ATTACHMENT 4

WESTINGHOUSE REPORT WCAP-13831, REVISION 1

VIRGINIA ELECTRIC AND POWER COMPANY

Westinghouse Class 3 (Non-Proprietary)

WCAP-13831
Revision 1

# HEATUP AND COOLDOWN LIMIT CURVES FOR NORMAL OPERATION FOR NORTH ANNA UNIT 1

J. M. Chicots
M. J. Malone

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for the Virginia Electric and Power Company

Approved by: Bruce A. Bishop for
T. A. Meyer, Manager
Structural Reliability and Plant Life Optimization

WESTINGHOUSE ELECTRIC CORPORATION
Nuclear and Advanced Technology Division
P.O. Box 355
Pittsburgh, Pennsylvania 15230-0355

## PREFACE

This report was revised to change the margin values for the upper shell forging and weld seams W05A and W05B. In addition, the maximum temperature difference between RCS fluid and reactor vessel 1/4-T and 3/4-T locations for heatup and cooldown rates was added.

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## 1. INTRODUCTION

Heatup and cooldown limit curves are calculated using the most limiting value of $RT_{NDT}$ (reference nil-ductility temperature) corresponding to the limiting beltline region material 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 unirradiated reactor vessel material fracture toughness properties and estimating the radiation-induced $\Delta RT_{NDT}$. The unirradiated $RT_{NDT}$ 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 majcr 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, $\Delta RT_{NDT}$ 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]. Regulatory Guide 1.99, Revision 2 is used for the calculation of ART values at 1/4-T and 3/4-T locations. T is the thickness of the vessel at the beltline region measured from the clad/base metal interface.

The pressure-temperature limit curves in Figures 1 through 4 of this report correspond to allowable pressure-temperature values at the limiting beltline region of the reactor vessel and do not include margins for instrumentation errors or for pressure differences between the wide-range pressure transmitter and the limiting reactor vessel beltline region.

## 2. 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[2]. The pre-irradiation fracture-toughness properties of the North Anna Unit 1 reactor vessel are presented in Table 1. The post-irradiation fracture toughness properties of the intermediate to lower shell circumferential weld and the lower shell forging were obtained directly from the North Anna Unit 1 Reactor Vessel Radiation Surveillance Program. Credible surveillance data are currently available for two capsules (Capsules U and V) for North Anna Unit 1.

## 3. 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 of the ASME Code[3]. The $K_{IR}$ curve is given by the following equation:

$$K_{IR} = 26.78 + 1.223 * e^{[0.0145 (T-RTNDT + 160)]} \qquad (1)$$

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[3] as follows:

$$C * K_{IM} + K_{IT} \leq K_{IR} \qquad (2)$$

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 $\Delta T$ 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. The heatup results in compressive stresses at the inside surface 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 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[4] 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 unirradiated $RT_{NDT}$ by at least 120°F for normal operation when the pressure exceeds 20 percent of the preservice hydrostatic test pressure (621 psig for North Anna Unit 1). Table 1 indicates that the limiting unirradiated $RT_{NDT}$ of -22°F occurs in the vessel flange of North Anna Unit 1, so the minimum allowable temperature of this region is 98°F at pressures greater than 621 psig. This limit is shown in Figures 1 through 4 whenever applicable.

## 4. HEATUP AND COOLDOWN PRESSURE-TEMPERATURE LIMIT CURVES

Pressure-temperature limit curves for normal heatup and cooldown of the primary reactor coolant system have been calculated for the pressure and temperature in the reactor vessel beltline region using the methods discussed in Section 3. Since indication of reactor vessel beltline pressure is not available on the plant, the pressure difference between the wide-range pressure transmitter and the limiting beltline region must be accounted for when using the pressure-temperature limit curves presented in this report.

Figures 1 through 3 present the heatup curves using heatup rates of 20°F/hr, 40°F/hr and 60°F/hr applicable for the end-of-license life (30.7 EFPY), respectively. Figure 4 presents the cooldown curves using cooldown rates up to 100°F/hr applicable for the end-of-license life (30.7 EFPY). No margins for possible instrumentation errors are included in the development of 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 reactor must not be made critical until pressure-temperature combinations are to the right of the criticality limit line shown in Figures 1 through 3. The straight line portion of the criticality limit is at the minimum permissible temperature for the 2485 psig inservice hydrostatic test as required by Appendix G to 10CFR Part 50. The governing equation for the hydrostatic test is defined in Appendix G to Section III of the ASME Code as follows:

$$1.5\, K_{IM} \leq K_{IR},$$

where,

$K_{IM}$ is the stress intensity factor covered by membrane (pressure) stress,

$K_{IR} = 26.78 + 1.233\, e^{[0.0145\,(T - RTNDT + 160)]}$,

T is the minimum permissible metal temperature, and

RTNDT is the metal reference nil-ductility temperature.

The curved portion of the criticality limit is shifted 40°F to the right of and parallel to the heatup curve as required by Appendix G to 10 CFR Part 50. It should be noted that there are other criteria which must be met before the reactor is made critical. For example, the reactor must not be made critical until a steam bubble is formed in the pressurizer. The leak test limit curve shown on the heatup curve in Figures 1 through 3 represents minimum temperature requirements at leak test pressures ranging from 2000 psig to 2485 psig. The leak test limit curve was determined by the same method used to compute the inservice hydrostatic test temperature. This method used a 1.5 safety factor on the pressure stress intensity factor as explained previously.

The leak limit curve shown in Figures 1 through 3 represents minimum temperature requirements at the leak test pressure specified by applicable codes[2,3]. The leak test limit curve was determined by methods of References 2 and 4.

Figures 1 through 4 define limits for ensuring prevention of nonductile failure for the North Anna Unit 1 reactor vessel.

The data points used to develop the heatup and cooldown pressure-temperature limit curves shown in Figures 1 through 4 are presented in Appendix A.

The maximum temperature difference between the RCS fluid and the vessel at the 1/4-T and 3/4-T locations for both heatup and cooldown cases were calculated and the results are provided in Table 4.

## 5. CALCULATION OF 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.

$\Delta RT_{NDT}$ 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 $\Delta RT_{NDT}$ at any depth (e.g., at 1/4-T or 3/4-T), the following formula must first be used to attenuate the fluence at the specific depth.

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

where x (in inches) is the depth into the vessel wall measured from the vessel clad/base metal interface. The resultant fluence is then put into equation (4) to calculate $\Delta RT_{NDT}$ at the specific depth. The calculated surface fluence at end-of-license life (30.7 EFPY) for North Anna Unit 1 is $3.95 \times 10^{19}$ n/cm$^2$ [6].

CF (°F) is the chemistry factor, obtained from Tables in Reference 1, using the values of the copper and nickel content as reported in Table 1. If plant-specific surveillance data has been deemed credible per Reg. Guide 1.99, Revision 2, it may be considered in the calculation of the chemistry factor.

All materials in the beltline region of North Anna Unit 1 reactor vessel were considered in determining the limiting material. The results of the ART's at 1/4-T and 3/4-T are summarized in Table 2. From Table 2, it can be seen that the limiting material is the circumferential weld seam for heatup and cooldown curves applicable up to end-of-license life (30.7 EFPY). Sample calculations to determine the ART values for the circumferential weld seam at end-of-license life (30.7 EFPY) are shown in Table 3.

TABLE 1
NORTH ANNA UNIT 1 REACTOR VESSEL TOUGHNESS TABLE
(Unirradiated)

| Material Description | Cu[1] (%) | Ni (%) | $IRT_{NDT}$ (°F) |
|---|---|---|---|
| Closure Head Flange | -- | 0.82 | -40 (b) |
| Vessel Flange | -- | 0.77 | -22 (b) |
| Upper Shell Forging 05 | 0.16 | 0.74 | 6 (c) |
| Intermediate Shell Forging 04 | 0.12 | 0.82 | 17 (a) |
| Lower Shell Forging 03 | 0.15 | 0.80 | 38 (a) |
| Intermediate Shell to Lower Shell Circumferential Weld Seam, W04 | 0.086 | 0.11 | 19 (a) |
| Weld Seam, W05A | 0.30 | 0.10 | 0 (c) |
| Weld Seam, W05B | 0.11 | 0.10 | 0 (c) |

(a) The initial $RT_{NDT}$ values for the plates and circumferential weld are measured values.

(b) Initial $RT_{NDT}$ values for the closure head and vessel flange were estimated per U.S. NRC Standard Review Plan [2]. These values are used for considering flange requirements for the heatup/cooldown curves[4].

(c) Initial $RT_{NDT}$ values are estimated per U.S. NRC Standard Review Plan [2]. For margin calculations for these materials, $\sigma_i$ = 30°F for forging and $\sigma_i$ = 20°F for welds per BAW-1911, Rev. 1, "Reactor Pressure Vessel and Surveillance Program Materials Licensing Information for North Anna Units 1 and 2", A. L. Lowe, dated August 1986.

Copper and nickel concentrations are taken from North Anna Unit 1 Table 7 "Beltline Materials" of BAW-2168, Revision 1, "Response to Generic Letter 92-01 for Virginia Electric and Power Company North Anna Units 1 and 2", dated September 1992.

TABLE 2

SUMMARY OF ADJUSTED REFERENCE TEMPERATURES (ART's)

AT 1/4-T and 3/4-T LOCATIONS FOR 30.7 EFPY

| **Material** | **1/4-T ART (°F)** | **3/4-T ART (°F)** |
|---|---|---|
| Upper Shell Forging 05 | 140.3 | 117.3 |
| Inter. Shell Forging 04 | 158.1 | 136.8 |
| Lower Shell Forging 03 | 215.2 (146.5) | 186.7 (128.3) |
| Circumferential Weld Seam, W04 | 137.5 (162.9)* | 119.1 (139.9)* |
| Weld Seam, W05A | 143.4 | 111.2 |
| Weld Seam, W05B | 82.4 | 65.4 |

ART numbers within ( ) are based on chemistry factors calculated using surveillance capsule data.

*These ART numbers were used to generate heatup and cooldown curves.

TABLE 3

CALCULATION OF ADJUSTED REFERENCE TEMPERATURES AT 30.7 EFPY FOR THE LIMITING NORTH ANNA UNIT 1 REACTOR VESSEL MATERIAL - CIRCUMFERENTIAL WELD SEAM

| Parameter | Regulatory Guide 1.99 -Revision 2 30.7 EFPY 1/4-T | 3/4-T |
|---|---|---|
| Chemistry Factor, CF (°F) | 93.09 | 93.09 |
| Fluence, f ($10^{19}$ n/cm$^2$)[a] | 2.492 | 0.992 |
| Fluence Factor, ff | 1.245 | 0.998 |
| $\Delta RT_{NDT}$ = CF x ff (°F) | 115.9 | 92.9 |
| Initial $RT_{NDT}$, I (°F) | 19 | 19 |
| Margin, M (°F) [b] | 28 | 28 |
| Revision 2 to Regulatory Guide 1.99 | | |
| Adjusted Reference Temperature, ART = Initial $RT_{NDT}$ + $\Delta RT_{NDT}$ + Margin | 162.9 | 139.9 |

(a) Fluence, f, is based upon $f_{surf}$ ($10^{19}$ n/cm$^2$, E>1 Mev) = 3.95 at end-of-license life (30.7 EFPY). The North Anna Unit 1 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, $\sigma_I$, is assumed to be 0°F since the initial $RT_{NDT}$ is a measured value. The standard deviation for $\Delta RT_{NDT}$ term, $\sigma_\Delta$, is 28°F for the weld, except that $\sigma_\Delta$ need not exceed 0.5 times the mean value of $\Delta RT_{NDT}$. $\sigma_\Delta$ is 14°F for the weld (half the value) when surveillance data is used.

TABLE 4

MAXIMUM TEMPERATURES DIFFERENCE BETWEEN RCS FLUID AND REACTOR VESSEL 1/4-T AND 3/4-T LOCATIONS FOR HEATUP AND COOLDOWN RATES

| Case | Rate °F/hr | Time (sec) | Water Temp. (°F) | 1/4-T Temp. (°F) | 3/4-T Temp. (°F) | ΔT @ 1/4-T (°F) | ΔT @ 3/4-T (°F) |
|---|---|---|---|---|---|---|---|
| Heatup | 60 | 28,800 | 550 | 535 | 519 | 15 | 31 |
| Cooldown | 100 | 5940 | 385 | 408 | 434 | 23 | ---- |
| | 100 | 6120 | 380 | 403 | 429 | ---- | 49 |

MATERIAL PROPERTY BASIS

LIMITING MATERIAL: CIRCUMFERENTIAL WELD SEAM

LIMITING ART AT 30.7 EFPY: 1/4-T, 162.9°F

3/4 39.9°F

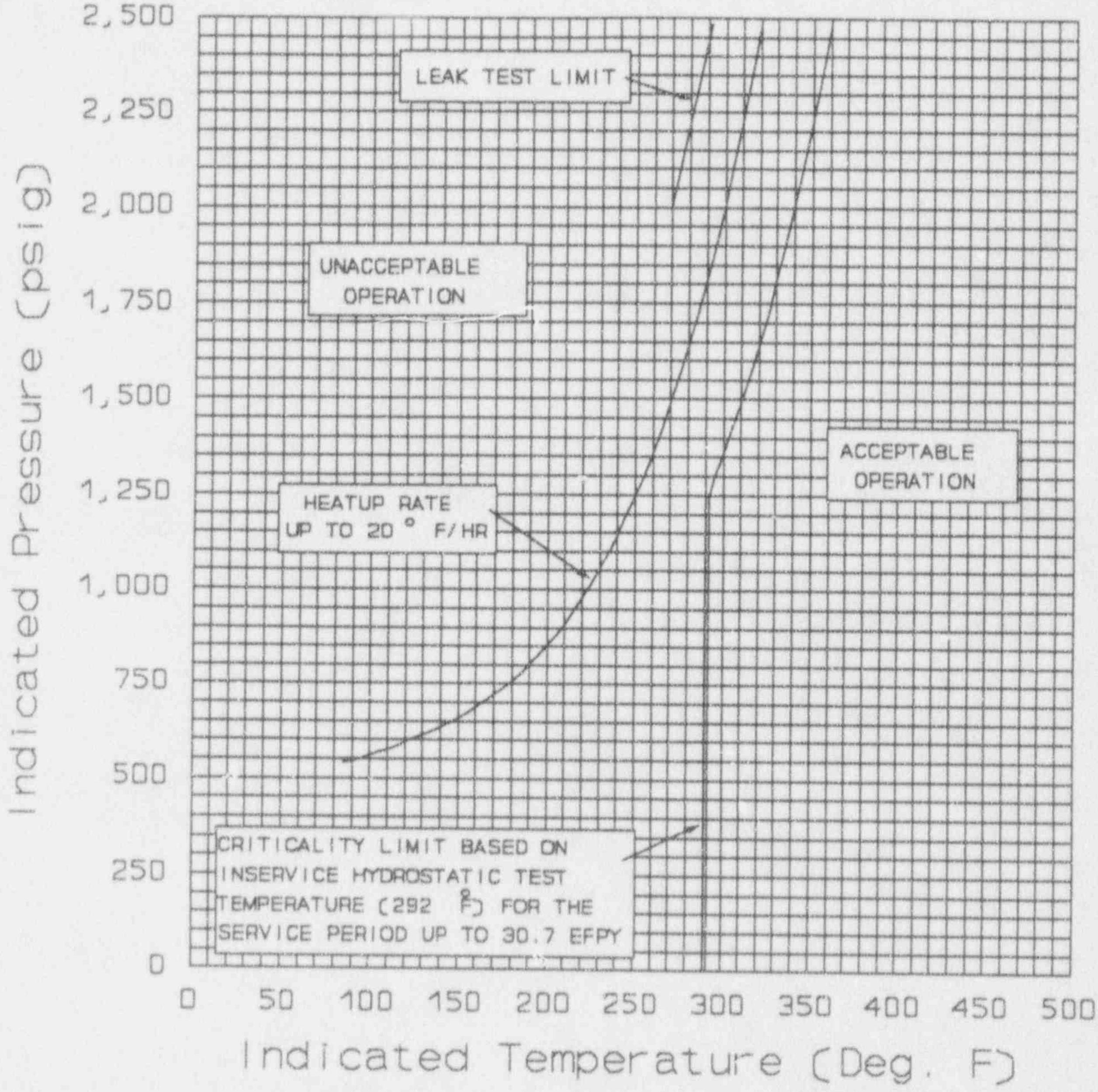


Figure 1. North Anna Unit 1 Reactor Coolant System Heatup Limitations (Heatup rate of 20°F/hr) Applicable for the First 30.7 EFPY (Without Margins for Instrumentation Errors)

MATERIAL PROPERTY BASIS

LIMITING MATERIAL: CIRCUMFERENTIAL WELD SEAM

LIMITING ART AT 30.7 EFPY: 1/4-T, 162.9°F

3/4-T, 139.9°F

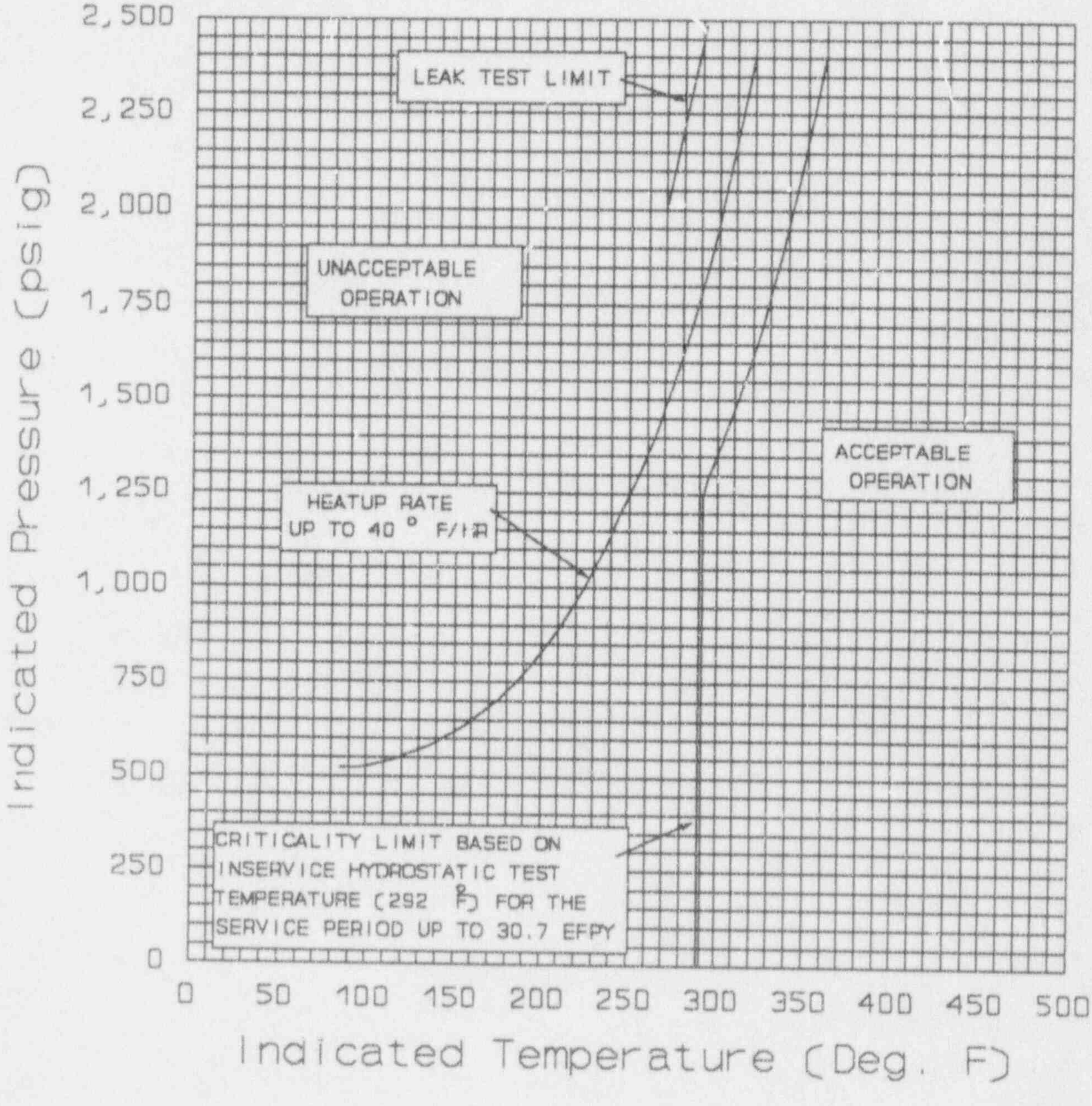


Figure 2. North Anna Unit 1 Reactor Coolant System Heatup Limitations (Heatup rate of 40°F/hr) Applicable for the First 30.7 EFPY (Without Margins for Instrumentation Errors)

MATERIAL PROPERTY BASIS

LIMITING MATERIAL: CIRCUMFERENTIAL WELD SEAM

LIMITING ART AT 30.7 EFPY: 1/4-T, 162.9°F

3/4-T, 139.9°F

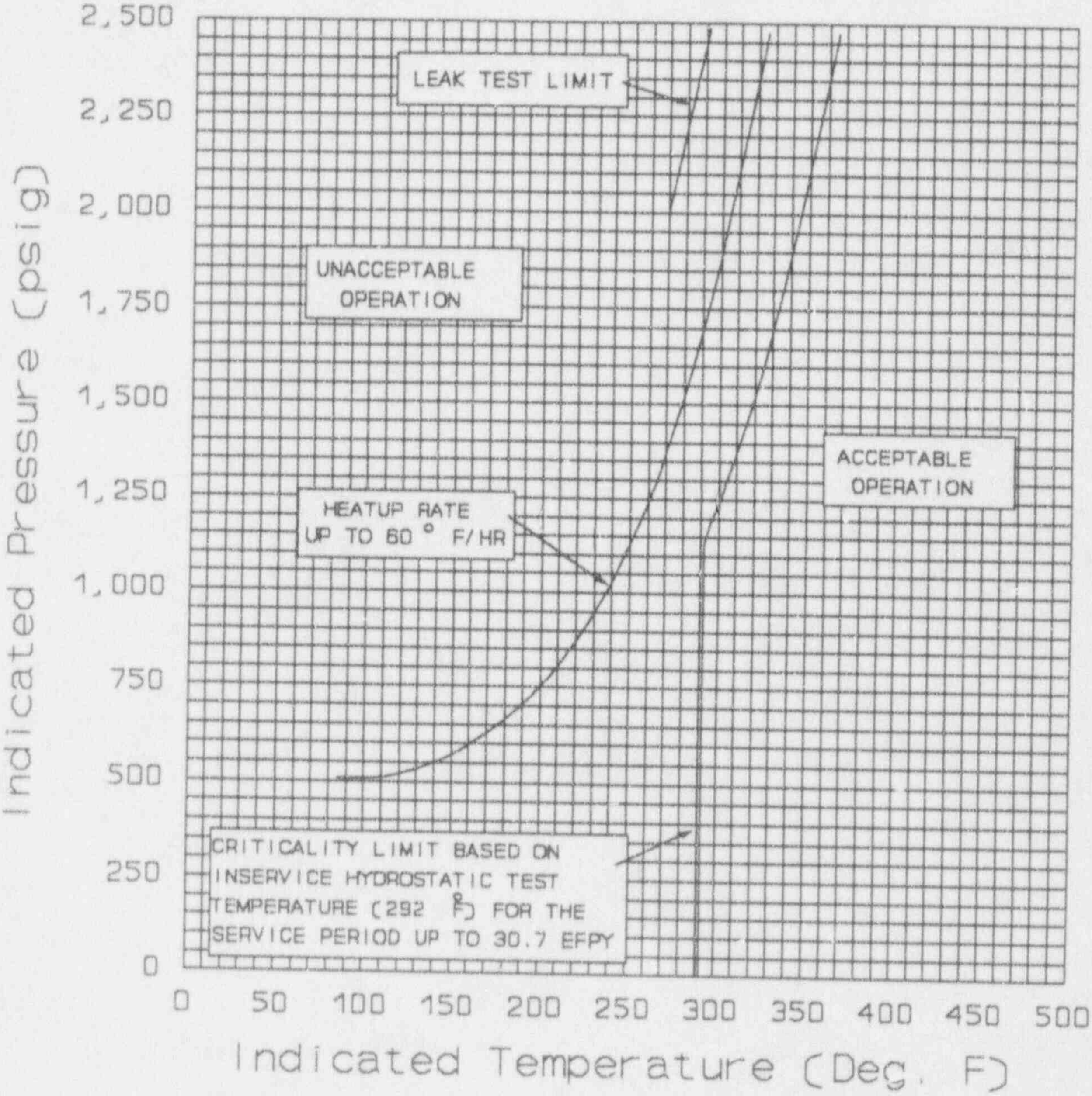


Figure 3. North Anna Unit 1 Reactor Coolant System Heatup Limitations (Heatup rate of 60°F/hr) Applicable for the First 30.7 EFPY (Without Margins for Instrumentation Errors)

MATERIAL PROPERTY BASIS

LIMITING MATERIAL: CIRCUMFERENTIAL WELD SEAM

LIMITING ART AT 30.7 EFPY: 1/4-T, 162.9°F

3/4-T, 139.9°F

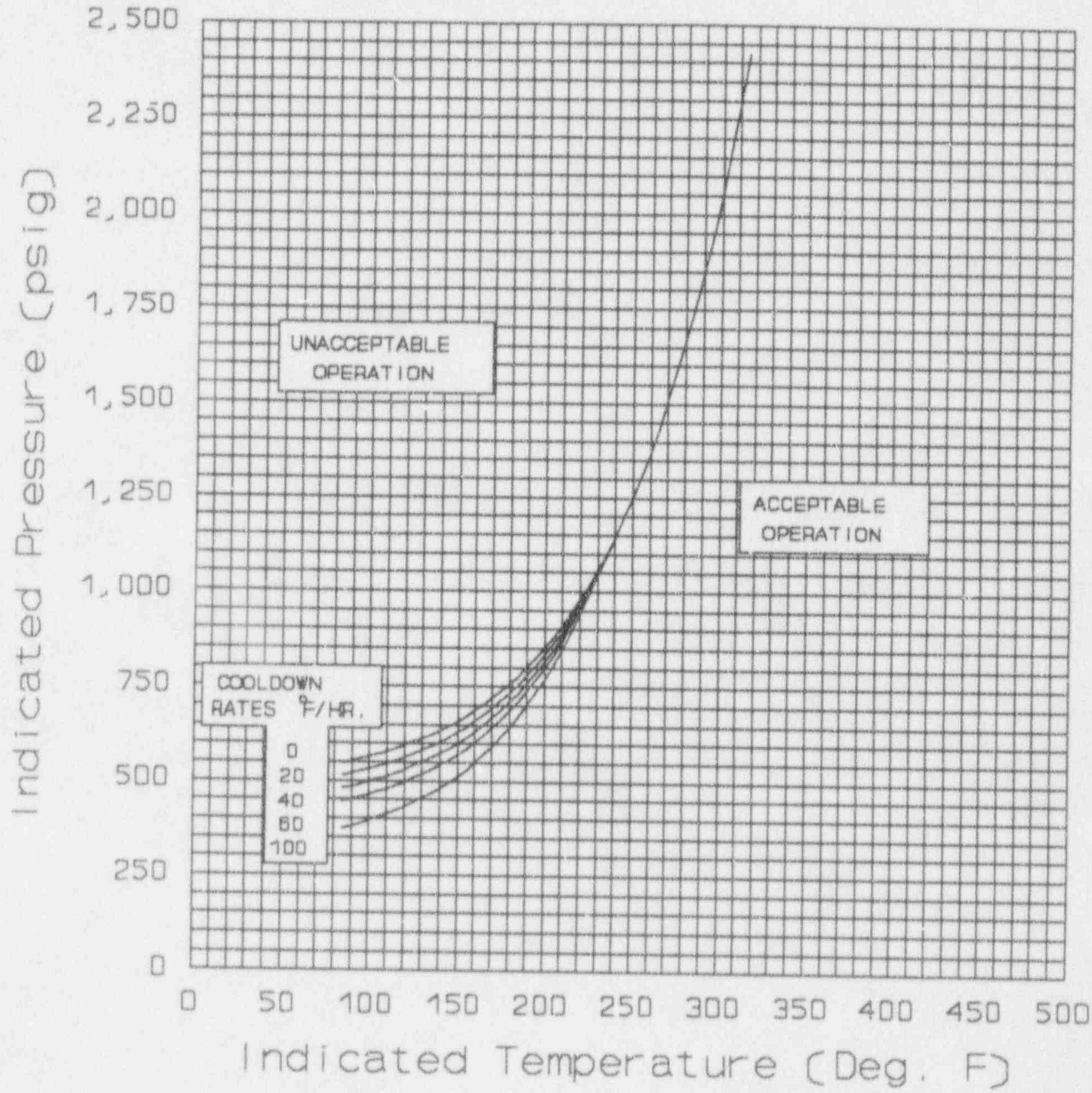


Figure 4. North Anna Unit 1 Reactor Coolant System Cooldown Limitations (Cooldown rates up to 100°F/hr) Applicable for the First 30.7 EFPY (Without Margins for Instrumentation Errors)

## 6. REFERENCES


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

[2] "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-0800, 1981.

[3] 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.

[4] 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.

[5] WCAP-8771, "Virginia Electric and Power Company North Anna Unit No. 1 Reactor Vessel Radiation Surveillance Program", J. A. Davidson and J. H. Phillips, September 1976. (Westinghouse Proprietary Class 3)

[6] WCAP-11777, "Analysis of Capsule U from the Virginia Electric and Power Company North Anna Unit 1 Reactor Vessel Radiation Surveillance Program", S. E. Yanichko, et al., February 1988. (Westinghouse Proprietary Class 3)

[7] Material Certification Report, Rotterdam Order No. 30661, Lab No. J 575.

# APPENDIX A

## DATA POINTS FOR HEATUP AND COOLDOWN CURVES

(Without Margins for Instrumentation Errors)

North Anna Unit 1 Heatup/Cooldown Curve Data WITHOUT margins

Cooldown Curves

| Steady State | | 20 DEG CD | | 40 DEG CD | | 60 DEG CD | | 100 DEG CD | |
|---|---|---|---|---|---|---|---|---|---|
| T | P | T | P | T | P | T | P | T | P |
| 85 | 547.28 | 85 | 513.64 | 85 | 479.44 | 85 | 444.59 | 85 | 373.22 |
| 90 | 552.65 | 90 | 519.15 | 90 | 485.12 | 90 | 450.47 | 90 | 379.55 |
| 95 | 558.42 | 95 | 525.11 | 95 | 491.20 | 95 | 456.85 | 95 | 386.47 |
| 100 | 564.63 | 100 | 531.51 | 100 | 497.82 | 100 | 463.73 | 100 | 393.94 |
| 105 | 571.30 | 105 | 538.32 | 105 | 505.00 | 105 | 471.20 | 105 | 402.08 |
| 110 | 578.47 | 110 | 545.75 | 110 | 512.72 | 110 | 479.24 | 110 | 410.82 |
| 115 | 586.06 | 115 | 553.77 | 115 | 521.07 | 115 | 487.96 | 115 | 420.38 |
| 120 | 594.35 | 120 | 562.38 | 120 | 530.05 | 120 | 497.26 | 120 | 430.71 |
| 125 | 603.26 | 125 | 571.68 | 125 | 539.66 | 125 | 507.42 | 125 | 441.85 |
| 130 | 612.85 | 130 | 581.66 | 130 | 550.10 | 130 | 518.37 | 130 | 453.95 |
| 135 | 623.15 | 135 | 592.32 | 135 | 561.39 | 135 | 530.21 | 135 | 467.07 |
| 140 | 634.08 | 140 | 603.90 | 140 | 573.52 | 140 | 542.85 | 140 | 481.22 |
| 145 | 645.99 | 145 | 616.38 | 145 | 586.50 | 145 | 556.63 | 145 | 496.47 |
| 150 | 658.79 | 150 | 629.77 | 150 | 600.59 | 150 | 571.46 | 150 | 513.00 |
| 155 | 672.54 | 155 | 644.11 | 155 | 615.80 | 155 | 587.36 | 155 | 530.89 |
| 160 | 687.19 | 160 | 659.65 | 160 | 632.00 | 160 | 604.60 | 160 | 550.08 |
| 165 | 703.09 | 165 | 676.35 | 165 | 649.64 | 165 | 623.22 | 165 | 570.94 |
| 170 | 720.16 | 170 | 694.21 | 170 | 668.61 | 170 | 643.11 | 170 | 593.30 |
| 175 | 738.39 | 175 | 713.59 | 175 | 688.90 | 175 | 664.74 | 175 | 617.60 |
| 180 | 758.16 | 180 | 734.23 | 180 | 710.90 | 180 | 687.84 | 180 | 643.63 |
| 185 | 779.19 | 185 | 756.66 | 185 | 734.42 | 185 | 712.94 | 185 | 671.90 |
| 190 | 802.03 | 190 | 780.55 | 190 | 759.89 | 190 | 739.76 | 190 | 702.20 |
| 195 | 826.34 | 195 | 806.48 | 195 | 787.16 | 195 | 768.82 | 195 | 734.90 |
| 200 | 852.71 | 200 | 834.16 | 200 | 816.63 | 200 | 799.99 | 200 | 770.25 |
| 205 | 880.84 | 205 | 864.11 | 205 | 848.28 | 205 | 833.52 | 205 | 808.29 |
| 210 | 911.22 | 210 | 896.19 | 210 | 882.22 | 210 | 869.58 | 210 | 849.22 |
| 215 | 943.79 | 215 | 930.62 | 215 | 918.77 | 215 | 908.64 | 215 | 893.33 |
| 220 | 978.72 | 220 | 967.62 | 220 | 958.26 | 220 | 950.46 | 220 | 940.81 |
| 225 | 1016.23 | 225 | 1007.61 | 225 | 1000.60 | 225 | 995.48 | 225 | 991.95 |
| 230 | 1056.54 | 230 | 1050.44 | 230 | 1046.08 | 230 | 1043.85 | 230 | 1046.95 |
| 235 | 1100.08 | 235 | 1096.47 | 235 | 1094.99 | 235 | 1095.93 | | |
| 240 | 1146.64 | 240 | 1145.88 | | | | | | |
| 245 | 1196.67 | | | | | | | | |
| 250 | 1250.15 | | | | | | | | |
| 255 | 1307.90 | | | | | | | | |
| 260 | 1369.68 | | | | | | | | |
| 265 | 1435.87 | | | | | | | | |
| 270 | 1507.19 | | | | | | | | |
| 275 | 1583.43 | | | | | | | | |
| 280 | 1664.94 | | | | | | | | |
| 285 | 1752.74 | | | | | | | | |
| 290 | 1846.59 | | | | | | | | |
| 295 | 1946.83 | | | | | | | | |
| 300 | 2054.44 | | | | | | | | |
| 305 | 2169.27 | | | | | | | | |
| 310 | 2292.07 | | | | | | | | |
| 315 | 2423.12 | | | | | | | | |

North Anna Unit 1 Heatup/Cooldown Curve Data WITHOUT margins

Heatup Curves

| 20 DEG HU | | Criticality Limit | | 40 DEG HU | | Criticality Limit | |
|---|---|---|---|---|---|---|---|
| T | P | T | P | T | P | T | P |
| 85 | 541.05 | 292 | 0 | 85 | 523.08 | 292 | 0 |
| 90 | 544.65 | 292 | 544.65 | 90 | 523.08 | 292 | 523.08 |
| 95 | 551.19 | 292 | 551.19 | 95 | 523.91 | 272 | 523.91 |
| 100 | 558.02 | 292 | 558.02 | 100 | 526.71 | 292 | 526.71 |
| 105 | 566.12 | 292 | 566.12 | 105 | 531.26 | 292 | 531.26 |
| 110 | 574.67 | 292 | 574.67 | 110 | 536.96 | 292 | 536.96 |
| 115 | 583.99 | 292 | 583.99 | 115 | 543.97 | 292 | 543.97 |
| 120 | 594.05 | 292 | 594.05 | 120 | 551.92 | 292 | 551.92 |
| 125 | 603.26 | 292 | 603.26 | 125 | 560.88 | 292 | 560.88 |
| 130 | 612.85 | 292 | 612.85 | 130 | 570.68 | 292 | 570.68 |
| 135 | 623.15 | 292 | 623.15 | 135 | 581.44 | 292 | 581.44 |
| 140 | 634.08 | 292 | 634.08 | 140 | 592.98 | 292 | 592.98 |
| 145 | 645.99 | 292 | 645.99 | 145 | 605.64 | 292 | 605.64 |
| 150 | 658.79 | 292 | 658.79 | 150 | 619.27 | 292 | 619.27 |
| 155 | 672.54 | 292 | 672.54 | 155 | 633.87 | 292 | 633.87 |
| 160 | 687.19 | 292 | 687.19 | 160 | 649.71 | 292 | 649.71 |
| 165 | 703.09 | 292 | 703.09 | 165 | 666.81 | 292 | 666.81 |
| 170 | 720.16 | 292 | 720.16 | 170 | 685.00 | 292 | 685.00 |
| 175 | 738.39 | 292 | 738.39 | 175 | 704.79 | 292 | 704.79 |
| 180 | 758.16 | 292 | 758.16 | 180 | 725.84 | 292 | 725.84 |
| 185 | 779.19 | 292 | 779.19 | 185 | 748.72 | 292 | 748.72 |
| 190 | 802.03 | 292 | 802.03 | 190 | 773.22 | 292 | 773.22 |
| 195 | 826.34 | 292 | 826.34 | 195 | 799.50 | 292 | 799.50 |
| 200 | 852.71 | 292 | 852.71 | 200 | 827.64 | 292 | 827.64 |
| 205 | 880.84 | 292 | 880.84 | 205 | 858.12 | 292 | 858.12 |
| 210 | 911.22 | 292 | 911.22 | 210 | 890.69 | 292 | 890.69 |
| 215 | 943.79 | 292 | 943.79 | 215 | 925.64 | 292 | 925.64 |
| 220 | 978.72 | 292 | 978.72 | 220 | 963.17 | 292 | 963.17 |
| 225 | 1016.23 | 292 | 1016.23 | 225 | 1003.73 | 292 | 1003.73 |
| 230 | 1056.54 | 292 | 1056.54 | 230 | 1047.10 | 292 | 1047.10 |
| 235 | 1100.08 | 292 | 1100.08 | 235 | 1093.62 | 292 | 1093.62 |
| 240 | 1146.64 | 292 | 1146.64 | 240 | 1143.61 | 292 | 1143.61 |
| 245 | 1196.62 | 292 | 1196.62 | 245 | 1196.62 | 292 | 1196.62 |
| 250 | 1247.04 | 292 | 1247.04 | 250 | 1246.41 | 292 | 1246.41 |
| 255 | 1301.47 | 295 | 1301.47 | 255 | 1297.66 | 295 | 1297.66 |
| 260 | 1359.75 | 300 | 1359.75 | 260 | 1352.70 | 300 | 1352.70 |
| 265 | 1422.22 | 305 | 1422.22 | 265 | 1411.55 | 305 | 1411.55 |
| 270 | 1489.11 | 310 | 1489.11 | 270 | 1474.75 | 310 | 1474.75 |
| 275 | 1561.00 | 315 | 1561.00 | 275 | 1542.37 | 315 | 1542.37 |
| 280 | 1637.87 | 320 | 1637.87 | 280 | 1614.86 | 320 | 1614.86 |
| 285 | 1720.33 | 325 | 1720.33 | 285 | 1692.59 | 325 | 1692.59 |
| 290 | 1808.46 | 330 | 1808.46 | 290 | 1775.78 | 330 | 1775.78 |
| 295 | 1903.06 | 335 | 1903.06 | 295 | 1864.70 | 335 | 1864.70 |
| 300 | 2004.25 | 340 | 2004.25 | 300 | 1960.06 | 340 | 1960.06 |
| 305 | 2112.48 | 345 | 2112.48 | 305 | 2061.90 | 345 | 2061.90 |
| 310 | 2227.78 | 350 | 2227.78 | 310 | 2170.71 | 350 | 2170.71 |
| 315 | 2351.31 | 355 | 2351.31 | 315 | 2286.98 | 355 | 2286.98 |
| 320 | 2482.97 | 360 | 2482.97 | 320 | 2410.81 | 360 | 2410.81 |

North Anna Unit 1 Heatup/Cooldown Curve Data WITHOUT margins

| Heatup Curves | | | | Leak Test Data (for all heatup curves) | |
|---|---|---|---|---|---|
| 60 DEG HU | | Criticality Limit | | | |
| T | P | T | P | T | P |
| 85 | 507.18 | 292 | 0 | 270 | 2000 |
| 90 | 507.18 | 292 | 507.18 | 292 | 2485 |
| 95 | 507.18 | 292 | 507.18 | | |
| 100 | 507.18 | 292 | 507.18 | | |
| 105 | 507.82 | 292 | 507.82 | | |
| 110 | 510.04 | 292 | 510.04 | | |
| 115 | 513.81 | 292 | 513.81 | | |
| 120 | 518.78 | 292 | 518.78 | | |
| 125 | 525.00 | 292 | 525.00 | | |
| 130 | 532.25 | 292 | 532.25 | | |
| 135 | 540.50 | 292 | 540.50 | | |
| 140 | 549.81 | 292 | 549.81 | | |
| 145 | 560.20 | 292 | 560.20 | | |
| 150 | 571.56 | 292 | 571.56 | | |
| 155 | 583.91 | 292 | 583.91 | | |
| 160 | 597.43 | 292 | 597.43 | | |
| 165 | 612.13 | 292 | 612.13 | | |
| 170 | 628.00 | 292 | 628.00 | | |
| 175 | 645.05 | 292 | 645.05 | | |
| 180 | 663.57 | 292 | 663.57 | | |
| 185 | 683.38 | 292 | 683.38 | | |
| 190 | 704.87 | 292 | 704.87 | | |
| 195 | 727.84 | 292 | 727.84 | | |
| 200 | 752.74 | 292 | 752.74 | | |
| 205 | 779.33 | 292 | 779.33 | | |
| 210 | 808.11 | 292 | 808.11 | | |
| 215 | 838.87 | 292 | 838.87 | | |
| 220 | 871.89 | 292 | 871.89 | | |
| 225 | 907.59 | 292 | 907.59 | | |
| 230 | 945.78 | 292 | 945.78 | | |
| 235 | 986.80 | 292 | 986.80 | | |
| 240 | 1030.80 | 292 | 1030.80 | | |
| 245 | 1078.06 | 292 | 1078.06 | | |
| 250 | 1128.77 | 292 | 1128.77 | | |
| 255 | 1183.17 | 295 | 1183.17 | | |
| 260 | 1241.50 | 300 | 1241.50 | | |
| 265 | 1304.05 | 305 | 1304.05 | | |
| 270 | 1371.17 | 310 | 1371.17 | | |
| 275 | 1443.05 | 315 | 1443.05 | | |
| 280 | 1520.15 | 320 | 1520.15 | | |
| 285 | 1602.65 | 325 | 1602.65 | | |
| 290 | 1691.00 | 330 | 1691.00 | | |
| 295 | 1785.48 | 335 | 1785.48 | | |
| 300 | 1886.69 | 340 | 1886.69 | | |
| 305 | 1994.52 | 345 | 1994.52 | | |
| 310 | 2110.10 | 350 | 2110.10 | | |
| 315 | 2229.35 | 355 | 2229.35 | | |
| 320 | 2346.49 | 360 | 2346.49 | | |
| 325 | 2471.30 | 365 | 2471.30 | | |

# APPENDIX B

# PRESSURIZED THERMAL SHOCK EVALUATION RESULTS

A limiting condition on reactor vessel integrity known as Pressurized Thermal Shock (PTS) may occur during a severe system transient such as a Loss-Of-Coolant-Accident (LOCA) or a steam line break. Such transients may challenge the integrity of a reactor vessel under the following conditions:

- severe overcooling of the inside surface of the vessel wall followed by high repressurization;
- significant degradation of vessel material toughness caused by radiation embrittlement; and
- the presence of a critical-size defect in the vessel wall.

In 1985 the Nuclear Regulatory Commission (NRC) issued a formal ruling on PTS. It established screening criteria on pressurized water reactor (PWR) vessel embrittlement as measured by the nil-ductility reference temperature, termed $RT_{PTS}$[B1]. $RT_{PTS}$ screening values were set for beltline axial welds, forging or plates and for beltline circumferential weld seams for the end-of-license plant operation. The screening criteria were determined using conservative fracture mechanics analysis techniques. All PWR vessels in the United States have been required to evaluate vessel embrittlement in accordance with the criteria through end-of-license. The NRC has amended its regulations for light water nuclear power plants to change the procedure for calculating radiation embrittlement. The revised PTS Rule was published in the Federal Register, May 15, 1991 with an effective date of June 14, 1991[B2]. This amendment makes the procedure for calculating $RT_{PTS}$ values consistent with the methods given in Regulatory Guide 1.99, Revision 2[B3].

The PTS Rule requires that the PTS submittal be updated whenever there are changes in core loadings, surveillance measurements or other information that indicates a significant change in projected $RT_{PTS}$ values.

The Rule outlines regulations to address the potential for PTS events on pressurized water reactor vessels in nuclear power plants that are operated with a license from the United States Nuclear Regulatory Commission (USNRC). PTS events have been shown from operating experience to be transients that result in a rapid and severe cooldown in the primary system coincident with a high or increasing primary system pressure. The PTS concern arises if one of these transients acts on the beltline region of a reactor vessel where a reduced fracture resistance exists because of neutron irradiation. Such an event may result in the propagation of flaws postulated to exist near the inner wall surface, thereby potentially affecting the integrity of the vessel.

The Rule establishes the following requirements for all domestic, operating PWRs:

All plants must submit projected values of $RT_{PTS}$ for reactor vessel beltline materials by giving values for time of submittal, the expiration date of the operating license, and the projected expiration date if a change in the operating license or renewal has been requested. This assessment must be submitted within six months after the effective date of this Rule if the value of $RT_{PTS}$ for any material is projected to exceed the screening criteria. Otherwise, it must be submitted with the next update of the pressure-temperature limits, or the next reactor vessel surveillance capsule report, or within 5 years from the effective date of this Rule change, whichever comes first. These values must be calculated based on the methodology specified in this rule. The submittal must include the following:

1) the bases for the projection (including any assumptions regarding core loading patterns), and
2) copper and nickel content and fluence values used in the calculations for each beltline material. (If these values differ from those previously submitted to the NRC, justification must be provided.)

Using the prescribed PTS Rule methodology, $RT_{PTS}$ values were generated for all beltline region materials of the North Anna Unit 1 reactor vessel as a function of present time (11 EFPY) and end-of-life (30.7 EFPY) fluence values.

Table B-1 provides a summary of the $RT_{PTS}$ values for all beltline region materials for 11 EFPY and end-of-license (30.7 EFPY), using the PTS Rule. As shown in Table B-1 all the $RT_{PTS}$ values remain below the NRC screening values for PTS using the fluence values for the present time (11 EFPY) and the projected fluence values for the end-of-license (30.7 EFPY).

TABLE B-1

NORTH ANNA UNIT 1 $RT_{PTS}$ VALUES (°F) FOR 11 EFPY AND 30.7 EFPY

| MATERIAL | 11 EFPY | 30.7 EFPY |
|---|---|---|
| Upper Shell Forging 05 | 105.2 | 132.9 |
| Intermediate Shell Forging 04 | 146.0 | 167.4 |
| Lower Shell Forging 03 | 199.1 (153.2) | 227.7 (171.5) |
| Circumferential Weld Seam, W04 | 140.9 (177.9) | 155.7 (201.0) |
| Weld Seam, W05A | 124.3 | 155.9 |
| Weld Seam, W05B | 90.6 | 104.0 |

Note values in ( ) were calculated based upon surveillance capsule data.

## REFERENCES


[B1] 10CFR Part 50, "Analysis of Potential Pressurized Thermal Shock Events," July 23, 1985.

[B2] 10CFR Part 50.61, "Fracture Toughness Requirements for Protection Against Pressurized Thermal Shock Events," May 15, 1991. (PTS Rule)

[B3] Regulatory Guide 1.99, Revision 2, "Radiation Embrittlement of Reactor Vessel Materials," U.S. Nuclear Regulatory Commission, May 1988.