NORTH ANNA POWER STATION

UNITS 1 and 2

TECHNICAL SPECIFICATION, CHANGE REQUEST

HEATUP/COOLDOWN LIMITATIONS
LOW TEMPERATURE / OVERPRESSURE PROTECTION

Attachment 1	Discussion of Proposed Changes
Attachment 2	Proposed Changes – Unit 1 - Final Copy
Attachment 3	Proposed Changes – Unit 2 - Final Copy
Attachment 4	Significant Hazards Considerations Determination (10 CFR 50.92 Eval ..ion)
Attachment 5	Additional Supporting Information (NA&F Technical Analysis) (BAW-2146) (WCAP-12503) (References List)

VIRGINIA ELECTRIC AND POWER COMPANY

ATTACHMENT 1

PROPOSED TECHNICAL SPECIFICATIONS CHANGES

TECHNICAL SPECIFICATION CHANGE REQUEST

DISCUSSION

NORTH ANNA POWER STATION - UNITS 1 & 2

VIRGINIA ELECTRIC AND POWER COMPANY

DISCUSSION OF PROPOSED CHANGES

Introduction

The North Anna Reactor Coolant Systems (RCS), specifically the Reactor Pressure Vessels (RPV), are protected from material failure during low temperature operations by imposing restrictions on RCS pressure. The heatup and cooldown curves as well as the Low Temperature/Overpressure Protection System (LTOP) setpoints, provide the restrictions to bound the area of operation and ensure RCS protection from non-ductile failure. The regulatory requirements for providing these restrictions and reevaluating them, are stipulated in 10 CFR 50, Appendix G.

The current heatup and cooldown curves and LTOP setpoints, will not be valid after 10 effective full power years (EFPY) cumulative core burnup. According to our most recent estimates in September, 1991, North Anna Unit 1 is expected to reach 10 EFPY in April, 1993 and Unit 2 in September, 1993. In anticipation of the expiration of these curves, Virginia Power has performed a safety evaluation to support implementing revised curves and setpoints. These new curve values and setpoints will be valid through 12 EFPY for North Anna Unit 1 and 17 EFPY for North Anna Unit 2.

Background

The heatup and cooldown curves are required by Appendix G of 10 CFR 50 and have been extrapolated to 12 EFPY and 17 EFPY for North Anna Units 1 and 2, respectively, by including the effects of the incremental radiation exposure on the reactor vessel beltline region. The results are referenced to the analyses of the North Anna Units 1 and 2 Capsule U results. The revised Appendix G curves were prepared using standard B&W and Westinghouse methodologies including those found in Regulatory Guide 1.99 Rev. 2. PORV setpoints were developed to provide bounding heatup and cooldown curve protection for the worst case mass and heat addition low temperature overpressure transients.

Technical Specification Changes

All Technical Specification changes described herein apply to North Anna Units 1 and 2 with the exception of Specification 3.4.1.2, which is only applicable to Unit 2.

Where possible, the Unit 2 specifications and associated bases have been modified to be consistent in content and format with the Unit 1 specifications. These changes are considered to be administrative in nature.

During the discussion of each specification, setpoints that are applicable to Unit 2 only, will be in parentheses.

Technical Specification 3.1.2.2
REACTIVITY CONTROL SYSTEMS - FLOW PATHS - OPERATING

An existing footnote to Technical Specification 3.1.2.2 has been revised to specify that only one boron 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 316°F (358°F for Unit 2). This requirement is provided to ensure consistency with the requirements of T.S. 3.1.2.4 (charging pump operability), and to ensure that actual operating conditions are consistent with those assumed in the mass addition transient analysis. The mass addition transient analysis assumes that only one charging pump will be operable below the temperature defined in T.S. 3.1.2.2. The 316°F value (358°F for Unit 2) corresponds to the pressurizer safety valve lift setpoint of 2485 psig, on the composite heatup and cooldown curve. Below these temperatures, the anticipated low temperature accidents may be adequately mitigated by the automatic action of the PORV. In addition, for Unit 2 only, this footnote has been applied to MODE 3 Applicability. This is necessary because the new Unit 2 setpoint involves MODE 3 operation between 350°F and 358°F.

Technical Specification 3.1.2.4
REACTIVITY CONTROL SYSTEMS - CHARGING PUMPS - OPERATING

An existing footnote to Technical Specification 3.1.2.4 has been revised to specify that 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 316°F (358°F for Unit 2). This requirement is provided to ensure that actual operating conditions are consistent with those assumed in the mass addition transient analysis. The mass addition transient analysis assumes that only one charging pump will be operable below the temperature defined in T.S. 3.1.2.4. The 316°F value (358°F for Unit 2) corresponds to the pressurizer safety valve lift setpoint of 2485 psig, on the composite heatup and cooldown curve.

In addition, for Unit 2 only, this footnote has been applied to the MODE 3 Applicability because the new Unit 2 setpoint involves MODE 3 operation between 350°F and 358°F.

Technical Specification 3.4.1.2
REACTOR COOLANT SYSTEM - SHUTDOWN -MODE 3

This change applies to Unit 2 only. A footnote was added to Specification 3.4.1.2 to specify that a reactor coolant pump shall not be started with one or more of the RCS cold leg temperatures less than or equal to 358°F unless the secondary water temperature of each steam generator is less than 50°F above each of the RCS cold leg temperatures. This is necessary because the new setpoint encompasses a small portion of MODE 3 (>350°F ≤358°F). This requirement is provided to ensure that actual operating conditions are consistent with those assumed in the heat addition transient analysis. The heat addition transient assumes that a 50 degree temperature differential exists between the secondary and primary sides of the steam generator when a reactor coolant pump is started. The 358°F value corresponds to the pressurizer safety valve lift setpoint of 2485 psig on the composite heatup and cooldown curve.

Technical Specification 3.4.1.3
REACTOR COOLANT SYSTEM - SHUTDOWN - MODES 4&5

An existing footnote to Technical Specification 3.4.1.3 has been revised to specify that a reactor coolant pump shall not be started with one or more of the RCS cold leg temperatures less than or equal to 316°F (358°F for Unit 2) unless the secondary water temperature of each steam generator is less than 50°F above each of the RCS cold leg temperatures. This requirement is provided to ensure that actual operating conditions are consistent with those assumed in the heat addition transient analysis. The heat addition transient assumes that a 50 degree temperature differential exists between the secondary and primary sides of the steam generator when a reactor coolant pump is started. The 316°F value (358°F for Unit 2) corresponds to the pressurizer safety valve lift setpoint of 2485 psig, on the composite heatup and cooldown curve.

Technical Specification 3.4.9.1
REACTOR COOLANT SYSTEM - PRESSURE/TEMPERATURE LIMITS

T.S. 3.4.9.1 indicates that T.S. Figures 3.2-2 and 3.4-3 provide pressure and temperature operating limitations at criticality. The criticality limit line has been eliminated from the heatup curve in Figure 3.4-2 in favor of the more restrictive T.S. 3.1.1.5 minimum temperature for criticality. The text of T.S. 3.4.9.1 has been modified to reflect this change.

Technical Specification Figures 3.4-2 and 3.4-3
PRESSURE/TEMPERATURE LIMITS - HEAT UP&COOLDOWN CURVES

Revised T.S. Figures 3.4-2 and 3.4-3 have been prepared presenting the revised Unit 1, 12 EFPY and Unit 2, 17 EFPY heatup and cooldown curves. The revised curves do not include allowances for temperature and measurement uncertainty. The bases have been modified to reflect this change. The development of these curves is discussed in greater detail in a later section. The criticality limit line has been excluded from the heatup curve in Figure 3.4-2 in favor of the more restrictive Technical Specification 3.1.1.5, Minimum Temperature For Criticality. (The Material Property Bases table, presented on these figures, has been added to the Bases.)

Technical Specification 3.4.9.3
RCS - OVERPRESSURE PROTECTION SYSTEMS

The LTOP setpoints have been changed to provide bounding heatup and cooldown curve protection. The development of these revised setpoints is discussed in greater detail in a later section. For Unit 2, LCO 3.9.4.3 (c), has been eliminated. The maximum pressurizer water volume requirement has been eliminated in favor of requiring low temperature overpressure protection via automatic operation of the LTOP system below 321°F. The maximum pressurizer water volume requirement was eliminated from the Unit 1 Specification 3.4.9.3 in a previous amendment[3].

Technical Specification 3.5.2
EMERGENCY CORE COOLING SUBSYSTEMS - Tavg ≥ 350°F

ACTION c, has been revised to allow the provisions of Specification 3.0.4 to be not applicable to ACTIONS a and b for one hour following heatup above 316°F (358°F for Unit 2) or prior to cooldown below 316°F (358°F for Unit 2).

Also, for Unit 2 only, a footnote and a # sign have been added which are applicable to LCO 3.5.2 (a) and (b). This added footnote specifies that 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 358°F for Unit 2 and is necessary because the new Unit 2 setpoint involves MODE 3 operation between 350°F and 358°F.

These changes ensure that actual operating conditions are consistent with those assumed in the mass addition transient analysis. The mass addition transient analysis assumes that only one charging pump will be operable below 316°F (358°F for Unit 2). The 316°F value (358°F for Unit 2) corresponds to the pressurizer safety valve lift setpoint of 2485 psig, on the composite heatup and cooldown curve.

Technical Specification 3.5.3
EMERGENCY CORE COOLING SUBSYSTEMS - Tavg < 350°F

The existing footnote in the Specification has been revised such that 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 316°F (358°F for Unit 2). These changes ensure that actual operating conditions are consistent with those assumed in the mass addition transient analysis. The mass addition transient analysis assumes that only one charging pump will be operable below 316°F (358°F for Unit 2). The 316°F value (358°F for Unit 2) corresponds to the pressurizer safety valve lift setpoint of 2485 psig, on the composite heatup and cooldown curve.

Conclusions

The attached Technical Analysis supports the following conclusions:

The heatup and cooldown curves required by Appendix G of 10 CFR 50 have been extrapolated to 12 EFPY and 17 EFPY for North Anna Units 1 and 2, respectively, by including the effects of the incremental radiation exposure on the reactor vessel beltline region. The results are referenced to the analyses of the North Anna Units 1 and 2 Capsule U results. The revised Appendix G curves were prepared using standard B&W and Westinghouse methodologies including Regulatory Guide 1.99 Rev. 2. LTOP setpoints were developed to provide bounding heatup and cooldown curve protection for the worst case mass and heat addition low temperature overpressure transients.

The next Unit 1 reactor vessel surveillance capsule (Capsule X) is scheduled to be removed after the tenth fuel cycle (10 EFPY) which allows sufficient time for analysis prior to exceeding 12 EFPY. The next Unit 2 reactor vessel surveillance capsule (Capsule W) is scheduled to be removed after the thirteenth fuel cycle (15 EFPY) which allows sufficient time for analysis prior to exceeding 17 EFPY.

The heatup and cooldown curves prepared by B&W and Westinghouse were determined in a conventional manner according to Section III of the ASME code as required by 10 CFR 50 Appendix G. Both steady-state and transient thermal conditions were considered in order to bound the possible combinations of pressure (i.e. membrane) and thermal stresses.

The new North Anna Unit 1 low temperature overpressure protection system PORV lift settings should be less than or equal to 450 psig whenever any RCS cold leg temperature is less than or equal to 270°F, and less than or equal to 300 psig whenever any RCS cold leg temperature is less than 150°F.

The new North Anna Unit 2 low temperature overpressure protection system PORV lift settings should be less than or equal to 510 psig whenever any RCS cold leg temperature is less than or equal to 321°F, and less than or equal to 360 psig whenever any RCS cold leg temperature is less than 210°F.

Pressurized Thermal Shock (PTS) evaluations were made for the limiting beltline locations. It was demonstrated that: (1) predicted end-of-license fluences do not result in RT_{PTS} values in excess of the screening criteria when calculated using the methodology of Regulatory Guide 1.99, Revision 2; (2) there is an excellent comparison between experimentally determined and calculated vessel fluences; and (3) the extrapolated fluences at the burnup limit to which the revised heatup and cooldown curves are applicable for each unit are significantly less than the extrapolated end-of-license fluences (which have been demonstrated to not result in a violation of PTS screening criteria). On this basis it may be concluded that there is neither a significant change in predicted RT_{PTS} values; nor is there a PTS concern for either unit up to the burnup limit to which the revised heatup and cooldown curves are valid.

ATTACHMENT 2

PROPOSED TECHNICAL SPECIFICATIONS CHANGES

TECHNICAL SPECIFICATION CHANGE REQUEST

FINAL COPY

NORTH ANNA POWER STATION - UNIT 1

VIRGINIA ELECTRIC AND POWER COMPANY

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% Δk/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 316°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% Δk/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 316°F or prior to cooldown below 316°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 316°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 316°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.

---

* A reactor coolant pump shall not be started with one or more of the RCS cold leg temperatures less than or equal to 316°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.

3/4.4.9 PRESSURE/TEMPERATURE LIMITS

REACTOR COOLANT SYSTEM

LIMITING CONDITION FOR OPERATION

3.4.9.1 The Reactor Coolant System (except the pressurizer) temperature and pressure shall be limited in accordance with the limit lines shown in Figures 3.4-2 and 3.4-3 during heatup, cooldown, and inservice leak and hydrostatic testing with:

a. A maximum heatup of 60°F in any one hour period.

b. A maximum cooldown of 100°F in any one hour period.

c. A maximum temperature change of less than or equal to 10°F in any one hour period during inservice hydrostatic and leak testing operations above the heatup and cooldown limit curves.

APPLICABILITY: At all times.

ACTION:

With any of the above limits exceeded, restore the temperature and/or pressure to within the limit within 30 minutes; perform an engineering evaluation to determine the effects of the out-of-limit condition on the structural integrity of the Reactor Coolant System; determine that the Reactor Coolant System remains acceptable for continued operations or be in at least HOT STANDBY within the next 6 hours and reduce the RCS Tavg and pressure to less than 200°F and 500 psig, respectively, within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.4.9.1.1 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.

4.4.9.1.2 The reactor vessel material irradiation surveillance specimens shall be removed and examined, to determine changes in material properties, at the intervals required by 10 CFR 50, Appendix H. The results of these examinations shall be used to update Figures 3.4-2 and 3.4-3.

Figure 3.4-2 Unit 1 RCS HEATUP P/T Limits
Valid to 12 EFPY Heatup Rates: 0-60°F/Hr.
(Margins for Instrument Errors NOT Included)

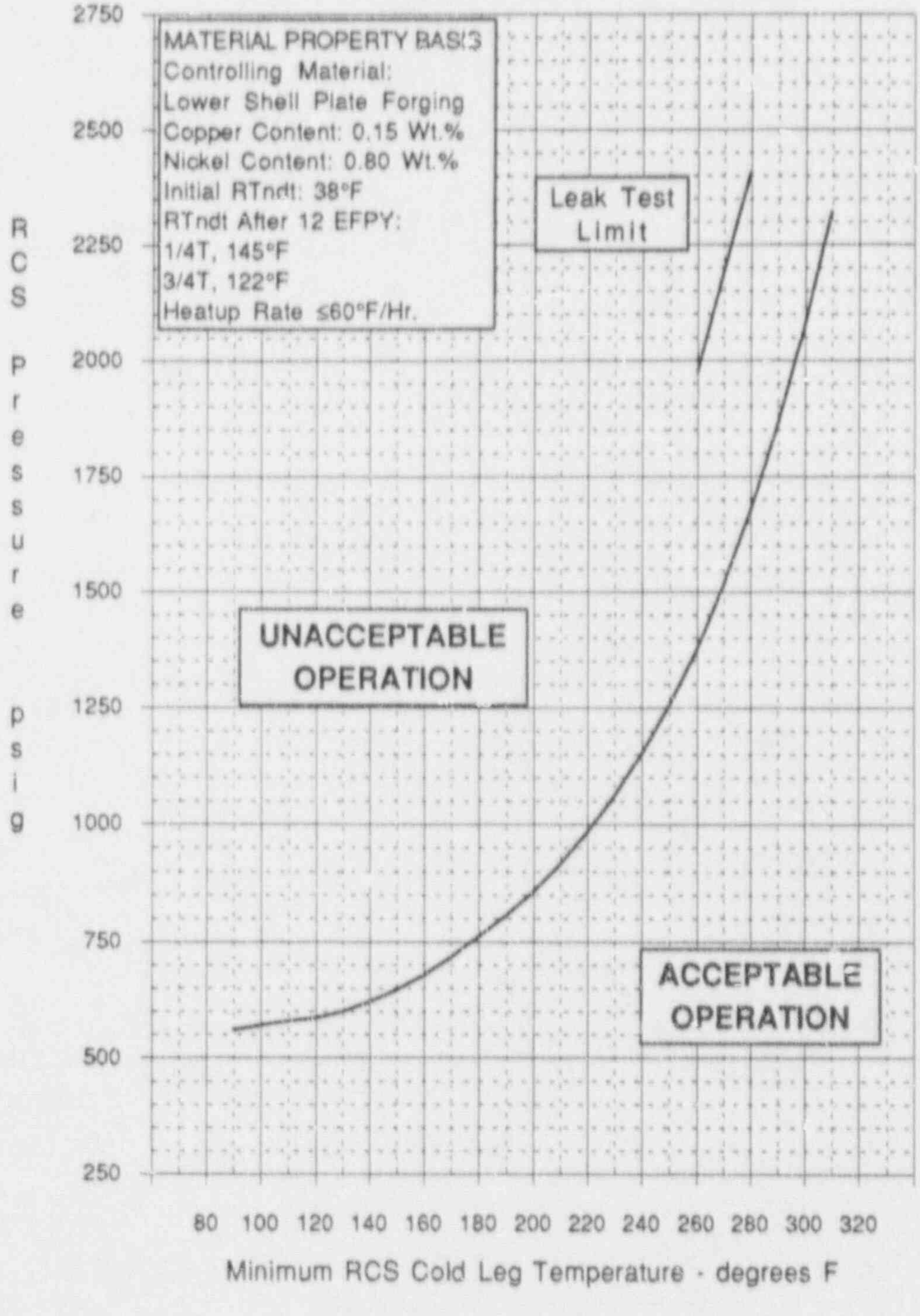
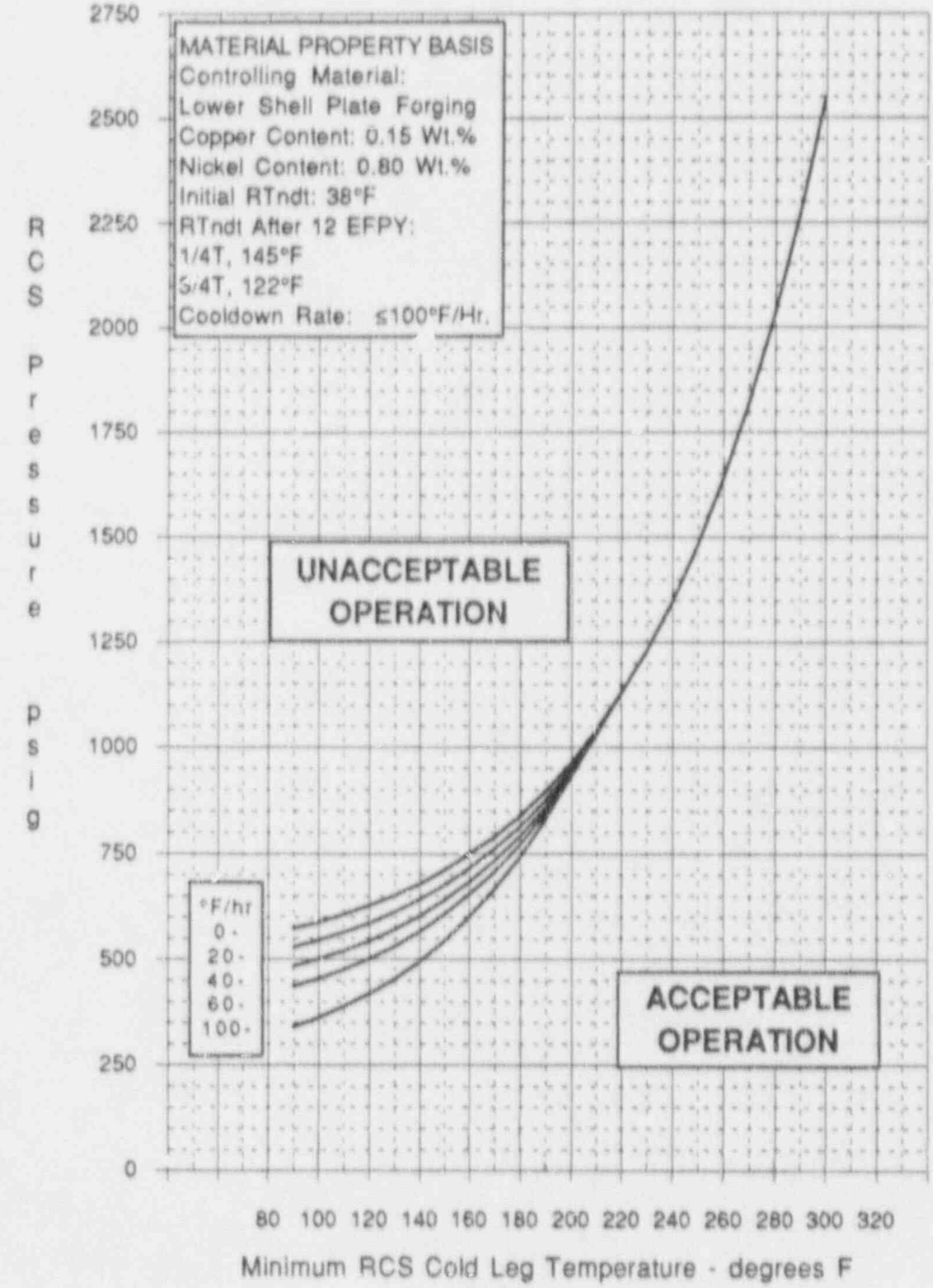


Figure 3.4-3 Unit 1 RCS COOLDOWN P/T Limits
Valid to 12 EFPY Cooldown Rates: 0-100°F/Hr.
(Margins for Instrument Errors NOT Included)



LIMITING CONDITION FOR OPERATION

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

a. Two power operated relief valves (PORVs) with a lift setting of: 1) less than or equal to 450 psig whenever any RCS cold leg temperature is less than or equal to 270°F, and 2) less than or equal to 390 psig whenever any RCS cold leg temperature is less than 150°F, or

b. A reactor coolant system vent of greater than or equal to 2.07 square inches.

APPLICABILITY: When the temperature of one or more of the RCS cold legs is less than or equal to 270°F, except when the reactor vessel head is removed.

ACTION:

a. With one PORV inoperable, either restore the inoperable PORV to OPERABLE status within 7 days or depressurize and vent the RCS through 2.07 square inch vent(s) within the next 8 hours; maintain the RCS in a vented condition until both PORVs have been restored to OPERABLE status.

b. With both PORVs inoperable, depressurize and vent the RCS through a 2.07 square inch vent(s) within 8 hours; maintain the RCS in a vented condition until both PORVs have been restored to OPERABLE status.

c. In the event either the PORVs or the RCS vent(s) are used to mitigate a 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.

d. The provisions of Specification 3.0.4 are not applicable.

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 316°F or prior to cooldown below 316°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 Tavg 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 316°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 316°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% Δk/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 316°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% Δk/k after xenon 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 a 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 316°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 staring 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

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 phenomenon which may occur following changes in THERMAL POWER.

Reducing Tavg 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

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 are prepared based upon the most limiting value of the predicted adjusted reference temperature at the end of 12 EFPY. The adjusted reference temperature was calculated using results from a capsule removed after the sixth fuel cycle. The results are documented in Westinghouse Report WCAP-11777, February 1988 and Babcock and Wilcox Report BAW-2146, October, 1991.

The reactor vessel materials have been tested to determine their initial RT_{NDT}. The results of these tests are shown in the UFSAR and WCAP-11777. Reactor operation and resultant fast neutron (E>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 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 12 EFPY. The reactor vessel beltline region material properties are listed in Table B.3.4-1.

The actual shift in RT_{NDT} of the vessel material will be established periodically during operation by removing and evaluating, in accordance with ASTM E185-70, reactor vessel material irradiation surveillance specimens installed near the inside wall of the reactor vessel in the core area. 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 ΔRT_{NDT} determined from the surveillance capsule is different from the calculated ΔRT_{NDT} for the equivalent capsule radiation exposure.

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 and WCAP-11777 to assure compliance with the requirements of Appendix H to 10 CFR Part 50.

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.

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.

Table B.3.4-1

MATERIAL PROPERTY BASIS

Controlling Material:	Lower Shell Plate Forging
Copper Content:	0.15 Wt.%
Nickel Content:	0.80 Wt.%
Initial RTndt:	38°F
RTndt After 12 EFPY:	1/4T, 145°F
	3/4T, 122°F
Cooldown Rate:	≤100°F/Hr.
Heatup Rate:	≤60°F/Hr.

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 316°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.

ATTACHMENT 3

PROPOSED TECHNICAL SPECIFICATIONS CHANGES

TECHNICAL SPECIFICATION CHANGE REQUEST

FINAL COPY

NORTH ANNA POWER STATION - UNIT 2

VIRGINIA ELECTRIC AND POWER COMPANY

FLOW PATHS - OPERATING

LIMITING CONDITION FOR OPERATION

3.1.2.2 At least two of the following three 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 358°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 358°F or prior to cooldown below 358°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 358°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 358°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 associa'ed steam generator and reactor coolant pump,*

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

APPLICABILITY: MODE 3

ACTIC.,:

a. With less than the above required loops OPERABLE, restore the required loops to OPERABLE status with in 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.

---

* A reactor coolant pump shall not be started with one or more of the RCS cold leg temperatures less than or equal to 358°F unless the secondary water temperature of each steam generator is less than 50°F above each of the RCS cold leg temperatures.

** 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.

---

* A reactor coolant pump shall not be started with one or more of the RCS cold leg temperatures less than or equal to 358°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.

3/4.4.9 PRESSURE/TEMPERATURE LIMITS

REACTOR COOLANT SYSTEM

LIMITING CONDITION FOR OPERATION

3.4.9.1 The Reactor Coolant System (except the pressurizer) temperature and pressure shall be limited in accordance with the limit lines shown in Figures 3.4-2 and 3.4-3 during heatup, cooldown, and inservice leak and hydrostatic testing with:

a. A maximum heatup of 60°F in any one hour period.

b. A maximum cooldown of 100°F in any one hour period.

c. A maximum temperature change of less than or equal to 10°F in any one hour period during inservice hydrostatic and leak testing operations above the heatup and cooldown limit curves.

APPLICABILITY: At all times.

ACTION:

With any of the above limits exceeded, restore the temperature and/or pressure to within the limit within 30 minutes; perform an engineering evaluation to determine the effects of the out-of-limit condition on the structural integrity of the Reactor Coolant System; determine that the Reactor Coolant System remains acceptable for continued operations or be in at least HOT STANDBY within the next 6 hours and reduce the RCS Tavg and pressure to less than 200°F and 500 psig, respectively, within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.4.9.1.1 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.

4.4.9.1.2 The reactor vessel material irradiation surveillance specimens shall be removed and examined, to determine changes in material properties, at the intervals required by 10 CFR 50, Appendix H. The results of these examinations shall be used to update Figures 3.4-2 and 3.4-3.

Figure 3.4-2 Unit 2 RCS HEATUP P/T Limits
Valid to 17 EFPY Heatup Rates: 0-60°F/Hr.
(Margins for Instrument Errors NOT Included)

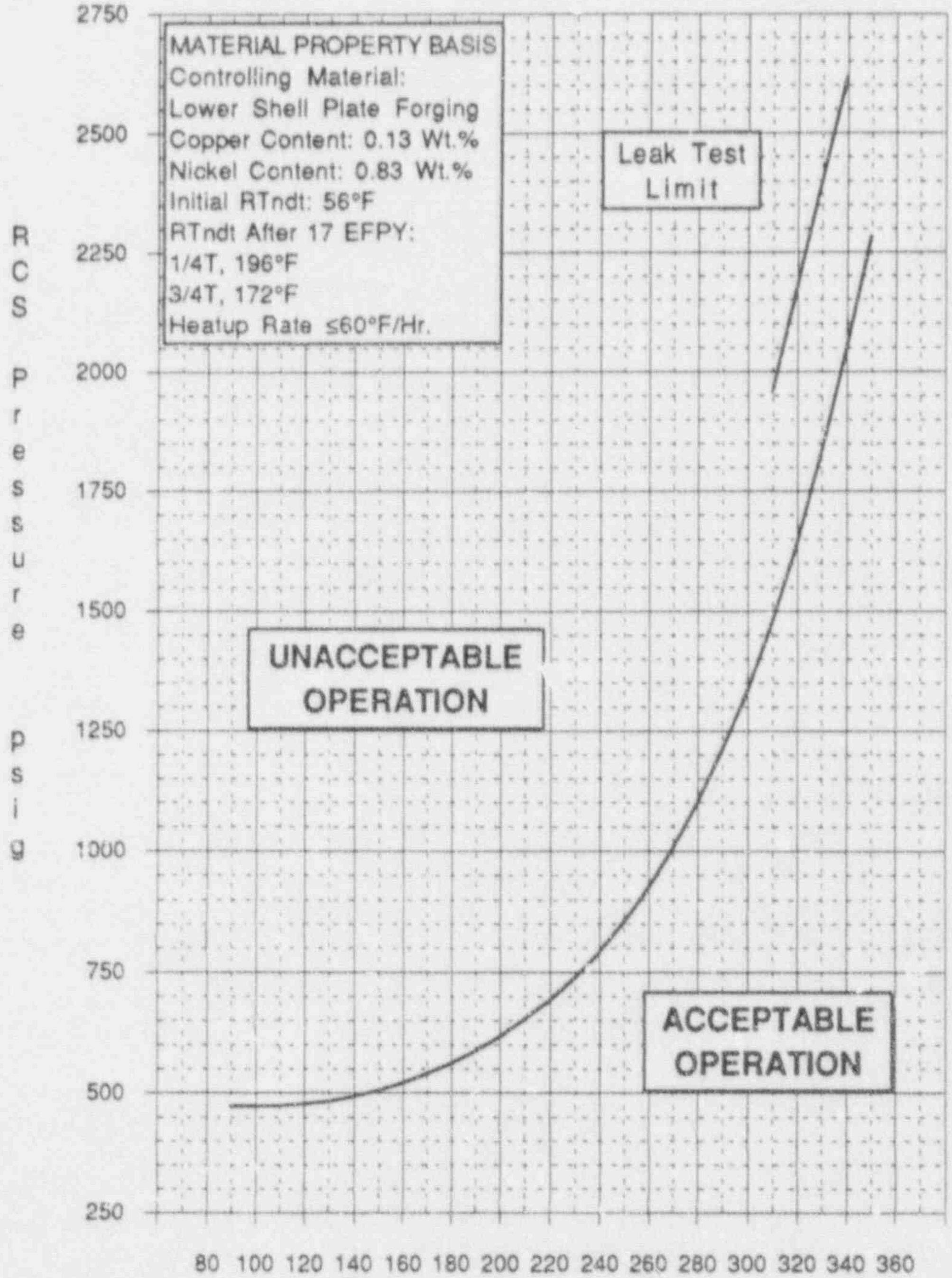
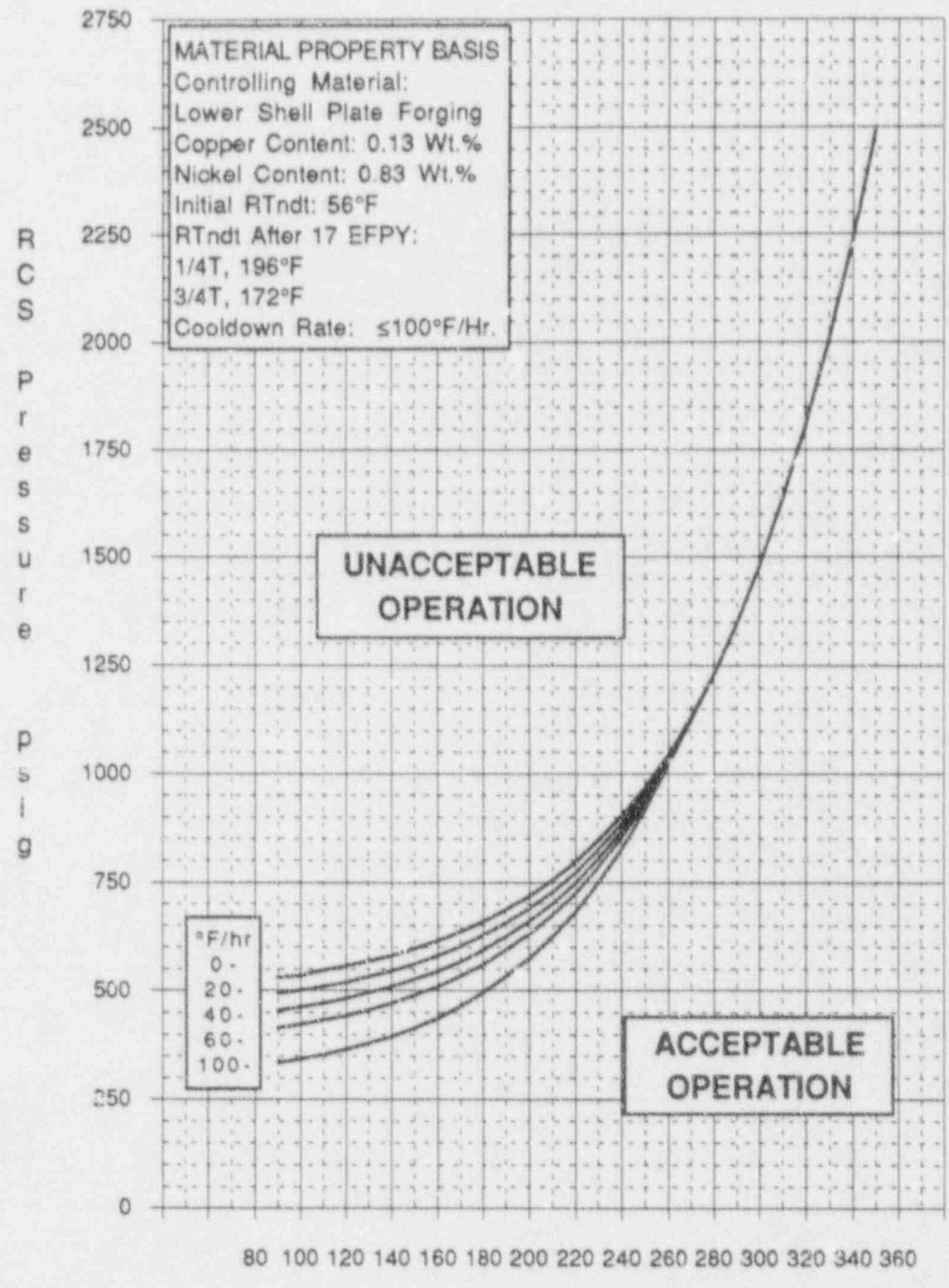


Figure 3.4-3 Unit 2 RCS COOLDOWN P/T Limits
Valid to 17 EFPY Cooldown Rates: 0-100°F/Hr.
(Margins for Instrument Errors NOT Included)



LIMITING CONDITION FOR OPERATION

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

a. Two power operated relief valves (PORVs) with a lift setting of: 1) less than or equal to 510 psig whenever any RCS cold leg temperature is less than or equal to 321°F, and 2) less than or equal to 360 psig whenever any RCS cold leg temperature is less than 210°F, or

b. A reactor coolant system vent of greater than or equal to 2.07 square inches.

APPLICABILITY: When the temperature of one or more of the RCS cold legs is less than or equal to 321°F, except when the reactor vessel head is removed.

ACTION:

a. With one PORV inoperable, either restore the inoperable PORV to OPERABLE status within 7 days or depressurize and vent the RCS through 2.07 square inch vent(s) within the next 8 hours; maintain the RCS in a vented condition until both PORVs have been restored to OPERABLE status.

b. With both PORVs inoperable, depressurize and vent the RCS through a 2.07 square inch vent(s) within 8 hours; maintain the RCS in a vented condition until both PORVs have been restored to OPERABLE status.

c. In the event either the PORVs or the RCS vent(s) are used to mitigate a 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.

d. The provisions of Specification 3.0.4 are not applicable.

ECCS SUBSYSTEMS - Tavg 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 Specifications 3.5.2.a and 3.5.2.b for one hour following heatup above 358°F or prior to cooldown below 358°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:

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 358°F.

ECCS SUBSYSTEMS - Tavg 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 Tavg 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 358°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 358°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 358°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 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 a 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 358°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 staring 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 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.

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 phenomenon which may occur following changes in THERMAL POWER.

Reducing Tavg 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

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 are prepared based upon the most limiting value of the predicted adjusted reference temperature at the end of 17 EFPY. The adjusted reference temperature was calculated using results from a capsule removed after the sixth fuel cycle. The results are documented in Westinghouse Report WCAP-12497, January 1990 and WCAP-12503, March, 1990.

The reactor vessel materials have been tested to determine their initial RT_{NDT}. The results of these tests are shown in the UFSAR and WCAP-12497. Reactor operation and resultant fast neutron (E>1 Mev) irradiation will cause an increase in the RT_{NDT}. Therefore, an adjusted refe ce temperature, based upon the fluence and copper content of the material in questior · be predicted using US NRC Regulatory Guide 1.98, Revision 2. The heatup and coo' .vn 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 in Table B.3.4-1.

The actual shift in RT_{NDT} of the vessel material will be established periodically during operation by removing and evaluating, in accordance with ASTM E185-70, reactor vessel material irradiation surveillance specimens installed near the inside wall of the reactor vessel in the core area. 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 ΔRT_{NDT} determined from the surveillance capsule is different from the calculated ΔRT_{NDT} for the equivalent capsule radiation exposure.

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 and WCAP-12497 to assure compliance with the requirements of Appendix H to 10 CFR Part 50.

The limitations imposed on pressurizer heatup and cooldown and spray water temperature differential are prov''ed 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 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 321°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 321°F. This temperature is the water temperature corresponding to a metal temperature of at least the limiting RT_{NDT}+90°F+ instrument uncertainty. Above 321°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.

Table B.3.4-1

MATERIAL PROPERTY BASIS

Controlling Material:	Lower Shell Plate Forging
Copper Content:	0.13 Wt.%
Nickel Content:	0.83 Wt.%
Initial RTndt:	56°F
RTndt After 17 EFPY:	1/4T, 196°F
	3/4T, 172°F
Cooldown Rate:	≤100°F/Hr.
Heatup Rate:	≤60°F/Hr.

3/4.4.10 STRUCTURAL INTEGRITY

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

The inspection program 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.

Pages B 3/4 4-9 thru B 3/4 4-16 have been deleted.

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 358°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 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.

ATTACHMENT 4

DETERMINATION OF NO SIGNIFICANT HAZARDS CONSIDERATIONS
(10 CFR 50.92 EVALUATION)

TECHNICAL SPECIFICATION CHANGE REQUEST

NORTH ANNA UNITS 1 & 2

VIRGINIA ELECTRIC AND POWER COMPANY

BASIS FOR NO SIGNIFICANT HAZARDS DETERMINATION

The North Anna Reactor Coolant Systems (RCS), specifically the Reactor Pressure Vessels (RPV), are protected from material failure during low temperature operations by imposing restrictions on RCS pressure. The heatup and cooldown curves as well as the Low Temperature/Over-pressure Protection System (LTOP) setpoints, provide the restrictions to bound the area of operation and ensure RCS protection from non-ductile failure. The regulatory requirements for providing these restrictions and reevaluating them, are stipulated in 10 CFR 50, Appendix G.

The current heatup and cooldown curves and LTOP setpoints, will not be valid after 10 effective full power years (EFPY) cumulative core burnup. According to our most recent estimates (September, 1991) North Anna Unit 1 is expected to reach 10 EFPY in April, 1993 and Unit 2 in September, 1993. In anticipation of the expiration of these curves, Virginia Power has performed a safety evaluation to support implementing revised curves and setpoints.

These revised curves required by Appendix G of 10 CFR 50 have been extrapolated to 12 EFPY and 17 EFPY for North Anna Units 1 and 2, respectively, by including the effects of the incremental radiation exposure on the reactor vessel beltline region. The revised curves were prepared using standard methodologies and Regulatory Guide 1.99 Rev. 2. LTOP setpoints were developed to provide bounding heatup and cooldown curve protection for the worst case mass and heat addition low temperature overpressure transients.

The proposed changes to Technical Specifications are required to support implementation of the revised heatup and cooldown pressure-temperature limitations on the Reactor Coolant System and the revised setpoints for the low temperature - overpressure protection (LTOP) system. Operation of North Anna Power Station Units 1 & 2 in accordance with this change will not involve a significant hazards consideration as defined in 10 CFR 50.92 because it will not:

1. result in a significant increase in the probability or consequence of an accident previously evaluated. The application of the revised pressure-temperature limitations and the revised LTOP setpoints, results in greater restrictions on the operation of the units and will insure that the requirements of 10 CFR 50, Appendix G, for fracture toughness of the Reactor Coolant System pressure boundary will continue to be satisfied.

2. create the possibility of a new or different kind of accident from any accident previously identified. There will be greater restrictions on the operation of North Anna Power Station Units 1 & 2 using the revised pressure-temperature limitations and the revised LTOP setpoints. These restrictions will insure that the requirements of 10 CFR 50, Appendix G, for fracture toughness of the Reactor Coolant System pressure boundary will continue to be satisfied. The proposed amendments will not result in other changes in the way the units are operated.

3. result in a significant reduction in a margin of safety. The revised pressure-temperature limitations and the revised LTOP setpoints will insure that the requirements of 10 CFR 50, Appendix G, for fracture toughness of the Reactor Coolant System pressure boundary will continue to be satisfied. The safety factors defined in the ASME Code and the requirements of 10 CFR 50 Appendix G provide the basis for the applicable safety margins. The plant specific information obtained from the testing of the sample vessel material and utilized to develop the revised pressure-temperature limitations and the revised LTOP setpoints will confirm that these safety margins are not reduced.

Therefore, pursuant to 10 CFR 50.92, based on the above considerations, it has been determined that this change does not involve a significant hazards consideration.

ATTACHMENT 5

ADDITIONAL SUPPORTING INFORMATION
(NA&F TECHNICAL ANALYSIS)
(BAW-2146)
(WCAP-12503)
(REFERENCE LIST)

TECHNICAL SPECIFICATION CHANGE REQUEST

NORTH ANNA POWER STATION UNITS 1 & 2

VIRGINIA ELECTRIC AND POWER COMPANY

NA&F TECHNICAL ANALYSIS

TECHNICAL ANALYSIS

Surveillance Capsule Analysis Results

The North Anna Units 1[4] and 2[5] reactor vessel materials surveillance program Capsule U analysis reports were transmitted to the NRC in June 1988[and March 1990[6], respectively. These reports provide the basis for the fluence projections and RT_{NDT} values used in the generation of revised heatup and cooldown curves.

North Anna Unit 1 Capsule Analysis Results

Capsule U was removed from North Anna Unit 1 at the end of the sixth cycle of operation. The capsule dosimeters were evaluated and found to have a cumulative fast neutron, E > 1.0 Mev, fluence of 8.28×10^{18} n/cm^2. The calculated fast neutron fluence based on actual cycle power distributions at the capsule location was 8.85×10^{18} n/cm^2 which compares favorably (within 7%) with the dosimeter fluence. The peak fluence at the inside surface of the reactor vessel was calculated to be 8.83×10^{18} n/cm^2 which shows that the capsule has been exposed to slightly more neutrons than the vessel[4].

The material property testing included Charpy V-notch impact testing and tension testing of several specimens located within the surveillance capsule. The Charpy tests are performed to determine the transition temperature increases at 30 ft-lb and 50 ft-lb points, and the decrease in the upper shelf energy. The tensile specimens were used to determine ultimate tensile strength and yield strength. The vessel specimens within Capsule U were obtained from the same girth weld and forging materials as those used in the reactor vessel beltline[4].

The irradiated specimens test results were compared to unirradiated specimen test results. The Charpy V-notch impact test results show the irradiation has increased the average 50 ft-lb transition temperature by 80 to 110°F depending on the specimen metal. Irradiation has increased the average 30 ft-lb transition temperature by 65 to 100°F. The upper shelf energy (average energy absorption at full shear) results show the worst decrease to be 25 ft-lb when comparing irradiated samples to unirradiated samples.

The lowest average upper shelf energy was determined to be 92 ft-lb which is greater than the 10 CFR 50 Appendix G low limit of 50 ft-lb.[11]. The Charpy impact test results from Capsule U were also satisfactorily compared to the Capsule V results. Tension test results show a slight increase in the ultimate tensile strength and the yield strength due to irradiation. Reference 4 should be consulted for specific test results.

In accordance with the methods prescribed by Regulatory Guide 1.99, Revision 2, the lower shell forging adjusted RT_{NDT} at the end of 12 EFPY for Unit 1 was calculated utilizing the data from the surveillance capsule analysis. Although the use of surveillance data for the calculation of RT_{NDT} results in a lower value of RT_{NDT} than is obtained when surveillance data is not used, the limiting 12 EFPY values of RT_{NDT} at the 1/4T and 3/4T locations were shown to occur in the Unit 1 lower shell forging.

The limiting material in the existing Unit 1, 10 EFPY curves was determined to be the lower shell circumferential well[3].

North Anna Unit 2 Capsule Analysis Results

Capsule U was removed from North Anna Unit 2 at the end of the sixth cycle of operation. The capsule dosimeters were evaluated and found to have a cumulative fast neutron, E > 1.0 Mev, fluence of 9.55×10^{18} n/cm^2. The calculated fast neutron fluence at the capsule location was 1.06×10^{19} n/cm^2, which compares favorably (within 11%) with the dosimeter fluence. The peak calculated fluence at the inside surface of the reactor vessel was calculated to be 8.02×10^{18} n/cm^2 which shows that the capsule has been exposed to slightly more neutrons than the vessel[5].

The material property testing included Charpy V-notch impact testing and tension testing of several specimens located within the surveillance capsule. The Charpy tests are performed to determine the transition temperature increases at 30 ft-lb and 50-ft-lb points, and the decrease in the upper shelf energy. The tensile specimens were used to determine ultimate tensile strength and yield strength. The vessel specimens within Capsule U were obtained from the same girth weld and forging materials at those used in the reactor vessel beltline[5].

The irradiated specimens test results were compared to unirradiated specimen test results. The Charpy V-notch impact test results show the irradiation has increased the average 50 ft-lb transition temperature between 25°F to 55°F depending on the specimen metal. Irradiation has increased the average 30 ft-lb transition temperature between 25 and 60°F depending on the specimen metal. The upper shelf energy (average energy absorption at full shear) results showed no decrease in the average upper shelf energy of the forging and weld metals. Both surveillance materials exhibit a more than adequate upper shelf level for continued safe plant operation[5]. Tension test results show a slight increase in the ultimate tensile strength and the yield strength due to irradiation. A comparison of the 30 ft-lb transition temperature increases for the Unit 2 surveillance material with predicted increases using the methods of NRC Regulatory Guide 1.99, Revision 2, demonstrated that forging and weld metal transition temperature increases were less than predicted. Reference 5 should be consulted for specific test results.

Heatup and Cooldown Curves

Heatup and Cooldown Curve Generation

10 CFR 50 Appendix G establishes fracture toughness requirements for the reactor vessel. Virginia Power utilizes two types of graphs to identify plant specific limits. The graphs are known as heatup and cooldown curves. The proposed heatup curve depicts two curves: the leak test limit and the heatup limit (up to 60°F/hr). The criticality limit curve has been omitted as it is redundant with respect to the existing T.S. 3.1.1.5 criticality limitation. The cooldown curve depicts a series of curves for a range of assumed cooldown rates (0, 20, 40, 60, and 100°F/hr). The control room operators use these Technical Specification graphs to ensure pressure and temperature are within acceptable values.

The heatup and cooldown curve analysis developed pressure-temperature relationships for steady-state conditions as well as finite heatup rate conditions assuming the presence of a 1/4T defect at the inside of the vessel wall. The thermal gradients during heatup produce compressive stresses at the inside of the wall that alleviate the tensile stresses produced by internal pressure. Therefore, steady-state conditions can be limiting for the inside wall so both heatup and steady state must be considered.

The heatup curve calculations must also consider the case of a 1/4T flaw at the outside surface. The thermal and pressure stresses never cancel for this situation. The thermal stresses are dependent on both the rate of heatup and the coolant temperature along the heatup ramp.

The use of a composite curve is required to make sure that the limiting condition is always protected against. For example, protection must be provided if the limiting location shifts from the inside to the outside surface. Therefore, a composite heatup curve is generated by comparing on a point-by-point basis the steady-state curve at the inside of the wall along with the various heatup rate curves at the outside surface. Thus at any given temperature, the allowable pressure is taken to be the most limiting of the values from each of the curves under consideration.

During cooldown, the controlling location of the flaw is always at the inside of the wall because the thermal gradients produce tensile stresses at the inside, which increase with increasing cooldown rates. Allowable pressure-temperature relations are generated for both steady-state and finite cooldown rate situations. A lower bound composite curve from the steady state and cooldown conditions is constructed for each cooldown rate of interest.

The use of a composite curve in the cooldown analysis is necessary because control of the cooldown procedure is based on measurement of reactor coolant temperature, whereas the limiting pressure is actually dependent on the material temperature at the tip of the assumed flaw. During cooldown the tip is at a higher temperature than the fluid adjacent to the vessel inner wall. This condition is not true for the steady-state situation. It follows that at any given reactor coolant temperature, the temperature gradient developed during cooldown results in a higher value of K_{IR} at the 1/4T location for finite cooldown rates than for steady-state operation. So, if the temperature changes such that K_{IR} increases faster than K_{IT}, the steady-state can be limiting.

Revised heatup and cooldown curves valid to 12 EFPY and 17 EFPY have been generated for North Anna Units 1[9] and 2[10], respectively. The reports documenting the development of these curves are presented in Appendices A and B.

Heatup and Cooldown Curve Interpretation

Babcock and Wilcox (B&W) and Westinghouse have provided Virginia Power with revised heatup and cooldown curves for North Anna 1[9] and 2[10], respectively. Figures 1 through 4 present these revised Technical Specification heatup curves and cooldown curves. Data for these figures is presented in the attached References 9 and 10. Although the proposed Technical Specifications heatup curves present only the curve generated for a 60°F/hr heatup rate, heatup curves were generated for heatup rates of 20°, 40°, and 60°F/hr. The cooldown curves were generated for 20°, 40°, 60°, and 100°F/hr cooldown rates. The steady state condition was also considered. As indicated in the figures, the curves are based on extrapolated fluences to permit operation to the specified cumulative core average burnup.

The proposed revised heatup curve does not contain the 10 CFR 50, Appendix G criticality limit. The criticality limit is not required since limiting condition for operation (LCO) 3.1.1.5 restricts the lowest operating loop average temperature to ≥ 541°F for Modes 1 and 2. This LCO defines a minimum temperature for criticality that provides substantially more margin to the heatup curve than the criticality limits required by 10 CFR 50, Appendix G.

A previous discussion of the composite cooldown curve indicated that the reactor vessel wall has a temperature gradient dependent on cooldown rate. Because of this temperature gradient, and because higher cooldown rates are not realistically possible at low RCS temperatures, lower achievable cooldown rates at lower temperatures were assumed. Table 1 presents the assumed maximum cooldown rate for various temperatures. These lower achievable cooldown rates permit the elimination of the most restrictive (low temperature) portions of the 20°, 40°, 60°, and 100°F/hr cooldown curves for the purpose of establishing LTOP setpoints.

Table 2 presents assumed maximum heatup rates for various temperatures. As in the case of the cooldown curves, these lower assumed heatup rates permit the elimination of the most restrictive portions of the 20° and 40°F/hr heatup curves for the purpose of establishing LTOP setpoints. The temperature-dependent heatup and cooldown rates will be incorporated into the Unit 1 and 2 plant operating procedures to ensure that they are not exceeded during normal operation.

After modifying the cooldown curve and heatup curve to reflect lower assumed cooldown and heatup rates at lower temperatures, the most limiting points from each curve were selected at each temperature to construct a composite curve for use in the development of revised LTOP PORV setpoints. Tables 3 and 4 present the composite, modified heatup and cooldown curve for Units 1 and 2, respectively.

Since the design basis transients are defined with operational assumptions related to the pressurizer safety valve setpoints, certain operational restrictions must be enforced to ensure the low temperature accident analysis assumptions are valid. The temperature on the composite, modified heatup and cooldown curve corresponding to the pressurizer safety valve setpoint of 2485 psig is 316°F (358°F for Unit 2). This point is used to bound all of the low temperature accident analyses. The mass addition transient assumes only one charging pump will be operable below 316°F (358°F for Unit 2). The heatup transient assumes whenever a RCP is started below 316°F (358°F for Unit 2) the temperature difference between the primary and secondary fluids in the Steam Generator is less than 50°F.

Because the Unit 2 temperature which corresponds to 2485 psig (358°F) is greater than 350°F, it was necessary to apply the single charging pump operability requirement to MODE 3. The implications of this change on postulated events at RCS temperatures equal to or less than 358°F must be considered. Because the stored energy of the core would be low at conditions where RCS temperatures are less than or equal to 358°F, the MODE 3 single charging pump operability requirement presents no safety concerns. It is reasoned that although one of the charging pumps would not be available for automatic initiation, it would still be available for manual initiation. Therefore, it has been concluded that extension of this requirement to 358°F has no impact of the ability of the operator to mitigate the consequences of postulated events.

PORV Setpoints

Background

Cold overpressure protection is provided to ensure that the normal operation heatup and cooldown curves are not violated during operation with a water solid system. The PORVs on the pressurizer are set at a pressure low enough to prevent violation of the composite, modified heatup and cooldown curve should a RCS pressure transient occur. The limits have been set by two design basis accidents: the inadvertent start of a charging pump and the startup of a reactor coolant pump in an RCS loop with a 50°F difference between the steam generator secondary fluid temperature and the RCS temperature. These transients represent the limiting mass addition and heat input transients and are analyzed with the RCS water solid. Only one PORV is required to operate during the transients.

Generic transient analyses have been used previously to determine LTOP setpoints which maintain acceptable pressure-temperature combinations on the Appendix G heatup and cooldown curves. The plant specific analysis allows actual plant characteristics to be modelled rather than the use of generic assumptions. The generic assumptions have excessive conservatism to allow a wide range of application. These generic conservative assumptions become operationally burdensome when PORV lift setpoints are too low to allow normal RCS operation without opening a PORV. A plant specific North Anna two loop RETRAN02/MOD03[15] model was developed to analyze possible setpoints. (Information supporting the use of the RETRAN Code version RETRAN02/MOD03 is presented in Attachment 3 to Reference 2.) The analysis revealed that the mass addition transient produces the most limiting results. The following sections describe the North Anna model development and the analysis to determine new PORV setpoints.

Mass Addition Transient

The inadvertent startup of a single charging pump was selected as the design basis mass addition transient based on previous UFSAR work (Reference 12, Section 5.2.2.2). The LTOP setpoints were determined such that a pressure overshoot allowance exists to prevent the composite, modified heatup and cooldown curve from being exceeded assuming an inadvertent charging pump startup during water solid operation. This overshoot allowance is required because of the valve opening characteristic associated with the air operated relief valves used on the pressurizer at North Anna[13],[14].

Inadvertent operation of a single charging pump was modeled assuming initial conditions as listed in Table 5. The initial RCS temperature, pressure, and PORV setpoint were varied to observe the effects of changes in these parameters. A range of RCS temperatures between 100 and 325°F were examined, as well as a range of initial pressures. The results revealed a gradually decreasing PORV setpoint pressure overshoot with increasing initial RCS temperature and PORV setpoint. The peak RCS pressure was found to be relatively insensitive to the initial RCS pressure.

Heat Addition Transient

The heat addition transient assumes the Technical Specification limit of a 50°F temperature difference between the steam generators and the RCS. A reactor coolant pump startup in one loop is also assumed to maximize the heat transfer. This scenario has been determined to be the design basis heat addition transient for LTOP setpoint determination relative to the composite, modified heatup and cooldown curve (Reference 12, Section 5.2.2.2).

The heat addition transient was modeled assuming the initial conditions listed in Table 6. The secondary to primary heat transfer modelling included a very conservative evaluation of the local secondary side convection heat transfer coefficient and an assumed constant bulk secondary side temperature (no credit taken for decreasing temperature due to secondary to primary heat transfer). The pump startup flow characteristic was also modelled in a conservative fashion. The analysis revealed that the results of the heat addition transient are easily bounded by those of the mass addition transient.

Revised LTOP Setpoints

Using the PORV setpoint overshoot results from the analysis described above performed with the North Anna plant-specific RETRAN[15] model, revised LTOP setpoints were determined. The revised setpoints were established using the composite, modified heatup and cooldown curves in Tables 3 and 4 which take credit for the assumed cooldown and heatup rates presented in Tables 1 and 2, and exclude measurement uncertainty.

It is reasoned that measurement uncertainties may be excluded from consideration in the development of LTOP setpoints on the basis that these uncertainties are insignificant when compared to the margin terms included in the ASME Section III Appendix G methods. Specifically, the pressure stress is multiplied by a factor of two, resulting in conservative stress intensity values. (As an example, a pressure/temperature limit which shows an allowable internal pressure of 400 psi is actually based upon a stress associated with an internal pressure of 800 psi.) In addition, the use of a lower bound allowable stress intensity, K_{IR}, which is shifted in accordance with Regulatory Guide 1.99 Revision 2 methods (i.e., 2σ margin on mean predicted shift) ensures a conservative measure of allowable stress intensity as a function of temperature in the heatup and cooldown curve calculations.

Instrumentation uncertainties have been excluded from consideration in other utility submittals on the basis of their insignificance relative to the conservatisms of stress intensity factors[19],[20].

Temperature measurement uncertainty was considered in the development of the minimum LTOP enabling temperatures. Automatic low temperature overpressurization protection is required whenever any RCS cold leg temperature is less than 270°F. (321°F for Unit 2). These temperatures are the RT_{NDT}+ΔT+90°F+instrument uncertainty. For Unit 1, the 12 EFPY RT_{NDT} temperature is 145°F for 1/4T and 122°F for 3/4T[9]. For Unit 2, the 17 EFPY RT_{NDT} temperature is 196°F for 1/4T and 172°F for 3/4T[10]. The ΔT is the maximum temperature difference between the water and metal (i.e., 15°F at the 1/4T; 32°F at the 3/4T). The instrument uncertainty added was 20°F. The 90°F addition is considered to be a reasonable range to require the automatic low temperature overpressurization protection. This is sufficient for automatic protection during startup and shutdown. Above 270°F (321°F for Unit 2), administrative control is adequate protection because of Appendix G fracture criteria. The analysis has an increased margin at higher

temperatures. In addition, operation of the RCS above 270°F (321°F for Unit 2) decreases the effects of the two design basis transients.

The concept of requiring automatic LTOP at the lower end and administrative control at the upper end of the Appendix G pressure/temperature limit curve is further discussed in NRC Generic Letter 88-11[16]. The Generic Letter states that due to the impact of implementation of Revision 2 of Regulatory Guide 1.99, Standard Review Plan Section 5.2.2 and Branch Technical Position RSB 5-2 will be revised to define the temperature where automatic protection is required to be enabled.

The PORV setpoints were optimized to values which limit the reactor vessel peak pressure during postulated overpressure transients to values less than those represented by the composite, modified heatup and cooldown curve, and which are not operationally restrictive because of too much safety margin. The proposed North Anna Unit 1 setpoints are ≤ 450 psig when the RCS temperature is ≤ 270°F, and ≤ 390 psig when the RCS temperature is ≤ 150°F. The proposed North Anna Unit 2 setpoints are ≤ 510 psig when the RCS temperature is ≤ 321°F, and ≤ 360 psig when the RCS temperature is ≤ 210°F.

Table 7 presents the current and proposed LTOP PORV setpoints. The setpoints are not dramatically changed, reflecting the net impact of excluding instrumentation uncertainties in the heatup and cooldown curves and the extrapolation of RT_{NDT} to a higher applicable burnup.

RT_{PTS} Evaluation

The results of the Capsule U analyses for both North Anna Unit 1[4] and Unit 2[5] reveal an excellent comparison between the experimentally determined fast neutron fluence in the surveillance capsule and the calculated fluence at the capsule center. These results confirm the analytical model used to predict vessel fluence to the end of the current operating license.

Reference 17 presented estimated fluences for North Anna Unit 1 and Unit 2 as a function of burnup. These values were utilized to estimate the end-of-license RT_{PTS} for the controlling materials in the North Anna 1 and 2 reactor vessel beltlines. Utilizing the method of Reg Guide 1.99, Revision 2 as specified in 10 CFR 50.61[18], the following results were obtained:

North Anna Unit 1 - Limiting Forging (Lower Shell) - (Estimated EOL Fluence = 6.79×10^{19} n/cm^2 - Inner Surface)
EOL RT_{PTS} = 240°F (Screening Criterion = 270°F)

North Anna Unit 1 - Limiting Weld (Circumferential Weld) -
(Estimated EOL Fluence = 6.79×10^{19} n/cm^2 - Inner Surface)
EOL RT_{PTS} = 148°F (Screening Criterion = 300°F)

North Anna Unit 2 - Limiting Forging (Lower Shell) - (Estimated EOL
Fluence = 6.96×10^{19} n/cm^2 - Inner Surface)
EOL RT_{PTS} = 230°F (Screening Criterion = 270°F)

North Anna Unit 2 - Limiting Weld (Circumferential Weld) -
(Estimated EOL Fluence = 6.96×10^{19} n/cm^2 - Inner Surface)
EOL RT_{PTS} = 60°F (Screening Criterion = 300°F)

The extrapolated fluence at the above locations at 12 EFPY (Unit 1) is estimated to be 1.64×10^{19} n/cm^2 (4). This is well within the fluences demonstrated above to not result in a PTS concern for Unit 1. The extrapolated fluence at the above locations at 17 EFPY (Unit 2) is estimated to be 2.35×10^{19} n/cm^2 (5). Again, this is well within the fluences demonstrated to not result in a PTS concern for Unit 2. Because of the excellent comparison between experimentally determined and calculated fluences, and because there are no expected changes in operating conditions that would significantly impact vessel fluence estimates, the EOL RT_{PTS} values presented above (based on previous fluence estimates) are still considered valid.

Measured ΔRT_{NDT} values from the capsule analyses and predicted values as calculated by Regulatory Guide 1.99 Revision 2 compare well, with the R.G. 1.99 Rev. 2 values generally being more limiting. This reinforces the conclusion that there is no PTS concern for North Anna Units 1 and 2 for burnups up to the respective applicable burnups for each unit.

Conclusions

The heatup and cooldown curves required by Appendix G of 10 CFR 50 have been extrapolated to 12 EFPY and 17 EFPY for North Anna Units 1 and 2, respectively, by including the effects of the incremental radiation exposure on the reactor vessel beltline region. The results are referenced to the analyses of the North Anna Units 1 and 2 Capsule U results. The revised Appendix G curves were prepared using standard B&W and Westinghouse methodologies methodology including Regulatory Guide 1.99 Rev. 2. PORV setpoints were developed to provide bounding heatup and cooldown curve protection for the worst case mass and heat addition low temperature overpressure transients.

The next Unit 1 reactor vessel surveillance capsule (Capsule X) is scheduled to be removed after the tenth fuel cycle (10 EFPY) which allows sufficient time for analysis prior to exceeding 12 EFPY. The next Unit 2 reactor vessel surveillance capsule (Capsule W) is scheduled to be removed after the thirteenth fuel cycle (15 EFPY) which allows sufficient time for analysis prior to exceeding 17 EFPY.

The heatup and cooldown curves prepared by B&W and Westinghouse were determined in a conventional manner according to Section III of the ASME code as required by 10 CFR 50 Appendix G. Both steady-state and transient thermal conditions were considered in order to bound the possible combinations of pressure (i.e. membrane) and thermal stresses.

The new North Anna Unit 1 low temperature overpressure protection system PORV lift settings should be less than or equal to 450 psig whenever any RCS cold leg temperature is less than or equal to 270°F, and less than or equal to 300 psig whenever any RCS cold leg temperature is less than 150°F.

The new North Anna Unit 2 low temperature overpressure protection system PORV lift settings should be less than or equal to 510 psig whenever any RCS cold leg temperature is less than or equal to 321°F, and less than or equal to 360 psig whenever any RCS cold leg temperature is less than 210°F.

PTS evaluations were made for the limiting beltline locations. It was demonstrated that (a) predicted end-of-license fluences do not result in RT_{PTS} values in excess of the screening criteria when calculated using the methodology of Regulatory Guide 1.99, Revision 2; (b) there is an excellent comparison between experimentally determined and calculated vessel fluences; and (c) the extrapolated fluences at the burnup limit to which the revised heatup and cooldown curves are applicable for each unit are significantly less than the extrapolated end-of-license fluences (which have been demonstrated to not result in a violation of PTS screening criteria). On this basis it may be concluded that there is neither a significant change in predicted RT_{PTS} values; nor is there a PTS concern for either unit up to the burnup limit to which the revised heatup and cooldown curves are valid.

List of Tables

Table 1: COOLDOWN RATES ASSUMED FOR VARIOUS TEMPERATURE RANGES

Temperature Range °F	Cooldown Rate °F/hr.
T>200°F	100°F/hr.
180°F<T≤200°F	60°F/hr.
150°F<T≤180°F	40°F/hr.
120°F<T≤150°F	20°F/hr.
T≤120°F	0°F/hr.

Table 2: HEATUP RATES ASSUMED FOR VARIOUS TEMPERATURE RANGES

Temperature Range °F	Heatup Rate °F/hr.
T≤150°F	20°F/hr.
170°F>T>150°F	40°F/hr.
T≥170°F	60°F/hr.

Table 3

COMPOSITE, MODIFIED HEATUP AND COOLDOWN CURVE - NORTH ANNA UNIT 1

(Data for LTOPS Setpoint Development)

Indicated Temperature (°F)	Composite Pressure Limit (psig)
75	550
80	554
85	559
90	565
95	572
100	579
105	587
110	595
115	604
120	614
125	597
130	610
135	624
140	639
145	655
150	673
155	659
160	681
165	705
170	715
175	737
180	760
185	782
190	805
195	829
200	855
205	883
210	914
215	946
220	981
225	1019
230	1059

Table 3 - (continued)

Indicated Temperature (°F)	Composite Pressure Limit (psig).
235	1103
240	1149
245	1199
250	1253
255	1311
260	1373
265	1440
270	1512
275	1590
280	1673
285	1762
290	1858
295	1962
300	2073
305	2192
310	2320
315	2458
320	2605

Table 4

COMPOSITE, MODIFIED HEATUP AND COOLDOWN CURVE - NORTH ANNA UNIT 2

(Data for LTOPS Setpoint Development)

Indicated Temperature (°F)	Composite Pressure Limit (psig)
85	513
90	514
95	518
100	522
105	527
110	532
115	538
120	545
125	525
130	531
135	537
140	545
145	552
150	561
155	534
160	544
165	555
170	539
175	550
180	561
185	574
190	586
195	601
200	616
205	596
210	622
215	649
220	679
225	711
230	737

Table 4 - (continued)

Indicated Temperature (°F)	Composite Pressure Limit (psig)
235	763
240	791
245	820
250	852
255	886
260	923
265	963
270	1005
275	1051
280	1099
285	1152
290	1208
295	1268
300	1333
305	1402
310	1477
315	1556
320	1642
325	1733
330	1830
335	1935
340	2046
345	2166
350	2283
355	2402

Table 5: INITIAL CONDITIONS FOR THE MASS ADDITIONS TRANSIENT

Reactor Coolant Temperature (°F)	100, 150, 200, 250, 300, 325
Reactor Coolant Pressure (psig)	200, 250, 300, 340, 380, 400
Maximum Charging Pump Flowrate (design head/flow curve)	705 gpm
Pressurizer Steam Volume	0 ft^3
Pressurizer Water Volume	1400 ft^3
Reactor Coolant System Flow	10%
PORV OPEN Setpoint	Variable
PORV CLOSED Setpoint	OPEN-15 psi

Table 6: INITIAL CONDITIONS FOR THE HEAT ADDITIONS TRANSIENT

Reactor Coolant Temperature	100°F
Reactor Coolant Pressure	280, 340(psig)
RCS/SG ΔT	50°F
Pressurizer Steam Volume	0 ft^3
Pressurizer Water Volume	1400 ft^3
RCP Speeds	
In Affected Loop, startup	10% - 100%
In Unaffected Loop, coastdown	10% - 0%
PORV OPEN Setpoint	Variable
PORV CLOSED Setpoint	OPEN-15 psi

Table 7: CURRENT AND PROPOSED LTOPS PORV SETPOINTS

NORTH ANNA - UNIT 1	
Current:	≤450 psig for Cold Leg T≤261°F ≤390 psig for Cold Leg T≤150°F
Proposed:	≤450 psig for Cold Leg T≤270°F ≤390 psig for Cold Leg T≤150°F
NORTH ANNA - UNIT 2	
Current:	≤520 psig for Cold Leg T≤340°F ≤375 psig for Cold Leg T≤190°F
Proposed:	≤510 psig for Cold Leg T≤321°F ≤360 psig for Cold Leg T≤210°F

BAW-2146

BAW-2146

October 1991

NORTH ANNA UNIT 1 PRESSURE-TEMPERATURE LIMITS FOR 12 EFPY AND NORTH ANNA UNIT 2 PRESSURE-TEMPERATURE LIMITS FOR 12 AND 15 EFPY

VIRGINIA ELECTRIC AND POWER COMPANY

by

A. D. Nana
M. J. Devan

B&W Document No. 77-2146-00
(See Section 3 for Document Signatures)

B&W NUCLEAR SERVICE COMPANY
Engineering and Plant Services Division
P. O. Box 10935
Lynchburg, Virginia 24506-0935

1. INTRODUCTION

This report presents the North Anna Unit 1 and Unit 2 reactor vessel beltline region pressure-temperature operating limits applicable to 12 EFPY. Also included in this report are the North Anna Unit 2 reactor vessel beltline region pressure-temperature operating limits for 15 EFPY. The data used to develop the 12 EFPY limitations for North Anna Unit 1 are based on the analysis of North Anna Unit 1 Reactor Vessel Surveillance Capsule U as reported in WCAP-11777,[1] and the data used to develop the 12 and 15 EFPY limitations for North Anna Unit 2 are based on the analysis of North Anna Unit 2 Reactor Vessel Surveillance Capsule U reported in WCAP-12497.[2] The report contains data which support the development of the pressure-temperature limits for normal operation, both heatup and cooldown, inservice leak and hydrostatic tests and reactor core operation applicable to 12 EFPY for North Anna Unit 1 and 2 and 15 EFPY for North Anna Unit 2. These limits are adequate for current operations and are justified by the data obtained from the surveillance capsules as presented in WCAP-11777 for North Anna Unit 1 and WCAP-12497 for North Anna Unit 2.

2. DETERMINATION OF REACTOR VESSEL BELTLINE REGION PRESSURE-TEMPERATURE LIMITS

The pressure-temperature limits of the reactor vessel beltline region of North Anna Unit 1 and Unit 2 are established in accordance with the requirements of 10CFR50, Appendix G.[3] The objective of these limits is to prevent nonductile failure during any normal operating condition, including anticipated operational occurrences and system hydrostatic tests. The loading conditions of interest include the following:

1. Normal operations, including heatup and cooldown.
2. Inservice leak and hydrostatic tests.
3. Reactor core operation.

The beltline regions of the North Anna Unit 1 and Unit 2 reactor vessels have been analyzed in accordance with 10CFR50, Appendix G. For the service period for which the limit curves are established, the maximum allowable pressure as a function of fluid temperature is obtained through a point-by-point comparison of the limits imposed by the beltline region. The maximum allowable pressure is taken to be the lowest of the allowable pressures at the one-quarter T and three-quarter T vessel wall locations (T = wall thickness measured from the inside surface) and due to steady state conditions.

<u>North Anna Unit 1</u>

The limit curves for North Anna Unit 1 are based on the predicted values of the adjusted reference temperatures of all the beltline region materials at the end of the twelfth EFPY. The twelfth EFPY was selected for this analysis to extend the applicability of the Unit 1 curves to provide sufficient time to perform the 10 EFPY surveillance capsule analysis and evaluation.

Unirradiated Charpy impact properties were determined for surveillance beltline region materials in accordance with 10CFR50, Appendixes G and H.[8] For beltline region materials for which the measured properties are not available, unirradiated Charpy impact properties and residual element compositions, as originally determined, are listed in Table 2-1. The adjusted reference temperatures are calculated by adding the predicted radiation-induced shifts in RT_{NDT} and the initial RT_{NDT}. The radiation-induced RT_{NDT} values were predicted as a function of the material's copper and nickel content and exposure to neutron fluence in accordance with the guidelines presented in Regulatory Guide 1.99, Revision 2.[4]

The neutron fluence and adjusted RT_{NDT} values of the beltline region materials at the end of the twelfth full-power year are listed in Table 2-1. The adjusted RT_{NDT} values are given for the one-quarter T and three-quarter T vessel wall locations.

Figures 2-1 through 2-3 show reactor vessel's pressure-temperature limit curves for normal heatup and inservice leak and hydrostatic tests. These figures also show the core criticality limits as required by 10CFR50, Appendix G. Figure 2-4 shows the vessel's pressure-temperature limit curves for normal cooldown. The above pressure-temperature limit curves are applicable through 12 EFPY. Protection against nonductile failure for the North Anna Unit 1 reactor beltline region is ensured by maintaining the reactor vessel downcomer pressure below the upper limits of the pressure-temperature limit curves. The acceptable pressure and temperature combination for reactor vessel operation are below and to the right of the limit curves. The reactor is not permitted to go critical until the pressure-temperature combinations are to the right of the criticality limit curve. To establish the pressure-temperature limits for protection against nonductile failure, the limits presented in Figures 2-1 through 2-4 must be adjusted by the pressure differential between the point of system pressure measurement and the pressure on the reactor vessel controlling the limit curves as well as accounting for possible instrument errors.

In addition to the North Anna Unit 1 revised heatup and cooldown curves, the original Babcock and Wilcox Report BAW-2146 documented revised curves for North Anna Unit 2 applicable to burnups of 12 and 15 EFPY. Virginia Power has removed the pages from BAW-2146 which document these curves as they are not being submitted for review and approval at this time.

Table 2-1. Data for Protection of Pressure-Temperature Limit Curves for North Anna Unit 1 Reactor Vessel -- Applicable Through 12 EFPY

Material Heat No.	Identification Type	Beltline Region Location	Material Chemical Composition, w/e[a] Copper	Material Chemical Composition, w/e[a] Nickel	Fluence[b] T/4 Wall location n/cm²	Fluence[b] 3/4T Wall location n/cm²	Chemistry Factor	Initial RT_{NDT}, F[c]	Radiation-Induced RT_{NDT} at End of 12 EFPY, F[d] T/4[e]	Radiation-Induced RT_{NDT} at End of 12 EFPY, F[d] 3/4T[e]	Margin T/4	Margin 3/4T	Adjusted RT_{NDT}, End of 12 EFPY, F[d] T/4	Adjusted RT_{NDT}, End of 12 EFPY, F[d] 3/4T
Forging 05	SA508,Cl.2	Nozzle Shell Forging	0.16	0.74	6.94E+17	2.76E+17	122	6	42	25	34	26	62	57
Forging 04	SA508,Cl.2	Interm. Shell Forging	0.12	[illegible]	1.04E+19	4.12E+18	88	17	87	65	34	34	138	116
Forging 03	SA508,Cl.2	Lower Shell Forging	[illegible]	0.80	1.04E+19	4.12E+18	115	38	118	87	34	34	188	159
Weld 05A	Subm. Arc	Noz. to Int. Cir. Weld (OD 94%)	[illegible]	0.10	6.94E+17	2.76E+17	138	0	48	79	62	49	119	78
Weld 05B	Subm. Arc	Noz. to Int. Cir. Weld (ID 6%)	[illegible]	0.10	N/A	N/A	58	0	N/A	N/A	N/A	N/A	N/A	N/A
Weld 04	Subm. Arc	Int. to Lower Cir. Weld	0.086	0.11	1.04E+19	4.12E+18	52	19	52	39	52	38	123	96
Forging 03	Calculated RT_{NDT} shift based on use of surveillance capsule data per Reg. Guide 1.99, Rev.2		0.15	0.80	1.04E+19	4.12E+18	89	38	90	67	17	17	[145][f]	[122][f]
Weld 04	Calculated RT_{NDT} shift based on use of surveillance capsule data per Reg. Guide 1.99, Rev. 2		0.086	0.11	1.04E+19	4.12E+18	93	19	94	70	28	[illegible]	141	117

[a] Materials chemical compositions per WCAP-11791, May 1988,[3] and BAW-1911, Revision 1, August 1986.[4]
[b] Fluence data per WCAP-11777, February 1989.[1]
[c] Initial RT_{NDT} values per BAW-1911, Revision 1, August 1986.[4]
[d] Radiation-induced and adjusted RT_{NDT} values calculated per Regulatory Guide 1.99, Revision 2, May 1988.[4]
[e] Reactor vessel thickness - 7.677 inches.[9]
[f] [ ] Controlling values of the adjusted RT_{NDT}.

In addition to the North Anna Unit 1 revised heatup and cooldown curves, the original Babcock and Wilcox Report BAW-2146 documented revised curves for North Anna Unit 2 applicable to burnups of 12 and 15 EFPY. Virginia Power has removed the pages from BAW-2146 which document these curves as they are not being submitted for review and approval at this time.

Figure 2-1. North Anna Unit 1 Reactor Vessel Pressure-Temperature Limit Curves for Normal Operation - Heatup, Applicable for First 12 EFPY Up to 20°F/hr

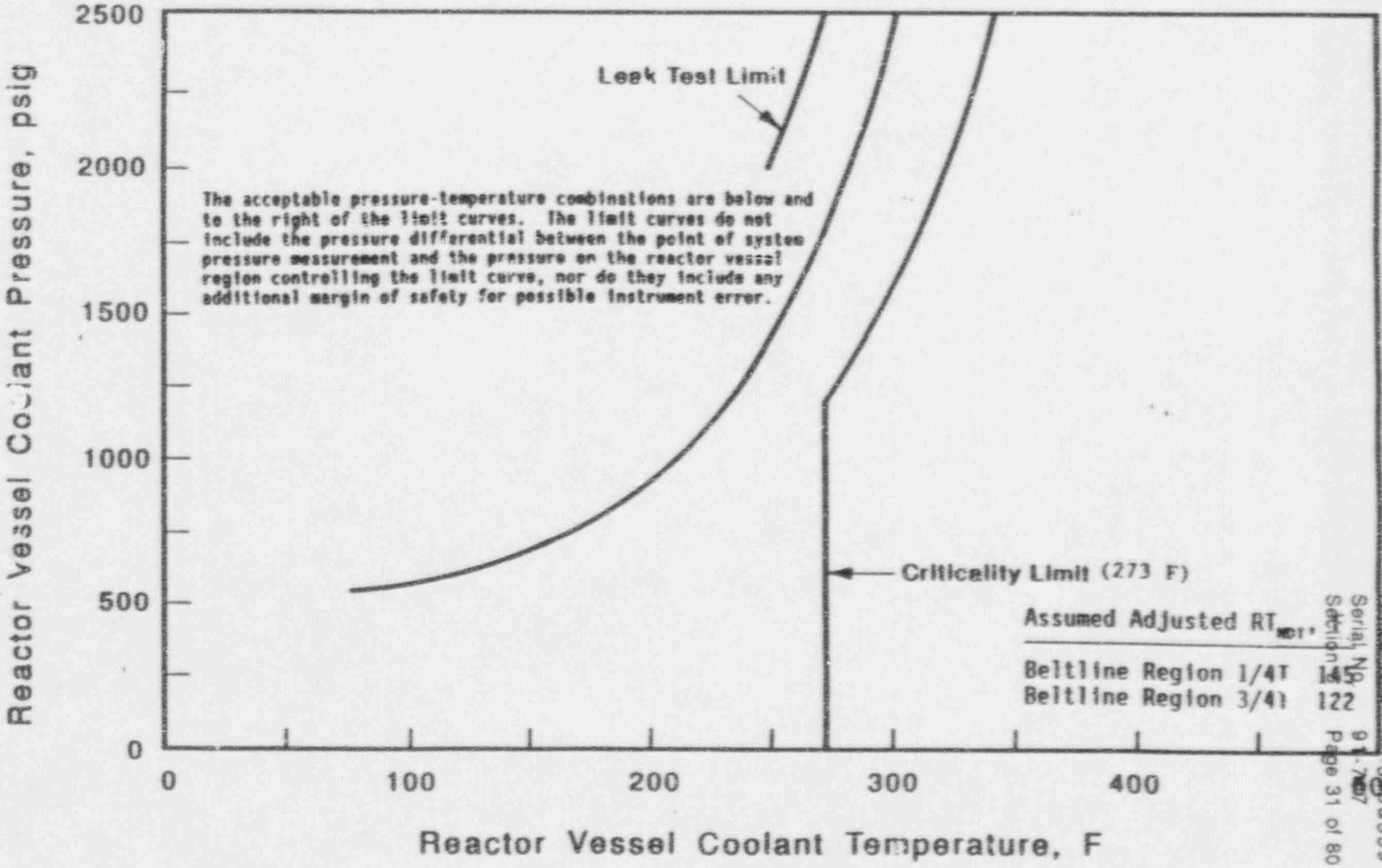
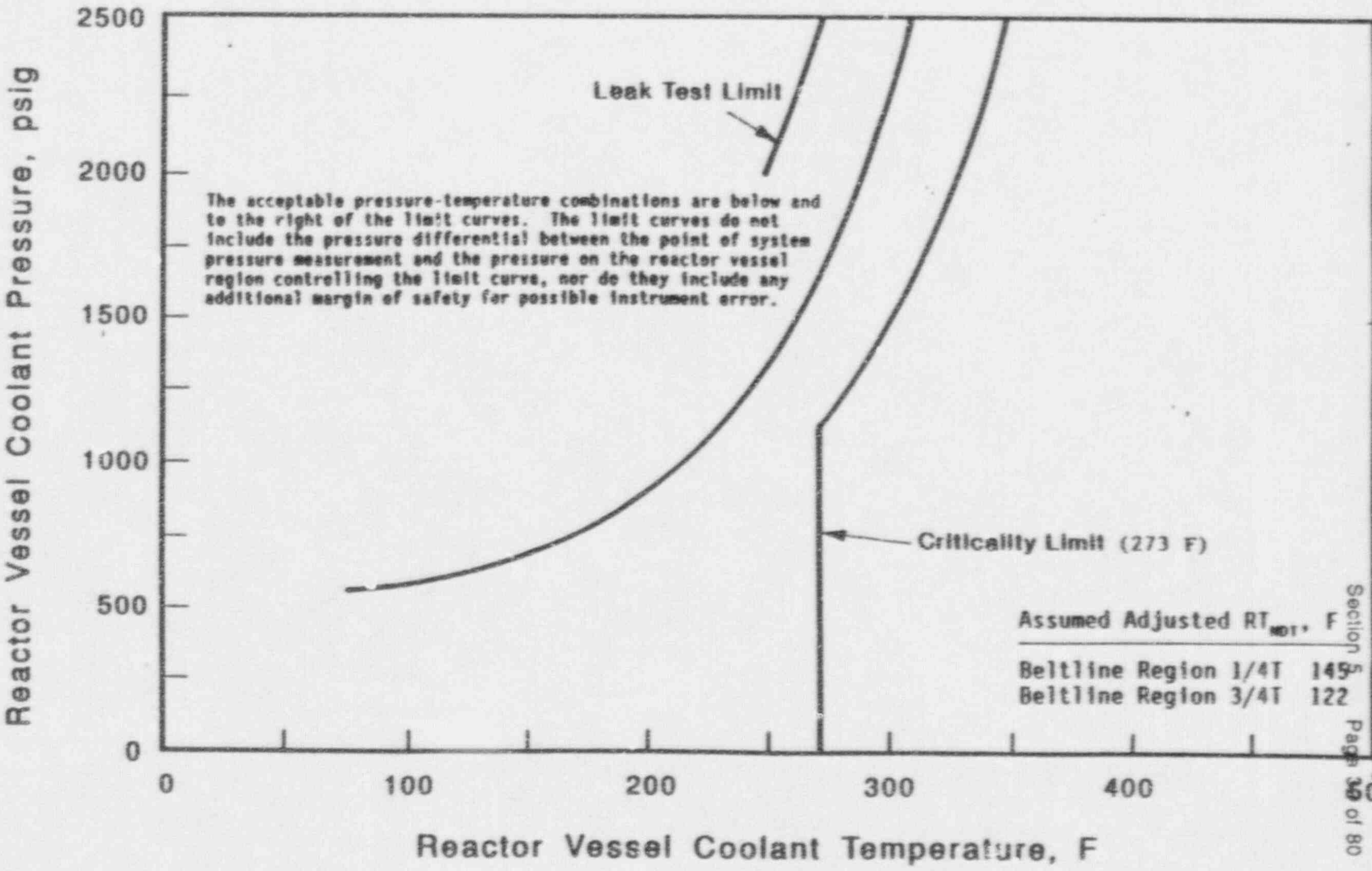


Figure 2-2. North Anna Unit 1 Reactor Vessel Pressure-Temperature Limit Curves for Normal Operation - Heatup, Applicable for First 12 EFPY Up to 40°F/hr



VIRGINIA ELECTRIC AND POWER COMPANY

Figure 2-3. North Anna Unit 1 Reactor Vessel Pressure-Temperature Limit Curves for Normal Operation - Heatup, Applicable for First 12 EFPY Up to 60°F/hr

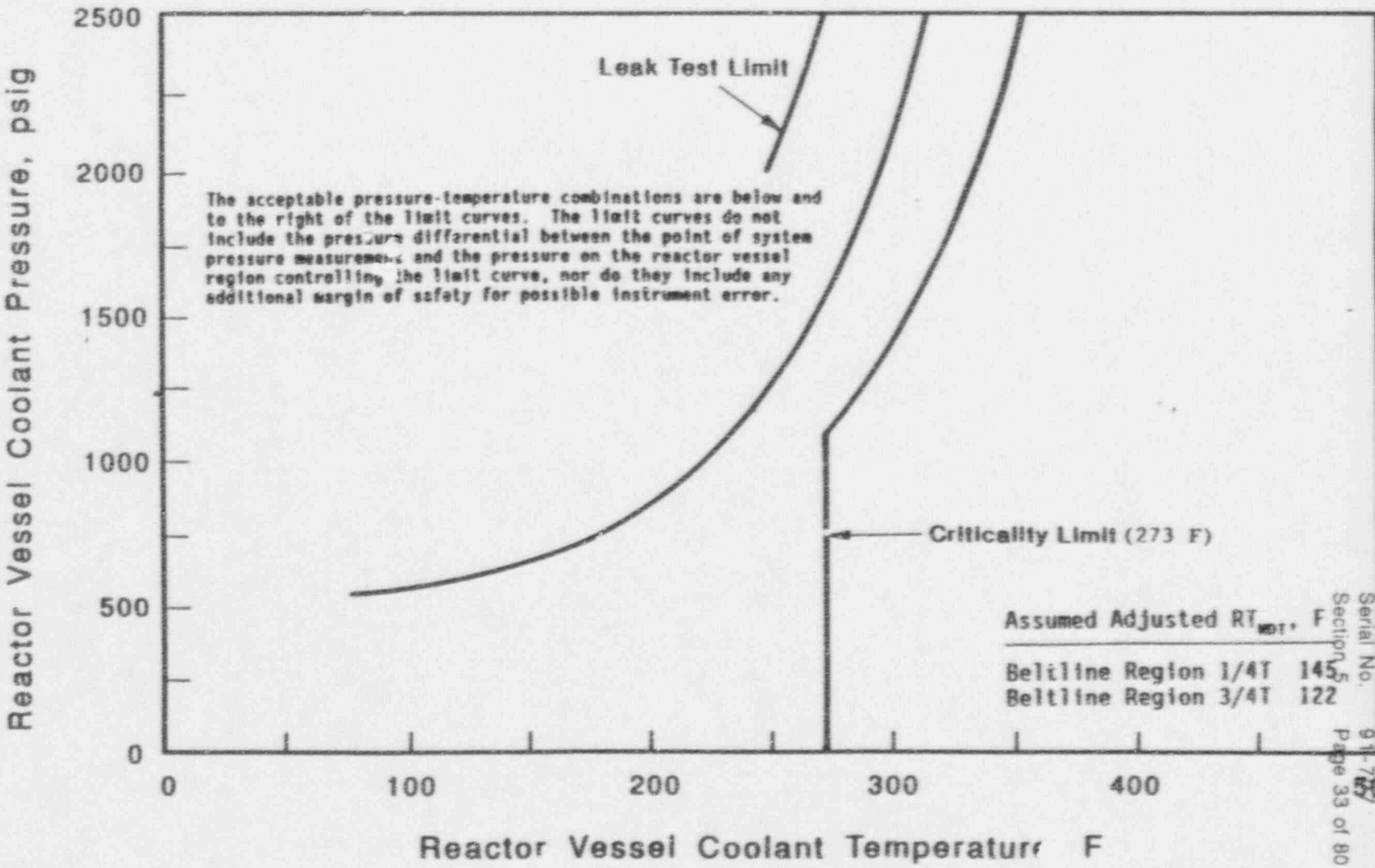
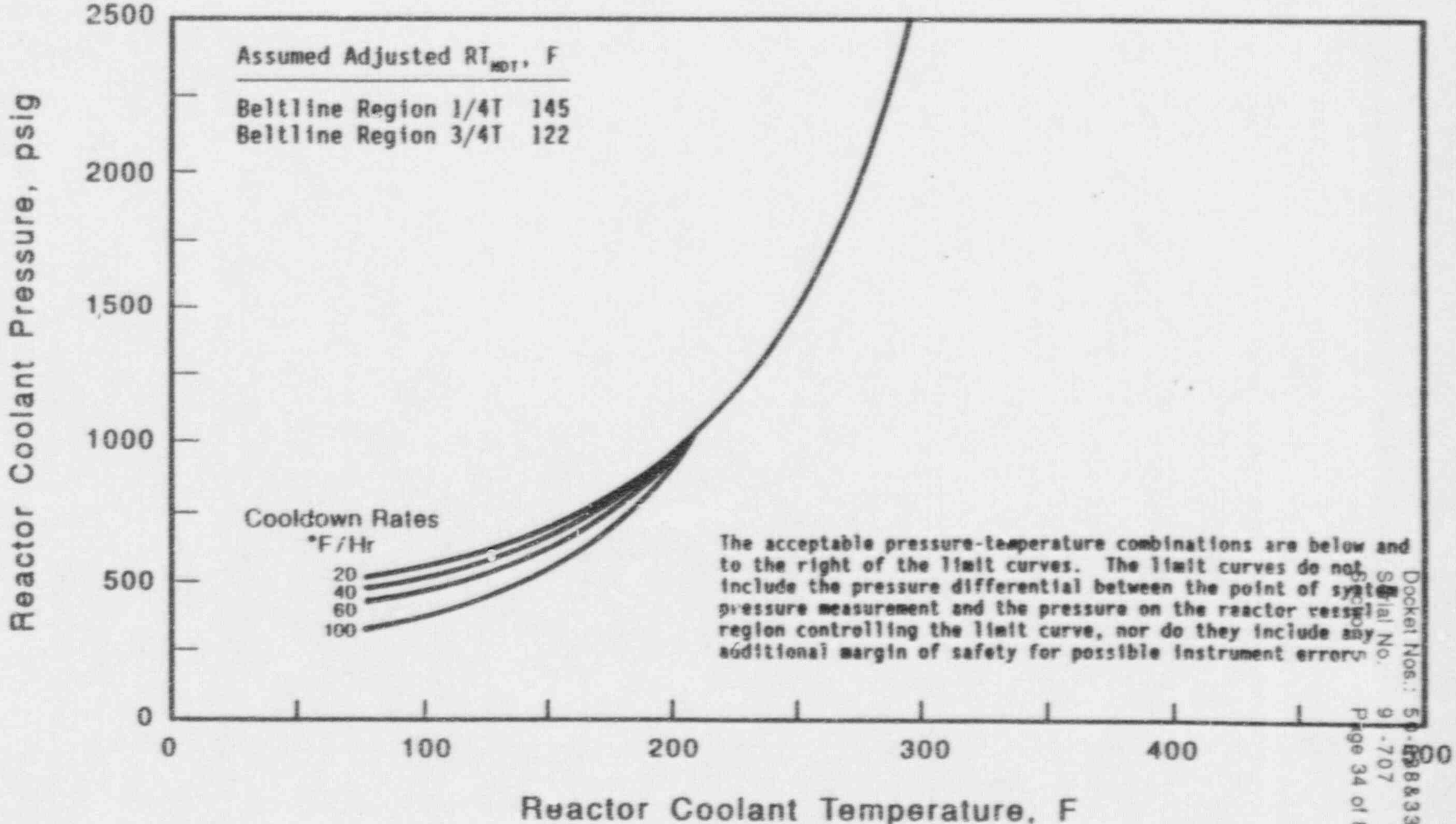


Figure 2-4. North Anna Unit 1 Reactor Vessel Pressure-Temperature Limit Curves for Normal Operation - Cooldown, Applicable for First 12 EFPY



In addition to the North Anna Unit 1 revised heatup and cooldown curves, the original Babcock and Wilcox Report BAW-2146 documented revised curves for North Anna Unit 2 applicable to burnups of 12 and 15 EFPY. Virginia Power has removed the pages from BAW-2146 which document these curves as they are not being submitted for review and approval at this time.

3. CERTIFICATION

The pressure-temperature operating limits for North Anna Units 1 and 2 reactor pressure vessel were calculated using approved procedures and established methods and techniques in accordance with the requirements of 10CFR50, Appendix G.

A D Nana 10/10/91
A. D. Nana Date
Materials and Structural Analysis

M J DeVan 10/10/91
M. J. Devan Date
Materials and Structural Analysis

This report has been reviewed for technical content and accuracy.

L B Gross 10/15/91
L. B. Gross, P.E. (Materials Analysis) Date
Materials and Structural Analysis

for K.K. Yoon 10/15/91
K. K. Yoon, P. E. (Fracture Analysis) Date
Materials and Structural Analysis

Verification of independent review.

K. E. Moore 10/15/91
K. E. Moore, Manager Date
Materials and Structural Analysis

This report has been approved for release.

T. L. Baldwin 10/15/91
T. L. Baldwin Date
Program Manager

4. REFERENCES

1. S. E. Yanichko, et al., Analysis of Capsule U from the Virginia Electric and Power Company North Anna Unit 1 Reactor Vessel Radiation Surveillance Program, WCAP-11777, Westinghouse Electric Corporation, Pittsburgh, Pennsylvania, February 1988.

2. E. Terek, et al., Analysis of Capsule U from the Virginia Electric and Power Company North Anna Unit 2 Reactor Vessel Radiation Surveillance Program, WCAP-12497, Westinghouse Electric Corporation, Pittsburgh, Pennsylvania, January 1990.

3. Code of Federal Regulations, Title 10, Part 50, Domestic Licensing of Production and Utilization Facilities, Appendix G, Fracture Toughness Requirements, November 30, 1986.

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

5. J. C. Schmertz, Analysis of Capsule U from the Virginia Electric and Power Company North Anna Unit 1 Reactor Vessel Radiation Surveillance Program, North Anna Unit 1 Reactor Vessel Heatup and Cooldown Limit Curves for Normal Operation, WCAP-11791, Westinghouse Electric Corporation, Pittsburgh, Pennsylvania, May 1988.

6. A. L. Lowe, Jr., Reactor Pressure Vessel and Surveillance Program Materials Licensing Information for North Anna Units 1 and 2, BAW-1911, Revision 1, Babcock & Wilcox, Lynchburg, Virginia, August 1985.

7. N. K. Ray, and J. M. Chicots, North Anna Unit 2 Reactor Vessel Heatup and Cooldown Limit Curves for Normal Operation (Capsule U), WCAP-12503, Westinghouse Electric Corporation, Pittsburgh, Pennsylvania, March 1990.

8. Code of Federal Regulations, Title 10, Part 50, Domestic Licensing of Production and Utilization Facilities, Appendix H, Reactor Vessel Material Surveillance Program Requirements, November 30, 1936.

9. VEPCO North Anna Power Station Units 1 and 2, Final Safety Analysis Report, Part B, Volume III-A, Paragraph 5.4, Figure 5.4-1, dated January 3, 1973, USNRC Docket Nos. 50-338 and 5-339.

APPENDIX A

North Anna Unit 1 Data Points
for Heatup and Cooldown
Applicable to 12 EFPY

North Anna Beltline Uncorrected P/T Limits
Unit 1
12 EFPY

NOTE: The P/T Limits provided:
a) do not account for location & instrument channel uncertainties

b) do not include Closure Head Limits.

Min P/T Limits									
Shell Fluid Temp	HU 20 F/hr	HU 40 F/hr	HU 60 F/hr	SS T/4	Shell Fluid Temp	CD 20 F/hr	CD 40 F/hr	CD 60 F/hr	CD 100 F/hr
75	550	549	548	552	570	3532	3532	3532	3532
80	554	552	552	558	565	3511	3512	3514	3516
85	559	557	555	565	560	3495	3491	3491	3494
90	565	562	560	572	555	3485	3473	3470	3471
95	572	567	565	579	550	3478	3458	3451	3469
100	579	574	570	587	545	3474	3447	3435	3429
105	587	580	576	596	540	3471	3437	3421	3409
110	595	588	578	605	535	3469	3430	3409	3392
115	604	596	581	615	530	3468	3424	3399	3376
120	614	605	585	626	525	3468	3419	3390	3361
125	625	615	591	637	520	3467	3415	3382	3347
130	636	625	599	650	515	3467	3413	3375	3335
135	648	636	608	663	510	3467	3410	3370	3323
140	662	648	619	677	505	3467	3409	3365	3313
145	676	661	631	692	500	3467	3407	3361	3304
150	691	675	645	709	495	3468	3406	3358	3295
155	707	690	660	727	490	3468	3406	3355	3287
160	725	706	677	746	485	3468	3405	3352	3280
165	743	723	695	766	480	3468	3405	3350	3274
170	764	742	715	788	475	3468	3405	3349	3268
175	785	762	737	812	470	3468	3404	3347	3263
180	809	783	760	837	465	3468	3404	3346	3258
185	834	806	782	865	460	3468	3404	3345	3254
190	861	831	805	894	455	3468	3404	3344	3250
195	890	857	829	926	450	3468	3404	3343	3246
200	921	886	855	960	445	3468	3404	3343	3243
205	954	917	883	997	440	3468	3404	3342	3240
210	990	950	914	1036	435	3468	3404	3342	3237
215	1029	985	946	1078	430	3468	3404	3341	3235
220	1071	1023	981	1126	425	3468	3404	3341	3232
225	1115	1064	1019	1173	420	3469	3404	3341	3230
230	1163	1108	1059	1225	415	3469	3404	3341	3229
235	1215	1156	1103	1282	410	3469	3405	3341	3227
240	1271	1206	1149	1343	405	3469	3405	3341	3226
245	1331	1261	'199	1408	400	3469	3405	3341	3225
250	1395	1320	.253	1478	395	3469	3405	3341	3224
255	1464	1383	1311	1554	390	3469	3405	3341	3223
260	1539	1451	1373	1635	385	3469	3406	3342	3222
265	1619	1524	1440	1723	380	3469	3406	3342	3221
270	1705	1603	1512	1817	375	3469	3406	3342	3221
275	1797	1688	1590	1917	370	3469	3406	3342	3220
280	1896	1779	1673	2026	365	3469	3406	3342	3220
285	2003	1876	1762	2143	360	3469	3406	3342	3219
290	2117	1981	1858	2268	355	3469	3406	3342	3219
295	2241	2094	1962	2403	350	3469	3406	3342	3219
300	2373	2215	2073	2548	345	3469	3406	3342	3219
305	2515	2345	2192	2704	340	3469	3406	3342	3218
310	2668	2485	2320	2872	335	3469	3406	3342	3218
315	2832	2635	2458	3052	330	3470	3406	3342	3218
320	3008	2796	2605	3246	325	3455	3406	3342	3217
325	3198	2970	2764	3455	320	3246	3246	3246	3217
330	3402	3156	2934	3532	315	3052	3052	3052	3052
335	3532	3356	3117	3532	310	2872	2872	2872	2872
340	3532	3532	3314	3532	305	2704	2704	2704	2704
345	3532	3532	3519	3532	300	2548	2548	2548	2548
350	3532	3532	3532	3532	295	2403	2403	2403	2403
355	3532	3532	3532	3532	290	2268	2268	2268	2268
360	3532	3532	3532	3532	285	2143	2143	2143	2143
365	3532	3532	3532	3532	280	2026	2026	2026	2026
370	3532	3532	3532	3532	275	1917	1917	1917	1918
375	3532	3532	3532	3532	270	1817	1817	1817	1817
380	3532	3532	3532	3532	265	1723	1723	1723	1723
385	3532	3532	3532	3532	260	1635	1635	1635	1635
390	3532	3532	3532	3532	255	1554	1554	1554	1554

395	3532	3532	3532	3532
400	3532	3532	3532	3532
405	3532	3532	3532	3532
410	3532	3532	3532	3532
415	3532	3532	3532	3532
420	3532	3532	3532	3532
425	3532	3532	3532	3532
430	3532	3532	3532	3532
435	3532	3532	3532	3532
440	3532	3532	3532	3532
445	3532	3532	3532	3532
450	3532	3532	3532	3532
455	3532	3532	3532	3532
460	3532	3532	3532	3532
465	3532	3532	3532	3532
470	3532	3532	3532	3532
475	3532	3532	3532	3532
480	3532	3532	3532	3532
485	3532	3532	3532	3532
490	3532	3532	3532	3532
495	3532	3532	3532	3532
500	3532	3532	3532	3532
505	3532	3532	3532	3532
510	3532	3532	3532	3532
515	3532	3532	3532	3532
520	3532	3532	3532	3532
525	3532	3532	3532	3532
530	3532	3532	3532	3532
535	3532	3532	3532	3532
540	3532	3532	3532	3532
545	3532	3532	3532	3532
550	3532	3532	3532	3532
555	3532	3532	3532	3532
560	3532	3532	3532	3532
565	3532	3532	3532	3532

250	1478	1478	1478	1478
245	1408	1408	1408	1408
240	1343	1343	1343	1343
235	1282	1282	1282	1282
230	1225	1225	1225	1225
225	1173	1173	1173	1173
220	1124	1124	1126	1124
215	1074	1075	1078	1078
210	1028	1025	1027	1036
205	985	978	976	987
200	946	935	928	930
195	908	894	884	877
190	874	857	843	828
185	842	822	805	782
180	812	789	769	739
175	785	759	736	700
170	759	731	705	663
165	735	705	677	629
160	713	681	651	598
155	692	659	627	568
150	673	638	604	541
145	655	619	583	516
140	639	601	564	492
135	624	584	546	471
130	610	569	529	451
125	597	555	514	433
120	584	542	499	416
115	573	530	486	400
110	562	519	474	386
105	553	508	463	373
100	544	499	453	360
95	535	490	444	349
90	527	482	435	339
85	520	474	427	330
80	513	467	419	321
75	507	461	412	313

Leak Test Data
(12 EFPY)

Pressure (psig)	Temperature (F)
1971	250.0
2297	265.0
2488	272.5

In addition to the North Anna Unit 1 revised heatup and cooldown curves, the original Babcock and Wilcox Report BAW-2146 documented revised curves for North Anna Unit 2 applicable to burnups of 12 and 15 EFPY. Virginia Power has removed the pages from BAW-2146 which document these curves as they are not being submitted for review and approval at this time.

WCAP-12503

WCAP-12503

NORTH ANNA UNIT 2 REACTOR VESSEL HEATUP AND COOLDOWN LIMIT CURVES FOR NORMAL OPERATION (CAPSULE U)

N. K. Ray
J. M. Chicots

Work Performed for Virginia Power Company

March 1990

Approved by: T. A. Meyer

T. A. Meyer, Manager
Structural Materials and Reliability Technology

Work Performed Under Shop Order VJGP-139

WESTINGHOUSE ELECTRIC CORPORATION
Nuclear and Advanced Technology Division
P.O. Box 2728
Pittsburgh, Pennsylvania 15230-2728

TABLE OF CONTENTS

LIST OF TABLES

LIST OF FIGURES

HEATUP AND COOLDOWN LIMIT CURVES FOR NORMAL OPERATION

1.0 INTRODUCTION

Heatup and cooldown limit curves are calculated using the most limiting value of RT_{NDT} (reference nil-ductility temperature) for the reactor vessel. The most limiting RT_{NDT} of the material in the core region of the reactor vessel is determined by using the preservice reactor vessel material fracture toughness properties and estimating the radiation-induced ΔRT_{NDT}. RT_{NDT} is designated as the higher of either the drop weight nil-ductility transition temperature (NDTT) or the temperature at which the material exhibits at least 50 ft-lb of impact energy and 35-mil lateral expansion (normal to the major working direction) minus 60°F.

RT_{NDT} increases as the material is exposed to fast-neutron radiation. Therefore, to find the most limiting RT_{NDT} at any time period in the reactor's life, ΔRT_{NDT} due to the radiation exposure associated with that time period must be added to the original unirradiated RT_{NDT}. The extent of the shift in RT_{NDT} is enhanced by certain chemical elements (such as copper and nickel) present in reactor vessel steels. The Nuclear Regulatory Commission (NRC) has published a method for predicting radiation embrittlement in Regulatory Guide 1.99 Rev. 2 (Radiation Embrittlement of Reactor Vessel Materials)[1]. The fluence values used in this report are from Surveillance Capsule U report [2].

2.0 FRACTURE TOUGHNESS PROPERTIES

The fracture-toughness properties of the ferritic material in the reactor coolant pressure boundary are determined in accordance with the NRC Regulatory Standard Review Plan[3]. The pertinent chemical and mechanical properties of the beltline region plate and weld materials of the North Anna Unit 2 reactor vessel are given in table 1.

The chemistry factors and "margin" (M) terms that are also shown in table 1 were determined in accordance with the Regulatory Guide 1.99 Revision 2. Chemistry factor and margin values in table 1 that were based upon credible surveillance measurements are also given for the beltline region materials where this data was available. Table 2 gives further information on the determination of the chemistry factors from this data in accordance with the applicable procedure from Revision 2 to Regulatory Guide 1.99. For those reactor vessel materials where credible surveillance data exists, the respective chemistry factor and margin terms are lower than the values based on material chemistry measurements. However, credible surveillance data is not available for lower shell plate which is the limiting material.

3.0 CRITERIA FOR ALLOWABLE PRESSURE-TEMPERATURE RELATIONSHIPS

The ASME approach for calculating the allowable limit curves for various heatup and cooldown rates specifies that the total stress intensity factor, K_I, for the combined thermal and pressure stresses at any time during heatup or cooldown cannot be greater than the reference stress intensity factor, K_{IR}, for the metal temperature at that time. K_{IR} is obtained from the reference fracture toughness curve, defined in Appendix G to the ASME Code[4]. The K_{IR} curve is given by the following equation:

$$K_{IR} = 26.78 + 1.223 \exp [0.0145 (T-RT_{NDT} + 160)] \quad (1)$$

where

K_{IR} = reference stress intensity factor as a function of the metal temperature T and the metal reference nil-ductility temperature RT_{NDT}

Therefore, the governing equation for the heatup-cooldown analysis is defined in Appendix G of the ASME Code[4] as follows:

$$C\, K_{IM} + K_{IT} \leq K_{IR} \quad (2)$$

where

K_{IM} = stress intensity factor caused by membrane (pressure) stress

K_{IT} = stress intensity factor caused by the thermal gradients

K_{IR} = function of temperature relative to the RT_{NDT} of the material

C = 2.0 for Level A and Level B service limits

C = 1.5 for hydrostatic and leak test conditions during which the reactor core is not critical

At any time during the heatup or cooldown transient, K_{IR} is determined by the metal temperature at the tip of the postulated flaw, the appropriate value for RT_{NDT}, and the reference fracture toughness curve. The thermal stresses resulting from the temperature gradients through the vessel wall are calculated and then the corresponding (thermal) stress intensity factors, K_{IT}, for the reference flaw are computed. From equation 2, the pressure stress intensity factors are obtained and, from these, the allowable pressures are calculated.

For the calculation of the allowable pressure versus coolant temperature during cooldown, the reference flaw of Appendix G to the ASME Code is assumed to exist at the inside of the vessel wall. During cooldown, the controlling location of the flaw is always at the inside of the wall because the thermal gradients produce tensile stresses at the inside, which increase with increasing cooldown rates. Allowable pressure-temperature relations are generated for both steady-state and finite cooldown rate situations. From these relations, composite limit curves are constructed for each cooldown rate of interest.

The use of the composite curve in the cooldown analysis is necessary because control of the cooldown procedure is based on the measurement of reactor coolant temperature, whereas the limiting pressure is actually dependent on the material temperature at the tip of the assumed flaw.

During cooldown, the 1/4 T vessel location is at a higher temperature than the fluid adjacent to the vessel ID. This condition, of course, is not true for the steady-state situation. It follows that, at any given reactor coolant temperature, the ΔT developed during cooldown results in a higher value of K_{IR} at the 1/4 T location for finite cooldown rates than for steady-state operation. Furthermore, if conditions exist so that the increase in K_{IR} exceeds K_{IT}, the calculated allowable pressure during cooldown will be greater than the steady-state value.

The above procedures are needed because there is no direct control on temperature at the 1/4 T location and, therefore, allowable pressures may unknowingly be violated if the rate of cooling is decreased at various intervals along a cooldown ramp. The use of the composite curve eliminates this problem and ensures conservative operation of the system for the entire cooldown period.

Three separate calculations are required to determine the limit curves for finite heatup rates. As is done in the cooldown analysis, allowable pressure-temperature relationships are developed for steady-state conditions as well as finite heatup rate conditions assuming the presence of a 1/4 T defect at the inside of the wall that alleviate the tensile stresses produced by internal pressure. The metal temperature at the crack tip lags the coolant temperature; therefore, the K_{IR} for the 1/4 T crack during heatup is lower than the K_{IR} for the 1/4 T crack during steady-state conditions at the same time coolant temperature. During heatup, especially at the end of the transient, conditions may exist so that the effects of compressive thermal stresses and lower K_{IR}'s do not offset each other, and the pressure-temperature curve based on steady-state conditions no longer represents a lower bound of all similar curves for finite heatup rates when the 1/4 T flaw is considered. Therefore, both cases have to be analyzed in order to ensure that at any coolant temperature the lower value of the allowable pressure calculated for steady-state and finite heatup rates is obtained.

The second portion of the heatup analysis concerns the calculation of the pressure-temperature limitations for the case in which a 1/4 T deep outside surface flaw is assumed. Unlike the situation at the vessel inside surface,

the thermal gradients established at the outside surface during heatup produce stresses which are tensile in nature and therefore tend to reinforce any pressure stresses present. These thermal stresses are dependent on both the rate of heatup and the time (or coolant temperature) along the heatup ramp. Since the thermal stresses at the outside are tensile and increase with increasing heatup rates, each heatup rate must be analyzed on an individual basis.

Following the generation of pressure-temperature curves for both the steady-state and finite heatup rate situations, the final limit curves are produced by constructing a composite curve based on a point-by-point comparison of the steady-state and finite heatup rate data. At any given temperature, the allowable pressure is taken to be the lesser of the three values taken from the curves under consideration. The use of the composite curve is necessary to set conservative heatup limitations because it is possible for conditions to exist wherein, over the course of the heatup ramp, the controlling condition switches from the inside to the outside, and the pressure limit must at all times be based on analysis of the most critical criterion.

Finally, the 1983 Amendment to 10CFR50[5] has a rule which addresses the metal temperature of the closure head flange and vessel flange regions. This rule states that the metal temperature of the closure flange regions must exceed the material RT_{NDT} by at least 120°F for normal operation when the pressure exceeds 20 percent of the preservice hydrostatic test pressure.

Table 1 indicates that the limiting RT_{NDT} of -22°F occurs in the closure head flange of North Anna Unit 2, so the minimum allowable temperature of this region is 98°F. These limits are shown on Figures 1 through 4, whenever applicable.

4.0 HEATUP AND COOLDOWN LIMIT CURVES

Limit curves for normal heatup and cooldown of the primary Reactor Coolant System have been calculated using the methods discussed in section 3.0. Figures 1 through 4 are applicable for the first 17 EFPY. No instrumentation error margins are considered in developing the heatup and cooldown curves.

Allowable combinations of temperature and pressure for specific temperature change rates are below and to the right of the limit lines shown in figures 1 through 4. This is in addition to other criteria which must be met before the reactor is made critical.

The leak limit curve shown in figures 1, 2, and 3 represent the minimum temperature requirement at the leak test pressure specified by applicable codes[3,4]. The leak test limit curve was determined by methods of references 3 and 5.

Figures 1 through 4 define limits for ensuring prevention of nonductile failure for the North Anna Unit 2 Primary Reactor Coolant System.

5. ADJUSTED REFERENCE TEMPERATURE

From Regulatory Guide 1.99 Rev. 2 [1] the adjusted reference temperature (ART) for each material in the beltline is given by the following expression.

$$ART = \text{Initial } RT_{NDT} + \Delta RT_{NDT} + \text{Margin} \qquad (3)$$

Initial RT_{NDT} is the reference temperature for the unirradiated material as defined in paragraph NB-2331 of Section III of the ASME Boiler and Pressure Vessel Code. If measured values of initial RT_{NDT} for the material in question are not available, generic mean values for that class of material may be used if there are sufficient test results to establish a mean and standard deviation for the class.

ΔRT_{NDT} is the mean value of the adjustment in reference temperature caused by irradiation and should be calculated as follows:

$$\Delta RT_{NDT} = [CF]f^{(0.28-0.10 \log f)} \qquad (4)$$

To calculate ΔRT_{NDT} at any depth (e.g., at 1/4T or 3/4T), the following formula must first be used to attenuate the fluence at the specific depth.

$$f_{(\text{depth } X)} = f_{\text{surface}}^{(e^{-.24x})} \qquad (5)$$

where x (in inches) is the depth into the vessel wall measured from the vessel inner (wetted) surface. The resultant fluence is then put into equation (4) to calculate ΔRT_{NDT} at the specific depth.

CF (°F) is the chemistry factor, obtained from reference 1. The fluence values from reference 2 are reproduced in table 3. The peak fluence value is used to project cumulative fluence up to 17 effective full power years. The calculation of ΔRT for the beltline region are materials shown in tables 4, 5, and 6. Table 7 summarizes the results of RT_{NDT} at 1/4T and 3/4T locations.

TABLE 1

NORTH ANNA UNIT 2 REACTOR VESSEL BELTLINE REGION MATERIAL PROPERTIES

	Cu (Wt.%)	Ni (Wt.%)	CF	I[a] (°F)	M[b] (°F)
Intermediate Shell Plate	.09	.83	58 (35.1)	75	34 (17)
Lower Shell Plate	.13	.83	96	56	34
Weld	.07	.05	38 (10.4)	-48	56 (28)
Closure Head Flange (d)	-	-	-	-31	-
Vessel Flange (d)	-	-	-	-22	-

(a) The initial RT_{NDT} (I) values for the plates and welds are measured values.

(b) Margin (M) as per Reg. Guide 1.99, rev. 2; the standard deviation for the initial RT_{NDT} margin term is assumed to be zero since the initial RT_{NDT} values were obtained from conservative (i.e., "upper bound") test results.

(c) Numbers in ( ) corresponds to surveillance capsule data.

(d) Initial RT_{NDT} values for closure head flange and vessel flange will be considered for the adjustment of heatup/cooldown curves.

TABLE 2
CALCULATION OF CHEMISTRY FACTOR USING NORTH ANNA UNIT 2
SURVEILLANCE CAPSULE DATA

Material	Capsule	f (10^{19} n/cm^2)	ff	ΔRT_{NDT} (°F)	ff x ΔRT_{NDT}	ff^2
Int. Shell	V	.241	.615	9	5.535	.3782
(Tangential)	U	.956	.987	25	24.675	.9742
Int. Shell	V	.241	.615	9	5.535	.3782
(Axial)	U	.956	.987	60	59.22	.9742
				Σ =	94.97	2.7048

o Chemistry Factor (plate) = $\frac{\Sigma(ff \times \Delta RTNDT)}{\Sigma(ff^2)} = \frac{94.97}{2.7048} = 35.1$

Material	Capsule	f	ff	ΔRT_{NDT}	ff x ΔRT_{NDT}	ff^2
Weld Metal	V	.241	.615	2	1.23	.3782
	U	.956	.987	13	12.831	.9742
				Σ =	14.061	1.3524

o Chemistry Factor (weld metal) = $\frac{14.061}{1.3524} = 10.4$

Notes:

f = fluence in 10^{19} n/cm^2
ff = fluence factor
Capsule V information are from reference 6.

TABLE 3

VESSEL NEUTRON EXPOSURE VALUES (E > 1.0 MeV) FOR USE IN GENERATION OF HEATUP/COOLDOWN CURVES [2]

	15 EFPY (n/cm^2)	32 EFPY (n/cm^2)
0° (a)	2.06×10^{19}	4.47×10^{19}
15°	1.03×10^{19}	2.23×10^{19}
30°	5.99×10^{18}	1.30×10^{19}
45°	4.19×10^{18}	9.11×10^{18}

(a) Maximum point on the pressure vessel

TABLE 4

CALCULATION OF ADJUSTED REFERENCE TEMPERATURES FOR NORTH ANNA UNIT 2 REACTOR VESSEL MATERIAL - LOWER SHELL

	Regulatory Guide 1.99 - Revision 2 17 EFPY	
	1/4 T	3/4 T
Chemistry Factor, CF	96	96
Fluence, f (10^{19} n/cm^2)(a)	1.483	.590
Fluence Factor, ff	1.109	.852

****

ΔRT_{NDT} = CF x ff (°F)	106.4	81.8
Initial RT_{NDT}, I (°F)	56	56
Margin, M (°F) (b)	34	34

****

Revision 2 to Regulatory Guide 1.99

Adjusted Reference Temperature, ART = Initial RT_{NDT} + ΔRT_{NDT} + Margin	196	172

****

(a) Fluence, f, is based upon f_{surf} (10^{19} n/cm^2, E>1 Mev) = 2.35 at 17 EFPY. The North Anna Unit 2 reactor vessel wall thickness is 7.677 inches at the beltline region.

(b) Margin is calculated as, $M = 2\,[\sigma_I{}^2 + \sigma_\Delta{}^2]^{0.5}$. The standard deviation for the initial RT_{NDT} margin term (σ_I) is assumed to be 0°F since the initial RT_{NDT} is a measured value.

TABLE 5

CALCULATION OF ADJUSTED REFERENCE TEMPERATURES FOR NORTH ANNA UNIT 2 REACTOR VESSEL MATERIAL - INTERMEDIATE SHELL

	Regulatory Guide 1.99 - Revision 2	
	17 EFPY	
	1/4 T	3/4 T
Chemistry Factor, CF (°F)	58 (35.1)	58 (35.1)
Fluence, f (10^{19} n/cm^2)(a)	1.483	.590
Fluence Factor, ff	1.109	.852
************************		
ΔRT_{NDT} = CF x ff (°F)	64.3 (38.9)	49.4 (29.9)
Initial RT_{NDT}, I (°F)	75	75
Margin, M (°F) (b)	34 (17)	34 (17)
************************		
Revision 2 to Regulatory Guide 1.99		
Adjusted Reference Temperature, ART = Initial RT_{NDT} + ΔRT_{NDT} + Margin	173 (131)	158 (122)

****

(a) Fluence, f, is based upon f_{surf} (10^{19} n/cm^2, E>1 Mev) = 2.35 at 17 EFPY. The North Anna Unit 2 reactor vessel wall thickness is 7.677 inches at the beltline region.

(b) Margin is calculated as, $M = 2\,[\sigma_1^{\ 2} + \sigma_\Delta^{\ 2}]^{0.5}$. The standard deviation for the initial RT_{NDT} margin term (σ_I) is assumed to be 0°F since the initial RT_{NDT} is a measured value. The standard deviation for ΔRT_{NDT}, ($\sigma\Delta$) is 17°F for the plate. σ_Δ is 8.5°F for the plate (cut into half) when surveillance data is used.

( ) numbers in parenthesis were calculated using surveillance capsule data.

TABLE 6

CALCULATION OF ADJUSTED REFERENCE TEMPERATURES FOR NORTH ANNA UNIT 2 REACTOR VESSEL MATERIAL - WELDS

	Regulatory Guide 1.99 - Revision 2 17 EFPY 1/4 T	Regulatory Guide 1.99 - Revision 2 17 EFPY 3/4 T
Chemistry Factor, CF (°F)	38 (10.4)	38 (10.4)
Fluence, f (10^{19} n/cm^2)[(a)]	1.483	.590
Fluence Factor, ff	1.109	.852
******		
ΔRT_{NDT} = CF x ff (°F)	42.0 (11.5)	32.0 (8.9)
Initial RT_{NDT}, I (°F)	-48	-48
Margin, M (°F) [(b)]	42 (11.5)	32 (9)
******		
Revision 2 to Regulatory Guide 1.99		
Adjusted Reference Temperature, ART = Initial RT_{NDT} + ΔRT_{NDT} + Margin	36 (-25)	16 (-30)

****

(a) Fluence, f, is based upon f_{surf} (10^{19} n/cm^2, E>1 Mev) = 2.35 at 17 EFPY. The North Anna Unit 2 reactor vessel wall thickness is 7.677 inches at the beltline region.

(b) Margin is calculated as, $M = 2\,[\sigma_I^2 + \sigma_\Delta^2]^{0.5}$. The standard deviation for the initial RT_{NDT} margin term (σ_I) is assumed to be 0°F since the initial RT_{NDT} is a measured value. The standard deviation for ΔRT_{NDT}, ($\sigma\Delta$) is 28°F for the weld. σ_Δ is 14°F for the weld (cut into half) when surveillance data is used. Also σ_Δ need not exceed 1/2 ΔRT_{NDT}.

( ) numbers in parenthesis were calculated using surveillance capsule data.

TABLE 7

SUMMARY OF ADJUSTED REFERENCE TEMPERATURES (ART) AT 1/4T AND 3/4T LOCATION

Component	17 EFPY	
	1/4T (°F)	3/4T (°F)
Intermediate Shell	173 (131)	158 (122)
Lower Shell	196.0*	172*
Weld	35 (-25)	15 (-30)

Numbers within ( ) are using chemistry factor based on surveillance capsule.

* Adjusted Reference Temperatures of 196°F at 1/4T and 172°F at 3/4T used to generate hoatup/cooldown curves applicable up to 17 EFPY.

MATERIAL PROPERTY BASIS

CONTROLLING MATERIAL: LOWER SHELL PLATE
INITIAL RT_{NDT}: 56°F
RT_{NDT} AFTER 17 EFPY: 1/4T, 196°F
3/4T, 172°F

CURVES APPLICABLE FOR HEATUP RATES UP TO 20°F/HR FOR THE SERVICE PERIOD UP TO 17 EFPY. CONTAINS NO MARGIN FOR POSSIBLE INSTRUMENT ERRORS.

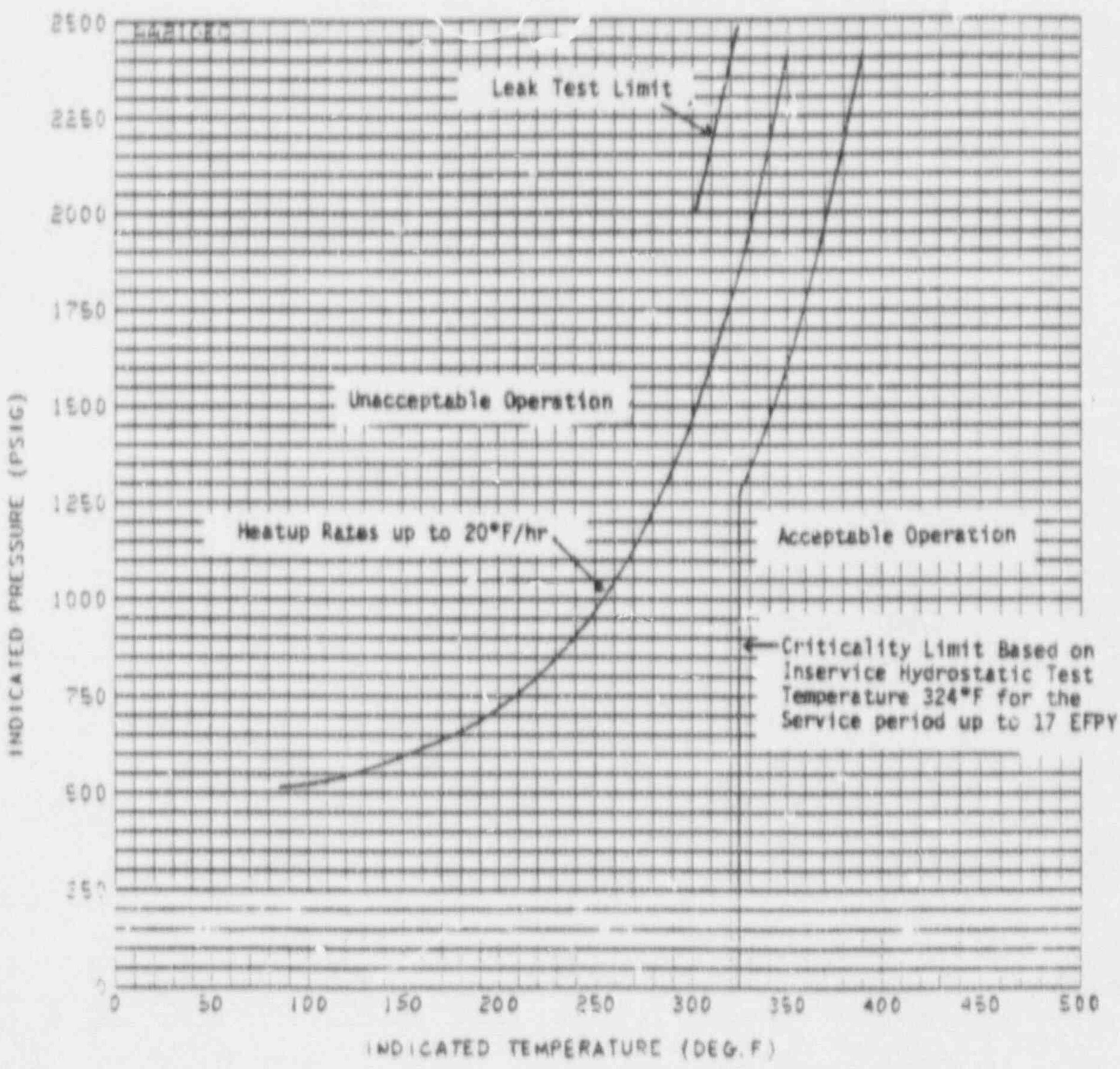


Figure A-2. North Anna Unit 2 Reactor Coolant System Heatup Limitations Applicable for the First 17 EFPY (Without Margins for Instrumentation Errors) Up to 20°F/hr

MATERIAL PROPERTY BASIS

CONTROLLING MATERIAL: LOWER SHELL PLATE
INITIAL RT_{NDT}: 56°F
RT_{NDT} AFTER 17 EFPY: 1/4T, 196°F
3/4T, 172°F

CURVES APPLICABLE FOR HEATUP RATES UP TO 40°F/HR FOR THE SERVICE PERIOD UP TO 17 EFPY. CONTAINS NO MARGIN FOR POSSIBLE INSTRUMENT ERRORS.

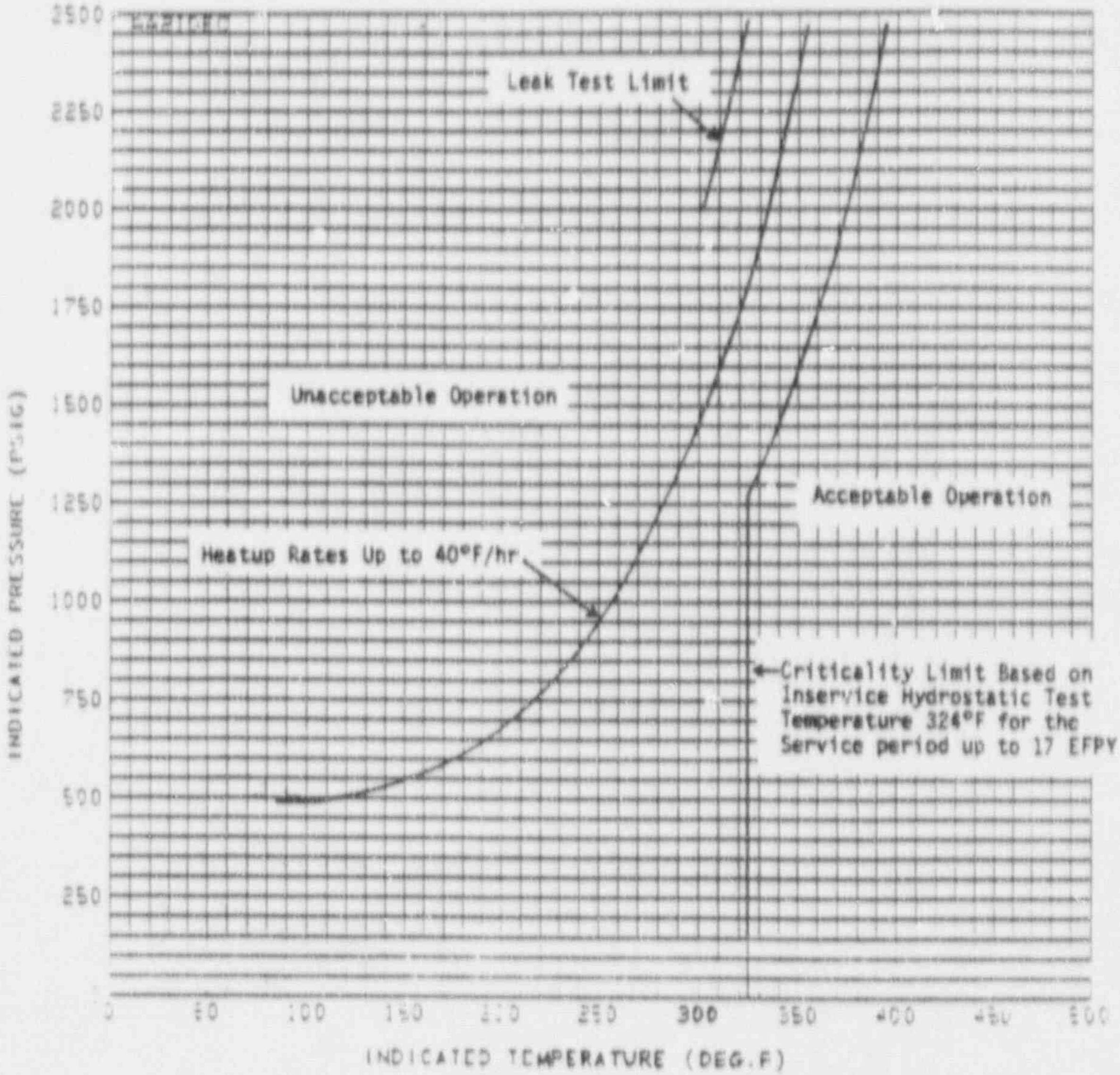


Figure A-3. North Anna Unit 2 Reactor Coolant System Heatup Limitations Applicable for the First 17 EFPY (Without Margins for Instrumentation Errors) Up to 40°F/hr

MATERIAL PROPERTY BASIS

CONTROLLING MATERIAL: LOWER SHELL PLATE

INITIAL RT_{NDT}: 56°F

RT_{NDT} AFTER 17 EFPY: 1/4T, 196°F

3/4T, 172°F

CURVES APPLICABLE FOR HEATUP RATES UP TO 60°F/HR FOR THE SERVICE PERIOD UP TO 17 EFPY. CONTAINS NO MARGIN FOR POSSIBLE INSTRUMENT ERRORS.

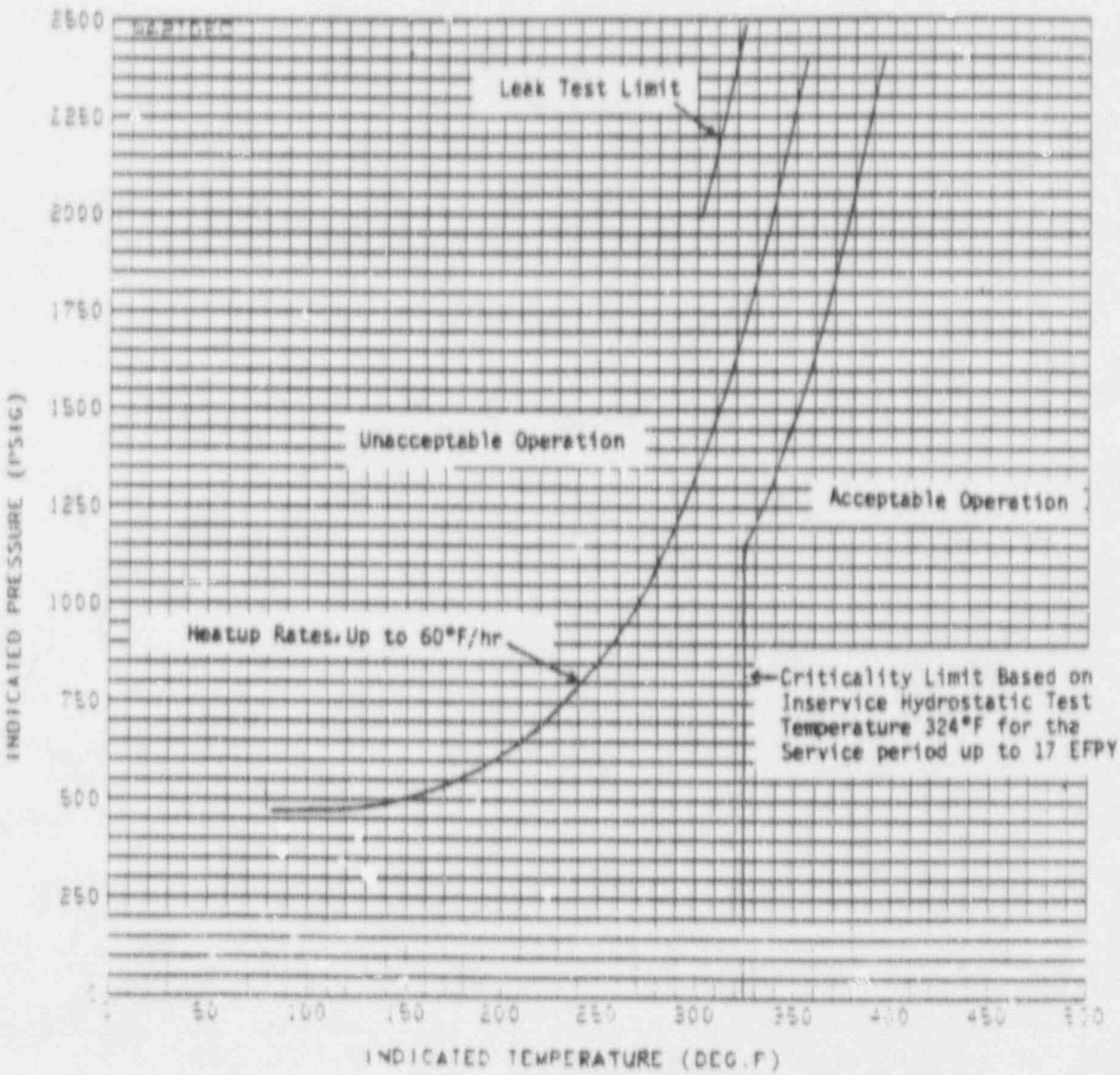


Figure A-4. North Anna Unit 2 Reactor Coolant System Heatup Limitations Applicable for the First 17 EFPY (Without Margins for Instrumentation Errors) Up to 60°F/hr

MATERIAL PROPERTY BASIS

CONTROLLING MATERIAL: LOWER SHELL PLATE
INITIAL RT_{NDT}: 56°F
RT_{NDT} AFTER 17 EFPY: 1/4T, 196°F
3/4T, 172°F

CURVES APPLICABLE FOR COOLDOWN RATES UP TO 100°F/HR FOR THE SERVICE PERIOD UP TO 17 EFPY. CONTAINS NO MARGIN FOR POSSIBLE INSTRUMENT ERRORS.

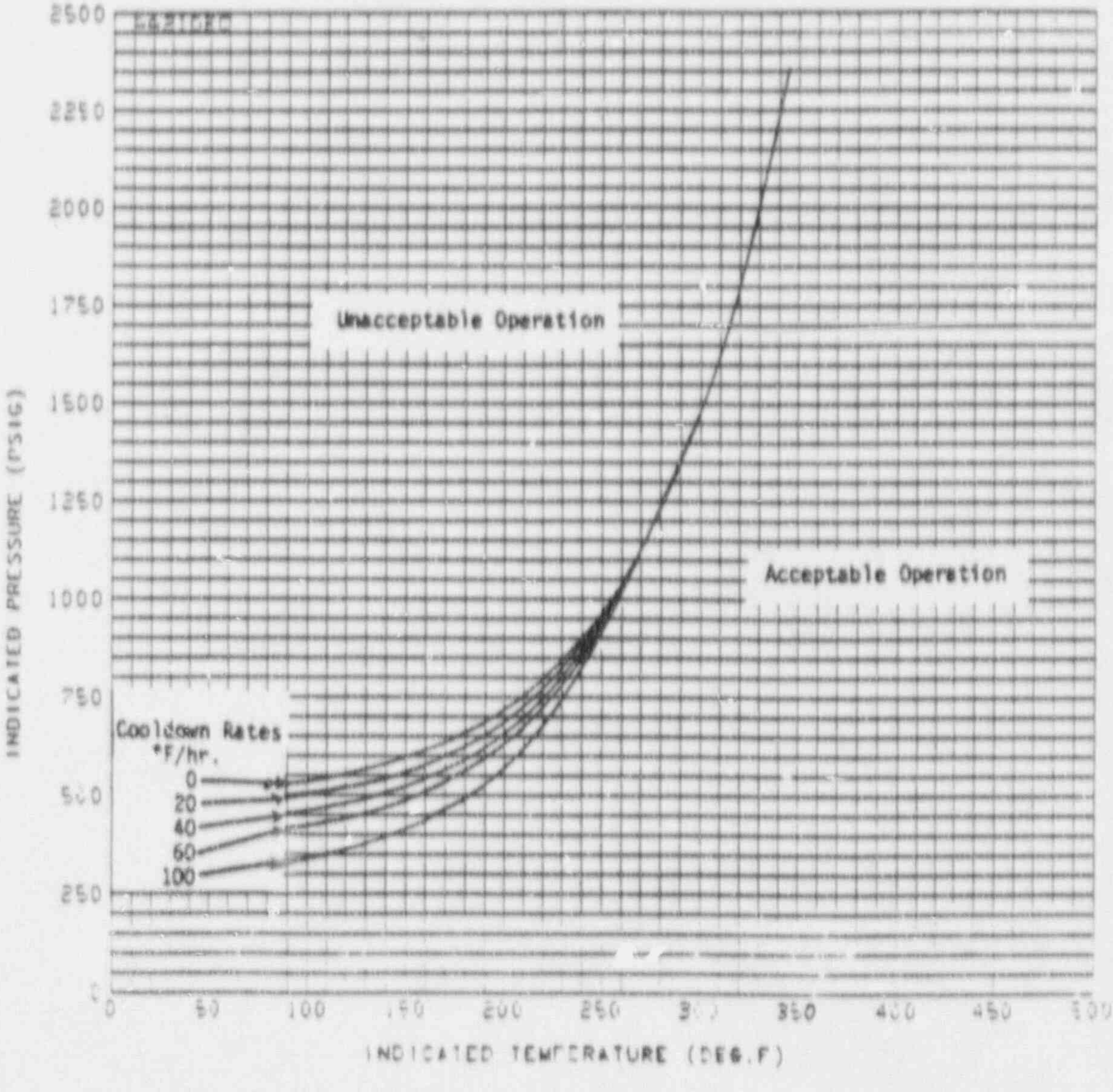


Figure A-5. North Anna Unit 2 Reactor Coolant System Cooldown Limitations Applicable for the First 17 EFPY (Without Margins for Instrumentation Errors)

6.0 REFERENCES

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

[2] Terek, E., Anderson, S. L., Albertin, L., WCAP-12497, January 1990, "Analysis of Capsule U from the Virginia Electric Power Company North Anna 2 Reactor Vessel Radiation Surveillance Program".

[3] "Fracture Toughness Requirements," Branch Technical Position MTEB 5-2, Chapter 5.3.2 in Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants, LWR Edition, NUREG-800, 1981.

[4] ASME Boiler and Pressure Vessel Code, Section III, Division 1 - Appendixes, "Rules for Construction of Nuclear Power Plant Components, Appendix G, Protection Against Nonductile Failure," pp. 558-563, 1986 Edition, American Society of Mechanical Engineers, New York, 1986.

[5] Code of Federal Regulations, 10CFR50, Appendix G, "Fracture Toughness Requirements," U.S. Nuclear Regulatory Commission, Washington, D.C., Federal Register, Vol. 48 No. 104, May 27, 1983.

[6] "Analysis of Capsule V, Virginia Electric Power Company North Anna Unit 2, Reactor Vessel Materials Surveillance Program", BAW-1794, Lowe, A. L. Jr. et. al. October, 1983.

APPENDIX A

Data Points for Heatup and Cooldown Curves

THE FOLLOWING DATA WERE CALCULATEDFOR THE INSERVICE HYDROSTATIC LEAK TEST.

MINIMUM INSERVICE LEAK TEST TEMPERATURE (17.000 EFPY)

PRESSURE (PSI)	TEMPERATURE (DEG.F)
2000	302
2485	324

PRESSURE (PSI)	PRESSURE STRESS (PSI)	1.5 K1M (PSI SQ.RT.IN.)
2000	21051	84522
2485	26156	105823

VGB HEATUP CURVE REG. GUIDE 1.99,REV.2 01/24/90

COMPOSITE CURVE PLOTTED FOR HEATUP PROFILE 2 HEATUP RATE(S) (DEG.F/HR) = 20.0

IRRADIATION PERIOD = 17.000 EFP YEARS
FLAW DEPTH = (1-ADWIN)T

	INDICATED TEMPERATURE (DEG.F)	INDICATED PRESSURE (PSI)		INDICATED TEMPERATURE (DEG.F)	INDICATED PRESSURE (PSI)		INDICATED TEMPERATURE (DEG.F)	INDICATED PRESSURE (PSI)
1	85.000	513.18	19	175.000	645.54	37	265.000	1082.23
2	90.000	514.28	20	180.000	657.85	38	270.000	1127.24
3	95.000	518.10	21	185.000	671.08	39	275.000	1175.32
4	100.000	522.09	22	190.000	685.29	40	280.000	1226.73
5	105.000	527.12	23	195.000	700.43	41	285.000	1278.89
6	110.000	532.41	24	200.000	71[illegible].88	42	290.000	1334.84
7	115.000	538.35	25	205.000	734.54	43	295.000	1394.87
8	120.000	544.65	26	210.000	753.3[illegible]	44	300.000	1459.10
9	125.000	551.44	27	215.000	773.79	45	305.000	1528.12
10	130.000	558.79	28	220.000	795.53	46	310.000	1601.98
11	135.000	566.76	29	225.000	819.14	47	315.000	1681.18
12	140.000	575.29	30	230.000	844.27	48	320.000	1766.00
13	145.000	584.52	31	235.000	871.53	49	325.000	1856.82
14	150.000	594.38	32	240.000	900.61	50	330.000	1954.10
15	155.000	604.47	33	245.000	932.01	51	335.000	2058.33
16	160.000	613.68	34	250.000	965.68	52	340.000	2169.49
17	165.000	623.58	35	255.000	1001.79	53	345.000	2288.42
18	170.000	634.24	36	260.000	1040.57	54	350.000	2415.28

COMPOSITE CURVE PLOTTED FOR HEATUP PROFILE 2 HEATUP RATE(S) (DEG.F/HR) = 40.0

IRRADIATION PERIOD 17.000 EFP YEARS
FLAW DEPTH = (1-AOWIN)T

	INDICATED TEMPERATURE (DEG.F)	INDICATED PRESSURE (PSI)		INDICATED TEMPERATURE (DEG.F)	INDICATED PRESSURE (PSI)		INDICATED TEMPERATURE (DEG.F)	INDICATED PRESSURE (PSI)
1	85.000	~~488.05~~ 493.06	20	180.000	613.92	38	270.000	1122.83
2	90.000	~~494.62~~	21	185.000	628.27	39	275.000	1174.89
3	95.000	493.06	22	190.000	643.65	40	280.000	1226.82
4	100.000	493.27	23	195.000	660.12	41	285.000	1278.19
5	105.000	495.06	24	200.000	677.85	42	290.000	1330.79
6	110.000	497.85	25	205.000	696.97	43	295.000	1387.15
7	115.000	501.67	26	210.000	717.58	44	300.000	1447.61
8	120.000	506.10	27	215.000	739.72	45	305.000	1512.06
9	125.000	511.40	28	220.000	763.38	46	310.000	1581.69
10	130.000	517.28	29	225.000	788.95	47	315.000	1655.78
11	135.000	523.86	30	230.000	816.32	48	320.000	1735.73
12	140.000	531.00	31	235.000	845.67	49	325.000	1820.99
13	145.000	538.83	32	240.000	877.42	50	330.000	1912.20
14	150.000	547.16	33	245.000	911.38	51	335.000	2009.92
15	155.000	556.35	34	250.000	947.79	52	340.000	2114.39
16	160.000	566.21	35	255.000	986.92	53	345.000	2225.88
17	165.000	576.88	36	260.000	1028.17	54	350.000	2345.12
18	170.000	588.36	37	265.000	1074.33	55	355.000	2472.24
19	175.000	600.63						

COMPOSITE CURVE PLOTTED FOR HEATUP PROFILE 2 HEATUP RATE(S) (DEG.F/HR) = 60.0

IRRADIATION PERIOD = 17.000 EFP YEARS
FLAW DEPTH = (1-AOWIN)T

	INDICATED TEMPERATURE (DEG.F)	INDICATED PRESSURE (PSI)
1	85.000	~~488.77~~
2	90.000	~~486.18~~
3	95.000	~~478.42~~ 473.41
4	100.000	~~475.60~~
5	105.000	~~473.87~~
6	110.000	473.41
7	115.000	474.31
8	120.000	476.18
9	125.000	479.10
10	130.000	482.79
11	135.000	487.34
12	140.000	492.59
13	145.000	498.61
14	150.000	505.22
15	155.000	512.67
16	160.000	520.82
17	165.000	529.76
18	170.000	539.46
19	175.000	549.90
20	180.000	561.29
21	185.000	573.63
22	190.000	586.93
23	195.000	601.16
24	200.000	616.62
25	205.000	633.29
26	210.000	651.06
27	215.000	670.37
28	220.000	691.09
29	225.000	713.28
30	230.000	737.25
31	235.000	762.91
32	240.000	790.59
33	245.000	820.26
34	250.000	852.06
35	255.000	886.40
36	260.000	923.17
37	265.000	962.66
38	270.000	1005.04
39	275.000	1050.55
40	280.000	1099.43
41	285.000	1151.82
42	290.000	1208.04
43	295.000	1268.40
44	300.000	1332.88
45	305.000	1402.35
46	310.000	1476.62
47	315.000	1556.29
48	320.000	1641.53
49	325.000	1732.71
50	330.000	1830.42
51	335.000	1934.85
52	340.000	2046.38
53	345.000	2165.60
54	350.000	2282.71
55	355.000	2402.28

VGB COOLDOWN CURVES REG. GUIDE 1.99,REV 2 01/24/90

THE FOLLOWING DATA WERE PLOTTED FOR COOLDOWN PROFILE 1 (STEADY-STATE COOLDOWN)

IRRADIATION PERIOD = 17.000 EFP YEARS
FLAW DEPTH - 1/4T

	INDICATED TEMPERATURE (DEG.F)	INDICATED PRESSURE (PSI)		INDICATED TEMPERATURE (DEG.F)	INDICATED PRESSURE (PSI)		INDICATED TEMPERATURE (DEG.F)	INDICATED PRESSURE (PSI)
1	85.000	526.54	19	175.000	645.54	37	265.000	1082.23
2	90.000	529.88	20	180.000	657.85	38	270.000	1127.24
3	95.000	533.48	21	185.000	671.08	39	275.000	1175.32
4	100.000	537.34	22	190.000	685.29	40	280.000	1226.82
5	105.000	541.48	23	195.000	700.43	41	285.000	1282.38
6	110.000	545.96	24	200.000	716.88	42	290.000	1342.01
7	115.000	550.68	25	205.000	734.54	43	295.000	1405.87
8	120.000	555.81	26	210.000	753.36	44	300.000	1474.37
9	125.000	561.36	27	215.000	773.79	45	305.000	1547.82
10	130.000	567.33	28	220.000	795.53	46	310.000	1626.80
11	135.000	573.74	29	225.000	819.14	47	315.000	1711.17
12	140.000	580.64	30	230.000	844.27	48	320.000	1801.58
13	145.000	588.05	31	235.000	871.53	49	325.000	1898.56
14	150.000	595.90	32	240.000	900.61	50	330.000	2002.43
15	155.000	604.47	33	245.000	932.01	51	335.000	2113.50
16	160.000	613.68	34	250.000	965.68	52	340.000	2231.98
17	165.000	623.58	35	255.000	1001.79	53	345.000	2358.87
18	170.000	634.24	36	260.000	1040.57			

THE FOLLOWING DATA WERE PLOTTED FOR COOLDOWN PROFILE 2 (20 DEG-F / HR COOLDOWN)

IRRADIATION PERIOD = 17.000 EFP YEARS
FLAW DEPTH = AOWIN T

	INDICATED TEMPERATURE (DEG.F)	INDICATED PRESSURE (PSI)		INDICATED TEMPERATURE (DEG.F)	INDICATED PRESSURE (PSI)		INDICATED TEMPERATURE (DEG.F)	INDICATED PRESSURE (PSI)
1	85.000	488.96	14	150.000	560.43	27	215.000	747.64
2	90.000	492.32	15	155.000	569.39	28	220.000	770.89
3	95.000	495.97	16	160.000	578.01	29	225.000	795.70
4	100.000	499.89	17	165.000	589.39	30	230.000	822.61
5	105.000	504.04	18	170.000	600.41	31	235.000	851.33
6	110.000	508.60	19	175.000	612.44	32	240.000	882.39
7	115.000	513.54	20	180.000	625.37	33	245.000	915.70
8	120.000	518.84	21	185.000	639.29	34	250.000	951.41
9	125.000	524.57	22	190.000	654.13	35	255.000	989.83
10	130.000	530.72	23	195.000	670.26	36	260.000	1031.31
11	135.000	537.38	24	200.000	687.59	37	265.000	1075.78
12	140.000	544.53	25	205.000	706.12	38	270.000	1123.49
13	145.000	552.14	26	210.000	726.21	39	275.000	1174.83

VGB COOLDOWN CURVES REG GUIDE 1.99,REV 2 01/24/90

THE FOLLOWING DATA WERE PLOTTED FOR COOLDOWN PROFILE 3 (40 DEG-F / HR COOLDOWN)

IRRADIATION PERIOD = 17.000 EFP YEARS
FLAW DEPTH = AOWIN T

	INDICATED TEMPERATURE (DEG.F)	INDICATED PRESSURE (PSI)		INDICATED TEMPERATURE (DEG.F)	INDICATED PRESSURE (PSI)		INDICATED TEMPERATURE (DEG.F)	INDICATED PRESSURE (PSI)
1	85.000	450.56	14	150.000	524.52	27	215.000	722.17
2	90.000	453.96	15	155.000	533.89	28	220.000	746.64
3	95.000	457.67	16	160.000	543.96	29	225.000	773.22
4	100.000	461.67	17	165.000	554.75	30	230.000	801.59
5	105.000	466.01	18	170.000	566.46	31	235.000	832.35
6	110.000	470.69	19	175.000	579.12	32	240.000	865.26
7	115.000	475.77	20	180.000	592.72	33	245.000	900.66
8	120.000	481.23	21	185.000	607.30	34	250.000	938.71
9	125.000	487.16	22	190.000	623.10	35	255.000	979.89
10	130.000	493.54	23	195.000	640.15	36	260.000	1023.98
11	135.000	500.46	24	200.000	658.35	37	265.000	1071.42
12	140.000	507.80	25	205.000	678.13	38	270.000	1122.35
13	145.000	515.86	26	210.000	698.23			

VGB COOLDOWN CURVES REG. GUIDE 1.99,REV 2 01/24/90

THE FOLLOWING DATA WERE PLOTTED FOR COOLDOWN PROFILE 4 (60 DEG-F / HR COOLDOWN)

IRRADIATION PERIOD = 17.000 EFP YEARS
FLAW DEPTH = AOWIN T

	INDICATED TEMPERATURE (DEG.F)	INDICATED PRESSURE (PSI)		INDICATED TEMPERATURE (DEG.F)	INDICATED PRESSURE (PSI)		INDICATED TEMPERATURE (DEG.F)	INDICATED PRESSURE (PSI)
1	85.000	411.52	14	150.000	488.13	27	215.000	687.07
2	90.000	414.92	15	155.000	497.96	28	220.000	723.29
3	95.000	418.70	16	160.000	508.45	29	225.000	751.38
4	100.000	422.79	17	165.000	519.91	30	230.000	781.78
5	105.000	427.25	18	170.000	532.25	31	235.000	814.39
6	110.000	432.[illegible]	19	175.000	545.60	32	240.000	849.44
7	115.000	437.31	20	180.000	559.86	33	245.000	887.38
8	120.000	442.96	21	185.000	575.39	34	250.000	928.04
9	125.000	449.04	22	190.000	592.10	35	255.000	971.85
10	130.000	455.67	23	195.000	610.03	36	260.000	1018.93
11	135.000	462.88	24	200.000	629.47	37	265.000	1069.60
12	140.000	470.64	25	205.000	650.30	38	270.000	1124.03
13	145.000	479.06	26	210.000	672.88			

VGB COOLDOWN CURVES REG. GUIDE 1.99, REV 2 01/24/90

THE FOLLOWING DATA WERE PLOTTED FOR COOLDOWN PROFILE 5 (100 DEG-F/HR COOLDOWN)

IRRADIATION PERIOD = 17.000 EFP YEARS
FLAW DEPTH = AOWIN T

	INDICATED TEMPERATURE (DEG.F)	INDICATED PRESSURE (PSI)		INDICATED TEMPERATURE (DEG.F)	INDICATED PRESSURE (PSI)		INDICATED TEMPERATURE (DEG.F)	INDICATED PRESSURE (PSI)
1	85.000	330.73	14	150.000	413.84	26	210.000	621.76
2	90.000	334.33	15	155.000	424.75	27	215.000	649.26
3	95.000	338.31	16	160.000	436.52	28	220.000	678.06
4	100.000	342.63	17	165.000	449.22	29	225.000	711.08
5	105.000	347.38	18	170.000	463.01	30	230.000	745.55
6	110.000	352.53	19	175.000	477.96	31	235.000	782.93
7	115.000	358.18	20	180.000	494.08	32	240.000	823.05
8	120.000	364.29	21	185.000	511.46	33	245.000	866.28
9	125.000	370.94	22	190.000	530.29	34	250.000	912.81
10	130.000	376.17	23	195.000	550.56	35	255.000	962.99
11	135.000	386.06	24	200.000	572.82	36	260.000	1016.82
12	140.000	394.59	25	205.000	596.15	37	265.000	1075.03
13	145.000	403.87						

21/32

1.35 1.62
25 1.33 Full

In addition to the North Anna Unit 2 revised heatup and cooldown curves, the original Westinghouse Report WCAP-12503 documents the results of a study prepared by Westinghouse for Virginia Power on the effects of accelerated cooldowns on heatup and cooldown limits. Virginia Power has removed the pages which document these results as they are not being submitted for review and approval at this time.

REFERENCE LISTS

REFERENCES

(1) Letter from W. R. Cartwright to USNRC, "Virginia Electric and Power Company, North Anna Power Station Unit 1, Proposed Technical Specification Change," NRC Letter Serial No. 88-202A, dated November 30, 1988.

(2) Letter from W. L. Stewart to USNRC, "Virginia Electric and Power Company, North Anna Power Station Unit 1, Proposed Technical Specification Change - Supplement," NRC Letter Serial No. 88-202B, dated June 19, 1989.

(3) Letter from USNRC to W. R. Cartwright, "North Anna Unit 1, Issuance of Amendment Re: Heatup and Cooldown Curves (TAC No. 72060)," NRC Letter Serial No. 88-499, dated June 30, 1989.

(4) S. E. Yanichko and L. Albertin: "Analysis of Capsule U from the Virginia Electric and Power Company North Anna Unit 1 Reactor Vessel Radiation Surveillance Program," WCAP-11777, dated February, 1990.

(5) E. Terek and L. Albertin: "Analysis of Capsule U from the Virginia Electric and Power Company North Anna Unit 2 Reactor Vessel Radiation Surveillance Program," WCAP-12497, dated January, 1990.

(6) Letter from W. L. Stewart to USNRC, "Virginia Electric and Power Company, North Anna Power Station Unit 2, Reactor Vessel Materials Surveillance Program," NRC Letter Serial No. 90-097, dated March 8, 1990.

(7) Letter from W. L. Stewart to USNRC, "Virginia Electric and Power Company, North Anna Power Station Units 1 and 2, Heatup and Cooldown Curves," NRC Letter Serial No. 91-067, dated February 14, 1991.

(8) Letter from W. L. Stewart to USNRC, "Virginia Electric and Power Company, North Anna Power Station Unit 1, Reactor Vessel Materials Surveillance Program," NRC Letter Serial No. 88-202, dated June 1, 1989.

(9) A. D. Nana, et al.: "North Anna Unit 1 Pressure-Temperature Limits for 12 EFPY and North Anna Unit 2 Pressure-Temperature Limits for 12 and 15 EFPY," BAW-2146, dated October, 1991.

(10) N. K. Ray, et al.: "North Anna Unit 2 Reactor Vessel Heatup and Cooldown Limit Curves for Normal Operation (Capsule U)," WCAP-12503, dated March, 1990.

(11) Code of Federal Regulations, Title 10, Part 50, Appendix G, "Energy: Domestic Licensing of Production and Utilization Facilities; Fracture Toughness Requirements," published January 1, 1988 by the Office of the Federal Register, National Archives and Records Administration.

(12) Updated Final Safety Analysis Report," North Anna Power Station, Units 1 and 2, Virginia Electric and Power Company.

(13) "EPRI PWR Safety and Relief Valve Test Program, Safety and Relief Valve Test Report," EPRI, NP-2628-SR, December, 1982.

(14) "Safety and Relief Valves in Light Water Reactors," EPRI, NP-4306-SR, December, 1985.

(15) "Reactor System Transient Analyses Using the RETRAN Computer Code," VEP-FRD-41, March, 1981; as supplemented by letter from W. L. Stewart to USNRC, "Virginia Electric and Power Company, Surry and North Anna Power Stations, Reactor System Transient Analysis" NRC Letter Serial No. 85-753, dated November 19, 1985.

(16) "NRC Position on Radiation Embrittlement of Reactor Vessel Materials and Its Impact on Plant Operations (Generic Letter 88-11)," U.S. Nuclear Regulatory Commission, July 12, 1988.

(17) C. C. Heinecke, et al.: "North Anna Units 1 and 2 Reactor Vessel Fluence and RT_{PTS} Evaluations," WCAP-11016, Rev. 3, dated January, 1988.

(18) Code of Federal Regulations, Title 10, Part 50.61, Federal Register Volume 56, No. 94, dated Wednesday May 15, 1991 (Incorporation of the Methodology of Regulatory Guide 1.99, Revision 2 into the 10 CFR 50.61 PTS Rule).

(19) Letter from J. H. Goldberg (Florida Power and Light) to USNRC, St. Lucie Unit 1, Docket No. 50-335, Proposed License Amendment, P-T Limits and LTOP Analysis, dated December 5, 1989.

(20) Letter from USNRC to J. H. Goldberg (Florida Power and Light), St. Lucie Unit 1 - Issuance of Amendment Re: Pressure/Temperature (P/T) Limits and Low Temperature Overpressure Protection (LTOP) Analysis (TAC No. 75386), Docket No. 50-335, dated June 11, 1990.