ATTACHMENT 2

PROPOSED TECHNICAL SPECIFICATIONS CHANGES

NORTH ANNA UNITS 1 AND 2

VIRGINIA ELECTRIC AND POWER COMPANY

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LIMITING CONDITIONS FOR OPERATION AND SURVEILLANCE REQUIREMENTS

BASES

REACTIVITY CONTROL SYSTEMS

FLOW PATHS – OPERATING

LIMITING CONDITION FOR OPERATION

3.1.2.2 Each of the following boron injection flow paths shall be OPERABLE:

a. The flow path from the boric acid tanks via a boric acid transfer pump and a charging pump to the Reactor Coolant System, and

b. The flow path from the refueling water storage tank via a charging pump to the Reactor Coolant System.

APPLICABILITY: MODES 1, 2, 3 AND 4#.

ACTION:

a. With the flow path from the boric acid tanks inoperable, restore the inoperable flow path to OPERABLE status within 72 hours or be in at least HOT STANDBY and borated to a SHUTDOWN MARGIN equivalent to at least 1.77% $\Delta k/k$ at 200°F within the next 6 hours; restore the flow path to OPERABLE status within the next 7 days or be in COLD SHUTDOWN within the next 30 hours.

b. With the flow path from the refueling water storage tank inoperable, restore the flow path to OPERABLE status within one hour or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.1.2.2 Each of the above required flow paths shall be demonstrated OPERABLE:

a. At least once per 7 days by verifying that the temperature of the heat traced portion of the flow path from the boric acid tanks is ≥ 115°F.

Only one boron injection flow path is required to be OPERABLE whenever the temperature of one or more of the RCS cold legs is less than or equal to 235°F.

CHARGING PUMPS – OPERATING

LIMITING CONDITION FOR OPERATION

3.1.2.4 At least two charging pumps shall be OPERABLE.

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

ACTION:

With only one charging pump OPERABLE, restore a second charging pump to OPERABLE status within 72 hours or be in at least HOT STANDBY and borated to a SHUTDOWN MARGIN equivalent to at least 1.77% $\Delta k/k$ at 200°F within the next 6 hours; restore a second charging pump to OPERABLE status within the next 7 days or be in COLD SHUTDOWN within the next 30 hours. The provisions of Specification 3.0.4 are not applicable for one hour following heatup above 235°F or prior to cooldown below 235°F.

SURVEILLANCE REQUIREMENTS

4.1.2.4.1 The above required charging pumps shall be demonstrated OPERABLE by verifying, that on recirculation flow, each pump develops a discharge pressure of ≥ 2410 psig when tested pursuant to Specification 4.0.5.

4.1.2.4.2 All charging pumps, except the above required OPERABLE pump, shall be demonstrated inoperable at least once per 12 hours whenever the temperature of one or more of the RCS cold legs is less than or equal to 235°F by verifying that the switches in the Control Room have been placed in the pull to lock position.

* A maximum of one centrifugal charging pump shall be OPERABLE whenever the temperature of one or more of the RCS cold legs is less than or equal to 235°F.

LIMITING CONDITION FOR OPERATION

3.4.1.3 a. At least two of the coolant loops listed below shall be OPERABLE:

1. Reactor Coolant Loop A and its associated steam generator and reactor coolant pump,*
2. Reactor Coolant Loop B and its associated steam generator and reactor coolant pump,*
3. Reactor Coolant Loop C and its associated steam generator and reactor coolant pump,*
4. Residual Heat Removal Subsystem A,**
5. Residual Heat Removal Subsystem B.**

b. At least one of the above coolant loops shall be in operation.***

APPLICABILITY: MODES 4 and 5

ACTION:

a. With less than the above required loops OPERABLE, immediately initiate corrective action to return the required loops to OPERABLE status as soon as possible; be in COLD SHUTDOWN within 20 hours.

b. With no coolant 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.

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* A reactor coolant pump shall not be started with one or more of the RCS cold leg temperatures less than or equal to 235°F unless the secondary water temperature of each steam generator is less than 50°F above each of the RCS cold leg temperatures.

** The offsite or emergency power source may be inoperable in MODE 5.

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

SAFETY AND RELIEF VALVES - OPERATING

SAFETY VALVES

LIMITING CONDITION FOR OPERATION

3.4.3.1 All pressurizer code safety valves shall be OPERABLE with a lift setting of 2485 PSIG ± 1%.*

APPLICABILITY: MODES 1, 2 and 3.

ACTION:

With one pressurizer code safety valve inoperable, either restore the inoperable valve to OPERABLE status within 15 minutes or be in HOT SHUTDOWN within 12 hours.

SURVEILLANCE REQUIREMENTS

4.4.3.1 No additional Surveillance Requirements other than those required by Specification 4.0.5.

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

REACTOR COOLANT SYSTEM

SAFETY AND RELIEF VALVES - OPERATING

RELIEF VALVES

LIMITING CONDITION FOR OPERATION

3.4.3.2 Both power-operated relief valves (PORVs) and their associated block valves shall be OPERABLE.

APPLICABILITY: MODES 1, 2, and 3.

ACTION:

a. With one or both PORVs inoperable but capable of being manually cycled, within 1 hour either restore the PORV(s) to OPERABLE status or close the associated block valve(s) with power maintained to the block valve(s); otherwise, be in at least HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours.

b. With one PORV inoperable and not capable of being manually cycled, within 1 hour either restore the PORV to OPERABLE status or capable of being manually cycled, or close its associated block valve and remove power from the block valve; restore the PORV to OPERABLE status within the following 72 hours or be in HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours.

c. With both PORVs inoperable and not capable of being manually cycled, within 1 hour either restore at least one PORV to OPERABLE status or capable of being manually cycled, or close its associated block valve and remove power from the block valve and be in HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours.

d. With one or both block valves inoperable, within 1 hour restore the block valve(s) to OPERABLE status or place its associated PORV(s) in manual control. Restore at least one block valve to OPERABLE status within the next hour if both block valves are inoperable; restore the remaining inoperable block valve to OPERABLE status within 72 hours; otherwise, be in at least HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours.

e. The provisions of Specification 3.0.4 are not applicable.

REACTOR COOLANT SYSTEM

SAFETY AND RELIEF VALVES – OPERATING

RELIEF VALVES

SURVEILLANCE REQUIREMENTS

4.4.3.2.1 In addition to the requirements of Specification 4.0.5, each PORV shall be demonstrated OPERABLE:

a. At least once per 31 days by performing a CHANNEL FUNCTIONAL TEST, excluding valve operation, and

b. At least once per 18 months by

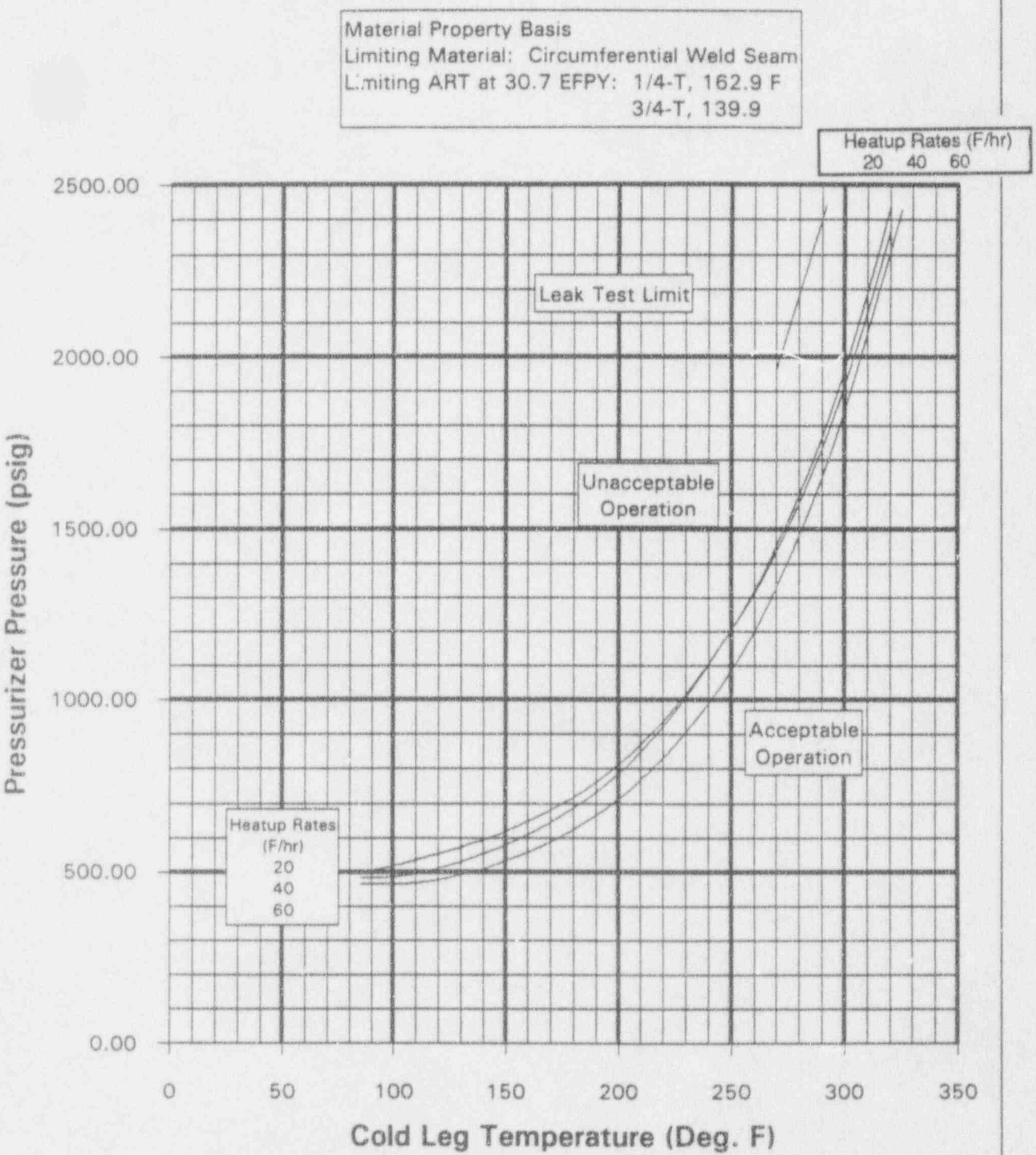
   1. Operating the PORV through one complete cycle of full travel during MODES 3 or 4, and

   2. Operating the solenoid air control valves and check valves on the associated accumulators in the PORV control systems through one complete cycle of full travel, and

   3. Performing a CHANNEL CALIBRATION of the actuation instrumentation.

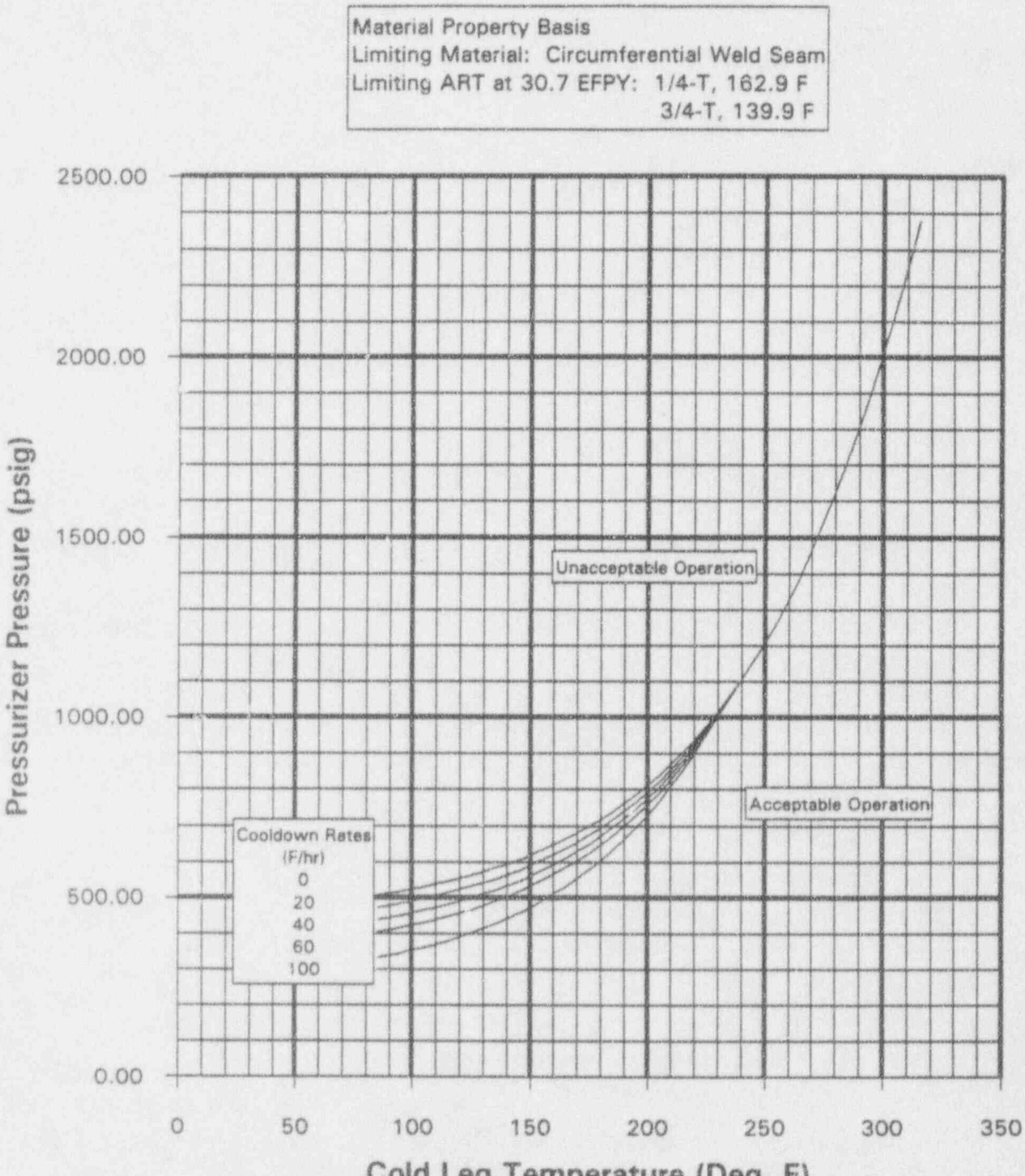
4.4.3.2.2 Each block valve shall be demonstrated OPERABLE at least once per 92 days by operating the valve through one complete cycle of full travel unless the block valve is closed in order to meet the requirements of ACTION b or c in Specification 3.4.3.2.

Figure 3.4-2 — **North Anna Unit 1**
Reactor Coolant System Heatup Limitations



North Anna Unit 1 Reactor Coolant System Heatup Limitations (Heatup Rates up to 60 F/hr) Applicable for the First 30.7 EFPY (Without Margins for Instrumentation Errors)

Figure 3.4-3 — **North Anna Unit 1**
Reactor Coolant System Cooldown Limitations



North Anna Unit 1 Reactor Coolant System Cooldown Limitations (Cooldown Rates up to 100 F/hr) Applicable for the First 30.7 EFPY (Without Margins for Instrumentation Errors)

LIMITING CONDITION FOR OPERATION

3.4.9.3 Two power-operated relief valves (PORVs) shall be OPERABLE with lift settings of (1) less than or equal to 500 psig whenever any RCS cold leg temperature is less than or equal to 235°F, and (2) less than or equal to 395 psig whenever any RCS cold leg temperature is less than 150°F.

APPLICABILITY: MODE 4 when the temperature of any RCS cold leg is less than or equal to 235°F, MODE 5, and MODE 6 when the head is on the reactor vessel and the RCS is not vented through a 2.07 square inch or larger vent.

ACTION:

a. With one PORV inoperable in MODE 4, restore the inoperable PORV to OPERABLE status within 7 days or depressurize and vent the RCS through at least a 2.07 square inch vent within the next 8 hours.

b. With one PORV inoperable in MODES 5 or 6, either (1) restore the inoperable PORV to OPERABLE status within 24 hours, or (2) complete depressurization and venting of the RCS through at least a 2.07 square inch vent within a total of 32 hours.

c. With both PORVs inoperable, complete depressurization and venting of the RCS through at least a 2.07 square inch vent within 8 hours.

d. With the RCS vented per ACTIONS a, b, or c, verify the vent pathway at least once per 31 days when the pathway is provided by a valve(s) that is locked, sealed, or otherwise secured in the open position; otherwise, verify the vent pathway ever 12 hours.

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

f. The provisions of Specification 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.4.9.3 Each PORV shall be demonstrated OPERABLE by:

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

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

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

d. Testing pursuant to Specification 4.0.5.

ECCS SUBSYSTEMS - Tavg ≥ 350°F

LIMITING CONDITION FOR OPERATION

3.5.2 Two independent ECCS subsystems shall be OPERABLE with each subsystem comprised of:

a. One OPERABLE centrifugal charging pump,

b. One OPERABLE low head safety injection pump,

c. An OPERABLE flow path capable of transferring fluid to the Reactor Coolant System when taking suction from the refueling water storage tank on a safety injection signal or from the containment sump when suction is transferred during the recirculation phase of operation or from the discharge of the outside recirculation spray pump.

APPLICABILITY: MODES 1, 2 and 3.

ACTION:

a. With one ECCS subsystem inoperable, restore the inoperable subsystem to OPERABLE status within 72 hours or be in HOT SHUTDOWN within the next 12 hours.

b. In the event the ECCS is actuated and injects water into the Reactor Coolant System, a Special Report shall be prepared and submitted to the Commission pursuant to Specification 6.9.2 within 90 days describing the circumstances of the actuation and the total accumulated actuation cycles to date.

c. The provisions of Specification 3.0.4 are not applicable to 3.5.2.a and 3.5.2.b for one hour following heatup above 235°F or prior to cooldown below 235°F.

ECCS SUBSYSTEMS – Tavg < 350°F

LIMITING CONDITION FOR OPERATION

3.5.3 As a minimum, one ECCS subsystem comprised of the following shall be OPERABLE:

a. One OPERABLE centrifugal charging pump#,

b. One OPERABLE low head safety injection pump#, and

c. An OPERABLE flow path capable of automatically transferring fluid to the reactor coolant system when taking suction from the refueling water storage tank or from the containment sump when the suction is transferred during the recirculation phase of operation or from the discharge of the outside recirculation spray pump.

APPLICABILITY: MODE 4.

ACTION:

a. With no ECCS subsystem OPERABLE because of the inoperability of either the centrifugal charging pump or the flow path from the refueling water storage tank, restore at least one ECCS subsystem to OPERABLE status within 1 hour or be in COLD SHUTDOWN within the next 20 hours.

b. With no ECCS subsystem OPERABLE because of the inoperability of the low head safety injection pump, restore at least one ECCS subsystem to OPERABLE status or maintain the Reactor Coolant System T_{avg} less than 350°F by use of alternate heat removal methods.

c. In the event the ECCS is actuated and injects water into the Reactor Coolant System, a Special Report shall be prepared and submitted to the Commission pursuant to Specification 6.9.2 within 90 days describing the circumstances of the actuation and the total accumulated actuation cycles to date.

\# A maximum of one centrifugal charging pump and one low head safety injection pump shall be OPERABLE whenever the temperature of one or more of the RCS cold legs is less than or equal to 235°F.

SURVEILLANCE REQUIREMENTS

4.5.3.1 The ECCS subsystem shall be demonstrated OPERABLE per the applicable Surveillance Requirements of 4.5.2.

4.5.3.2 All charging pumps and safety injection pumps, except the above required OPERABLE pumps, shall be demonstrated inoperable at least once per 12 hours whenever the temperature of one or more of the RCS cold legs is less than or equal to 235°F by verifying that the switches in the Control Room are in the pull to lock position.

3/4.1.2 BORATION SYSTEMS (Continued)

With the RCS average temperature above 200°F, a minimum of two separate and redundant boron injection systems are provided to ensure single functional capability in the event an assumed failure renders one of the systems inoperable. Allowable out-of-service periods ensure that minor component repair or corrective action may be completed without undue risk to overall facility safety from injection system failures during the repair period.

The boration capability of either system is sufficient to provide a SHUTDOWN MARGIN from expected operating conditions of 1.77% $\Delta k/k$ after xenon decay and cooldown to 200°F. This expected boration capability requirement occurs at EOL from full power equilibrium xenon conditions and requires 6,000 gallons of 12,950 ppm borated water from the boric acid storage tanks or 54,200 gallons of 2300 ppm borated water from the refueling water storage tank.

The limitation for a maximum of one centrifugal charging pump to be OPERABLE and the Surveillance Requirement to verify all charging pumps except the required OPERABLE pump to be inoperable below 235°F provides assurance that a mass addition pressure transient can be relieved by the operation of a single PORV.

With the RCS temperature below 200°F, one injection system is acceptable without single failure consideration on the basis of the stable reactivity condition of the reactor and the additional restrictions prohibiting CORE ALTERATIONS and positive reactivity change in the event the single injection system becomes inoperable.

The boron capability required below 200°F is sufficient to provide a SHUTDOWN MARGIN of 1.77% $\Delta k/k$ after zenon decay and cooldown from 200°F to 140°F. This condition requires either 1378 gallons of 12,950 ppm borated water from the boric acid storage tanks or 3400 gallons of 2300 ppm borated water from the refueling water storage tank.

The contained water volume limits include allowance for water not available because of discharge line location and other physical characteristics. The OPERABILITY of one boron injection system during REFUELING insures that this system is available for reactivity control while in MODE 6.

3/4.4.1 REACTOR COOLANT LOOPS

The plant is designed to operate with all reactor coolant loops in operation and maintain the DNBR above the design limit during all normal operations and anticipated transients. In MODES 1 and 2 with one reactor coolant loop not in operation, this specification requires that the plant be in at least HOT STANDBY within 1 hour.

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

In MODES 4 and 5, a single reactor coolant loop or RHR loop provides sufficient heat removal capability for removing decay heat, but single failure considerations require that at least two loops be OPERABLE. Thus, if the reactor coolant loops are not OPERABLE, this specification requires two RHR loops to be OPERABLE.

After the reactor has shutdown and entered into MODE 3 for at least 100 hours, a minimum RHR system flow rate of 2000 gpm in MODE 5 is permitted, provided there is sufficient decay heat removal to maintain the RCS temperature less than or equal to 140°F. Since the decay heat power production rate decreases with time after reactor shutdown, the requirements for RHR system decay heat removal also decrease. Adequate decay heat removal is provided as long as the reactor has been shutdown for at least 100 hours after entry into MODE 3 and RHR flow is sufficient to maintain the RCS temperature less than or equal to 140°F. The reduced flow rate provides additional margin to vortexing at the RHR pump suction while in Mid Loop Operation. During a reduction in reactor coolant system boron concentration the Specification 3.1.1.3.1 requirement to maintain 3000 gpm flow rate provides sufficient coolant circulation to minimize the effect of a boron dilution incident and to prevent boron stratification.

The restrictions on starting a Reactor Coolant Pump with one or more RCS cold legs less than or equal to 235°F are provided to prevent RCS pressure transients, caused by energy additions from the secondary system which could exceed the limits of Appendix G to 10 CFR Part 50. The RCS will be protected against overpressure transients and will not exceed the limits of Appendix G by restricting starting of the RCPs to when the secondary water temperature of each steam generator is less than 50°F above each of the RCS cold leg temperatures.

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

The requirement to maintain the boron concentration of an isolated loop greater than or equal to the boron concentration of the operating loops ensures that no reactivity addition to the core could occur during startup of an isolated loop. Verification of the boron concentration in an idle loop prior to opening the cold leg stop valve provides a reassurance of the adequacy of the boron concentration in the isolated loop. Operating the isolated loop on recirculating flow for at least 90 minutes prior to opening its cold leg stop valve ensures adequate mixing of the coolant in this loop and prevents any reactivity effects due to boron concentration stratification.

Startup of an idle loop will inject cool water from the loop into the core. The reactivity transient resulting from this cool water injection is minimized by delaying isolated loop startup until its temperature is within 20°F of the operating loops. Making the reactor subcritical prior to loop startup prevents any power spike which could result from this cool water induced reactivity transient.

3/4.4.2 AND 3/4.4.3 SAFETY AND RELIEF VALVES

The pressurizer code safety valves operate to prevent the RCS from being pressurized above its Safety Limit of 2735 psig. Each safety valve is designed to relieve 380,000 lbs per hour of saturated steam at the valve set point. The relief capacity of a single safety valve is adequate to relieve any overpressure condition which could occur during hot shutdown. In the event that no safety valves are OPERABLE, an operating RHR loop, connected to the RCS, or the power operated relief valves (PORVs) will provide overpressure relief capability and will prevent RCS overpressurization.

During operation, all pressurizer code safety valves must be OPERABLE to prevent the RCS from being pressurized above its safety limit of 2735 psig. The combined relief capacity of all of these valves is greater than the maximum surge rate resulting from a complete loss of load assuming no reactor trip until the first Reactor Protective System trip setpoint is reached (i.e., no credit is taken for a direct reactor trip on the loss of load) and also assuming no operation of the power operated relief valves or steam dump valves.

Demonstration of the safety valves' lift settings will occur only during shutdown and will be performed in accordance with the provisions of Section XI of the ASME Boiler and Pressure Vessel Code.

The power operated relief valves and steam bubble function to relieve RCS pressure during all design transients up to and including the design step load decrease with steam dump. Operation of the power operated relief valves minimizes the undesirable opening of the spring-loaded pressurizer code safety valves. Each PORV has a remotely operated block valve to provide a positive shutoff capability should a relief valve become inoperable.

The OPERABILITY of the PORVs and block valves is determined on the basis of their being capable of performing the following functions:

a) Manual control of PORVs to control reactor coolant system pressure. This is a function that may be used to mitigate certain accidents and for plant shutdown.

b) Maintaining the integrity of the reactor coolant pressure boundary. This function is related to controlling identified leakage and ensuring the ability to detect unidentified reactor coolant pressure boundary leakage.

c) Manual control of the block valve to (1) unblock an isolated PORV to allow it to be used for manual control of reactor coolant system pressure (Item a, above), and (2) isolate a PORV with excessive seat leakage (Item b, above).

d) Automatic control of PORVs to control reactor coolant system pressure. This function reduces challenges to the code safety valves for overpressurization events.

e) Manual control of a block valve to isolate a stuck-open PORV.

Surveillance Requirements provide the assurance that the PORVs and block valves can perform their functions. Specification 4.4.3.2.1 addresses the PORVs and Specification 4.4.3.2.2 addresses the block valves. The block valves are exempt from the surveillance requirements to cycle the valves when they have been closed to comply with the ACTION requirements. This precludes the need to cycle the valves with full system differential pressure or when maintenance is being performed to restore an inoperable PORV to operable status.

Surveillance Requirement 4.4.3.2.1.b.2 provides for the testing of the mechanical and electrical aspects of control systems for the PORVs.

Testing of PORVs in HOT STANDBY or HOT SHUTDOWN is required in order to simulate the temperature and pressure environmental effects on PORVs. Testing at COLD SHUTDOWN is not considered to be a representative test for assessing PORV performance under normal plant operating conditions.

3/4.4.4 PRESSURIZER

The limit on the maximum water volume in the pressurizer assures that the parameter is maintained within the normal steady state envelope of operation assumed in the SAR. The limit is consistent with the initial SAR assumptions. The 12 hour periodic surveillance is sufficient to ensure that the parameter is restored to within its limit following expected transient operation. The maximum water volume also ensures that a steam bubble is formed and thus the RCS is not a hydraulically solid system.

The ACTION statement permitting POWER OPERATION to continue for limited time periods with the primary coolant's specific activity > 1.0 ~μCi/gram DOSE EQUIVALENT I-131, but within the allowable limit shown on Figure 3.4-1, accommodates possible iodine spiking phenomenon which may occur following changes in THERMAL POWER.

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

3/4.4.9 PRESSURE / TEMPERATURE LIMITS

Reactor Coolant System

All components in the Reactor Coolant System are designed to withstand the effects of cyclic loads due to system temperature and pressure changes. These cyclic loads are introduced by normal load transients, reactor trips, and startup and shutdown operations. The various categories of load cycles used for design purposes are provided in Section 5.2 of the UFSAR. During startup and shutdown, the rates of temperature and pressure changes are limited so that the maximum specified heatup and cooldown rates are consistent with the design assumptions and satisfy the stress limits for cyclic operation.

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

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

The heatup limit curve, Figure 3.4-2, is a composite curve which was prepared by determining the most conservative case, with either the inside or outside wall controlling, for any heatup rate up to 60°F per hour. The cooldown limit curves of Figure 3.4-3 are composite curves which were prepared based upon the same type analysis with the exception that the controlling location is always the inside wall where the cooldown thermal gradients tend to produce tensile stresses while producing compressive stresses at the outside wall. The heatup and cooldown curves were prepared based upon the most limiting value of the predicted adjusted reference temperature at the end of 30.7 EFPY. The most recent capsule analysis results are documented in Westinghouse Report WCAP-11777, February 1988. The heatup and cooldown curves are documented in Westinghouse Report WCAP-13831, Rev. 1, August 1993.

The reactor vessel materials have been tested to determine their initial RT_{NDT}. Reactor operation and resultant fast neutron (E > 1 Mev) irradiation will cause an increase in the RT_{NDT}. An adjusted reference temperature, based upon the fluence and copper content of the material in question, can be predicted using US NRC Regulatory Guide 1.98, Revision 2. The heatup and cooldown limit curves (Figure 3.4-2 and Figure 3.4-3) include predicted adjustments for this shift in RT_{NDT} at the end of 30.7 EFPY. The reactor vessel beltline region material properties are listed on Figures 3.4-2 and 3.4-3.

The actual shift in the RT_{NDT} of the vessel material will be established periodically by removal and evaluation of the reactor vessel material specimens installed on the inside wall of the thermal shield. The surveillance capsule withdrawal schedule was prepared in accordance with the requirements of ASTM E-185 and is presented in the UFSAR. Regulatory Guide 1.99, Revision 2, provides guidance for calculation of the shift in RT_{NDT} using measured data. Dosimetry from the surveillance capsule is used to determine the neutron fluence to which the material specimens were exposed, and to support calculational estimates of the neutron fluence to the reactor vessel.

The pressure-temperature limit lines shown on Figure 3.4-2 for inservice leak and hydrostatic testing have been provided to assure compliance with the minimum temperature requirements of Appendix G to 10 CFR 50. The minimum temperature for criticality specified in T.S. 3.1.1.5 assures compliance with the criticality limits of 10 CFR 50 Appendix G.

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

Pressurizer

The limitations imposed on pressurizer heatup and cooldown 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.

Low-Temperature Overpressure Protection

The OPERABILITY of two PORVs or an RCS vent opening of greater than 2.07 square inches ensures that the RCS will be protected from pressure transients which could exceed the limits of Appendix G to 10 CFR Part 50 when one or more of the RCS cold legs are less than or equal to 235°F. Either PORV has adequate relieving capability to protect the RCS from overpressurization when the transient is limited to either (1) the start of an idle RCP with the secondary water temperature of the steam generator less than or equal to 50°F above the RCS cold leg temperatures or (2) the start of a charging pump and its injection into a water-solid RCS.

Automatic or passive low temperature overpressure protection (LTOP) is required whenever any RCS cold leg temperature is less than 235°F. This temperature is the water temperature corresponding to a metal temperature of at least the limiting RT_{NDT} + 50°F + instrument uncertainty. Above 235°F administrative control is adequate protection to ensure the limits of the heatup curve (Figure 3.4-2) and the cooldown curve (Figure 3.4-3) are not violated. The concept of requiring automatic LTOP at the lower end, and administrative control at the upper end, of the Appendix G curves is further discussed in NRC Generic Letter 88-11.

ECCS SUBSYSTEMS (Continued)

With the RCS temperature below 350°F, one OPERABLE ECCS subsystem is acceptable without single failure consideration on the basis of the stable reactivity condition of the reactor and the limited core cooling requirements.

The limitation for a maximum of one centrifugal charging pump and one low head safety injection pump to be OPERABLE and the Surveillance Requirement to verify all charging pumps and low head safety injection pumps except the required OPERABLE pump to be inoperable below 235°F provides assurance that a mass addition pressure transient can be relieved by the operation of a single PORV.

The Surveillance Requirements provided to ensure OPERABILITY of each component ensures that at a minimum, the assumptions used in the safety analyses are met and that subsystem OPERABILITY is maintained.

3/4.5.4 BORON INJECTION SYSTEM

The OPERABILITY of the boron injection system as part of the ECCS ensures that sufficient negative reactivity is injected into the core to counteract any positive increase in reactivity caused by RCS system cooldown. RCS cooldown can be caused by inadvertent depressurization, a loss-of-coolant accident or a steam line rupture.

The limits on injection tank minimum contained volume and boron concentration ensure that the assumptions used in the steam line break analysis are met.

The OPERABILITY of the redundant heat tracing channels associated with the boron injection system ensure that the solubility of the boron solution will be maintained above the solubility limit of 111°F at 15,750 ppm boron.

SPECIAL REPORTS

6.9.2 Special reports shall be submitted to the Regional Administrator, Region II, within the time period specified for each report. These reports shall be submitted pursuant to the requirement of the applicable specification:

a. Inservice Inspection Reviews, Specification 4.0.5, shall be reported within 90 days of completion.

b. MODERATOR TEMPERATURE COEFFICIENT. Specification 3.1.1.4.

c. RADIATION MONITORING INSTRUMENTATION. Specification 3.3.3.1, Table 3.3-6, Action 35.

d. SEISMIC INSTRUMENTATION. Specifications 3.3.3.3 and 4.3.3.3.2.

e. METEOROLOGICAL INSTRUMENTATION. Specification 3.3.3.4.

f. Deleted.

g. LOOSE PARTS MONITORING SYSTEMS. Specification 3.3.3.9.

h. Deleted.

i. LOW-TEMPERATURE OVERPRESSURE PROTECTION. Specification 3.4.9.3.

j. EMERGENCY CORE COOLING SYSTEMS. Specification 3.5.2 and 3.5.3.

k. SETTLEMENT OF CLASS 1 STRUCTURES. Specification 3.7.12.

l. GROUND WATER LEVEL - SERVICE WATER RESERVOIR. Specification 3.7.13.

m. Deleted.

n. RADIOACTIVE EFFLUENTS. As required by the ODCM.

o. RADIOLOGICAL ENVIRONMENTAL MONITORING. As required by the ODCM.

p. SEALED SOURCE CONTAMINATION. Specification 4.7.11.1.3.

q. REACTOR COOLANT SYSTEM STRUCTURAL INTEGRITY. Specification 4.4.10. For any abnormal degradation of the structural integrity of the reactor vessel or the Reactor Coolant System pressure boundary detected during the performance of Specification 4.4.10, an initial report shall be submitted within 10 days after detection and a detailed report submitted within 90 days after the completion of Specification 4.4.10.

LIMITING CONDITIONS FOR OPERATION AND SURVEILLANCE REQUIREMENTS

BASES

SECTION PAGE

REACTIVITY CONTROL SYSTEMS

FLOW PATHS – OPERATING

LIMITING CONDITION FOR OPERATION

3.1.2.2 At least two of the following boron injection flow paths shall be OPERABLE:

a. The flow path from the boric acid tanks via a boric acid transfer pump and a charging pump to the Reactor Coolant System.

b. Two flow paths from the refueling water storage tank via charging pumps to the Reactor Coolant System.

APPLICABILITY: MODES 1, 2, 3 and 4#.

ACTION:

With only one of the above required boron injection flow paths to the Reactor Coolant System OPERABLE, restore at least two boron injection flow paths to the Reactor Coolant System to OPERABLE status within 72 hours or be in at least HOT STANDBY and borated to a SHUTDOWN MARGIN equivalent to at least 1.77% delta k/k at 200°F within the next 6 hours; restore at least two flow paths to OPERABLE status within the next 7 days or be in COLD SHUTDOWN within the next 30 hours.

SURVEILLANCE REQUIREMENTS

4.1.2.2 Each of the above required flow paths shall be demonstrated OPERABLE:

a. At least once per 7 days by verifying that the temperature of the heat traced portion of the flow path from the boric acid tanks is greater than or equal to 115°F when it is a required water source.

\# Only one boron injection flow path is required to be OPERABLE whenever the temperature of one or more of the RCS cold legs is less than or equal to 270°F.

CHARGING PUMPS - OPERATING

LIMITING CONDITION FOR OPERATION

3.1.2.4 At least two charging pumps shall be OPERABLE.

APPLICABILITY: MODES 1, 2, 3 and 4#.

ACTION:

With only one charging pump OPERABLE, restore a second charging pump to OPERABLE status within 72 hours or be in at least HOT STANDBY and borated to a SHUTDOWN MARGIN equivalent to at least 1.77% delta k/k at 200°F within the next 6 hours; restore a second charging pump to OPERABLE status within the next 7 days or be in COLD SHUTDOWN within the next 30 hours. The provisions of Specification 3.0.4 are not applicable for one hour following heatup above 270°F or prior to cooldown below 270°F.

SURVEILLANCE REQUIREMENTS

4.1.2.4.1 The above required charging pumps shall be demonstrated OPERABLE by verifying, that on recirculation flow, each pump develops a discharge pressure of greater than or equal to 2410 psig when tested pursuant to Specification 4.0.5.

4.1.2.4.2 All charging pumps, except the above required OPERABLE pump, shall be demonstrated inoperable at least once per 12 hours whenever the temperature of one or more of the RCS cold legs is less than or equal to 270°F by verifying that the control switch is in the pull to lock position.

A maximum of one centrifugal charging pump shall be OPERABLE whenever the temperature of one or more of the RCS cold legs is less than or equal to 270°F.

REACTOR COOLANT SYSTEM

HOT STANDBY

LIMITING CONDITION FOR OPERATION

3.4.1.2 a. At least two of the reactor coolant loops listed below shall be OPERABLE:

1. Reactor Coolant Loop A and its associated steam generator and reactor coolant pump,
2. Reactor Coolant Loop B and its associated steam generator and reactor coolant pump,
3. Reactor Coolant Loop C and its associated steam generator and reactor coolant pump,

b. At least one of the above coolant loops shall be in operation.*, **

APPLICABILITY: MODE 3

ACTION:

a. With less than the above required loops OPERABLE, restore the required loops to OPERABLE status within 72 hours or be in HOT SHUTDOWN within the next 12 hours.

b. With no coolant loop in operation, suspend all operations involving a reduction in boron concentration of the Reactor Coolant System and immediately initiate corrective actions to return the required coolant loop to operation.

SURVEILLANCE REQUIREMENTS

4.4.1.2.1 At least the above required reactor coolant pumps, if not in operation, shall be determined to be OPERABLE once per 7 days by verifying correct breaker alignments and indicated power availability.

4.4.1.2.2 At least one cooling loop shall be verified to be in operation and circulating coolant at least once per 12 hours.

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

** The requirement to have one coolant loop in operation is exempted during the performance of the boron mixing tests as stipulated in License Condition 2.C(15)(f) and 2.C(20)(b).

LIMITING CONDITION FOR OPERATION

3.4.1.3 a. At least two of the coolant loops listed below shall be OPERABLE:

1. Reactor Coolant Loop A and its associated steam generator and reactor coolant pump,*
2. Reactor Coolant Loop B and its associated steam generator and reactor coolant pump,*
3. Reactor Coolant Loop C and its associated steam generator and reactor coolant pump,*
4. Residual Heat Removal Subsystem A,**
5. Residual Heat Removal Subsystem B.**

b. At least one of the above coolant loops shall be in operation.***

APPLICABILITY: MODES 4 and 5

ACTION:

a. With less than the above required loops OPERABLE, immediately initiate corrective action to return the required loops to OPERABLE status as soon as possible; be in COLD SHUTDOWN within 20 hours.

b. With no coolant 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.

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* A reactor coolant pump shall not be started with one or more of the RCS cold leg temperatures less than or equal to 270°F unless the secondary water temperature of each steam generator is less than 50°F above each of the RCS cold leg temperatures.

** The offsite or emergency power source may be inoperable in MODE 5.

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

REACTOR COOLANT SYSTEM

SAFETY AND RELIEF VALVES – OPERATING

RELIEF VALVES

LIMITING CONDITION FOR OPERATION

3.4.3.2 Both power-operated relief valves (PORVs) and their associated block valves shall be OPERABLE.

APPLICABILITY: MODES 1, 2, and 3.

ACTION:

a. With one or both PORVs inoperable but capable of being manually cycled, within 1 hour either restore the PORV(s) to OPERABLE status or close the associated block valve(s) with power maintained to the block valve(s); otherwise, be in at least HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours.

b. With one PORV inoperable and not capable of being manually cycled, within 1 hour either restore the PORV to OPERABLE status or capable of being manually cycled, or close its associated block valve and remove power from the block valve; restore the PORV to OPERABLE status within the following 72 hours or be in HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours.

c. With both PORVs inoperable and not capable of being manually cycled, within 1 hour either restore at least one PORV to OPERABLE status or capable of being manually cycled, or close its associated block valve and remove power from the block valve and be in HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours.

d. With one or both block valves inoperable, within 1 hour restore the block valve(s) to OPERABLE status or place its associated PORV(s) in manual control. Restore at least one block valve to OPERABLE status within the next hour if both block valves are inoperable; restore the remaining inoperable block valve to OPERABLE status within 72 hours; otherwise, be in at least HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours.

e. The provisions of Specification 3.0.4 are not applicable.

RELIEF VALVES

SURVEILLANCE REQUIREMENTS

4.4.3.2.1 In addition to the requirements of Specification 4.0.5, each PORV shall be demonstrated OPERABLE:

a. At least once per 31 days by performing a CHANNEL FUNCTIONAL TEST, excluding valve operation, and

b. At least once per 18 months by:

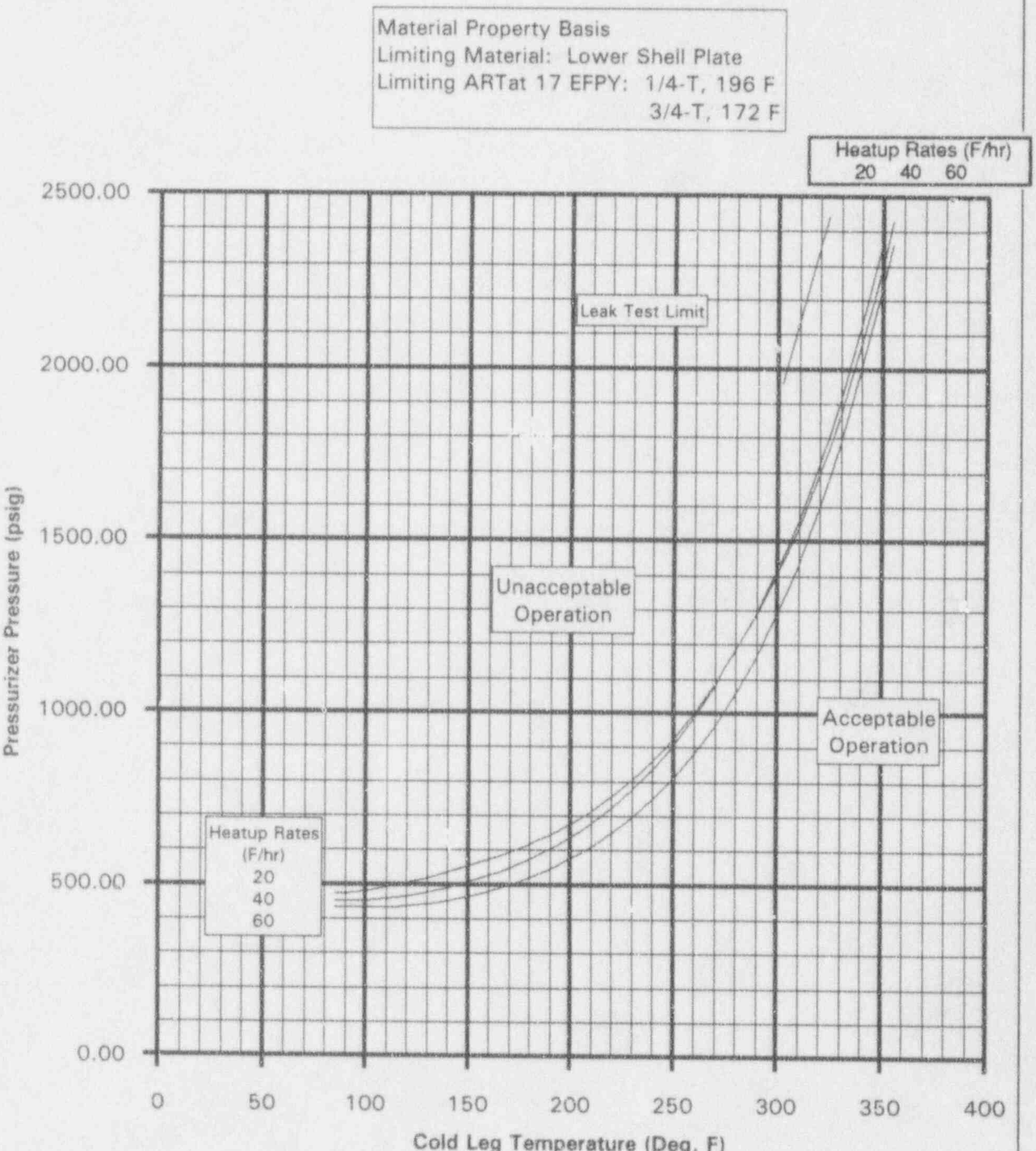
   1. Operating the PORV through one complete cycle of full travel during MODES 3 or 4, and

   2. Operating the solenoid air control valves and check valves on the associated accumulators in the PORV control systems through one complete cycle of full travel, and

   3. Performing a CHANNEL CALIBRATION of the actuation instrumentation.

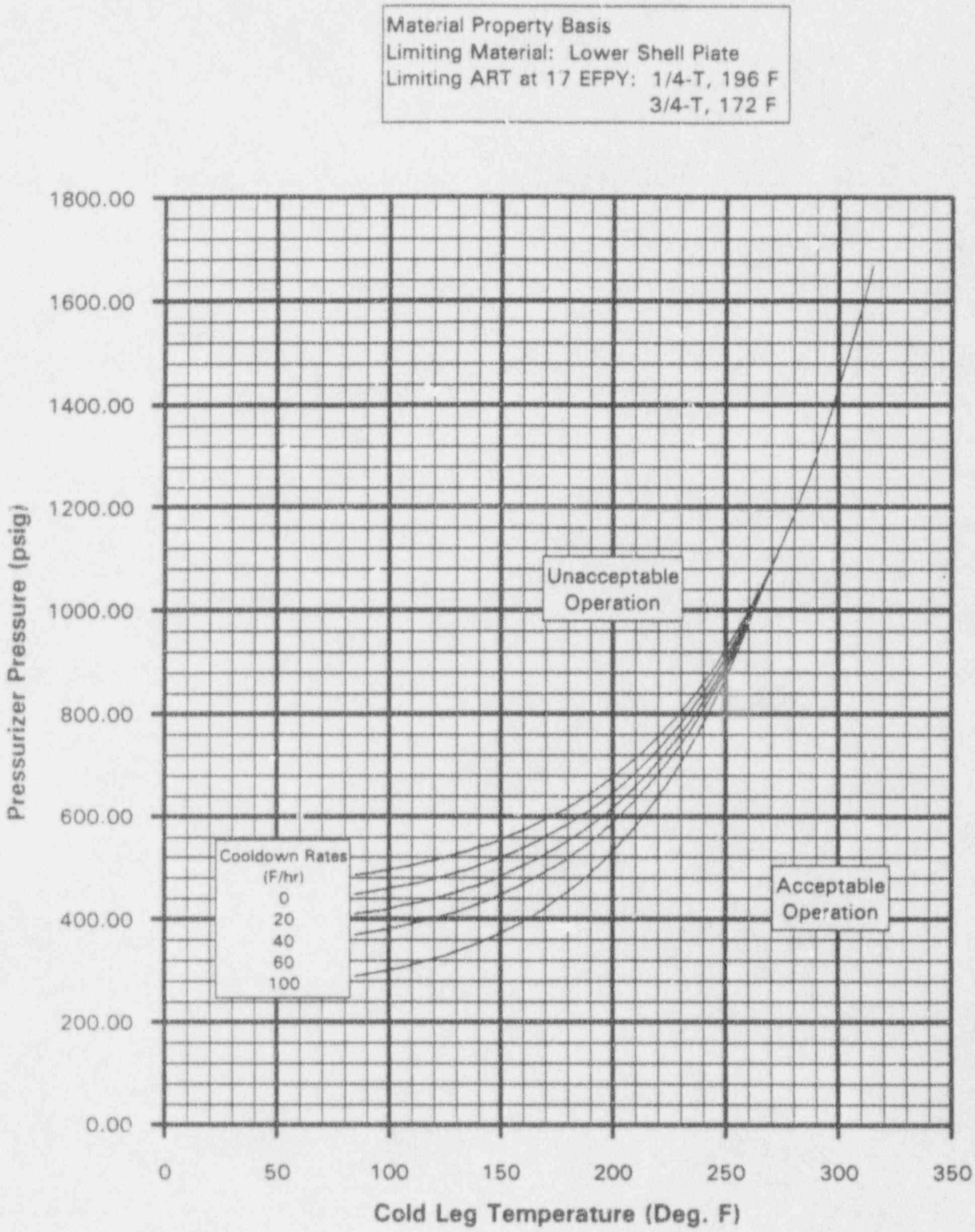
4.4.3.2.2 Each block valve shall be demonstrated OPERABLE at least once per 92 days by operating the valve through one complete cycle of full travel unless the block valve is closed in order to meet the requirements of ACTION b or c in Specification 3.4.3.2.

Figure 3.4-2 — North Anna Unit 2
Reactor Coolant System Heatup Limitations



North Anna Unit 2 Reactor Coolant System Heatup Limitations (Heatup Rates up to 60 F/hr) Applicable for the First 17 EFPY (Without Margins for Instrumentation Errors)

Figure 3.4-3 — **North Anna Unit 2**
Reactor Coolant System Cooldown Limitations



North Anna Unit 2 Reactor Coolant System Cooldown Limitations (Cooldown Rates up to 100 F/hr) Applicable for the First 17 EFPY (Without Margins for Instrumentation Errors)

LOW-TEMPERATURE OVERPRESSURE PROTECTION

LIMITING CONDITION FOR OPERATION

3.4.9.3 Two power-operated relief valves (PORVs) shall be OPERABLE with lift settings of (1) less than or equal to 415 psig whenever any RCS cold leg temperature is less than or equal to 270°F, and (2) less than or equal to 375 psig whenever any RCS cold leg temperature is less than 130°F.

APPLICABILITY: MODE 4 when the temperature of any RCS cold leg is less than or equal to 270°F, MODE 5, and MODE 6 when the head is on the reactor vessel and the RCS is not vented through a 2.07 square inch or larger vent.

ACTION:

a. With one PORV inoperable in MODE 4, restore the inoperable PORV to OPERABLE status within 7 days or depressurize and vent the RCS through at least a 2.07 square inch vent within the next 8 hours.

b. With one PORV inoperable in MODES 5 or 6, either (1) restore the inoperable PORV to OPERABLE status within 24 hours, or (2) complete depressurization and venting of the RCS through at least a 2.07 square inch vent within a total of 32 hours.

c. With both PORVs inoperable, complete depressurization and venting of the RCS through at least a 2.07 square inch vent within 8 hours.

d. With the RCS vented per ACTIONS a, b, or c, verify the vent pathway at least once per 31 days when the pathway is provided by a valve(s) that is locked, sealed, or otherwise secured in the open position; otherwise, verify the vent pathway ever 12 hours.

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

f. The provisions of Specification 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.4.9.3 Each PORV shall be demonstrated OPERABLE by:

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

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

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

d. Testing pursuant to Specification 4.0.5.

EMERGENCY CORE COOLING SYSTEMS

ECCS SUBSYSTEMS – T_{avg} GREATER THAN 350°F

LIMITING CONDITION FOR OPERATION

3.5.2 Two independent ECCS subsystems shall be OPERABLE with each subsystem comprised of:

a. One OPERABLE centrifugal charging pump,

b. One OPERABLE low head safety injection pump,

c. An OPERABLE flow path capable of transferring fluid to the Reactor Coolant System when taking suction from the refueling water storage tank on a safety injection signal or from the containment sump when suction is transferred during the recirculation phase of operation.

APPLICABILITY: MODES 1, 2 and 3.

ACTION:

a. With one ECCS subsystem inoperable, restore the inoperable subsystem to OPERABLE status within 72 hours or be in HOT SHUTDOWN within the next 12 hours.

b. In the event the ECCS is actuated and injects water into the Reactor Coolant System, a Special Report shall be prepared and submitted to the Commission pursuant to Specification 6.9.2 within 90 days describing the circumstances of the actuation and the total accumulated actuation cycles to date. The current value of the usage factor for each affected safety injection nozzle shall be provided in this Special Report whenever its value exceeds 0.70.

c. The provisions of Specification 3.0.4 are not applicable to 3.5.2.a and 3.5.2.b for one hour following heatup above 270°F or prior to cooldown below 270°F.

SURVEILLANCE REQUIREMENTS

4.5.2 Each ECCS subsystem shall be demonstrated OPERABLE:

a. At least once per 12 hours by verifying that the following valves are in the indicated positions with power to the valve operators removed:

ECCS SUBSYSTEMS - T_{avg} LESS THAN 350°F

LIMITING CONDITION FOR OPERATION

3.5.3 As a minimum, one ECCS subsystem comprised of the following shall be OPERABLE:

a. One OPERABLE centrifugal charging pump#,

b. One OPERABLE low head safety injection pump#, and

c. An OPERABLE flow path capable of automatically transferring fluid to the reactor coolant system when taking suction from the refueling water storage tank or from the containment sump when the suction is transferred during the recirculation phase of operation.

APPLICABILITY: MODE 4.

ACTION:

a. With no ECCS subsystem OPERABLE because of the inoperability of either the centrifugal charging pump or the flow path from the refueling water storage tank, restore at least one ECCS subsystem to OPERABLE status within 1 hour or be in COLD SHUTDOWN within the next 20 hours.

b. With no ECCS subsystem OPERABLE because of the inoperability of the low head safety injection pump, restore at least one ECCS subsystem to OPERABLE status or maintain the Reactor Coolant System T_{avg} less than 350°F by use of alternate heat removal methods.

c. In the event the ECCS is actuated and injects water into the Reactor Coolant System, a Special Report shall be prepared and submitted to the Commission pursuant to Specification 6.9.2 within 90 days describing the circumstances of the actuation and the total accumulated actuation cycles to date. The current value of the usage factor for each affected safety injection nozzle shall be provided in this Special Report whenever its value exceeds 0.70.

A maximum of one centrifugal charging pump and one low head safety injection pump shall be OPERABLE whenever the temperature of one or more of the RCS cold legs is less than or equal to 270°F.

SURVEILLANCE REQUIREMENTS

4.5.3.1 The ECCS subsystem shall be demonstrated OPERABLE per the applicable Surveillance Requirements of 4.5.2.

4.5.3.2 All charging pumps and safety injection pumps, except the above required OPERABLE pumps, shall be demonstrated inoperable at least once per 12 hours whenever the temperature of one or more of the RCS cold legs is less than or equal to 270°F by verifying that the control switch is in the pull to lock position.

3/4.1.2 BORATION SYSTEMS

The boron injection system ensures that negative reactivity control is available during each mode of facility operation. The components required to perform this function include 1) borated water sources, 2) charging pumps, 3) separate flow paths, 4) boric acid transfer pumps, 5) associated heat tracing systems, and 6) an emergency power supply from OPERABLE diesel generators.

With the RCS average temperature above 200°F, a minimum of two boron injection flow paths are required to ensure single functional capability in the event an assumed failure renders one of the flow paths inoperable. The boration capability of either flow path is sufficient to provide a SHUTDOWN MARGIN from expected operation conditions of 1.77% delta k/k after xenon decay and cooldown to 200°F. The maximum expected boration capability requirement occurs at EOL from full power equilibrium xenon conditions and requires 6000 gallons of 12,950 ppm borated water from the boric acid storage tanks or 54,200 gallons of 2300 ppm borated water from the refueling water storage tank.

With the RCS temperature below 200°F, one injection system is acceptable without single failure consideration on the basis of the stable reactivity condition of the reactor and the additional restrictions prohibiting CORE ALTERATIONS and positive reactivity change in the event the single injection system becomes inoperable.

The limitation for a maximum of one centrifugal charging pump to be OPERABLE and the Surveillance Requirement to verify all charging pumps except the required OPERABLE pump to be inoperable below 270°F provides assurance that a mass addition pressure transient can be relieved by the operation of a single PORV.

The boron capability required below 200°F is sufficient to provide a SHUTDOWN MARGIN of 1.77% delta k/k after zenon decay and cooldown from 200°F to 140°F. This condition requires either 1378 gallons of 12,950 ppm borated water from the boric acid storage tanks or 3400 gallons of 2300 ppm borated water from the refueling water storage tank.

3/4.4.1 REACTOR COOLANT LOOPS

The plant is designed to operate with all reactor coolant loops in operation and maintain the DNBR above design limit during all normal operations and anticipated transients. In MODES 1 and 2 with one reactor coolant loop not in operation, this specification requires that the plant be in at least HOT STANDBY within 1 hour.

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

In MODES 4 and 5, a single reactor coolant loop or RHR loop provides sufficient heat removal capability for removing decay heat, but single failure considerations require that at least two loops be OPERABLE. Thus, if the reactor coolant loops are not OPERABLE, this specification requires two RHR loops to be OPERABLE.

After the reactor has shutdown and entered into MODE 3 for at least 100 hours, a minimum RHR system flow rate of 2000 gpm in MODE 5 is permitted, provided there is sufficient decay heat removal to maintain the RCS temperature less than or equal to 140°F. Since the decay heat power production rate decreases with time after reactor shutdown, the requirements for RHR system decay heat removal also decrease. Adequate decay heat removal is provided as long as the reactor has been shutdown for at least 100 hours after entry into MODE 3 and RHR flow is sufficient to maintain the RCS temperature less than or equal to 140°F. The reduced flow rate provides additional margin to vortexing at the RHR pump suction while in Mid Loop Operation. During a reduction in reactor coolant system boron concentration the Specification 3.1.1.3.1 requirement to maintain 3000 gpm flow rate provides sufficient coolant circulation to minimize the effect of a boron dilution incident and to prevent boron stratification.

The restrictions on starting a Reactor Coolant Pump with one or more RCS cold legs less than or equal to 270°F are provided to prevent RCS pressure transients, caused by energy additions from the secondary system which could exceed the limits of Appendix G to 10 CFR Part 50. The RCS will be protected against overpressure transients and will not exceed the limits of Appendix G by restricting starting of the RCPs to when the secondary water temperature of each steam generator is less than 50°F above each of the RCS cold leg temperatures.

The requirement to maintain the boron concentration of an isolated loop greater than or equal to the boron concentration of the operating loops ensures that no reactivity addition to the core could occur during startup of an isolated loop. Verification of the boron concentration in an idle loop prior to opening the cold leg stop valve provides a reassurance of the adequacy of the boron concentration in the isolated loop. Operating the isolated loop on recirculating flow for at least 90 minutes prior to opening its cold leg stop valve ensures adequate mixing of the coolant in this loop and prevents any reactivity effects due to boron concentration stratifications.

Startup of an idle loop will inject cool water from the loop into the core. The reactivity transient resulting from this cool water injection is minimized by delaying isolated loop startup until its temperature is within 20°F of the operating loops. Making the reactor subcritical prior to loop startup prevents any power spike which could result from this cool water induced reactivity transient.

3/4.4.2 AND 3/4.4.3 SAFETY AND RELIEF VALVES

The pressurizer code safety valves operate to prevent the RCS from being pressurized above its Safety Limit of 2735 psig. Each safety valve is designed to relieve 380,000 lbs per hour of saturated steam at the valve set point. The relief capacity of a single safety valve is adequate to relieve any overpressure condition which could occur during hot shutdown. In the event that no safety valves are OPERABLE, an operating RHR loop, connected to the RCS, or the power operated relief valves (PORVs) will provide overpressure relief capability and will prevent RCS overpressurization.

During operation, all pressurizer code safety valves must be OPERABLE to prevent the RCS from being pressurized above its safety limit of 2735 psig. The combined relief capacity of all of these valves is greater than the maximum surge rate resulting from a complete loss of load assuming no reactor trip until the first Reactor Protective System trip setpoint is reached (i.e., no credit is taken for a direct reactor trip on the loss of load) and also assuming no operation of the power operated relief valves or steam dump valves.

Demonstration of the safety valves' lift settings will occur only during shutdown and will be performed in accordance with the provisions of Section XI of the ASME Boiler and Pressure Vessel Code.

The power operated relief valves and steam bubble function to relieve RCS pressure during all design transients up to and including the design step load decrease with steam dump. Operation of the PORVs minimizes the undesirable opening of the spring-loaded pressurizer code safety valves. Each PORV has a remotely operated block valve to provide a positive shutoff capability should a relief valve become inoperable.

The OPERABILITY of the PORVs and block valves is determined on the basis of their being capable of performing the following functions:

a) Manual control of PORVs to control reactor coolant system pressure. This is a function that may be used to mitigate certain accidents and for plant shutdown.

b) Maintaining the integrity of the reactor coolant pressure boundary. This function is related to controlling identified leakage and ensuring the ability to detect unidentified reactor coolant pressure boundary leakage.

c) Manual control of the block valve to (1) unblock an isolated PORV to allow it to be used for manual control of reactor coolant system pressure (Item a, above), and (2) isolate a PORV with excessive seat leakage (Item b, above).

d) Automatic control of PORVs to control reactor coolant system pressure. This function reduces challenges to the code safety valves for overpressurization events.

e) Manual control of a block valve to isolate a stuck-open PORV.

Surveillance Requirements provide the assurance that the PORVs and block valves can perform their functions. Specification 4.4.3.2.1 addresses the PORVs and Specification 4.4.3.2.2 addresses the block valves. The block valves are exempt from the surveillance requirements to cycle the valves when they have been closed to comply with the ACTION requirements. This precludes the need to cycle the valves with full system differential pressure or when maintenance is being performed to restore an inoperable PORV to operable status.

Surveillance Requirement 4.4.3.2.1.b.2 provides for the testing of the mechanical and electrical aspects of control systems for the PORVs.

Testing of PORVs in HOT STANDBY or HOT SHUTDOWN is required in order to simulate the temperature and pressure environmental effects on PORVs. Testing at COLD SHUTDOWN is not considered to be a representative test for assessing PORV performance under normal plant operating conditions.

3/4.4.4 PRESSURIZER

The limit on the maximum water volume in the pressurizer assures that the parameter is maintained within the normal steady state envelope of operation assumed in the SAR. The limit is consistent with the initial SAR assumptions. The 12-hour periodic surveillance is sufficient to ensure that the parameter is restored to within its limit following expected transient operation. The maximum water volume also ensures that a steam bubble is formed and thus the RCS is not a hydraulically solid system. The requirement that a minimum number of pressurizer heaters be OPERABLE ensures that the plant will be able to establish natural circulation.

The ACTION statement permitting POWER OPERATION to continue for limited time periods with the primary coolant's specific activity greater than 1.0 ~μCi/gram DOSE EQUIVALENT I-131, but within the allowable limit shown on Figure 3.4-1, accommodates possible iodine spiking phenomenon which may occur following changes in THERMAL POWER.

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

3/4.4.9 PRESSURE / TEMPERATURE LIMITS

Reactor Coolant System

All components in the Reactor Coolant System are designed to withstand the effects of cyclic loads due to system temperature and pressure changes. These cyclic loads are introduced by normal load transients, reactor trips, and startup and shutdown operations. The various categories of load cycles used for design purposes are provided in Section 5.2 of the UFSAR. During startup and shutdown, the rates of temperature and pressure changes are limited so that the maximum specified heatup and cooldown rates are consistent with the design assumptions and satisfy the stress limits for cyclic operation.

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

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

The heatup limit curve, Figure 3.4-2, is a composite curve which was prepared by determining the most conservative case, with either the inside or outside wall controlling, for any heatup rate up to 60°F per hour. The cooldown limit curves of Figure 3.4-3 are composite curves which were prepared based upon the same type analysis with the exception that the controlling location is always the inside wall where the cooldown thermal gradients tend to produce tensile stresses while producing compressive stresses at the outside wall. The heatup and cooldown curves were prepared based upon the most limiting value of the predicted adjusted reference temperature at the end of 17 EFPY. The most recent capsule analysis results are documented in Westinghouse Reports WCAP-12497, January 1990. The heatup and cooldown curves are documented in Westinghouse Report WCAP-12503, March, 1990.

The reactor vessel materials have been tested to determine their initial RT_{NDT}. Reactor operation and resultant fast neutron (E > 1 Mev) irradiation will cause an increase in the RT_{NDT}. An adjusted reference temperature, based upon the fluence and copper content of the material in question, can be predicted using US NRC Regulatory Guide 1.98, Revision 2. The heatup and cooldown limit curves (Figure 3.4-2 and Figure 3.4-3) include predicted adjustments for this shift in RT_{NDT} at the end of 17 EFPY. The reactor vessel beltline region material properties are listed on Figures 3.4-2 and 3.4-3.

The actual shift in the RT_{NDT} of the vessel material will be established periodically by removal and evaluation of the reactor vessel material specimens installed on the inside wall of the thermal shield. The surveillance capsule withdrawal schedule was prepared in accordance with the requirements of ASTM E-185 and is presented in the UFSAR. Regulatory Guide 1.99, Revision 2, provides guidance for calculation of the shift in RT_{NDT} using measured data. Dosimetry from the surveillance capsule is used to determine the neutron fluence to which the material specimens were exposed, and to support calculational estimates of the neutron fluence to the reactor vessel.

The pressure-temperature limit lines shown on Figure 3.4-2 for inservice leak and hydrostatic testing have been provided to assure compliance with the minimum temperature requirements of Appendix G to 10 CFR 50. The minimum temperature for criticality specified in T.S. 3.1.1.5 assures compliance with the criticality limits of 10 CFR 50 Appendix G.

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

Pressurizer

The limitations imposed on pressurizer heatup and cooldown 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.

Low-Temperature Overpressure Protection

The OPERABILITY of two PORVs or an RCS vent opening of greater than 2.07 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 270°F. Either PORV has adequate relieving capability to protect the RCS from overpressurization when the transient is limited to either (1) the start of an idle RCP with the secondary water temperature of the steam generator less than or equal to 50°F above the RCS cold leg temperatures or (2) the start of a charging pump and its injection into a water-solid RCS.

Automatic or passive low temperature overpressure protection (LTOP) is required whenever any RCS cold leg temperature is less than 270°F. This temperature is the water temperature corresponding to a metal temperature of at least the limiting RT_{NDT} + 50°F + instrument uncertainty. Above 270°F administrative control is adequate protection to ensure the limits of the heatup curve (Figure 3.4-2) and the cooldown curve (Figure 3.4-3) are not violated. The concept of requiring automatic LTOP at the lower end, and administrative control at the upper end, of the Appendix G curves is further discussed in NRC Generic Letter 88-11.

3/4.4.10 STRUCTURAL INTEGRITY

3/4.4.10.1 ASME CODE CLASS 1, 2 and 3 COMPONENTS

The inspection programs for ASME Code Class 1, 2 and 3 Reactor Coolant System components ensure that the structural integrity of these components will be maintained at an acceptable level throughout the life of the plant. To the extent applicable, the inspection program for components is in compliance with Section XI of the ASME Boiler and Pressure Vessel Code.

ECCS SUBSYSTEMS (Continued)

With the RCS temperature below 350°F, one OPERABLE ECCS subsystem is acceptable without single failure consideration on the basis of the stable reactivity condition of the reactor and the limited core cooling requirements.

The limitation for a maximum of one centrifugal charging pump and one low head safety injection pump to be OPERABLE and the Surveillance Requirement to verify all charging pumps and low head safety injection pumps except the required OPERABLE pump to be inoperable below 270°F provides assurance that a mass addition pressure transient can be relieved by the operation of a single PORV.

The Surveillance Requirements provided to ensure OPERABILITY of each component ensures that at a minimum, the assumptions used in the safety analyses are met and that subsystem OPERABILITY is maintained. Surveillance Requirements for the throttle valve position stops and flow balance testing provide assurance that proper flow resistance and pressure drop in the piping system to each injection point is necessary to: (1) prevent total pump flow from exceeding runout conditions when the system is in its minimum resistance configuration, (2) provide the proper flow split between injection points in accordance with the assumptions used in the ECCS - LOCA analyses, and (3) provide an acceptable level of total ECCS flow to all injection points equal to or above that assumed in the ECCS - LOCA analyses.

3/4.5.4 BORON INJECTION SYSTEM

The OPERABILITY of the boron injection system as part of the ECCS ensures that sufficient negative reactivity is injected into the core to counteract any positive increase in reactivity caused by RCS system cooldown. RCS cooldown can be caused by inadvertent depressurization, a loss-of-coolant accident or a steam line rupture.

The limits on injection tank minimum contained volume and boron concentration ensure that the assumptions used in the steam line break analysis are met. The contained water volume limit includes an allowance for water not usable because of tank discharge line location or other physical characteristics.

The OPERABILITY of the redundant heat tracing channels associated with the boron injection system ensure that the solubility of the boron solution will be maintained above the solubility limit of 111°F at 15,750 ppm boron.

SPECIAL REPORTS

6.9.2 Special reports shall be submitted to the Regional Administrator, Region II, within the time period specified for each report. These reports shall be submitted pursuant to the requirement of the applicable specification:

a. Inservice Inspection Reviews, Specification 4.0.5, shall be reported within 90 days of completion.

b. MODERATOR TEMPERATURE COEFFICIENT. Specification 3.1.1.4.

c. Deleted.

d. RADIATION MONITORING INSTRUMENTATION. Specification 3.3.3.1, Table 3.3-6, Action 35.

e. Deleted.

f. LOW-TEMPERATURE OVERPRESSURE PROTECTION. Specification 3.4.9.3.

g. EMERGENCY CORE COOLING SYSTEMS. Specification 3.5.2 and 3.5.3.

h. SETTLEMENT OF CLASS 1 STRUCTURES. Specification 3.7.12.

i. GROUND WATER LEVEL - SERVICE WATER RESERVOIR. Specification 3.7.13.

j. Deleted.

k. Deleted.

l. RADIOACTIVE EFFLUENTS. As required by the ODCM.

m. RADIOLOGICAL ENVIRONMENTAL MONITORING. As required by the ODCM.

n. SEALED SOURCE CONTAMINATION. Specification 4.7.11.1.3.

o. REACTOR COOLANT SYSTEM STRUCTURAL INTEGRITY. Specification 4.4.10. For any abnormal degradation of the structural integrity of the reactor vessel or the Reactor Coolant System pressure boundary detected during the performance of Specification 4.4.10, an initial report shall be submitted within 10 days after detection and a detailed report submitted within 90 days after the completion of Specification 4.4.10.

p. CONTAINMENT STRUCTURAL INTEGRITY. Specification 4.6.1.6. For any abnormal degradation of the containment structure detected during the performance of Specification 4.6.1.6, an initial report shall be submitted within 10 days after completion of Specification 4.6.1.6. A final report, which includes (1) a description of the condition of the liner plate and concrete, (2) inspection procedure, (3) the tolerance on cracking, and (4) the corrective actions taken, shall be submitted within 90 days after the completion of Specification 4.6.1.6.

ATTACHMENT 3

SIGNIFICANT HAZARDS CONSIDERATION

VIRGINIA ELECTRIC AND POWER COMPANY

Significant Hazards Consideration

The North Anna Units 1 and 2 Reactor Coolant Systems are protected from material failure by the imposition of restrictions on allowable pressure and temperature, and on heatup and cooldown rate. The Low Temperature Overpressure Protection System (LTOPS) ensures that material integrity limits are not exceeded during the design basis overpressurization accidents. Equipment operability requirements are imposed to ensure that the assumptions of the accident analyses remain valid. The operating restrictions, setpoints, and equipment operability requirements must be revised to extend their applicability to a higher cumulative burnup, and to improve operational flexibility.

The current pressure/temperature operating limits and LTOPS setpoints are valid to 12 EFPY and 17 EFPY for North Anna Units 1 and 2, respectively. According to the most recent estimates, the burnup applicability limits will be exceeded by North Anna Unit 1 in Spring of 1996. The North Anna Unit 2 pressure/temperature operating limits and LTOPS setpoints remain valid well into the year 2002. The proposed North Anna Unit 1 Technical Specifications include revised pressure/temperature operating limits valid to end-of-license. Although the North Anna Unit 2 Technical Specification pressure/temperature operating limits are not being changed, the Unit 2 LTOPS setpoints and associated reactor vessel integrity protection philosophy are being changed. The reactor vessel integrity protection philosophy which supports the proposed Technical Specifications changes provides improved operational flexibility while maintaining an adequate margin of safety as demonstrated by the safety analysis. The probability and consequences of the design basis low temperature overpressurization accidents are not increased.

Virginia Electric and Power Company has reviewed the Technical Specification changes against the criteria of 10 CFR 50.92 and has concluded that the changes do not pose a significant hazards consideration. Specifically, operation of Surry Power Station in accordance with the Technical Specification changes will not:

a. involve a significant increase in the probability or consequences of an accident previously evaluated. The safety analysis demonstrates that the proposed reactor vessel protection philosophy, and the associated pressure/temperature limits, LTOPS setpoints, and component operability requirements, ensure that reactor vessel integrity will be maintained during normal operation and design basis accident conditions. Specifically, adherence to the heatup/cooldown rate-dependent pressure/temperature operating limits ensures that the assumed design basis flaw will not propagate during normal operation. Below the LTOPS enabling temperature, automatic actuation of the PORVs ensures that the assumed design basis flaw will not propagate under design basis low-temperature overpressurization accident conditions. Two pressurizer safety valve are sufficient to relieve the overpressurization due to the inadvertent startup of two charging pumps at water solid conditions without propagation of the assumed design basis flaw. The proposed changes to address the concerns of Generic Letter 90-06 (Generic Issues 70 and 94) improve LTOPS availability and reliability by instituting requirements for PORV, block valve, and control system testing and allowed outage times for these components. Although these changes do not reduce the probability of occurrence or the consequences of the LTOPS design basis (mass and heat addition) transients, the changes provide increased assurance that pressure relieving devices will perform their design function when required.

b. create the possibility of a new or different kind of accident from any accident previously evaluated. The proposed Technical Specifications modify pressure/temperature operating limits, LTOPS setpoints and enabling temperatures, and component operability requirements. The revised pressure/temperature operating limits, and LTOPS setpoints and enabling temperatures are only slightly different than those currently in the Technical Specifications. No operating limits or setpoints are added or deleted by the proposed changes. Therefore, it may be concluded that the operating limits and setpoint changes do not create the possibility of a new or different kind of accident. With regard to component operability requirements, restrictions on the number of charging pumps which may be operable, the number of PORVs which must be operable, and the allowable temperature difference between the steam generator primary and secondary remain unchanged. Only the setpoint temperature at which these restrictions apply have been modified. The proposed changes are entirely consistent with the reactor vessel integrity protection philosophy which ensures that the design basis reactor vessel flaw will not propagate under normal operation or postulated accident conditions. Further, the proposed changes do not invalidate the any component design criteria or the assumptions of any UFSAR Chapter 15 accident analyses. In addition, modifications have been made to the Technical Specifications to improve availability and reliability of PORVs and associated block valves. These changes have been made in accordance with NRC guidance in Generic Letter 90-06. It may be concluded that none of the proposed changes creates the possibility of a new or different kind of accident from any previously evaluated.

c. involve a significant reduction in a margin of safety. As described above, the reactor vessel integrity protection philosophy ensures that the design basis assumed flaw will not propagate under normal operation or design basis accident conditions. Adherence to the Technical Specification pressure/temperature operating limits ensures that the margin to vessel fracture provided by the ASME Section XI methodology is maintained. With regard to LTOPS protection, the safety analysis demonstrates that the proposed LTOPS design ensures margins consistent with those provided by ASME Section XI Appendix G methods. This conclusion is based on industry experience with LTOPS events and engineering evaluation. Specifically, both industry experience and engineering evaluation demonstrate that LTOPS design basis events may be expected to occur at essentially isothermal conditions. Engineering evaluation demonstrates that any reduction in allowable pressure due to thermal stresses which may be expected to occur during low temperature operation is insignificant when compared to margins provided by the ASME Section XI Appendix G methods for calculating pressure/temperature operating limits. Use of the isothermal pressure/temperature limit curve as the design limit for establishing low temperature PORV lift setpoints has been approved for other utilities by the NRC. This design maximizes the operating margin above the minimum RCS pressure for reactor coolant pump (RCP) operation, thereby minimizing the probability of undesired PORV lifts during RCP startup. The proposed changes to address the concerns of Generic Letter 90-06 (Generic Issues 70 and 94) improve LTOPS availability and reliability by instituting requirements for PORV, block valve, and control system testing and allowed outage times for these components. Although these changes do not increase the margin of safety demonstrated by the analysis of the LTOPS design basis (mass and heat addition) transients, the changes provide increased assurance that pressure relieving devices will perform their design function when required.

Virginia Electric and Power Company concludes that the activities associated with these proposed Technical Specification changes satisfy the no significant hazards consideration criteria of 10 CFR 50.92 and, accordingly, a no significant hazards consideration finding is justified.