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 the service period indicated on Figure 3.4-2 and 3.4-3.

The reactor vessel materials have been tested to determine their initial RT<sub>NOT</sub>; the results of these test are shown in Table B 3/4.4-1. Reactor operation and resultant fast neutron (E greater than 1 Mev) irradiation will cause an increase in the RT<sub>NOT</sub>. Therefore, an adjusted reference temperature, based upon the fluence and copper and phosphorous content of the material in question, can be predicted using FSAR Table 5.2-5 and the recommendations of Regulatory Guide 1.99, Revision 1, "Effects of Residual Elements on Predicted Radiation Damage to Reactor Vessel Materials." The heatup and cooldown limit curves, Figures 3.4-2 and 3.4-3, include predicted adjustments for this shift in RT<sub>NOT</sub> 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<sub>NOT</sub> of the vessel material will be established periodically during operation by removing and evaluating, in accordance with ASTM E185-73 and 10 CFR Appendix H, reactor vessel material irradiation surveillance specimens installed near the inside wall of the reactor vessel in the core area. The surveillance specimen withdrawal schedule is shown in Table 4.4-5. 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 taking into account the location of the sample closer to the core than the vessel wall by means of the Lead Factor. The heatup and cooldown curves must be recalculated when the delta RT<sub>NOT</sub> determined from the surveillance capsule is different from the calculated delta RT<sub>NOT</sub> for the equivalent capsule radiation exposure.

The pressure-temperature limit lines shown on Figure 3.4-2 and 3.4-3 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.

The maximum RT<sub>NOT</sub> for all reactor coolant system pressure-retaining materials, with the exception of the reactor pressure vessel, has been determined to be 90°F. The Lowest Service Temperature limit line shown on Figure 3.4-2 and 3.4-3 is based upon this RT<sub>NOT</sub> 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<sub>NOT</sub> + 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.

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.

## REACTOR COOLANT SYSTEM

### BASES

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#### PRESSURE/TEMPERATURE LIMITS (Continued)

The OPERABILITY of the Shutdown Cooling System relief valve or a RCS vent opening of greater than 5.6 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 less than or equal to 235°F. The Shutdown Cooling System relief valve 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 less than or equal to 100°F above the RCS cold leg temperatures or (2) inadvertent safety injection actuation with two HPSI pumps injecting into a water-solid RCS with full charging capacity and letdown isolated.

#### 3/4.4.9 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).

Components of the reactor coolant system were designed to provide access to permit inservice inspections in accordance with Section XI of the ASME Boiler and Pressure Vessel Code, 1974 Edition and Addenda through Summer 1975.

NPF-10-278

ATTACHMENT "B"

UNIT 2 PROPOSED SPECIFICATION

LIMITING CONDITION FOR OPERATION AND SURVEILLANCE REQUIREMENTS

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## REACTOR COOLANT SYSTEM

### HOT SHUTDOWN

#### LIMITING CONDITION FOR OPERATION

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3.4.1.3 At least two of the loop(s)/train(s) listed below shall be OPERABLE and at least one Reactor Coolant and/or shutdown cooling loops shall be in operation. \*

- a. Reactor Coolant Loop 1 and its associated steam generator and at least one associated Reactor Coolant pump,\*\*
- b. Reactor Coolant Loop 2 and its associated steam generator and at least one associated Reactor Coolant pump,\*\*
- c. Shutdown Cooling Train A,
- d. Shutdown Cooling Train B.

APPLICABILITY: MODE 4

#### ACTION:

- a. With less than the above required Reactor Coolant loops and/or shutdown cooling trains OPERABLE, immediately initiate corrective action to return the required loops/trains to OPERABLE status as soon as possible; if the remaining OPERABLE loop is a shutdown cooling train, be in COLD SHUTDOWN within 24 hours.
- b. With no Reactor Coolant loop or shutdown cooling train in operation, suspend all operations involving a reduction in boron concentration of the Reactor Coolant System and immediately initiate corrective action to return the required coolant loop/ train to operation.

\* All Reactor Coolant pumps and shutdown cooling pumps may be de-energized for up to 1 hour provided (1) no operations are permitted that would cause dilution of the Reactor Coolant System, boron concentration, and (2) core outlet temperature is maintained at least 100°F below saturation temperature.

\*\* A Reactor Coolant pump shall not be started with one or more of the Reactor Coolant System cold leg temperatures less than or equal to that specified in Table 3.4-3 when the secondary water temperature of each steam generator is greater than 100°F above each of the Reactor Coolant system cold leg temperatures.

LIMITING CONDITION FOR OPERATION

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3.4.1.4.1

- a. At least one of the following loop(s)/trains listed below shall be OPERABLE and in operation\*:
- 1. Reactor Coolant Loop 1 and its associated steam generator and at least one associated Reactor Coolant Pump\*\*
  - 2. Reactor Coolant Loop 2 and its associated steam generator and at least one associated Reactor Coolant Pump\*\*
  - 3. Shutdown Cooling Train A
  - 4. Shutdown Cooling Train B
- b. One additional Reactor Coolant Loop/shutdown cooling train shall be OPERABLE, or
- c. The secondary side water level of each steam generator shall be greater than 10% (wide range).

APPLICABILITY: MODE 5, with Reactor Coolant loops filled.

ACTION:

- a. With less than the above required shutdown trains/loops OPERABLE or with less than the required steam generator level, immediately initiate corrective action to return the required trains/loops to OPERABLE status or restore the required level as soon as possible.
- b. With no loop/train in operation, suspend all operations involving a reduction in boron concentration of the Reactor Coolant System and immediately initiate corrective action to return the required loop/train to operation.

\* All reactor coolant pumps and shutdown cooling pumps may be de-energized for up to 1 hour provided (1) no operations are permitted that would cause dilution of the Reactor Coolant System boron concentration, and (2) core outlet temperature is maintained at least 100°F below saturation temperature.

\*\* A Reactor Coolant pump shall not be started with one or more of the Reactor Coolant System cold leg temperatures less than or equal to that specified in Table 3.4-3 when the secondary water temperature of each steam generator is greater than 100°F above each of the Reactor Coolant system cold leg temperatures.

3/4.4.8 PRESSURE/TEMPERATURE LIMITS

REACTOR COOLANT SYSTEM

LIMITING CONDITION FOR OPERATION

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3.4.8.1 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 and Figure 3.4-3 during heatup, cooldown, criticality, boltup, and inservice leak and hydrostatic testing with:

- a. A maximum heatup of 100°F in any one hour period with RC cold leg temperature less than 112°F. A maximum heatup of 30°F in any one hour period with RC cold leg temperature less than 163°F. A maximum heatup of 60°F in any one hour period with RC cold leg temperature greater than 163°F.
- b. A maximum cooldown of 100°F in any one hour period with RC cold leg temperature less than 103°F. A maximum cooldown of 30°F in any one hour period with RC cold leg temperatures less than 145°F. A maximum cooldown of 100°F in any one hour period with RC temperature greater than 145°F.
- c. A maximum temperature change of less than or equal to 100°F in any one hour period during inservice hydrostatic and leak testing operations above the heatup and cooldown limit curves.
- d. A minimum temperature of 86°F to tension reactor vessel head bolts.

APPLICABILITY: At all times.

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 structural integrity 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 Tavg and pressure to less than 200°F and 500 psia, respectively, within the following 30 hours.

SURVEILLANCE REQUIREMENTS

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4.4.8.1.1 The Reactor Coolant System temperature and pressure shall be determined to be within the limits at least once per 30 minutes during system heatup, cooldown, and inservice leak and hydrostatic testing operations.

## REACTOR COOLANT SYSTEM

### 3/4.4.8 PRESSURE/TEMPERATURE LIMITS

## REACTOR COOLANT SYSTEM

### SURVEILLANCE REQUIREMENTS (Continued)

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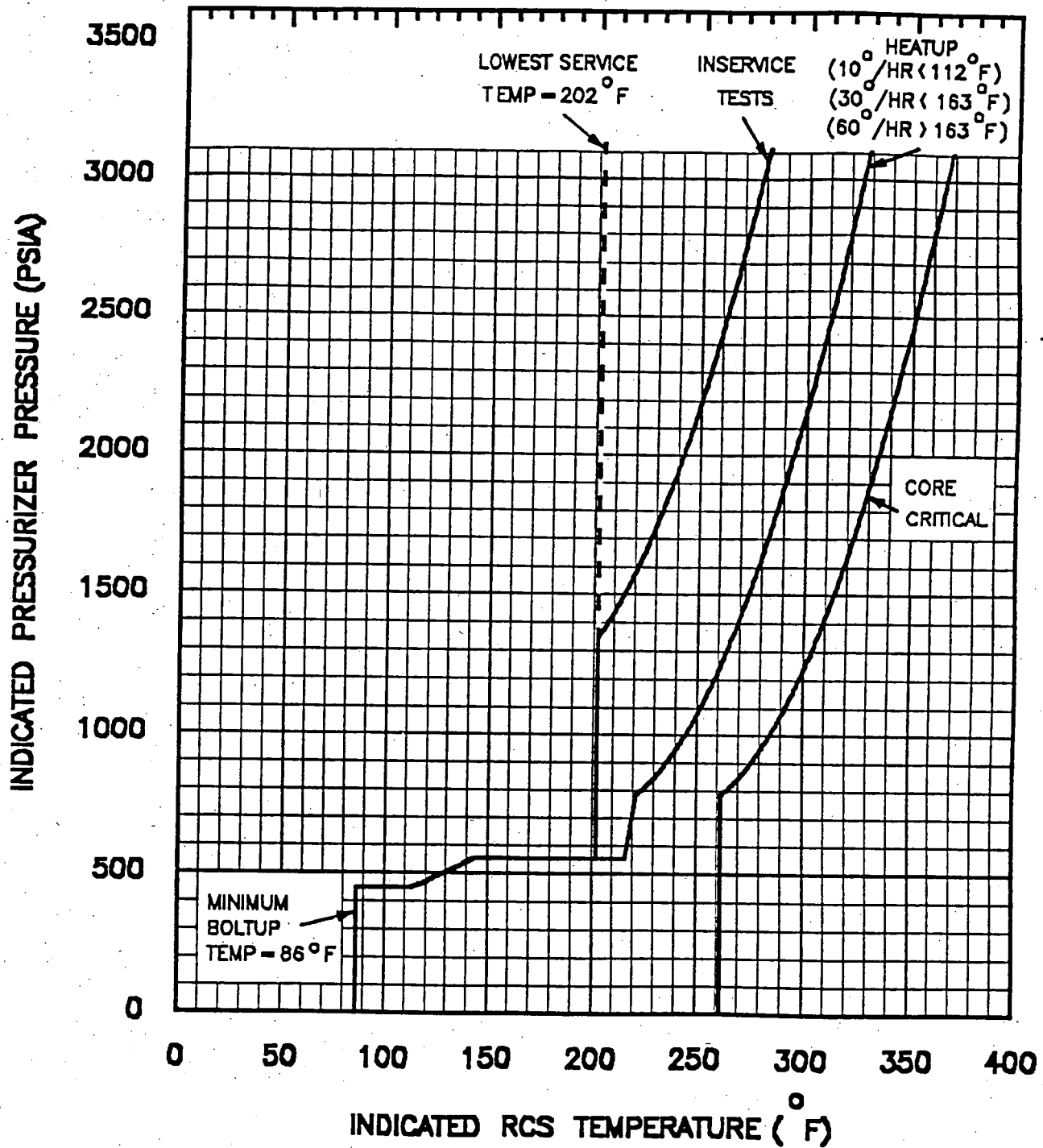
4.4.8.1.2 The reactor vessel material irradiation surveillance specimens shall be removed and examined, to determine changes in material properties, at the intervals required by 10 CFR 50 Appendix H in accordance with the schedule in Table 4.4-5. The results of these examinations shall be used to update Figures 3.4-2 and 3.4-3. Recalculate the Adjusted Reference Temperature based on the greater of the following:

- a. The mean value of shift in reference temperature for plates C-6404-3\*, or
- b. The predicted shift in reference temperature for weld seams 3-203A or 3-203B as determined by Regulatory Guide 1.99, Revision 2, "Radiation Embrittlement of Reactor Vessel Materials," May 1988.

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\* The most limiting material in the reactor vessel in accordance with the new Regulatory Guide 1.99, Revision 2, "Radiation Embrittlement of Reactor Vessel Materials," May 1988, has changed and are plates C-6404-3. Calculative procedures provided in the new guide should be used to obtain the mean values of shift in  $RT_{NDT}$  of C-6404-3 plates. Calculations are based on the actual shift in reference temperature as determined by impact testing on the existing plate C-6404-2 surveillance material.

**FIGURE 3.4-2**  
**SONGS 2 RCS HEATUP PRESSURE/TEMPERATURE**  
**LIMITATIONS FOR 4-10 EFY**



**FIGURE 3.4-3**  
**SONGS 2 RCS COOLDOWN PRESSURE/TEMPERATURE**  
**LIMITATIONS FOR 4-10 EFY**

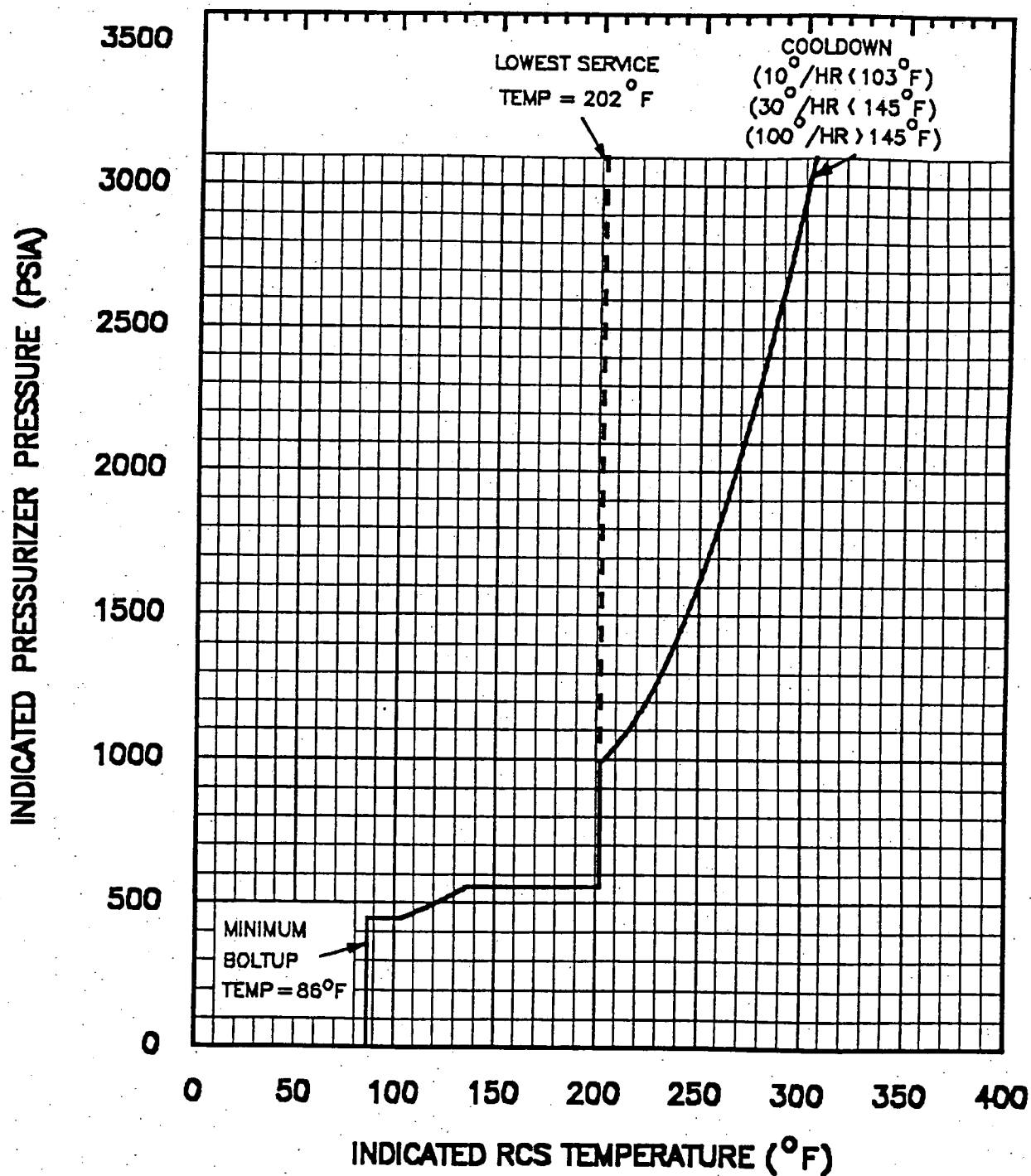


Table 3.4-3

Low Temperature RCS Overpressure Protection Range

<u>Operating Period, EFPY</u>	<u>Cold Leg Temperature, °F</u>	
	<u>During Heatup</u>	<u>During Cooldown</u>
4 to 10	≤ 312	≤ 287

REACTOR COOLANT SYSTEM  
RCS Temperature  $\leq 310^{\circ}\text{F}$   
OVERPRESSURE PROTECTION SYSTEMS

LIMITING CONDITION FOR OPERATION

3.4.8.3.1 At least one of the following overpressure protection systems shall be OPERABLE:

- a. The Shutdown Cooling System Relief Valve (PSV9349) with:
  - 1) A lift setting of  $406 \pm 10$  psig\*, and
  - 2) Relief Valve isolation valves 2HV9337, 2HV9339, 2HV9377 and 2HV9378 open, or,
- b. The Reactor Coolant System depressurized with an RCS vent of greater than or equal to 5.6 square inches.

APPLICABILITY: MODE 4 when the temperature of any one RCS cold leg is less than or equal to that specified in Table 3.4-3; MODE 5; MODE 6 with the reactor vessel head on.

ACTION:

- a. With the SDCS Relief Valve inoperable, reduce  $T_{avg}$  to less than  $200^{\circ}\text{F}$ , depressurize and vent the RCS through a greater than or equal to 5.6 square inch vent within the next 8 hours.
- b. With one or both SDCS Relief Valve isolation valves in a single SDCS Relief Valve isolation valve pair (valve pair 2HV9337 and 2HV9339 or valve pair 2HV9377 and 2HV9378) closed, open the closed valve(s) within 7 days or reduce  $T_{avg}$  to less than  $200^{\circ}\text{F}$ , depressurize and vent the RCS through a greater than or equal to 5.6 inch vent within the next 8 hours.
- c. In the event either the SDCS Relief Valve or an RCS vent is 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 SDCS Relief Valve or RCS vent on the transient and any corrective action necessary to prevent recurrence.
- d. The provisions of Specification 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.4.8.3.1.1 The SDCS Relief Valve shall be demonstrated OPERABLE by:

- a. Verifying at least once per 72 hours when the SDCS Relief Valve is being used for overpressure protection that SDCS Relief Valve isolation valves 2HV9337, 2HV9339, 2HV9377 and 2HV9378 are open.

\* For valve temperatures less than or equal to  $130^{\circ}\text{F}$ .



REACTOR COOLANT SYSTEM  
RCS Temperature > 300°F  
OVERPRESSURE PROTECTION SYSTEMS

LIMITING CONDITION FOR OPERATION

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3.4.8.3.2 At least one of the following overpressure protection systems shall be OPERABLE:

a. The Shutdown Cooling System Relief Valve (PSV9349) with:

- 1) A lift setting of  $406 \pm 10$  psig\*, and
- 2) Relief Valve isolation on valves 2HV9337, 2HV9339, 2HV9377 and 2HV9378 open, or,

b. A minimum of one pressurizer code safety valve with a lift setting of  $2500 \text{ psia} \pm 1\%$ \*\*.

APPLICABILITY: MODE 4 with RCS temperature above that specified in Table 3.4-3.

ACTION:

- a. With no safety or relief valve OPERABLE, be in COLD SHUTDOWN and vent the RCS through a greater than or equal to 5.6 square inch vent within the next 8 hours.
- b. In the event the SDCS Relief Valve or an RCS vent is 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 SDCS Relief Valve code safety valve or RCS vent on the transient and any corrective action necessary to prevent recurrence.

SURVEILLANCE REQUIREMENTS

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4.4.8.3.2.1 The SDCS Relief Valve shall be demonstrated OPERABLE by:

- a. Verifying at least once per 72 hours that the SDCS Relief Valve isolation valves 2HV9337, 2HV9339, 2HV9377 and 2HV9378 are open when the SDCS Relief Valve is being used for overpressure protection.
- b. Verifying relief valve setpoint at least once per 30 months when tested pursuant to Specification 4.0.5.

\* For valve temperatures less than or equal to 130°F

\*\* The lift setting pressure shall correspond to ambient conditions of the valve at nominal operating temperature and pressure.

REACTOR COOLANT SYSTEM

OVERPRESSURE PROTECTION SYSTEMS

SURVEILLANCE REQUIREMENTS (Continued)

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4.4.8.3.2.2 The pressurizer code safety valve has no additional surveillance requirements other than those required by Specification 4.0.5.

4.4.8.3.2.3 The RCS vent shall be verified to be open at least once per 12 hours when the vent is being used for overpressure protection, except when the vent pathway is provided with a valve which is locked, seated, or otherwise secured in the open position, then verify these valves open at least once per 31 days.

### 3/4.4 REACTOR COOLANT SYSTEM

#### BASES

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#### 3/4.4.1 REACTOR COOLANT LOOPS AND COOLANT CIRCULATION

The plant is designed to operate with both reactor coolant loops and associated reactor coolant pumps in operation, and maintain DNBR greater than 1.31 during all normal operations and anticipated transients. As a result, in MODES 1 and 2 with one reactor coolant loop not in operation, this specification requires that the plant be in at least HOT STANDBY within 1 hour since no safety analysis has been conducted for operation with less than 4 reactor coolant pumps or less than two reactor coolant loops in operation.

In MODE 3, a single reactor coolant loop provides sufficient heat removal capability for removing decay heat; however, single failure considerations require that two loops be OPERABLE.

In MODE 4, and in MODE 5 with reactor coolant loops filled, a single reactor coolant loop or shutdown cooling train provides sufficient heat removal capability for removing decay heat; but single failure considerations require that at least two loops/trains (either RCS or shutdown cooling) be OPERABLE.

In MODE 5 with reactor coolant loops not filled, a single shutdown cooling train provides sufficient heat removal capability for removing decay heat; but single failure considerations, and the unavailability of the steam generators as a heat removing component, require that at least two shutdown cooling trains be OPERABLE.

The operation of one Reactor Coolant Pump or one shutdown cooling pump provides adequate flow to ensure mixing, prevent stratification and produce gradual reactivity changes during boron concentration reductions in the Reactor Coolant System. The reactivity change rate associated with boron reductions will, therefore, be within the capability of operator recognition and control.

The restrictions on starting a Reactor Coolant Pump in Modes 4 and 5 with one or more RCS cold legs less than or equal to that specified in Table 3.4-3 are provided to prevent RCS pressure transients, caused by energy additions from the secondary system, which could exceed the limits of Appendix G to 10 CFR Part 50. The RCS will be protected against overpressure transients and will not exceed the limits of Appendix G by either (1) restricting the water volume in the pressurizer and thereby providing a volume for the primary coolant to expand into or (2) by restricting starting of the RCPs to when the secondary water temperature of each steam generator is less than 100°F above each of the RCS cold leg temperatures.

#### 3/4.4.2 SAFETY VALVES

The pressurizer code safety valves operate to prevent the RCS from being pressurized above its Safety Limit of 2750 psia. Each safety valve is designed to relieve  $4.6 \times 10^5$  lbs per hour of saturated steam at the valve

3/4.4.2 SAFETY VALVES (Continued)

setpoint plus 3% accumulation. The relief capacity of a single safety valve is adequate to relieve any overpressure condition which could occur during shutdown with RCS cold leg temperature greater than that specified in Table 3.4-3. In the event that no safety valves are OPERABLE and for RCS cold leg temperature less than or equal to that specified in Table 3.4-3, the operating shutdown cooling relief valve, connected to the RCS, provides overpressure relief capability and will prevent RCS overpressurization.

SPECIFIC ACTIVITY (Continued)

Reducing  $T_{avg}$  to less than 500°F prevents the release of activity should a steam generator tube rupture since the saturation pressure of the primary coolant is below the lift pressure of the atmospheric steam relief valves. The surveillance requirements provide adequate assurance that excessive specific activity levels in the primary coolant will be detected in sufficient time to take corrective action. Information obtained on iodine spiking will be used to assess the parameters associated with spiking phenomena. A reduction in frequency of isotopic analyses following power changes may be permissible if justified by the data obtained.

3/4.4.8 PRESSURE/TEMPERATURE LIMITS

All components in the Reactor Coolant System are designed to withstand the effects of cyclic loads due to system temperature and pressure changes. These cyclic loads are introduced by normal load transients, reactor trips, and startup and shutdown operations. The various categories of load cycles used for design purposes are provided in Section 3.9.1.1 of the FSAR. During startup and shutdown, the rates of temperature and pressure changes are limited so that the maximum specified heatup and cooldown rates are consistent with the design assumptions and satisfy the stress limits for cyclic operation.

During heatup, the thermal gradients in the reactor vessel wall produce thermal stresses which are tensile at the reactor vessel outside surface. Since reactor vessel internal pressure always produces tensile stresses at both the inside and outside surface locations, the total applied stress is greatest at the outside surface location. However, since neutron irradiation damage is larger at the inside surface location when compared to the outside surface, the inside surface flaw may be more limiting. Consequently, for the heatup analysis both the inside and outside surface flaw locations must be analyzed for the specific pressure and thermal loadings to determine which is more limiting.

During cooldown, the thermal gradients through the reactor vessel wall produce thermal stresses which are tensile at the reactor vessel inside surface and which are compressive at the reactor vessel outside surface. Since reactor vessel internal pressure always produces tensile stresses at both the inside and outside surface locations, the total applied stress is greatest at the inside surface location. Since the neutron irradiation damage is also greatest at the inside surface location the inside surface flaw is the limiting location. Consequently, only the inside surface flaw must be evaluated for the cooldown analysis.

The heatup and cooldown limit curves Figures, 3.4-2 and 3.4-3, are composite curves which were prepared by determining the most conservative case, with either the inside or outside wall controlling, for any heatup

PRESSURE/TEMPERATURE LIMITS (Continued)

rate of up to 600°F/hour or cooldown rate of up to 1000°F/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, and they include adjustments for possible errors in the pressure and temperature sensing instruments.

The reactor vessel materials have been tested to determine their initial RT<sub>NDT</sub>; the results of these test are shown in Table B 3/4.4-1. Reactor operation and resultant fast neutron (E greater than 1 Mev) irradiation will cause an increase in the RT<sub>NDT</sub>. Therefore, an adjusted reference temperature, based upon the fluence and copper and nickel content of the material in question, can be predicted using FSAR Table 5.2-5 and the recommendations of Regulatory Guide 1.99, Revision 2, "Radiation Embrittlement of Reactor Vessel Materials." The heatup and cooldown limit curves, Figures 3.4-2 and 3.4-3, include predicted adjustments for this shift in RT<sub>NDT</sub>, 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<sub>NDT</sub> of the vessel material will be established periodically during operation by removing and evaluating, in accordance with ASTM E185-73 and 10 CFR50 Appendix H, reactor vessel material irradiation surveillance specimens installed near the inside wall of the reactor vessel in the core area. The surveillance specimen withdrawal schedule is shown in Table 4.4-5. 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 taking into account the location of the sample closer to the core than the vessel wall by means of the Lead Factor. The heatup and cooldown curves must be recalculated when the delta RT<sub>NDT</sub> determined from the surveillance capsule is different from the calculated delta RT<sub>NDT</sub> the equivalent capsule radiation exposure.

The pressure-temperature limit lines shown on Figure 3.4-2 and 3.4-3 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.

The maximum RT<sub>NDT</sub> for all reactor coolant system pressure-retaining materials, with the exception of the reactor pressure vessel, has been determined to be 900°F. The Lowest Service Temperature limit line shown on Figure 3.4-2 and 3.4-3 is based upon this RT<sub>NDT</sub> 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<sub>NDT</sub> + 1000°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.

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.

PRESSURE/TEMPERATURE LIMITS (Continued)

The OPERABILITY of the Shutdown Cooling System relief valve or a RCS vent opening of greater than 5.6 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 less than or equal to that specified in Table 3.4-3. The Shutdown Cooling System relief valve has adequate relieving capability to protect the RCS from over-pressurization when the transient is limited to either (1) the start of an idle RCP with the secondary water temperature of the steam generator less than or equal to 100°F above the RCS cold leg temperatures or (2) inadvertent safety injection actuation with two HPSI pumps injecting into a water-solid RCS with full charging capacity and letdown isolated.

3/4.4.9 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).

Components of the reactor coolant system were designed to provide access to permit inservice inspections in accordance with Section XI of the ASME Boiler and Pressure Vessel Code, 1974 Edition and Addenda through Summer 1975.