ATTACHMENT A

EXISTING TECHNICAL SPECIFICATIONS AND BASES
UNIT 2

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LIMITING CONDITION FOR OPERATION AND SURVEILLANCE REQUIREMENTS

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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, boltup, and inservice leak and hydrostatic testing with:

a. A maximum heatup of 10°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 10°F in any one hour period with RC cold leg temperatures 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 10°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 T_{avg} 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.

3/4.4.8 PRESSURE/TEMPERATURE LIMITS

REACTOR COOLANT SYSTEM

SURVEILLANCE REQUIREMENTS (Continued)

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.

*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.

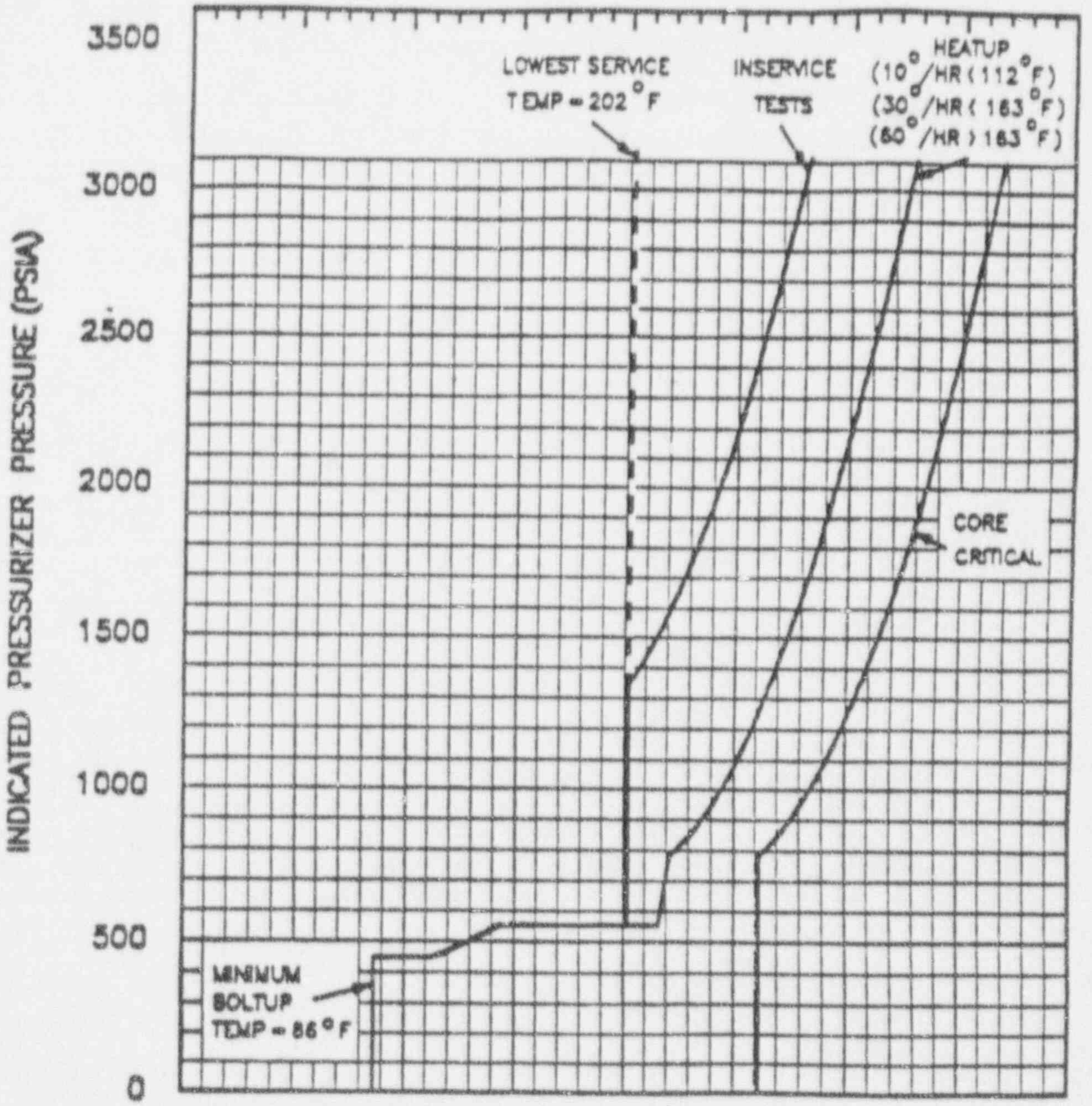

Figure 3.4-2

RCS HEATUP PRESSURE/TEMPERATURE LIMITATIONS FOR 4-8 EFPY

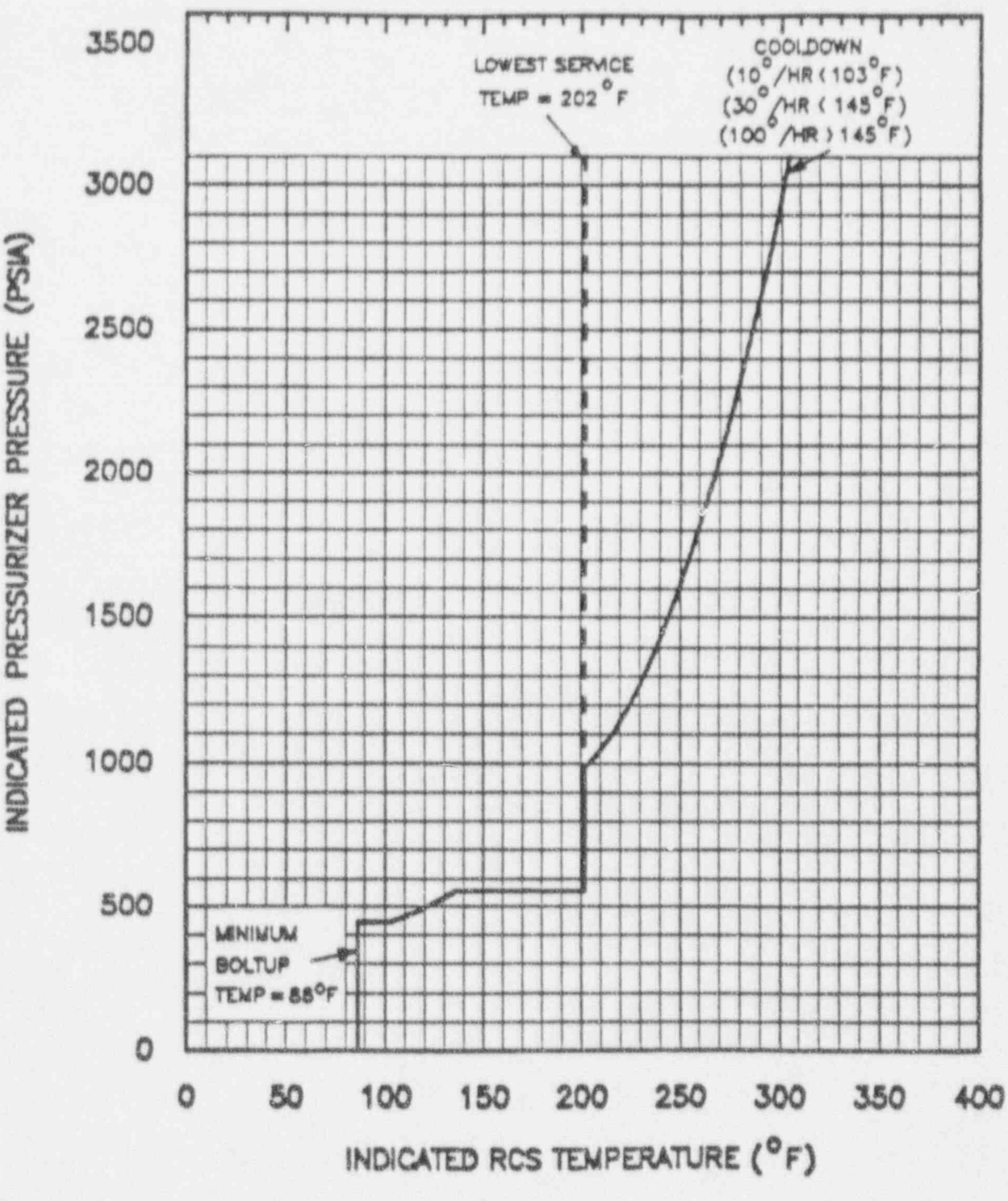

Figure 3.4-3

RCS COOLDOWN PRESSURE/TEMPERATURE LIMITATIONS FOR 4-8 EFPY

Table 3.4-3

Low Temperature RCS Overpressure Protection Range

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

LIMITING CONDITION FOR OPERATION

3.4.8.3.1 No more than two high-pressure safety injection pumps shall be OPERABLE and 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 ± 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 the enable temperatures specified in Table 3.4-3; MODE 5; and MODE 6 when the head is on the reactor vessel and the RCS is not vented.

ACTION:

a. With the SDCS Relief Valve inoperable, reduce T_{avg} to less than 200°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) or power-lock open the OPERABLE SDCS Relief Valve isolation valve pair within 24 hours, or reduce T_{avg} to less than 200°F, depressurize and vent the RCS through a greater than or equal to 5.6 inch vent within the next 8 hours.

c. **With more than two high-pressure safety injection pumps OPERABLE, secure the third high-pressure safety injection pump by racking out its motor circuit breaker or locking close its discharge valve within 8 hours.**

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

OVERPRESSURE PROTECTION SYSTEMS

RCS TEMPERATURE > 312°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 ± 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 psia ± 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

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.

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, 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_{NDT}; the results of these tests are shown in Table B 3/4.4-1. Reactor operation and resultant fast neutron (E greater than I Mev) irradiation will cause an increase in the RT_{NDT}. 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_{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 and 10 CFR 50 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_{NDT} determined from the surveillance capsule is different from the calculated delta RT_{NDT} 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_{NDT} 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_{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.

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.

TABLE B 3/4.4-1

REACTOR VESSEL TOUGHNESS

Piece No.	Code No.	Material	Vessel Location	Drop Weight Results	Temperature of Charpy V-Notch @ 30 ft - lb	Temperature of Charpy V-Notch @ 50 ft - lb	Minimum Upper Shelf Cv energy for Longitudinal Direction-ft lb
215-01	C-6403-1	A533GRBCL-1	Upper Shell Plate	40	15	35	130
215-01	C-6403-2	"	"	0	20	25	133
215-01	C-6403-3	"	"	-10	20	45	131
215-03	C-6404-1	"	Intermediate Shell Plate	-30	10	50	145
215-03	C-6404-2	"	"	-20	20	50	155
215-03	C-6404-3	"	"	-20	10	50	131
215-02	C-6404-4	"	Lower Shell Plate	-10	-5	25	124
215-02	C-6404-5	"	"	-20	10	25	134
215-02	C-6404-6	"	"	-10	-20	0	151
238-02	C-6401	A508C1-2	Vessel Flange Forging	-10	-70	-35	148
209-02	C-6402	"	Closure Head Flange Forging	-10	-90	-40	142
205-02	C-6410-1	"	Inlet Nozzle Forging	20	-40	-35	130
205-02	C-6410-2	"	"	0	-20	-5	135
205-02	C-6410-3	"	"	0	-15	-15	140
205-02	C-6410-4	"	"	0	-65	-50	140
205-06	C-6411-1	"	Outlet Nozzle Forging	-100	-30	-10	140
205-06	C6411-2	"	"	0	-35	-10	140
232-01	C-6424	A533GRBCL-1	Bottom Head Torus	-50	-20	10	122
232-02	C-6425	"	Bottom Head Dome	-50	-30	-20	136
205-03	C-6428-1	A508CL-1	Inlet Nozzle Forging S/E	-30	-70	-50	174
205-03	C-6428-2	"	"	-30	-70	-50	174
205-03	C-6428-3	"	"	-30	-70	-50	174
205-03	C-6428-4	"	"	-30	-70	-50	174
205-07	C-6429-1	"	Outlet Nozzle Ext. Forging	-30	-40	-25	229
205-07	C-6429-1	"	"	-30	-40	-25	229
231-02	C-6430-1	A533GRBCL-1	Closure Head Peels	+10	20	55	118
231-02	C-6431-1	"	"	-20	10	50	100
231-02	C-6432-1	"	"	-10	-15	45	115
231-02	C-6432	"	Closure Head Dome	-10	-15	45	115

ATTACHMENT B

PROPOSED TECHNICAL SPECIFICATIONS AND BASES
UNIT 2

LIMITING CONDITION FOR OPERATION AND SURVEILLANCE REQUIREMENTS

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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 With the reactor vessel head bolts tensioned*, ~~I~~ the Reactor Coolant System (except the pressurizer) temperature and pressure shall be limited in accordance with the limit lines shown on Figures 3.4-2, ~~and Figure~~ 3.4-34, 3.4-5, 3.4-6, and 3.4-7 during heatup, cooldown, criticality, ~~boltup,~~ and inservice leak and hydrostatic testing with:

a. ~~A maximum heatup of 10 °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~~ 1-hour period with RCS cold leg temperature greater than ~~163~~ or equal to 86°F. Sup 1

b. ~~A maximum cooldown of 10°F in any one hour period with RC cold leg temperatures less than 103°F.~~ A maximum cooldown ~~of 30°F~~ as specified by Figure 3.4-5 in any ~~one~~ 1-hour period with RCS cold leg temperature~~s~~ less than or equal to ~~145~~ 160°F. A maximum cooldown of 100°F in any ~~one~~ 1-hour period with RCS cold leg temperature greater than ~~145~~ 160°F. Sup 1

c. A maximum temperature change of ~~less than or equal to~~ 10°F in any ~~one~~ 1-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.

With the reactor vessel head bolts detensioned, the Reactor Coolant System (except the pressurizer) temperature shall be limited to a maximum heatup or cooldown of 60°F in any 1-hour period.

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_{avg} and pressure to less than 200°F and 500 psia, respectively, within the following 30 hours.

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* With the reactor vessel head bolts detensioned, RCS cold leg temperature may be less than 86°F.

REACTOR COOLANT SYSTEM

SURVEILLANCE REQUIREMENTS ~~(Continued)~~ | sup1

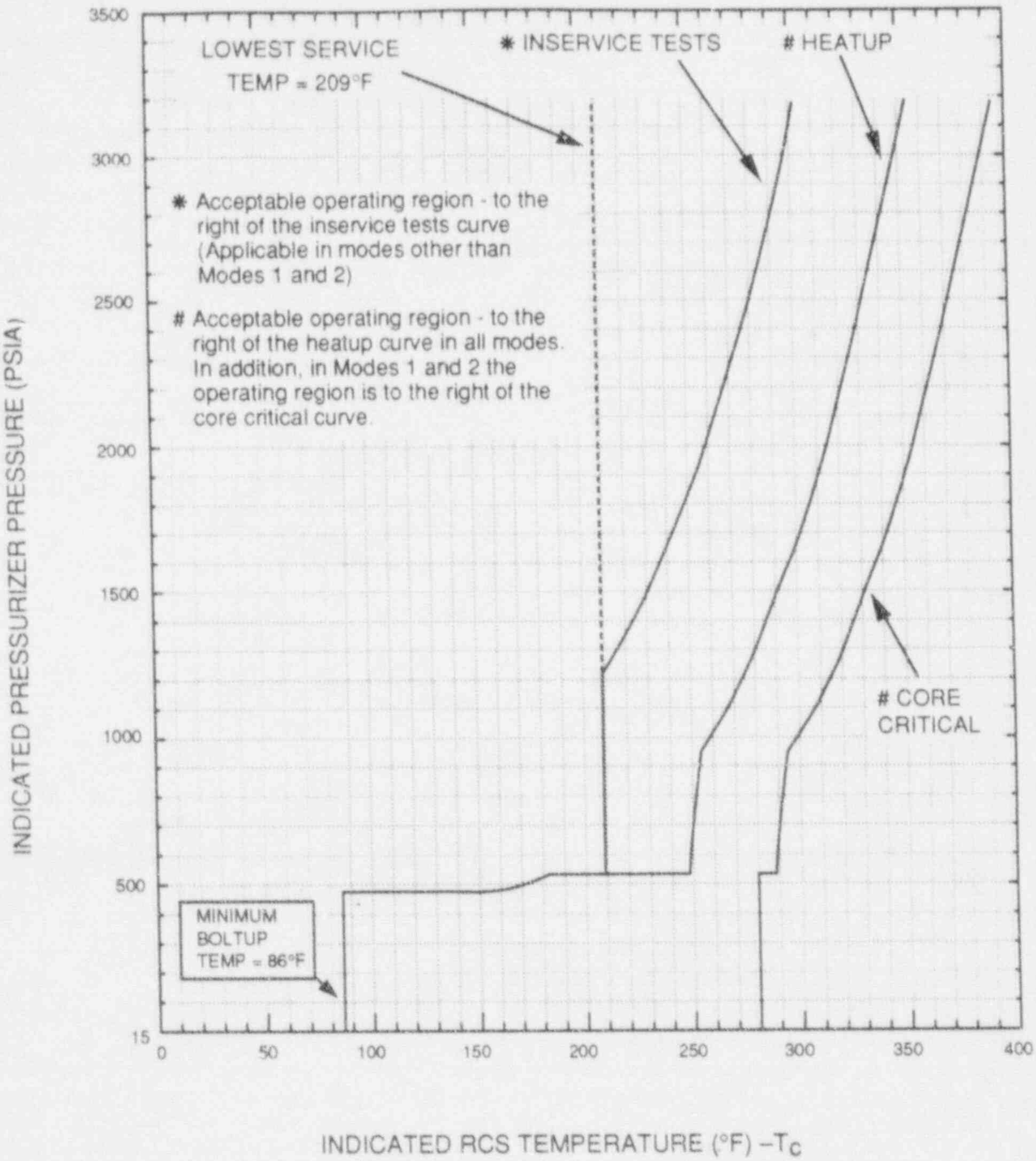
4.4.8.1.1 The Reactor Coolant System temperature and pressure shall be determined to be within the limits at least once per 30 minutes during system heatup, cooldown, and inservice leak and hydrostatic testing operations.

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~~ 3.4-4 through 3.4-7. Recalculate the Adjusted Reference Temperature ~~based on the greater of the following:~~ in accordance with Regulatory Guide 1.99, Revision 2, "Radiation Embrittlement of Reactor Vessel Materials," May 1988.

~~a. The mean value of shift in reference temperature for plates C-6403-3*, 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.~~

~~*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.~~



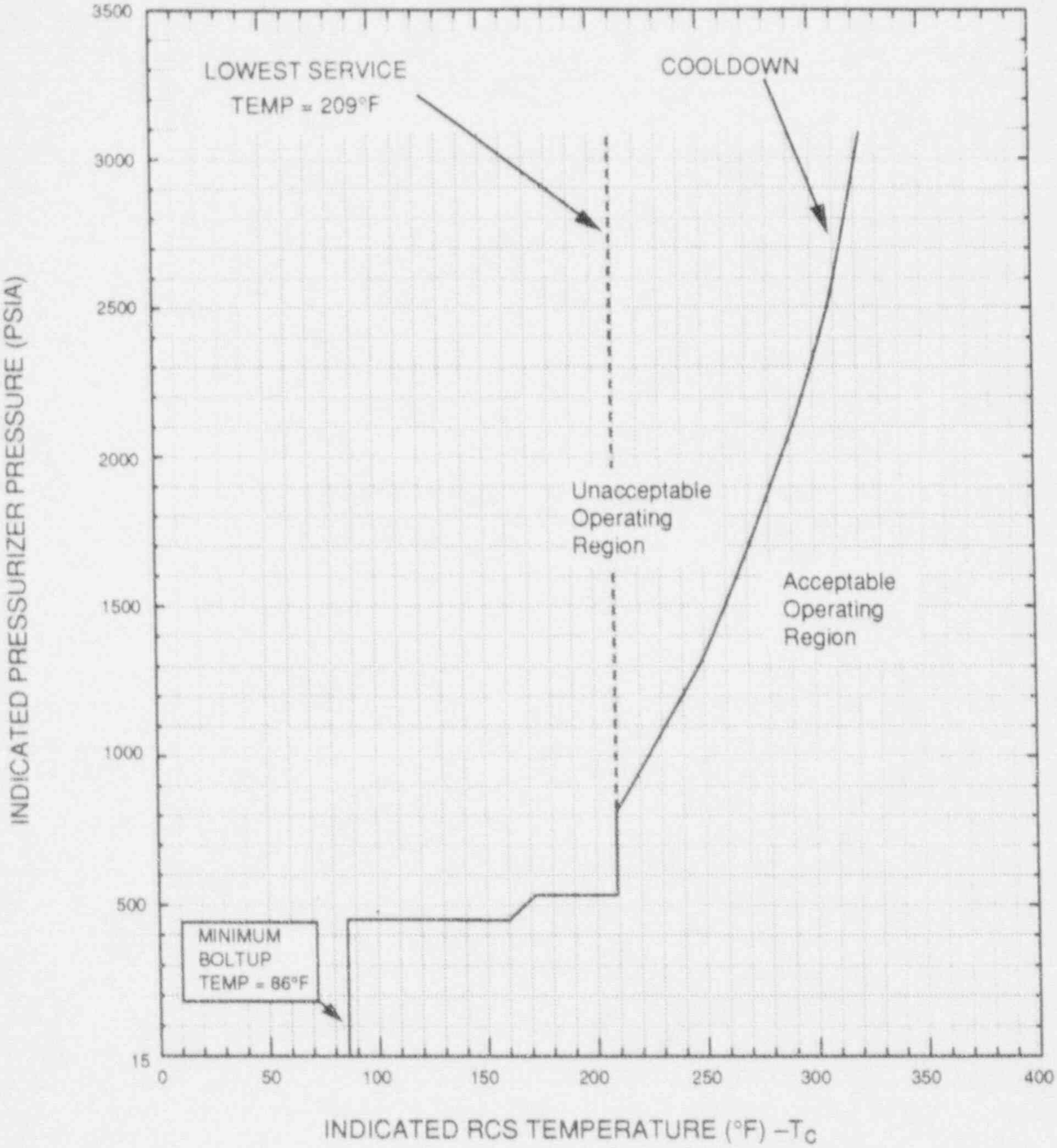
sup 1

FIGURE 3.4-2

SONGS 2 HEATUP RCS PRESSURE/TEMPERATURE
LIMITATIONS ~~FOR 4-8~~ UNTIL 20 EFPY
Normal Operation

sup 1

(Figure 3.4-3 - Not Used)

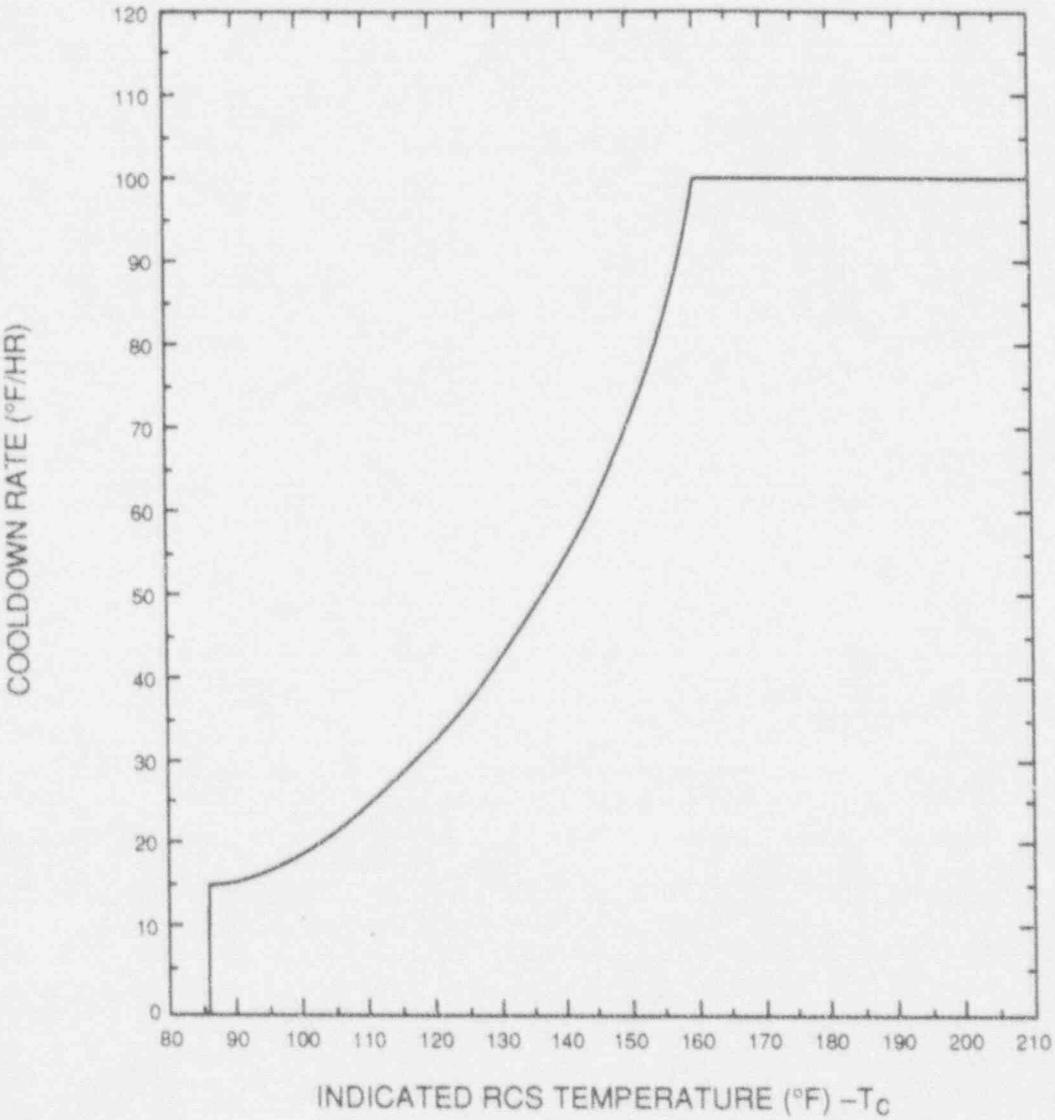


sup1

FIGURE ~~3.4-3~~ 3.4-4

SONGS 2 COOLDOWN RCS PRESSURE/TEMPERATURE
LIMITATIONS ~~FOR 4-8~~ UNTIL 20 EFPY
Normal Operation

sup1



Sup 1

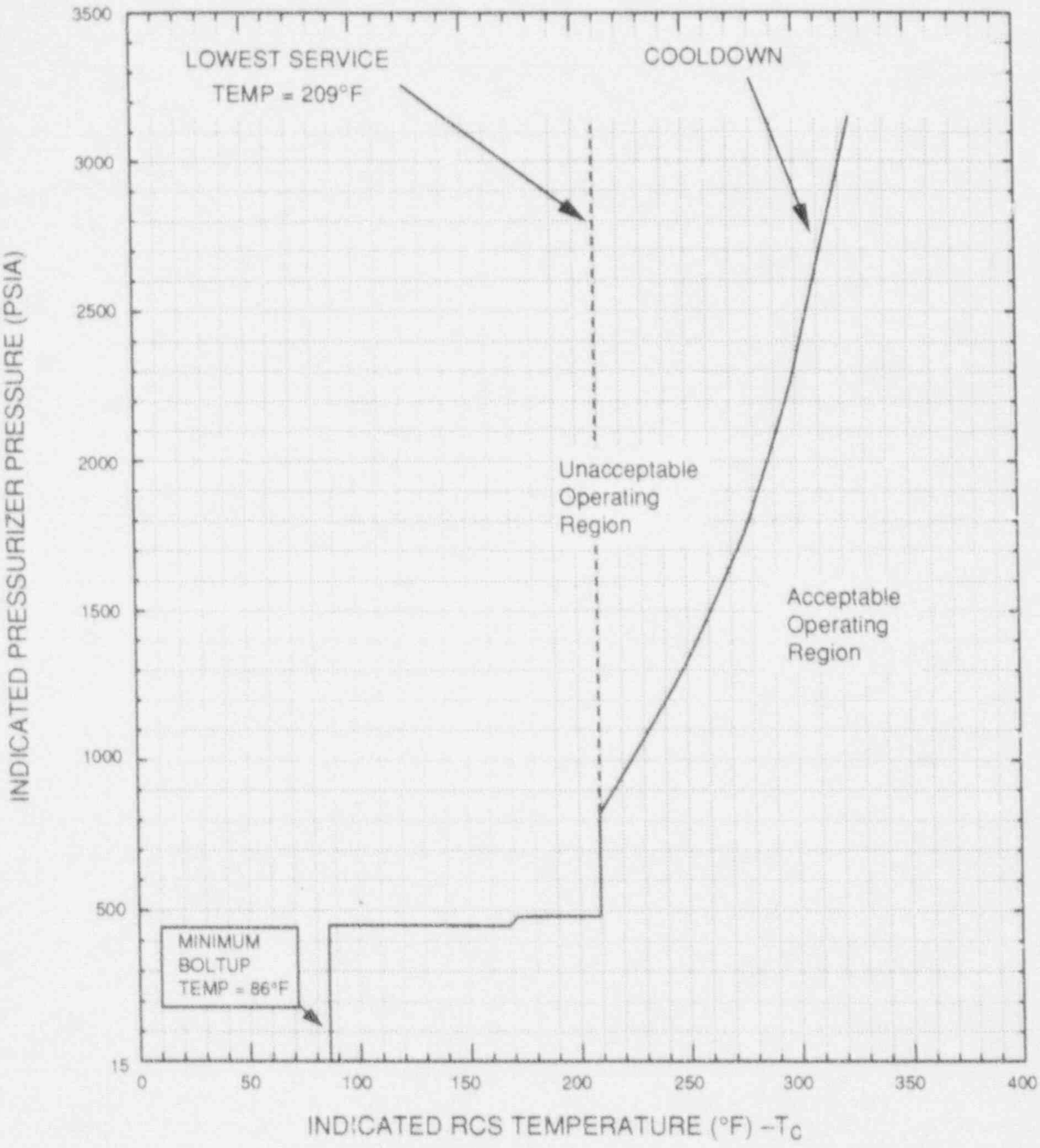
NOTE: A MAXIMUM COOLDOWN RATE OF 100°F/HR IS ALLOWED AT ANY TEMPERATURE ABOVE 160°F

Sup 1

FIGURE 3.4-5

SONGS 2 RCS PRESSURE/TEMPERATURE LIMITS
MAXIMUM ALLOWABLE COOLDOWN RATES (UNTIL 20 EFPY)
Normal Operation

Sup 1

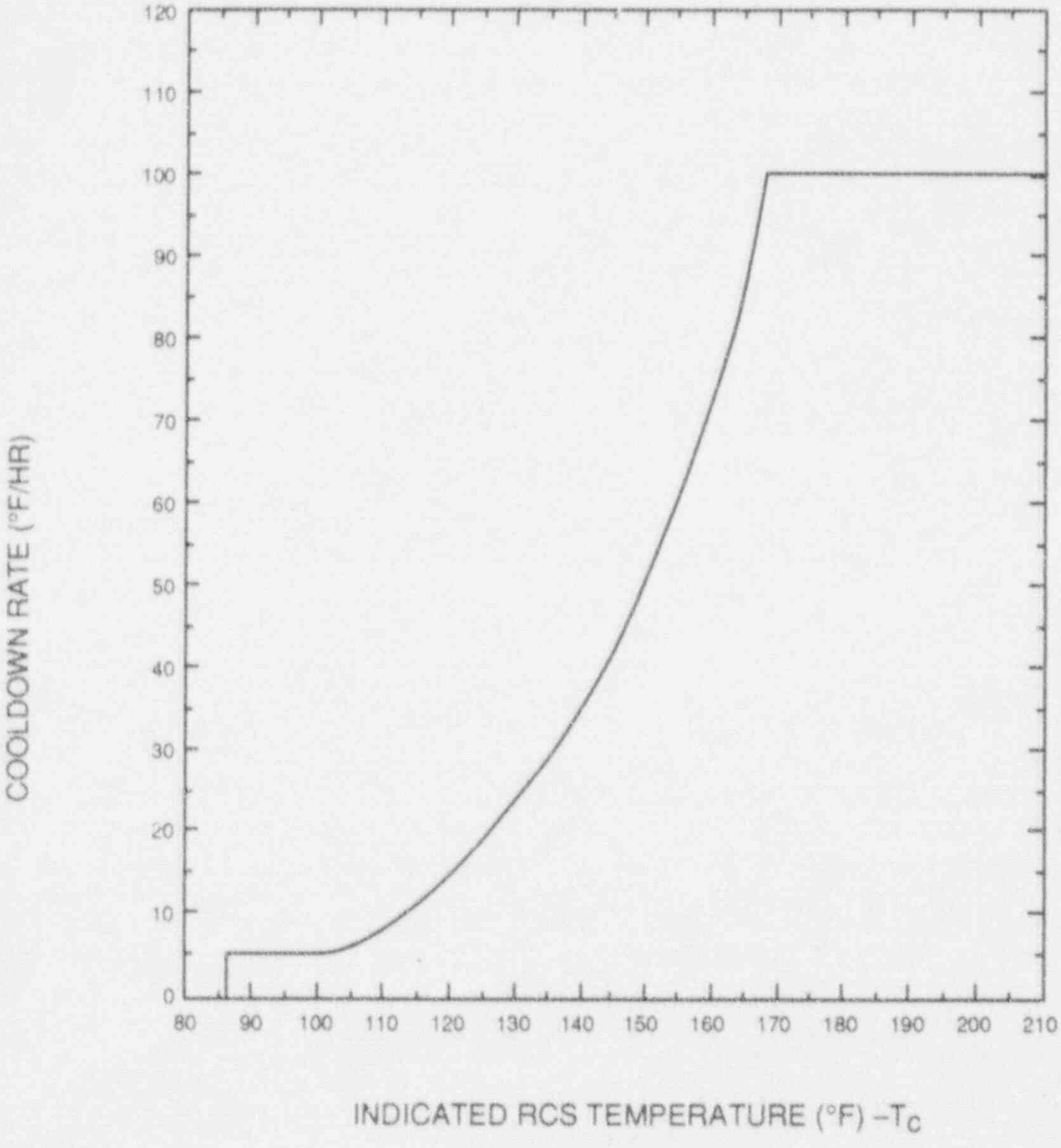


sup 1

FIGURE 3.4-6

SONGS 2 COOLDOWN RCS PRESSURE/TEMPERATURE
LIMITATIONS UNTIL 20 EFPY
Remote Shutdown Operation

sup 1



sup 1

NOTE: A MAXIMUM COOLDOWN RATE OF 100°F/HR IS ALLOWED AT ANY TEMPERATURE ABOVE 168°F

sup 1

FIGURE 3.4-7

SONGS 2 RCS PRESSURE/TEMPERATURE LIMITS
MAXIMUM ALLOWABLE COOLDOWN RATES (UNTIL 20 EFPY)
Remote Shutdown Operation

sup 1

TABLE 3.4-3

Low Temperature RCS Overpressure Protection Range

Operating Period, EFPY	Cold Leg Temperature, °F	
	During Heatup	During Cooldown
~~4 to 10~~ Until 20 (Normal Operation)	≤ ~~312~~ 256	≤ ~~287~~ 238
Until 20 (Remote Shutdown Operation)	*	≤ 238

Sup 1

* Heatup operations are not normally performed from the Remote Shutdown panels.

REACTOR COOLANT SYSTEM

OVERPRESSURE PROTECTION SYSTEMS

RCS TEMPERATURE ≤ ~~312~~ 256°F

|sup'

LIMITING CONDITION FOR OPERATION

3.4.8.3.1 No more than two high-pressure safety injection pumps shall be OPERABLE and 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 ± 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 the enable temperatures specified in Table 3.4-3; MODE 5; and MODE 6 when the head is on the reactor vessel and the RCS is not vented.

ACTION:

a. With the SDCS Relief Valve inoperable, reduce T_{avg} to less than 200°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) or power-lock open the OPERABLE SDCS Relief Valve isolation valve pair within 24 hours, or reduce T_{avg} to less than 200°F, depressurize and vent the RCS through a greater than or equal to 5.6 inch vent within the next 8 hours.

c. With more than two high-pressure safety injection pumps OPERABLE, secure the third high-pressure safety injection pump by racking out its motor circuit breaker or locking close its discharge valve within 8 hours.

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* The lift setting pressure applicable to ~~For~~ valve temperatures of less than or equal to 130°F.

REACTOR COOLANT SYSTEM

OVERPRESSURE PROTECTION SYSTEMS

RCS TEMPERATURE >~~312~~ 256°F Sup 1

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 ± 10 psig*, and

2) Relief Valve isolation valves 2HV9337, 2HV9339, 2HV9377, and 2HV9378 open~~, or,~~ Sup 1

or,

b. A minimum of one pressurizer code safety valve with a lift setting of 2500 psia ± 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

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.~~

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*The lift setting pressure applicable to ~~For~~ valve temperatures of 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.

PRESSURE/TEMPERATURE LIMITS (Continued)

The heatup and cooldown limit curves for normal operation (Figures 3.4-2 and ~~3.4-3~~ 3.4-4) and the cooldown limit curve for remote shutdown operation (Figure 3.4-6) 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 limit curves for Remote Shutdown operation are determined using the Total Loop Uncertainties (TLUs) for temperature and pressure for the Remote Shutdown Panel instruments in which the pressure TLUs are higher than those for the Control Room shutdown instruments. 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~~ instrument uncertainties, and static and dynamic heads.

The reactor vessel materials ~~have been~~ were tested prior to reactor startup to determine their initial RT_{NDT}; the results of these tests and the updates resulting from the evaluation of material properties in response to Generic Letter 92-01, "Reactor Vessel Structural Integrity," Revision 1 are shown in Table B 3/4.4-1. Reactor operation and resultant fast neutron (E greater than ~~1~~ 1 MeV) irradiation will cause an increase in the RT_{NDT}. 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 limit curve (Figure 3.4-2) and the cooldown limit curves, Figures 3.4-~~3~~ 4 and 3.4-6, 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~~ instrument uncertainties, and static and dynamic heads. Sup1

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 and 10 CFR 50 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_{NDT} determined from the surveillance capsule is different from the calculated delta RT_{NDT} 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.

PRESSURE/TEMPERATURE LIMITS (Continued)

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 90°F. The Lowest Service Temperature limit line shown on Figures 3.4-2, 3.4-4, and 3.4-~~3~~ 6 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. sup 1

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 Low Temperature Overpressure Protection (LTOP) enable temperatures are based upon the recommendations of NUREG-0800 Branch Technical Position (BTP) RSB 5-2, Revision 1, "Overpressurization Protection of Pressurized Water Reactors While Operating at Low Temperatures." BTP RSB 5-2, Revision 1 defines the enable temperature as "the water temperature corresponding to a metal temperature of at least RT_{NDT} + 90°F at the beltline location (1/4t or 3/4t) that is controlling in the Appendix G limit calculations."

TABLE B 3/4.4-1

REACTOR VESSEL TOUGHNESS

Piece No.	Code No.	Material	Vessel Location	Drop Weight Results	Temperature of Charpy V-Notch @ 30 ft - lb	@ 50 ft - lb	Minimum Upper Shelf Cv energy for Longitudinal Direction-ft lb	
215-01	C-6403-1	A533GRBCL-1	Upper Shell Plate	40	15	35	130	
215-01	C-6403-2	A533GRBCL-1	Upper Shell Plate	0	20	25	133	
215-01	C-6403-3	A533GRBCL-1	Upper Shell Plate	-10	20	45	131	
215-03	C-6404-1	A533GRBCL-1	Intermed. Shell Plate	-30	~~10~~ 40	~~50~~ 80	~~145~~ 119	
215-03	C-6404-2	A533GRBCL-1	Intermed. Shell Plate	-20	~~20~~ 70	~~50~~ 80	~~155~~ 113	Sup1
215-03	C-6404-3	A533GRBCL-1	Intermed. Shell Plate	-20	~~10~~ 70	~~50~~ 80	~~131~~ 99	
215-02	C-6404-4	A533GRBCL-1	Lower Shell Plate	-10	~~-5~~ -40	~~25~~ 80	~~124~~ 104	
215-02	C-6404-5	A533GRBCL-1	Lower Shell Plate	-20	~~10~~ 50	~~25~~ 70	~~134~~ 118	
215-02	C-6404-6	A533GRBCL-1	Lower Shell Plate	-10	~~-20~~ 50	~~0~~ 50	~~151~~ 124	
238-02	C-6401	A508Cl-2	Vessel Flange Forging	-10	-70	-35	148	
209-02	C-6402	A508Cl-2	Closure Head Flange Forging	-10	-90	-40	142	
205-02	C-6410-1	A508Cl-2	Inlet Nozzle Forging	20	-40	-35	130	
205-02	C-6410-2	A508Cl-2	Inlet Nozzle Forging	0	-20	-5	135	
205-02	C-6410-3	A508Cl-2	Inlet Nozzle Forging	0	-15	-15	140	
205-02	C-6410-4	A508Cl-2	Inlet Nozzle Forging	0	-65	-50	140	
205-06	C-6411-1	A508Cl-2	Outlet Nozzle Forging	-100	-30	-10	140	
205-06	C-6411-2	A508Cl-2	Outlet Nozzle Forging	0	-35	-10	140	Sup1
232-01	C-6424	A533GRBCL-1	Bottom Head Torus	-50	-20	10	122	
232-02	C-6425	A533GRBCL-1	Bottom Head Dome	-50	-30	-20	136	
205-03	C-6428-1	A508CL-1	Inlet Nozzle Forging S/E	-30	-70	-50	174	
205-03	C-6428-2	A508CL-1	Inlet Nozzle Forging S/E	-30	-70	50	174	
205-03	C-6428-3	A508CL-1	Inlet Nozzle Forging S/E	-30	-70	-50	174	
205-03	C-6428-4	A508CL-1	Inlet Nozzle Forging S/E	-30	-70	-50	174	

TABLE B 3/4.4-1 (Continued)

Piece No.	Code No.	Material	Vessel Location	Drop Weight Results	Temperature of Charpy V-Notch @ 30 ft - lb	Temperature of Charpy V-Notch @ 50 ft - lb	Minimum Upper Shelf Cv energy for Longitudinal Direction-ft lb
205-07	C-6429-1	A508CL-1	Outlet Nozzle Ext. Forging	-30	-40	-25	229
205-07	C-6429-1	A508CL-1	Outlet Nozzle Ext. Forging	-30	-40	-25	229
231-02	C-6430-1	A533GRBCL-1	Closure Head Peels	+10	20	55	118
231-02	C-6431-1	A533GRBCL-1	Closure Head Peels	-20	10	50	100
231-02	C-6432-1	A533GRBCL-1	Closure Head Peels	-10	-15	45	115
231-02	C-6432	A533GRBCL-1	Closure Head Dome	-10	-15	45	115

ENCLOSURE 3

TECHNICAL SPECIFICATION PAGES CONTAINING THE CHANGES WHICH WERE PREVIOUSLY REQUESTED IN AMENDMENT APPLICATION NO. 117 (PCN-354) DATED SEPTEMBER 3, 1992, AND ARE BEING REQUESTED IN THIS LICENSE AMENDMENT APPLICATION NO. 118 (PCN-335) | Sup 1

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PCN 354

REACTOR COOLANT SYSTEM

SURVEILLANCE REQUIREMENTS ~~(Continued)~~ PCN 335 sup 1

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~~ as 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~~ 3.4-4 through 3.4-7. Recalculate the Adjusted Reference Temperature ~~based on the greater of the following:~~ in accordance with Regulatory Guide 1.99, Revision 2, "Radiation Embrittlement of Reactor Vessel Materials," May 1988. PCN 354 PCN 335

a. ~~The mean value of shift in reference temperature for plates C-6403-3*, 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.~~

~~*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.~~ PCN 335

PRESSURE/TEMPERATURE LIMITS (Continued)

The heatup and cooldown limit curves for normal operation (Figures 3.4-2 and ~~3.4-3~~ 3.4-4) and the cooldown limit curve for remote shutdown operation (Figure 3.4-6) 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 limit curves for Remote Shutdown operation are determined using the Total Loop Uncertainties (TLUs) for temperature and pressure for the Remote Shutdown Panel instruments in which the pressure TLUs are higher than those for the Control Room shutdown instruments. 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~~ instrument uncertainties, and static and dynamic heads. PCN 335 PCN 335 PCN 335

The reactor vessel materials ~~have been~~ were tested prior to reactor startup to determine their initial RT_{NDT}; the results of these tests and the updates resulting from the evaluation of material properties in response to Generic Letter 92-01, "Reactor Vessel Structural Integrity," Revision 1 are shown in Table B 3/4.4-1. Reactor operation and resultant fast neutron (E greater than ~~1~~ 1 MeV) irradiation will cause an increase in the RT_{NDT}. 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 limit curve (Figure 3.4-2) and the cooldown limit curves, Figures 3.4-~~3~~ 4 and 3.4-6, 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~~ instrument uncertainties, and static and dynamic heads. PCN 335 Sup 1 PCN 335 PCN 335 PCN 335

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 and 10 CFR 50 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~~ maintained in the FSAR. 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_{NDT} determined from the surveillance capsule is different from the calculated delta RT_{NDT} for the equivalent capsule radiation exposure. PCN 354 PCN 335

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. PCN 335