NPF-38-148

ATTACHMENT A

LIST OF TABLES (Continued)

HOT 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 and/or shutdown cooling loops OPERABLE, immediately initiate corrective action to return the required loops to OPERABLE status as soon as possible; if the remaining OPERABLE loop is a shutdown cooling loop, be in COLD SHUTDOWN within 24 hours.

b. With no reactor coolant or shutdown cooling loop 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 to operation.

*All reactor coolant pumps and shutdown cooling pumps (LPSI pumps) may be deenergized 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 285°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 At least two of the loop(s)/trains listed below shall be OPERABLE and at least one reactor coolant and/or shutdown cooling loop 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 5 with reactor coolant loops filled**.

ACTION:

a. With less than the above required reactor coolant and/or shutdown cooling loops OPERABLE or with less than the required steam generator level, immediately initiate corrective action to return the required loops to OPERABLE status or to restore the required level as soon as possible.

b. With no reactor coolant or shutdown cooling loop 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 to operation.

SURVEILLANCE REQUIREMENTS

4.4.1.4.1 The required reactor coolant pump(s), if not in operation, shall be determined to be OPERABLE once per 7 days by verifying correct breaker alignments and indicated power availability.

4.4.1.4.2 The required steam generator(s) shall be determined OPERABLE by verifying the secondary side water level to be $\geq$50% of wide range indication at least once per 12 hours.

4.4.1.4.3 At least one reactor coolant loop or shutdown cooling train shall be verified to be in operation and circulating reactor coolant at least once per 12 hours.

*All reactor coolant pumps and shutdown cooling pumps (LPSI pumps) may be deenergized 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 285°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.

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 Figures 3.4-2 and 3.4-3 during heatup, cooldown, criticality, and inservice leak and hydrostatic testing with:

a. A maximum heatup rate of 30°F per hour with Reactor Coolant System cold leg temperature less than 200°F.

b. A maximum heatup rate of 50°F per hour with Reactor Coolant System cold leg temperature greater than 200°F and less than or equal to 345°F.

c. A maximum heatup rate of 60°F per hour with Reactor Coolant System cold leg temperature greater than 345°F.

d. A maximum cooldown rate of 10°F per hour with Reactor Coolant System cold leg temperature less than 135°F.

e. A maximum cooldown rate of 30°F per hour with Reactor Coolant System cold leg temperature greater than or equal to 135°F and less than or equal to 200°F.

f. A maximum cooldown rate of 100°F per hour with Reactor Coolant System cold leg temperature greater than 200°F.

g. A maximum temperature change of less than or equal to 10°F in any 1-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_{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.

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 Part 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. The adjusted reference temperature resulting from neutron irradiation shall be calculated based on the greater of the following:

a. Actual shift in the RT_{NDT} as measured by impact testing of 88114/0145 weld metal;

b. Predicted shift in RT_{NDT} for E8018/BOCA weld metal as determined by Regulatory Guide 1.99, "Effects of Residual Elements on Predicted Radiation Damage to Reactor Vessel Materials."

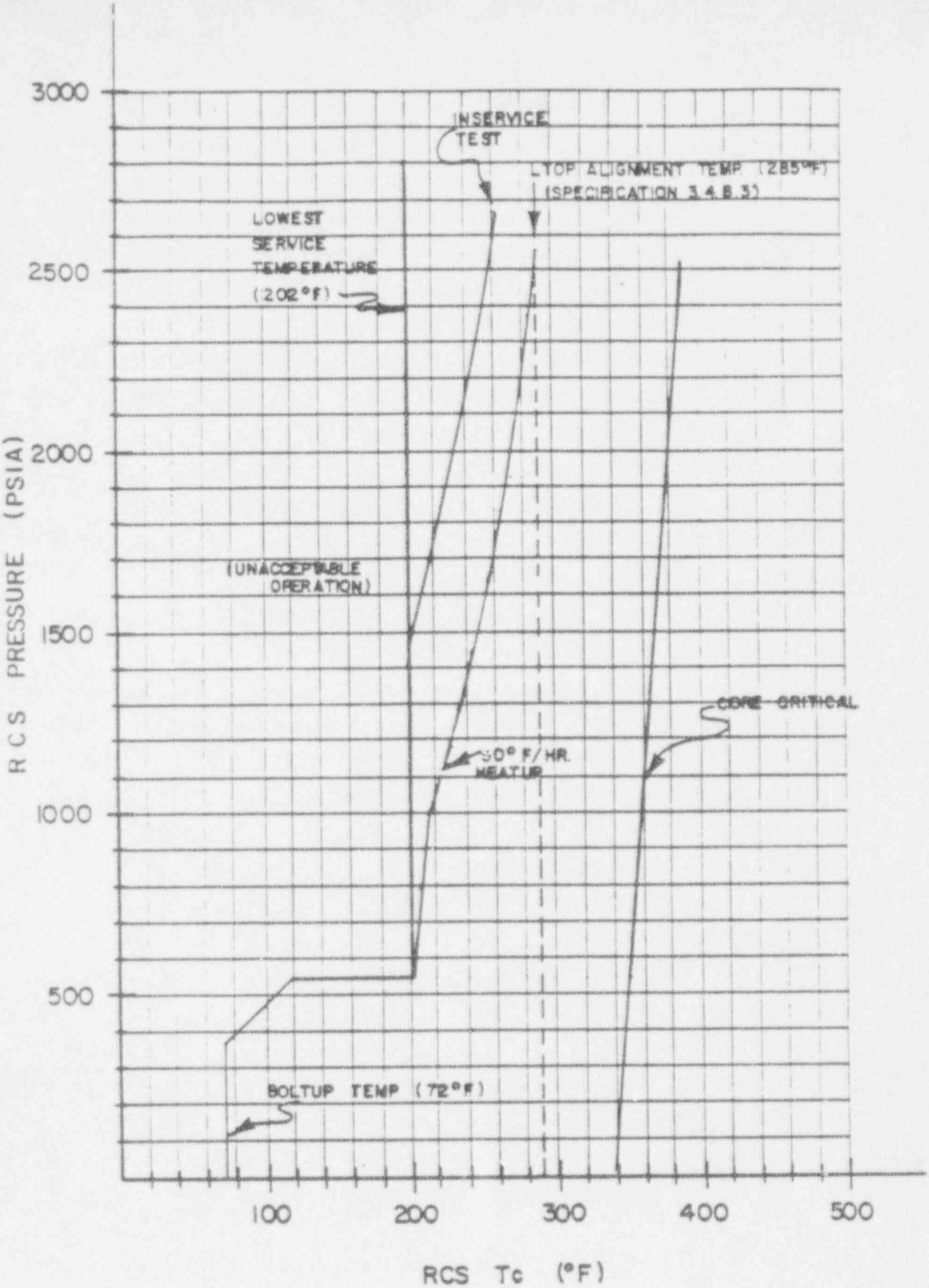


FIGURE 3.4-2

REACTOR COOLANT SYSTEM PRESSURE/TEMPERATURE LIMITATIONS FOR 0-8 EFFECTIVE FULL POWER YEARS (HEATUP)

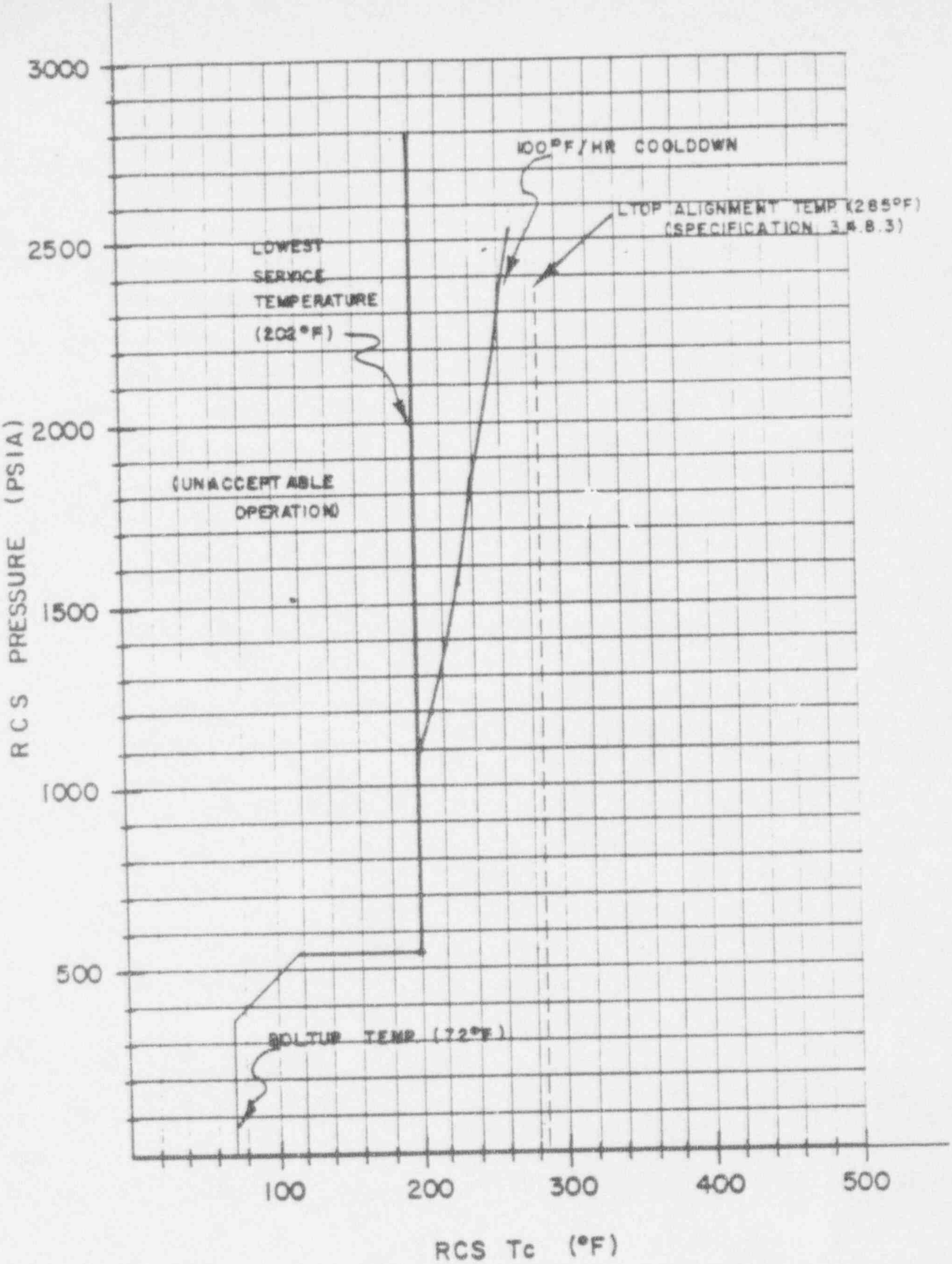


FIGURE 3.4-3

REACTOR COOLANT SYSTEM PRESSURE/TEMPERATURE LIMITATIONS
FOR 0-8 EFFECTIVE FULL POWER YEARS (COOLDOWN)

TABLE 4.4-5

REACTOR VESSEL MATERIAL SURVEILLANCE PROGRAM - WITHDRAWAL SCHEDULE

CAPSULE NUMBER	VESSEL LOCATION	LEAD FACTOR	WITHDRAWAL TIME (EFPY)*
1	83°	1.50	Standby
2	97°	1.50	4.0 EFPY
3	104°	1.50	11.0 EFPY
6	284°	1.50	18.0 EFPY
4	263°	1.50	Standby
5	277°	1.50	Standby

*Withdrawal time may be modified to coincide with those refueling outages or plant shutdowns most closely approaching the withdrawal schedule.

OVERPRESSURE PROTECTION SYSTEMS

LIMITING CONDITION FOR OPERATION

3.4.8.3 Two Shutdown Cooling (SDC) System suction line relief valves (SI-406A and SI-406B) shall be OPERABLE with a lift setting of less than or equal to 430 psia.

APPLICABILITY: MODE 4 when the temperature of any RCS cold leg is less than or equal to 285°F#, MODE 5, and MODE 6 when the head is on the reactor vessel and the RCS is not vented through a 5.6 square inch or larger vent.

ACTION:

a. With one SDC System suction line relief valve inoperable in MODE 4, restore the inoperable valve to OPERABLE status within 7 days, or depressurize and vent the RCS through at least a 5.6 square inch vent within the next 8 hours.

b. With one SDC System suction line relief valve inoperable in MODES 5, or 6, either (1) restore the inoperable valve to OPERABLE status within 24 hours, or (2) complete depressurization and venting of the RCS through at least a 5.6 square inch vent within a total of 32 hours.

c. With both SDC System suction line relief valves inoperable, complete depressurization and venting of the RCS through at least a 5.6 square inch vent within 8 hours.

d. In the event either the SDC System suction line relief valve(s) 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 SDC System suction line relief valve(s) or RCS vent(s) on the transient, and any corrective action necessary to prevent recurrence.

e. The provisions of Specification 3.0.4 are not applicable.

#260°F during inservice leak and hydrostatic testing with Reactor Coolant System temperature changes restricted in accordance with Specification 3.4.8.1g.

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 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 Figures 3.4-2 and 3.4-3. The limitations on the Reactor Coolant System heatup and cooldown rates are further restricted due to stress limitations in the Reactor Coolant Pump. As part of the LOCA support scheme, the Reactor Coolant Pump has a ring around the suction nozzle of the pump. The support skirt is welded to the ring. Due to this design, the heatup and cooldown rates must be limited to maintain acceptable thermal stresses.

The reactor vessel materials have been tested to determine their initial RT_{NDT}; 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_{NDT}. Therefore, an adjusted reference temperature, based upon the fluence and copper content of the material in question, can be predicted using FSAR Table 5.3-1 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_{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 E185-82 and 10 CFR Part 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. 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 Figures 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 Part 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 and 3.4-3 is based upon this RT_{NDT} since Article NB-2332 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 (as corrected for elevation and instrument error).

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 the shutdown cooling system relief valve or an 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 285°F. Each shutdown cooling system relief valve has adequate relieving capability to protect the RCS from overpressurization when the transient is 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 injection into a water-solid RCS. The limiting transient includes simultaneous, inadvertent operation of three HPSI pumps, three charging pumps, and all pressurizer backup heaters in operation. Since SIAS starts only two HPSI pumps, a 20% margin is realized.

The restrictions on starting a reactor coolant pump in MODE 4 and with the reactor coolant loops filled in MODE 5, with one or more RCS cold legs less than or equal to 285°F, are provided in Specification 3.4.1.3 and 3.4.1.4 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 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. Maintaining the steam generator less than 100°F above each of the Reactor Coolant System cold leg temperatures (even with the RCS filled solid) or maintaining a large surge volume in the pressurizer ensures that this transient is less severe than the limiting transient considered above.

The automatic isolation setpoint of the shutdown cooling isolation valves is sufficiently high to preclude inadvertent isolation of the shutdown cooling relief valves during a pressure transient.

NPF-38-148

ATTACHMENT B

LIST OF TABLES (Continued)

DELETE

HOT 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 and/or shutdown cooling loops OPERABLE, immediately initiate corrective action to return the required loops to OPERABLE status as soon as possible; if the remaining OPERABLE loop is a shutdown cooling loop, be in COLD SHUTDOWN within 24 hours.

b. With no reactor coolant or shutdown cooling loop 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 to operation.

*All reactor coolant pumps and shutdown cooling pumps (LPSI pumps) may be deenergized 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 10F 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 ~~285°F~~ 272°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.

COLD SHUTDOWN - LOOPS FILLED

LIMITING CONDITION FOR OPERATION

3.4.1.4 At least two of the loop(s)/trains listed below shall be OPERABLE and at least one reactor coolant and/or shutdown cooling loop 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 5 with reactor coolant loops filled**.

ACTION:

a. With less than the above required reactor coolant and/or shutdown cooling loops OPERABLE or with less than the required steam generator level, immediately initiate corrective action to return the required loops to OPERABLE status or to restore the required level as soon as possible.

b. With no reactor coolant or shutdown cooling loop 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 to operation.

SURVEILLANCE REQUIREMENTS

4.4.1.4.1 The required reactor coolant pump(s), if not in operation, shall be determined to be OPERABLE once per 7 days by verifying correct breaker alignments and indicated power availability.

4.4.1.4.2 The required steam generator(s) shall be determined OPERABLE by verifying the secondary side water level to be > 50% of wide range indication at least once per 12 hours.

4.4.1.4.3 At least one reactor coolant loop or shutdown cooling train shall be verified to be in operation and circulating reactor coolant at least once per 12 hours.

---

*All reactor coolant pumps and shutdown cooling pumps (LPSI pumps) may be deenergized 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 10F 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 ~~285°F~~ 272°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.

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 Figures 3.4-2 and 3.4-3 during heatup, cooldown, criticality, and inservice leak and hydrostatic testing with:

a. A maximum heatup rate of 30°F per hour with Reactor Coolant System cold leg temperature less than 200°F.

b. A maximum heatup rate of 50°F per hour with Reactor Coolant System cold leg temperature greater than 200°F and less than or equal to 345°F.

c. A maximum heatup rate of 60°F per hour with Reactor Coolant System cold leg temperature greater than 345°F.

d. A maximum cooldown rate of 10°F per hour with Reactor Coolant System cold leg temperature less than 135°F.

e. A maximum cooldown rate of 30°F per hour with Reactor Coolant System cold leg temperature greater than or equal to 135°F and less than or equal to 200°F.

f. A maximum cooldown rate of 100°F per hour with Reactor Coolant System cold leg temperature greater than 200°F.

g. A maximum temperature change of less than or equal to 10°F in any 1-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_{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.

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 Part 50 Appendix H in accordance with the

SURVEILLANCE REQUIREMENTS (Continued)

Reactor Vessel material surveillance program - withdrawal schedule in ~~Table 4.4-5~~ FSAR Table 5.3-10. The results of these examinations shall be used to update Figures 3.4-2 and 3.4-3. ~~The adjusted reference temperature resulting from neutron irradiation shall be calculated based on the greater of the following:~~

~~a. Actual shift in the RT_{NDT} as measured by impact testing of 88114/0145 weld metal;~~

~~b. Predicted shift in RT_{NDT} for E8018/BOCA weld metal as determined by Regulatory Guide 1.99, "Effects of Residual Elements on Predicted Radiation Damage to Reactor Vessel Materials."~~

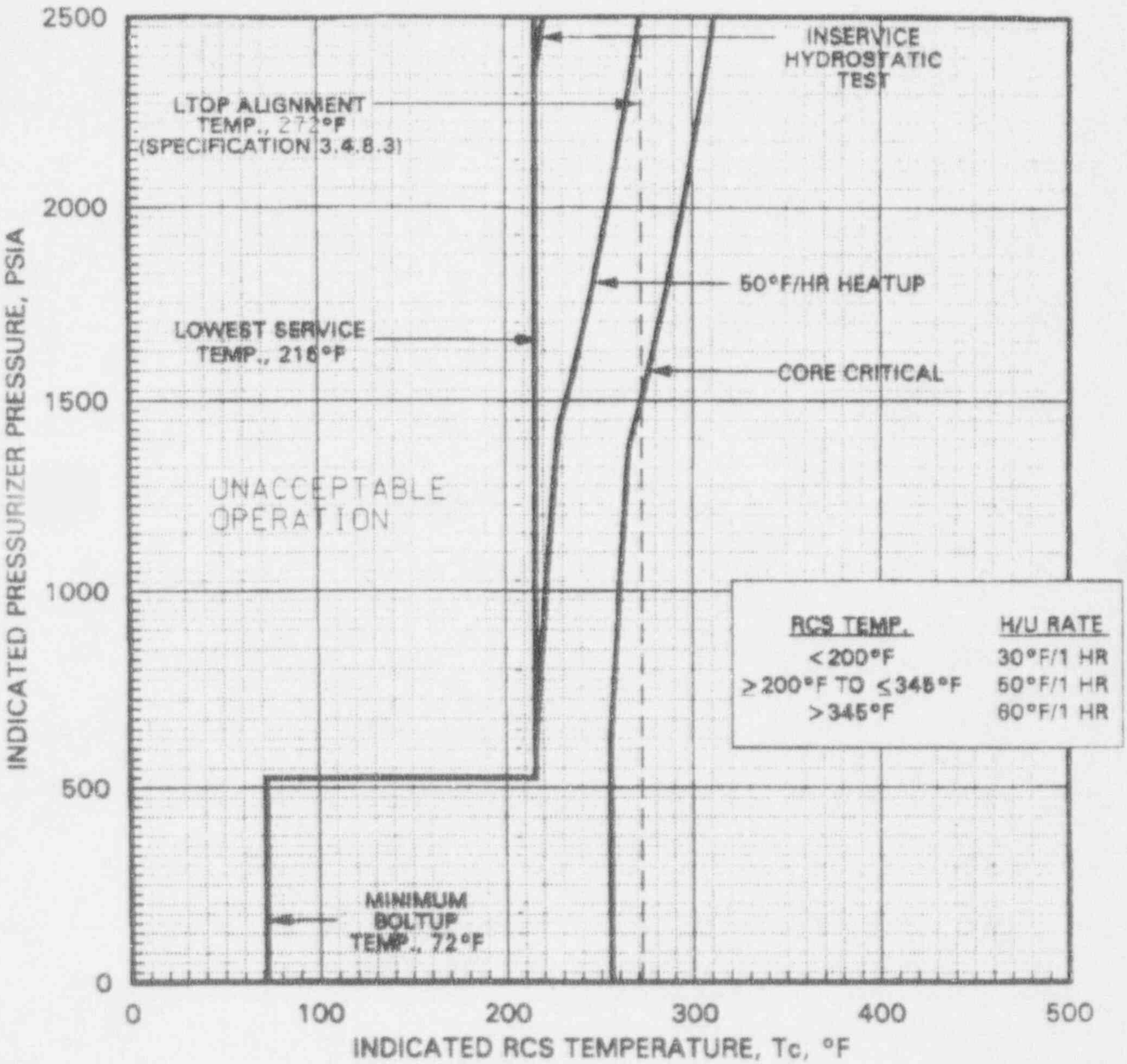


FIGURE 3.4-2

WATERFORD UNIT 3 HEATUP CURVE
REACTOR COOLANT SYSTEM PRESSURE-TEMPERATURE LIMITS
0 - 20 EFPY (PEAK SURFACE FLUENCE = 2.29×10^{19} n/cm^2)

FILENAME: tecspcl.dgn

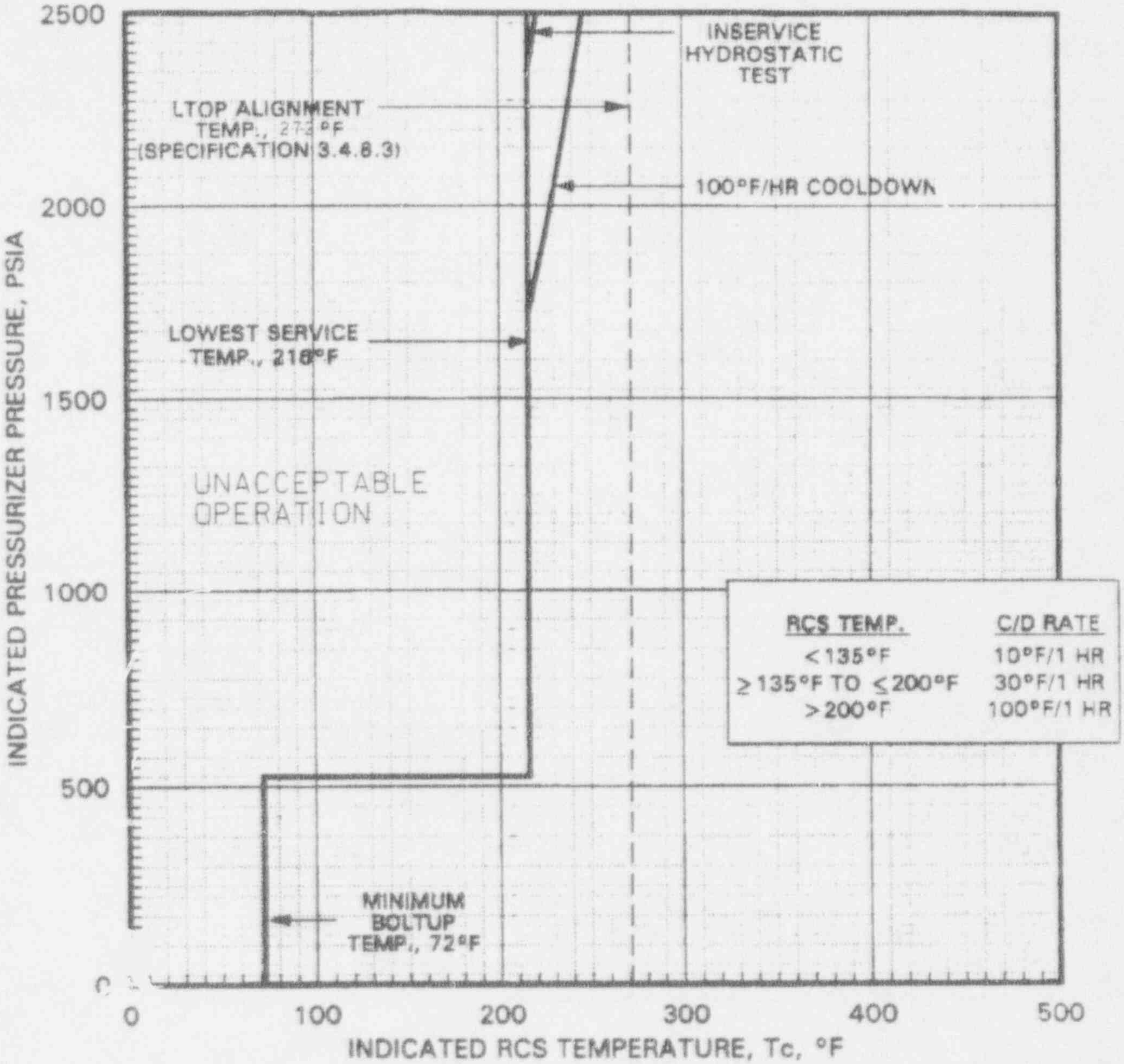


FIGURE 3.4-3

WATERFORD UNIT 3 COOLDOWN CURVE

REACTOR COOLANT SYSTEM PRESSURE-TEMPERATURE LIMITS

0 - 20 EFPY (PEAK SURFACE FLUENCE = 2.29 x 10^{19} n/cm^2)

FILENAME: tecspc2.dgn

~~TABLE 4.4-5~~

~~REACTOR VESSEL MATERIAL SURVEILLANCE PROGRAM - WITHDRAWAL SCHEDULE~~

~~CAPSULE NUMBER~~	~~VESSEL LOCATION~~	~~LEAD FACTOR~~	~~WITHDRAWAL TIME (EFPY) *~~
~~1~~	~~83º~~	~~1.50~~	~~Standby~~
~~2~~	~~97~~	~~1.50~~	~~4.0 EFPY~~
~~3~~	~~104~~	~~1.50~~	~~11.0 EFPY~~
~~6~~	~~284~~	~~1.50~~	~~18.0 EFPY~~
4	~~263~~	~~1.50~~	~~Standby~~
~~5~~	~~277~~	~~1.50~~	~~Standby~~

TABLE 4.4-5

This TABLE has been deleted.

~~*Withdrawal time may be modified to coincide with those refueling outages or plant shutdowns most closely approaching the withdrawal schedule.~~

LIMITING CONDITION FOR OPERATION

3.4.8.3 Two Shutdown Cooling (SDC) System suction line relief valves (SI-406A and SI-406B) shall be OPERABLE with a lift setting of less than or equal to 430 psia.

APPLICABILITY: MODE 4 when the temperature of any RCS cold leg is less than or equal to ~~285~~ 272°F#, MODE 5, and MODE 6 when the head is on the reactor vessel and the RCS is not vented through a 5.6 square inch or larger vent.

ACTION:

a. With one SDC System suction line relief valve inoperable in MODE 4, restore the inoperable valve to OPERABLE status within 7 days, or depressurize and vent the RCS through at least 5.6 square inch vent within the next 8 hours.

b. With one SDC System suction line relief valve inoperable in MODES 5, or 6, either (1) restore the inoperable valve to OPERABLE status within 24 hours, or (2) complete depressurization and venting of the RCS through at least 5.6 square inch vent within a total of 32 hours.

c. With both SDC System suction line relief valves inoperable, complete depressurization and venting of the RCS through at least 5.6 square inch vent within 8 hours.

d. In the event either the SDC System suction line relief valve(s) 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 SDC System suction line relief valve(s) or RCS vent(s) on the transient, and any corrective action necessary to prevent recurrence.

e. The provisions of Specification 3.0.4 are not applicable.

260F during inservice leak and hydrostatic testing with Reactor Coolant System temperature changes restricted in accordance with Specification 3.4.8.1g.

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 per hour or cooldown rate ~~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 Figures 3.4-2 and 3.4-3. The limitations on the Reactor Coolant System heatup and cooldown rates are further restricted due to stress limitations in the Reactor Coolant Pump. As part of the LOCA support scheme, the Reactor Coolant Pump has a ring around the suction nozzle of the pump. The support skirt is welded to the ring. Due to this design, the heatup and cooldown rates must be limited to maintain acceptable thermal stresses.

The reactor vessel materials have been tested to determine their initial RT_{NDT}; 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_{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 5.3-1 and the recommendations of Regulatory Guide 1.99, Revision ~~1, "Effects of Residual Elements on Predicted Radiation Damage to Reactor Vessel Materials."~~ 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 Part 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 FSAR Table ~~4.4-5~~ 5.3-10. 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 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 Figures 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 Part 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 and 3.4-3 is based upon this RT_{NDT} since Article NB-2332 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 (as corrected for elevation and instrument error).

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 the shutdown cooling system relief valve or an 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 ~~285~~ 272 °F. Each shutdown cooling system relief valve has adequate relieving capability to protect the RCS from overpressurization when the transient is 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 injection into a water-solid RCS. The limiting transient includes simultaneous, inadvertent operation of three HPSI pumps, three charging pumps, and all pressurizer backup heaters in operation. Since SIAS starts only two HPSI pumps, a 20% margin is realized.

The restrictions on starting a reactor coolant pump in MODES 4 and with the reactor coolant loops filled in MODE 5, with one or more RCS cold legs less then or equal to ~~285~~ 272 °F, are provided in Specification 3.4.1.3 and 3.4.1.4 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 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. Maintaining the steam generator less than 100°F above each of the Reactor Coolant System cold leg temperatures (even with the RCS filled solid) or maintaining a large surge volume in the pressurizer ensures that this transient is less severe than the limiting transient considered above.

~~The automatic isolation setpoint of the shutdown cooling isolation valves is sufficiently high to preclude inadvertent isolation of the shutdown cooling relief valves during a pressure transient.~~

NPF-38-148

ATTACHMENT C

Final Report

Development of Reactor Coolant System Pressure Temperature

Limits for 20 Effective Full Power Years (Fluence = $2.29 \times 10^{19} n/cm^2$)

and Recommended Surveillance Withdrawal Schedule for

Waterford Unit 3

DEVELOPMENT OF

REACTOR COOLANT SYSTEM

PRESSURE TEMPERATURE LIMITS

FOR

20 EFFECTIVE FULL POWER YEARS

(FLUENCE = 2.29×10^{19} n/cm^2)

AND

RECOMMENDED

SURVEILLANCE WITHDRAWAL SCHEDULE

FOR

WATERFORD UNIT 3

Prepared For:

ENTERGY OPERATIONS, INC.
WATERFORD STEAM ELECTRIC STATION UNIT 3
KILLONA, LOUISIANA 70066-0751

By:

ABB COMBUSTION ENGINEERING NUCLEAR SERVICES
COMBUSTION ENGINEERING, INC.
REACTOR VESSEL INTEGRITY GROUP
1000 PROSPECT HILL ROAD
WINDSOR, CONNECTICUT 06095-0500

OCTOBER, 1993

FINAL REPORT

C-MECH-ER-021, REV. 00

EXECUTIVE SUMMARY

This report provides reactor coolant system pressure-temperature limits valid through twenty (20) effective full power years' (EFPY) operation for Waterford Steam Electric Station Unit 3 (Waterford Unit 3). The reactor coolant system pressure-temperature limits currently provided by the Waterford Unit 3 Technical Specifications are valid through 8 EFPY and have been revised to account for additional embrittlement of the reactor vessel beltline materials due to continued neutron irradiation. The change in the reactor vessel beltline fracture toughness properties have been estimated in accordance with Regulatory Guide 1.99 Revision 2 using a projected peak surface fluence of 2.29×10^{19} n/cm^2. This fluence was based on the results obtained from the W-97 surveillance capsule evaluation report. The limiting material was identified as plate M-1004-2 having adjusted reference temperatures at the 1/4t and 3/4t locations of 65.4°F and 54.0°F, respectively. The reactor coolant system pressure-temperature limits, including the revised beltline analysis, have been developed in accordance with 10 CFR 50 Appendix G requirements.

10 CFR 50 Appendix G and 10 CFR §50.61 requirements for end-of-life (32 EFPY) fracture toughness properties were reviewed and shown to be in compliance. Specifically, all beltline materials will have greater than 50 ft-lbs upper-shelf energy projected at the 1/4t locations given an end-of-life peak surface fluence of 3.69×10^{19} n/cm^2. The values of RT_{PTS} for all beltline materials are well below the screening criteria provided by 10 CFR §50.61.

Low Temperature Overpressure Protection (LTOP) Enable temperatures were developed using the guidance of Standard Review Plan 5.2.2 and ASME Boiler and Pressure Vessel Code, Section XI, Code Case N-514 (not yet approved for use by the NRC).

Lastly, a revised surveillance capsule withdrawal schedule has been developed based on consideration of the W-97 surveillance capsule results. This has been developed in accordance with the regulatory requirements of 10 CFR 50 Appendix H. Consideration has been given to the future ASTM E-185-93.

TABLE OF CONTENTS

LIST OF TABLES

LIST OF FIGURES

1.0 INTRODUCTION

Reactor coolant system (RCS) pressure-temperature (P-T) limits provide protection against nonductile failure for ferritic pressure boundary components during normal operation heatup, cooldown and inservice testing. These limits are required to be developed to meet the requirements of 10 CFR 50, Appendix G (Reference 1) and must be updated periodically to account for continued embrittlement of the reactor vessel beltline due to neutron irradiation. The rate of embrittlement is monitored by the withdrawal of surveillance capsules and mechanical testing of material specimens within the capsules and dosimetry evaluations. 10 CFR 50, Appendix H (Reference 2) provides the requirements for the reactor vessel material surveillance program.

The first capsule (W-97) was recently removed from the Waterford Steam Electric Station Unit 3 (Waterford Unit 3). Upon removal and completion of material testing and dosimetry calculations, a report (Report No. BAW-2177, Babcock & Wilcox Nuclear Services (Reference 3)) summarizing the results was provided to the NRC by Entergy Operations, Inc. (under Reference 4 cover). Along with the summary was a commitment to update the Waterford Unit 3 RCS P-T limits and review the adequacy of the surveillance withdrawal schedule, both contained in the Waterford Unit 3 Technical Specifications (Reference 5).

This report provides revised RCS P-T limits for Waterford Unit 3 valid through twenty (20) effective full power years (EFPY) operation, along with a description of the data and methods associated with their development. In addition, a revised surveillance withdrawal schedule has been provided along with the basis for its development.

2.0 RCS PRESSURE-TEMPERATURE LIMITS

2.1 OVERVIEW

The basis for the development of reactor coolant system pressure-temperature limitations for the Waterford Unit 3 Steam Electric Station is provided in the subsequent subsections. RCS P-T limits were developed to meet the requirements associated with 10 CFR Part 50 Appendix A (Ref. 6), Design Criterion 14 and Design Criterion 31. These design criteria require that the reactor coolant pressure boundary be designed, fabricated, erected, and tested in order to have an extremely low probability of abnormal leakage, rapid failure, and gross rupture. The criteria also require that the reactor coolant pressure boundary be designed with sufficient margin to assure that when stressed under operating, maintenance, and testing the boundary behaves in a non-brittle manner and the probability of rapidly propagating fracture is minimized.

Specific requirements regarding the development of pressure-temperature limits are provided by 10 CFR 50 Appendix G (Ref. 1) which forms the general basis for these limitations. 10 CFR Part 50 Appendix G mandates margins of safety against fracture equivalent to those recommended in the ASME Boiler and Pressure Vessel Code, Section III, Appendix G, Protection Against Nonductile Failure (Ref. 7, referred to herein as ASME Code, Appendix G). The general guidance provided by the ASME Code, Appendix G procedures has been utilized to develop the revised Waterford Unit 3 reactor vessel beltline pressure-temperature limits with the requisite margins of safety for the heatup, cooldown and inservice hydro test conditions.

Within the RCS, the reactor vessel beltline region is subjected to sufficient neutron irradiation to alter the material fracture toughness properties of the beltline material. The effects of neutron irradiation has been included in the development of the beltline pressure-temperature limits and based on the irradiation damage prediction methods of Regulatory Guide 1.99 Revision 02 (Ref. 8). This methodology has been used to calculate the limiting material Adjusted Reference Temperatures for Waterford Unit 3 and has utilized peak I.D. surface fluence value of 2.29×10^{19} n/cm^2 corresponding to approximately 20 Effective Full Power Years (EFPY) of operation.

The following sections describe the methodology associated with the development of the reactor vessel beltline pressure-temperature limits. The transients analyzed were the isothermal condition, linear heatup rates of 30, 50, and 60°F/hr, and linear cooldown rates of 10, 30, and 100°F/hr. The beltline limits for inservice hydrostatic test corresponding to an isothermal condition were also developed.

In addition to the beltline P-T limits, the additional requirements of 10 CFR 50 Appendix G and ASME Code have been developed to provide P-T limits appropriate for the RCS.

2.2 BASIC DATA

<u>Reactor Vessel Data</u>			Reference
Design Pressure	=	2500 psia	9
Design Temperature	=	650°F	9
Operating Pressure	=	2250 psia	9
Beltline Thickness	=	8.625 in	9
Inside Radius (to wetted surface)	=	86.971 in	9
Cladding Thickness	=	0.21875 in	9

<u>Material-SA 533 Grade B Class 1</u>			Reference
Thermal Conductivity	=	23.8 BTU/hr-ft-°F	10
Youngs Modulus	=	28×10^6 psi	10
Coefficient of Thermal Expansion	=	7.77×10^{-6}in/in/°F	10
Specific Heat	=	0.122 BTU/lb-°F	
Density	=	490 lbm/ft^3	11

<u>Stainless Steel Cladding</u>			
Thermal Conductivity	=	10.1 BTU/hr-ft-°F	10
Specific Heat	=	0.126 BTU/lb-°F	
Density	=	493 lbm/ft^3	11

Convective film coefficient on vessel inside surface = 1000 BTU/hr-ft^2-°F

2.3 ADJUSTED REFERENCE TEMPERATURE PROJECTIONS

To permit the development of revised beltline P-T limits, it was necessary to calculate the adjusted reference temperatures (ARTs) for the Waterford Unit 3 beltline materials to establish the limiting material. The adjusted reference temperatures of each reactor vessel beltline material were calculated at the postulated crack tip inherent with the beltline P-T limit analysis (1/4t and 3/4t) locations after 20 EFPY of operation. The controlling material for Waterford Unit 3 was determined by comparing ART values for each material.

The adjusted reference temperatures (ART) were calculated using the procedures in Regulatory Position 1.1 of Regulatory Guide 1.99 Revision 2 (Ref. 8). The procedure for calculating ART values for a material in the beltline is given by the following expression:

$$ART = \text{Initial } RT_{NDT} + \Delta RT_{NDT} + \text{Margin} \qquad (1)$$

The initial RT_{NDT} is the reference temperature for the unirradiated material. The ΔRT_{NDT} is the mean value of the adjustment in the reference temperature caused by irradiation and is given by the following expression:

$$\Delta RT_{NDT} = (CF)\ f^{(0.28 - 0.10 \log f)} \qquad (2)$$

CF is the chemistry factor for the beltline materials which is a function of the weight percent copper and nickel for the material. Regulatory Guide 1.99 Revision 2 provides chemistry factors for welds and for base metal plates and forgings. The term f is the neutron fluence at any depth in the vessel. The neutron fluence at any depth is given by the following expression:

$$f = f_{surf}\ e^{(-0.24x)} \qquad (3)$$

The term f_{surf} is the neutron fluence (having units of 10^{19} n/cm^2 E > 1MeV) at the inner wetted surface of the vessel, and x is the depth into the vessel wall from the

inner wetted surface (inches). In this instance, the reactor vessel stainless steel cladding is ignored.

Margin is the quantity that is added to obtain a conservative upper bound value of ART. The margin term is given by the following expression:

$$\text{Margin} = 2(\sigma_I^2 + \sigma_\Delta^2)^{1/2} \qquad (4)$$

The terms σ_I and σ_Δ represent the standard deviations associated with the initial RT_{NDT} and the mean value for the reference temperature shift, respectively.

The following information provides the basis for the calculated ART values associated with Waterford Unit 3 reactor vessel beltline:

1) Unirradiated beltline material data was obtained from Tables 5.2-6 and 5.2-13 of Reference 12, including copper content, nickel content and initial reference temperature (RT_{NDT}). This data is summarized in Table 1 for Waterford Unit 3. It should be noted that in the process of establishing Charpy upper-shelf energy values for the Waterford Unit 3 reactor vessel beltline welds, an inconsistency was identified for longitudinal seam 101-142B. A weld repair was identified which identified a different weld wire heat and flux, along with a different initial RT_{NDT} and nickel content (w%). This information was considered in the determination of the limiting beltline material. However, the determination as to the extent and location of the weld repair was not within the scope of this effort. The results of this inconsistency have had no bearing on the results of this evaluation.

2) Peak neutron fluence for the Waterford Unit 3 beltline region was determined to be 2.29×10^{19} n/cm^2 (E > 1MeV) at 20 EFPY. This was calculated by linear interpolation using fluence values for 6 EFPY and 32 EFPY of 6.47×10^{18} n/cm^2 and 3.69×10^{19} n/cm^2, respectively, obtained from the surveillance capsule evaluation report (Ref. 3).

3) Calculations were based on the procedures in Regulatory Position 1.1 of NRC Regulatory Guide 1.99, Rev. 2 (Ref. 8). Uncertainty in initial RT_{NDT} was taken as 0°F for measured values of initial RT_{NDT} (σ_I values for the beltline

materials were not required due to measured values of initial RT_{NDT}. See subsequent discussion for the associated technical basis.)

4) The effect of an 8°F reduction in reactor coolant cold leg temperature on RT_{NDT} shift was considered as recommended by Reference 13. This reduction in reactor coolant cold leg temperature, from 553°F to 545°F, is still within the irradiation temperature range for which the methods provided by in Reference 8 are valid. Consequently, the margin term, σ_Δ, currently accounts for uncertainty in the prediction of RT_{NDT} shift which could result from this type of irradiation environment variability and was assessed to be adequate.

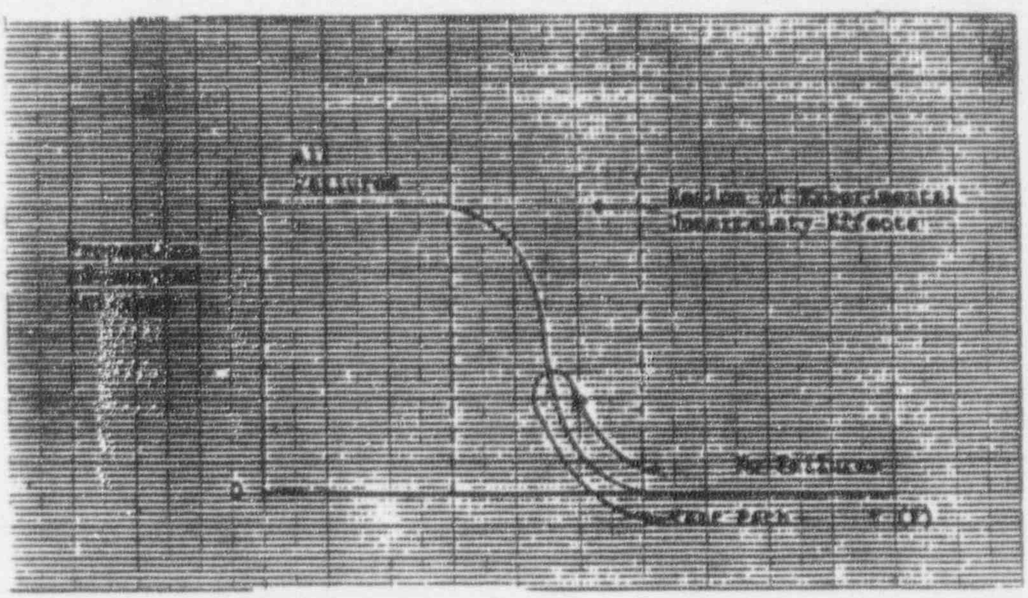
According to Position 1.1 of Regulatory Guide 1.99, Revision 2 (Ref. 8), the uncertainty in the value of initial RT_{NDT} is to be estimated from the precision of test method when a "measured" value of initial RT_{NDT} is available. RT_{NDT} is derived in accordance with NB2300 of the ASME Boiler and Pressure Vessel Code, Section III. It involves both a series of drop weight (ASTM E208) and Charpy impact (ASTM E23) tests on the material. The RT_{NDT} resulting from these two test methods of evaluation are conservatively biased. The elements of this conservatism include:

1) Selection of RT_{NDT} is the higher of NDTT or T_{CV} - 60°F. The drop-weight test is performed to obtain NDTT and a full Charpy impact curve is developed to obtain T_{CV} for a given material. The combination of the two test methods gives protection against the possibility of errors in conducting either test and, with the full Charpy curve, demonstrates that the material is typical of reactor pressure vessel steel. Choice of the more conservative of the two (i.e., the higher of NDTT or T_{CV} - 60°F) assures that tests at temperatures above the reference temperature will yield increasing values of toughness, and verifies the temperature dependence of the fracture toughness implicit in the K_{IR} curve (ASME Code, Section III, Appendix G).

2) Selection of the most adverse Charpy results for T_{CV}. In accordance with NB2300, a temperature, T_{CV}, is established at which three Charpy specimens exhibit at least 35 mils lateral expansion and not less than 50 ft-lb absorbed energy. The three specimens will typically exhibit a range of lateral expansion and absorbed energy consistent with the variables inherent in the test: specimen temperature, testing equipment, operator, and test specimen (e.g.,

dimensional tolerance and material homogeneity). All of these variables are controlled using process and procedural controls, calibration and operator training, and they are conservatively bounded by using the lowest measurement of the three specimens. Furthermore, two related criteria are used, lateral expansion and absorbed energy, where consistency between the two measurements provides further assurance that they are realistic and the material will exhibit the intended strength, ductility and toughness implicit in the K_{IR} curve.

3) Inherent conservatism in the protocol used in performing the drop-weight test. The drop-weight test procedure was carefully designed to assure attainment of explicit values of deflection and stress concentration, eliminating a specific need to account for below nominal test conditions and thereby guaranteeing a conservative direction of these uncertainty components. In addition, the test protocol calls for decreasing temperature until the <u>first</u> failure is encountered, followed by increasing the test temperature 10°F above the point where the <u>last</u> failure is encountered. This in fact assures that one has biased the resulting estimate toward a low failure probability region of the temperature versus failure rate function diagrammed below. The effect of this protocol is to conservatively accommodate the integrated uncertainty components.



Given the three elements of conservatism described above, values of initial RT_{NDT} obtained in accordance with NB2300 will result in a conservative measure of the reference temperature. The conservative bias of the NB2300 methodology and the drop-weight test protocol essentially eliminate the uncertainty which might result from the precision of an individual drop-weight or Charpy impact test. Therefore, when measured values of RT_{NDT} are available, the estimate of uncertainty in initial RT_{NDT} is taken as zero.

Adjusted reference temperatures for all beltline materials at the 1/4t and 3/4t locations through 20 EFPY were calculated using Regulatory Guide 1.99 Revision 2 and the controlling material can be established from the results of the material evaluation shown in Table 2. The term "controlling" means having the highest ART for a given time and position within the vessel wall. The highest, or limiting, ARTs are then used to develop the beltline pressure-temperature limits for the corresponding time period.

In the case of Waterford Unit 3, the limiting material at both the 1/4t and 3/4 locations after 20 EFPY is plate M-1004-2 based on the predicted ART values of 65.4°F and 54.0°F, respectively.

2.4 CALCULATION OF REACTOR VESSEL BELTLINE P-T LIMITS

2.4.1 General Method

The analytical procedure for developing reactor vessel pressure-temperature limits utilizes the methods of Linear Elastic Fracture Mechanics (LEFM) found in the ASME Boiler and Pressure Vessel Code (ASME Code) Section III, Appendix G (Reference 7) in accordance with the requirements of 10 CFR Part 50 Appendix G (Reference 1). For these analyses, the Mode I (opening mode) stress intensity factors are used for the solution basis.

The general method utilizes Linear Elastic Fracture Mechanics procedures. Linear Elastic Fracture Mechanics relates the size of a flaw with the allowable loading which precludes crack initiation. This relation is based upon a mathematical stress analysis of the beltline material fracture toughness properties as prescribed in Reference 7.

The reactor vessel beltline region is analyzed assuming a semi-elliptical surface flaw oriented in the axial direction with a depth of one quarter of the reactor vessel beltline thickness and an aspect ratio (depth to length) of one to six. This postulated flaw is analyzed at both the inside diameter location (referred to as the 1/4t location) and the outside diameter location (referred to as the 3/4t location) to assure the most limiting condition is achieved. The above flaw geometry and orientation is the maximum postulated defect size (reference flaw) described in Paragraph G-2120 of Reference 7.

At each of the postulated flaw tip locations, the Mode I stress intensity factor, K_I, produced by each of the specified loadings is calculated. The summation of the K_I values is then compared to a reference stress intensity, K_{IR}, which is the critical value of K_I for the material and temperature involved. The result of this method is a relation of pressure versus temperature for each condition analyzed providing reactor vessel operating limits which preclude brittle fracture.

In accordance with the ASME Code Section III Appendix G requirements, the general equations for determining the allowable pressure for any assumed rate of temperature change during Service Level A and B operation are:

$$2K_{IM} + K_{IT} < K_{IR} \quad (5)$$

and

$$1.5K_{IM} + K_{IT} < K_{IR} \text{ (Inservice Hydrostatic Test)} \quad (6)$$

where,

K_{IM} = Allowable pressure stress intensity factor, $\mathrm{Ksi}\sqrt{\mathrm{in}}$

K_{IT} = Thermal stress intensity factor, $\mathrm{Ksi}\sqrt{\mathrm{in}}$

K_{IR} = Reference stress intensity, $\mathrm{Ksi}\sqrt{\mathrm{in}}$

The reference stress intensity K_{IR} is defined by Paragraph G-2110 of Reference 7 as

$$K_{IR} = 26.78 + 1.223\, e^{[0.0145(T\text{-}ART + 160)]} \qquad (7)$$

where,

K_{IR}	=	reference stress intensity factor, $Ksi\sqrt{in}$
T	=	temperature at the postulated crack tip, °F
ART	=	adjusted reference nil ductility temperature at the postulated crack tip, °F

At any instant during the postulated heatup or cooldown, K_{IR} is calculated based on the metal temperature at the tip of the flaw and the adjusted reference temperature at that flaw location. The temperature gradients across the reactor vessel wall are also calculated for any instant during the heatup or cooldown (see Section 2.4.2) and the corresponding thermal stress intensity factor, K_{IT}, is determined. The thermal stress intensity is subtracted from the available K_{IR} to determine the allowable pressure stress intensity factor and consequently the allowable pressure.

The pressure-temperature limits provided in this report account for the temperature differential between the reactor vessel base metal and the reactor coolant bulk fluid temperature. Corrections to account for pressure differentials between the location of concern and the location of measurement due to elevation differences and RCS flow are included in the development of the pressure-temperature limits. Uncertainties for the temperature and pressure instrumentation loops associated with control room indications are also included. Consequently, the P-T limits are provided on coordinates of indicated pressurizer pressure versus indicated RCS temperature.

The reactor coolant system pressure measurement is taken from the pressurizer. The differential pressure due to the elevation difference between the reactor vessel beltline wall and the pressurizer was conservatively

established and equal to 36.04 psia for all temperatures. The pressure differential due to the flow induced pressure drop between the reactor vessel inlet nozzle and outlet nozzle was established based on four (4) reactor coolant pumps operating and is equal to 34.71 psi. The pressure differential associated with hot leg flow induced pressure drop was estimated to be 0.196 psi. The uncertainty associated with the pressure indication instrument loop was established as ±28.34 psi for the narrow range instrument and ±114.89 for the wide range instrument. This information was combined to determine the following pressure correction factor utilized in the development of the P-T curves:

Actual Pressure (P) Range	Total Pressure Correction Factor (PCF)
$P < 200$ psia	-186 psi
200 psia $\leq P <$ 850 psia	-100 psi
850 psia $\leq P \leq$ 3000 psia	-186 psi

The uncertainty associated with the temperature indication instrument loop was also included. This value was established to be ±25.6°F (Reference 14).

By explicitly accounting for the temperature differential between the flaw tip base metal temperature and the reactor coolant bulk fluid temperature, and the pressure differentials between the beltline region of the reactor vessel and the pressurizer including the uncertainties associated with the indication loops, the P-T limits are correctly represented on coordinates of indicated pressurizer pressure and indicated cold leg temperature.

2.4.2 Thermal Analysis Methodology

The Mode I thermal stress intensity factor is obtained through a detailed thermal analysis of the reactor vessel beltline wall using a computer code. In this code a one dimensional three noded isoparametric finite element is used for performing the radial conduction-convection transient heat transfer analysis. The vessel wall is divided into 24 elements and an accurate distribution of temperature as a function of radial location and transient time is calculated. The code utilizes a convective boundary condition on the inside

wall of the vessel and an insulated boundary on the outside wall of the vessel. Variation of material properties through the vessel wall are permitted allowing for the change in material thermal properties between the cladding and the base metal.

In general, the temperature distribution through the reactor vessel wall is governed by a partial differential equation,

$$\rho C_p \frac{\partial T}{\partial t} = K \frac{\partial^2 T}{\partial r^2} + \frac{1}{r} \frac{\partial T}{\partial r}$$

subject to the following boundary conditions at the inside and outside wall surface locations:

At $r = r_i$ $\qquad -K \frac{\partial T}{\partial r} = h\,(T - T_c)$

At $r = r_o$ $\qquad \frac{\partial T}{\partial r} = 0$

where,

ρ	=	density, lb/ft^3
C_p	=	specific heat, BTU/lb-°F
K	=	thermal conductivity, BTU/hr-ft-°F
T	=	vessel wall temperature, °F
r	=	radius, ft
t	=	time, hr
h	=	convective heat transfer coefficient, BTU/hr-ft^2-°F
T_c	=	RCS coolant temperature, °F
r_i, r_o	=	inside and outside radii of vessel wall, ft

The above is solved numerically using a finite element model to determine wall temperature as a function of radius, time, and thermal rate.

Thermal stress intensity factors are calculated using a superposition technique and influence coefficients specifically generated for this purpose. The influence coefficients depend upon the geometry of the maximum postulated

defect, the geometry of the reactor vessel beltline region (i.e., r_o/r_i, a/c, a/t where a=crack depth, c=crack half length, and t=vessel wall thickness), and the assumed unit loading. The alternate method employed utilized a third order polynomial fit of the temperature profile and the respective influence coefficient (uniform, linear, quadratic and cubic) to calculate each profile contribution to K_{IT}. The total K_{IT} was the summation of all the contributions. The influence coefficients were calculated using a detailed 2-dimensional finite element model of the reactor vessel. The influence coefficients were corrected for 3-dimensional effects using ASTM Special Technical Publication 677 (Reference 15).

ASME Code Section III Appendix G recognizes the limitations of the method it provides for calculating K_{IT} because of the assumed temperature profile. An alternate method for calculating K_{IT} was employed as required by Subsubparagraph G-2214.3 of Reference 7 to account for the varying temperature profiles (and consequently varying thermal stress intensities) resulting from a detailed heat transfer analysis.

2.4.3 Heatup Limit Analysis

During a heatup transient, the thermal bending stress is compressive at the reactor vessel inside wall and tensile at the reactor vessel outside wall. Internal pressure creates a tensile stress at both the inside wall and the outside wall locations. Consequently, the outside wall location has the larger total stress when compared to the inside wall. However, neutron embrittlement, the shift in material RT_{NDT}, and the associated reduction in fracture toughness are greater at the inside location than the outside. Therefore, both the inside and outside flaw locations must be analyzed to assure that the most limiting condition is achieved.

As described in Section 2.4.1, the reference stress intensity factor is calculated for the metal temperature at the tip of the flaw and the adjusted reference temperature at the flaw location. During heatup, the reference stress intensity is calculated for both the 1/4t and 3/4t locations. The temperature profile through the wall and the metal temperatures at the tip of the flaw are calculated for the transient history using the finite element method described in

Section 2.4.2. This information is used to calculate the thermal stress intensity factor at the 1/4t and 3/4t locations using the calculated wall gradient and thermal influence coefficients. The allowable pressure stress intensity is then determined by rearranging equation 5 to give

$$K_{IM} = (K_{IR} - K_{IT})/2 \tag{8}$$

The allowable pressure is then derived from the calculated allowable pressure stress intensity factor. Influence coefficients due to unit pressure loadings have been developed through detailed finite element analyses. These influence coefficients permit expedient conversion of the stress intensity factor to an allowable pressure.

It is interesting to note that a sign change occurs in the thermal stress through the reactor vessel beltline wall. Considering a reference flaw at the 1/4t location, it can be shown that the thermal stress tends to alleviate the pressure stress. This would indicate that the isothermal steady state condition would represent the controlling P-T limit. However, the isothermal condition may not always provide the limiting pressure-temperature limit for the 1/4t location during a heatup transient. This is due to the correction of the base metal temperature to the Reactor Coolant System (RCS) fluid temperature at the inside wall by accounting for clad and film temperature differentials.

For a given heatup rate (non-isothermal), the differential temperature through the clad and film increases as a function of thermal rate resulting in a higher RCS fluid temperature at the inside wall than the isothermal condition for the same flaw tip temperature and pressure. Therefore, to ensure the accurate representation of the 1/4t pressure-temperature limit during heatup, both the isothermal and heatup rate dependent pressure-temperature limits are calculated to ensure the limiting condition was achieved. The limits for both heatup and cooldown account for clad and film differential temperatures and for the gradual buildup of wall differential temperatures with time.

At the 3/4t location the pressure stress and thermal stresses are both tensile, resulting in the maximum stress at that location. Pressure-temperature limits were calculated for the 3/4t location accounting for clad and film differential temperature and the buildup of wall temperature gradients with time using the method described in Section 2.4.2. The allowable pressure based upon a flaw at the 3/4t location was derived in the same way as if the flaw were at the 1/4t location. With the K_{IR} and K_{IT} both calculated, equation 8 is used to solve for the allowable pressure stress intensity, K_{IM}. The allowable pressure is then calculated using K_{IM}.

To develop composite pressure-temperature limits for the heatup transient, the isothermal, 1/4t heatup, and 3/4t heatup pressure-temperature limits are compared for a given thermal rate. The most restrictive pressure-temperature limits are then compiled for each analyzed temperature throughout the transient duration, resulting in a composite limit curve for the reactor vessel beltline for the heatup event.

Table 3 provides the results pressure-temperature limits for linear heatup rates of 30, 50, and 60°F/hr. The allowable pressure is in units of psia, and the temperature is in units of °F. Figure 1 provides a graphical presentation of the heatup pressure-temperature limits found in Table 3. It is permissible to linearly interpolate between the heatup pressure-temperature limits.

2.4.4 Cooldown Limit Analysis

During cooldown, membrane and thermal bending stresses act together in tension at the reactor vessel inside wall. This results in the pressure stress intensity factor, K_{IM}, and the thermal stress intensity factor, K_{IT}, acting in unison creating a high stress intensity factor. At the reactor vessel outside wall the tensile pressure stress and the compressive thermal stress act in opposition resulting in a lower total stress intensity factor than at the inside wall location. Also neutron embrittlement, the shift in RT_{NDT}, and the associated reduction in fracture toughness are less severe at the outside wall when compared to the inside wall location. Consequently, the inside flaw location is always more limiting and is analyzed for the cooldown event.

The reference stress intensity is once again determined using the material metal temperature and adjusted reference temperature at the 1/4t location. From the method provided in Section 2.4.2, the through wall temperature gradient is calculated for the assumed cooldown rate to determine the thermal stress intensity factor. In general the thermal stress intensity factors are found using the temperature profile through the wall as a function of transient time as described in Section 2.4.2. They are then subtracted from the available K_{IR} value to find the allowable pressure stress intensity factor and consequently the allowable pressure.

Pressure-temperature curves are generated for cooldown transients the same way they are generated for heatup transients. The allowable pressure stress intensity at the 1/4t location is calculated using equation 8, and the allowable pressure is calculated from the allowable pressure stress intensity factor, K_{IM}.

To develop a composite pressure-temperature limit for a specific cooldown event, the isothermal pressure-temperature limit must be calculated. The isothermal pressure-temperature limit is then compared to the limit associated with the specific cooling rate, and the more restrictive of the two limits is chosen to result in a composite limit curve for the reactor vessel beltline.

Table 4 provides the results for the isothermal condition and the linear rates of 10, 30, and 100°F/hr cooldown. The allowable pressure is in units of psia, and the temperature is in units of °F. Figure 2 provides a graphical presentation of the cooldown pressure-temperature limits found in Table 4. It is permissible to linearly interpolate between the linear cooldown pressure-temperature limits.

2.4.5 Hydrostatic Test Limit Analysis

Both 10 CFR Part 50 Appendix G and the ASME Code Appendix G require the development of pressure-temperature limits which are applicable to inservice hydrostatic tests. For hydrostatic tests performed subsequent to loading fuel into the reactor vessel, the minimum test temperature is determined by evaluating the Mode I stress intensity factors. The evaluation of these factors is performed in the same manner as that for normal operation

heatup and cooldown conditions with two differences. Equation 6 shows that the safety factor applied to the pressure stress intensity factor is 1.5 instead of 2.0. Also, the inservice hydrostatic test limit is established based upon an isothermal condition. This eliminates the thermal stress intensity factor, K_{IT}, from equation 6.

The inservice hydrostatic test limit is provided for 20 EFPY in Table 5 and is shown in Figure 3. The minimum temperature for the inservice hydrostatic test pressure can be conservatively determined using the guidance of ASME Code Section XI, Subarticle 2500 (Reference 16) using the curve developed for inservice hydrostatic test. A test temperature (indicated) equal to 219.8°F is necessary for the selected test pressure of 2475 psia (1.1 times normal operating pressure).

2.5 CORE CRITICAL LIMITS

Pressure-temperature limits for core critical operation are specified in 10 CFR 50, Appendix G to provide additional margin during actual power operation. The pressure-temperature limit for core critical operation is based upon two criteria. These criteria are that the reactor vessel must be at a temperature equal to or greater than the minimum temperature required for the inservice hydrostatic test (219.8°F) and be at least 40°F higher than the minimum pressure-temperature curve for normal operation heatup or cooldown.

Note that the core critical limits established above are solely based upon fracture mechanics considerations and do not consider core reactivity safety analyses which can control the temperature at which the core can be brought critical.

2.6 FLANGE LIMITS

As stated in Reference 1, the temperature of the closure flange regions must exceed the initial RT_{NDT} of the material by at least 120°F for normal operation. The temperature must exceed the initial RT_{NDT} by at least 90°F for hydrostatic tests and leak testing when the pressure exceeds 20% of preservice hydrostatic test pressure.

Accounting for instrument uncertainty, the flange limits were calculated to be (given an initial RT_{NDT} of 20°F) 165.6°F for normal operation and 135.6°F for hydrostatic testing. These are the minimum temperatures in the flange region for the pressure to exceed 20% of the preservice hydrostatic test pressure.

A review of the original design basis was performed which identified flange limits developed using the guidance of ASME Code, Section III, Appendix G. This provided a flange limit associated with a heatup rate of 50°F/hr which was adjusted with the correction factor from Section 2.4.1. The flange limit is provided below in terms of indicated pressure and temperature.

50°F/hr heatup	Indicated T_c (°F)	Indicated P_{all} (psia)
	215.6	618
	265.6	3991

The design basis value is more restrictive while meeting 10 CFR 50 Appendix G requirements and shall be used in the development of the Technical Specification.

2.7 LOWEST SERVICE TEMPERATURE

The Lowest Service Temperature is the minimum allowable temperature at which pressures can exceed the pre-operational system hydrostatic test pressure (625 psia uncorrected). This temperature is defined by Paragraph NB-2332 of ASME Code Section III (Reference 17) as to the most limiting RT_{NDT} for the balance of Reactor Coolant System (RCS) components plus 100°F.

The maximum RT_{NDT} for the balance of the RCS components was conservatively established to be 90°F and was associated with the reactor coolant pump. Therefore, the Lowest Service Temperature is equal to 100°F + 90°F + 25.6°F = 215.6°F, where 25.6°F is the temperature instrument uncertainty.

2.8 MINIMUM PRESSURE

The minimum pressure limit is defined as 20% of the pre-operational hydrostatic test pressure. Therefore, the minimum pressure is 625 psia uncorrected. After application of the appropriate pressure correction factor, the indicated minimum

pressure was calculated to be 525 psia. The minimum pressure shall not be exceeded prior to achieving a coolant temperature equal to the Lowest Service Temperature.

2.9 MINIMUM BOLTUP TEMPERATURE

The minimum boltup temperature is the minimum allowable temperature for the flange to be stressed by the full intended bolt preload and by pressures less than or equal to 20% of the pre-operational system hydrostatic test pressure. The minimum boltup temperature is defined by Paragraph G-2222 of ASME Code Appendix G (Reference 7) as the initial RT_{NDT} for the material of the higher stressed region of the reactor vessel plus any effects for irradiation. The maximum initial RT_{NDT} associated with the stressed region determined to be 20°F. The minimum boltup temperature including temperature instrument uncertainty is 20°F + 25.6°F = 45.6°F. However, for additional conservatism it is recommended that the currently specified indicated temperature of 72°F continue to be used.

3.0 LTOP ENABLE TEMPERATURES

Standard Review Plan 5.2.2, Overpressure Protection (Reference 18), has defined the temperature at which the Low Temperature Overpressure Protection (LTOP) system should be operable during startup and shutdown conditions. This temperature, known as the LTOP enable temperature, is defined 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 calculations. The ASME Section XI Code Case N-514 (Ref. 19) suggests a coolant temperature which corresponds to a metal temperature of at least (ART + 50°F) at the beltline location or 200°F, whichever is greater. The results of this method (referred to as the N-514 method below) are included for information only at this time.

The LTOP enable temperature for cooldown is based on the isothermal case. The 1/4t location is limiting for both cooldown and isothermal cases. During a cooldown transient, the temperature at any point in the wall is higher than the coolant temperature. Therefore, it is conservative to determine the LTOP enable temperature for cooldown based on an isothermal case. In an isothermal condition the water temperature is equal to ART + 90°F. Including instrument uncertainty, the LTOP enable temperature for cooldown was established to be 181°F.

LTOP enable temperatures for heatup were determined for each heatup rate at both the 1/4t and 3/4t locations. The values and controlling locations are provided below.

LTOP Enable Temperatures (SRP 5.2.2)
Heatup, 20 EFPY

	1/4t flaw location	3/4t flaw location
30°F/hr	193.7°F (3/4t limiting)	192.3°F (3/4t limiting)
50°F/hr	201.8°F (3/4t limiting)	206.8°F (3/4t limiting)
60°F/hr	205.6°F (3/4t limiting)	213.6°F (3/4t limiting)

Review of these values along with consideration of the allowable heatup rates and respective temperature ranges provide an LTOP enable temperature of 206.8°F (207°F) including instrument uncertainty. Above this value, the primary safety valves provide adequate protection.

The LTOP enable temperatures were calculated using the criteria (ART + 50°F) of Code Case N-514. In the case of both heatup and cooldown, the values were less than 200°F. Consequently, the LTOP enable temperatures for heatup and cooldown would be 200°F based on Code Case N-514.

4.0 PRESSURIZED THERMAL SHOCK (PTS) SCREENING CRITERIA

In accordance with 10 CFR §50.61 (Ref. 20), the pressurized thermal shock (PTS) criteria were evaluated with this update of the pressure-temperature limits to ensure a complete submittal. Once the value of RT_{PTS} is calculated for all beltline materials, it must not exceed the PTS screening criteria of 270°F for plates, forgings, and axial weld materials and 300°F for circumferential welds. As required by 10 CFR §50.61, the RT_{PTS} is calculated using the following equation

$$RT_{PTS} = I + M + \Delta RT_{PTS} \tag{9}$$

where I is the initial reference temperature of the material, M is a margin added to cover uncertainty in the initial RT_{NDT}, and ΔRT_{PTS} is the shift in the initial reference temperature caused by irradiation. Measured values of I were used, so M was set to 56°F for weld materials and 34°F for base metals. The ΔRT_{PTS} term is calculated at the vessel inside surface using the same equation for calculating ΔRT_{NDT} (eq. 2). Material information from Table 1 and an estimated peak surface fluence of 3.69×10^{19} n/cm^2 corresponding to 32 EFPY (from Ref. 3) were used to calculate RT_{PTS} for all the beltline materials, and the results are presented in Table 6.

5.0 END-OF-LIFE UPPER SHELF ENERGY

10 CFR 50 Appendix G requires that the reactor vessel beltline material maintain a Charpy upper-shelf energy (USE) of 50 ft-lbs throughout its operational lifetime. Regulatory Guide 1.99 Rev. 2 (Reference 8) provides a method to predict the decrease in Charpy USE which is based on initial Charpy USE, accumulated fluence and copper content (w%). To ensure compliance with 10 CFR 50, Appendix G the decrease at end-of-life (32 EFPY) was calculated through strict application of Regulatory Position 1.2 of Reference 8. Values of initial Charpy USE were obtained, in most instances, from Reference 12. To evaluate the beltline welds, Waterford Unit 3 original fabrication records were used to ascertain the initial Charpy USE values. The initial and end-of-life (1/4) Charpy USE values are summarized in Table 6.

6.0 SURVEILLANCE CAPSULE WITHDRAWAL SCHEDULE

The Waterford 3 reactor has 6 surveillance capsules designed to monitor the changes in beltline material properties (Ref. 21). The governing withdrawal schedule for these capsules as required by 10 CFR 50, Appendix H, is defined in Table 5.3-10 of the Waterford 3 Final Safety Analysis Report (FSAR) (Ref. 22). This current withdrawal schedule is presented in Table 7 along with the capsule identification number and original target fluence as presented in Reference 3.

Capsule 2, located at the 97 degree position, (also referred to as capsule W-97) was removed, and the encapsulated specimens were tested. A major result in the W-97 capsule report pertinent to the capsule removal schedule was a change in the capsule lead factors. The lead factor is defined as the ratio of neutron flux density at the location of the specimens in a surveillance capsule to the neutron flux density of the inside surface at the peak fluence location (Ref. 23). For capsules W-104 and W-284 the lead factor was revised from 1.5 to 0.81 (Ref. 3) and for the remaining capsules (W-83, W-97, W-263 and W-277) the lead factor was revised from 1.5 to 1.26 in Reference 3.

A revised schedule was developed using the lead factors provided by Reference 3 and the guidance of ASTM E185-82 in accordance with current 10 CFR 50, Appendix H requirements. Factors external to the ASTM E185-82 procedure that were also considered included:

1. Coordination with the generation of P-T limits and LTOP evaluation beyond 20 EFPY. - If additional surveillance capsule information is to be used to support the generation of P-T limits and an LTOP evaluation beyond 20 EFPY, the next capsule withdrawal must allow for enough time to analyze the encapsulated materials as well as develop new P-T limits and LTOP requirements prior to 20 EFPY.

2. Potential for use of Position 2 of Regulatory Guide 1.99, Rev. 2 - Surveillance capsule data may be used in conjunction with Position 2 of R.G. 1.99, Rev. 2 to predict mean shift in reference temperature (ΔRT_{NDT}) and decrease in upper shelf energy (USE) once credible surveillance data is obtained. One requirement for credibility is that, "the surveillance data for the correlation

monitor material in the capsule should fall within the scatter band of the data base for that material" (Reference 8).

Capsule W-97 did not contain correlation material (Ref. 21), so the next capsule withdrawn must contain correlation material in order to allow for the use of Position 2 of R.G. 1.99, Rev. 2. The two capsules that contain correlation material are W-104 and W-263 (Ref. 21).

3. The reactor coolant cold leg temperature for Waterford 3 has been reduced by 8°F from 553°F to 545°F (Ref. 13). The effect, if any, of this temperature reduction on the reactor vessel beltline materials must be monitored.

   The new operating condition was evaluated, and it was determined that the requirements of 10 CFR 50, Appendix G and 10 CFR 50.61 are not affected by the temperature reduction of the cold leg (Ref. 13). However, variations in the adjusted reference temperature (ART) and upper shelf energy (USE) of the surveillance material from predicted decreases must be monitored to verify the validity of the previous studies. This evaluation should be made at the time of the second surveillance capsule withdrawal, and modifications to the shift in ART and USE predictions can be made if necessary. The timing of the second capsule withdrawal should be such that significant variations from predictions can be detected early enough to ensure that the P-T limits based on the ART predictions remain conservative.

4. The surveillance capsule withdrawal schedule should be managed with consideration given to plant license renewal. Enough capsules must be tested to assure confidence in beltline material properties, but capsules should also be conserved to allow for future testing beyond the current design lifetime.

The guidelines provided in Section 7, "Irradiation Requirements" and Subsection 7.6, "Number of Surveillance Capsules and Withdrawal Schedule" of ASTM E185-82 (Ref. 23) are currently required by 10 CFR 50, Appendix H for establishing the surveillance capsule withdrawal schedule. A review of the proposed revised standards (Ref. 24) showed no changes affecting the method for determining the withdrawal schedule. Therefore, future modifications to 10 CFR 50 Appendix H by reference to this revised ASTM E185-93 standard are not expected to alter the capsule withdrawal

requirements.

According to ASME E185-82 (Ref. 23), the peak vessel inside fluence at EOL and the corresponding transition temperature shift must be estimated to determine the number of capsules required for removal. Waterford 3 has a peak EOL fluence of 3.69×10^{19} n/cm^2 (Ref. 3) and a 1/4t fluence of 2.20×10^{19} n/cm^2 (using equation 3 of R.G. 1.99, Rev. 2 to attenuate fluence to the 1/4t location).

Based on the calculations of RT_{PTS} (Table 6), the largest shift in reference temperature (ΔRT_{NDT}) at EOL is 59.4°F (note that the method in 10 CFR 50.61 for calculating ΔRT_{PTS} and the R.G. 1.99, Rev. 2 method for calculating ΔRT_{NDT} produce equivalent results). Using ASTM E185-82, it was determined that 3 capsules must be withdrawn in the following order.

First Capsule: (Removed and tested)

Second Capsule: At 15 EFPY or at the time when the accumulated neutron fluence of the capsule corresponds to the approximate EOL fluence at the reactor vessel inner wall location, whichever comes first.

Third (Final) Capsule: At EOL but not less than once or greater than twice the peak EOL vessel fluence. This may be modified on the basis of previous tests. This capsule may be held without testing following withdrawal.

The second capsule to be withdrawn could be either W-104 or W-263 to obtain credible surveillance data. However, capsule W-104 has a low lead factor (0.81), whereas capsule W-263 has a high lead factor (1.26). It is preferred to withdraw capsule W-263 for the second capsule, as the capsule fluence would be greater than the peak surface fluence received by the vessel. Capsule W-263 would be expected to receive a fluence equivalent to the EOL fluence at the reactor vessel inner wall at 25.4 EFPY (32 EFPY/1.26).

Given the criteria for withdrawal of the second capsule (above), capsule W-263 should be withdrawn at 15 EFPY. The capsule fluence corresponding to 15 EFPY was estimated to be 2.18×10^{19} n/cm^2 using the lead factor of 1.26 and linear interpolation of the EOL vessel fluence of 3.69×10^{19} n/cm^2 given in Reference 3.

Modifying the withdrawal schedule to meet the current edition of ASTM standards calls for the last capsule to be removed between 25.4 and 50.8 EFPY. It is suggested that capsule withdrawal occur no later than 32 EFPY because this time corresponds to the plant EOL. This will correspond to a capsule fluence of 4.65×10^{19} n/cm^2.

Given the requirements of 10 CFR 50, Appendix H and ASTM E185-82 along with the plant-specific considerations for Waterford 3, Table 8 presents the recommended schedule for the Waterford 3 reactor vessel surveillance capsule removal program:

This schedule meets ASTM E185-82 requirements for capsule withdrawal (Ref. 23) as currently required by 10 CFR 50, Appendix H. It allows for detection of any effect on ΔRT_{NDT} or decrease in USE which could result from the reduction in cold leg temperature. This schedule should make available credible surveillance data for analyses following Position 2 of Regulatory Guide 1.99, Revision 2 (Ref. 8), and it provides for capsule withdrawal and testing prior to a P-T limit modification following 20 EFPY. This schedule will also allow for a sufficient number of standby capsules (3) to be maintained for possible license renewal or to provide for other future contingencies.

7.0 RESULTS

Revised reactor vessel beltline P-T limits associated with twenty (20) EFPY have been developed for heatup, cooldown, and inservice hydrostatic test. The reactor vessel P-T limits are provided in tabular form in Tables 3, 4 and 5 and graphically in Figures 1, 2 and 3. The beltline limits were considered in conjunction with the other limits discussed in Sections 2.5-2.9 to produce new Technical Specification Figures 3.4-2 and 3.4-3 for heatup and cooldown, respectively. These Technical Specification figures meet the requirements of 10 CFR 50, Appendix G and guidance of ASME Section III, Appendix G, and they will continue to support the Waterford 3 Technical Specifications through 20 EFPY.

The heatup and cooldown rates, along with the associated temperature ranges remain the same as currently specified in the Technical Specifications (8 EFPY). The Technical Specification Figures for 20 EFPY provide a less restrictive controlling pressure than the current P-T limits for 0-8 EFPY, so the current SDCS relief valve setpoint and other administrative controls previously established for LTOP will continue to provide adequate protection.

An evaluation of the PTS screening criteria was performed in accordance with 10 CFR 50.61. Values of RT_{PTS} were calculated for all beltline materials at end of life and are shown in Table 6. The values of RT_{PTS} for all beltline materials were far below the screening criteria and therefore meet the requirements of 10 CFR 50.61.

The end of life USE was evaluated for all beltline materials in accordance with References 1 and 8. The decrease in USE resulting from irradiation was calculated at the 1/4t location, and the results are shown in Table 6. All values for end of life USE are well above 50 ft·lbs and satisfy the requirements of 10 CFR 50, Appendix G.

A revised surveillance withdrawal schedule was developed based on the results of the W-97 capsule evaluation report. This revised schedule, Table 8, has been developed to meet utility goals and the requirements of 10 CFR 50, Appendix H.

Table 1
RPV Beltline Materials

Product	Material ID	Initial RT_{NDT} (°F)	Copper Content	Nickel Content	Charpy Longitudinal USE(ft · lbs)
Plate	M-1003-1	-30	0.02	0.71	144
Plate	M-1003-2	-50	0.02	0.67	149
Plate	M-1003-3	-42	0.02	0.70	138
Plate	M-1004-1	-15	0.03	0.62	163
Plate	M-1004-2	22	0.03	0.58	144
Plate	M-1004-3	-10	0.03	0.62	145
Weld	101-124 A	-60	0.02	0.96	106[a]
Weld	101-124 B,C	-60	0.02	0.96	131[a]
Weld	101-142 A,B,C	-80	0.03	<0.20	129[a]
Weld	101-171	-70	0.05	0.16	166[a]
Weld (repair)	101-142 B	-40[a]	0.03[a]	0.97[a]	110[a]

(a) Values were calculated using information from weld Certified Material Test Reports and Weld Inspection Forms.

Table 2

ART Values for Beltline Materials at 20 EFPY

Product	Matl. ID	Init. RT_{NDT}	ΔRT_{NDT} (1/4t)	ΔRT_{NDT} (3/4t)	σ_I	σ_Δ (1/4t)	σ_Δ (3/4t)	M 1/4t	M 3/4t	ART 1/4t	ART 3/4t
Plate	M-1003-1	-30	21.7	16.0	0	10.9	8.0	21.7	16.0	13.4	2.0
Plate	M-1003-2	-50	21.7	16.0	0	10.9	8.0	21.7	16.0	-6.6	-18.0
Plate	M-1003-3	-42	21.7	16.0	0	10.9	8.0	21.7	16.0	1.4	-10.0
Plate	M-1004-1	-15	21.7	16.0	0	10.9	8.0	21.7	16.0	28.4	17.0
Plate	M-1004-2	22	21.7	16.0	0	10.9	8.0	21.7	16.0	65.4	54.0
Plate	M-1004-3	-10	21.7	16.0	0	10.9	8.0	21.7	16.0	33.4	22.0
Weld	101-124 A,B,C	-60	29.3	21.5	0	14.7	10.8	29.3	21.5	-1.4	-17.0
Weld	101-142 A,B,C	-80	38.0	27.9	0	19.0	14.0	38.0	27.9	-4.0	-24.2
Weld	101-171	-70	48.2	35.4	0	24.1	17.7	48.2	35.4	26.4	0.8
Weld (repair)	101-142 B	-40	44.5	32.7	0	22.3	16.4	44.5	32.7	49.0	25.4

Table 3

Beltline P-T Limits
Heatup, 20 EFPY (psia)

RCS Temp (°F)	Isothermal	RCS Temp (°F)	30°F/hr	RCS Temp (°F)	50°F/hr	RCS Temp (°F)	60°F/hr
72	572.0	72	572.0	72	572.0	72	572.0
75.6	582.0	75.6	582.0	75.6	582.0	75.6	582.0
85.6	610.8	85.6	610.8	85.6	610.8	85.6	610.8
95.6	644.1	95.6	618.4	95.6	598.9	95.6	594.5
105.6	682.6	105.6	632.7	105.6	594.0	105.6	583.5
115.6	727.1	115.6	659.8	115.6	601.4	115.6	583.7
120.0	749.9 *	125.6	692.6	125.6	619.9	125.6	594.5
120.1	664.0 *	135.6	746.8	135.6	648.7	135.6	615.1
125.6	692.6	136.1	749.9 *	145.6	687.6	145.6	645.5
135.6	752.1	136.2	664.0 *	155.6	736.9	155.6	685.5
145.6	820.9	145.6	719.9	158.0	749.9 *	165.6	736.1
155.6	900.4	155.6	789.7	158.1	664.0 *	168.0	749.9 *
165.6	992.4	165.6	871.9	165.6	711.1	168.1	664.0 *
175.6	1098.6	175.6	967.3	175.6	783.2	175.6	711.6
185.6	1221.5	185.6	1078.5	185.6	868.6	185.6	785.7
195.6	1363.5	195.6	1206.9	195.6	968.8	195.6	873.0
--	--	199.9	1271.0 *	200.0	1020.3 *	--	--
205.6	1527.7	205.6	1356.0	205.6	1085.9	205.6	976.1
215.6	1717.5	215.6	1528.0	215.6	1222.2	215.6	1096.2
225.6	1937.0	225.6	1727.6	225.6	1380.5	225.6	1236.8
--	--	--	--	228.0	1424.6 *	--	--
235.6	2190.7	235.6	1957.6	235.6	1564.2	235.6	1399.7
245.6	2483.9	245.6	2224.5	245.6	1777.0	245.6	1589.5
246.2	2500.0 *	255.6	2523.5	255.6	2023.4	255.6	1808.8
265.6	--	265.6	--	265.6	2306.5	265.6	2063.8
--	--	--	--	271.4	2500.0 *	--	--
275.6	--	275.6	--	275.6	2636.5	275.6	2358.0
285.6	--	285.6	--	285.6	--	285.6	2699.7

* – interpolated value

Table 4

Beltline P-T Limits
Cooldown, 20 EFPY (psia)

RCS Temp (°F)	Isothermal	RCS Temp (°F)	10°F/hr	RCS Temp (°F)	30°F/hr	RCS Temp (°F)	100°F/hr
65.6	--	65.6	526.0	65.6	465.7	65.6	280.5
72.0	572.0	72.0	542.9 *	72.0	484.9 *	72.0	310.1
75.6	582.0	75.6	552.4	75.6	495.7	75.6	326.7
85.6	610.8	85.6	583.2	85.6	530.4	85.6	380.0
95.6	644.1	95.6	618.5	95.6	570.4	95.6	441.7
105.6	682.6	105.6	659.6	105.6	616.9	105.6	513.0
115.6	727.1	115.6	706.8	115.6	670.3	115.6	595.3
120.0	749.9 *	123.6	749.9 *	125.6	692.6	125.6	690.6
120.1	664.0 *	123.7	664.0 *	128.2	749.9 *	131.1	749.9
125.6	692.6	125.6	675.7	128.3	664.0 *	131.2	664.0
--	--	134.9	734.4 *	135.0	713.4 *	--	--
135.6	752.1	135.6	738.8	135.6	717.8	135.6	714.6
145.6	820.9	145.6	812.2	145.6	800.8	145.6	820.9
155.6	900.4	155.6	896.6	155.6	896.2	155.6	900.4
165.6	992.4	165.6	992.4	165.6	992.4	165.6	992.4
175.6	1098.6	175.6	1098.6	175.6	1098.6	175.6	1098.6
185.6	1221.5	185.6	1221.5	185.6	1221.5	185.6	1221.5
195.6	1363.5	195.6	1363.5	195.6	1363.5	195.6	1363.5
--	--	--	--	200.0	1436.7 *	200.1	1437.4 *
205.6	1527.7	205.6	1527.7	205.6	1527.7	205.6	1527.7
215.6	1717.5	215.6	1717.5	215.6	1717.5	215.6	1717.5
225.6	1937.0	225.6	1937.0	225.6	1937.0	225.6	1937.0
235.6	2190.7	235.6	2190.7	235.6	2190.7	235.6	2190.7
245.6	2483.9	245.6	2483.9	245.6	2483.9	245.6	2483.9
246.2	2500.0 *	246.2	2500.0 *	246.2	2500.0 *	246.2	2500.0

* -- interpolated value

Table 5

Beltline P-T Limits
Hydrostatic Test, 20 EFPY

RCS Temp (°F)	Pall
72	710.2
75.6	723.3
85.6	761.7
95.6	806.1
105.6	857.5
115.6	916.8
125.6	985.5
135.6	1064.8
145.6	1156.5
155.6	1262.6
165.6	1385.1
175.6	1526.8
185.6	1690.7
195.6	1880.0
205.6	2099.0
215.6	2352.1
219.8	2475 *
220.7	2500 *
225.6	2644.6

* – interpolated value

Table 6

RT$_{PTS}$ and End of Life USE Evaluations

Product	Matl. ID	Initial RT$_{NDT}$ (°F)	ΔRT$_{PTS}$ (°F)	RT$_{PTS}$ (°F)	Initial USE Transverse (ft · lbs)	End of Life USE (ft · lbs)
Plate	M-1003-1	-30	26.8	30.8	93.6	72.3
Plate	M-1003-2	-50	26.8	10.8	96.9	74.8
Plate	M-1003-3	-42	26.8	18.8	89.7	69.2
Plate	M-1004-1	-15	26.8	45.8	106.0	81.8
Plate	M-1004-2	22	26.8	82.8	93.6	72.3
Plate	M-1004-3	-10	26.8	50.8	94.3	72.8
Weld	101-124A	-60	36.1	32.1	106	81.8
Weld	101-124B,C	-60	36.1	32.1	131	101.1
Weld	101-142A,B,C	-80	46.8	22.8	129	99.6
Weld	101-171	-70	59.4	45.4	166	128.2
Weld (repair)	101-142B	-40	54.9	70.9	110	84.9

Table 7

WSES-FSAR-Unit 3, Capsule Assembly Removal Schedule

Capsule No.	Capsule I.D.	Azimuthal Location (deg.)	Lead Factor	Removal Time (EFPY)	Target Fluence (n/cm^2)
1	W-83	83	1.5	Standby	---
2	W-97	97	1.5	3.5 - 4.5	0.6×10^{19}
3	W-104	104	1.5	10 - 12	1.6×10^{19}
6	W-284	284	1.5	16 - 20	2.5×10^{19}
4	W-263	263	1.5	Standby	---
5	W-277	277	1.5	Standby	---

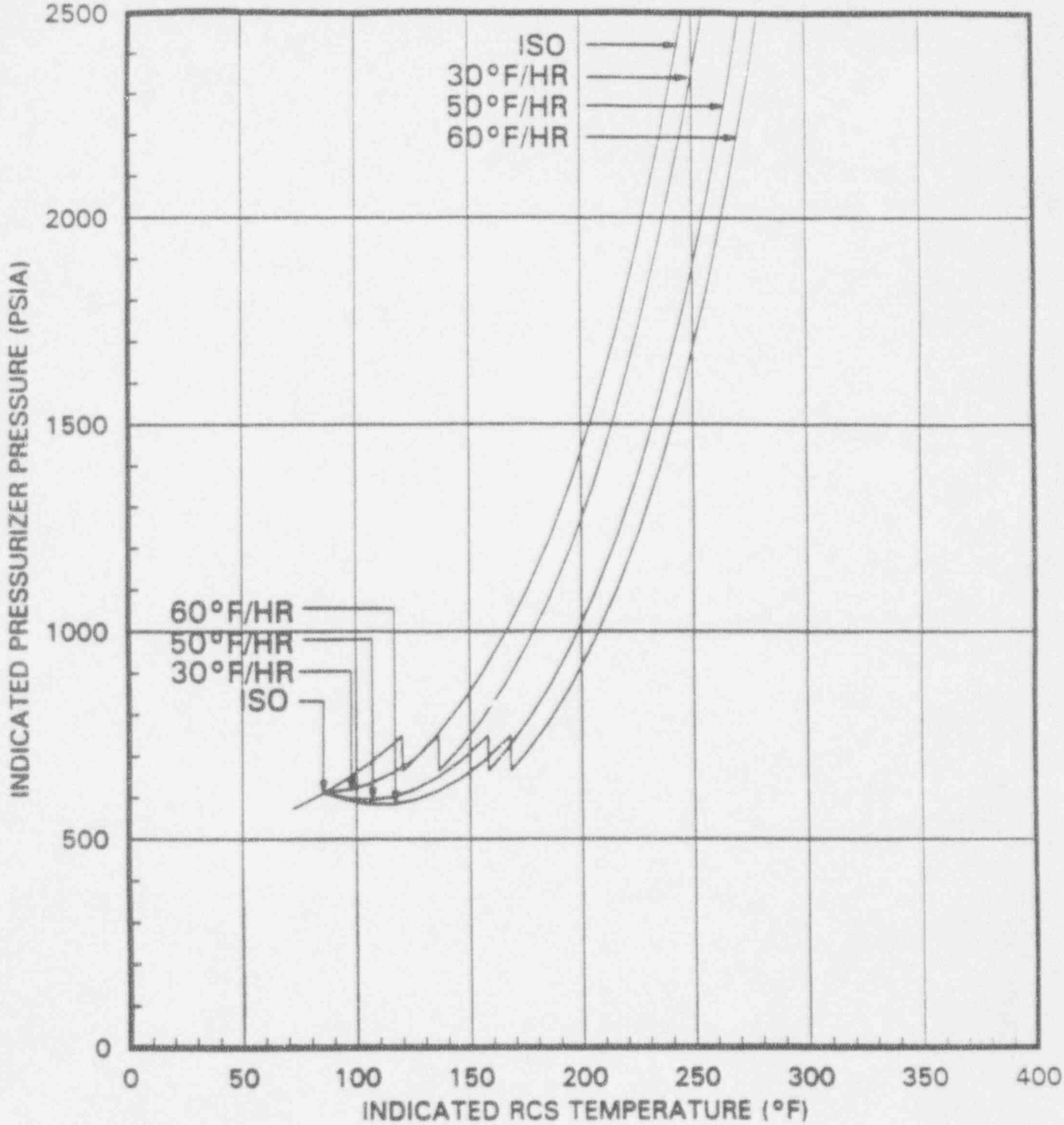
Table 8

Proposed Capsule Removal Schedule Meeting ASTM E185-82 Requirements

Capsule No.	Capsule I.D.	Azimuthal Location (deg.)	Lead Factor	Removal Time (EFPY)	Target Fluence (n/cm^2)
1	W-83	83	1.26	Standby	---
2*	W-97	97	1.26	4.44	6.47×10^{18}
3	W-104	104	0.81	Standby	---
4	W-263	263	1.26	15	2.18×10^{19}
5	W-277	277	1.26	25.4 to 50.8 Recommended ≤32	3.69×10^{19} to 7.38×10^{19} Recommended $\leq 4.65 \times 10^{19}$
6	W-284	284	0.81	Standby	---

*Values represent actual data on removed capsule.

FIGURE 1
WATERFORD UNIT 3
APPENDIX G BELTLINE P-T LIMITS, HEATUP



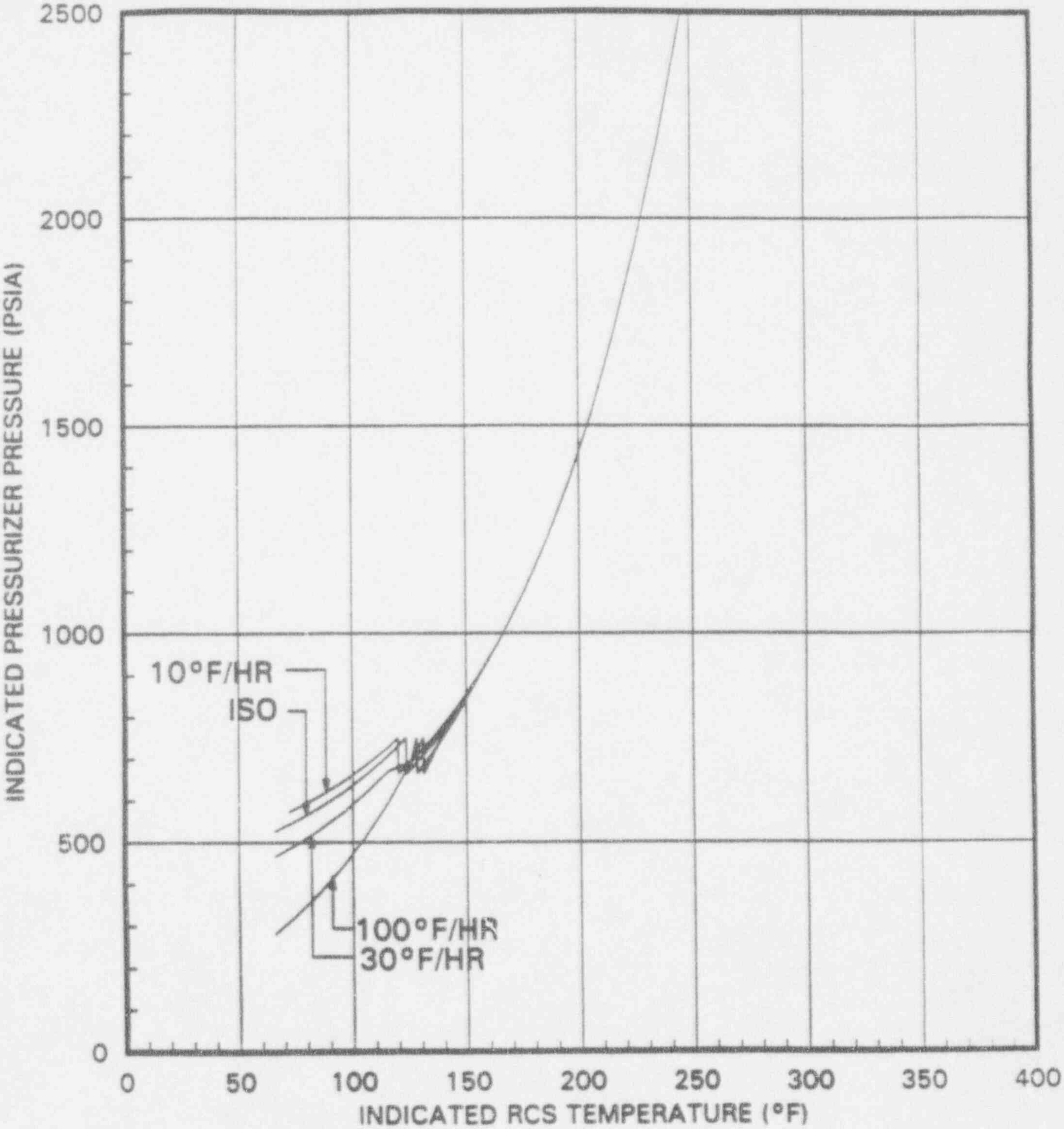
0 psia < Prcs < 200 psia, ΔP = -186 psi
200 psia ≤ Prcs < 850 psia, ΔP = -100 psi
850 psia ≤ Prcs ≤ 3000 psia, ΔP = -186 psi
ΔT = +25.6°F

ART
1/4t = 65.4°F
3/4t = 54.0°F

FIGURE 2

WATERFORD UNIT 3

APPENDIX G BELTLINE P-T LIMITS, COOLDOWN

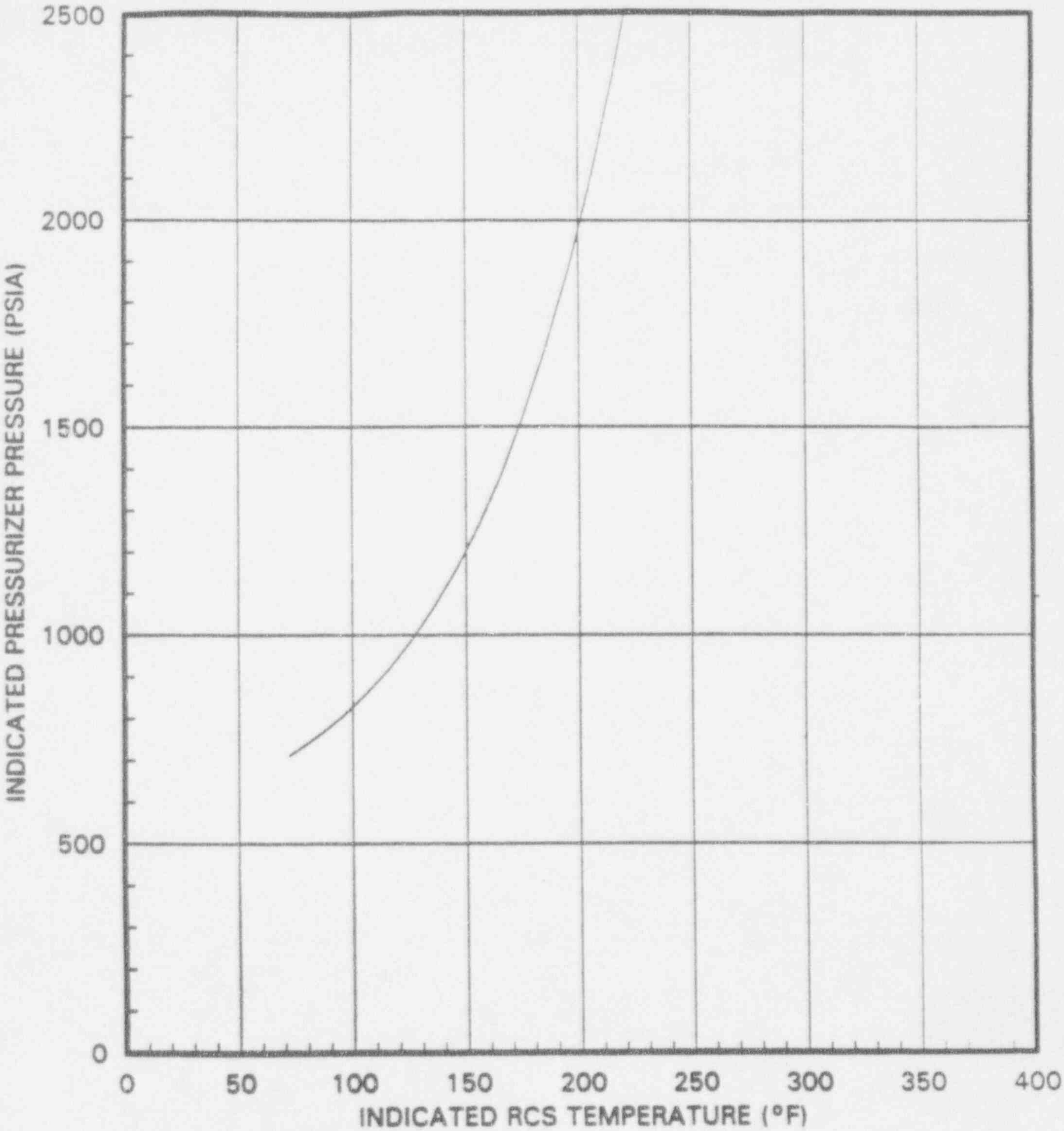


0 psia < Prcs < 200 psia, ΔP = -186 psi
200 psia ≤ Prcs < 850 psia, ΔP = -100 psi
850 psia ≤ Prcs ≤ 3000 psia, ΔP = -186 psi
ΔT = +25.6°F

ART
1/4t = 65.4°F
3/4t = 54.0°F

FIGURE 3

WATERFORD UNIT 3
APPENDIX G BELTLINE P-T LIMITS, HYDROSTATIC



0 psia < Prcs < 200 psia, ΔP = -186 psi
200 psia ≤ Prcs < 850 psia, ΔP = -100 psi
850 psia ≤ Prcs ≤ 3000 psia, ΔP = -186 psi
ΔT = +25.6°F

ART
1/4t = 65.4°F
3/4t = 54.0°F

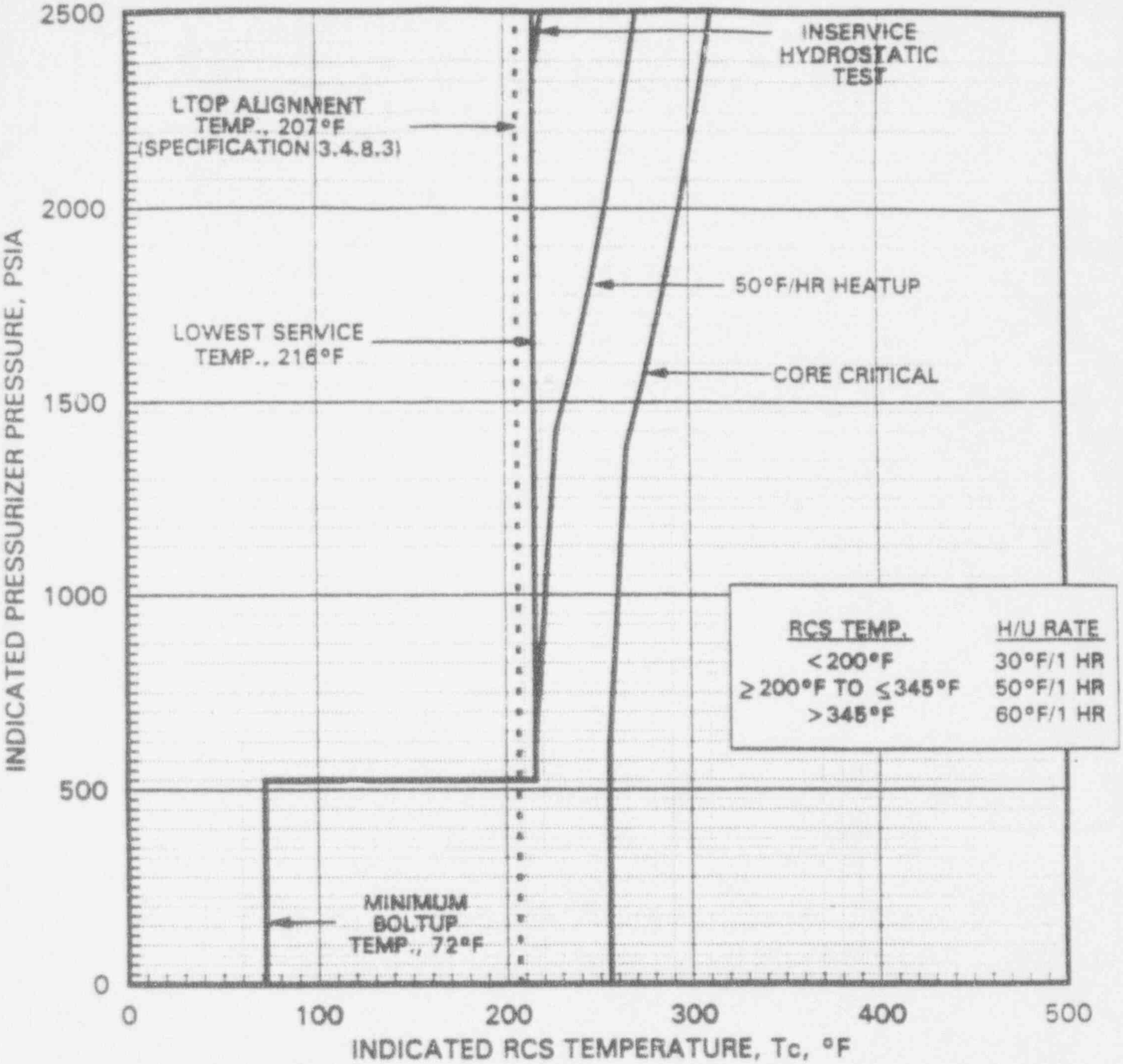


FIGURE 3.4-2

WATERFORD UNIT 3 HEATUP CURVE

REACTOR COOLANT SYSTEM PRESSURE-TEMPERATURE LIMITS

0 - 20 EFPY (PEAK SURFACE FLUENCE = 2.29×10^{19} n/cm^2)

3/4 4-30

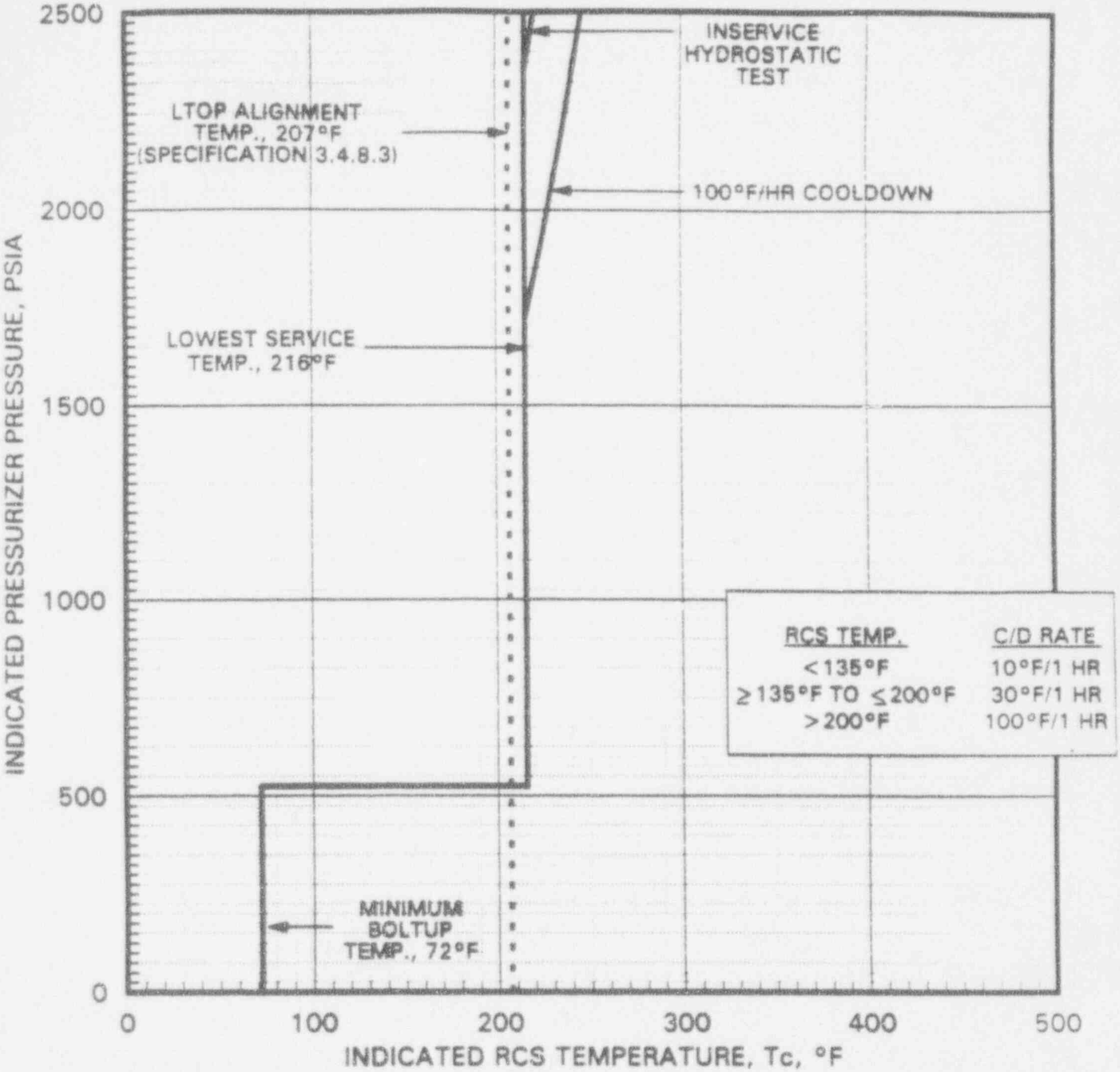


FIGURE 3.4-3

WATERFORD UNIT 3 COOLDOWN CURVE

REACTOR COOLANT SYSTEM PRESSURE-TEMPERATURE LIMITS

0 - 20 EFPY (PEAK SURFACE FLUENCE = 2.29×10^{19} n/cm^2)

8.0 REFERENCES

1. Code of Federal Regulations, 10 CFR Part 50, Appendix G, "Fracture Toughness Requirements", dated August 31, 1992.

2. Code of Federal Regulations, 10 CFR Part 50, Appendix H, "Reactor Vessel Material Surveillance Program Requirements," dated April 30, 1993.

3. Report No. BAW-2177, "Analysis of Capsule W-97", B&W Nuclear Services Company, dated November, 1992.

4. R. Burski to T. Murley, "Waterford 3 SES, Docket No. 50-382, License No. NPF-38, 10 CFR 50 Appendix H.III.A - Reactor Vessel Material Surveillance Program Requirements - Report of Test Results," dated November 25, 1992.

5. Waterford 3 Technical Specifications, Amendment No. 84, Section 3/4.4.8, Reactor Coolant System Pressure Temperature Limits.

6. Code of Federal Regulations, 10 CFR Part 50, Appendix A, "General Design Criteria for Nuclear Power Plants", dated January 1988.

7. ASME Boiler and Pressure Vessel Code Section III, Appendix G, "Protection Against Nonductile Failure", 1989 Edition.

8. Regulatory Guide 1.99, "Radiation Embrittlement of Reactor Vessel Materials", U.S. Nuclear Regulatory Commission, Revision 2, May 1988.

9. Instruction Manual, Reactor Vessel Assembly, Waterford III, Louisiana Power and Light, C.E. Book No. 74170, Vol. I, dated April, 1977.

10. ASME Boiler and Pressure Vessel Code, Section III, Appendix I, "Design Stress Intensity Values, Allowable Stresses, Material Properties, and Design Fatigue Curves", 1989 Edition.

11. Fundamentals of Heat and Mass Transfer, Incropera and DeWitt, 2nd Edition, Copyright 1985.

12. Revised FSAR Table 5.3-13, License Document Change Request Form, LDCR No. 93-0001, dated August 21, 1992.

13. ABB/CE Report No. C-MECH-ER-014, Rev. 00, "An Assessment of the Waterford 3 Reduction in Operating Temperature on NSSS Structural Integrity," dated September 1993.

14. R. O'Quinn to C. Stewart, "Design Input and Assumptions, Entergy Contract No. W-1068-0505 ESR No. C-93-003, Revision of Pressure-Temperature Limits".

15. "Semi-Elliptical Cracks in a Cylinder Subjected to Stress Gradients", J. Heliot, R.C. Labbens and Pellisser - Tanon ASTM Special Technical Publication 677, August 1979.

16. ASME Boiler and Pressure Vessel Code, Section XI, "Rules for Inservice Inspection of Nuclear Power Plant Components," 1989 Edition.

17. ASME Boiler and Pressure Vessel Code, Section III, "Rules for Construction of Nuclear Power Plant Components," 1989 Edition.

18. U.S. NRC Standard Review Plan 5.2.2, "Overpressure Protection," Revision 2, dated November 1988.

19. ASME Boiler and Pressure Vessel Code Case N-514, "Low Temperature Overpressure Protection, Section XI, Division 1", February 12, 1992.

20. 10 CFR 50.61, "Fracture Toughness Requirements for Protection Against Pressurized Thermal Shock Events", August 31, 1992.

21. Report No. C-NLM-003, Rev. 1, "Program for Irradiation Surveillance of Waterford Unit Three Reactor Vessel Materials," Combustion Engineering Inc., dated October 30, 1974.

22. "Waterford 3 SES Updated Final Safety Analysis Report," Docket No. 50-382 Operating License NPF-38, Controlled Copy No. 222

23. ASTM Designation E 185-82, "Standard Practice for Conducting Surveillance Tests for Light Water Cooled Nuclear Power Reactor Vessels," Annual Book of ASTM Standards, Vol. 12.02, American Society for Testing and Materials, Philadelphia, PA.

24. ASTM Designation E 185 (Revision approved in June 1993), "Standard Practice for Conducting Surveillance Tests for Light Water Cooled Nuclear Power Reactor Vessels," Annual Book of ASTM Standards, Vol. 12.02, American Society for Testing and Materials, Philadelphia, PA.

APPENDIX A

Discussion of the

Development of Technical Specification

Figures 3.4-2 and 3.4-3

TECHNICAL SPECIFICATION FIGURE DEVELOPMENT

Figures A1 and A2 provide a graphical representation of all the limits which are required in the development of the RCS P-T limits. These were developed with heatup and cooldown rates and associated temperature ranges which were consistent with the current Technical Specifications (8 EFPY). These rates and temperature ranges are summarized below.

Heatup

RCS Cold Leg Temperature	HU Rate
<200°F	≤30°F/hr
200°F to 345°F	≤50°F/hr
>345°F	≤60°F/hr

Cooldown

RCS Cold Leg Temperature	HU Rate
>200°F	≤100°F/hr
200°F to 135°F	≤30°F/hr
<135°F	≤10°F/hr

The most limiting pressures were then determined for each temperature range for inclusion in the Technical Specification figures. A description is provided below for heatup and cooldown. Note: No consideration was given to Low Temperature Overpressure Protection (LTOP). However, it can be noted that the 8 EFPY Technical Specification figures provided a lower controlling pressure which should have been the basis for the LTOP setpoint, heatup and cooldown rates (and associated temperature intervals) and LTOP administrative procedures. Consequently, the existing LTOP basis should be adequate.

Heatup

To develop the heatup Technical Specification figure, the most controlling pressure for a given temperature is chosen. Figure A1 provides a graphical depiction of the required limits.

Between the minimum boltup temperature (72°F) and 200°F, the allowable heatup rate has been established as less than or equal to 30°F/hr. Figure A1 shows the beltline limit associated with this temperature range. As depicted, the minimum pressure (525 psia) is more restrictive than the beltline limit. Consequently, the minimum pressure requirement is shown on the Technical Specification figure within this temperature range.

A heatup rate of 50°F/hr is permitted from 200°F to 345°F. Figure A1 shows the beltline P-T limit associated with 50°F/hr. Over the specified temperature range, the reactor vessel flange limit associated with a 50°F/hr is also depicted. Review of Figure A1 shows that below the lowest service temperature (215.6°F) the controlling pressure is associated with the minimum pressure requirements (525 psia). At the lowest service temperature (215.6°F), pressures are permitted to exceed 20% of preservice hydrostatic test. However, design pressure (2500 psia) cannot be obtained until 271.4°F because of brittle fracture requirement associated with the reactor vessel flange region and the reactor vessel beltline region. The reactor vessel flange provides the controlling pressure for temperatures above 215.6°F and up to 227.4°F. The beltline then provides the controlling pressure up to 271.4°F.

At temperatures greater than 345°F, a heatup rate of 60°F/hr is allowed although normal operation pressures are permitted.

For convenience, the requirements for inservice hydrostatic test are provided on the heatup figure. 10 CFR 50 Appendix G requires a temperature of at least 135.6°F with respect to the vessel flange region. However, ASME Code requires a temperature of at least 215.6°F (the lowest service temperature) prior to exceeding the minimum pressure (525 psia). In addition, the beltline controls the pressure until 220.7°F where a pressure of 2500 psia is permitted. The purpose of this curve is solely to establish the permissible temperature for inservice hydrostatic tests in accordance with ASME Code Section XI requirements. This temperature has been conservatively assessed to be 219.8°F. Below this temperature the normal heatup or cooldown P-T limits apply.

The preceding will provide the appropriate P-T limit for normal operation and inservice hydrostatic test.

Utilizing the limit developed for heatup (it is more restrictive than cooldown), core critical limits are established per the requirements of 10 CFR 50 Appendix G. This is depicted in

Figure A1. However, for graphical illustration the curve has been conservatively modified as indicated.

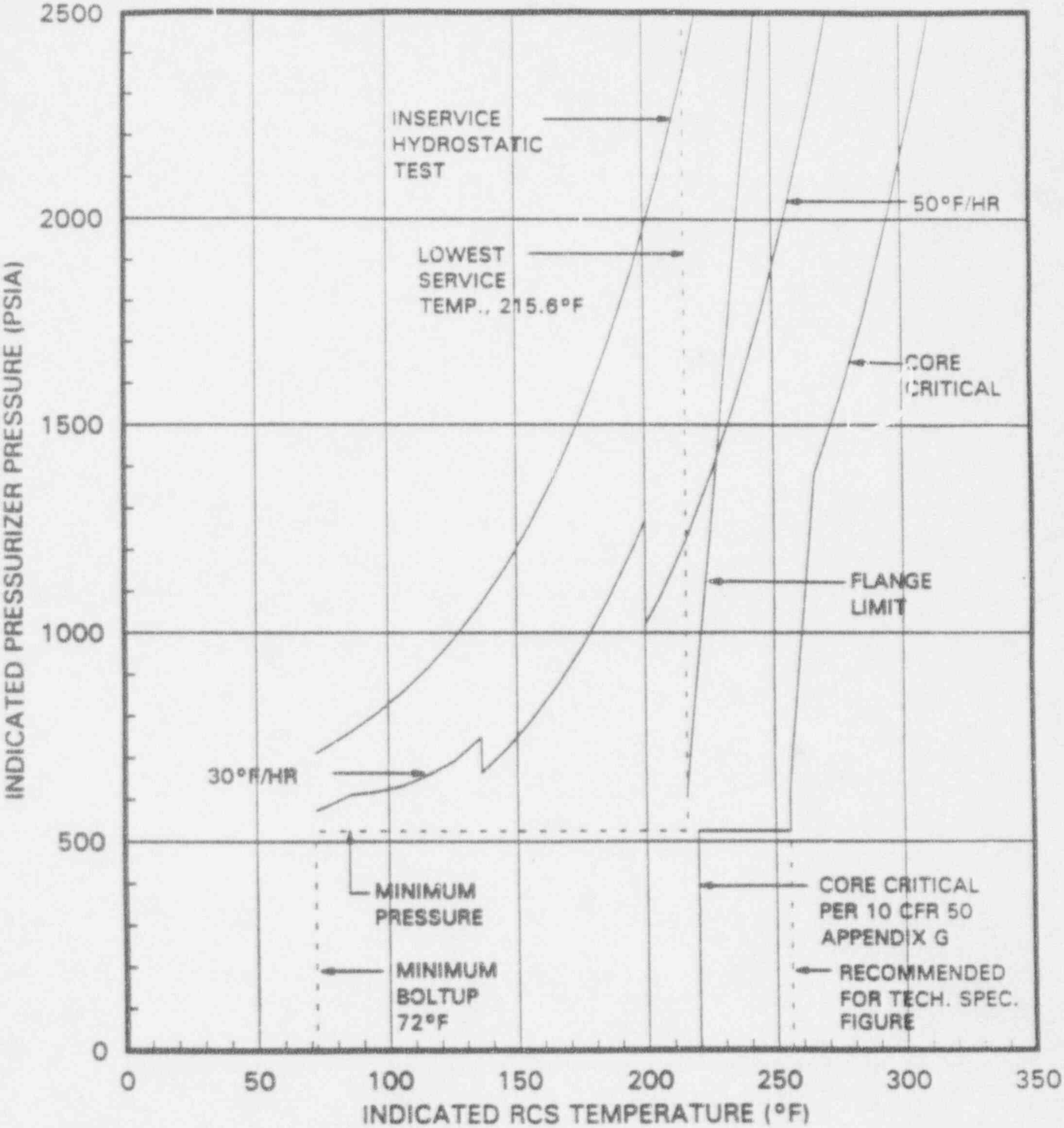
Cooldown

Figure A2 depicts the limits associated with cooldown. Selection of the controlling pressure provides the curve presented in the Technical Specification figure.

As with the heatup, the minimum pressure (525 psia) is controlling with respect to the beltline up to the lowest service temperature (215.6°F). At temperatures greater than the lowest service temperature, the beltline P-T limit associated with 100°F/hr cooldown provides the controlling pressure.

Inservice hydrostatic test limits are also provided and the discussion presented in the heatup section is still valid.

FIGURE A1
WATERFORD UNIT 3
RCS P-T LIMITS, HEATUP



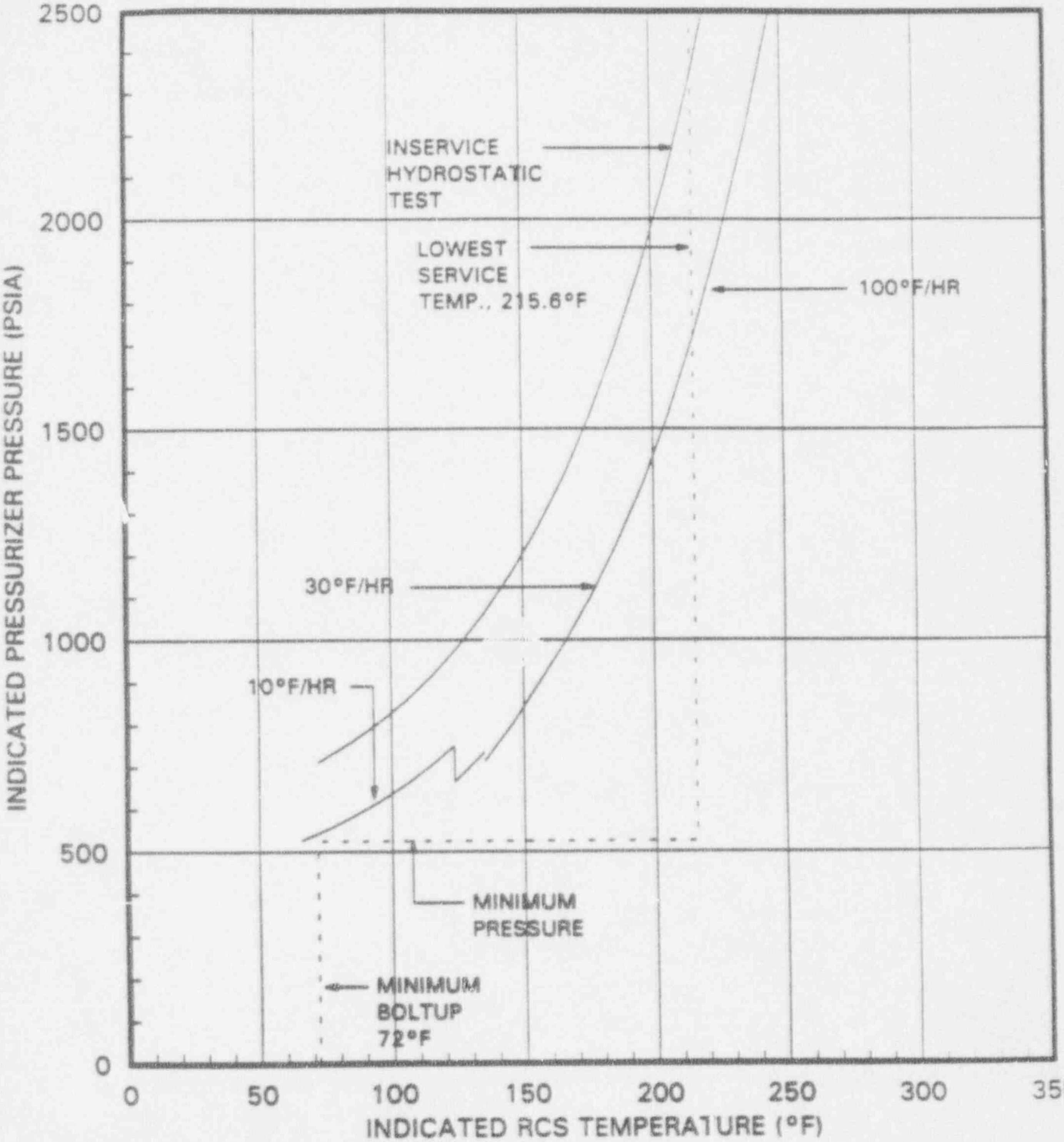
0 psia < Prcs < 200 psia, ΔP = -186 psi
200 psia ≤ Prcs < 850 psia, ΔP = -100 psi
850 psia ≤ Prcs ≤ 3000 psia, ΔP = -186 psi
ΔT = +25.6°F

ART
1/4t = 65.4°F
3/4t = 54.0°F

FIGURE A2

WATERFORD UNIT 3

RCS P-T LIMITS, COOLDOWN



0 psia < Prcs < 200 psia, ΔP = -186 psi
200 psia ≤ Prcs < 850 psia, ΔP = -100 psi
850 psia ≤ Prcs ≤ 3000 psia, ΔP = -186 psi
ΔT = +25.6°F

ART
1/4t = 65.4°F
3/4t = 54.0°F

NPF-38-148

ATTACHMENT D

Final Letter Report

Review of Reactor Vessel Beltline Pressure-
Temperature Limits for 0 - 8 EFPY

ABB

October 11, 1993
C-MECH-93-072

Mr. R.C. O'Quinn
Entergy Operations, Inc.
Waterford Steam Electric Station Unit 3
P.O. Box B
Killona, LA 70066-0751

Subject: **Review of Reactor Vessel Beltline Pressure-Temperature Limits for 0-8 EFPY**

Dear Mr. O'Quinn:

This letter provides a summary of the results pertaining to the assessment of the reactor vessel beltline Pressure-Temperature (P-T) limits associated with the Waterford 3 Reactor Coolant System (RCS) currently documented in Reference 1. The effort was performed under Entergy Contract No. W-1068-0505, ESR C-93-003 and was prompted due to an identified non-conservatism regarding the adjusted reference temperature for the limiting beltline material as documented in Reference 2. The effort was performed in accordance with the CE Nuclear Services Quality Assurance Manual for Quality Class 1 work. The contents of this report have been independently reviewed to insure the accuracy of its contents.

Background

The RCS P-T limits which constitute both the operating and licensing basis of the Waterford Unit 3 plant through eight (8) effective full power years (EFPY) operation are intended to provide protection against brittle fracture for the ferritic pressure boundary components. These limits are based on the materials reference nil ductility transition temperature (RT_{NDT}). With respect to the beltline, the RT_{NDT} is adjusted to account for the loss in ductility experienced by these materials over time as a result of neutron radiation. It is the adjusted reference temperature (ART) that is used in calculating the beltline P-T limits.

ABB Combustion Engineering Nuclear Power

Combustion Engineering, Inc
1000 Prospect Hill Road
Post Office Box 500
Telephone (203) 688-1911
Fax (203) 285-9512

The method for calculating the adjusted reference temperature currently acceptable to the NRC is provided by Regulatory Guide 1.99 Revision 2 (Reference 3). However, this Regulatory Guide was not the standard when the 8 EFPY RCS P-T limits were developed. Application of Regulatory Guide 1.99 Revision 2 provides a higher ART corresponding to the outer diameter crack tip location (3/4t) using the design basis input parameters. Consequently, the ART's were recalculated for the design basis conditions and improved fracture mechanics methods were applied to show the conservatism inherent in beltline P-T limits.

Analysis Highlights

The original basis associated with the beltline P-T limits were reviewed and the limiting material was shown to be Plate M-1004-2. The following information represents the design basis input used to predict the ART and was consistent with the current information (Reference 4) associated with this plate material.

Plate M-1004-2
Initial RT_{NDT} = 22°F
Copper Content = 0.03 w%
Nickel Content = 0.58 w%

As documented in the design basis calculation, a peak surface fluence of 9.2×10^{18} n/cm^2 was utilized to predict the ART values at the 1/4t and 3/4t locations, 97°F and 30°F respectively. Strict application of Regulatory Guide 1.99 Revision 2 provides 1/4t and 3/4t ART values of 55.2°F and 44.6°F.

Comparison of these values show the ART at the 3/4t location to be non-conservative. The significance of this is that in developing the beltline P-T limits, the 3/4t location can provide the controlling pressure during a heatup transient. Recalculation of the beltline limits using the same method would likely provide more restrictive beltline limits. Consequently, the beltline limits were recalculated using improved fracture mechanics methods to show that the basis beltline curves were conservative and in compliance with the requirements of Appendix G to 10 CFR 50 (Reference 5).

Appendix G to 10 CFR 50 requires that the beltline P-T limits be calculated using the guidance of ASME Boiler and Pressure Vessel Code, Section III, Appendix G (Reference 6, referred to herein as ASME Code Appendix G). ABB/CE has developed an improved fracture mechanics methodology which meets the requirements of Appendix G to 10 CFR 50 and the ASME Code Appendix G guidance. This method employs influence coefficients and the principles of superposition to determine the stress intensity factors resulting from internal pressure and thermal loads (K_{IM} and K_{IT}, respectively). A transient convection/conduction heat transfer analysis is performed utilizing a one-dimensional isoparametric finite element model. The heat transfer results are utilized to determine the allowable material stress intensity factor, K_{IR} and the applied thermal stress intensity factor, K_{IT}, as a function of time. The allowable pressure stress intensity factor is determined and subsequently the allowable pressure. The results of this evaluation are allowable pressure versus temperature for the analyzed thermal rate.

To assure the limiting condition is achieved the isothermal condition is also compared to the analyzed rate conditions. In this instance, the rates associated with the Waterford Unit 3 Technical Specification (Reference 1) were analyzed.

To properly index the limits in terms of indicated pressurizer pressure and indicated cold leg temperature, it is necessary to consider the elevation difference between the reactor vessel and the pressurizer, the flow induced pressure losses between the reactor vessel inlet nozzle and the surge line nozzle, and include the effects of instrument uncertainty. The original design basis values were utilized.

The beltline P-T limits were compared to the current Waterford Unit 3 Technical Specification heatup figure to assess whether the current beltline curves were adequate in providing the requisite protection from brittle fracture.

Results

The ART values associated with the limiting reactor vessel beltline material (Plate M-1004-2) through 8 EFPY were calculated for the 1/4t and 3/4t locations in accordance with Regulatory Guide 1.99 Revision 2 as 55.2°F and 44.6°F respectively. The beltline P-T

limits were calculated utilizing the ART values for the limiting beltline material and other pertinent design basis information. Through the use of improved fracture mechanics methods the beltline P-T limits were re-evaluated for Waterford Unit 3. The comparison of the design basis beltline limits, as provided by the Waterford 3 Technical Specifications, to those calculated with the improved fracture mechanics techniques are provided in Figure 1.

Conclusions

Review of the limiting beltline material ART values at the 1/4t and 3/4t locations showed the 3/4t value to be non-conservative when predicted using the guidance provided by Regulatory Guide 1.99 Revision 2. While the reactor vessel beltline 3/4t location can provide the controlling pressure during heatup, re-evaluation of the beltline P-T limits using improved fracture mechanics techniques showed the current basis provided by the Waterford Unit 3 Technical Specifications to be more restrictive. Since the applied method used to re-evaluate the beltline meets the requirements of 10 CFR 50 Appendix G, the reactor vessel beltline P-T limits provided by the Technical Specification have provided the requisite margins of safety in light of the non-conservative ART.

If you should have any questions regarding this report, please contact me at (203)-285-2294 or Mr. Carl Gimbrone at (203)-285-2567.

Sincerely,

COMBUSTION ENGINEERING, INC.

Craig D Stewart

Craig Stewart
Project Engineer

cc: G. Bundick
C. Gimbrone

References

(1) Waterford Unit 3 Technical Specifications, Section 3/4.4.8, Pressure/Temperature Limits, Amendment No. 84.

(2) Letter No. C-MECH-92-079, "Review of Waterford 3 Appendix G RCS Pressure-Temperature Limits," C.D. Stewart, dated Nov. 11, 1992.

(3) Regulatory Guide 1.99 Revision 2, "Radiation Embrittlement of Reactor Vessel Materials," dated May 1988.

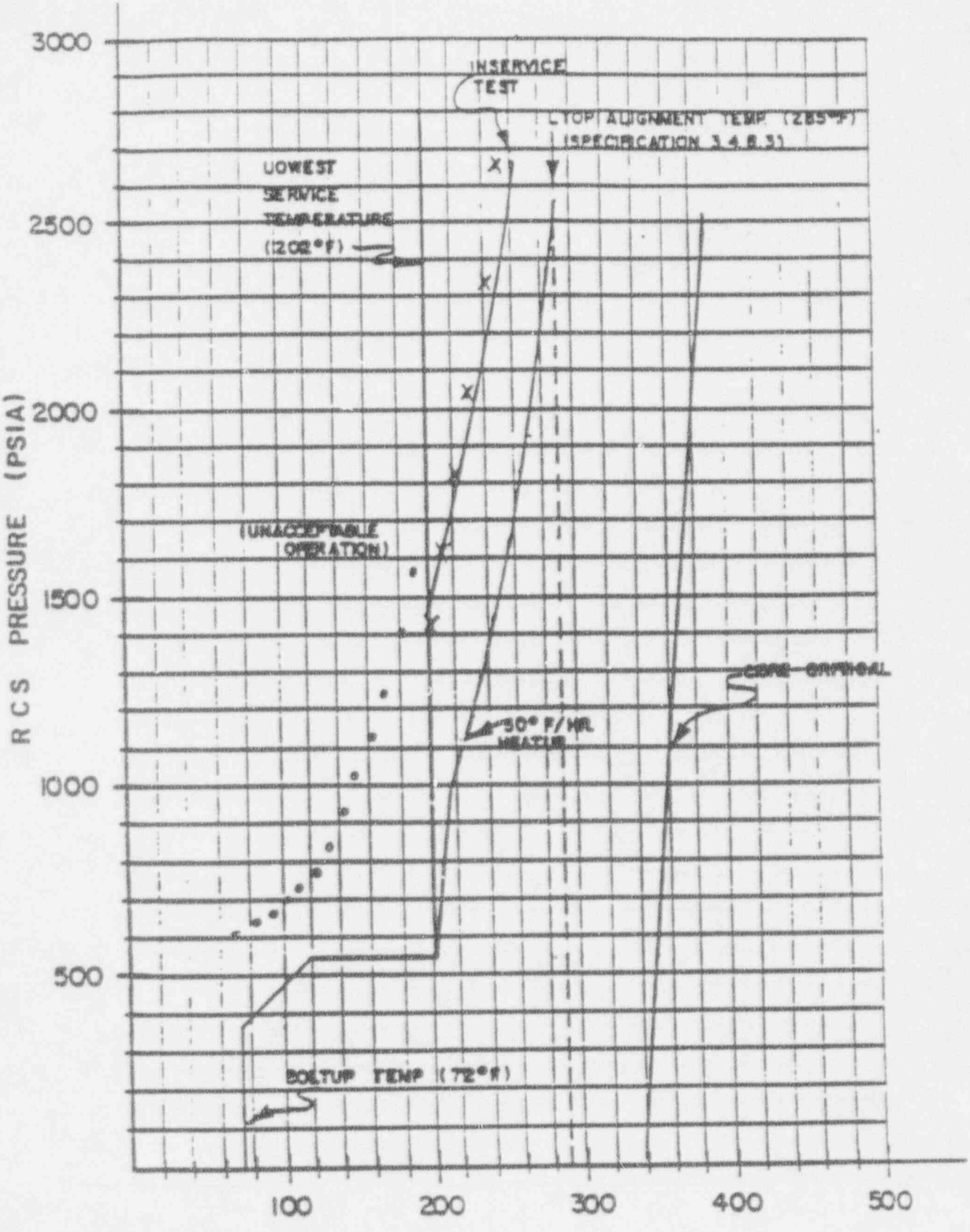
(4) "Updated Final Safety Analysis Report," Waterford 3 SES, Docket No. 50-382, Operating License NPF-38, Controlled Copy No. 222 (As amended by LDCR No. 93-0001, J. B. Perez, dated August 21, 1992).

(5) 10 CFR 50 Appendix G, "Fracture Toughness Requirements," dated August 31, 1992.

(6) ASME Boiler and Pressure Vessel Code, Section III, Appendix G, "Protection Against Nonductile Failure," 1989 Edition.

FIGURE 1

Beltline Limits for 8 EFPY, Heatup



• - 30°F/hr heatup
X - 50°F/hr heatup