ENCLOSURE 4

TENNESSEE VALLEY AUTHORITY
SEQUOYAH NUCLEAR PLANT (SQN)
UNITS 1 AND 2

PRESSURE TEMPERATURE LIMITS REPORTS (PTLR)

sponsibility for the correctness of design, details and dimensions.
Letter No. N10004
Date: June 06, 2002
TENNESSEE VALLEY AUTHORITY
SOEP (N) BY D. L. Lundy

# Tennessee Valley Authority

# Sequoyah Unit 1

# Pressure Temperature Limits Report

# Revision 3, May 2002

PROJECT Sequoyah DISCIPLINE N
CONTRACT 91NNP-86305B UNIT 1
DESC. RCS Pressure-Temperature Limit Report
DWG/DOC NO. PTLR-1
SHEET - OF - REV 03
DATE 06/06/02 ECN/DCN - FILE N2N-048

RIMS, WTC A-K

This report affects TS 3.4.9.1, RCS Pressure/Temperature Limits (P/T) Limits. All TS requirements associated with Low Temperature Overpressure Protection System (LTOPS) are contained in TS 3.4.12, RCS Overpressure Protection System.

## 2.0 RCS Pressure and Temperature Limits

The limits for TS 3.4.9.1 are presented in the subsection which follow and were developed using the NRC approved methodologies specified in TS 6.9.1.15 with exception of ASME Code Case N-640[14] (Use of $K_{Ic}$), WCAP-15315[15] (Elimination of the Flange Requirement), 1996 Version of Appendix G[4] and the revised fluences[7]. The operability requirements associated with LTOPS are specified in TS LCO 3.4.12 and were determined to adequately protect the RCS against brittle fracture in the event of an LTOP Transient in accordance with the methodology specified in TS 6.9.1.15.

2.1 RCS Pressure/Temperature (P/T) Limits (LCO - 3.4.9.1)

2.1.1 The minimum boltup temperature is 50°F

2.1.2 The RCS temperature rate-of-change limits are:

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

2.1.3 The RCD P/T limits for heatup, cooldown, inservice hydrostatic and leak testing, and criticality are specified by Figures 2-1 and 2-2.

## 3.0 Low Temperature Overpressure Protection System (LCO 3.4.12)

The lift setpoints for the pressurizer Power Operated Relief Valves (PORVs) are presented in the subsection which follow. These lift setpoints have been developed using the NRC-approved methodologies specified in Specification 3.4.12.

reactor midplane/beltline or for instrument inaccuracies. The pressure difference between the pressurizer transmitter and the reactor vessel midplane/beltline with four reactor coolant pumps in operation is 68.3 psi
(Ref. 12).

Note: These setpoints include allowance for the 50°F thermal transport effect for heat injection transients. A demonstrated accuracy calculation (Reference 13) has been performed to confirm that the setpoints will maintain the system pressure within the established limits when the pressure difference between the pressure transmitter and reactor midplane and maximum temperature/pressure instrument uncertainties are applied to the setpoints.

### 3.2 Arming Temperature

The LTOPS arming temperature is based upon the methodology defined in the Sequoyah Nuclear Plant Unit 1 Technical Specifications Administrative Controls Section 6.9.1.15. The arming temperature shall be ≤ 350°F.

## 4.0 Reactor Vessel Material Surveillance Program

The reactor vessel material irradiation surveillance specimens shall be removed and examined to determine changes in material properties. The removal schedule is provided in Table 4-1. The results of these examinations shall be used to update Figures 2-1, 2-2 and 3-1.

The pressure vessel steel surveillance program (WCAP-8233[1]) is in compliance with Appendix H to 10 CFR 50, "Reactor Vessel Material Surveillance Program Requirements"[2]. The material test requirements and the acceptance standard utilize the reference nil-ductility temperature $RT_{NDT}$, which is determined in accordance with ASTM E208[3]. The empirical relationship between $RT_{NDT}$ and the fracture toughness of the reactor vessel steel is developed in accordance with Code Case N-640 of Section XI of the ASME Boiler and Pressure Vessel Code, Appendix G, "Fracture Toughness Criteria for Protection Against Failure"[4]. The surveillance capsule removal schedule meets the requirements of ASTM E185-82[5]. The removal schedule is provided in Table 4-1.

Table 5-2 shows calculations of the surveillance material chemistry factors using surveillance capsule data. Note that in the calculation of the surveillance weld chemistry factor, the ratio procedure from Regulatory Guide 1.99, Revision 2 was followed. The ratio in question is equal to 0.90.

Table 5-3 provides the required Sequoyah Unit 1 reactor vessel toughness data.

Table 5-4 provides a summary of the fluence values used in the generation of the heatup and cooldown limit curves and the PTS evaluation.

Table 5-5 and 5-6 show the calculation of the 1/4T and 3/4T adjusted reference temperature at 32 EFPY for each beltline material in the Sequoyah Unit 1 reactor vessel. The limiting beltline material was the lower shell 04.

Table 5-7 provides a summary of the adjusted reference temperature (ART) values of the Sequoyah Unit 1 reactor vessel beltline materials at the 1/4T and 3/4T locations for 32 EFPY.

Table 5-8 provides $RT_{PTS}$ values for Sequoyah Unit 1 at 32 EFPY.

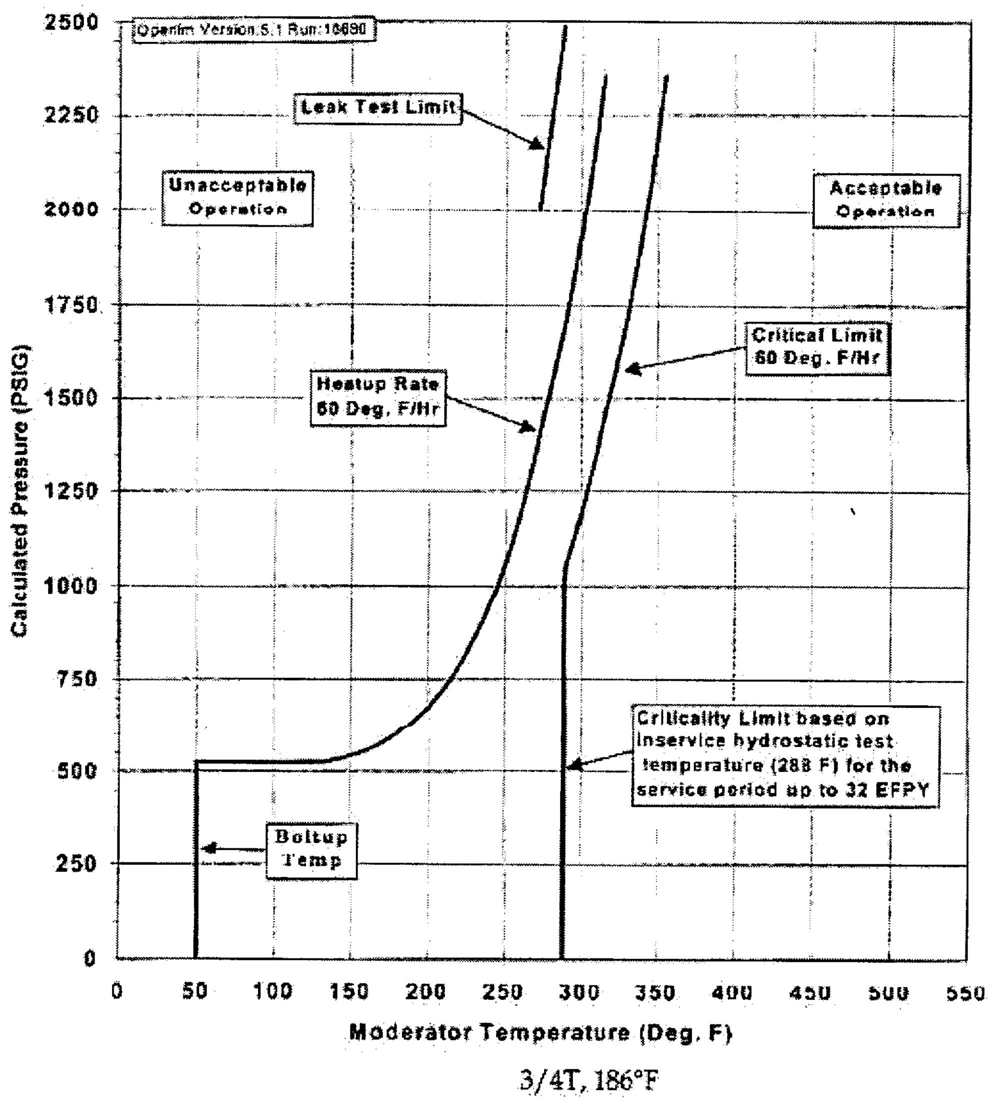
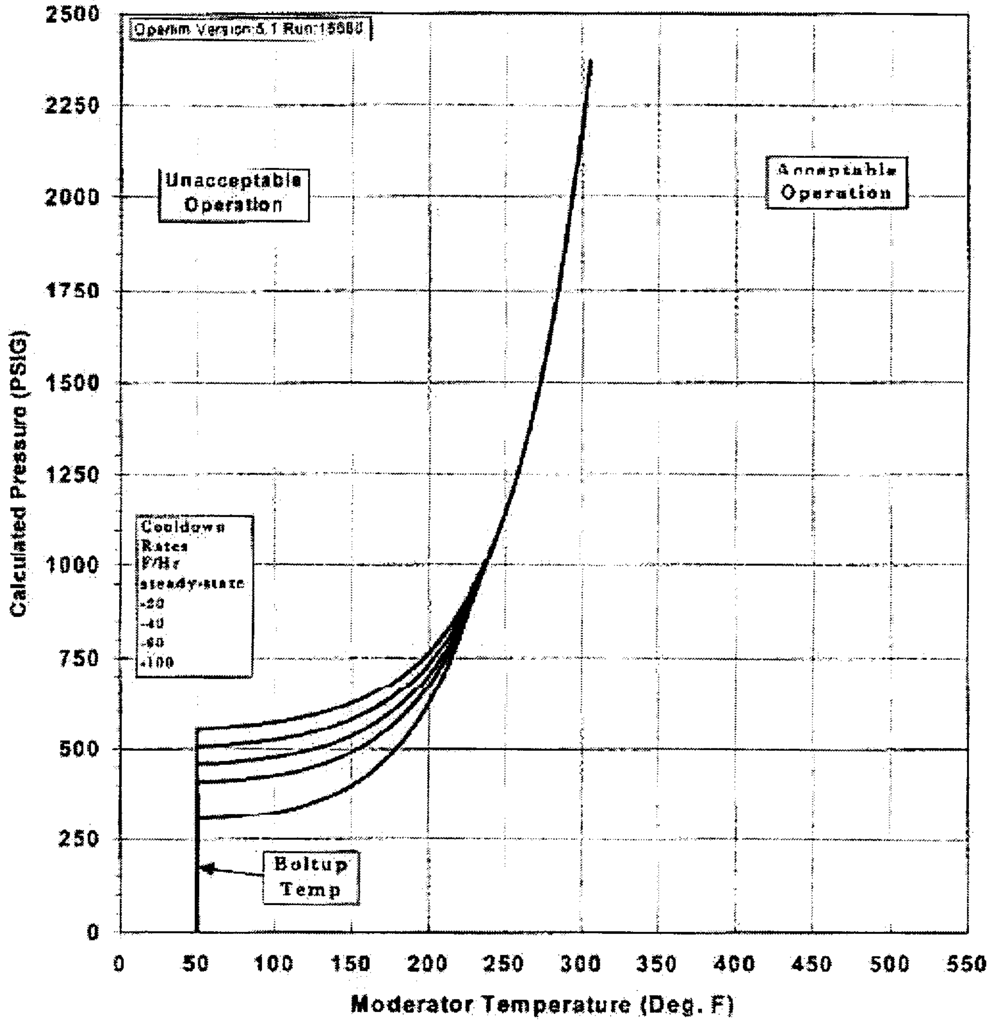



Figure 2-1 Sequoyah Unit 1 Reactor Coolant System Heatup Limitations (Heatup Rate of 60°F/hr) Applicable for the First 32 EFPY (w/Margins for Instrumentation Error of 10°F and 60 psig)

*(Plotted Data provided on Table 2-1)*




3/4T, 186°F

Figure 2-2 Sequoyah Unit 1 Reactor Coolant System Cooldown Limitations (Cooldown Rates up to 100°F/hr) Applicable for the First 32 EFPY (w/Margins for Instrumentation Error of 10°F and 60 psig) *(Plotted Data provided on Table 2-2)*

| T | P | T | P | T | P |
|---|---|---|---|---|---|
| 50 | 0 | 288 | 0 | 272 | 2000 |
| 50 | 525 | 288 | 525 | 288 | 2485 |
| 55 | 525 | 288 | 525 | | |
| 60 | 525 | 288 | 525 | | |
| 65 | 525 | 288 | 525 | | |
| 70 | 525 | 288 | 526 | | |
| 75 | 525 | 288 | 528 | | |
| 105 | 525 | 288 | 557 | | |
| 110 | 525 | 288 | 566 | | |
| 115 | 525 | 288 | 577 | | |
| 120 | 525 | 288 | 589 | | |
| 125 | 525 | 288 | 603 | | |
| 130 | 526 | 288 | 618 | | |
| 135 | 528 | 288 | 635 | | |
| 140 | 531 | 288 | 654 | | |
| 145 | 536 | 288 | 675 | | |
| 150 | 542 | 288 | 698 | | |
| 155 | 549 | 288 | 724 | | |
| 160 | 557 | 288 | 752 | | |
| 165 | 566 | 288 | 784 | | |
| 170 | 577 | 288 | 819 | | |
| 175 | 589 | 288 | 858 | | |
| 180 | 603 | 288 | 900 | | |
| 185 | 618 | 288 | 948 | | |
| 190 | 635 | 288 | 1000 | | |
| 195 | 654 | 290 | 1057 | | |
| 200 | 675 | 295 | 1121 | | |
| 205 | 698 | 300 | 1191 | | |
| 210 | 724 | 305 | 1269 | | |
| 215 | 752 | 310 | 1354 | | |
| 220 | 784 | 315 | 1449 | | |
| 225 | 819 | 320 | 1531 | | |
| 230 | 858 | 325 | 1617 | | |
| 235 | 900 | 330 | 1712 | | |
| 240 | 948 | 335 | 1817 | | |
| 245 | 1000 | 340 | 1933 | | |
| 250 | 1057 | 345 | 2060 | | |

| T | P | T | P | |
|---|---|---|---|---|
| 260 | 1191 | 355 | 2357 | |
| 265 | 1269 | | | |
| 270 | 1354 | | | |
| 275 | 1449 | | | |
| 280 | 1531 | | | |
| 285 | 1617 | | | |
| 290 | 1712 | | | |
| 295 | 1817 | | | |
| 300 | 1933 | | | |
| 305 | 2060 | | | |
| 310 | 2201 | | | |
| 315 | 2357 | | | |

| T | P | T | P | T | P | T | P | T | P |
|---|---|---|---|---|---|---|---|---|---|
| 50 | 0 | 50 | 0 | 50 | 0 | 50 | 0 | 50 | 0 |
| 50 | 552 | 50 | 503 | 50 | 457 | 50 | 408 | 50 | 305 |
| 55 | 553 | 55 | 505 | 55 | 458 | 55 | 409 | 55 | 306 |
| 60 | 555 | 60 | 507 | 60 | 459 | 60 | 410 | 60 | 307 |
| 65 | 556 | 65 | 509 | 65 | 460 | 65 | 411 | 65 | 308 |
| 70 | 558 | 70 | 510 | 70 | 462 | 70 | 412 | 70 | 309 |
| 75 | 560 | 75 | 512 | 75 | 464 | 75 | 414 | 75 | 311 |
| 80 | 561 | 80 | 514 | 80 | 465 | 80 | 416 | 80 | 313 |
| 85 | 564 | 85 | 516 | 85 | 468 | 85 | 418 | 85 | 315 |
| 90 | 566 | 90 | 518 | 90 | 470 | 90 | 420 | 90 | 318 |
| 95 | 569 | 95 | 521 | 95 | 473 | 95 | 423 | 95 | 321 |
| 100 | 571 | 100 | 524 | 100 | 476 | 100 | 426 | 100 | 325 |
| 105 | 575 | 105 | 527 | 105 | 479 | 105 | 430 | 105 | 329 |
| 110 | 578 | 110 | 531 | 110 | 483 | 110 | 434 | 110 | 333 |
| 115 | 582 | 115 | 535 | 115 | 487 | 115 | 438 | 115 | 338 |
| 120 | 586 | 120 | 540 | 120 | 492 | 120 | 443 | 120 | 344 |
| 125 | 591 | 125 | 545 | 125 | 497 | 125 | 449 | 125 | 351 |
| 130 | 596 | 130 | 550 | 130 | 503 | 130 | 456 | 130 | 358 |
| 135 | 602 | 135 | 556 | 135 | 510 | 135 | 463 | 135 | 367 |
| 140 | 608 | 140 | 563 | 140 | 517 | 140 | 471 | 140 | 376 |
| 145 | 616 | 145 | 571 | 145 | 525 | 145 | 479 | 145 | 387 |
| 150 | 623 | 150 | 579 | 150 | 534 | 150 | 489 | 150 | 399 |
| 155 | 632 | 155 | 588 | 155 | 544 | 155 | 500 | 155 | 412 |
| 160 | 642 | 160 | 599 | 160 | 556 | 160 | 512 | 160 | 427 |
| 165 | 652 | 165 | 610 | 165 | 568 | 165 | 526 | 165 | 443 |
| 170 | 664 | 170 | 623 | 170 | 582 | 170 | 541 | 170 | 461 |
| 175 | 677 | 175 | 637 | 175 | 597 | 175 | 558 | 175 | 482 |
| 180 | 691 | 180 | 652 | 180 | 614 | 180 | 577 | 180 | 505 |
| 185 | 707 | 185 | 669 | 185 | 633 | 185 | 597 | 185 | 530 |
| 190 | 724 | 190 | 688 | 190 | 654 | 190 | 620 | 190 | 558 |
| 195 | 743 | 195 | 709 | 195 | 677 | 195 | 646 | 195 | 590 |
| 200 | 764 | 200 | 733 | 200 | 702 | 200 | 674 | 200 | 624 |
| 205 | 788 | 205 | 759 | 205 | 731 | 205 | 705 | 205 | 663 |
| 210 | 814 | 210 | 787 | 210 | 762 | 210 | 740 | 210 | 706 |
| 215 | 843 | 215 | 819 | 215 | 797 | 215 | 779 | 215 | 754 |

| Steady State | | [illegible] | | [illegible] | | [illegible] | | [illegible] | |
|---|---|---|---|---|---|---|---|---|---|
| T | P | T | P | T | P | T | P | T | P |
| 220 | 874 | 220 | 853 | 220 | 836 | 220 | 821 | 220 | 806 |
| 225 | 909 | 225 | 892 | 225 | 878 | 225 | 869 | 225 | 865 |
| 230 | 948 | 230 | 935 | 230 | 925 | 230 | 921 | 230 | 930 |
| 235 | 991 | 235 | 982 | 235 | 978 | 235 | 979 | | |
| 240 | 1038 | 240 | 1034 | 240 | 1036 | | | | |
| 245 | 1090 | | | | | | | | |
| 250 | 1148 | | | | | | | | |
| 255 | 1212 | | | | | | | | |
| 260 | 1283 | | | | | | | | |
| 265 | 1360 | | | | | | | | |
| 270 | 1447 | | | | | | | | |
| 275 | 1542 | | | | | | | | |
| 280 | 1647 | | | | | | | | |
| 285 | 1763 | | | | | | | | |
| 290 | 1892 | | | | | | | | |
| 295 | 2034 | | | | | | | | |
| 300 | 2191 | | | | | | | | |
| 305 | 2364 | | | | | | | | |

| | Setpoint (psig) | Setpoint (psig) |
|---|---|---|
| 50 | 490 | 465 |
| 100 | 500 | 475 |
| 135 | 540 | 510 |
| 175 | 575 | 540 |
| 200 | 610 | 370 |
| 250 | 745 | 685 |
| 280 | 745 | 685 |
| 405 | 745 | 685 |
| 450 | 2350 | 2350 |

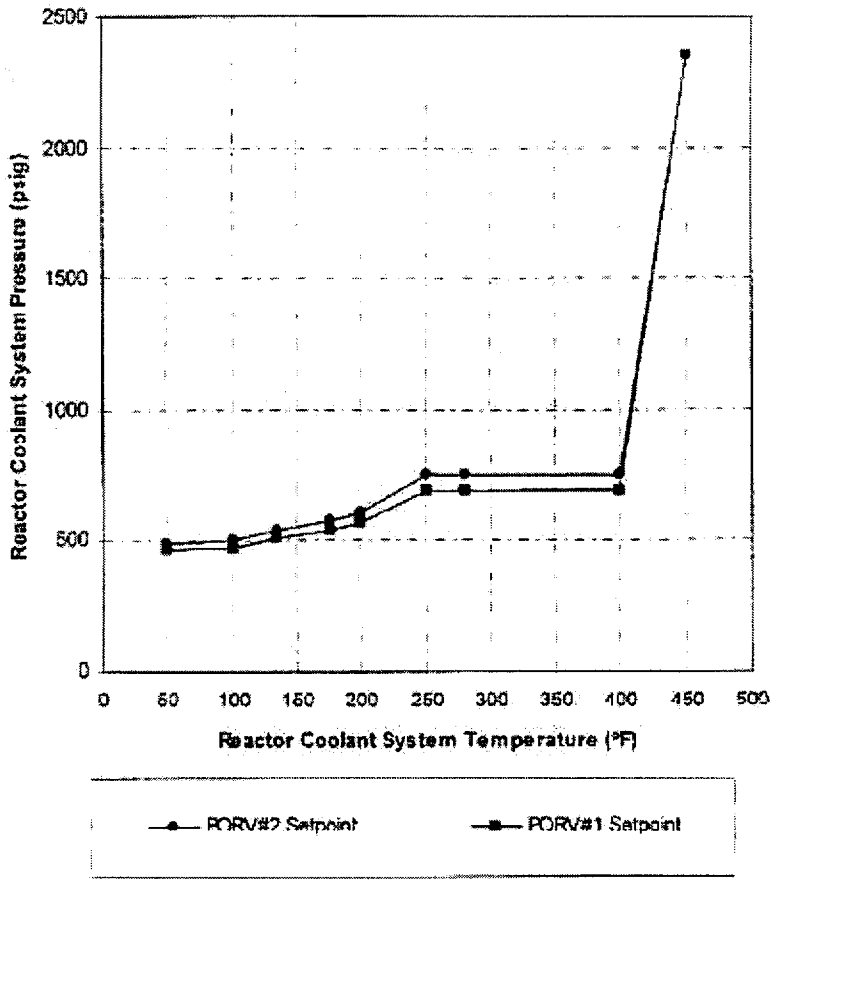



Figure 3-1: Sequoyah Unit 1 LTOPS Selected Setpoints *(Plotted Data provided on Table 3-1)*

| [illegible] | [illegible] | [illegible] | [illegible] | [illegible] |
|---|---|---|---|---|
| T | 40° | 3.39 | 1.03 | $2.61 \times 10^{18}$ (c) |
| U | 140° | 3.47 | 3.00 | $7.96 \times 10^{18}$ (c) |
| X | 220° | 3.47 | 5.27 | $1.32 \times 10^{19}$ (c) |
| Y | 320° | 3.43 | 10.03 | $2.19 \times 10^{19}$ (c,d) |
| S | 4° | 1.08 | Standby | (d,e) |
| V | 176° | 1.08 | Standby | (d,e) |
| W | 184° | 1.08 | Standby | (d,e) |
| Z | 356° | 1.08 | Standby | (d,e) |

Notes:

(a) Updated in Capsule Y dosimetry analysis (WCAP-15224[7]).

(b) Effective Full Power Years (EFPY) from plant startup.

(c) Plant specific evaluation.

(d) This fluence is not less than once or greater than twice the peak end of license (32 EFPY) fluence

(e) Capsules S, V, W and Z will reach a fluence of $2.74 \times 10^{19}$ (E > 1.0 MeV), the 48 EFPY peak vessel fluence at approximately 44 EFPY, respectively.

| Material | Capsule | Fluence (x $10^{19}$ n/cm$^2$) | 30 ft-lb Transition Temperature Shift | | Upper Shelf Energy Decrease | |
|---|---|---|---|---|---|---|
| | | | Predicted (°F) [a] | Measured (°F) [b] | Predicted (%) [a] | Measured (%)[c] |
| Lower Shell Forging 04 (Tangential) | T | 0.261 | 59.85 | 67.52 | 16 | 16 |
| | U | 0.796 | 89.3 | 109.7 | 20.5 | 21 |
| | X | 1.32 | 102.6 | 145.12 | 23 | 8 |
| | Y | 2.19 | 114.95 | 129.87 | 26.5 | 23 |
| Lower Shell Forging 04 (Axial) | T | 0.261 | 59.85 | 50.59 | 16 | 0 |
| | U | 0.796 | 89.3 | 67.59 | 20.5 | 19 |
| | X | 1.32 | 102.6 | 103.34 | 23 | 22 |
| | Y | 2.19 | 114.95 | 133.35 | 26.5 | 19 |
| Weld Metal | T | 0.261 | 111.13 | 127.79 | 35 | 30 |
| | U | 0.796 | 165.82 | 144.92 | 42 | 26 |
| | X | 1.32 | 190.51 | 159.02 | 45 | 21 |
| | Y | 2.19 | 213.44 | 163.8 | 48 | 28 |
| HAZ Metal | T | 0.261 | -- | 45.48 | -- | 20 |
| | U | 0.796 | -- | 78.94 | -- | 26 |
| | X | 1.32 | -- | 95.89 | -- | 3 |
| | Y | 2.19 | -- | 73.3 | -- | 10 |

Notes:

(a) Based on Regulatory Guide 1.99, Revision 2, methodology using the mean weight percent values of copper and nickel of the surveillance material.

(b) Calculated using measured Charpy data plotted using CVGRAPH, Version 4.1[8].

(c) Values are based on the definition of upper shelf energy given in ASTM E185-82.

| | | | | | | |
|---|---|---|---|---|---|---|
| Lower Shell Forging 04 (Tangential) | T | 2.61E+18 | 0.63 | 67.52°F | 42.54°F | 0.40 |
| | U | 7.96E+18 | 0.94 | 109.7°F | 103.12°F | 0.88 |
| | X | 1.32E+19 | 1.08 | 145.12°F | 156.73°F | 1.16 |
| | Y | 2.19E+19 | 1.21 | 129.87°F | 157.14°F | 1.47 |
| Lower Shell Forging 04 (Axial) | T | 2.61E+18 | 0.63 | 50.59°F | 31.87°F | 0.40 |
| | U | 7.96E+18 | 0.94 | 67.59°F | 63.53°F | 0.88 |
| | X | 1.32E+19 | 1.08 | 103.34°F | 111.61°F | 1.16 |
| | Y | 2.19E+19 | 1.21 | 133.35°F | 161.35°F | 1.47 |
| | | | | SUM: | 827.89°F | 7.82 |
| | $CF_{04} = \Sigma(FF * RT_{NDT}) \div \Sigma(FF^2) = (827.89) \div (7.82) = 105.9°F$ | | | | | |
| Surveillance Weld Material[d] | T | 2.61E+18 | 0.63 | 115.0°F | 72.5°F | 0.40 |
| | U | 7.96E+18 | 0.94 | 130.4°F | 122.6°F | 0.88 |
| | X | 1.32E+19 | 1.08 | 143.1°F | 154.5°F | 1.16 |
| | Y | 2.19E+19 | 1.21 | 147.4°F | 178.4°F | 1.47 |
| | | | | SUM: | 528.0°F | 3.91 |
| | $CF_{Surv.\ Weld} = \Sigma(FF * RT_{NDT}) \div \Sigma(FF^2) = (528.0°F) \div (3.91) = 135.0°F$ | | | | | |

Notes:

(a) f = Calculated fluence from capsule Y dosimetry analysis results[7], (x $10^{19}$ n/cm$^2$, E > 1.0 MeV).

(b) FF = fluence factor = $f^{(0.28 - 0.1*\log f)}$.

(c) $\Delta RT_{NDT}$ values are the measured 30 ft lb shift values taken from App. B of Ref. 7, rounded to one decimal point.

(d) The surveillance weld metal $\Delta RT_{NDT}$ values have been adjusted by a ratio factor of 0.90.

| | | | |
|---|---|---|---|
| (Heat # 980807/281489) | | | |
| Lower Shell Forging 04 (Heat # 980919/281587) | 0.13 | 0.76 | 73°F |
| Surveillance Weld[b, d, e] ⇒ | 0.387 | 0.11 | --- |
| Rotterdam Test[c, e] ⇒ | 0.30 | --- | --- |
| Rotterdam Test[c, e] ⇒ | 0.25 | --- | --- |
| Rotterdam Test[c, e] ⇒ | 0.46 | --- | --- |
| Best Estimate of the Intermediate to Lower Shell Forging Circumferential Weld Seam W05[d, e] | 0.35 | 0.11 | -40°F |

Notes:

(a) The Initial $RT_{NDT}$ values are measured values

(b) These copper and nickel values are best estimate values for only the surveillance weld metal and is the average of three data points [0.424 (WCAP-10340, Rev.1), 0.406 (WCAP-10340, Rev.1), 0.33 (WCAP-8233) copper and 0.084 (WCAP-10340, Rev.1), 0.085 (WCAP-10340, Rev.1), 0.17 (WCAP-8233) nickel.]. These values are treated as one data point in the calculation of the best estimate average for the inter. to lower shell circ. weld shown above. Originally the 0.424 / 0.406 and 0.084 / 0.085 values were reported as single points, 0.41 - 0.42 and 0.08 (Per WCAP-10340, Rev. 1[6]), but it is actually made up of two data points. Sample TW58 from capsule T was broken into two samples, TW58a and TW58b, thus providing the two data points.

(c) From NRC Reactor Vessel Integrity Database (RVID) and ultimately from Rotterdam Weld Certifications.

(d) Circumferential Weld Seam W05 was fabricated with weld wire type SMIT 40, Heat # 25295, Flux type SMIT 89, lot # 2275. The surveillance weld was fabricated with weld wire type SMIT 40, Heat # 25295, Flux type SMIT 89, lot # 1103 and is representative of the intermediate to lower shell circumferential weld.

(e) The surveillance weld and the three Rotterdam tests are averaged together for the Best Estimate of the Intermediate to Lower Shell Forging Circumferential Weld Seam.

| EFPY | 0° | 15° | 30° | 45° |
|---|---|---|---|---|
| 10.03 | 0.205 | 0.321 | 0.409 | 0.637 |
| 20 | 0.387 | 0.596 | 0.761 | 1.18 |
| 32 | 0.605 | 0.928 | 1.19 | 1.84 |
| 48 | 0.896 | 1.37 | 1.75 | 2.72 |

| | | | | | | | |
|---|---|---|---|---|---|---|---|
| Intermediate Shell Forging 05 | Position 1.1 | 115.6 | 1.029 | 40 | 119.0 | 34 | 193 |
| Lower Shell Forging 04 | Position 1.1 | 95 | 1.029 | 73 | 97.8 | 34 | 205 |
| | Position 2.1 | 105.9 | 1.029 | 73 | 109.0 | 34 | 216 |
| Intermediate to Lower Shell | Position 1.1 | 161.3 | 1.029 | -40 | 166.0 | 56 | 182 |
| Circumferential Weld Seam | Position 2.1 | 135.0 | 1.029 | -40 | 138.9 | 56 | 155 |

Notes:

(1) Initial $RT_{NDT}$ values measured values.

(2) ART = Initial $RT_{NDT}$ + $\Delta RT_{NDT}$ + Margin (°F)

(3) $\Delta RT_{NDT}$ = CF * FF

(4) $M = 2 * (\sigma_i^2 + \sigma_\Delta^2)^{1/2}$

Table 5-6

Sequoyah Unit 1 Calculation of the ART Values for the 3/4T Location @ 32 EFPY

| Material | RG 1.99 R2 Method | CF (°F) | FF | $IRT_{NDT}$ (1) (°F) | $\Delta RT_{NDT}$ (3) (°F) | Margin (4) (°F) | ART (2) (°F) |
|---|---|---|---|---|---|---|---|
| Intermediate Shell Forging 05 | Position 1.1 | 115.6 | 0.747 | 40 | 86.4 | 34 | 160 |
| Lower Shell Forging 04 | Position 1.1 | 95 | 0.747 | 73 | 71.0 | 34 | 178 |
| | Position 2.1 | 105.9 | 0.747 | 73 | 79.1 | 34 | 186 |
| Intermediate to Lower Shell | Position 1.1 | 161.3 | 0.747 | -40 | 120.5 | 56 | 137 |
| Circumferential Weld Seam | Position 2.1 | 135.0 | 0.747 | -40 | 100.8 | 56 | 117 |

Notes:

(1) Initial $RT_{NDT}$ values measured values.

(2) ART = Initial $RT_{NDT}$ + $\Delta RT_{NDT}$ + Margin (°F)

(3) $\Delta RT_{NDT}$ = CF * FF

(4) $M = 2 * (\sigma_i^2 + \sigma_\Delta^2)^{1/2}$

| | | | |
|---|---|---|---|
| Intermediate Shell Forging 05 | Position 1.1 | 193 | 160 |
| Lower Shell Forging 04 | Position 1.1 | 205 | 178 |
| | Position 2.1 | **216** | **186** |
| Intermediate to Lower Shell Circumferential Weld Seam | Position 1.1 | 182 | 137 |
| | Position 2.1 | 155 | 117 |

Table 5-8

$RT_{PTS}$ Calculations for Sequoyah Unit 1 Beltline Region Materials at 32 EFPY

| Material | Fluence (n/cm$^2$, E>1.0 MeV) | FF | CF (°F) | $\Delta RT_{PTS}$[c] (°F) | Margin (°F) | $RT_{NDT(U)}$[a] (°F) | $RT_{PTS}$[b] (°F) |
|---|---|---|---|---|---|---|---|
| Intermediate Shell Forging 05 | 1.84 | 1.167 | 115.6 | 134.9 | 34 | 40 | 209 |
| Lower Shell Forging 04 | 1.84 | 1.167 | 95.0 | 110.9 | 34 | 73 | 218 |
| Lower Shell Forging 04 (Using S/C Data) | 1.84 | 1.167 | 105.9 | 123.6 | 34 | 73 | 231 |
| Circumferential Weld Metal | 1.84 | 1.167 | 161.3 | 188.2 | 56 | -40 | 204 |
| Circumferential Weld Metal (Using S/C Data) | 1.84 | 1.167 | 135.0 | 157.5 | 56 | -40 | 174 |

Notes:
(a) Initial $RT_{NDT}$ values are measured values
(b) $RT_{PTS} = RT_{NDT(U)} + \Delta RT_{PTS} + \text{Margin}$ (°F)
(c) $\Delta RT_{PTS} = CF * FF$

2. Code of Federal Regulations, 10CFR50, Appendix H, *Reactor Vessel Material Surveillance Program Requirements*, U.S. Nuclear Regulatory Commission, Washington, D.C.

3. ASTM E208, *Standard Test Method for Conducting Drop-Weight Test to Determine Nil Ductility Transition Temperature of Ferritic Steels*, in ASTM Standards, Section 3, American Society for Testing and Materials, Philadelphia, PA.

4. Section XI of the ASME Boiler and Pressure Vessel Code, Appendix G, *Fracture Toughness Criteria for Protection Against Failure*

5. ASTM E185-82, Annual Book of ASTM Standards, Section 12, Volume 12.02, *Standard Practice for Conducting Surveillance Tests for Light-Water Cooled Nuclear Power Reactor Vessels.*

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

7. WCAP-15224, *Analysis of Capsule Y From the Tennessee Valley Authority Sequoyah Unit 1 Reactor Vessel Radiation Surveillance Program.* T.J. Laubham. et. al., Dated June 1999.

8. CVGRAPH, Hyperbolic Tangent Curve-Fitting Program, Version 4.1, developed by ATI Consulting, March 1999.

9. WCAP-10340, Revision 1, "Analysis of Capsule T from the Tennessee Valley Authority Sequoyah Unit 1 Reactor Vessel Radiation Surveillance Program", S.E. Yanichko, et. al., February 1984.

10. WCAP-14040-NP-A, Revision 2, "Methodology used to Develop Cold Overpressure Mitigating System Setpoints and RCS Heatup and Cooldown Limit Curves", J.D. Andrachek, et. al., January 1996.

11. WCAP-15293, Revision 1, "Sequoyah Unit 1 Heatup and Cooldown Limit Curves for Normal Operation and PTLR Support Documentation", J.H. Ledger, April 2001.

12. Westinghouse Letter to TVA, TVA-93-105, "Cold Overpressure Mitigation System Code Case and Delta-P Calculation", May 19, 1993.

13. Calculation SQN-IC-014, "Demonstrated Accuracy Calculation for Cold Overpressure Protection System."

14. ASME Code Case N-640, "Alternative Reference Fracture Toughness for Development of P-T Limit Curves for Section XI, Division 1", February 26, 1999.

design, details and dimensions.
Letter No. N10004
Date: June 06, 2002
TENNESSEE VALLEY AUTHORITY
SQEP (N) BY D. L. Lundy

# Tennessee Valley Authority

# Sequoyah Unit 2

# Pressure Temperature Limits Report

# Revision 3, May 2002

PROJECT Sequoyah DISCIPLINE N
CONTRACT 91NNP-86305B UNIT 2
DESC. RCS Pressure-Temperature Limit Report
DWG/DOC NO. PTLR-2
SHEET - OF - REV 03
DATE 06/06/02 ECN/DCN - FILE N2N-048

RIMS, WTC A-K

This report affects TS 3.4.9.1, RCS Pressure/Temperature Limits (P/T) Limits. All TS requirements associated with Low Temperature Overpressure Protection System (LTOPS) are contained in TS 3.4.12, RCS Overpressure Protection System.

## 2.0 RCS Pressure and Temperature Limits

The limits for TS 3.4.9.1 are presented in the subsection which follows and were developed using the NRC approved methodologies specified in TS 6.9.1.15 with exception of ASME Code Case N-640[11] (Use of $K_k$), WCAP-15315[12] (Elimination of the Flange Requirement), 1996 Version of Appendix G[4] and the revised fluences[7]. The operability requirements associated with LTOPS are specified in TS LCO 3.4.12 and were determined to adequately protect the RCS against brittle fracture in the event of an LTOP Transient in accordance with the methodology specified in TS 6.9.1.15.

2.1 RCS Pressure/Temperature (P/T) Limits (LCO - 3.4.9.1)

2.1.1 The minimum boltup temperature is 50°F

2.1.2 The RCS temperature rate-of-change limits are:

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

2.1.3 The RCD P/T limits for heatup, cooldown, inservice hydrostatic and leak testing, and criticality are specified by Figures 2-1 and 2-2.

## 3.0 Low Temperature Overpressure Protection System (LCO 3.4.12)

The lift setpoints for the pressurizer Power Operated Relief Valves (PORVs) are presented in the subsection which follows. These lift setpoints have been developed using the NRC-approved methodologies specified in Specification 3.4.12.

reactor midplane/beltline or for instrument inaccuracies. The pressure difference between the pressurizer transmitter and the reactor vessel midplane/beltline with four reactor coolant pumps in operation is 68.3 psi
(Ref. 13).

Note: These setpoints include allowance for the 50°F thermal transport effect for heat injection transients. A demonstrated accuracy calculation (Reference 14) has been performed to confirm that the setpoints will maintain the system pressure within the established limits when the pressure difference between the pressure transmitter and reactor midplane and maximum temperature/pressure instrument uncertainties are applied to the setpoints.

### 3.2 Arming Temperature

The LTOPS arming temperature is based upon the methodology defined in the Sequoyah Nuclear Plant Unit 2 Technical Specifications Administrative Controls Section 6.9.1.15. The arming temperature shall be ≤ 350°F.

## 4.0 Reactor Vessel Material Surveillance Program

The reactor vessel material irradiation surveillance specimens shall be removed and examined to determine changes in material properties. The removal schedule is provided in Table 4-1. The results of these examinations shall be used to update Figures 2-1, 2-2 and 3-1.

The pressure vessel steel surveillance program (WCAP-8513[1]) is in compliance with Appendix H to 10 CFR 50, "Reactor Vessel Material Surveillance Program Requirements"[2]. The material test requirements and the acceptance standard utilize the reference nil-ductility temperature $RT_{NDT}$, which is determined in accordance with ASTM E208[3]. The empirical relationship between $RT_{NDT}$ and the fracture toughness of the reactor vessel steel is developed in accordance with Code Case N-640 of Section XI of the ASME Boiler and Pressure Vessel Code, Appendix G, "Fracture Toughness Criteria for Protection Against Failure"[4]. The surveillance capsule removal schedule meets the requirements of ASTM E185-82[5]. The removal schedule is provided in Table 4-1.

Table 5-2 shows calculations of the surveillance material chemistry factors using surveillance capsule data. Note that in the calculation of the surveillance weld chemistry factor, the ratio procedure from Regulatory Guide 1.99, Revision 2 was followed. The ratio in question is equal to 0.93.

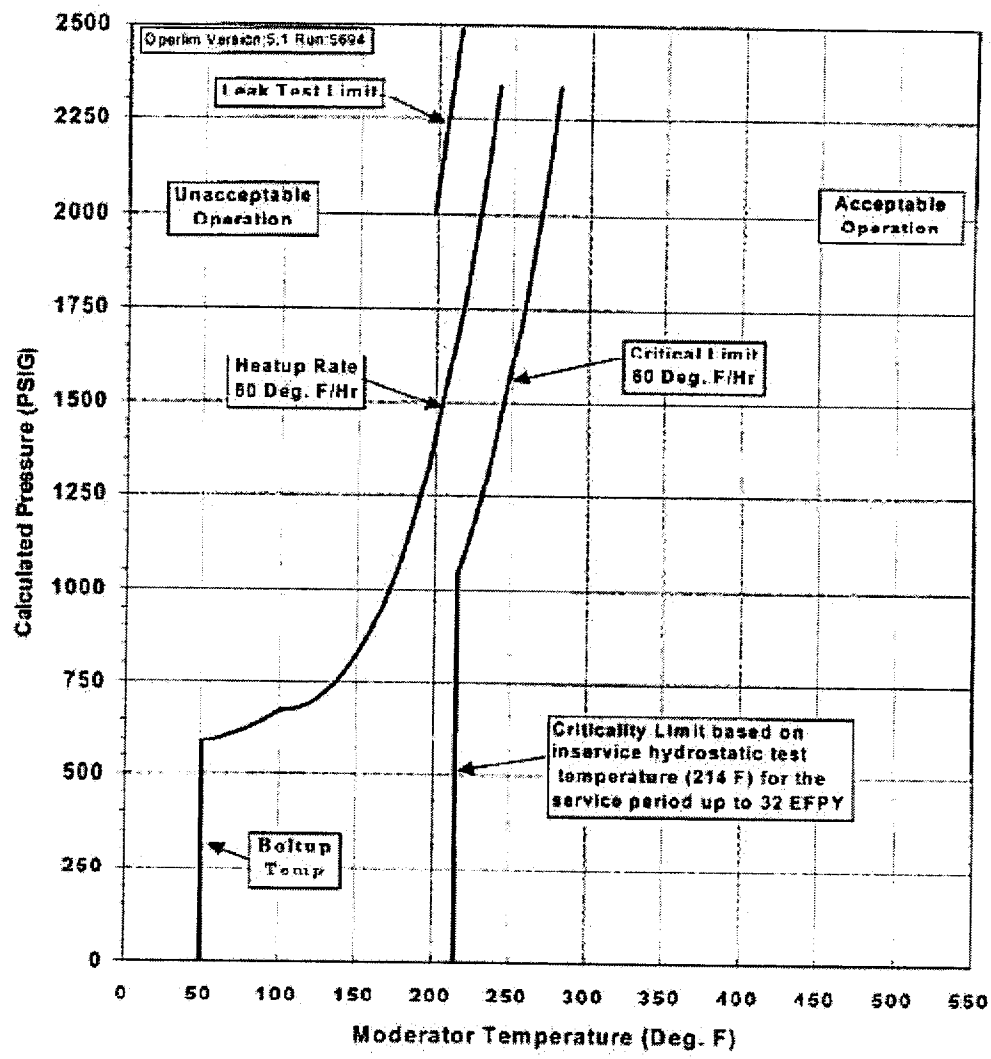
Table 5-3 provides the required Sequoyah Unit 2 reactor vessel toughness data.

Table 5-4 provides a summary of the fluence values used in the generation of the heatup and cooldown limit curves and the PTS evaluation.

Table 5-5 and 5-6 show the calculation of the 1/4T and 3/4T adjusted reference temperature at 32 EFPY for each beltline material in the Sequoyah Unit 2 reactor vessel. The limiting beltline material was the intermediate shell 05.

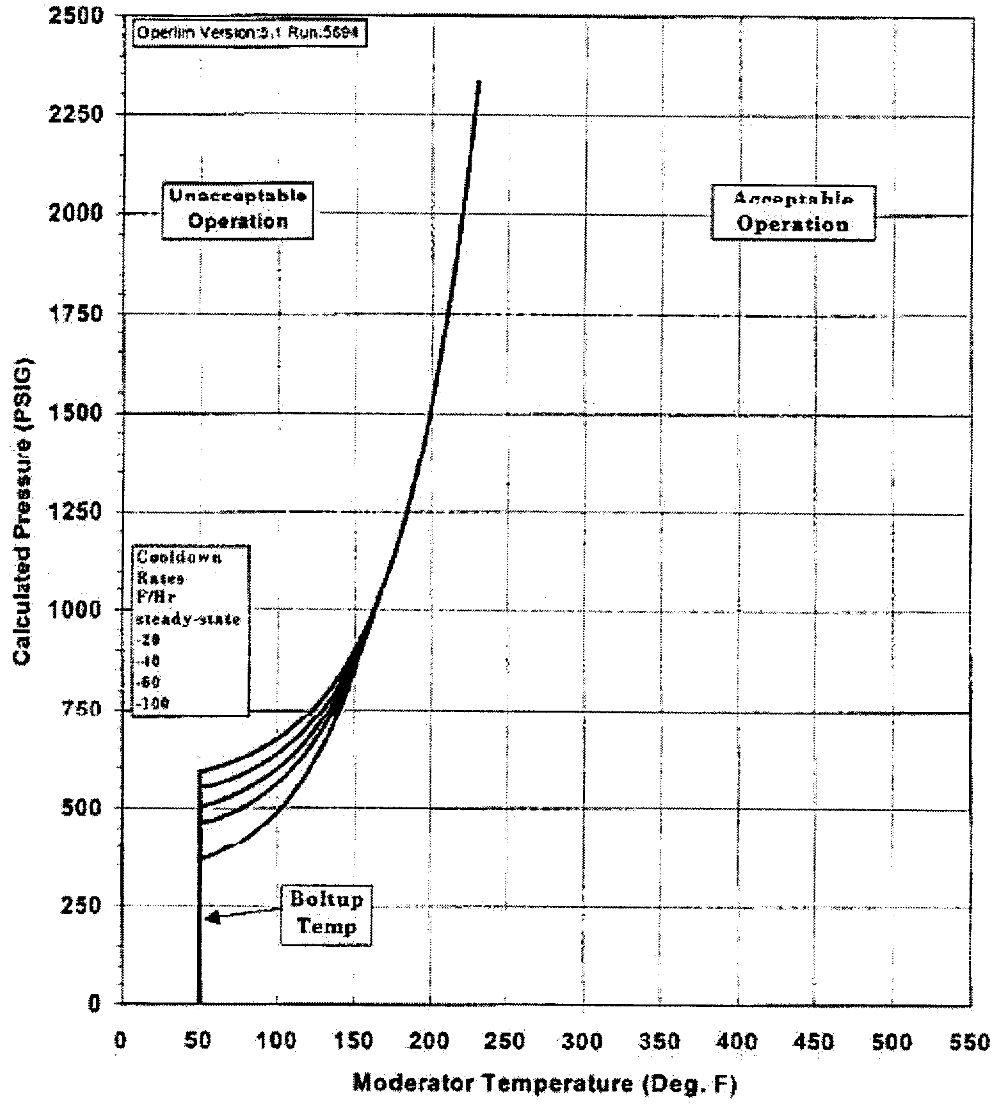
Table 5-7 provides a summary of the adjusted reference temperature (ART) values of the Sequoyah Unit 2 reactor vessel beltline materials at the 1/4T and 3/4T locations for 32 EFPY.

Table 5-8 provides $RT_{PTS}$ values for Sequoyah Unit 2 at 32 EFPY.




3/4T, 115°F

Figure 2-1 **Sequoyah Unit 2 Reactor Coolant System Heatup Limitations (Heatup Rate of 60°F/hr) Applicable for the First 32 EFPY (w/ Margins for Instrumentation Error of 10°F and 60 psig)** ***(Plotted Data provided on Table 2-1)***




3/4T, 115°F

Figure 2-2 Sequoyah Unit 2 Reactor Coolant System Cooldown Limitations (Cooldown Rates up to 100°F/hr) Applicable for the First 32 EFPY (w/ Margins for Instrumentation Error of 10°F and 60 psig) (*Plotted Data provided on Table 2-2*)

| T | P | T | P | T | P |
|---|---|---|---|---|---|
| 50 | 0 | 214 | 0 | 186 | 2000 |
| 50 | 591 | 214 | 607 | 203 | 2485 |
| 55 | 595 | 214 | 614 | | |
| 60 | 601 | 214 | 622 | | |
| 65 | 607 | 214 | 675 | | |
| 70 | 614 | 214 | 678 | | |
| 75 | 622 | 214 | 685 | | |
| 105 | 675 | 214 | 695 | | |
| 110 | 678 | 214 | 708 | | |
| 115 | 685 | 214 | 725 | | |
| 120 | 695 | 214 | 745 | | |
| 125 | 708 | 214 | 768 | | |
| 130 | 725 | 214 | 795 | | |
| 135 | 745 | 214 | 825 | | |
| 140 | 768 | 214 | 860 | | |
| 145 | 795 | 214 | 898 | | |
| 150 | 825 | 214 | 942 | | |
| 155 | 860 | 214 | 990 | | |
| 160 | 898 | 215 | 1043 | | |
| 165 | 942 | 220 | 1103 | | |
| 170 | 990 | 225 | 1168 | | |
| 175 | 1043 | 230 | 1241 | | |
| 180 | 1103 | 235 | 1322 | | |
| 185 | 1168 | 240 | 1411 | | |
| 190 | 1241 | 245 | 1510 | | |
| 195 | 1322 | 250 | 1602 | | |
| 200 | 1411 | 255 | 1697 | | |
| 205 | 1510 | 260 | 1801 | | |
| 210 | 1602 | 265 | 1916 | | |
| 215 | 1697 | 270 | 2043 | | |
| 220 | 1801 | 275 | 2183 | | |
| 225 | 1916 | 280 | 2338 | | |
| 230 | 2043 | | | | |
| 235 | 2183 | | | | |
| 240 | 2338 | | | | |

| Steady State | | 20F | | 40F | | 60F | | 100F | |
|---|---|---|---|---|---|---|---|---|---|
| T | P | T | P | T | P | T | P | T | P |
| 50 | 0 | 50 | 0 | 50 | 0 | 50 | 0 | 50 | 0 |
| 50 | 591 | 50 | 552 | 50 | 503 | 50 | 461 | 50 | 366 |
| 55 | 595 | 55 | 554 | 55 | 508 | 55 | 466 | 55 | 372 |
| 60 | 601 | 60 | 558 | 60 | 514 | 60 | 470 | 60 | 380 |
| 65 | 607 | 65 | 564 | 65 | 521 | 65 | 478 | 65 | 389 |
| 70 | 614 | 70 | 572 | 70 | 529 | 70 | 486 | 70 | 399 |
| 75 | 622 | 75 | 580 | 75 | 538 | 75 | 496 | 75 | 410 |
| 80 | 630 | 80 | 589 | 80 | 548 | 80 | 506 | 80 | 423 |
| 85 | 640 | 85 | 599 | 85 | 559 | 85 | 518 | 85 | 437 |
| 90 | 650 | 90 | 610 | 90 | 571 | 90 | 531 | 90 | 453 |
| 95 | 661 | 95 | 623 | 95 | 584 | 95 | 546 | 95 | 470 |
| 100 | 674 | 100 | 636 | 100 | 599 | 100 | 562 | 100 | 490 |
| 105 | 688 | 105 | 652 | 105 | 616 | 105 | 580 | 105 | 512 |
| 110 | 703 | 110 | 668 | 110 | 634 | 110 | 600 | 110 | 536 |
| 115 | 720 | 115 | 687 | 115 | 654 | 115 | 622 | 115 | 563 |
| 120 | 739 | 120 | 707 | 120 | 676 | 120 | 647 | 120 | 593 |
| 125 | 760 | 125 | 730 | 125 | 701 | 125 | 674 | 125 | 626 |
| 130 | 783 | 130 | 755 | 130 | 729 | 130 | 704 | 130 | 663 |
| 135 | 809 | 135 | 783 | 135 | 759 | 135 | 738 | 135 | 704 |
| 140 | 837 | 140 | 814 | 140 | 793 | 140 | 775 | 140 | 749 |
| 145 | 868 | 145 | 848 | 145 | 831 | 145 | 816 | 145 | 800 |
| 150 | 902 | 150 | 885 | 150 | 872 | 150 | 862 | 150 | 856 |
| 155 | 940 | 155 | 927 | 155 | 918 | 155 | 913 | 155 | 918 |
| 160 | 982 | 160 | 973 | 160 | 968 | 160 | 969 | | |
| 165 | 1028 | 165 | 1024 | 165 | 1025 | | | | |
| 170 | 1080 | | | | | | | | |
| 175 | 1136 | | | | | | | | |
| 180 | 1199 | | | | | | | | |
| 185 | 1268 | | | | | | | | |
| 190 | 1344 | | | | | | | | |
| 195 | 1429 | | | | | | | | |
| 200 | 1522 | | | | | | | | |
| 205 | 1625 | | | | | | | | |
| 210 | 1739 | | | | | | | | |
| 215 | 1865 | | | | | | | | |
| 220 | 2004 | | | | | | | | |
| 225 | 2158 | | | | | | | | |
| 230 | 2328 | | | | | | | | |

| | Setpoint (psig) | Setpoint (psig) |
|---|---|---|
| 50 | 510 | 485 |
| 100 | 580 | 555 |
| 135 | 640 | 610 |
| 174 | 745 | 682 |
| 200 | 745 | 685 |
| 250 | 745 | 685 |
| 278 | 745 | 685 |
| 400 | 745 | 685 |
| 450 | 2350 | 2350 |

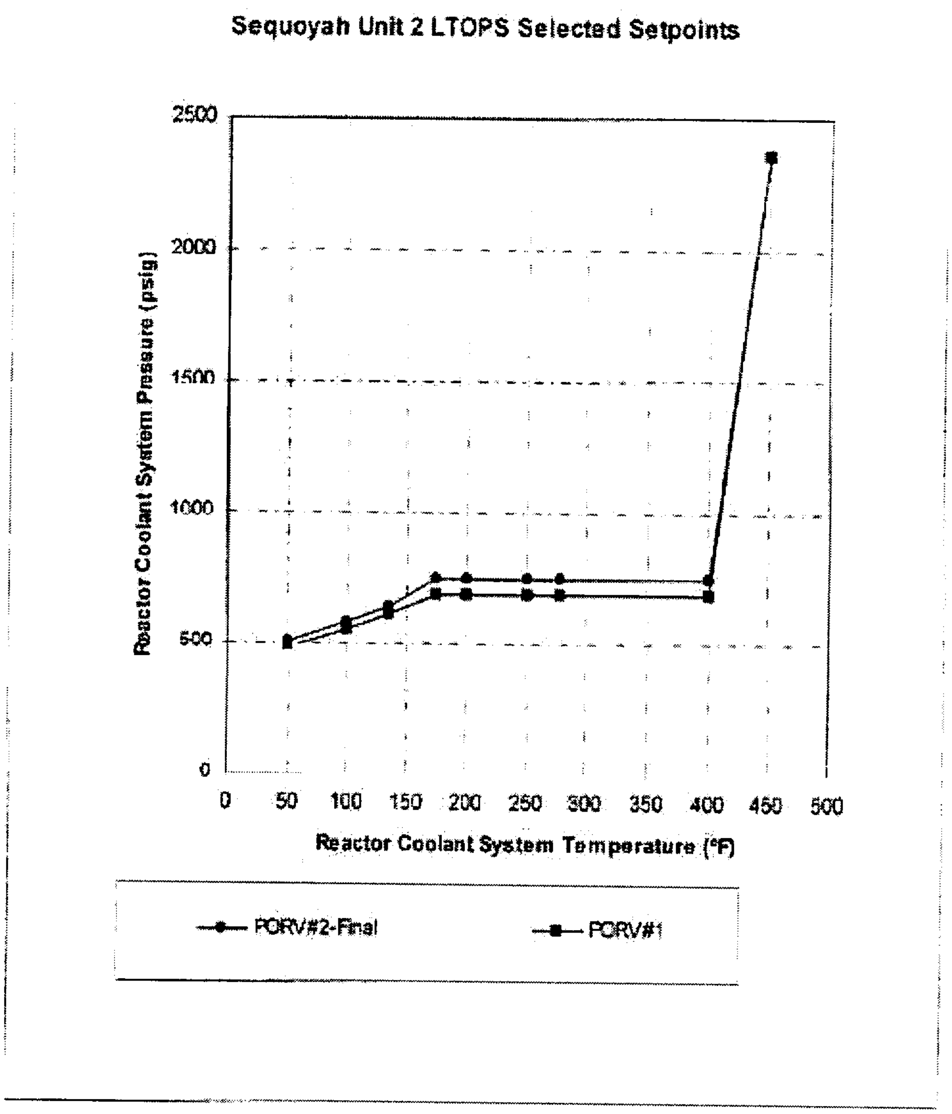



Figure 3-1: Sequoyah Unit 2 LTOP Selected Setpoints *(Plotted Data provided on Table 3-1)*

| | | | | |
|---|---|---|---|---|
| T | 40° | 3.33 | 1.04 | $2.61 \times 10^{18}$ (c) |
| U | 140° | 3.40 | 2.93 | $6.92 \times 10^{18}$ (c) |
| X | 220° | 3.39 | 5.36 | $1.22 \times 10^{19}$ (c) |
| Y | 320° | 3.35 | 10.54 | $2.14 \times 10^{19}$ (c,d) |
| S | 4° | 1.09 | Standby | (e) |
| V | 176° | 1.09 | Standby | (e) |
| W | 184° | 1.09 | Standby | (e) |
| Z | 356° | 1.09 | Standby | (e) |

Notes:

(a) Updated in Capsule Y dosimetry analysis (WCAP-15320[7]).

(b) Effective Full Power Years (EFPY) from plant startup.

(c) Plant specific evaluation.

(d) This fluence is not less than once or greater than twice the peak end of license (32 EFPY) fluence

(e) Capsules S, V, W and Z will reach a fluence of $2.71 \times 10^{19}$ ($E > 1.0$ MeV), the 48 EFPY peak vessel fluence at approximately 44 EFPY.

| Material | Capsule | Fluence ($\times 10^{19}$ n/cm$^2$) | 30 ft-lb Transition Temperature Shift | | Upper Shelf Energy Decrease | |
|---|---|---|---|---|---|---|
| | | | Predicted (°F) (a) | Measured (°F) (b) | Predicted (%) (a) | Measured (%) (c) |
| Intermediate Shell Forging 05 (Tangential) | T | 0.261 | 60.33 | 63.65 | 17 | 12 |
| | U | 0.692 | 85.22 | 79.31 | 21 | 16 |
| | X | 1.22 | 100.23 | 85.7 | 23 | 8 |
| | Y | 2.14 | 114.67 | 134.12 | 26 | 22 |
| Intermediate Shell Forging 05 (Axial) | T | 0.261 | 60.33 | 48.73 | 17 | 7 |
| | U | 0.692 | 85.22 | 66.06 | 21 | 9 |
| | X | 1.22 | 100.23 | 110.04 | 23 | 2 |
| | Y | 2.14 | 114.67 | 89.21 | 26 | 22 |
| Weld Metal | T | 0.261 | 43.12 | 74.56 | 20 | 2 |
| | U | 0.692 | 60.91 | 130.38 | 25 | 6 |
| | X | 1.22 | 71.63 | 44.22 | 29 | 35 |
| | Y | 2.14 | 81.96 | 86.91 | 33 | 3 |
| HAZ Metal | T | 0.261 | -- | 24.58 | -- | 2 |
| | U | 0.692 | -- | 64.03 | -- | 14 |
| | X | 1.22 | -- | 28.29 | -- | 19 |
| | Y | 2.14 | -- | 50.32 | -- | 39 |

Notes:

(a) Based on Regulatory Guide 1.99, Revision 2, methodology using the mean weight percent values of copper and nickel of the surveillance material.

(b) Calculated using measured Charpy data plotted using CVGRAPH, Version 4.1[5].

(c) Values are based on the definition of upper shelf energy given in ASTM E185-82.

| Material | Capsule | Capsule f[a] | FF[b] | $\Delta RT_{NDT}$[c] | $FF*\Delta RT_{NDT}$ | $FF^2$ |
|---|---|---|---|---|---|---|
| Intermediate Shell Forging 05 (Tangential) | T | 2.61E+18 | 0.635 | 63.7 | 40.45 | 0.403 |
| | U | 6.92E+18 | 0.897 | 79.3 | 71.13 | 0.805 |
| | X | 1.22E+19 | 1.055 | 85.7 | 90.41 | 1.113 |
| | Y | 2.14E+19 | 1.207 | 134.1 | 161.86 | 1.457 |
| Intermediate Shell Forging 05 (Axial) | T | 2.61E+18 | 0.635 | 48.7 | 30.92 | 0.403 |
| | U | 6.92E+18 | 0.897 | 66.1 | 59.29 | 0.805 |
| | X | 1.22E+19 | 1.055 | 110.0 | 116.05 | 1.113 |
| | Y | 2.14E+19 | 1.207 | 89.2 | 107.66 | 1.457 |
| | | | | SUM: | 677.77°F | 7.556 |
| | $CF_{05} = \Sigma(FF * RT_{NDT}) \div \Sigma(FF^2) = (677.77) \div (7.556) = 89.7°F$ | | | | | |
| Surveillance Weld Material[d] | T | 2.61E+18 | 0.635 | 69.4 (74.6) | 44.07 | 0.403 |
| | U | 6.92E+18 | 0.897 | 121.3 (130.4) | 108.81 | 0.805 |
| | X | 1.22E+19 | 1.055 | 41.1 (44.2) | 43.36 | 1.113 |
| | Y | 2.14E+19 | 1.207 | 80.8 (86.9) | 97.53 | 1.457 |
| | | | | SUM: | 293.77°F | 3.778 |
| | $CF_{Surv.\ Weld} = \Sigma(FF * RT_{NDT}) \div \Sigma(FF^2) = (293.77°F) \div (3.778) = 77.8°F$ | | | | | |

Notes:

(a) f = Calculated fluence from capsule Y dosimetry analysis results[7], ($x\ 10^{19}$ n/cm$^2$, E > 1.0 MeV).

(b) FF = fluence factor = $f^{(0.28 - 0.1*\log f)}$.

(c) $\Delta RT_{NDT}$ values are the measured 30 ft-lb shift values taken from App. B of Ref. 7, rounded to one decimal point.

(d) The surveillance weld metal $\Delta RT_{NDT}$ values have been adjusted by a ratio factor of 0.93.

| | | | |
|---|---|---|---|
| (Heat # 981334 / 781416) | | | |
| Vessel Flange (Heat # 980893 / 780931) | --- | --- | -22°F |
| Intermediate Shell 05 (Heat # 288757 / 981057) | 0.13 | 0.76 | 10°F |
| Lower Shell Forging 04 (Heat # 990469 / 293323) | 0.14 | 0.76 | 22°F |
| Intermediate to Lower Shell Forging Circumferential Weld Seam[b] | 0.12 | 0.11 | -4°F |
| Surveillance Weld[b] | 0.13 | 0.11 | --- |

Notes:

(a) The Initial $RT_{NDT}$ values are measured values

(b) Circumferential Weld Seam was fabricated with weld wire type SMIT 89, Heat # 4278, Flux type SMIT 89, lot # 1211 and is representative of the intermediate to lower shell circumferential weld.

| EFPY | 0° | 15° | 30° | 45° |
|---|---|---|---|---|
| 10.54 | 0.211 | 0.336 | 0.426 | 0.637 |
| 20 | 0.38 | 0.60 | 0.773 | 1.16 |
| 32 | 0.593 | 0.934 | 1.21 | 1.82 |
| 48 | 0.878 | 1.38 | 1.80 | 2.71 |

| | | | | | | | |
|---|---|---|---|---|---|---|---|
| Intermediate Shell Forging 05 | Position 1.1 | 95 | 1.027 | 10 | 97.6 | 34 | 142 |
| | Position 2.1 | 89.7 | 1.027 | 10 | 92.1 | 34 | 136 |
| Lower Shell Forging 04 | Position 1.1 | 104 | 1.027 | -22 | 106.8 | 34 | 119 |
| Intermediate to Lower Shell Circumferential Weld Seam | Position 1.1 | 63 | 1.027 | -4 | 64.7 | 56 | 117 |
| | Position 2.1 | 77.8 | 1.027 | -4 | 79.9 | 56 | 132 |

Notes:

(1) Initial $RT_{NDT}$ values measured values.
(2) ART = Initial $RT_{NDT}$ + $\Delta RT_{NDT}$ + Margin (°F)
(3) $\Delta RT_{NDT}$ = CF * FF
(4) $M = 2 * (\sigma_i^2 + \sigma_\Delta^2)^{1/2}$

Table 5-6
Sequoyah Unit 2 Calculation of the ART Values for the 3/4T Location @ 32 EFPY

| Material | RG 1.99 R2 Method | CF (°F) | FF | $IRT_{NDT}$[1] (°F) | $\Delta RT_{NDT}$[3] (°F) | Margin[4] (°F) | ART[2] (°F) |
|---|---|---|---|---|---|---|---|
| Intermediate Shell Forging 05 | Position 1.1 | 95 | 0.745 | 10 | 70.8 | 34 | 115 |
| | Position 2.1 | 89.7 | 0.745 | 10 | 66.8 | 34 | 111 |
| Lower Shell Forging 04 | Position 1.1 | 104 | 0.745 | -22 | 77.5 | 34 | 90 |
| Intermediate to Lower Shell Circumferential Weld Seam | Position 1.1 | 63 | 0.745 | -4 | 46.9 | 56 | 99 |
| | Position 2.1 | 77.8 | 0.745 | -4 | 58.0 | 56 | 110 |

Notes:

(1) Initial $RT_{NDT}$ values measured values.
(2) ART = Initial $RT_{NDT}$ + $\Delta RT_{NDT}$ + Margin (°F)
(3) $\Delta RT_{NDT}$ = CF * FF
(4) $M = 2 * (\sigma_i^2 + \sigma_\Delta^2)^{1/2}$

| | | | |
|---|---|---|---|
| Intermediate Shell Forging 05 | Position 1.1 | 142 | 115 |
| | Position 2.1 | 136 | 111 |
| Lower Shell Forging 04 | Position 1.1 | 119 | 90 |
| Intermediate to Lower Shell Circumferential Weld Seam | Position 1.1 | 117 | 99 |
| | Position 2.1 | 132 | 110 |

Table 5-8
$RT_{PTS}$ Calculations for Sequoyah Unit 2 Beltline Region Materials at 32 EFPY

| Material | Fluence ($n/cm^2$, E>1.0 MeV) | FF | CF (°F) | $\Delta RT_{PTS}$ [c] (°F) | Margin (°F) | $RT_{NDT(U)}$ [a] (°F) | $RT_{PTS}$ [b] (°F) |
|---|---|---|---|---|---|---|---|
| Intermediate Shell Forging 05 | 1.82 | 1.164 | 95 | 110.6 | 34 | 10 | 155 |
| Intermediate Shell Forging 05 (Using S/C Data) | 1.82 | 1.164 | 89.7 | 104.4 | 34 | 10 | 148 |
| Lower Shell Forging 04 | 1.82 | 1.164 | 104 | 121.1 | 34 | -22 | 133 |
| Circumferential Weld Metal | 1.82 | 1.164 | 63 | 73.3 | 56 | -4 | 125 |
| Circumferential Weld Metal (Using S/C Data) | 1.82 | 1.164 | 77.8 | 90.6 | 56 | -4 | 143 |

Notes:

(a) Initial $RT_{NDT}$ values are measured values

(b) $RT_{PTS} = RT_{NDT(U)} + \Delta RT_{PTS} + \text{Margin}$ (°F)

(c) $\Delta RT_{PTS} = CF * FF$

2. Code of Federal Regulations, 10CFR50, Appendix H, *Reactor Vessel Material Surveillance Program Requirements*, U.S. Nuclear Regulatory Commission, Washington, D.C.

3. ASTM E208, *Standard Test Method for Conducting Drop-Weight Test to Determine Nil-Ductility Transition Temperature of Ferritic Steels*, in ASTM Standards, Section 3, American Society for Testing and Materials, Philadelphia, PA.

4. Section XI of the ASME Boiler and Pressure Vessel Code, Appendix G, *Fracture Toughness Criteria for Protection Against Failure*

5. ASTM E185-82, Annual Book of ASTM Standards, Section 12, Volume 12.02, *Standard Practice for Conducting Surveillance Tests for Light-Water Cooled Nuclear Power Reactor Vessels*.

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

7. WCAP-15320, *Analysis of Capsule Y From the Tennessee Valley Authority Sequoyah Unit 2 Reactor Vessel Radiation Surveillance Program*, T.J. Laubham, et. al., Dated November 1999.

8. CVGRAPH, Hyperbolic Tangent Curve-Fitting Program, Version 4.1, developed by ATI Consulting, March 1999.

9. WCAP-14040-NP-A, Revision 2, "Methodology used to Develop Cold Overpressure Mitigating System Setpoints and RCS Heatup and Cooldown Limit Curves", J.D. Andrachek, et. al., January 1996.

10. WCAP-15321, Revision 1, "Sequoyah Unit 2 Heatup and Cooldown Limit Curves for Normal Operation and PTLR Support Documentation", J.H. Ledger, et.al., April 2001.

11. ASME Code Case N-640, "Alternative Reference Fracture Toughness for Development of P-T Limit Curves for Section XI, Division 1", February 26, 1999.

12. WCAP-15315, "Reactor Vessel Closure Head/Vessel Flange Requirements Evaluation For Operating PWR and BWR Plants", W. Bamford, et.al., October 1999.

13. Westinghouse Letter to TVA, TVA-93-105, "Cold Overpressure Mitigation System Code Case and Delta-P Calculation", May 19, 1993.

14. Calculation SQN-IC-014, Demonstrated Accuracy Calculation for Cold Overpressure Protection System".

ENCLOSURE 5

TENNESSEE VALLEY AUTHORITY
SEQUOYAH NUCLEAR PLANT (SQN)
UNITS 1 AND 2

TOPICAL REPORTS

WCAP-15293, Revision 1 (Unit 1)
WCAP-15321, Revision 1 (Unit 2)

Westinghouse Non-Proprietary Class 3

WCAP-15293
Revision 1

# Sequoyah Unit 1 Heatup and Cooldown Limit Curves for Normal Operation and PTLR Support Documentation

Westinghouse Electric Company LLC

05173758

WCAP-15293, Revision 1

# Sequoyah Unit 1
# Heatup and Cooldown Limit Curves
# for Normal Operation and PTLR Support Documentation

J. H. Ledger

April 2001

Prepared by the Westinghouse Electric Company LLC
for the Tennessee Valley Authority

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# PREFACE

This report has been technically reviewed and verified by:

T. J. Laubham

**Revision 1:**

An error was detected in the "OPERLIM" Computer Program that Westinghouse uses to generate pressure-temperature (PT) limit curves. This error potentially effects the heatup curves when the 1996 Appendix G Methodology is used in generating the PT curves. It has been determined that WCAP-15293 Rev. 0 was impacted by this error. Thus, this revision provides corrected curves from WCAP-15293 Rev. 0.

Note that only the 60°F/hr heatup curves were affected by this error. The 100°F/hr heatup and all cooldown curves were not affected by the computer error and thus remain valid.

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## EXECUTIVE SUMMARY

This report provides the methodology and results of the generation of heatup and cooldown pressure temperature limit curves for normal operation of the Sequoyah Unit 1 reactor vessel. In addition, Pressure Temperature Limits Report (PTLR) support information, such as LTOPS Setpoint, PTS , EOL USE and Withdrawal Schedule, is documented herein in Appendices. The PT curves were generated based on the latest available reactor vessel information (Capsule Y analysis, WCAP-15224[7] and the latest Pressure-Temperature (P-T) Limit Curves from WCAP-12970[13]). The Sequoyah Unit 1 heatup and cooldown pressure-temperature limit curves have been updated based on the use of the ASME Code Case N-640[3], which allows the use of the $K_{Ic}$ methodology, and a justification to lower the reactor vessel flange temperature requirement (Reference WCAP-15315[18]).

# 1 INTRODUCTION

Heatup and cooldown limit curves are calculated using the adjusted $RT_{NDT}$ (reference nil-ductility temperature) corresponding to the limiting beltline region material of the reactor vessel. The adjusted $RT_{NDT}$ of the limiting material in the core region of the reactor vessel is determined by using the unirradiated reactor vessel material fracture toughness properties, estimating the radiation-induced $\Delta RT_{NDT}$, and adding a margin. 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 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, $\Delta RT_{NDT}$ due to the radiation exposure associated with that time period must be added to the unirradiated $RT_{NDT}$ ($IRT_{NDT}$). 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, Revision 2, "Radiation Embrittlement of Reactor Vessel Materials."[1] Regulatory Guide 1.99, Revision 2, is used for the calculation of Adjusted Reference Temperature (ART) values ($IRT_{NDT} + \Delta RT_{NDT}$ + margins for uncertainties) at the 1/4T and 3/4T locations, where T is the thickness of the vessel at the beltline region measured from the clad/base metal interface.

The heatup and cooldown curves documented in this report were generated using the most limiting ART values and the NRC approved methodology documented in WCAP-14040-NP-A, Revision 2[2], "Methodology Used to Develop Cold Overpressure Mitigating System Setpoints and RCS Heatup and Cooldown Limit Curves" with exception of the following: 1) The fluence values used in this report are calculated fluence values, not the best estimate fluence values (See Appendix B). 2) The $K_{Ic}$ critical stress intensities are used in place of the $K_{Ia}$ critical stress intensities. This methodology is taken from approved ASME Code Case N-640[3]. 3) The reactor vessel flange temperature requirement has been reduced. Justification has been provided in WCAP-15315[18]. 4) The 1996 Version of Appendix G to Section XI[4] will be used rather than the 1989 version.

# 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 Standard Review Plan[5]. The beltline material properties of the Sequoyah Unit 1 reactor vessel is presented in Table 1.

Best estimate copper (Cu) and nickel (Ni) weight percent values used to calculate chemistry factors (CF) in accordance with Regulatory Guide 1.99, Revision 2, are provided in Table 3. Additionally, surveillance capsule data is available for four capsules (Capsules T, U, X and Y) already removed from the Sequoyah Unit 1 reactor vessel. This surveillance capsule data was also used to calculate CF values per Position 2.1 of Regulatory Guide 1.99, Revision 2. These CF values are presented in Table 2 and 3.

The NRC Standard Review Plan and Regulatory Guide 1.99, Revision 2 methodology used to develop the heatup and cooldown curves documented in this report is the same as that documented in WCAP-14040, Revision 2.

TABLE 1

Reactor Vessel Beltline Material Unirradiated Toughness Properties

| Material Description | Cu (%) | Ni(%) | Initial $RT_{NDT}$[(a)] |
|---|---|---|---|
| Intermediate Shell Forging 05 (Heat # 980807/281489) | 0.15 | 0.86 | 40°F |
| Lower Shell Forging 04 (Heat # 980919/281587) | 0.13 | 0.76 | 73°F |
| Surveillance Weld[(b, d, e)] ⇒ | 0.387 | 0.11 | --- |
| Rotterdam Test[(c, e)] ⇒ | 0.30 | --- | --- |
| Rotterdam Test[(c, e)] ⇒ | 0.25 | --- | --- |
| Rotterdam Test[(c, e)] ⇒ | 0.46 | --- | --- |
| Best Estimate of the Intermediate to Lower Shell Forging Circumferential Weld Seam W05[(d, e)] | 0.35 | 0.11 | -40°F |

Notes:

(a) The Initial $RT_{NDT}$ values are measured values

(b) These copper and nickel values are best estimate values for only the surveillance weld metal and is the average of three data points [0.424 (WCAP-10340, Rev.1), 0.406 (WCAP-10340, Rev.1), 0.33 (WCAP-8233) copper and 0.084 (WCAP-10340, Rev.1), 0.085 (WCAP-10340, Rev.1), 0.17 (WCAP-8233) nickel.]. These values are treated as one data point in the calculation of the best estimate average for the inter. to lower shell circ. weld shown above. Originally the 0.424 / 0.406 and 0.084 / 0.085 values were reported as single points, 0.41 - 0.42 and 0.08 (Per WCAP-10340, Rev. 1[6]), but it is actually made up of two data points. Sample TW58 from capsule T was broken into two samples, TW58a and TW58b, thus providing the two data points.

(c) From NRC Reactor Vessel Integrity Database (RVID) and ultimately from Rotterdam Weld Certifications.

(d) Circumferential Weld Seam W05 was fabricated with weld wire type SMIT 40, Heat # 25295, Flux type SMIT 89, lot # 2275. The surveillance weld was fabricated with weld wire type SMIT 40, Heat # 25295, Flux type SMIT 89, lot # 1103 and is representative of the intermediate to lower shell circumferential weld.

(e) The surveillance weld and the three Rotterdam tests are averaged together for the Best Estimate of the Intermediate to Lower Shell Forging Circumferential Weld Seam.

The chemistry factors were calculated using Regulatory Guide 1.99 Revision 2, Positions 1.1 and 2.1. Position 1.1 uses the Tables from the Reg. Guide along with the best estimate copper and nickel weight percents. Position 2.1 uses the surveillance capsule data from all capsules withdrawn to date. The fluence values used to determine the CFs in Table 2 are the calculated fluence values at the surveillance capsule locations. Hence, the calculated fluence values were used for all cases.

The measured $\Delta RT_{NDT}$ values for the weld data were adjusted using the ratio procedure given in Position 2.1 of Regulatory Guide 1.99, Revision 2. All fluence values were obtained from the recent Sequoyah Unit 1 capsule analysis[7] which calculated the fluences using the ENDF/B-VI scattering cross-section data set. The fluence values used are also documented in Appendix C of this report.

TABLE 2

Calculation of Chemistry Factors using Sequoyah Unit 1 Surveillance Capsule Data

| Material | Capsule | Capsule f[a] | FF[b] | $\Delta RT_{NDT}$[c] | FF*$\Delta RT_{NDT}$ | $FF^2$ |
|---|---|---|---|---|---|---|
| Lower Shell Forging 04 (Tangential) | T | 2.61E+18 | 0.63 | 67.52°F | 42.54°F | 0.40 |
| | U | 7.96E+18 | 0.94 | 109.7°F | 103.12°F | 0.88 |
| | X | 1.32E+19 | 1.08 | 145.12°F | 156.73°F | 1.16 |
| | Y | 2.19E+19 | 1.21 | 129.87°F | 157.14°F | 1.47 |
| Lower Shell Forging 04 (Axial) | T | 2.61E+18 | 0.63 | 50.59°F | 31.87°F | 0.40 |
| | U | 7.96E+18 | 0.94 | 67.59°F | 63.53°F | 0.88 |
| | X | 1.32E+19 | 1.08 | 103.34°F | 111.61°F | 1.16 |
| | Y | 2.19E+19 | 1.21 | 133.35°F | 161.35°F | 1.47 |
| | | | | SUM: | 827.89°F | 7.82 |
| | $CF_{04} = \Sigma(FF * RT_{NDT}) \div \Sigma(FF^2) = (827.89) \div (7.82) = 105.9°F$ | | | | | |
| Surveillance Weld Material[d] | T | 2.61E+18 | 0.63 | 115.0°F | 72.5°F | 0.40 |
| | U | 7.96E+18 | 0.94 | 130.4°F | 122.6°F | 0.88 |
| | X | 1.32E+19 | 1.08 | 143.1°F | 154.5°F | 1.16 |
| | Y | 2.19E+19 | 1.21 | 147.4°F | 178.4°F | 1.47 |
| | | | | SUM: | 528.0°F | 3.91 |
| | $CF_{Surv. Weld} = \Sigma(FF * RT_{NDT}) \div \Sigma(FF^2) = (528.0°F) \div (3.91) = 135.0°F$ | | | | | |

Notes:

(a) f = Calculated fluence from capsule Y dosimetry analysis results [7], (x $10^{19}$ n/cm$^2$, E > 1.0 MeV).

(b) FF = fluence factor = $f^{(0.28 - 0.1*\log f)}$.

(c) $\Delta RT_{NDT}$ values are the measured 30 ft-lb shift values taken from App. B of Ref. 7, rounded to one decimal point.

(d) The surveillance weld metal $\Delta RT_{NDT}$ values have been adjusted by a ratio factor of 0.90.

**TABLE 3**

**Summary of the Sequoyah Unit 1 Reactor Vessel Beltline Material Chemistry Factors**

| Material | Reg. Guide 1.99, Rev. 2 Position 1.1 CF's | Reg. Guide 1.99, Rev. 2 Position 2.1 CF's |
|---|---|---|
| Intermediate Shell Forging 05 | 115.6°F | --- |
| Lower Shell Forging 04 | 95°F | 105.9°F |
| Circumferential Weld W05 (Heat # 25295) | 161.3°F | 135.0°F |
| Surveillance Weld Metal (Heat # 25295) | 178.7°F | --- |

# 3 CRITERIA FOR ALLOWABLE PRESSURE-TEMPERATURE RELATIONSHIPS

## 3.1 Overall Approach

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, $K_{Ic}$, for the metal temperature at that time. $K_{Ic}$ is obtained from the reference fracture toughness curve, defined in Code Case N-640, "Alternative Reference Fracture Toughness for Development of PT Limit Curves for Section XI"[3 & 4] of the ASME Appendix G to Section XI. The $K_{Ic}$ curve is given by the following equation:

$$K_{Ic} = 33.2 + 20.734 * e^{[0.02(T - RT_{NDT})]} \qquad (1)$$

where,

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

This $K_{Ic}$ curve is based on the lower bound of static critical $K_I$ values measured as a function of temperature on specimens of SA-533 Grade B Class1, SA-508-1, SA-508-2, SA-508-3 steel.

## 3.2 Methodology for Pressure-Temperature Limit Curve Development

The governing equation for the heatup-cooldown analysis is defined in Appendix G of the ASME Code as follows:

$$C^* K_{Im} + K_{It} < K_{Ic} \qquad (2)$$

where,

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

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

$K_{Ic}$ = 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

For membrane tension, the corresponding $K_I$ for the postulated defect is:

$$K_{Im} = M_m \times (pR_i / t) \tag{3}$$

where, $M_m$ for an inside surface flaw is given by:

$M_m$ = 1.85 for $\sqrt{t} < 2$,
$M_m$ = $0.926\sqrt{t}$ for $2 \le \sqrt{t} \le 3.464$,
$M_m$ = 3.21 for $\sqrt{t} > 3.464$

Similarly, $M_m$ for an outside surface flaw is given by:

$M_m$ = 1.77 for $\sqrt{t} < 2$,
$M_m$ = $0.893\sqrt{t}$ for $2 \le \sqrt{t} \le 3.464$,
$M_m$ = 3.09 for $\sqrt{t} > 3.464$

and p = internal pressure, Ri = vessel inner radius, and t = vessel wall thickness.

For bending stress, the corresponding $K_I$ for the postulated defect is:

$K_{Ib}$ = $M_b$ * Maximum Stress, where $M_b$ is two-thirds of $M_m$

The maximum $K_I$ produced by radial thermal gradient for the postulated inside surface defect of G-2120 is $K_{It} = 0.953 \times 10^{-3} \times CR \times t^{2.5}$, where CR is the cooldown rate in °F/hr., or for a postulated outside surface defect, $K_{It} = 0.753 \times 10^{-3} \times HU \times t^{2.5}$, where HU is the heatup rate in °F/hr.

The through-wall temperature difference associated with the maximum thermal $K_I$ can be determined from Fig. G-2214-1. The temperature at any radial distance from the vessel surface can be determined from Fig. G-2214-2 for the maximum thermal $K_I$.

(a) The maximum thermal $K_I$ relationship and the temperature relationship in Fig. G-2214-1 are applicable only for the conditions given in G-2214.3(a)(1) and (2).

(b) Alternatively, the $K_I$ for radial thermal gradient can be calculated for any thermal stress distribution and at any specified time during cooldown for a ¼-thickness inside surface defect using the relationship:

$$K_{It} = (1.0359C_0 + 0.6322C_1 + 0.4753C_2 + 0.3855C_3) * \sqrt{\pi a} \tag{4}$$

or similarly, $K_{IT}$ during heatup for a ¼-thickness outside surface defect using the relationship:

$$K_{It} = (1.043C_0 + 0.630C_1 + 0.481C_2 + 0.401C_3) * \sqrt{\pi a} \qquad (5)$$

where the coefficients $C_0$, $C_1$, $C_2$ and $C_3$ are determined from the thermal stress distribution at any specified time during the heatup or cooldown using the form:

$$\sigma(x) = C_0 + C_1(x/a) + C_2(x/a)^2 + C_3(x/a)^3 \qquad (6)$$

and $x$ is a variable that represents the radial distance from the appropriate (i.e., inside or outside) surface to any point on the crack front and $a$ is the maximum crack depth.

Note, that equations 3, 4 and 5 were implemented in the OPERLIM computer code, which is the program used to generate the pressure-temperature (P-T) limit curves. No other changes were made to the OPERLIM computer code with regard to P-T calculation methodology. Therefore, the P-T curve methodology is unchanged from that described in WCAP-14040, "Methodology used to Develop Cold Overpressure Mitigating System Setpoints and RCS Heatup and Cooldwon Limit Curves"[2] Section 2.6 (equations 2.6.2-4 and 2.6.3-1) with the exceptions just described above.

At any time during the heatup or cooldown transient, $K_{Ic}$ is determined by the metal temperature at the tip of a postulated flaw at the 1/4T and 3/4T location, 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 1/4T vessel location is at a higher temperature than the fluid adjacent to the vessel inner diameter. This condition, of course, is not true for the steady-state situation. It follows that, at any given reactor coolant temperature, the $\Delta T$ (temperature) developed during cooldown results in a higher value of $K_{Ic}$ at the 1/4T location for finite cooldown rates than for steady-state operation. Furthermore, if conditions exist so that the increase in $K_{Ic}$ 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 1/4T 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 1/4T 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 $K_{Ic}$ for the 1/4T crack during heatup is lower than the $K_{Ic}$ for the 1/4T 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 $K_{Ic}$ values 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 1/4T 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 1/4T flaw located at the 1/4T location from the outside surface 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.

### 3.3 Closure Head/Vessel Flange Requirements

10 CFR Part 50, Appendix G 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 (3107 psi), which is 621 psig for Sequoyah Unit 1 reactor vessel. However, per WCAP-15315, "Reactor Vessel Closure Head/Vessel Flange Requirements Evaluation For Operating PWR and BWR Plants"[18], this requirement is no longer necessary when using the methodology of Code Case N-640[3]. Hence, Sequoyah Unit 1 heatup and cooldown limit curves will be generated without flange requirements included.

# 4 CALCULATION OF ADJUSTED REFERENCE TEMPERATURE

From Regulatory Guide 1.99, Revision 2, the adjusted reference temperature (ART) for each material in the beltline region is given by the following expression:

$$ART = \text{Initial } RT_{NDT} + \Delta RT_{NDT} + \text{Margin} \tag{7}$$

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[8]. 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:

$$\Delta RT_{NDT} = CF * f^{(0.28 - 0.10 \log f)} \tag{8}$$

To calculate $\Delta RT_{NDT}$ 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.

$$f_{(depth\ x)} = f_{surface} * e^{(-0.24x)} \tag{9}$$

where x inches (vessel beltline thickness is 8.45 inches) is the depth into the vessel wall measured from the vessel clad/base metal interface. The resultant fluence is then placed in Equation 8 to calculate the $\Delta RT_{NDT}$ at the specific depth.

The Westinghouse Radiation Engineering and Analysis Group evaluated the vessel fluence projections as a part of WCAP-15224 and are also presented in a condensed version in Table 4 of this report. The evaluation used the ENDF/B-VI scattering cross-section data set. This is consistent with methods presented in WCAP-14040-NP-A, "Methodology Used to Develop Cold Overpressure Mitigating System Setpoints and RCS Heatup and Cooldown Limit Curves"[2]. Table 4 contains the calculated vessel surface fluences values at various azimuthal locations and Tables 5 and 6 contains the 1/4T and 3/4T calculated fluences and fluence factors, per the Regulatory Guide 1.99, Revision 2, used to calculate the ART values for all beltline materials in the Sequoyah Unit 1 reactor vessel. Additionally, the surveillance capsule fluence values are presented in Table 7.

TABLE 4

Neutron Fluence Projections at Key Locations on the Reactor Vessel Clad/Base Metal Interface ($10^{19}$ n/cm$^2$, E > 1.0 MeV)

| EFPY | Azimuthal Location | | | |
|---|---|---|---|---|
| | 0° | 15° | 30° | 45° |
| 10.03 | 0.205 | 0.321 | 0.409 | 0.637 |
| 20 | 0.387 | 0.596 | 0.761 | 1.18 |
| 32 | 0.605 | 0.928 | 1.19 | 1.84 |
| 48 | 0.896 | 1.37 | 1.75 | 2.72 |

TABLE 5

Summary of the Vessel Surface, 1/4T and 3/4T Fluence Values used for the Generation of the 32 EFPY Heatup/Cooldown Curves

| Material | Surface | ¼ T[(a)] | ¾ T[(a)] |
|---|---|---|---|
| Intermediate Shell Forging 05 | $1.84 \times 10^{19}$ | $1.11 \times 10^{19}$ | $4.02 \times 10^{18}$ |
| Lower Shell Forging 04 | $1.84 \times 10^{19}$ | $1.11 \times 10^{19}$ | $4.02 \times 10^{18}$ |
| Circumferential Weld Seam W05 (Heat 25295) | $1.84 \times 10^{18}$ | $1.11 \times 10^{19}$ | $4.02 \times 10^{17}$ |

Note:

(a) 1/4T and 3/4T = $F_{(Surface)} * e^{(-0.24*x)}$, where x is the depth into the vessel wall (i.e. 8.45*0.25 or 0.75)

TABLE 6

Summary of the Calculated Fluence Factors used for the Generation of the 32 EFPY Heatup and Cooldown Curves

| EFPY | 1/4T FF | 3/4T FF |
|---|---|---|
| 32 | 1.029 | 0.747 |

TABLE 7
Integrated Neutron Exposure of the Sequoyah Unit 1 Surveillance Capsules Tested To Date

| Capsule | Fluence |
|---|---|
| T | $2.61 \times 10^{18}$ n/cm$^2$, (E > 1.0 MeV) |
| U | $7.96 \times 10^{18}$ n/cm$^2$, (E > 1.0 MeV) |
| X | $1.32 \times 10^{19}$ n/cm$^2$, (E > 1.0 MeV) |
| Y | $2.19 \times 10^{19}$ n/cm$^2$, (E > 1.0 MeV) |

Margin is calculated as, $M = 2\sqrt{\sigma_i^2 + \sigma_\Delta^2}$. The standard deviation for the initial $RT_{NDT}$ margin term, is $\sigma_i$ 0°F when the initial $RT_{NDT}$ is a measured value, and 17°F when a generic value is available. The standard deviation for the $\Delta RT_{NDT}$ margin term, $\sigma_\Delta$, is 17°F for plates or forgings, and 8.5°F for plates or forgings when surveillance data is used. For welds, $\sigma_\Delta$ is equal to 28°F when surveillance capsule data is not used, and is 14°F (half the value) when credible surveillance capsule data is used. $\sigma_\Delta$ need not exceed 0.5 times the mean value of $\Delta RT_{NDT}$.

Based on the surveillance program credibility evaluation presented in Appendix D to WCAP-15224, the Sequoyah Unit 1 surveillance program data is non-credible. In addition, following the guidance provided by the NRC in recent industry meeting, Table Chemistry Factor for the lower shell forging 04 was determined to be non-conservative. Hence, the adjusted reference temperature (ART) must be calculated using Position 2.1 along with the full margin term. Both Regulatory Guide 1.99, Revision 2, Position 1.1 and 2.1 have been shown herein. Contained in Tables 8 and 9 are the calculations of the 32 EFPY ART values used for generation of the heatup and cooldown curves.

TABLE 8
Calculation of the ART Values for the 1/4T Location @ 32 EFPY

| Material | RG 1.99 R2 Method | CF (°F) | FF | $IRT_{NDT}$[(1)] (°F) | $\Delta RT_{NDT}$[(3)] (°F) | Margin[(4)] (°F) | ART[(2)] (°F) |
|---|---|---|---|---|---|---|---|
| Intermediate Shell Forging 05 | Position 1.1 | 115.6 | 1.029 | 40 | 119.0 | 34 | 193 |
| Lower Shell Forging 04 | Position 1.1 | 95 | 1.029 | 73 | 97.8 | 34 | 205 |
| | Position 2.1 | 105.9 | 1.029 | 73 | 109.0 | 34 | 216 |
| Intermediate to Lower Shell | Position 1.1 | 161.3 | 1.029 | -40 | 166.0 | 56 | 182 |
| Circumferential Weld Seam | Position 2.1 | 135.0 | 1.029 | -40 | 138.9 | 56 | 155 |

Notes:
(1) Initial $RT_{NDT}$ values measured values.
(2) ART = Initial $RT_{NDT}$ + $\Delta RT_{NDT}$ + Margin (°F)
(3) $\Delta RT_{NDT} = CF * FF$
(4) $M = 2 * (\sigma_i^2 + \sigma_\Delta^2)^{1/2}$

TABLE 9
Calculation of the ART Values for the 3/4T Location @ 32 EFPY

| Material | RG 1.99 R2 Method | CF (°F) | FF | $IRT_{NDT}$[(1)] (°F) | $\Delta RT_{NDT}$[(3)] (°F) | Margin[(4)] (°F) | ART[(2)] (°F) |
|---|---|---|---|---|---|---|---|
| Intermediate Shell Forging 05 | Position 1.1 | 115.6 | 0.747 | 40 | 86.4 | 34 | 160 |
| Lower Shell Forging 04 | Position 1.1 | 95 | 0.747 | 73 | 71.0 | 34 | 178 |
| | Position 2.1 | 105.9 | 0.747 | 73 | 79.1 | 34 | 186 |
| Intermediate to Lower Shell | Position 1.1 | 161.3 | 0.747 | -40 | 120.5 | 56 | 137 |
| Circumferential Weld Seam | Position 2.1 | 135.0 | 0.747 | -40 | 100.8 | 56 | 117 |

Notes:
(1) Initial $RT_{NDT}$ values measured values.
(2) ART = Initial $RT_{NDT}$ + $\Delta RT_{NDT}$ + Margin (°F)
(3) $\Delta RT_{NDT} = CF * FF$
(4) $M = 2 * (\sigma_i^2 + \sigma_\Delta^2)^{1/2}$

The lower shell forging 04 is the limiting beltline material for the 1/4T and 3/4T case (See Tables 8 and 9). Contained in Table 10 is a summary of the limiting ARTs to be used in the generation of the Sequoyah Unit 1 reactor vessel heatup and cooldown curves.

TABLE 10
Summary of the Limiting ART Values Used in the Generation of the Sequoyah Unit 1 Heatup/Cooldown Curves

| EFPY | 1/4T Limiting ART | 3/4T Limiting ART |
|---|---|---|
| 32 | 216°F | 186°F |

# 5 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 Sections 3.0 and 4.0 of this report. This approved methodology is also presented in WCAP-14040-NP-A, Revision 2 with exception to those items discussed in Section 1 of this report.

Figures 1, 2, 4 and 5 present the heatup curves with (10°F and 60 psig) and without margins for possible instrumentation errors using heatup rates of 60 and 100°F/hr applicable for the first 32 EFPY. Figures 3 and 6 present the cooldown curves with (10°F and 60 psig) and without margins for possible instrumentation errors using cooldown rates of 0, 20, 40, 60 and 100°F/hr applicable for 32 EFPY. 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 6. This is in addition to other criteria which must be met before the reactor is made critical, as discussed below in the following paragraphs.

The reactor must not be made critical until pressure-temperature combinations are to the right of the criticality limit line shown in Figures 1, 2, 4 and 5. 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 10 CFR Part 50. The governing equation for the hydrostatic test is defined in Code Case N-640[3] (approved in February 1999) as follows:

$$1.5\ K_{Im} < K_{Ic}$$

where,

> $K_{Im}$ is the stress intensity factor covered by membrane (pressure) stress,
> $K_{Ic} = 33.2 + 20.734\ e^{[0.02\ (T - RT_{NDT})]}$,
> T is the minimum permissible metal temperature, and
> $RT_{NDT}$ is the metal reference nil-ductility temperature.

The criticality limit curve specifies pressure-temperature limits for core operation to provide additional margin during actual power production as specified in Reference 10. The pressure-temperature limits for core operation (except for low power physics tests) are that the reactor vessel must be at a temperature equal to or higher than the minimum temperature required for the inservice hydrostatic test, and at least 40°F higher than the minimum permissible temperature in the corresponding pressure-temperature curve for heatup and cooldown calculated as described in Section 3.0 of this report. For the heatup and cooldown curves without margins for instrumentation errors, the minimum temperature for the in service hydrostatic leak tests for the Sequoyah Unit 1 reactor vessel at 32 EFPY is 277°F. The vertical line drawn from these points on the pressure-temperature curve, intersecting a curve 40°F higher than the pressure-temperature limit curve, constitutes the limit for core operation for the reactor vessel.

Figures 1 through 6 define all of the above limits for ensuring prevention of nonductile failure for the Sequoyah Unit 1 reactor vessel. The data points used for the heatup and cooldown pressure-temperature limit curves shown in Figures 1 through 6 are presented in Tables 11 and 14.

MATERIAL PROPERTY BASIS

LIMITING MATERIAL: LOWER SHELL FORGING 04
LIMITING ART VALUES AT 32 EFPY: ¼T, 216°F
¾T, 186°F

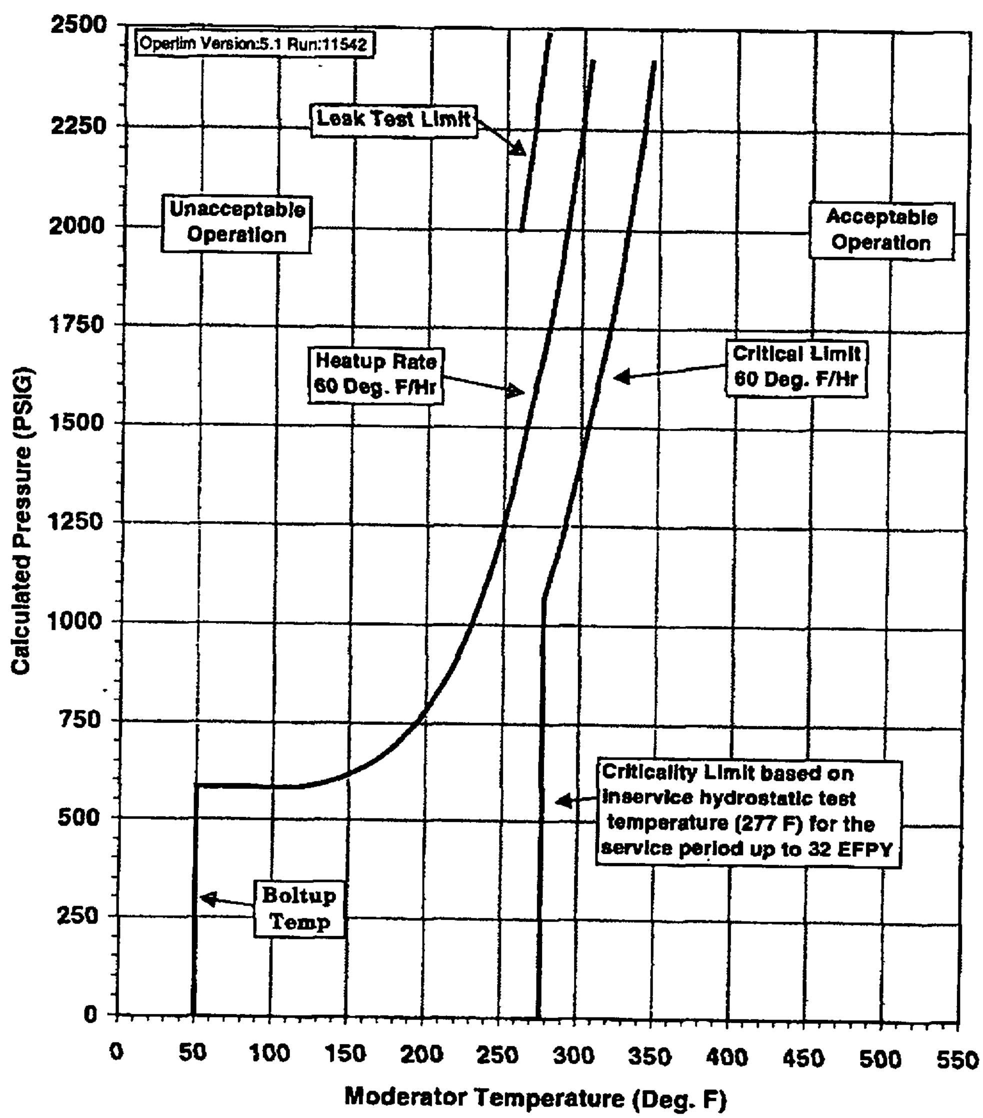


FIGURE 1 Sequoyah Unit 1 Reactor Coolant System Heatup Limitations (Heatup Rate of 60°F/hr) Applicable for the First 32 EFPY (Without Margins for Instrumentation Errors)

MATERIAL PROPERTY BASIS

LIMITING MATERIAL: LOWER SHELL FORGING 04
LIMITING ART VALUES AT 32 EFPY: ¼T, 216°F
¾T, 186°F

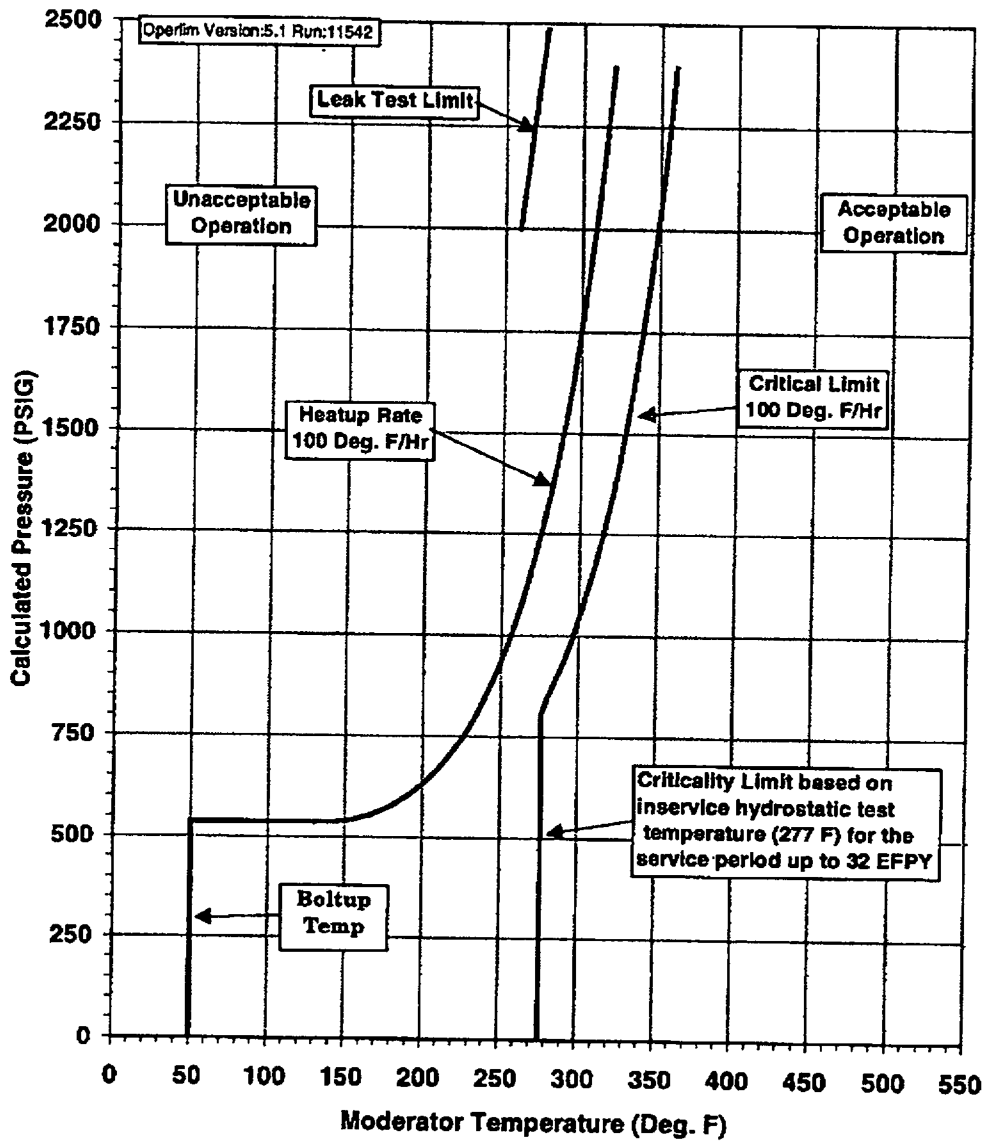


FIGURE 2 Sequoyah Unit 1 Reactor Coolant System Heatup Limitations (Heatup Rate of 100°F/hr) Applicable for the First 32 EFPY (Without Margins for Instrumentation Errors)

MATERIAL PROPERTY BASIS

LIMITING MATERIAL: LOWER SHELL FORGING 04
LIMITING ART VALUES AT 32 EFPY: 1/4T, 216°F
3/4T, 186°F

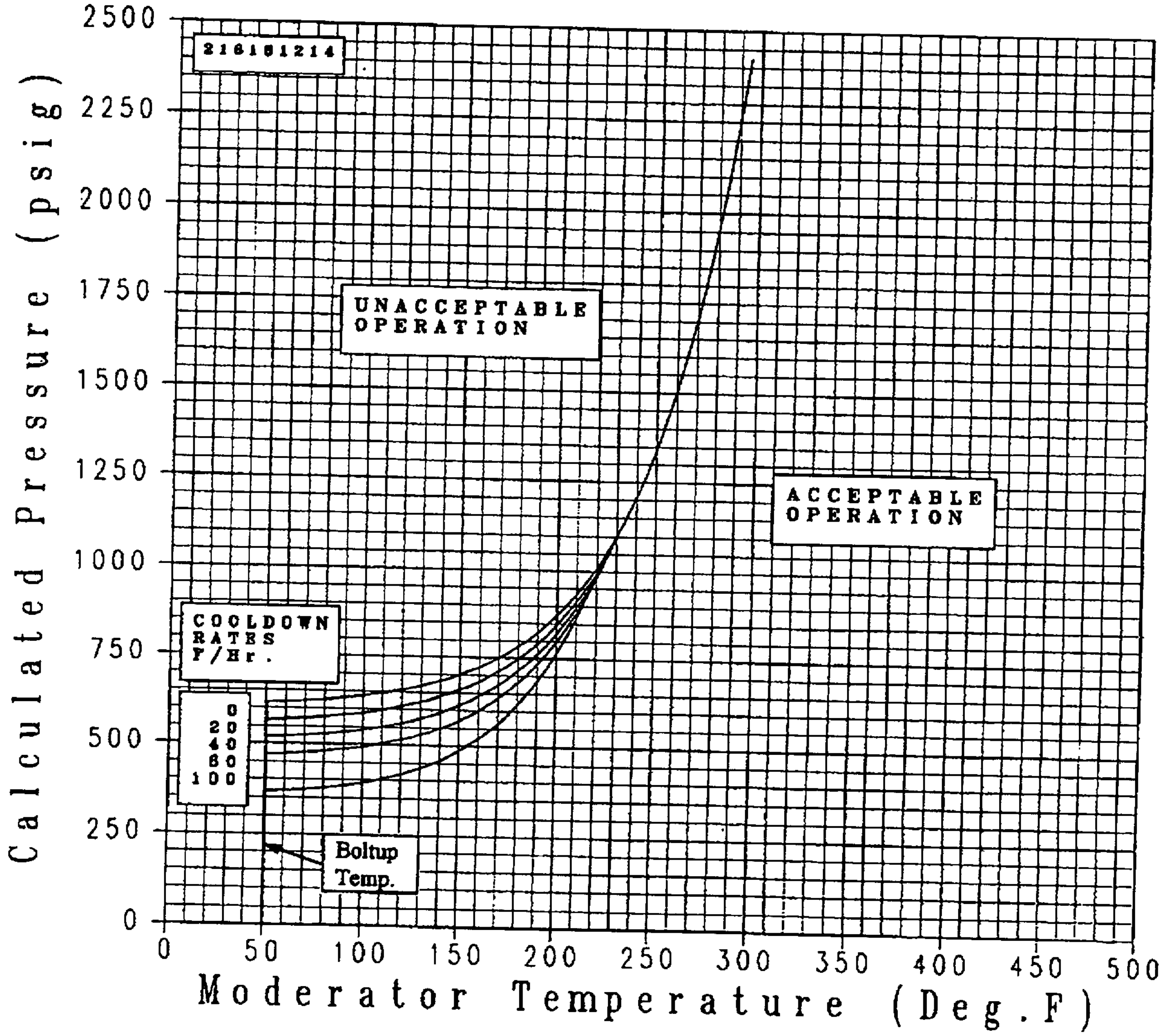


**Figure 3** **Sequoyah Unit 1 Reactor Coolant System Cooldown Limitations (Cooldown Rates up to 100°F/hr) Applicable for the First 32 EFPY (Without Margins for Instrumentation Errors)**

MATERIAL PROPERTY BASIS

LIMITING MATERIAL: LOWER SHELL FORGING 04
LIMITING ART VALUES AT 32 EFPY: ¼T, 216°F
¾T, 186°F

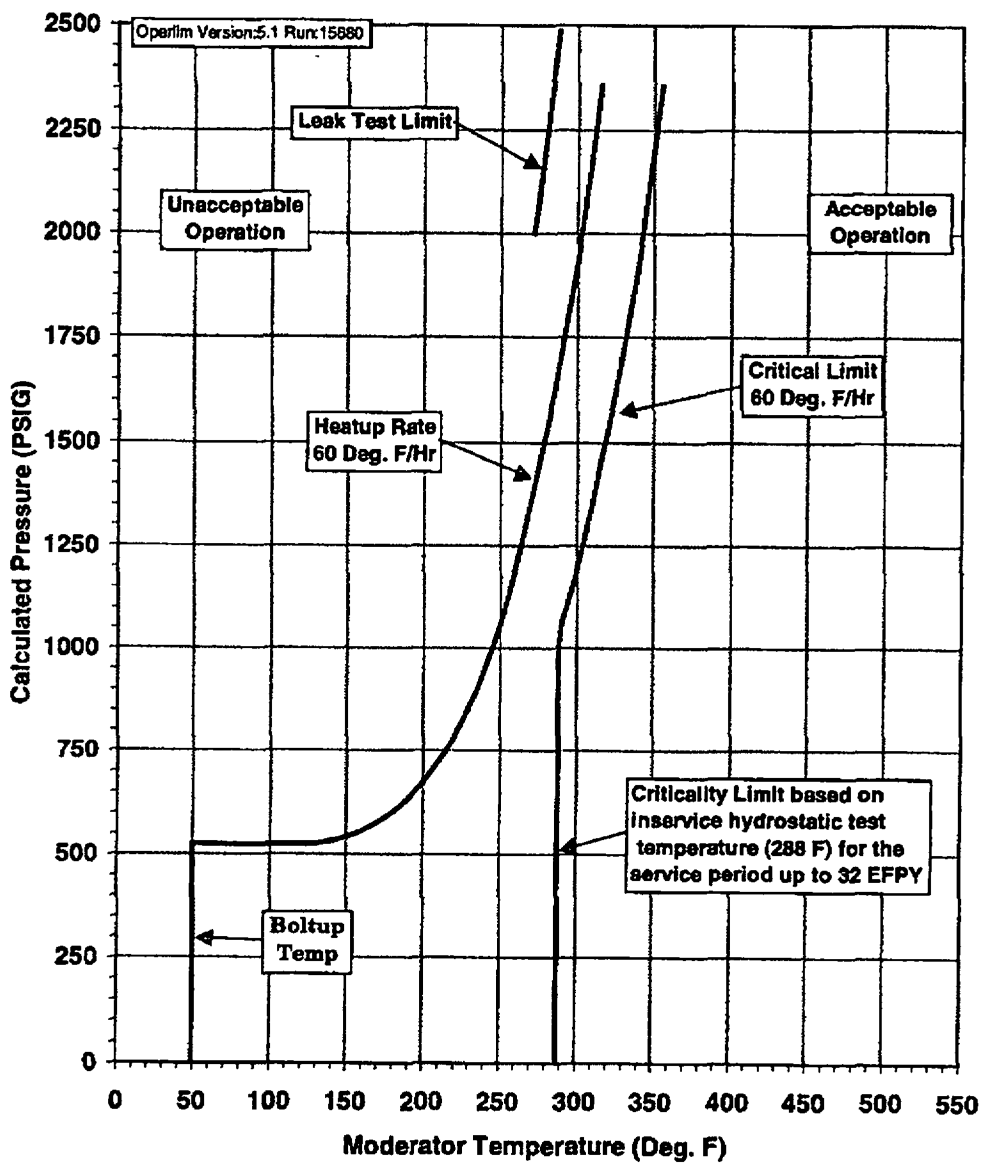


FIGURE 4 Sequoyah Unit 1 Reactor Coolant System Heatup Limitations (Heatup Rate of 60°F/hr) Applicable for the First 32 EFPY (With Margins for Instrumentation Errors of 10°F and 60 psig)

MATERIAL PROPERTY BASIS

LIMITING MATERIAL: LOWER SHELL FORGING 04
LIMITING ART VALUES AT 32 EFPY: ¼T, 216°F
¾T, 186°F

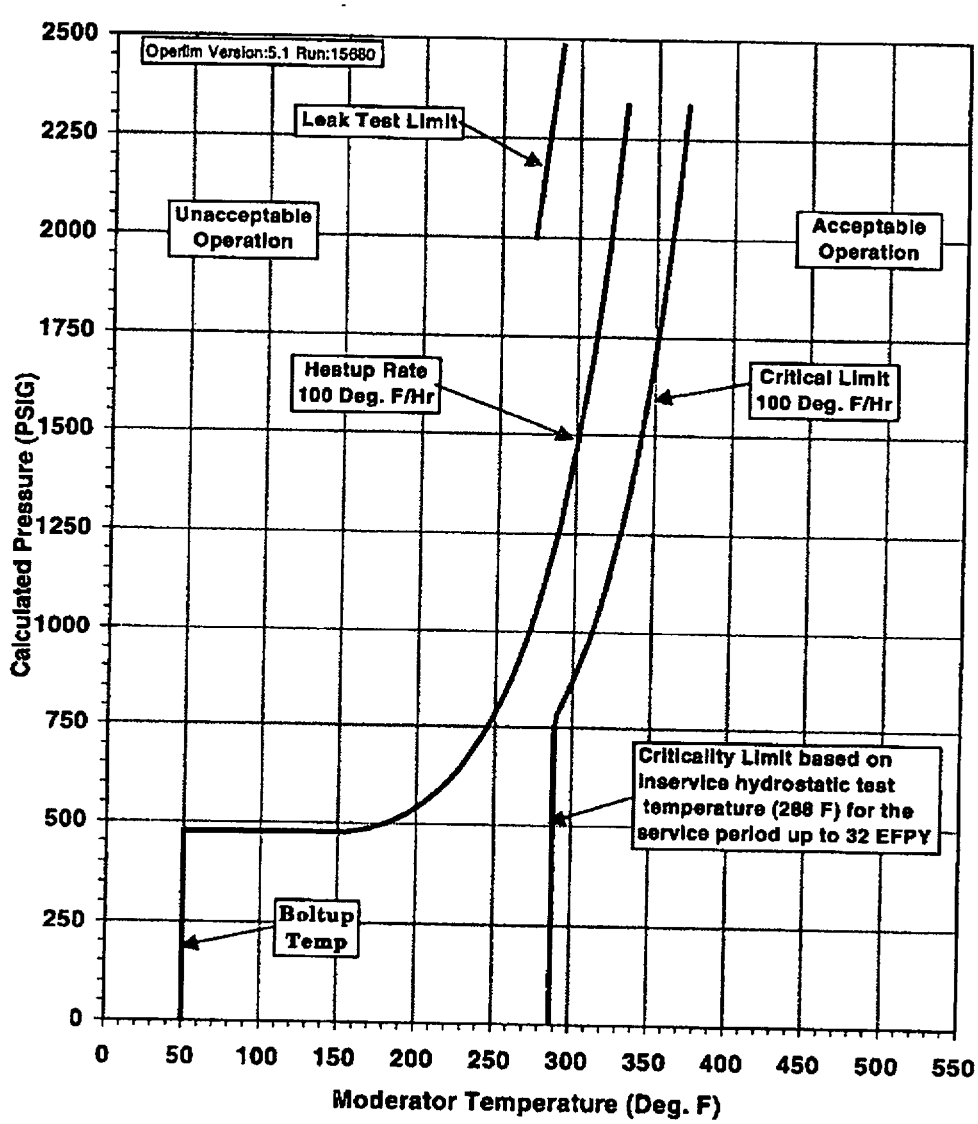


FIGURE 5 Sequoyah Unit 1 Reactor Coolant System Heatup Limitations (Heatup Rate of 100°F/hr) Applicable for the First 32 EFPY (With Margins for Instrumentation Errors of 10°F and 60 psig)

MATERIAL PROPERTY BASIS

LIMITING MATERIAL: LOWER SHELL FORGING 04
LIMITING ART VALUES AT 32 EFPY: 1/4T, 216°F
3/4T, 186°F

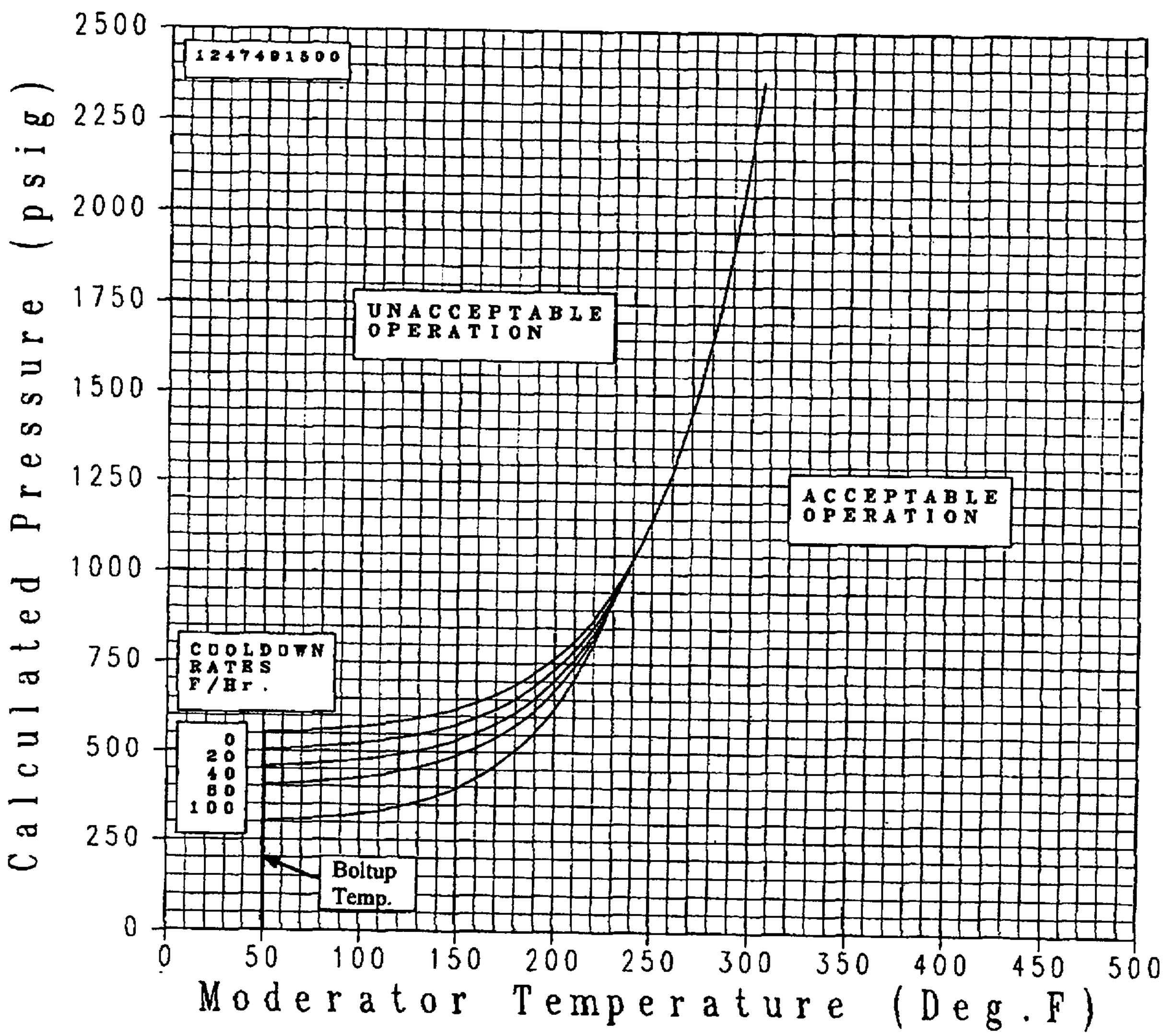


**Figure 6** **Sequoyah Unit 1 Reactor Coolant System Cooldown Limitations (Cooldown Rates up to 100°F/hr) Applicable for the First 32 EFPY (With Margins for Instrumentation Errors of 10°F and 60 psig)**

TABLE 11
32 EFPY Heatup Curve Data Points Using 1996 App.G
(without Uncertainties for Instrumentation Errors)

| Heatup Curves | | | | | | | | | |
|---|---|---|---|---|---|---|---|---|---|
| 60 Heatup | | 60 Limit Critical | | 100 Heatup | | 100 Limit Critical | | Leak Test Limit | |
| T | P | T | P | T | P | T | P | T | P |
| 50 | 0 | 277 | 0 | 50 | 0 | 277 | 0 | 260 | 2000 |
| 50 | 585 | 277 | 585 | 50 | 537 | 277 | 537 | 277 | 2485 |
| 55 | 585 | 277 | 585 | 55 | 537 | 277 | 537 | | |
| 60 | 585 | 277 | 586 | 60 | 537 | 277 | 538 | | |
| 65 | 585 | 277 | 588 | 65 | 537 | 277 | 538 | | |
| 70 | 585 | 277 | 591 | 70 | 537 | 277 | 540 | | |
| 75 | 585 | 277 | 596 | 75 | 537 | 277 | 543 | | |
| 80 | 585 | 277 | 602 | 80 | 537 | 277 | 547 | | |
| 85 | 585 | 277 | 609 | 85 | 537 | 277 | 553 | | |
| 90 | 585 | 277 | 617 | 90 | 537 | 277 | 560 | | |
| 95 | 585 | 277 | 626 | 95 | 537 | 277 | 568 | | |
| 100 | 585 | 277 | 637 | 100 | 537 | 277 | 577 | | |
| 105 | 585 | 277 | 649 | 105 | 537 | 277 | 588 | | |
| 110 | 585 | 277 | 663 | 110 | 537 | 277 | 601 | | |
| 115 | 585 | 277 | 678 | 115 | 537 | 277 | 615 | | |
| 120 | 586 | 277 | 695 | 120 | 537 | 277 | 631 | | |
| 125 | 588 | 277 | 714 | 125 | 537 | 277 | 649 | | |
| 130 | 591 | 277 | 735 | 130 | 537 | 277 | 669 | | |
| 135 | 596 | 277 | 758 | 135 | 537 | 277 | 691 | | |
| 140 | 602 | 277 | 784 | 140 | 537 | 277 | 716 | | |
| 145 | 609 | 277 | 812 | 145 | 538 | 277 | 744 | | |
| 150 | 617 | 277 | 844 | 150 | 540 | 277 | 774 | | |
| 155 | 626 | 277 | 879 | 155 | 543 | 277 | 808 | | |
| 160 | 637 | 277 | 918 | 160 | 547 | 280 | 846 | | |
| 165 | 649 | 277 | 960 | 165 | 553 | 285 | 888 | | |
| 170 | 663 | 277 | 1008 | 170 | 560 | 290 | 934 | | |
| 175 | 678 | 277 | 1060 | 175 | 568 | 295 | 985 | | |
| 180 | 695 | 280 | 1117 | 180 | 577 | 300 | 1041 | | |
| 185 | 714 | 285 | 1181 | 185 | 588 | 305 | 1104 | | |
| 190 | 735 | 290 | 1251 | 190 | 601 | 310 | 1172 | | |
| 195 | 758 | 295 | 1329 | 195 | 615 | 315 | 1248 | | |
| 200 | 784 | 300 | 1414 | 200 | 631 | 320 | 1332 | | |
| 205 | 812 | 305 | 1509 | 205 | 649 | 325 | 1424 | | |
| 210 | 844 | 310 | 1591 | 210 | 669 | 330 | 1526 | | |
| 215 | 879 | 315 | 1677 | 215 | 691 | 335 | 1638 | | |
| 220 | 918 | 320 | 1772 | 220 | 716 | 340 | 1762 | | |
| 225 | 960 | 325 | 1877 | 225 | 744 | 345 | 1898 | | |
| 230 | 1008 | 330 | 1993 | 230 | 774 | 350 | 2048 | | |
| 235 | 1060 | 335 | 2120 | 235 | 808 | 355 | 2214 | | |
| 240 | 1117 | 340 | 2261 | 240 | 846 | 360 | 2397 | | |
| 245 | 1181 | 345 | 2417 | 245 | 888 | | | | |
| 250 | 1251 | | | 250 | 934 | | | | |

TABLE 11 (Continued)
32 EFPY Heatup Curve Data Points Using 1996 App.G
(without Uncertainties for Instrumentation Errors)

| Heatup Curves | | | | | | | | | |
|---|---|---|---|---|---|---|---|---|---|
| 60 Heatup | | 60 Limit Critical | | 100 Heatup | | 100 Limit Critical | | Leak Test Limit | |
| T | P | T | P | T | P | T | P | T | P |
| 255 | 1329 | | | 255 | 985 | | | | |
| 260 | 1414 | | | 260 | 1041 | | | | |
| 265 | 1509 | | | 265 | 1104 | | | | |
| 270 | 1591 | | | 270 | 1172 | | | | |
| 275 | 1677 | | | 275 | 1248 | | | | |
| 280 | 1772 | | | 280 | 1332 | | | | |
| 285 | 1877 | | | 285 | 1424 | | | | |
| 290 | 1993 | | | 290 | 1526 | | | | |
| 295 | 2120 | | | 295 | 1638 | | | | |
| 300 | 2261 | | | 300 | 1762 | | | | |
| 305 | 2417 | | | 305 | 1898 | | | | |
| | | | | 310 | 2048 | | | | |
| | | | | 315 | 2214 | | | | |
| | | | | 320 | 2397 | | | | |

TABLE 12
32 EFPY Cooldown Curve Data Points Using 1996 App. G
(without Uncertainties for Instrumentation Errors)

| Cooldown Curves | | | | | | | | | |
|---|---|---|---|---|---|---|---|---|---|
| Steady State | | 20F | | 40F | | 60F | | 100F | |
| T | P | T | P | T | P | T | P | T | P |
| 50 | 0 | 50 | 0 | 50 | 0 | 50 | 0 | 50 | 0 |
| 50 | 615 | 50 | 567 | 50 | 519 | 50 | 470 | 50 | 367 |
| 55 | 616 | 55 | 569 | 55 | 520 | 55 | 471 | 55 | 368 |
| 60 | 618 | 60 | 570 | 60 | 522 | 60 | 472 | 60 | 369 |
| 65 | 620 | 65 | 572 | 65 | 524 | 65 | 474 | 65 | 371 |
| 70 | 621 | 70 | 574 | 70 | 525 | 70 | 476 | 70 | 373 |
| 75 | 624 | 75 | 576 | 75 | 528 | 75 | 478 | 75 | 375 |
| 80 | 626 | 80 | 578 | 80 | 530 | 80 | 480 | 80 | 378 |
| 85 | 629 | 85 | 581 | 85 | 533 | 85 | 483 | 85 | 381 |
| 90 | 631 | 90 | 584 | 90 | 536 | 90 | 486 | 90 | 385 |
| 95 | 635 | 95 | 587 | 95 | 539 | 95 | 490 | 95 | 389 |
| 100 | 638 | 100 | 591 | 100 | 543 | 100 | 494 | 100 | 393 |
| 105 | 642 | 105 | 595 | 105 | 547 | 105 | 498 | 105 | 398 |
| 110 | 646 | 110 | 600 | 110 | 552 | 110 | 503 | 110 | 404 |
| 115 | 651 | 115 | 605 | 115 | 557 | 115 | 509 | 115 | 411 |
| 120 | 656 | 120 | 610 | 120 | 563 | 120 | 516 | 120 | 418 |
| 125 | 662 | 125 | 616 | 125 | 570 | 125 | 523 | 125 | 427 |
| 130 | 668 | 130 | 623 | 130 | 577 | 130 | 531 | 130 | 436 |
| 135 | 676 | 135 | 631 | 135 | 585 | 135 | 539 | 135 | 447 |
| 140 | 683 | 140 | 639 | 140 | 594 | 140 | 549 | 140 | 459 |
| 145 | 692 | 145 | 648 | 145 | 604 | 145 | 560 | 145 | 472 |
| 150 | 702 | 150 | 659 | 150 | 616 | 150 | 572 | 150 | 487 |
| 155 | 712 | 155 | 670 | 155 | 628 | 155 | 586 | 155 | 503 |
| 160 | 724 | 160 | 683 | 160 | 642 | 160 | 601 | 160 | 521 |
| 165 | 737 | 165 | 697 | 165 | 657 | 165 | 618 | 165 | 542 |
| 170 | 751 | 170 | 712 | 170 | 674 | 170 | 637 | 170 | 565 |
| 175 | 767 | 175 | 729 | 175 | 693 | 175 | 657 | 175 | 590 |
| 180 | 784 | 180 | 748 | 180 | 714 | 180 | 680 | 180 | 618 |
| 185 | 803 | 185 | 769 | 185 | 737 | 185 | 706 | 185 | 650 |
| 190 | 824 | 190 | 793 | 190 | 762 | 190 | 734 | 190 | 684 |
| 195 | 848 | 195 | 819 | 195 | 791 | 195 | 765 | 195 | 723 |
| 200 | 874 | 200 | 847 | 200 | 822 | 200 | 800 | 200 | 766 |

TABLE 12 - (Continued)
32 EFPY Cooldown Curve Data Points Using 1996 App. G
(without Uncertainties for Instrumentation Errors)

| Cooldown Curves | | | | | | | | | |
|---|---|---|---|---|---|---|---|---|---|
| Steady State | | 20F | | 40F | | 60F | | 100F | |
| T | P | T | P | T | P | T | P | T | P |
| 205 | 903 | 205 | 879 | 205 | 857 | 205 | 839 | 205 | 814 |
| 210 | 934 | 210 | 913 | 210 | 896 | 210 | 881 | 210 | 866 |
| 215 | 969 | 215 | 952 | 215 | 938 | 215 | 929 | 215 | 925 |
| 220 | 1008 | 220 | 995 | 220 | 985 | 220 | 981 | 220 | 990 |
| 225 | 1051 | 225 | 1042 | 225 | 1038 | 225 | 1039 | | |
| 230 | 1098 | 230 | 1094 | 230 | 1096 | | | | |
| 235 | 1150 | | | | | | | | |
| 240 | 1208 | | | | | | | | |
| 245 | 1272 | | | | | | | | |
| 250 | 1343 | | | | | | | | |
| 255 | 1420 | | | | | | | | |
| 260 | 1507 | | | | | | | | |
| 265 | 1602 | | | | | | | | |
| 270 | 1707 | | | | | | | | |
| 275 | 1823 | | | | | | | | |
| 280 | 1952 | | | | | | | | |
| 285 | 2094 | | | | | | | | |
| 290 | 2251 | | | | | | | | |
| 295 | 2424 | | | | | | | | |

TABLE 13
32 EFPY Heatup Curve Data Points Using 1996 App.G
(with Uncertainties for Instrumentation Errors of 10°F and 60 psig)

| Heatup Curves | | | | | | | | | |
|---|---|---|---|---|---|---|---|---|---|
| 60 Heatup | | 60 Limit Critical | | 100 Heatup | | 100 Limit Critical | | Leak Test Limit | |
| T | P | T | P | T | P | T | P | T | P |
| 50 | 0 | 288 | 0 | 50 | 0 | 288 | 0 | 272 | 2000 |
| 50 | 525 | 288 | 525 | 50 | 477 | 288 | 477 | 288 | 2485 |
| 55 | 525 | 288 | 525 | 55 | 477 | 288 | 477 | | |
| 60 | 525 | 288 | 525 | 60 | 477 | 288 | 477 | | |
| 65 | 525 | 288 | 525 | 65 | 477 | 288 | 477 | | |
| 70 | 525 | 288 | 526 | 70 | 477 | 288 | 478 | | |
| 75 | 525 | 288 | 528 | 75 | 477 | 288 | 478 | | |
| 80 | 525 | 288 | 531 | 80 | 477 | 288 | 480 | | |
| 85 | 525 | 288 | 531 | 85 | 477 | 288 | 483 | | |
| 90 | 525 | 288 | 536 | 90 | 477 | 288 | 487 | | |
| 95 | 525 | 288 | 542 | 95 | 477 | 288 | 493 | | |
| 100 | 525 | 288 | 549 | 100 | 477 | 288 | 500 | | |
| 105 | 525 | 288 | 557 | 105 | 477 | 288 | 508 | | |
| 110 | 525 | 288 | 566 | 110 | 477 | 288 | 517 | | |
| 115 | 525 | 288 | 577 | 115 | 477 | 288 | 528 | | |
| 120 | 525 | 288 | 589 | 120 | 477 | 288 | 541 | | |
| 125 | 525 | 288 | 603 | 125 | 477 | 288 | 541 | | |
| 130 | 526 | 288 | 618 | 130 | 477 | 288 | 555 | | |
| 135 | 528 | 288 | 635 | 135 | 477 | 288 | 571 | | |
| 140 | 531 | 288 | 654 | 140 | 477 | 288 | 589 | | |
| 145 | 536 | 288 | 675 | 145 | 477 | 288 | 609 | | |
| 150 | 542 | 288 | 698 | 150 | 477 | 288 | 631 | | |
| 155 | 549 | 288 | 724 | 155 | 478 | 288 | 656 | | |
| 160 | 557 | 288 | 752 | 160 | 480 | 288 | 684 | | |
| 165 | 566 | 288 | 784 | 165 | 483 | 288 | 714 | | |
| 170 | 577 | 288 | 819 | 170 | 487 | 288 | 748 | | |
| 175 | 589 | 288 | 858 | 175 | 493 | 290 | 786 | | |
| 180 | 603 | 288 | 900 | 180 | 500 | 295 | 828 | | |
| 185 | 618 | 288 | 948 | 185 | 508 | 300 | 874 | | |
| 190 | 635 | 288 | 1000 | 190 | 517 | 305 | 925 | | |
| 195 | 654 | 290 | 1057 | 195 | 528 | 310 | 981 | | |
| 200 | 675 | 295 | 1121 | 200 | 541 | 315 | 1044 | | |
| 205 | 698 | 300 | 1191 | 205 | 555 | 320 | 1112 | | |
| 210 | 724 | 305 | 1269 | 210 | 571 | 325 | 1188 | | |
| 215 | 752 | 310 | 1354 | 215 | 589 | 330 | 1272 | | |
| 220 | 784 | 315 | 1449 | 220 | 609 | 335 | 1364 | | |
| 225 | 819 | 320 | 1531 | 225 | 631 | 340 | 1466 | | |
| 230 | 858 | 325 | 1617 | 230 | 656 | 345 | 1578 | | |
| 235 | 900 | 330 | 1712 | 235 | 684 | 350 | 1702 | | |
| 240 | 948 | 335 | 1817 | 240 | 714 | 355 | 1838 | | |
| 245 | 1000 | 340 | 1933 | 245 | 748 | 360 | 1988 | | |
| 250 | 1057 | 345 | 2060 | 250 | 786 | 365 | 2154 | | |

TABLE 13 (Continued)
32 EFPY Heatup Curve Data Points Using 1996 App.G
(with Uncertainties for Instrumentation Errors of 10°F and 60 psig)

| Heatup Curves | | | | | | | | | |
|---|---|---|---|---|---|---|---|---|---|
| 60 Heatup | | 60 Limit Critical | | 100 Heatup | | 100 Limit Critical | | Leak Test Limit | |
| T | P | T | P | T | P | T | P | T | P |
| 255 | 1121 | 350 | 2201 | 255 | 828 | 370 | 2337 | | |
| 260 | 1191 | 355 | 2357 | 260 | 874 | | | | |
| 265 | 1269 | | | 265 | 925 | | | | |
| 270 | 1354 | | | 270 | 981 | | | | |
| 275 | 1449 | | | 275 | 1044 | | | | |
| 280 | 1531 | | | 280 | 1112 | | | | |
| 285 | 1617 | | | 285 | 1188 | | | | |
| 290 | 1712 | | | 290 | 1272 | | | | |
| 295 | 1817 | | | 295 | 1364 | | | | |
| 300 | 1933 | | | 300 | 1466 | | | | |
| 305 | 2060 | | | 305 | 1578 | | | | |
| 310 | 2201 | | | 310 | 1702 | | | | |
| 315 | 2357 | | | 315 | 1838 | | | | |
| | | | | 320 | 1988 | | | | |
| | | | | 325 | 2154 | | | | |
| | | | | 330 | 2337 | | | | |

TABLE 14
32 EFPY Cooldown Curve Data Points Using 1996 App. G
(with Uncertainties for Instrumentation Errors of 10°F and 60 psig)

| Cooldown Curves | | | | | | | | | |
|---|---|---|---|---|---|---|---|---|---|
| Steady State | | 20F | | 40F | | 60F | | 100F | |
| T | P | T | P | T | P | T | P | T | P |
| 50 | 0 | 50 | 0 | 50 | 0 | 50 | 0 | 50 | 0 |
| 50 | 552 | 50 | 503 | 50 | 457 | 50 | 408 | 50 | 305 |
| 55 | 553 | 55 | 505 | 55 | 458 | 55 | 409 | 55 | 306 |
| 60 | 555 | 60 | 507 | 60 | 459 | 60 | 410 | 60 | 307 |
| 65 | 556 | 65 | 509 | 65 | 460 | 65 | 411 | 65 | 308 |
| 70 | 558 | 70 | 510 | 70 | 462 | 70 | 412 | 70 | 309 |
| 75 | 560 | 75 | 512 | 75 | 464 | 75 | 414 | 75 | 311 |
| 80 | 561 | 80 | 514 | 80 | 465 | 80 | 416 | 80 | 313 |
| 85 | 564 | 85 | 516 | 85 | 468 | 85 | 418 | 85 | 315 |
| 90 | 566 | 90 | 518 | 90 | 470 | 90 | 420 | 90 | 318 |
| 95 | 569 | 95 | 521 | 95 | 473 | 95 | 423 | 95 | 321 |
| 100 | 571 | 100 | 524 | 100 | 476 | 100 | 426 | 100 | 325 |
| 105 | 575 | 105 | 527 | 105 | 479 | 105 | 430 | 105 | 329 |
| 110 | 578 | 110 | 531 | 110 | 483 | 110 | 434 | 110 | 333 |
| 115 | 582 | 115 | 535 | 115 | 487 | 115 | 438 | 115 | 338 |
| 120 | 586 | 120 | 540 | 120 | 492 | 120 | 443 | 120 | 344 |
| 125 | 591 | 125 | 545 | 125 | 497 | 125 | 449 | 125 | 351 |
| 130 | 596 | 130 | 550 | 130 | 503 | 130 | 456 | 130 | 358 |
| 135 | 602 | 135 | 556 | 135 | 510 | 135 | 463 | 135 | 367 |
| 140 | 608 | 140 | 563 | 140 | 517 | 140 | 471 | 140 | 376 |
| 145 | 616 | 145 | 571 | 145 | 525 | 145 | 479 | 145 | 387 |
| 150 | 623 | 150 | 579 | 150 | 534 | 150 | 489 | 150 | 399 |
| 155 | 632 | 155 | 588 | 155 | 544 | 155 | 500 | 155 | 412 |
| 160 | 642 | 160 | 599 | 160 | 556 | 160 | 512 | 160 | 427 |
| 165 | 652 | 165 | 610 | 165 | 568 | 165 | 526 | 165 | 443 |
| 170 | 664 | 170 | 623 | 170 | 582 | 170 | 541 | 170 | 461 |
| 175 | 677 | 175 | 637 | 175 | 597 | 175 | 558 | 175 | 482 |
| 180 | 691 | 180 | 652 | 180 | 614 | 180 | 577 | 180 | 505 |
| 185 | 707 | 185 | 669 | 185 | 633 | 185 | 597 | 185 | 530 |
| 190 | 724 | 190 | 688 | 190 | 654 | 190 | 620 | 190 | 558 |
| 195 | 743 | 195 | 709 | 195 | 677 | 195 | 646 | 195 | 590 |
| 200 | 764 | 200 | 733 | 200 | 702 | 200 | 674 | 200 | 624 |
| 205 | 788 | 205 | 759 | 205 | 731 | 205 | 705 | 205 | 663 |
| 210 | 814 | 210 | 787 | 210 | 762 | 210 | 740 | 210 | 706 |
| 215 | 843 | 215 | 819 | 215 | 797 | 215 | 779 | 215 | 754 |

TABLE 14 - (Continued)
32 EFPY Cooldown Curve Data Points Using 1996 App. G
(with Uncertainties for Instrumentation Errors of 10°F and 60 psig)

| Cooldown Curves | | | | | | | | | |
|---|---|---|---|---|---|---|---|---|---|
| Steady State | | 20F | | 40F | | 60F | | 100F | |
| T | P | T | P | T | P | T | P | T | P |
| 220 | 874 | 220 | 853 | 220 | 836 | 220 | 821 | 220 | 806 |
| 225 | 909 | 225 | 892 | 225 | 878 | 225 | 869 | 225 | 865 |
| 230 | 948 | 230 | 935 | 230 | 925 | 230 | 921 | 230 | 930 |
| 235 | 991 | 235 | 982 | 235 | 978 | 235 | 979 | | |
| 240 | 1038 | 240 | 1034 | 240 | 1036 | | | | |
| 245 | 1090 | | | | | | | | |
| 250 | 1148 | | | | | | | | |
| 255 | 1212 | | | | | | | | |
| 260 | 1283 | | | | | | | | |
| 265 | 1360 | | | | | | | | |
| 270 | 1447 | | | | | | | | |
| 275 | 1542 | | | | | | | | |
| 280 | 1647 | | | | | | | | |
| 285 | 1763 | | | | | | | | |
| 290 | 1892 | | | | | | | | |
| 295 | 2034 | | | | | | | | |
| 300 | 2191 | | | | | | | | |
| 305 | 2364 | | | | | | | | |

# 6 REFERENCES


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

2. WCAP-14040-NP-A, Revision 2, "Methodology used to Develop Cold Overpressure Mitigating system Setpoints and RCS Heatup and Cooldown Limit Curves", J.D. Andrachek, et. al., January 1996.

3. ASME Code Case N-640, "Alternative Reference Fracture Toughness for Development of P-T Limit Curves for Section XI, Division 1", February 26, 1999.

4. Section XI of the ASME Boiler and Pressure Vessel Code, Appendix G, "Fracture Toughness Criteria for Protection Against Failure.", Dated 1989 & December 1995.

5. "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.

6. WCAP-10340, Revision 1, "Analysis of Capsule T from the Tennessee Valley Authority Sequoyah Unit 1 Reactor Vessel Radiation Surveillance Program", S.E. Yanichko, et. al., February 1984.

7. WCAP-15224, "Analysis of Capsule Y From the Tennessee Valley Authority Sequoyah Unit 1 Reactor Vessel Radiation Surveillance Program", T.J. Laubham, et. al., Dated June 1999.

8. 1989 Section III, Division 1 of the ASME Boiler and Pressure Vessel Code, Paragraph NB-2331, "Material for Vessels."

9. CVGRAPH, Hyperbolic Tangent Curve-Fitting Program, Version 4.1, developed by ATI Consulting, March 1999.

10. Code of Federal Regulations, 10 CFR Part 50, Appendix G, "Fracture Toughness Requirements," U.S. Nuclear Regulatory Commission, Washington, D.C., Federal Register, Volume 60, No. 243, dated December 19, 1995.

11. WCAP-7924-A, "Basis for Heatup and Cooldown Limit Curves," W. S. Hazelton, et al., April 1975.

12. Calc. No. 92-016, "WOG USE Program - Onset of Upper Shelf Energy Calculations", J. M. Chicots, dated 11/12/92. File # WOG-108/4-18 (MUHP-5080).

13. WCAP-12970, "Heatup and Cooldown Limit Curves for Normal Operation Sequoyah Unit 1", J.M. Chicots, et. al., Dated June, 1991.

14. Westinghouse Letter TVA-91-242

15. Westinghouse Letter TVA-91-243

16. TVA letter No. N9664, "TASK N99-017 - Reactor Coolant System Pressure and Temperature Limit Report Development - N2N-048," W. M. Justice, August 17, 1999.

17. TVA letter No. N9667, "TASK N99-017 - Reactor Coolant System Pressure and Temperature Limit Report Development - N2N-048," W. M. Justice, August 20, 1999.

18. WCAP-15315, "Reactor Vessel Closure Head / Vessel Flange Requirements Evaluation for Operating PWR and BWR Plants", W. Bamford, et. al, October 1999.

19. Westinghouse Letter TVA-93-105.

# APPENDIX A

# LTOPS SETPOINTS

## 1.1 INTRODUCTION

Westinghouse has been requested to develop Low Temperature Overpressure Protection System (LTOPS) setpoints for Sequoyah Unit 1, for a vessel exposure of 32 EFPY. The LTOPS setpoints for the Sequoyah, Units 1 and 2, were last revised by Westinghouse in June of 1991 using pressure-temperature limits supplied by Tennessee Valley Authority for a vessel exposure of 16 EFPY. The results of this analysis were reported via References 14 and 15. The June 1991 analysis was based on the September 1989 analysis which addressed: Eagle-21 implementation and Appendix G limits based on Regulatory Guide 1.99. This Section documents the development of new Sequoyah Unit 1 COMS setpoints for 32 EFPY.

The Low Temperature Overpressure Protection System (LTOPS) is designed to provide the capability, during relatively low temperatures Reactor Coolant System (RCS) operation (typically less than 350°F), to protect the reactor vessel from being exposed to conditions of fast propagating brittle fracture. The LTOPS is provided in addition to the administrative controls, to prevent overpressure transients and as a supplement to the RCS overpressure mitigation function of the Residual Heat Removal System (RHRS) relief valves. LTOP consists of pressurizer PORVs and actuation logic from the wide range pressure channels. Once the system is enabled, no operator action is involved for the LTOPS to perform its intended pressure mitigation function.

LTOPS setpoints are conservatively selected to prevent exceeding the pressure/temperature limits established by 10 CFR Part 50 Appendix G requirements.

Two specific transients have been defined as the design basis for LTOPS. Each of these transient scenarios assume that the RCS is in a water-solid condition and that the RHRS is isolated from the RCS. The first transient consists of a heat injection scenario in which a reactor coolant pump in a single loop is started with the RCS temperature as much as 50°F lower than the steam generator secondary side temperature and the RHRS has been inadvertently isolated. This results in a sudden heat input to the RCS from the steam generators, creating an increasing pressure transient. The second transient has been defined as a mass injection scenario into the RCS caused by the simultaneous isolation of the RHRS, isolation of letdown and failure of the normal charging flow controls to the full flow condition. The resulting mass injection/letdown mismatch causes an increasing pressure transient.

## 1.2 LTOPS SETPOINT DETERMINATION

Westinghouse has developed new LTOPS setpoints for Sequoyah Unit 1, based on a vessel exposure of 32 EFPY using the methodology established in WCAP-14040 (Ref. 2). This methodology maximizes the available operating margin for setpoint selection while maintaining an appropriate level of protection in support of reactor vessel integrity. Note, Appendix G pressure limit relaxation allowed by ASME Code Case N-514 was not applied.

Plant design characteristics are unchanged (i.e., heat injection and mass injection transients characteristics and related plant responses have not been altered). Therefore, a complete reanalysis is not required. The new LTOPS setpoints were developed using results of the previous heat and mass injection transient analyses.

### 1.2.1 Pressure Limits Selection

The function of the LTOPS is to protect the reactor vessel from fast propagating brittle fracture. This has been implemented by choosing LTOPS setpoints, which prevent exceeding the limits prescribed by the applicable pressure/temperature characteristic for the specific reactor vessel material in accordance with rules given in Appendix G to 10CFR50. The LTOPS design basis takes credit for the fact that overpressure events most likely occur during isothermal conditions in the RCS. Therefore, it is appropriate to utilize the steady-state Appendix G limit. The Sequoyah Unit 1, 10CFR50 Appendix G curve for 32 EFPY is shown by Figure A-1. This curve sets the nominal upper limit on the pressure which should not be exceeded during RCS increasing pressure transients based on reactor vessel material properties.

When a relief valve is actuated to mitigate an increasing pressure transient, the system pressure then decreases, as the relief valve releases coolant, until a reset pressure is reached where the valve is signaled to close. Note that the pressure continues to decrease below the reset pressure as the valve re-closes. The nominal lower limit on the pressure during the transient is typically established based solely on an operational consideration for the RCP #1 seal to maintain a nominal differential pressure across the seal faces for proper film-riding performance. The RCP #1 seal limit is shown in Figure 1.

The nominal upper limit (based on the minimum of the steady-state 10CFR50 Appendix G requirement) and the nominal lower limit (based on RCP #1 seal performance criteria) create a pressure range from which the setpoints for both PORVs may be selected.

### 1.2.2 Mass Input Consideration

For a particular mass input transient to the RCS, the relief valve will be signaled to open at a specific pressure setpoint. However, there will be a pressure overshoot during the delay time before the valve starts to move and during the time the valve is moving to the full open position. This overshoot is dependent on the dynamics of the system and the input parameters, and results in a maximum system pressure somewhat higher than the set pressure. Similarly there will be a pressure undershoot, while the valve is relieving, both due to the reset pressure being below the setpoint and to the delay in stroking the valve closed.

The previous Sequoyah analyses of multiple mass input cases were used to determine the relationship between setpressures and resulting overshoots/undershoots.

### 1.2.3 Heat Input Consideration

The heat input case is done similarly to the mass input case except that the locus of transient pressure values versus selected setpoints may be determined for several values of the initial RCS temperature. This heat input evaluation provides a range of acceptable setpoints dependent on the reactor coolant temperature, whereas the mass input case is limited to the most restrictive low temperature condition only (i.e., the mass injection transient is not sensitive to temperature).

The previous Sequoyah analyses of multiple heat input cases were used to determine the relationship between setpressures and resulting overshoots/undershoots.

### 1.2.4 Final Setpoint Selection

Appendix G limits described in Section 1.2.1, were conservatively adjusted accounting for the pressure difference (ΔP) between the wide range pressure transmitter and the reactor vessel limiting beltline region of 68.3 psi for 4 RCPs in operation (See Reference 19).

The results of the analyses described in Section 1.2.2 & 1.2.3, and the adjusted Appendix G limit were used to define the maximum allowable setpoints for which the overpressure will not exceed the pressure limit applicable at a specific reactor vessel temperature. The maximum allowable setpoints are shown in Figure A-1.

Per Ref. 17, Sequoyah Demonstrated Accuracy Calculation SQN-IC014 establishes the instrument loop inaccuracy of the Sequoyah temperature and pressure instrument channels associated with the LTOPS. Previously, the LTOPS setpoints have been provided to TVA without application of instrument uncertainties. The TVA calculation quantified the instrument channel uncertainties, applied them to the nominal Westinghouse setpoints and evaluated the result against the safety limits established by the 10CFR50, Appendix G steady state heatup curve. The TVA demonstrated accuracy calculation will be revised to reflect the LTOPS setpoints calculated by Westinghouse under the subject task. As such, it is not necessary for Westinghouse to include instrument uncertainties in the nominal LTOPS setpoint calculation. Note, the heat injection results were adjusted to include 50°F thermal transport effect (difference in temperature between the RCS and steam generator at transient initiation).

The maximum allowable setpoints, adjusted to produce a smoother curve and reduced to nine data points, becomes the setpoints for PORV#2. A setpoint at a minimum temperature of 50°F was selected, as requested by TVA (Ref. 16). Each of the two PORVs may have a different pressure setpoint such that only one valve will open at a time and mitigate the transient (i.e., staggered setpoints). The second valve operates only if the first fails to open on command. This design supports a single failure assumption as well as minimizing the potential for both PORVs to open simultaneously, a condition which may create excessive pressure undershoot and challenge the RCP #1 seal performance criteria. The PORV#1 setpoints were selected by adjusting the setpoint PORV#2 in relationship to the overshoots discussed in Sections 1.2.2 and 1.2.3. The selected setpoints for PORV #1 and PORV #2 are shown in Table A-1 and Figure A-2. These setpoints were evaluated using the undershoots discussed in Sections 1.2.2 and 1.2.3 to ensure that they protect against the RCP # 1 seal limit.

In summary, the selection of the setpoints for LTOPS considered the use of nominal upper and lower pressure limits. The upper limits are specified by the minimum of the steady-state cooldown curve as calculated in accordance with Appendix G to 10CFR50 (adjusted to account for four RCPs in operation). The lower pressure extreme is specified by the reactor coolant pump #1 seal minimum differential pressure performance criteria. The selected setpoints, shown in Table A-1 and Figure A-2, provide protection against Appendix G, and RCP #1 seal limit violations. Note, these setpoints do not address instrumentation uncertainties.

## 1.3 ARMING AND ENABLE TEMPERATURES FOR LTOPS

The LTOPS arming temperature is traditionally based on the temperature corresponding to when Appendix G pressure equals 2500 psia. Based on this methodology the LTOPS arming temperature for Sequoyah conservatively continues to be 350°F.

The enable temperature is the temperature below which the LTOPS system is required to be operable, based on vessel materials concerns. ASME Code Case N-514 requires the LTOPS to be in operation at coolant temperatures less than 200°F or at coolant temperatures less than a temperature corresponding to a reactor vessel metal temperature less than $RT_{NDT}$ + 50°F, whichever is greater. $RT_{NDT}$ is the highest Adjusted Reference Temperature (ART) for the limiting belt-line material at a distance one fourth of the vessel section thickness from the vessel inside surface (i.e., clad/base metal interface), as determined by Regulatory Guide 1.99, Revision 2. The minimum required enable temperature for the Sequoyah Unit 1 Reactor Vessel is 295°F at 32 EFPY of operation.

Table A-1
Selected Setpoints, Sequoyah Unit 1

| Trcs (Deg.F) | PORV#2 Setpoint (psig) | PORV#1 Setpoint (psig) |
|---|---|---|
| 50 | 490 | 465 |
| 100 | 500 | 475 |
| 135 | 540 | 510 |
| 175 | 575 | 540 |
| 200 | 610 | 570 |
| 250 | 745 | 685 |
| 280 | 745 | 685 |
| 405 | 745 | 685 |
| 450 | 2350 | 2350 |

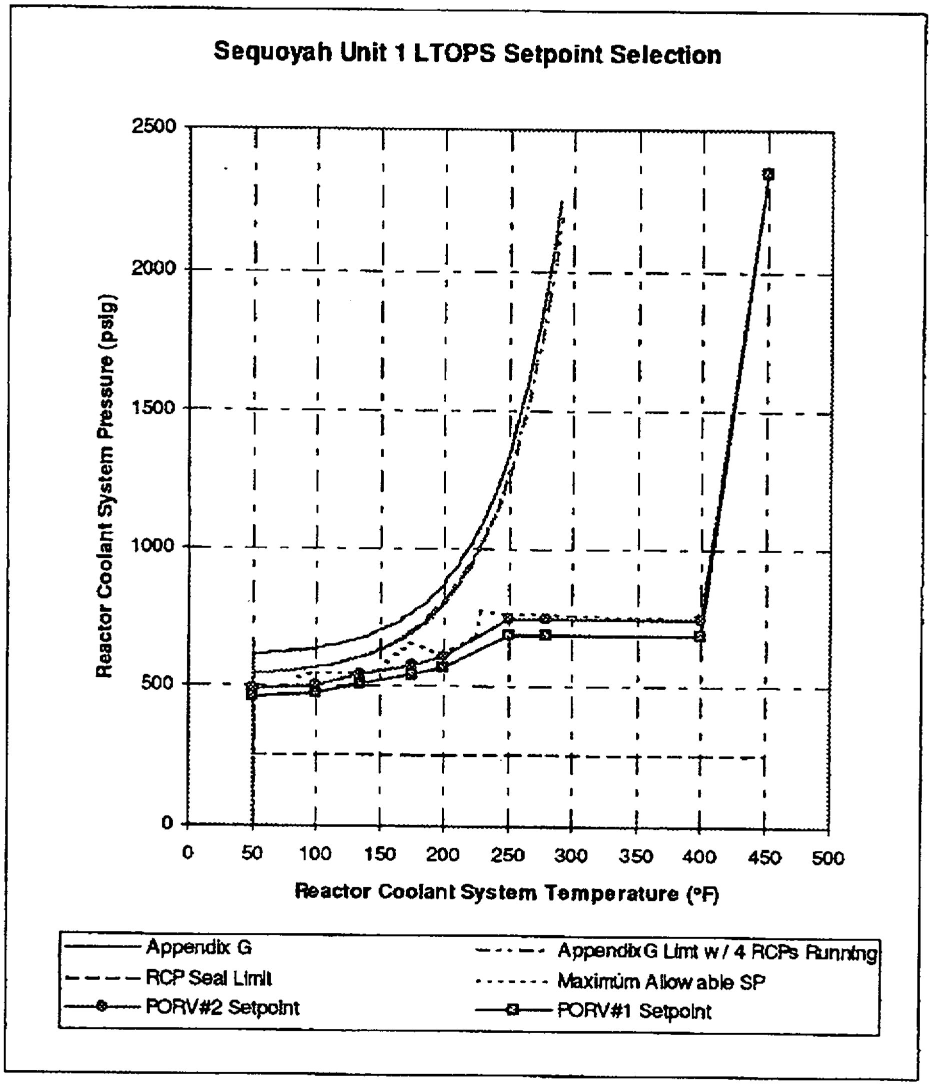


**Figure A-1: Sequoyah Unit 1 LTOPS Setpoint Selection**

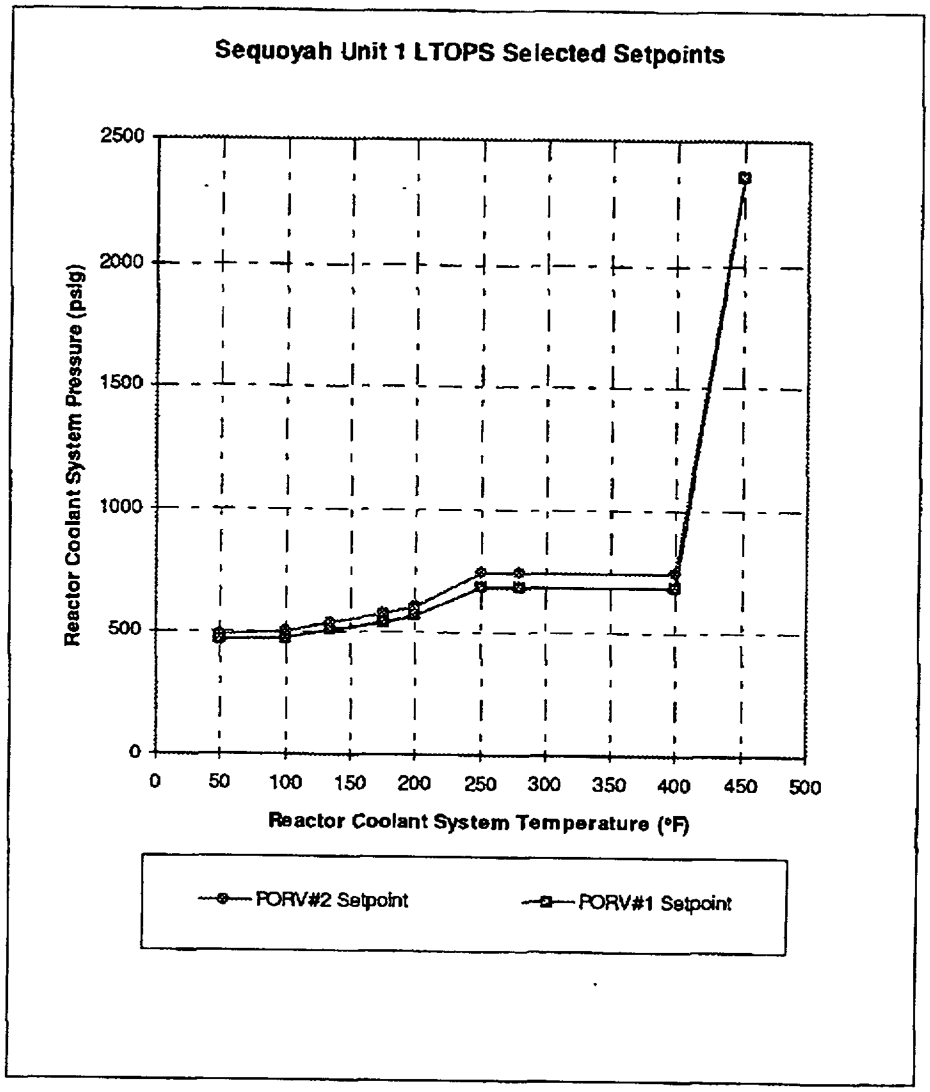


Figure A-2 - Sequoyah Unit 1 LTOPS Selected Setpoints

# APPENDIX B

# PRESSURIZED THERMAL SHOCK (PTS) RESULTS

**PTS Calculations:**

The PTS Rule requires that for each pressurized water nuclear power reactor for which an operating license has been issued, the licensee shall have projected values of $RT_{PTS}$, accepted by the NRC, for each reactor vessel beltline material for the EOL fluence of the material. This assessment must specify the basis for the projected value of $RT_{PTS}$ for each vessel beltline material, including the assumptions regarding core loading patterns, and must specify the copper and nickel contents and the fluence value used in the calculation. This assessment must be updated whenever there is a significant change in projected values of $RT_{PTS}$, or upon request for a change in the expiration date for operation of the facility. (Changes to $RT_{PTS}$ values are considered significant if either the previous value or the current value, or both values, exceed the screening criterion prior to the expiration of the operating license, including any renewed term, if applicable, for the plant.

To verify that $RT_{NDT}$, for each vessel beltline material is a bounding value for the specific reactor vessel, licensees shall consider plant-specific information that could affect the level of embrittlement. This information includes but is not limited to the reactor vessel operating temperature and any related surveillance program results. (Surveillance program results mean any data that demonstrates the embrittlement trends for the limiting beltline material, including but not limited to data from test reactors or from surveillance programs at other plants with or without surveillance program integrated per 10 CFR Part 50, Appendix H.)

**Calculations:**

Tables B-1 and B-2 contain the results of the calculations for each of the beltline region materials in the Sequoyah Unit 1 Reactor Vessel. Per TVA, the EOL is 32 EFPY and the Life Extension EOL is 48 EFPY.

TABLE B-1
$RT_{PTS}$ Calculations for Sequoyah Unit 1 Beltline Region Materials at 32 EFPY

| Material | Fluence ($n/cm^2$, E>1.0 MeV) | FF | CF (°F) | $\Delta RT_{PTS}$(c) (°F) | Margin (°F) | $RT_{NDT(U)}$(a) (°F) | $RT_{PTS}$(b) (°F) |
|---|---|---|---|---|---|---|---|
| Intermediate Shell Forging 05 | 1.84 | 1.167 | 115.6 | 134.9 | 34 | 40 | 209 |
| Lower Shell Forging 04 | 1.84 | 1.167 | 95.0 | 110.9 | 34 | 73 | 218 |
| Lower Shell Forging 04 (Using S/C Data) | 1.84 | 1.167 | 105.9 | 123.6 | 34 | 73 | 231 |
| Circumferential Weld Metal | 1.84 | 1.167 | 161.3 | 188.2 | 56 | -40 | 204 |
| Circumferential Weld Metal (Using S/C Data) | 1.84 | 1.167 | 135.0 | 157.5 | 56 | -40 | 174 |

Notes:
(a) Initial $RT_{NDT}$ values are measured values
(b) $RT_{PTS} = RT_{NDT(U)} + \Delta RT_{PTS} + \text{Margin}$ (°F)
(c) $\Delta RT_{PTS} = CF * FF$

TABLE B-2
$RT_{PTS}$ Calculations for Sequoyah Unit 1 Beltline Region Materials at 48 EFPY

| Material | Fluence ($n/cm^2$, E>1.0 MeV) | FF | CF (°F) | $\Delta RT_{PTS}$(c) (°F) | Margin (°F) | $RT_{NDT(U)}$(a) (°F) | $RT_{PTS}$(b) (°F) |
|---|---|---|---|---|---|---|---|
| Intermediate Shell Forging 05 | 2.72 | 1.267 | 115.6 | 146.5 | 34 | 40 | 221 |
| Lower Shell Forging 04 | 2.72 | 1.267 | 95.0 | 120.4 | 34 | 73 | 227 |
| Lower Shell Forging 04 (Using S/C Data) | 2.72 | 1.267 | 105.9 | 134.2 | 34 | 73 | 241 |
| Circumferential Weld Metal | 2.72 | 1.267 | 161.3 | 204.4 | 56 | -40 | 220 |
| Circumferential Weld Metal (Using S/C Data) | 2.72 | 1.267 | 135.0 | 171.0 | 56 | -40 | 187 |

Notes:
(a) Initial $RT_{NDT}$ values are measured values
(b) $RT_{PTS} = RT_{NDT(U)} + \Delta RT_{PTS} + \text{Margin}$ (°F)
(c) $\Delta RT_{PTS} = CF * FF$

All of the beltline materials in the Sequoyah Unit 1 reactor vessel are below the screening criteria values of 270°F and 300°F at 32 and 48 EFPY.

# APPENDIX C

# CALCULATED FLUENCE DATA

The best estimate exposure of the Sequoyah Unit 1 reactor vessel presented in WCAP-15224[7] was developed using a combination of absolute plant specific transport calculations and all available plant specific measurement data. The evaluation is consistent with the methodology accepted by the NRC and documented in WCAP-14040-NP-A[2].

Combining this measurement data base with the plant-specific calculations, the best estimate vessel exposure is obtained from the following relationship:

$$\Phi_{Best\ Est.} = K\ \Phi_{Calc.}$$

where:

$\Phi_{Best\ Est.}$ = The best estimate fast neutron exposure at the location of interest.

K = The plant specific best estimate/calculation (BE/C) bias factor derived from the surveillance capsule dosimetry data.

$\Phi_{Calc.}$ = The absolute calculated fast neutron exposure at the location of interest.

For Sequoyah Unit 1, the derived plant specific bias factors were 1.14, 1.14, 1.14 for $\Phi(E > 1.0\ MeV)$, $\Phi(E > 0.1\ MeV)$, and dpa, respectively. Bias factors of this magnitude developed with BUGLE-96 are within expected tolerances for fluence calculated using the ENDF/B-VI based cross-section library.

Table C-1 presents the reactor vessel fast neutron (E > 1.0 MeV) exposure projections using the absolute plant specific calculations. Table C-2 presents the calculated and measured fluences at the capsules.

Table C-1
Azimuthal Variations Of The Neutron Exposure Projections
On The Reactor Vessel Clad/Base Metal Interface At Core Midplane

Calculated

| | 0° | 15° | 30° | 45°[a] |
|---|---|---|---|---|
| | | 10.03 EFPY | | |
| E>1.0 MeV | 2.05E+18 | 3.21E+18 | 4.09E+18 | 6.37E+18 |
| E>0.1 MeV | 4.07E+18 | 6.41E+18 | 8.41E+18 | 1.34E+19 |
| dpa | 2.60E-03 | 4.06E-03 | 5.22E-03 | 8.08E-03 |
| | | 20 EFPY | | |
| E>1.0 MeV | 3.87E+18 | 5.96E+18 | 7.61E+18 | 1.18E+19 |
| E>0.1 MeV | 8.68E+18 | 1.34E+19 | 1.76E+19 | 2.80E+19 |
| dpa | 5.55E-03 | 8.48E-03 | 1.09E-02 | 1.69E-02 |
| | | 32 EFPY | | |
| E>1.0 MeV | 6.05E+18 | 9.28E+18 | 1.19E+19 | 1.84E+19 |
| E>0.1 MeV | 1.42E+19 | 2.18E+19 | 2.86E+19 | 4.56E+19 |
| dpa | 9.09E-03 | 1.38E-02 | 1.78E-02 | 2.76E-02 |
| | | 48 EFPY | | |
| E>1.0 MeV | 8.96E+18 | 1.37E+19 | 1.75E+19 | 2.72E+19 |
| E>0.1 MeV | 2.16E+19 | 3.30E+19 | 4.33E+19 | 6.91E+19 |
| dpa | 1.38E-02 | 2.09E-02 | 2.69E-02 | 4.18E-02 |

Note:
a) Maximum neutron exposure projection

Table C-2
Comparison Of Calculated And Best Estimate Integrated Neutron Exposure Of Sequoyah Unit 1 Surveillance Capsules T, U, X, and Y

CAPSULE T

| | Calculated | Best Estimate | BE/C |
|---|---|---|---|
| $\Phi(E > 1.0\ MeV)$ [n/cm$^2$] | 2.61E+18 | 2.89E+18 | 1.10 |
| $\Phi(E > 0.1\ MeV)$ [n/cm$^2$] | 8.74E+18 | 9.62E+18 | 1.10 |
| dpa | 4.34E-03 | 4.80E-03 | 1.11 |

CAPSULE U

| | Calculated | Best Estimate | BE/C |
|---|---|---|---|
| $\Phi(E > 1.0\ MeV)$ [n/cm$^2$] | 7.96E+18 | 9.69E+18 | 1.22 |
| $\Phi(E > 0.1\ MeV)$ [n/cm$^2$] | 2.66E+19 | 3.16E+19 | 1.19 |
| dpa | 1.32E-02 | 1.59E-02 | 1.21 |

CAPSULE X

| | Calculated | Best Estimate | BE/C |
|---|---|---|---|
| $\Phi(E > 1.0\ MeV)$ [n/cm$^2$] | 1.32E+19 | 1.50E+19 | 1.14 |
| $\Phi(E > 0.1\ MeV)$ [n/cm$^2$] | 4.42E+19 | 5.09E+19 | 1.15 |
| dpa | 2.20E-02 | 2.51E-02 | 1.15 |

CAPSULE Y

| | Calculated | Best Estimate | BE/C |
|---|---|---|---|
| $\Phi(E > 1.0\ MeV)$ [n/cm$^2$] | 2.19E+19 | 2.43E+19 | 1.11 |
| $\Phi(E > 0.1\ MeV)$ [n/cm$^2$] | 7.31E+19 | 8.15E+19 | 1.12 |
| dpa | 3.63E-02 | 4.05E-02 | 1.12 |

AVERAGE BE/C RATIOS

| | BE/C |
|---|---|
| $\Phi(E > 1.0\ MeV)$ [n/cm$^2$] | 1.14 |
| $\Phi(E > 0.1\ MeV)$ [n/cm$^2$] | 1.14 |
| dpa | 1.14 |

# APPENDIX D

# UPDATED SURVEILLANCE MATERIAL 30 FT-LB TRANSITION TEMPERATURE SHIFTS AND UPPER SHELF ENERGY DECREASES

## TABLE D-1
Measured 30 ft-lb Transition Temperature Shifts of all Available Surveillance Data

| Material | Capsule | Fluence (x $10^{19}$ n/cm$^2$) | 30 ft-lb Transition Temperature Shift | | Upper Shelf Energy Decrease | |
|---|---|---|---|---|---|---|
| | | | Predicted (°F) [a] | Measured (°F) [b] | Predicted (%) [a] | Measured (%)[c] |
| Lower Shell Forging 04 (Tangential) | T | 0.261 | 59.85 | 67.52 | 16 | 16 |
| | U | 0.796 | 89.3 | 109.7 | 20.5 | 21 |
| | X | 1.32 | 102.6 | 145.12 | 23 | 8 |
| | Y | 2.19 | 114.95 | 129.87 | 26.5 | 23 |
| Lower Shell Forging 04 (Axial) | T | 0.261 | 59.85 | 50.59 | 16 | 0 |
| | U | 0.796 | 89.3 | 67.59 | 20.5 | 19 |
| | X | 1.32 | 102.6 | 103.34 | 23 | 22 |
| | Y | 2.19 | 114.95 | 133.35 | 26.5 | 19 |
| Weld Metal | T | 0.261 | 111.13 | 127.79 | 35 | 30 |
| | U | 0.796 | 165.82 | 144.92 | 42 | 26 |
| | X | 1.32 | 190.51 | 159.02 | 45 | 21 |
| | Y | 2.19 | 213.44 | 163.8 | 48 | 28 |
| HAZ Metal | T | 0.261 | -- | 45.48 | -- | 20 |
| | U | 0.796 | -- | 78.94 | -- | 26 |
| | X | 1.32 | -- | 95.89 | -- | 3 |
| | Y | 2.19 | -- | 73.3 | -- | 10 |

Notes:
(a) Based on Regulatory Guide 1.99, Revision 2, methodology using the mean weight percent values of copper and nickel of the surveillance material.
(b) Calculated using measured Charpy data plotted using CVGRAPH, Version 4.1 (Reference 9)
(c) Values are based on the definition of upper shelf energy given in ASTM E185-82.

# APPENDIX E

## REACTOR VESSEL BELTLINE MATERIAL PROJECTED END OF LICENSE UPPER SHELF ENERGY VALUES

TABLE E-1
Predicted End-of-License (32 EFPY) USE Calculations for all the Beltline Region Materials

| Material | Weight % of Cu | 1/4T EOL Fluence ($10^{19}$ n/cm$^2$) | Unirradiated USE[a] (ft-lb) | Projected USE Decrease (%) | Projected EOL USE (ft-lb) |
|---|---|---|---|---|---|
| Intermediate Shell Forging 05 | 0.15 | 1.11 | 79 | 24 | 60 |
| Lower Shell Forging 04<br>Using S/C Data | 0.13 | 1.11 | 72 | 22 | 56 |
| Intermediate to Lower Shell Circumferential Weld Seam W05<br>Using S/C Data | 0.35 | 1.11 | 113 | 42 | 66 |

Notes:
(a) These values were obtained from Reference 12.

# APPENDIX F

# UPDATED SURVEILLANCE CAPSULE REMOVAL SCHEDULE

The following surveillance capsule removal schedule meets the requirements of ASTM E185-82 and is recommended for future capsules to be removed from the Sequoyah Unit 1 reactor vessel. This recommended removal schedule is applicable to 32 EFPY of operation.

**Table 7-1 Sequoyah Unit 1 Reactor Vessel Surveillance Capsule Withdrawal Schedule**

| Capsule | Location | Lead Factor(a) | Removal Time (EFPY)(b) | Fluence ($n/cm^2$, E>1.0 MeV)(a) |
|---|---|---|---|---|
| T | 40° | 3.39 | 1.03 | $2.61 \times 10^{18}$ (c) |
| U | 140° | 3.47 | 3.00 | $7.96 \times 10^{18}$ (c) |
| X | 220° | 3.47 | 5.27 | $1.32 \times 10^{19}$ (c) |
| Y | 320° | 3.43 | 10.03 | $2.19 \times 10^{19}$ (c,d) |
| S | 4° | 1.08 | Standby | (d,e) |
| V | 176° | 1.08 | Standby | (d,e) |
| W | 184° | 1.08 | Standby | (d,e) |
| Z | 356° | 1.08 | Standby | (d,e) |

Notes:

(a) Updated in Capsule Y dosimetry analysis (Reference 7).

(b) Effective Full Power Years (EFPY) from plant startup.

(c) Plant specific evaluation.

(d) This fluence is not less than once or greater than twice the peak end of license (32 EFPY) fluence

(e) Capsules S, V, W and Z will reach a fluence of $2.74 \times 10^{19}$ (E > 1.0 MeV), the 48 EFPY peak vessel fluence at approximately 44 EFPY, respectively.

# APPENDIX G

# ENABLE TEMPERATURE CALCULATIONS AND RESULTS

**Enable Temperature Calculation:**

ASME Code Case N-514 requires the low temperature overpressure (LTOP or COMS) system to be in operation at coolant temperatures less than 200°F or at coolant temperatures less than a temperature corresponding to a reactor vessel metal temperature less than $RT_{NDT}$ + 50°F, whichever is greater. $RT_{NDT}$ is the highest adjusted reference temperature (ART) for the limiting beltline material at a distance one fourth of the vessel section thickness from the vessel inside surface (ie. clad/base metal interface), as determined by Regulatory Guide 1.99, Revision 2.

<u>32 EFPY</u>

The highest calculated 1/4T ART for the Sequoyah Unit 1 reactor vessel beltline region at 32 EFPY is 216°F.

From the OPERLIM computer code output for the Sequoyah Unit 1 32 EFPY P-T limit curves without margins (Configuration # 1389796830, *operlim.film* File) the maximum $\Delta T_{metal}$ is:

Cooldown Rate (Steady-State Cooldown):
max ($\Delta T_{metal}$) at 1/4T = 0°F

Heatup Rate of 100°F/Hr:
max ($\Delta T_{metal}$) at 1/4T = 28.924°F

**Enable Temperature (ENBT) = $RT_{NDT}$ + 50 + max ($\Delta T_{metal}$), °F**
**= (216 + 50 + 28.924) °F**
**= 294.924°F**

The minimum required enable temperature for the Sequoyah Unit 1 Reactor Vessel is 295°F at 32 EFPY of operation.

05173760



WESTINGHOUSE NON-PROPRIETARY CLASS 3

**WCAP-15321, Revision 1**

# Sequoyah Unit 2 Heatup and Cooldown Limit Curves for Normal Operation and PTLR Support Documentation

**J. H. Ledger**

**April 2001**

Prepared by the Westinghouse Electric Company LLC
for the Tennessee Valley Authority

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Westinghouse Non-Proprietary Class 3

WCAP-15321
Revision 1

♦ ♦ ♦ ♦ ♦ ♦ ♦ ♦

# Sequoyah Unit 2 Heatup and Cooldown Limit Curves for Normal Operation and PTLR Support Documentation

Westinghouse Electric Company LLC

## PREFACE

This report has been technically reviewed and verified by:

T. J. Laubham

**Revision 1:**

An error was detected in the "OPERLIM" Computer Program that Westinghouse uses to generate pressure-temperature (PT) limit curves. This error potentially effects the heatup curves when the 1996 Appendix G Methodology is used in generating the PT curves. It has been determined that WCAP-15321 Rev. 0 was impacted by this error. Thus, this revision provides corrected curves from WCAP-15321 Rev. 0.

Note that only the 60°F/hr heatup curves were affected by this error. The 100°F/hr heatup and all cooldown curves were not affected by the computer error and thus remain valid.

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## EXECUTIVE SUMMARY

This report provides the methodology and results of the generation of heatup and cooldown pressure temperature limit curves for normal operation of the Sequoyah Unit 2 reactor vessel. In addition, Pressure Temperature Limits Report (PTLR) support information, such as LTOPS Setpoint, PTS , EOL USE and Withdrawal Schedule, is documented herein under the Appendices. The PT curves were generated based on the latest available reactor vessel information (Capsule Y analysis, WCAP-15321[7] and the latest Pressure-Temperature (P-T) Limit Curves from WCAP-12971[13]). The Sequoyah Unit 2 heatup and cooldown pressure-temperature limit curves have been updated based on the use of the ASME Code Case N-640[3], which allows the use of the $K_{Ic}$ methodology, and the elimination of the reactor vessel flange temperature requirement (Ref, WCAP-15315[6]).

# 1 INTRODUCTION

Heatup and cooldown limit curves are calculated using the adjusted $RT_{NDT}$ (reference nil-ductility temperature) corresponding to the limiting beltline region material of the reactor vessel. The adjusted $RT_{NDT}$ of the limiting material in the core region of the reactor vessel is determined by using the unirradiated reactor vessel material fracture toughness properties, estimating the radiation-induced $\Delta RT_{NDT}$, and adding a margin. 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 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, $\Delta RT_{NDT}$ due to the radiation exposure associated with that time period must be added to the unirradiated $RT_{NDT}$ ($IRT_{NDT}$). 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, Revision 2, "Radiation Embrittlement of Reactor Vessel Materials."[1] Regulatory Guide 1.99, Revision 2, is used for the calculation of Adjusted Reference Temperature (ART) values ($IRT_{NDT}$ + $\Delta RT_{NDT}$ + margins for uncertainties) at the 1/4T and 3/4T locations, where T is the thickness of the vessel at the beltline region measured from the clad/base metal interface.

The heatup and cooldown curves documented in this report were generated using the most limiting ART values and the NRC approved methodology documented in WCAP-14040-NP-A, Revision 2[2], "Methodology Used to Develop Cold Overpressure Mitigating System Setpoints and RCS Heatup and Cooldown Limit Curves" with exception of the following: 1) The fluence values used in this report are calculated fluence values, not the best estimate fluence values (See Appendix B). 2) The $K_{Ic}$ critical stress intensities are used in place of the $K_{Ia}$ critical stress intensities. This methodology is taken from approved ASME Code Case N-640[3]. 3) The reactor vessel flange temperature requirement has been eliminated. Justification has been provided in WCAP-15315[6]. 4) The 1996 Version of Appendix G to Section XI[4] will be used rather than the 1989 version.

# 2 FRACTURE TOUGHNESS PROPERTIES

The fracture-toughness properties of the ferritic materials in the reactor coolant pressure boundary are determined in accordance with the NRC Standard Review Plan[5]. The beltline material properties of the Sequoyah Unit 2 reactor vessel is presented in Table 1.

Best estimate copper (Cu) and nickel (Ni) weight percent values used to calculate chemistry factors (CF) in accordance with Regulatory Guide 1.99, Revision 2, are provided in Table 1. Additionally, surveillance capsule data is available for four capsules (Capsules T, U, X and Y) already removed from the Sequoyah Unit 2 reactor vessel. This surveillance capsule data was also used to calculate CF values per Position 2.1 of Regulatory Guide 1.99, Revision 2 in Table 2. These CF values are summarized in Table 3.

The Regulatory Guide 1.99, Revision 2 methodology used to develop the heatup and cooldown curves documented in this report is the same as that documented in WCAP-14040, Revision 2.

TABLE 1

Reactor Vessel Beltline Material Unirradiated Toughness Properties

| Material Description | Cu (%) | Ni(%) | Initial $RT_{NDT}$[a] |
|---|---|---|---|
| Intermediate Shell Forging 05 (Heat #288757 / 981057) | 0.13 | 0.76 | 10°F |
| Lower Shell Forging 04 (Heat # 990469 / 293323) | 0.14 | 0.76 | -22°F |
| Intermediate to Lower Shell Forging Circumferential Weld Seam[b] | 0.12 | 0.11 | -4°F |
| Surveillance Weld[b] | 0.13 | 0.11 | --- |

Notes:

(a) The Initial $RT_{NDT}$ values are measured values

(b) Circumferential Weld Seam was fabricated with weld wire type SMIT 89, Heat # 4278, Flux type SMIT 89, lot # 1211 and is representative of the intermediate to lower shell circumferential weld.

The chemistry factors were calculated using Regulatory Guide 1.99 Revision 2, Positions 1.1 and 2.1. Position 1.1 uses the Tables from the Reg. Guide along with the best estimate copper and nickel weight percents. Position 2.1 uses the surveillance capsule data from all capsules withdrawn to date. The fluence values used to determine the CFs in Table 2 are the calculated fluence values at the surveillance capsule locations. Hence, the calculated fluence values were used for all cases.

The measured $\Delta RT_{NDT}$ values for the weld data were adjusted using the ratio procedure given in Position 2.1 of Regulatory Guide 1.99, Revision 2. All fluence values were obtained from the recent Sequoyah Unit 2 capsule analysis[7] which calculated the fluences using the ENDF/B-VI scattering cross-section data set. The fluence values used are also documented in Appendix C of this report.

TABLE 2

Calculation of Chemistry Factors using Sequoyah Unit 2 Surveillance Capsule Data

| Material | Capsule | Capsule f(a) | FF(b) | $\Delta RT_{NDT}$(c) | $FF*\Delta RT_{NDT}$ | $FF^2$ |
|---|---|---|---|---|---|---|
| Intermediate Shell Forging 05 (Tangential) | T | 2.61E+18 | 0.635 | 63.7 | 40.45 | 0.403 |
| | U | 6.92E+18 | 0.897 | 79.3 | 71.13 | 0.805 |
| | X | 1.22E+19 | 1.055 | 85.7 | 90.41 | 1.113 |
| | Y | 2.14E+19 | 1.207 | 134.1 | 161.86 | 1.457 |
| Intermediate Shell Forging 05 (Axial) | T | 2.61E+18 | 0.635 | 48.7 | 30.92 | 0.403 |
| | U | 6.92E+18 | 0.897 | 66.1 | 59.29 | 0.805 |
| | X | 1.22E+19 | 1.055 | 110.0 | 116.05 | 1.113 |
| | Y | 2.14E+19 | 1.207 | 89.2 | 107.66 | 1.457 |
| | | | | SUM: | 677.77°F | 7.556 |
| | $CF_{05} = \Sigma(FF * RT_{NDT}) \div \Sigma(FF^2) = (677.77) \div (7.556) = 89.7°F$ | | | | | |
| Surveillance Weld Material(d) | T | 2.61E+18 | 0.635 | 69.4 (74.6) | 44.07 | 0.403 |
| | U | 6.92E+18 | 0.897 | 121.3 (130.4) | 108.81 | 0.805 |
| | X | 1.22E+19 | 1.055 | 41.1 (44.2) | 43.36 | 1.113 |
| | Y | 2.14E+19 | 1.207 | 80.8 (86.9) | 97.53 | 1.457 |
| | | | | SUM: | 293.77°F | 3.778 |
| | $CF_{Surv.\ Weld} = \Sigma(FF * RT_{NDT}) \div \Sigma(FF^2) = (293.77°F) \div (3.778) = 77.8°F$ | | | | | |

Notes:

(a) f = Calculated fluence from capsule Y dosimetry analysis results [7], ($x\ 10^{19}\ n/cm^2$, E > 1.0 MeV).

(b) FF = fluence factor = $f^{(0.28 - 0.1*\log f)}$.

(c) $\Delta RT_{NDT}$ values are the measured 30 ft-lb shift values taken from App. B of Ref. 7, rounded to one decimal point.

(d) The surveillance weld metal $\Delta RT_{NDT}$ values have been adjusted by a ratio factor of 0.93.

TABLE 3

Summary of the Sequoyah Unit 2 Reactor Vessel Beltline Material Chemistry Factors

| Material | Reg. Guide 1.99, Rev. 2 Position 1.1 CF's | Reg. Guide 1.99, Rev. 2 Position 2.1 CF's |
|---|---|---|
| Intermediate Shell Forging 05 | 95°F | 89.7°F |
| Lower Shell Forging 04 | 104°F | --- |
| Circumferential Weld W05 (Heat # 4278) | 63°F | 77.8°F |
| Surveillance Weld Metal (Heat # 4278) | 67.9°F | --- |

# 3 CRITERIA FOR ALLOWABLE PRESSURE-TEMPERATURE RELATIONSHIPS

## 3.1 Overall Approach

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, $K_{Ic}$, for the metal temperature at that time. $K_{Ic}$ is obtained from the reference fracture toughness curve, defined in Code Case N-640, "Alternative Reference Fracture Toughness for Development of PT Limit Curves for Section XI"[3 & 4] of the ASME Appendix G to Section XI. The $K_{Ic}$ curve is given by the following equation:

$$K_{Ic} = 33.2 + 20.734 * e^{[0.02(T - RT_{NDT})]} \quad (1)$$

where,

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

This $K_{Ic}$ curve is based on the lower bound of static critical $K_I$ values measured as a function of temperature on specimens of SA-533 Grade B Class1, SA-508-1, SA-508-2, SA-508-3 steel.

## 3.2 Methodology for Pressure-Temperature Limit Curve Development

The governing equation for the heatup-cooldown analysis is defined in Appendix G of the ASME Code as follows:

$$C^* K_{Im} + K_{It} < K_{Ic} \quad (2)$$

where,

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

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

$K_{Ic}$ = 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

For membrane tension, the corresponding $K_I$ for the postulated defect is:

$$K_{Im} = M_m \times (pR_i / t) \qquad (3)$$

where, $M_m$ for an inside surface flaw is given by:

$M_m$ = 1.85 for $\sqrt{t} < 2$,
$M_m$ = $0.926\sqrt{t}$ for $2 \leq \sqrt{t} \leq 3.464$,
$M_m$ = 3.21 for $\sqrt{t} > 3.464$

Similarly, $M_m$ for an outside surface flaw is given by:

$M_m$ = 1.77 for $\sqrt{t} < 2$,
$M_m$ = $0.893\sqrt{t}$ for $2 \leq \sqrt{t} \leq 3.464$,
$M_m$ = 3.09 for $\sqrt{t} > 3.464$

and p = internal pressure, Ri = vessel inner radius, and t = vessel wall thickness.

For bending stress, the corresponding $K_I$ for the postulated defect is:

$K_{Ib}$ = $M_b$ * Maximum Stress, where $M_b$ is two-thirds of $M_m$

The maximum $K_I$ produced by radial thermal gradient for the postulated inside surface defect of G-2120 is $K_{It} = 0.953 \times 10^{-3} \times CR \times t^{2.5}$, where CR is the cooldown rate in °F/hr., or for a postulated outside surface defect, $K_{It} = 0.753 \times 10^{-3} \times HU \times t^{2.5}$, where HU is the heatup rate in °F/hr.

The through-wall temperature difference associated with the maximum thermal $K_I$ can be determined from Fig. G-2214-1. The temperature at any radial distance from the vessel surface can be determined from Fig. G-2214-2 for the maximum thermal $K_I$.

(a) The maximum thermal $K_I$ relationship and the temperature relationship in Fig. G-2214-1 are applicable only for the conditions given in G-2214.3(a)(1) and (2).

(b) Alternatively, the $K_I$ for radial thermal gradient can be calculated for any thermal stress distribution and at any specified time during cooldown for a ¼-thickness inside surface defect using the relationship:

$$K_{It} = (1.0359C_0 + 0.6322C_1 + 0.4753C_2 + 0.3855C_3) * \sqrt{\pi a} \qquad (4)$$

or similarly, $K_{IT}$ during heatup for a ¼-thickness outside surface defect using the relationship:

$$K_{It} = (1.043C_0 + 0.630C_1 + 0.481C_2 + 0.401C_3) * \sqrt{\pi a} \quad (5)$$

where the coefficients $C_0$, $C_1$, $C_2$ and $C_3$ are determined from the thermal stress distribution at any specified time during the heatup or cooldown using the form:

$$\sigma(x) = C_0 + C_1(x/a) + C_2(x/a)^2 + C_3(x/a)^3 \quad (6)$$

and $x$ is a variable that represents the radial distance from the appropriate (i.e., inside or outside) surface to any point on the crack front and $a$ is the maximum crack depth.

Note, that equations 3, 4 and 5 were implemented in the OPERLIM computer code, which is the program used to generate the pressure-temperature (P-T) limit curves. No other changes were made to the OPERLIM computer code with regard to P-T calculation methodology. Therefore, the P-T curve methodology is unchanged from that described in WCAP-14040, "Methodology used to Develop Cold Overpressure Mitigating System Setpoints and RCS Heatup and Cooldwon Limit Curves"[2] Section 2.6 (equations 2.6.2-4 and 2.6.3-1) with the exceptions just described above.

At any time during the heatup or cooldown transient, $K_{Ic}$ is determined by the metal temperature at the tip of a postulated flaw at the 1/4T and 3/4T location, 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 1/4T vessel location is at a higher temperature than the fluid adjacent to the vessel inner diameter. This condition, of course, is not true for the steady-state situation. It follows that, at any given reactor coolant temperature, the $\Delta T$ (temperature) developed during cooldown results in a higher value of $K_{Ic}$ at the 1/4T location for finite cooldown rates than for steady-state operation. Furthermore, if conditions exist so that the increase in $K_{Ic}$ 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 1/4T 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 1/4T 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 $K_{Ic}$ for the 1/4T crack during heatup is lower than the $K_{Ic}$ for the 1/4T 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 $K_{Ic}$ values 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 1/4T 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 1/4T flaw located at the 1/4T location from the outside surface 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.

### 3.3 Closure Head/Vessel Flange Requirements

10 CFR Part 50, Appendix G 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 (3107 psi), which is 621 psig for Sequoyah Unit 2. However, per WCAP-15315, "Reactor Vessel Closure Head/Vessel Flange Requirements Evaluation For Operating PWR and BWR Plants"[6], this requirement is no longer necessary when using the methodology of Code Case N-640[8]. Hence, Sequoyah Unit 2 heatup and cooldown limit curves will be generated without flange requirements included.

# 4 CALCULATION OF ADJUSTED REFERENCE TEMPERATURE

From Regulatory Guide 1.99, Revision 2, the adjusted reference temperature (ART) for each material in the beltline region is given by the following expression:

$$ART = \text{Initial } RT_{NDT} + \Delta RT_{NDT} + \text{Margin} \quad (7)$$

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[8]. 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:

$$\Delta RT_{NDT} = CF * f^{(0.28 - 0.10 \log f)} \quad (8)$$

To calculate $\Delta RT_{NDT}$ 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.

$$f_{(depth\ x)} = f_{surface} * e^{(-0.24x)} \quad (9)$$

where x inches (vessel beltline thickness is 8.45 inches) is the depth into the vessel wall measured from the vessel clad/base metal interface. The resultant fluence is then placed in Equation 8 to calculate the $\Delta RT_{NDT}$ at the specific depth.

The Westinghouse Radiation Engineering and Analysis Group evaluated the vessel fluence projections as a part of WCAP-15320[7] and are also presented in a condensed version in Table 4 of this report. The evaluation used the ENDF/B-VI scattering cross-section data set. This is consistent with methods presented in WCAP-14040-NP-A, "Methodology Used to Develop Cold Overpressure Mitigating System Setpoints and RCS Heatup and Cooldown Limit Curves"[2]. Table 4 contains the calculated vessel surface fluences values at various azimuthal locations and Tables 5 and 6 contains the 1/4T and 3/4T calculated fluences and fluence factors, per the Regulatory Guide 1.99, Revision 2, used to calculate the ART values for all beltline materials in the Sequoyah Unit 2 reactor vessel. Additionally, the surveillance capsule fluence values are presented in Table 7.

TABLE 4

Neutron Fluence Projections at Key Locations on the Reactor Vessel Clad/Base Metal Interface

($10^{19}$ n/cm$^2$, E > 1.0 MeV)

| EFPY | Azimuthal Location | | | |
|---|---|---|---|---|
| | 0° | 15° | 30° | 45° |
| 10.54 | 0.211 | 0.336 | 0.426 | 0.637 |
| 20 | 0.38 | 0.60 | 0.773 | 1.16 |
| 32 | 0.593 | 0.934 | 1.21 | 1.82 |
| 48 | 0.878 | 1.38 | 1.80 | 2.71 |

TABLE 5

Summary of the Vessel Surface, 1/4T and 3/4T Fluence Values used for the Generation of the 32 EFPY Heatup/Cooldown Curves

| Material | Surface | ¼ T[a] | ¾ T[a] |
|---|---|---|---|
| Intermediate Shell Forging 05 | $1.82 \times 10^{19}$ | $1.10 \times 10^{19}$ | $3.98 \times 10^{18}$ |
| Lower Shell Forging 04 | $1.82 \times 10^{19}$ | $1.10 \times 10^{19}$ | $3.98 \times 10^{18}$ |
| Circumferential Weld Seam (Heat 4278) | $1.82 \times 10^{18}$ | $1.10 \times 10^{19}$ | $3.98 \times 10^{18}$ |

Note:
(a) 1/4T and 3/4T = $F_{(Surface)} * e^{(-0.24*x)}$, where x is the depth into the vessel wall (i.e. 8.45*0.25 or 0.75)

TABLE 6

Summary of the Calculated Fluence Factors used for the Generation of the 32 EFPY Heatup and Cooldown Curves

| EFPY | 1/4T FF | 3/4T FF |
|---|---|---|
| 32 | 1.027 | 0.745 |

TABLE 7
Integrated Neutron Exposure of the Sequoyah Unit 2 Surveillance Capsules Tested To Date

| Capsule | Fluence |
|---|---|
| T | $2.61 \times 10^{18}$ n/cm², (E > 1.0 MeV) |
| U | $6.92 \times 10^{18}$ n/cm², (E > 1.0 MeV) |
| X | $1.22 \times 10^{19}$ n/cm², (E > 1.0 MeV) |
| Y | $2.14 \times 10^{19}$ n/cm², (E > 1.0 MeV) |

Margin is calculated as, $M = 2\sqrt{\sigma_I^2 + \sigma_\Delta^2}$. The standard deviation for the initial $RT_{NDT}$ margin term, is $\sigma_i$ 0°F when the initial $RT_{NDT}$ is a measured value, and 17°F when a generic value is available. The standard deviation for the $\Delta RT_{NDT}$ margin term, $\sigma_\Delta$, is 17°F for plates or forgings, and 8.5°F for plates or forgings when surveillance data is used. For welds, $\sigma_\Delta$ is equal to 28°F when surveillance capsule data is not used, and is 14°F (half the value) when credible surveillance capsule data is used. $\sigma_\Delta$ need not exceed 0.5 times the mean value of $\Delta RT_{NDT}$.

Based on the surveillance program credibility evaluation presented in Appendix D to WCAP-15320, the Sequoyah Unit 2 surveillance program data is non-credible. In addition, following the guidance provided by the NRC in recent industry meeting, Table Chemistry Factor for the intermediate shell forging 05 was determined to be conservative. Hence, the adjusted reference temperature (ART) must be calculated using Position 1.1 along with the full margin term. Both Regulatory Guide 1.99, Revision 2, Position 1.1 and 2.1 have been shown herein for completeness. Contained in Tables 8 and 9 are the calculations of the 32 EFPY ART values used for generation of the heatup and cooldown curves.

TABLE 8
Calculation of the ART Values for the 1/4T Location @ 32 EFPY

| Material | RG 1.99 R2 Method | CF (°F) | FF | $IRT_{NDT}$(1) (°F) | $\Delta RT_{NDT}$(3) (°F) | Margin(4) (°F) | ART(2) (°F) |
|---|---|---|---|---|---|---|---|
| Intermediate Shell Forging 05 | Position 1.1 | 95 | 1.027 | 10 | 97.6 | 34 | 142 |
| | Position 2.1 | 89.7 | 1.027 | 10 | 92.1 | 34 | 136 |
| Lower Shell Forging 04 | Position 1.1 | 104 | 1.027 | -22 | 106.8 | 34 | 119 |
| Intermediate to Lower Shell | Position 1.1 | 63 | 1.027 | -4 | 64.7 | 56 | 117 |
| Circumferential Weld Seam | Position 2.1 | 77.8 | 1.027 | -4 | 79.9 | 56 | 132 |

Notes:
(1) Initial $RT_{NDT}$ values measured values.
(2) ART = Initial $RT_{NDT}$ + $\Delta RT_{NDT}$ + Margin (°F)
(3) $\Delta RT_{NDT} = CF * FF$
(4) $M = 2 *(\sigma_i^2 + \sigma_\Delta^2)^{1/2}$

TABLE 9
Calculation of the ART Values for the 3/4T Location @ 32 EFPY

| Material | RG 1.99 R2 Method | CF (°F) | FF | $IRT_{NDT}$(1) (°F) | $\Delta RT_{NDT}$(3) (°F) | Margin(4) (°F) | ART(2) (°F) |
|---|---|---|---|---|---|---|---|
| Intermediate Shell Forging 05 | Position 1.1 | 95 | 0.745 | 10 | 70.8 | 34 | 115 |
| | Position 2.1 | 89.7 | 0.745 | 10 | 66.8 | 34 | 111 |
| Lower Shell Forging 04 | Position 1.1 | 104 | 0.745 | -22 | 77.5 | 34 | 90 |
| Intermediate to Lower Shell | Position 1.1 | 63 | 0.745 | -4 | 46.9 | 56 | 99 |
| Circumferential Weld Seam | Position 2.1 | 77.8 | 0.745 | -4 | 58.0 | 56 | 110 |

Notes:
(1) Initial $RT_{NDT}$ values measured values.
(2) ART = Initial $RT_{NDT}$ + $\Delta RT_{NDT}$ + Margin (°F)
(3) $\Delta RT_{NDT} = CF * FF$
(4) $M = 2 *(\sigma_i^2 + \sigma_\Delta^2)^{1/2}$

The intermediate shell forging 05 is the limiting beltline material for the 1/4T and 3/4T case (See Tables 8 and 9). Contained in Table 10 is a summary of the limiting ARTs to be used in the generation of the Sequoyah Unit 2 reactor vessel heatup and cooldown curves.

TABLE 10
Summary of the Limiting ART Values Used in the Generation of the Sequoyah Unit 2 Heatup/Cooldown Curves

| EFPY | 1/4T Limiting ART | 3/4T Limiting ART |
|---|---|---|
| 32 | 142°F | 115°F |

# 5 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 Sections 3.0 and 4.0 of this report. This approved methodology is also presented in WCAP-14040-NP-A, Revision 2 with exception to those items discussed in Section 1 of this report.

Figures 1, 2, 4 and 5 present the heatup curves with (10°F and 60 psig) and without margins for possible instrumentation errors using heatup rates of 60 and 100°F/hr applicable for the first 32 EFPY. Figures 3and 6 presents the cooldown curves with (10°F and 60 psig) and without margins for possible instrumentation errors using cooldown rates of 0, 20, 40, 60 and 100°F/hr applicable for 32 EFPY. 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 6. This is in addition to other criteria which must be met before the reactor is made critical, as discussed below in the following paragraphs.

The reactor must not be made critical until pressure-temperature combinations are to the right of the criticality limit line shown in Figures 1,2, 4 and 5. 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 10 CFR Part 50. The governing equation for the hydrostatic test is defined in Code Case N-640[3] (approved in February 1999) as follows:

$$1.5\ K_{Im} < K_{Ic}$$

where,

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

$K_{Ic} = 33.2 + 20.734\ e^{[0.02(T - RT_{NDT})]}$,

T is the minimum permissible metal temperature, and

$RT_{NDT}$ is the metal reference nil-ductility temperature.

The criticality limit curve specifies pressure-temperature limits for core operation to provide additional margin during actual power production as specified in Reference 10. The pressure-temperature limits for core operation (except for low power physics tests) are that the reactor vessel must be at a temperature equal to or higher than the minimum temperature required for the inservice hydrostatic test, and at least 40°F higher than the minimum permissible temperature in the corresponding pressure-temperature curve for heatup and cooldown calculated as described in Section 3.0 of this report. For the heatup and cooldown curves without margins for instrumentation errors, the minimum temperature for the in service hydrostatic leak tests for the Sequoyah Unit 2 reactor vessel at 32 EFPY is 214°F. The vertical line drawn from these points on the pressure-temperature curve, intersecting a curve 40°F higher than the pressure-temperature limit curve, constitutes the limit for core operation for the reactor vessel.

Figures 1 through 6 define all of the above limits for ensuring prevention of nonductile failure for the Sequoyah Unit 2 reactor vessel. The data points used for the heatup and cooldown pressure-temperature limit curves shown in Figures 1 through 6 are presented in Tables 11 through 14.

MATERIAL PROPERTY BASIS

LIMITING MATERIAL: INTERMEDIATE SHELL FORGING 05

LIMITING ART VALUES AT 32 EFPY: ¼T, 142°F

¾T, 115°F

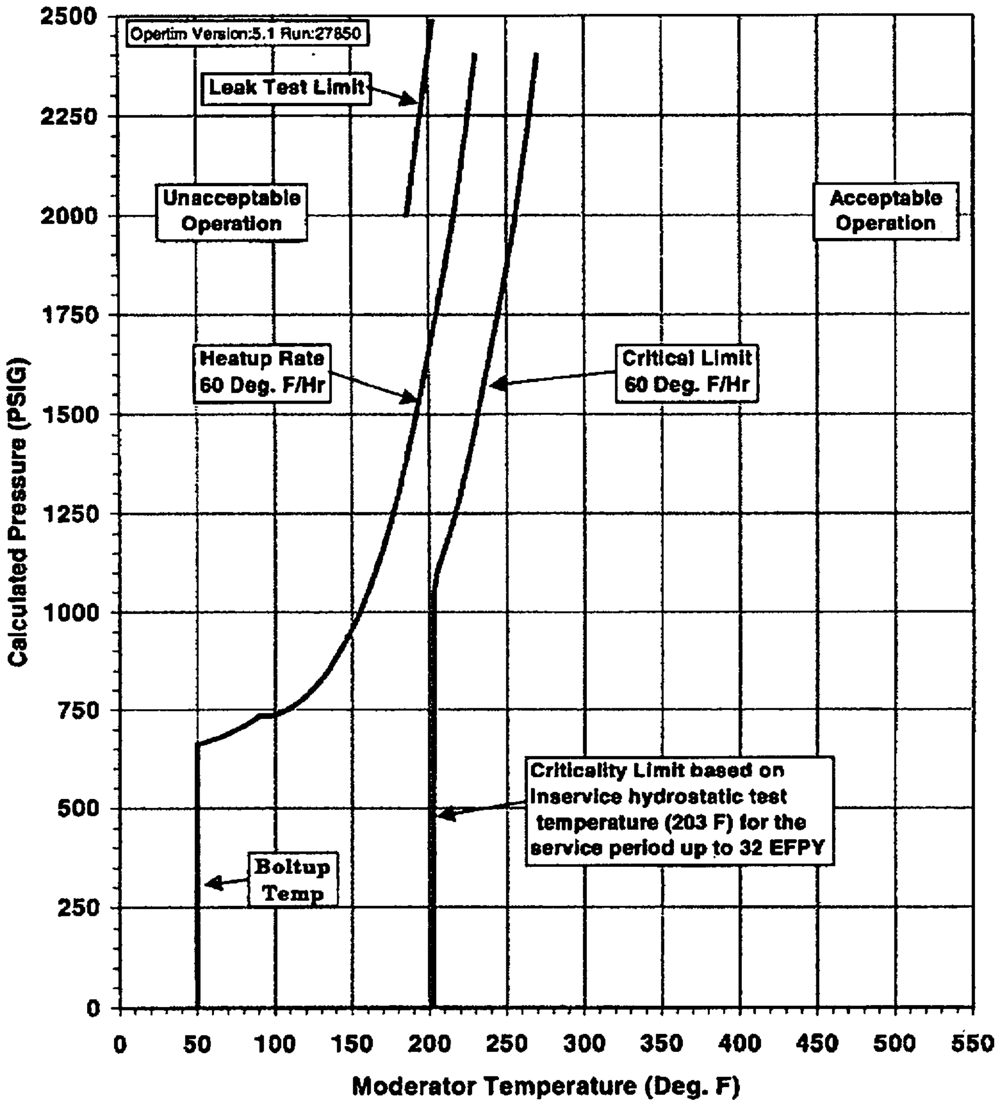


FIGURE 1 Sequoyah Unit 2 Reactor Coolant System Heatup Limitations (Heatup Rate of 60°F/hr) Applicable for the First 32 EFPY (Without Margins for Instrumentation Errors)

MATERIAL PROPERTY BASIS

LIMITING MATERIAL: INTERMEDIATE SHELL FORGING 05
LIMITING ART VALUES AT 32 EFPY: ¼T, 142°F
¾T, 115°F

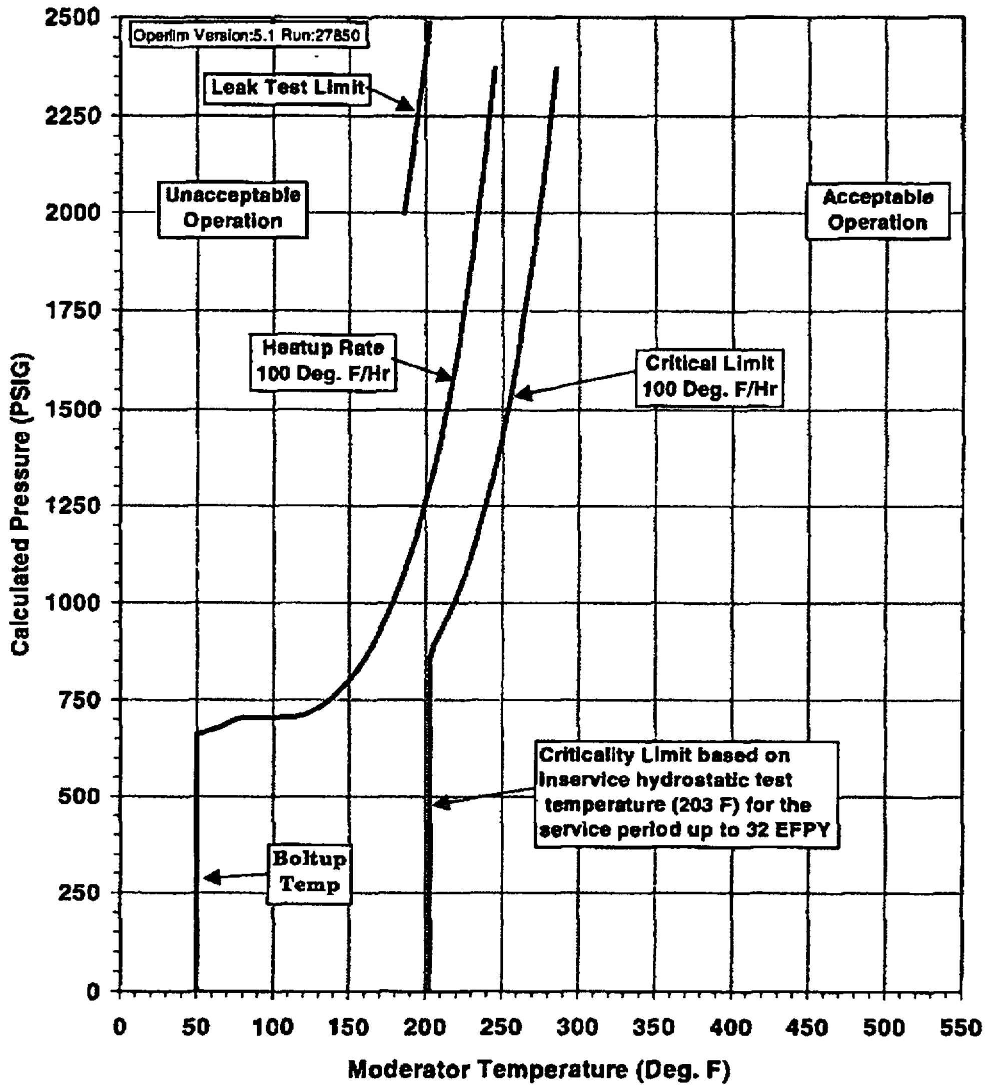


FIGURE 2 Sequoyah Unit 2 Reactor Coolant System Heatup Limitations (Heatup Rate of 100°F/hr) Applicable for the First 32 EFPY (Without Margins of for Instrumentation Errors)

MATERIAL PROPERTY BASIS

LIMITING MATERIAL: INTERMEDIATE SHELL FORGING 05
LIMITING ART VALUES AT 32 EFPY: 1/4T, 142°F
3/4T, 115°F

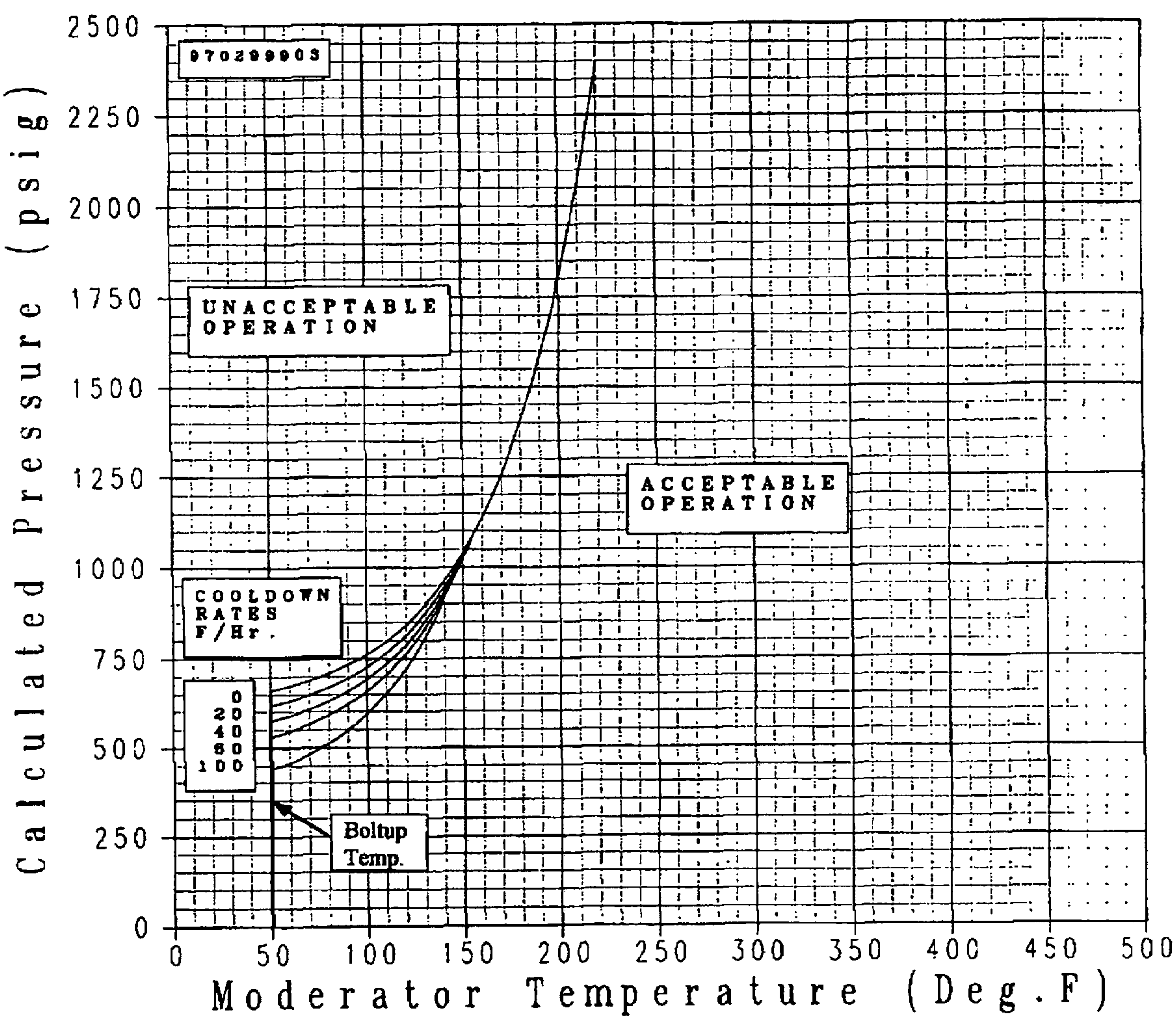


Figure 3 **Sequoyah Unit 2 Reactor Coolant System Cooldown Limitations (Cooldown Rates up to 100°F/hr) Applicable for the First 32 EFPY (Without Margins for Instrumentation Errors)**

MATERIAL PROPERTY BASIS

LIMITING MATERIAL: INTERMEDIATE SHELL FORGING 05
LIMITING ART VALUES AT 32 EFPY: ¼T, 142°F
¾T, 115°F

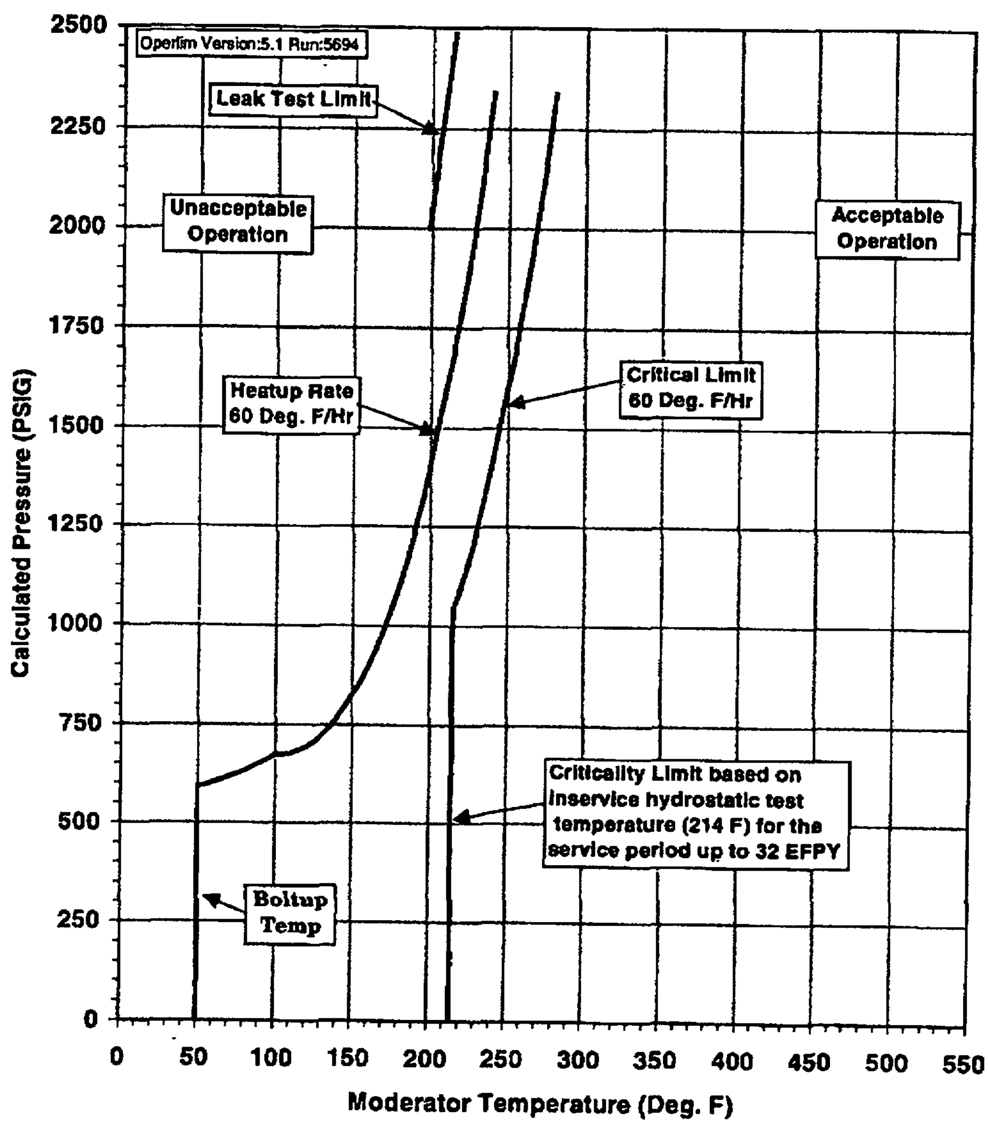


FIGURE 4 Sequoyah Unit 2 Reactor Coolant System Heatup Limitations (Heatup Rate of 60°F/hr) Applicable for the First 32 EFPY (With Margins for Instrumentation Errors of 10°F and 60 psig)

MATERIAL PROPERTY BASIS

LIMITING MATERIAL: INTERMEDIATE SHELL FORGING 05
LIMITING ART VALUES AT 32 EFPY: ¼T, 142°F
¾T, 115°F

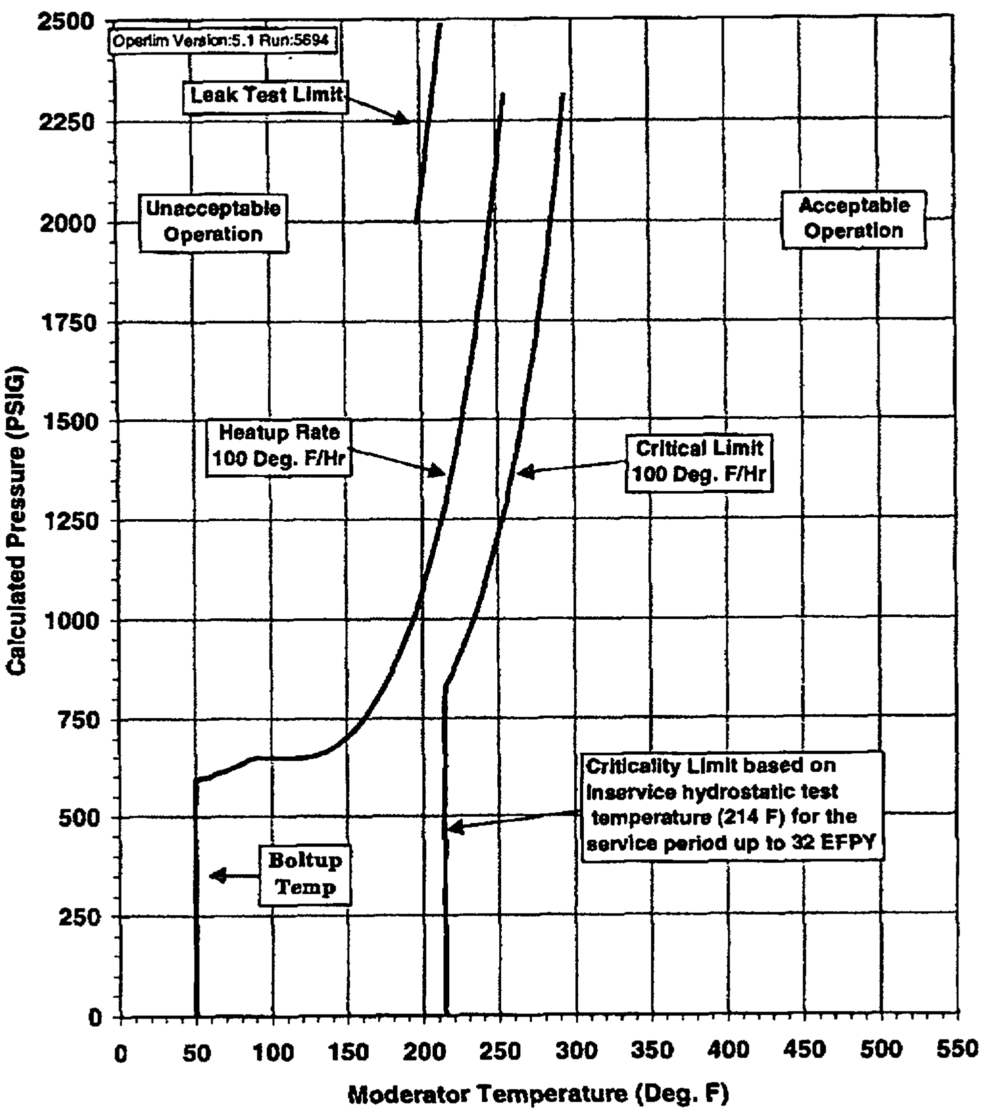


FIGURE 5 Sequoyah Unit 2 Reactor Coolant System Heatup Limitations (Heatup Rate of 100°F/hr) Applicable for the First 32 EFPY (With Margins of for Instrumentation Errors of 10°F and 60 psig)

MATERIAL PROPERTY BASIS

LIMITING MATERIAL: INTERMEDIATE SHELL FORGING 05
LIMITING ART VALUES AT 32 EFPY: 1/4T, 142°F
3/4T, 115°F

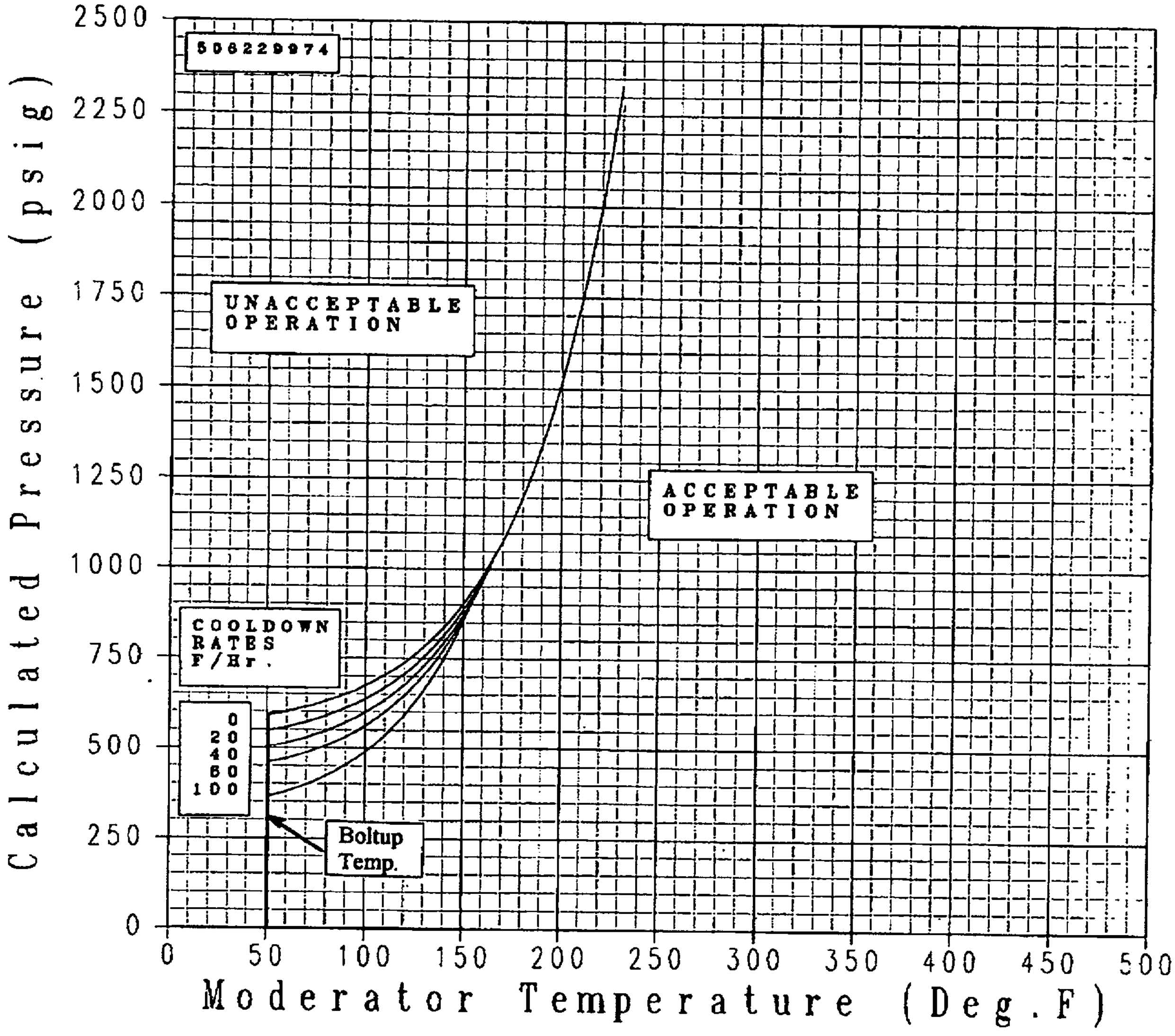


Figure 6 Sequoyah Unit 2 Reactor Coolant System Cooldown Limitations (Cooldown Rates up to 100°F/hr) Applicable for the First 32 EFPY (Without Margins for Instrumentation Errors of 10°F and 60 psig)

TABLE 11
32 EFPY Heatup Curve Data Points Using 1996 App.G
(without Uncertainties for Instrumentation Errors)

| Heatup Curves | | | | | | | | | |
|---|---|---|---|---|---|---|---|---|---|
| 60 Heatup | | 60 Limit Critical | | 100 Heatup | | 100 Limit Critical | | Leak Test Limit | |
| T | P | T | P | T | P | T | P | T | P |
| 50 | 0 | 203 | 0 | 50 | 0 | 203 | 0 | 186 | 2000 |
| 50 | 661 | 203 | 735 | 50 | 661 | 203 | 706 | 203 | 2485 |
| 55 | 667 | 203 | 738 | 55 | 667 | 203 | 707 | | |
| 60 | 674 | 203 | 745 | 60 | 674 | 203 | 708 | | |
| 65 | 682 | 203 | 755 | 65 | 682 | 203 | 713 | | |
| 70 | 690 | 203 | 768 | 70 | 690 | 203 | 721 | | |
| 75 | 700 | 203 | 785 | 75 | 700 | 203 | 731 | | |
| 80 | 710 | 203 | 805 | 80 | 706 | 203 | 745 | | |
| 85 | 721 | 203 | 828 | 85 | 706 | 203 | 761 | | |
| 90 | 734 | 203 | 855 | 90 | 706 | 203 | 780 | | |
| 95 | 735 | 203 | 885 | 95 | 706 | 203 | 803 | | |
| 100 | 738 | 203 | 920 | 100 | 706 | 203 | 829 | | |
| 105 | 745 | 203 | 958 | 105 | 706 | 203 | 858 | | |
| 110 | 755 | 203 | 1002 | 110 | 706 | 205 | 892 | | |
| 115 | 768 | 203 | 1050 | 115 | 708 | 210 | 929 | | |
| 120 | 785 | 205 | 1103 | 120 | 713 | 215 | 971 | | |
| 125 | 805 | 210 | 1163 | 125 | 721 | 220 | 1019 | | |
| 130 | 828 | 215 | 1228 | 130 | 731 | 225 | 1071 | | |
| 135 | 855 | 220 | 1301 | 135 | 745 | 230 | 1129 | | |
| 140 | 885 | 225 | 1382 | 140 | 761 | 235 | 1194 | | |
| 145 | 920 | 230 | 1471 | 145 | 780 | 240 | 1266 | | |
| 150 | 958 | 235 | 1570 | 150 | 803 | 245 | 1346 | | |
| 155 | 1002 | 240 | 1662 | 155 | 829 | 250 | 1434 | | |
| 160 | 1050 | 245 | 1757 | 160 | 858 | 255 | 1531 | | |
| 165 | 1103 | 250 | 1861 | 165 | 892 | 260 | 1639 | | |
| 170 | 1163 | 255 | 1976 | 170 | 929 | 265 | 1758 | | |
| 175 | 1228 | 260 | 2103 | 175 | 971 | 270 | 1889 | | |
| 180 | 1301 | 265 | 2243 | 180 | 1019 | 275 | 2034 | | |
| 185 | 1382 | 270 | 2398 | 185 | 1071 | 280 | 2194 | | |
| 190 | 1471 | | | 190 | 1129 | 285 | 2371 | | |
| 195 | 1570 | | | 195 | 1194 | | | | |
| 200 | 1662 | | | 200 | 1266 | | | | |
| 205 | 1757 | | | 205 | 1346 | | | | |

TABLE 11 (Continued)
**32 EFPY Heatup Curve Data Points Using 1996 App.G**
**(without Uncertainties for Instrumentation Errors)**

| Heatup Curves | | | | | | | | | |
|---|---|---|---|---|---|---|---|---|---|
| 60 Heatup | | 60 Limit Critical | | 100 Heatup | | 100 Limit Critical | | Leak Test Limit | |
| T | P | T | P | T | P | T | P | T | P |
| 210 | 1861 | | | 210 | 1434 | | | | |
| 215 | 1976 | | | 215 | 1531 | | | | |
| 220 | 2103 | | | 220 | 1639 | | | | |
| 225 | 2243 | | | 225 | 1758 | | | | |
| 230 | 2398 | | | 230 | 1889 | | | | |
| | | | | 235 | 2034 | | | | |
| | | | | 240 | 2194 | | | | |
| | | | | 245 | 2371 | | | | |

TABLE 12
32 EFPY Cooldown Curve Data Points Using 1996 App. G
(without Uncertainties for Instrumentation Errors)

| Cooldown Curves | | | | | | | | | |
|---|---|---|---|---|---|---|---|---|---|
| Steady State | | 20F | | 40F | | 60F | | 100F | |
| T | P | T | P | T | P | T | P | T | P |
| 50 | 0 | 50 | 0 | 50 | 0 | 50 | 0 | 50 | 0 |
| 50 | 661 | 50 | 618 | 50 | 574 | 50 | 530 | 50 | 440 |
| 55 | 667 | 55 | 624 | 55 | 581 | 55 | 538 | 55 | 449 |
| 60 | 674 | 60 | 632 | 60 | 589 | 60 | 546 | 60 | 459 |
| 65 | 682 | 65 | 640 | 65 | 598 | 65 | 556 | 65 | 470 |
| 70 | 690 | 70 | 649 | 70 | 608 | 70 | 566 | 70 | 483 |
| 75 | 700 | 75 | 659 | 75 | 619 | 75 | 578 | 75 | 497 |
| 80 | 710 | 80 | 670 | 80 | 631 | 80 | 591 | 80 | 513 |
| 85 | 721 | 85 | 683 | 85 | 644 | 85 | 606 | 85 | 530 |
| 90 | 734 | 90 | 696 | 90 | 659 | 90 | 622 | 90 | 550 |
| 95 | 748 | 95 | 712 | 95 | 676 | 95 | 640 | 95 | 572 |
| 100 | 763 | 100 | 728 | 100 | 694 | 100 | 660 | 100 | 596 |
| 105 | 780 | 105 | 747 | 105 | 714 | 105 | 682 | 105 | 623 |
| 110 | 799 | 110 | 767 | 110 | 736 | 110 | 707 | 110 | 653 |
| 115 | 820 | 115 | 790 | 115 | 761 | 115 | 734 | 115 | 686 |
| 120 | 843 | 120 | 815 | 120 | 789 | 120 | 764 | 120 | 723 |
| 125 | 869 | 125 | 843 | 125 | 819 | 125 | 798 | 125 | 764 |
| 130 | 897 | 130 | 874 | 130 | 853 | 130 | 835 | 130 | 809 |
| 135 | 928 | 135 | 908 | 135 | 891 | 135 | 876 | 135 | 860 |
| 140 | 962 | 140 | 945 | 140 | 932 | 140 | 922 | 140 | 916 |
| 145 | 1000 | 145 | 987 | 145 | 978 | 145 | 973 | 145 | 978 |
| 150 | 1042 | 150 | 1033 | 150 | 1028 | 150 | 1029 | | |
| 155 | 1088 | 155 | 1084 | 155 | 1085 | | | | |
| 160 | 1140 | | | | | | | | |
| 165 | 1196 | | | | | | | | |
| 170 | 1259 | | | | | | | | |
| 175 | 1328 | | | | | | | | |
| 180 | 1404 | | | | | | | | |
| 185 | 1489 | | | | | | | | |
| 190 | 1582 | | | | | | | | |
| 195 | 1685 | | | | | | | | |
| 200 | 1799 | | | | | | | | |
| 205 | 1925 | | | | | | | | |
| 210 | 2064 | | | | | | | | |
| 215 | 2218 | | | | | | | | |
| 220 | 2388 | | | | | | | | |

TABLE 13
32 EFPY Heatup Curve Data Points Using 1996 App.G
(with Uncertainties for Instrumentation Errors of 10°F and 60 psig)

| Heatup Curves | | | | | | | | | |
|---|---|---|---|---|---|---|---|---|---|
| 60 Heatup | | 60 Limit Critical | | 100 Heatup | | 100 Limit Critical | | Leak Test Limit | |
| T | P | T | P | T | P | T | P | T | P |
| 50 | 0 | 214 | 0 | 50 | 0 | 214 | 0 | 198 | 2000 |
| 50 | 591 | 214 | 607 | 50 | 591 | 214 | 607 | 214 | 2485 |
| 55 | 595 | 214 | 614 | 55 | 595 | 214 | 614 | | |
| 60 | 601 | 214 | 622 | 60 | 601 | 214 | 622 | | |
| 65 | 607 | 214 | 675 | 65 | 607 | 214 | 657 | | |
| 70 | 614 | 214 | 678 | 70 | 614 | 214 | 650 | | |
| 75 | 622 | 214 | 685 | 75 | 622 | 214 | 647 | | |
| 80 | 630 | 214 | 695 | 80 | 630 | 214 | 646 | | |
| 85 | 640 | 214 | 708 | 85 | 640 | 214 | 648 | | |
| 90 | 650 | 214 | 725 | 90 | 646 | 214 | 653 | | |
| 95 | 661 | 214 | 745 | 95 | 646 | 214 | 661 | | |
| 100 | 674 | 214 | 768 | 100 | 646 | 214 | 671 | | |
| 105 | 675 | 214 | 795 | 105 | 646 | 214 | 685 | | |
| 110 | 678 | 214 | 825 | 110 | 646 | 214 | 701 | | |
| 115 | 685 | 214 | 860 | 115 | 646 | 214 | 720 | | |
| 120 | 695 | 214 | 898 | 120 | 646 | 214 | 743 | | |
| 125 | 708 | 214 | 942 | 125 | 648 | 214 | 769 | | |
| 130 | 725 | 214 | 990 | 130 | 653 | 214 | 798 | | |
| 135 | 745 | 215 | 1043 | 135 | 661 | 215 | 832 | | |
| 140 | 768 | 220 | 1103 | 140 | 671 | 220 | 869 | | |
| 145 | 795 | 225 | 1168 | 145 | 685 | 225 | 911 | | |
| 150 | 825 | 230 | 1241 | 150 | 701 | 230 | 959 | | |
| 155 | 860 | 235 | 1322 | 155 | 720 | 235 | 1011 | | |
| 160 | 898 | 240 | 1411 | 160 | 743 | 240 | 1069 | | |
| 165 | 942 | 245 | 1510 | 165 | 769 | 245 | 1134 | | |
| 170 | 990 | 250 | 1602 | 170 | 798 | 250 | 1206 | | |
| 175 | 1043 | 255 | 1697 | 175 | 832 | 255 | 1286 | | |
| 180 | 1103 | 260 | 1801 | 180 | 869 | 260 | 1374 | | |
| 185 | 1168 | 265 | 1916 | 185 | 911 | 265 | 1471 | | |
| 190 | 1241 | 270 | 2043 | 190 | 959 | 270 | 1579 | | |
| 195 | 1322 | 275 | 2183 | 195 | 1011 | 275 | 1698 | | |
| 200 | 1411 | 280 | 2338 | 200 | 1069 | 280 | 1829 | | |
| 205 | 1510 | | | 205 | 1134 | 285 | 1974 | | |
| 210 | 1602 | | | 210 | 1206 | 290 | 2134 | | |
| 215 | 1697 | | | 215 | 1286 | 295 | 2311 | | |
| 220 | 1801 | | | 220 | 1374 | | | | |

TABLE 13 (Continued)
32 EFPY Heatup Curve Data Points Using 1996 App.G
(with Uncertainties for Instrumentation Errors of 10°F and 60 psig)

| Heatup Curves | | | | | | | | | |
|---|---|---|---|---|---|---|---|---|---|
| 60 Heatup | | 60 Limit Critical | | 100 Heatup | | 100 Limit Critical | | Leak Test Limit | |
| T | P | T | P | T | P | T | P | T | P |
| 225 | 1916 | | | 225 | 1471 | | | | |
| 230 | 2043 | | | 230 | 1579 | | | | |
| 235 | 2183 | | | 235 | 1698 | | | | |
| 240 | 2338 | | | 240 | 1829 | | | | |
| | | | | 245 | 1974 | | | | |
| | | | | 250 | 2134 | | | | |
| | | | | 255 | 2311 | | | | |

TABLE 14
32 EFPY Cooldown Curve Data Points Using 1996 App. G
(with Uncertainties for Instrumentation Errors of 10°F and 60 psig)

| Cooldown Curves | | | | | | | | | |
|---|---|---|---|---|---|---|---|---|---|
| Steady State | | 20F | | 40F | | 60F | | 100F | |
| T | P | T | P | T | P | T | P | T | P |
| 50 | 0 | 50 | 0 | 50 | 0 | 50 | 0 | 50 | 0 |
| 50 | 591 | 50 | 552 | 50 | 503 | 50 | 461 | 50 | 366 |
| 55 | 595 | 55 | 554 | 55 | 508 | 55 | 466 | 55 | 372 |
| 60 | 601 | 60 | 558 | 60 | 514 | 60 | 470 | 60 | 380 |
| 65 | 607 | 65 | 564 | 65 | 521 | 65 | 478 | 65 | 389 |
| 70 | 614 | 70 | 572 | 70 | 529 | 70 | 486 | 70 | 399 |
| 75 | 622 | 75 | 580 | 75 | 538 | 75 | 496 | 75 | 410 |
| 80 | 630 | 80 | 589 | 80 | 548 | 80 | 506 | 80 | 423 |
| 85 | 640 | 85 | 599 | 85 | 559 | 85 | 518 | 85 | 437 |
| 90 | 650 | 90 | 610 | 90 | 571 | 90 | 531 | 90 | 453 |
| 95 | 661 | 95 | 623 | 95 | 584 | 95 | 546 | 95 | 470 |
| 100 | 674 | 100 | 636 | 100 | 599 | 100 | 562 | 100 | 490 |
| 105 | 688 | 105 | 652 | 105 | 616 | 105 | 580 | 105 | 512 |
| 110 | 703 | 110 | 668 | 110 | 634 | 110 | 600 | 110 | 536 |
| 115 | 720 | 115 | 687 | 115 | 654 | 115 | 622 | 115 | 563 |
| 120 | 739 | 120 | 707 | 120 | 676 | 120 | 647 | 120 | 593 |
| 125 | 760 | 125 | 730 | 125 | 701 | 125 | 674 | 125 | 626 |
| 130 | 783 | 130 | 755 | 130 | 729 | 130 | 704 | 130 | 663 |
| 135 | 809 | 135 | 783 | 135 | 759 | 135 | 738 | 135 | 704 |
| 140 | 837 | 140 | 814 | 140 | 793 | 140 | 775 | 140 | 749 |
| 145 | 868 | 145 | 848 | 145 | 831 | 145 | 816 | 145 | 800 |
| 150 | 902 | 150 | 885 | 150 | 872 | 150 | 862 | 150 | 856 |
| 155 | 940 | 155 | 927 | 155 | 918 | 155 | 913 | 155 | 918 |
| 160 | 982 | 160 | 973 | 160 | 968 | 160 | 969 | | |
| 165 | 1028 | 165 | 1024 | 165 | 1025 | | | | |
| 170 | 1080 | | | | | | | | |
| 175 | 1136 | | | | | | | | |
| 180 | 1199 | | | | | | | | |
| 185 | 1268 | | | | | | | | |
| 190 | 1344 | | | | | | | | |
| 195 | 1429 | | | | | | | | |
| 200 | 1522 | | | | | | | | |
| 205 | 1625 | | | | | | | | |
| 210 | 1739 | | | | | | | | |
| 215 | 1865 | | | | | | | | |
| 220 | 2004 | | | | | | | | |
| 225 | 2158 | | | | | | | | |
| 230 | 2328 | | | | | | | | |

# 6 REFERENCES


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3. ASME Code Case N-640, "Alternative Reference Fracture Toughness for Development of P-T Limit Curves for Section XI, Division 1", February 26, 1999.

4. Section XI of the ASME Boiler and Pressure Vessel Code, Appendix G, "Fracture Toughness Criteria for Protection Against Failure.", Dated 1989 & December 1995.

5. "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.

6. WCAP-15315, "Reactor Vessel Closure Head/Vessel Flange Requirements Evaluation For Operating PWR and BWR Plants", W. Bamford, et.al., October 1999.

7. WCAP-15320, "Analysis of Capsule Y From the Tennessee Valley Authority Sequoyah Unit 2 Reactor Vessel Radiation Surveillance Program", T.J. Laubham, et. al., Dated December 1999.

8. 1989 Section III, Division 1 of the ASME Boiler and Pressure Vessel Code, Paragraph NB-2331, "Material for Vessels."

9. CVGRAPH, Hyperbolic Tangent Curve-Fitting Program, Version 4.1, developed by ATI Consulting, March 1999.

10. Code of Federal Regulations, 10 CFR Part 50, Appendix G, "Fracture Toughness Requirements," U.S. Nuclear Regulatory Commission, Washington, D.C., Federal Register, Volume 60, No. 243, dated December 19, 1995.

11. WCAP-7924-A, "Basis for Heatup and Cooldown Limit Curves," W. S. Hazelton, et al., April 1975.

12. Calc. No. 92-016, "WOG USE Program - Onset of Upper Shelf Energy Calculations", J. M. Chicots, dated 11/12/92. File # WOG-108/4-18 (MUHP-5080).

13. WCAP-12971, "Heatup and Cooldown Limit Curves for Normal Operation Sequoyah Unit 2". J.M. Chicots, et. al., Dated June, 1991.

14. Westinghouse Letter TVA-91-242

15. Westinghouse Letter TVA-91-243

16. TVA letter Number N9664, "TASK N99-017 - Reactor Coolant System Pressure and Temperature Limit Report Development - N2N-048," W. M. Justice, August 17, 1999.

17. TVA letter Number N9667, "TASK N99-017 - Reactor Coolant System Pressure and Temperature Limit Report Development - N2N-048," W. M. Justice, August 20, 1999.

18. Westinghouse Letter TVA-93-105.

# APPENDIX A

# LTOPS SETPOINTS

## 1.1 INTRODUCTION

Westinghouse has been requested to develop Low Temperature Overpressure Protection System (LTOPS) setpoints for Sequoyah Unit 2, for a vessel exposure of 32 EFPY. The LTOPS setpoints for the Sequoyah, Units 1 and 2, were last revised by Westinghouse in June of 1991 using pressure-temperature limits supplied by Tennessee Valley Authority for a vessel exposure of 16 EFPY. The results of this analysis were reported via References 14 and 15. The June 1991 analysis was based on the September 1989 analysis which addressed: Eagle-21 implementation and Appendix G limits based on Regulatory Guide 1.99. This Section documents the development of new Sequoyah Unit 2 COMS setpoints for 32 EFPY.

The Low Temperature Overpressure Protection System (LTOPS) is designed to provide the capability, during relatively low temperatures Reactor Coolant System (RCS) operation (typically less than 350°F), to protect the reactor vessel from being exposed to conditions of fast propagating brittle fracture. The LTOPS is provided in addition to the administrative controls, to prevent overpressure transients and as a supplement to the RCS overpressure mitigation function of the Residual Heat Removal System (RHRS) relief valves. LTOP consists of pressurizer PORVs and actuation logic from the wide range pressure channels. Once the system is enabled, no operator action is involved for the LTOPS to perform its intended pressure mitigation function.

LTOPS setpoints are conservatively selected to prevent exceeding the pressure/temperature limits established by 10 CFR Part 50 Appendix G requirements.

Two specific transients have been defined as the design basis for LTOPS. Each of these transient scenarios assume that the RCS is in a water-solid condition and that the RHRS is isolated from the RCS. The first transient consists of a heat injection scenario in which a reactor coolant pump in a single loop is started with the RCS temperature as much as 50°F lower than the steam generator secondary side temperature and the RHRS has been inadvertently isolated. This results in a sudden heat input to the RCS from the steam generators, creating an increasing pressure transient. The second transient has been defined as a mass injection scenario into the RCS caused by the simultaneous isolation of the RHRS, isolation of letdown and failure of the normal charging flow controls to the full flow condition. The resulting mass injection/letdown mismatch causes an increasing pressure transient.

## 1.2 LTOPS SETPOINT DETERMINATION

Westinghouse has developed new LTOPS setpoints for Sequoyah Unit 2, based on a vessel exposure of 32 EFPY using the methodology established in WCAP-14040 (Ref. 2). This methodology maximizes the available operating margin for setpoint selection while maintaining an appropriate level of protection in support of reactor vessel integrity. Note, Appendix G pressure limit relaxation allowed by ASME Code Case N-514 was not applied.

Plant design characteristics are unchanged (i.e., heat injection and mass injection transients characteristics and related plant responses have not been altered). Therefore, a complete reanalysis is not required. The new LTOPS setpoints were developed using results of the previous heat and mass injection transient analyses.

### 1.2.1 Pressure Limits Selection

The function of the LTOPS is to protect the reactor vessel from fast propagating brittle fracture. This has been implemented by choosing LTOPS setpoints, which prevent exceeding the limits prescribed by the applicable pressure/temperature characteristic for the specific reactor vessel material in accordance with rules given in Appendix G to 10CFR50. The LTOPS design basis takes credit for the fact that overpressure events most likely occur during isothermal conditions in the RCS. Therefore, it is appropriate to utilize the steady-state Appendix G limit. The Sequoyah Unit 2, 10CFR50 Appendix G curve for 32 EFPY is shown by Figure A-1. This curve sets the nominal upper limit on the pressure which should not be exceeded during RCS increasing pressure transients based on reactor vessel material properties.

When a relief valve is actuated to mitigate an increasing pressure transient, the system pressure then decreases, as the relief valve releases coolant, until a reset pressure is reached where the valve is signaled to close. Note that the pressure continues to decrease below the reset pressure as the valve re-closes. The nominal lower limit on the pressure during the transient is typically established based solely on an operational consideration for the RCP #1 seal to maintain a nominal differential pressure across the seal faces for proper film-riding performance. The RCP #1 seal limit is shown in Figure A-1.

The nominal upper limit (based on the minimum of the steady-state 10CFR50 Appendix G requirement) and the nominal lower limit (based on RCP #1 seal performance criteria) create a pressure range from which the setpoints for both PORVs may be selected.

### 1.2.2 Mass Input Consideration

For a particular mass input transient to the RCS, the relief valve will be signaled to open at a specific pressure setpoint. However, there will be a pressure overshoot during the delay time before the valve starts to move and during the time the valve is moving to the full open position. This overshoot is dependent on the dynamics of the system and the input parameters, and results in a maximum system pressure somewhat higher than the set pressure. Similarly there will be a pressure undershoot, while the valve is relieving, both due to the reset pressure being below the setpoint and to the delay in stroking the valve closed.

The previous Sequoyah analyses of multiple mass input cases were used to determine the relationship between setpressures and resulting overshoots/undershoots.

### 1.2.3 Heat Input Consideration

The heat input case is done similarly to the mass input case except that the locus of transient pressure values versus selected setpoints may be determined for several values of the initial RCS temperature. This heat input evaluation provides a range of acceptable setpoints dependent on the reactor coolant temperature, whereas the mass input case is limited to the most restrictive low temperature condition only (i.e., the mass injection transient is not sensitive to temperature).

The previous Sequoyah analyses of multiple heat input cases were used to determine the relationship between setpressures and resulting overshoots/undershoots.

### 1.2.4 Final Setpoint Selection

Appendix G limits described in Section 1.2.1, were conservatively adjusted accounting for the pressure difference ($\Delta P$) between the wide range pressure transmitter and the reactor vessel limiting beltline region of 68.3 psi for 4 RCPs in operation (See Reference 18).

The results of the analyses described in Section 1.2.2 & 1.2.3, and the adjusted Appendix G limit were used to define the maximum allowable setpoints for which the overpressure will not exceed the pressure limit applicable at a specific reactor vessel temperature. The maximum allowable setpoints are shown in Figure A-1.

Per Ref. 17, Sequoyah Demonstrated Accuracy Calculation SQN-IC014 establishes the instrument loop inaccuracy of the Sequoyah temperature and pressure instrument channels associated with the LTOPS. Previously, the LTOPS setpoints have been provided to TVA without application of instrument uncertainties. The TVA calculation quantified the instrument channel uncertainties, applied them to the nominal Westinghouse setpoints and evaluated the result against the safety limits established by the 10CFR50, Appendix G steady state heatup curve. The TVA demonstrated accuracy calculation will be revised to reflect the LTOPS setpoints calculated by Westinghouse under the subject task. As such, it is not necessary for Westinghouse to include instrument uncertainties in the nominal LTOPS setpoint calculation. Note, the heat injection results were adjusted to include 50°F thermal transport effect (difference in temperature between the RCS and steam generator at transient initiation).

The maximum allowable setpoints, adjusted to produce a smoother curve and reduced to nine data points, becomes the setpoints for PORV#2. A setpoint at a minimum temperature of 50°F was selected, as requested by TVA (Ref. 16). Each of the two PORVs may have a different pressure setpoint such that only one valve will open at a time and mitigate the transient (i.e., staggered setpoints). The second valve operates only if the first fails to open on command. This design supports a single failure assumption as well as minimizing the potential for both PORVs to open simultaneously, a condition which may create excessive pressure undershoot and challenge the RCP #1 seal performance criteria. The PORV#1 setpoints were selected by adjusting the setpoint PORV#2 in relationship to the overshoots discussed in Sections 1.2.2 and 1.2.3. The selected setpoints for PORV #1 and PORV #2 are shown in Table A-1 and Figure A-2. These setpoints were evaluated using the undershoots discussed in Sections 1.2.2 and 1.2.3 to ensure that they protect against the RCP # 1 seal limit.

In summary, the selection of the setpoints for LTOPS considered the use of nominal upper and lower pressure limits. The upper limits are specified by the minimum of the steady-state cooldown curve as calculated in accordance with Appendix G to 10CFR50 (adjusted to account for four RCPs in operation). The lower pressure extreme is specified by the reactor coolant pump #1 seal minimum differential pressure performance criteria. The selected setpoints, shown in Table A-1 and Figure A-2, provide protection against Appendix G and RCP #1 seal limit violations. Note, these setpoints do not address instrumentation uncertainties.

## 1.3 ARMING AND ENABLE TEMPERATURES FOR LTOPS

The LTOPS arming temperature is traditionally based on the temperature corresponding to when Appendix G pressure equals 2500 psia. Based on this methodology the LTOPS arming temperature for Sequoyah conservatively continues to be 350°F.

The enable temperature is the temperature below which the LTOPS system is required to be operable, based on vessel materials concerns. ASME Code Case N-514 requires the LTOPS to be in operation at coolant temperatures less than 200°F or at coolant temperatures less than a temperature corresponding to a reactor vessel metal temperature less than $RT_{NDT}$ + 50°F, whichever is greater. $RT_{NDT}$ is the highest Adjusted Reference Temperature (ART) for the limiting belt-line material at a distance one fourth of the vessel section thickness from the vessel inside surface (i.e., clad/base metal interface), as determined by Regulatory Guide 1.99, Revision 2. The minimum required enable temperature for the Sequoyah Unit 2 Reactor Vessel is 225°F at 32 EFPY of operation.

Table A-1
Selected Setpoints, Sequoyah Unit 2

| Trcs (Deg.F) | PORV#2 Setpoint (psig) | PORV#1 Setpoint (psig) |
|---|---|---|
| 50 | 510 | 485 |
| 100 | 580 | 555 |
| 135 | 640 | 610 |
| 174 | 745 | 682 |
| 200 | 745 | 685 |
| 250 | 745 | 685 |
| 278 | 745 | 685 |
| 400 | 745 | 685 |
| 450 | 2350 | 2350 |

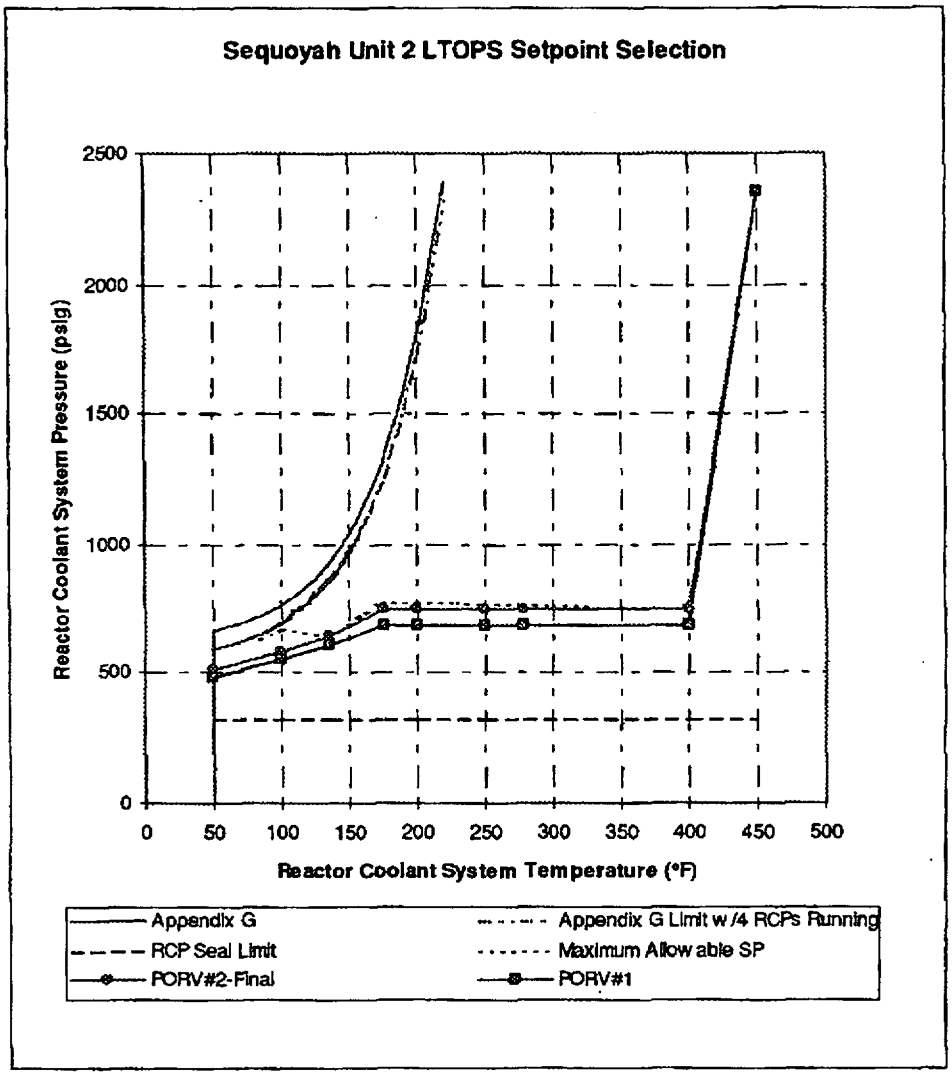


Figure A-1 Sequoyah Unit 2 LTOPS Setpoint Selection

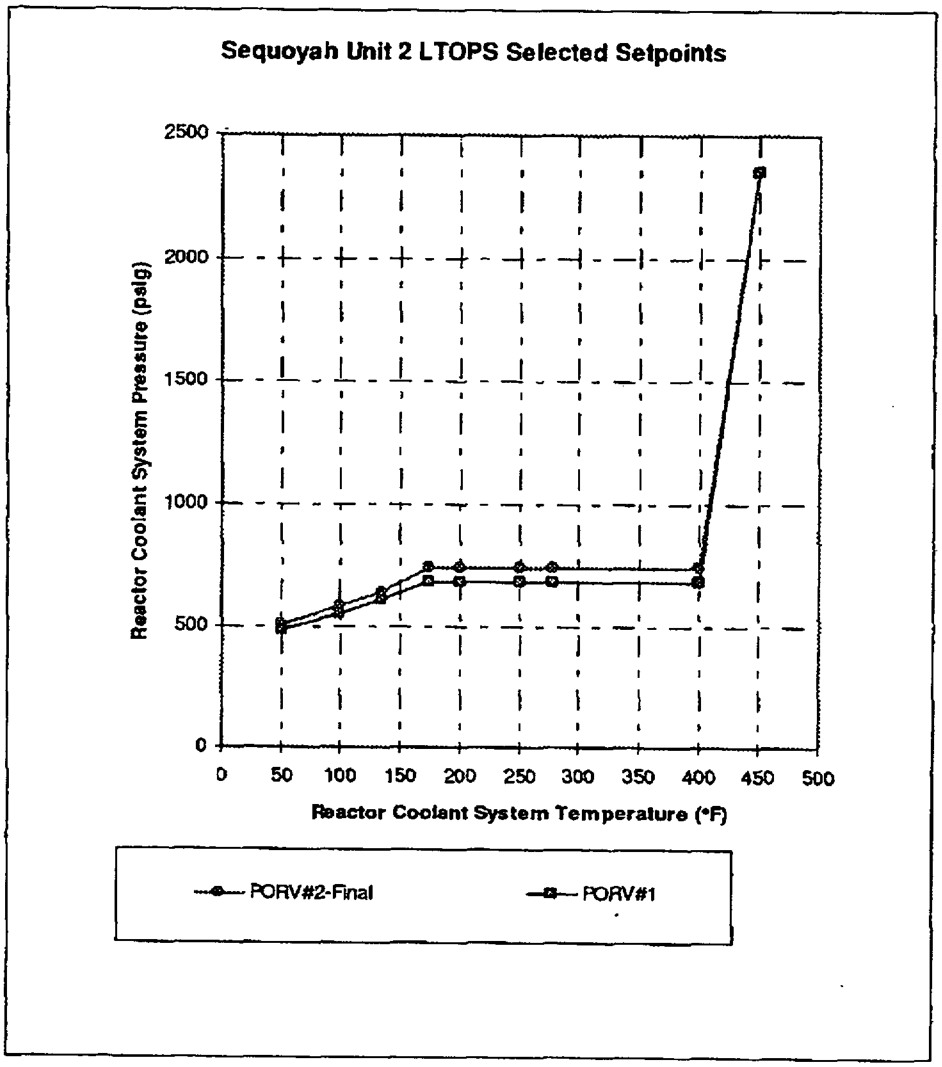


Figure A-2 - Sequoyah Unit 2 LTOPS Selected Setpoints

# APPENDIX B

# PRESSURIZED THERMAL SHOCK (PTS) RESULTS

**PTS Calculations:**

The PTS Rule requires that for each pressurized water nuclear power reactor for which an operating license has been issued, the licensee shall have projected values of $RT_{PTS}$, accepted by the NRC, for each reactor vessel beltline material for the EOL fluence of the material. This assessment must specify the basis for the projected value of $RT_{PTS}$ for each vessel beltline material, including the assumptions regarding core loading patterns, and must specify the copper and nickel contents and the fluence value used in the calculation. This assessment must be updated whenever there is a significant change in projected values of $RT_{PTS}$, or upon request for a change in the expiration date for operation of the facility. (Changes to $RT_{PTS}$ values are considered significant if either the previous value or the current value, or both values, exceed the screening criterion prior to the expiration of the operating license, including any renewed term, if applicable, for the plant.

To verify that $RT_{NDT}$, for each vessel beltline material is a bounding value for the specific reactor vessel, licensees shall consider plant-specific information that could affect the level of embrittlement. This information includes but is not limited to the reactor vessel operating temperature and any related surveillance program results. (Surveillance program results mean any data that demonstrates the embrittlement trends for the limiting beltline material, including but not limited to data from test reactors or from surveillance programs at other plants with or without surveillance program integrated per 10 CFR Part 50, Appendix H.)

**Calculations:**

Tables B-1 and B-2 contain the results of the calculations for each of the beltline region materials in the Sequoyah Unit 2 Reactor Vessel. Per TVA, the EOL is 32 EFPY and the Life Extension EOL is 48 EFPY.

TABLE B-1

$RT_{PTS}$ Calculations for Sequoyah Unit 2 Beltline Region Materials at 32 EFPY

| Material | Fluence ($n/cm^2$, E>1.0 MeV) | FF | CF (°F) | $\Delta RT_{PTS}$[c] (°F) | Margin (°F) | $RT_{NDT(U)}$[a] (°F) | $RT_{PTS}$[b] (°F) |
|---|---|---|---|---|---|---|---|
| Intermediate Shell Forging 05 | 1.82 | 1.164 | 95 | 110.6 | 34 | 10 | 155 |
| Intermediate Shell Forging 05 (Using S/C Data) | 1.82 | 1.164 | 89.7 | 104.4 | 34 | 10 | 148 |
| Lower Shell Forging 04 | 1.82 | 1.164 | 104 | 121.1 | 34 | -22 | 133 |
| Circumferential Weld Metal | 1.82 | 1.164 | 63 | 73.3 | 56 | -4 | 125 |
| Circumferential Weld Metal (Using S/C Data) | 1.82 | 1.164 | 77.8 | 90.6 | 56 | -4 | 143 |

Notes:

(a) Initial $RT_{NDT}$ values are measured values

(b) $RT_{PTS} = RT_{NDT(U)} + \Delta RT_{PTS}$ + Margin (°F)

(c) $\Delta RT_{PTS}$ = CF * FF

TABLE B-2

$RT_{PTS}$ Calculations for Sequoyah Unit 2 Beltline Region Materials at 48 EFPY

| Material | Fluence ($n/cm^2$, E>1.0 MeV) | FF | CF (°F) | $\Delta RT_{PTS}$[c] (°F) | Margin (°F) | $RT_{NDT(U)}$[a] (°F) | $RT_{PTS}$[b] (°F) |
|---|---|---|---|---|---|---|---|
| Intermediate Shell Forging 05 | 2.71 | 1.266 | 95 | 120.3 | 34 | 10 | 164 |
| Intermediate Shell Forging 05 (Using S/C Data) | 2.71 | 1.266 | 89.7 | 113.6 | 34 | 10 | 158 |
| Lower Shell Forging 04 | 2.71 | 1.266 | 104 | 131.7 | 34 | -22 | 144 |
| Circumferential Weld Metal | 2.71 | 1.266 | 63 | 79.8 | 56 | -4 | 132 |
| Circumferential Weld Metal (Using S/C Data) | 2.71 | 1.266 | 77.8 | 98.5 | 56 | -4 | 151 |

Notes:

(a) Initial $RT_{NDT}$ values are measured values

(b) $RT_{PTS} = RT_{NDT(U)} + \Delta RT_{PTS}$ + Margin (°F)

(c) $\Delta RT_{PTS}$ = CF * FF

All of the beltline materials in the Sequoyah Unit 2 reactor vessel are below the screening criteria values of 270°F and 300°F at 32 and 48 EFPY.

# APPENDIX C

# CALCULATED FLUENCE DATA

The best estimate exposure of the Sequoyah Unit 2 reactor vessel presented in WCAP-15320[7] was developed using a combination of absolute plant specific transport calculations and all available plant specific measurement data. The evaluation is consistent with the methodology accepted by the NRC and documented in WCAP-14040-NP-A[2].

Combining this measurement data base with the plant-specific calculations, the best estimate vessel exposure is obtained from the following relationship:

$$\Phi_{Best\,Est.} = K\,\Phi_{Calc.}$$

where:

$\Phi_{Best\,Est.}$ = The best estimate fast neutron exposure at the location of interest.

K = The plant specific best estimate/calculation (BE/C) bias factor derived from the surveillance capsule dosimetry data.

$\Phi_{Calc.}$ = The absolute calculated fast neutron exposure at the location of interest.

For Sequoyah Unit 2, the derived plant specific bias factors were 0.93, 0.98, 0.96 for $\Phi(E > 1.0\ MeV)$, $\Phi(E > 0.1\ MeV)$, and dpa, respectively. Bias factors of this magnitude developed with BUGLE-96 are within expected tolerances for fluence calculated using the ENDF/B-VI based cross-section library.

Table C-1 presents the reactor vessel fast neutron (E > 1.0 MeV) exposure projections using the absolute plant specific calculations. Table C-2 presents the calculated and measured fluences at the capsules.

**Table C-1**
**Azimuthal Variations Of The Neutron Exposure Projections**
**On The Reactor Vessel Clad/Base Metal Interface At Core Midplane**

Calculated

| | 0° | 15° | 30° | 45°[a] |
|---|---|---|---|---|
| | | 10.54 EFPY | | |
| E>1.0 MeV | 2.11E+18 | 3.36E+18 | 4.26E+18 | 6.37E+18 |
| E>0.1 MeV | 5.37E+18 | 8.50E+18 | 1.11E+19 | 1.70E+19 |
| dpa | 3.43E-03 | 5.39E-03 | 6.88E-03 | 1.03E-02 |
| | | 20 EFPY | | |
| E>1.0 MeV | 3.80E+18 | 6.00E+18 | 7.73E+18 | 1.16E+19 |
| E>0.1 MeV | 9.65E+18 | 1.52E+19 | 2.01E+19 | 3.10E+19 |
| dpa | 6.16E-03 | 9.61E-03 | 1.25E-02 | 1.88E-02 |
| | | 32 EFPY | | |
| E>1.0 MeV | 5.93E+18 | 9.34E+18 | 1.21E+19 | 1.82E+19 |
| E>0.1 MeV | 1.51E+19 | 2.36E+19 | 3.16E+19 | 4.88E+19 |
| dpa | 9.63E-03 | 1.50E-02 | 1.96E-02 | 2.95E-02 |
| | | 48 EFPY | | |
| E>1.0 MeV | 8.78E+18 | 1.38E+19 | 1.80E+19 | 2.71E+19 |
| E>0.1 MeV | 2.23E+19 | 3.49E+19 | 4.68E+19 | 7.24E+19 |
| dpa | 1.42E-02 | 2.21E-02 | 2.91E-02 | 4.38E-02 |

Note:
a) Maximum neutron exposure projection

Table C-2
Comparison Of Calculated And Best Estimate Integrated Neutron Exposure Of Sequoyah Unit 2 Surveillance Capsules T, U, X, and Y

CAPSULE T

| | Calculated | Best Estimate | BE/C |
|---|---|---|---|
| $\Phi$(E > 1.0 MeV) [n/cm$^2$] | 2.61E+18 | 2.57E+18 | 0.98 |
| $\Phi$(E > 0.1 MeV) [n/cm$^2$] | 8.74E+18 | 8.98E+18 | 1.03 |
| dpa | 4.34E-03 | 4.36E-03 | 1.01 |

CAPSULE U

| | Calculated | Best Estimate | BE/C |
|---|---|---|---|
| $\Phi$(E > 1.0 MeV) [n/cm$^2$] | 6.92E+18 | 6.03E+18 | 0.87 |
| $\Phi$(E > 0.1 MeV) [n/cm$^2$] | 2.31E+19 | 2.11E+19 | 0.91 |
| dpa | 1.15E-02 | 1.03E-02 | 0.90 |

CAPSULE X

| | Calculated | Best Estimate | BE/C |
|---|---|---|---|
| $\Phi$(E > 1.0 MeV) [n/cm$^2$] | 1.22E+19 | 1.04E+19 | 0.85 |
| $\Phi$(E > 0.1 MeV) [n/cm$^2$] | 4.09E+19 | 3.63E+19 | 0.89 |
| dpa | 2.03E-02 | 1.77E-02 | 0.87 |

CAPSULE Y

| | Calculated | Best Estimate | BE/C |
|---|---|---|---|
| $\Phi$(E > 1.0 MeV) [n/cm$^2$] | 2.14E+19 | 2.18E+19 | 1.02 |
| $\Phi$(E > 0.1 MeV) [n/cm$^2$] | 7.14E+19 | 7.72E+19 | 1.08 |
| dpa | 3.54E-02 | 3.70E-02 | 1.05 |

AVERAGE BE/C RATIOS

| | BE/C |
|---|---|
| $\Phi$(E > 1.0 MeV) [n/cm$^2$] | 0.93 |
| $\Phi$(E > 0.1 MeV) [n/cm$^2$] | 0.98 |
| dpa | 0.96 |

# APPENDIX D

## UPDATED SURVEILLANCE MATERIAL 30 FT-LB TRANSITION TEMPERATURE SHIFTS AND UPPER SHELF ENERGY DECREASES

TABLE D-1

Measured 30 ft-lb Transition Temperature Shifts of all Available Surveillance Data

| Material | Capsule | Fluence (x $10^{19}$ n/cm$^2$) | 30 ft-lb Transition Temperature Shift | | Upper Shelf Energy Decrease | |
|---|---|---|---|---|---|---|
| | | | Predicted (°F) [a] | Measured (°F) [b] | Predicted (%) [a] | Measured (%)[c] |
| Intermediate Shell Forging 05 (Tangential) | T | 0.261 | 60.33 | 63.65 | 17 | 12 |
| | U | 0.692 | 85.22 | 79.31 | 21 | 16 |
| | X | 1.22 | 100.23 | 85.7 | 23 | 8 |
| | Y | 2.14 | 114.67 | 134.12 | 26 | 22 |
| Intermediate Shell Forging 05 (Axial) | T | 0.261 | 60.33 | 48.73 | 17 | 7 |
| | U | 0.692 | 85.22 | 66.06 | 21 | 9 |
| | X | 1.22 | 100.23 | 110.04 | 23 | 2 |
| | Y | 2.14 | 114.67 | 89.21 | 26 | 22 |
| Weld Metal | T | 0.261 | 43.12 | 74.56 | 20 | 2 |
| | U | 0.692 | 60.91 | 130.38 | 25 | 6 |
| | X | 1.22 | 71.63 | 44.22 | 29 | 35 |
| | Y | 2.14 | 81.96 | 86.91 | 33 | 3 |
| HAZ Metal | T | 0.261 | -- | 24.58 | -- | 2 |
| | U | 0.692 | -- | 64.03 | -- | 14 |
| | X | 1.22 | -- | 28.29 | -- | 19 |
| | Y | 2.14 | -- | 50.32 | -- | 39 |

Notes:

(a) Based on Regulatory Guide 1.99, Revision 2, methodology using the mean weight percent values of copper and nickel of the surveillance material.

(b) Calculated using measured Charpy data plotted using CVGRAPH, Version 4.1 (Reference 9)

(c) Values are based on the definition of upper shelf energy given in ASTM E185-82.

# APPENDIX E

# REACTOR VESSEL BELTLINE MATERIAL PROJECTED END OF LICENSE UPPER SHELF ENERGY VALUES

TABLE E-1
Predicted End-of-License (32 EFPY) USE Calculations for all the Beltline Region Materials

| Material | Weight % of Cu | 1/4T EOL Fluence ($10^{19}$ n/cm$^2$) | Unirradiated USE[(a)] (ft-lb) | Projected USE Decrease (%) | Projected EOL USE (ft-lb) |
|---|---|---|---|---|---|
| Intermediate Shell Forging 05 Using S/C Data | 0.13 | 1.10 | 93 | 18.5 | 76 |
| Lower Shell Forging 04 | 0.14 | 1.10 | 100 | 23 | 77 |
| Intermediate to Lower Shell Circumferential Weld Seam Using S/C Data | 0.12 | 1.10 | 102 | 35 | 66 |

Notes:
(a) These values were obtained from Reference 12.

# APPENDIX F

# UPDATED SURVEILLANCE CAPSULE REMOVAL SCHEDULE

The following surveillance capsule removal schedule meets the requirements of ASTM E185-82 and is recommended for future capsules to be removed from the Sequoyah Unit 2 reactor vessel. This recommended removal schedule is applicable to 32 EFPY of operation.

**Table F-1 Sequoyah Unit 2 Reactor Vessel Surveillance Capsule Withdrawal Schedule**

| Capsule | Location | Lead Factor[a] | Removal Time (EFPY)[b] | Fluence ($n/cm^2$, E>1.0 MeV)[a] |
|---|---|---|---|---|
| T | 40° | 3.33 | 1.04 | $2.61 \times 10^{18}$ (c) |
| U | 140° | 3.40 | 2.93 | $6.92 \times 10^{18}$ (c) |
| X | 220° | 3.39 | 5.36 | $1.22 \times 10^{19}$ (c) |
| Y | 320° | 3.35 | 10.54 | $2.14 \times 10^{19}$ (c,d) |
| S | 4° | 1.09 | Standby | (d,e) |
| V | 176° | 1.09 | Standby | (d,e) |
| W | 184° | 1.09 | Standby | (d,e) |
| Z | 356° | 1.09 | Standby | (d,e) |

Notes:

(a) Updated in Capsule Y dosimetry analysis (Reference 7).

(b) Effective Full Power Years (EFPY) from plant startup.

(c) Plant specific evaluation.

(d) This fluence is not less than once or greater than twice the peak end of license (32 EFPY) fluence

(e) Capsules S, V, W and Z will reach a fluence of $2.71 \times 10^{19}$ (E > 1.0 MeV), the 48 EFPY peak vessel fluence at approximately 44 EFPY, respectively. If vessel fluence data is needed at the EOL for Life Extension, it is recommended that one or more of the Standby Capsules be moved to a higher flux location within the next few cycles of operation.

# APPENDIX G

# ENABLE TEMPERATURE CALCULATIONS AND RESULTS

**Enable Temperature Calculation:**

ASME Code Case N-514 requires the low temperature overpressure (LTOP or COMS) system to be in operation at coolant temperatures less than 200°F or at coolant temperatures less than a temperature corresponding to a reactor vessel metal temperature less than $RT_{NDT}$ + 50°F, whichever is greater. $RT_{NDT}$ is the highest adjusted reference temperature (ART) for the limiting beltline material at a distance one fourth of the vessel section thickness from the vessel inside surface (ie. clad/base metal interface), as determined by Regulatory Guide 1.99, Revision 2.

32 EFPY

The highest calculated 1/4T ART for the Sequoyah Unit 2 reactor vessel beltline region at 32 EFPY is 142°F.

From the OPERLIM computer code output for the Sequoyah Unit 2 32 EFPY P-T limit curves without margins (Configuration # 1676409813, *operlim.film* File) the maximum $\Delta T_{metal}$ is:

Cooldown Rate (Steady-State Cooldown):
max ($\Delta T_{metal}$) at 1/4T = 0°F

Heatup Rate of 100°F/Hr:
max ($\Delta T_{metal}$) at 1/4T = 28.924°F

**Enable Temperature (ENBT)** $= RT_{NDT} + 50 + \max(\Delta T_{metal})$, °F
$= (142 + 50 + 28.924)$ °F
$= 220.924$°F

The minimum required enable temperature for the Sequoyah Unit 2 Reactor Vessel is **225°F** at 32 EFPY of operation.

**ENCLOSURE 6**

**TENNESSEE VALLEY AUTHORITY**
**SEQUOYAH NUCLEAR PLANT (SQN)**
**UNITS 1 AND 2**

**JUSTIFICATION FOR EXEMPTION ALLOWING USE OF AMERICAN SOCIETY OF MECHANICAL ENGINEERS (ASME) CODE CASE N-640**

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In accordance with 10 CFR 50.12, "Specific exemptions," TVA is requesting an exemption from the requirements of 10 CFR 50.60, "Acceptance criteria for fracture prevention measures for lightwater nuclear power reactors for normal operation." The exemption would permit the use of the ASME Boiler and Pressure Vessel (B&PV) Code, Section XI Code Case N-640, "Alternative Requirement Fracture Toughness for Development of P-T Limit Curves for ASME Section XI, Division 1," in lieu of 10 CFR 50, Appendix G, paragraph IV.A.2.b. The proposed exemption meets the criteria of 10 CFR 50.12 as discussed below.

<u>10 CFR 50.12(a) Requirements</u>

10 CFR 50.12 states that the Commission may grant an exemption from requirements contained in 10 CFR 50 provided that the following is met:

1. The requested exemption is authorized by law.

   No law exists which precludes the activities covered by this exemption request. 10 CFR 50.60(b) allows the use of alternatives to 10 CFR 50, Appendix G, when an exemption is granted by the Commission under 10 CFR 50.12.

2. The requested exemption does not present an undue risk to the public health and safety.

   Revised P-T limit curves were developed for Sequoyah Units 1 and 2. The revised P-T limits use the $K_{Ic}$, fracture toughness curve shown on ASME XI, Appendix A, Figure A-4200-1, in lieu of the $K_{Ia}$, fracture toughness curve of ASME XI, Appendix G, Figure G-2210-1, as the lower bound for fracture toughness. The other margins involved with the ASME B&PV Code, Section XI, Appendix G process of determining P-T limit curves remain unchanged.

   Use of the $K_{Ic}$ curve in determining the lower bound fracture toughness in the development of P-T operating limits curve reduces the excess conservatism in the current Appendix G approach, the application of which could, in fact, reduce overall plant safety. The $K_{Ic}$ curve models the slow heat-up and cooldown process of a reactor pressure vessel. Use of this approach is justified given the initial conservatism built into the $K_{Ia}$ curve when the curve was codified in 1974. The initial conservatism was deemed necessary due to limited knowledge of reactor pressure vessel material fracture toughness. Since 1974, additional knowledge has been gained about the fracture toughness of reactor pressure vessel materials and their fracture response to applied loads. The additional knowledge demonstrates the lower bound fracture toughness provided by the $K_{Ia}$ curve is well beyond the margin of safety required to protect against potential reactor pressure vessel failure. The lower bound $K_{Ic}$ fracture toughness provides an adequate margin of safety to protect against potential reactor pressure vessel failure and does not present an undue risk to public health and safety.

P-T limit curves that are based on the $K_{Ic}$ fracture toughness limits enhance overall plant safety by opening the pressure-temperature operating window. Two primary safety benefits would be realized with no decrease to the margin of safety.

- Challenges to the operators would be reduced since the requirements for maintaining high vessel temperature during pressure testing would be lessened.

- Enhanced personnel safety would result because of the lower temperatures which would exist during the conduct of inspections in primary containment.

3. The requested exemption is consistent with the common defense and security.

   The subject of this exemption does not affect national defense or security issues. The common defense and security are not impacted by approval of this exemption request.

4. Special circumstances are present which necessitate the request for an exemption to the regulations of 10 CFR 50.60.

   In accordance with 10 CFR 50.12(a)(2), NRC will consider granting an exemption to the regulations if special circumstances are present. This requested exemption meets the special circumstances of the following paragraphs of 10 CFR 50.12:

   - (a)(2)(ii) - demonstrates the underlying purpose of the regulation will continue to be achieved;

   - (a)(2)(iii) - would result in undue hardship or other costs that are significant if the regulation is enforced and;

- (a)(2)(v) - will provide only temporary relief from the applicable regulation and the licensee has made good faith efforts to comply with the regulation.

10 CFR 50.12(a)(2)(ii): ASME B&PV Code, Section XI, Appendix G, provides procedures for determining allowable loading on the reactor pressure vessel and is approved for that purpose by 10 CFR 50, Appendix G. Application of these procedures in the determination of P-T operating and test curves satisfy the underlying requirement that:

- The reactor coolant pressure boundary be operated in a regime having sufficient margin to ensure, when stressed, the reactor pressure vessel boundary behaves in a non-brittle manner, and the probability of a rapidly propagating fracture is minimized, and

- P-T operating and test limit curves provide adequate margin in consideration of uncertainties in determining the effects of irradiation on material properties.

The requirements of ASME B&PV Code, Section XI, Appendix G, were conservatively developed based on the level of knowledge existing in 1974 concerning reactor pressure vessel materials and the estimated effects of operation. Since 1974, the level of experience and knowledge about these topics has increased. This increased experience and knowledge permits relaxation of ASME B&PV Code, Section XI, Appendix G, requirements via application of ASME Code Case N-640, while maintaining the underlying purpose of the ASME B&PV Code and the NRC regulations to ensure an acceptable margin of safety.

10 CFR 50.12(a)(2)(iii): The reactor pressure vessel pressure-temperature operating window is defined by the P-T limit curves developed in accordance with the ASME B&PV Code, Section XI, Appendix G requirements. Continued operation of Sequoyah Units 1 and 2 with these P-T limit curves without the relief provided by ASME Code Case N-640 would unnecessarily restrict the P-T operating window. This restriction challenges the operations staff during required pressure tests. The operator must maintain a high temperature within a limited operating window. In addition, the higher temperatures result in greater physical stress on the inspection personnel working in the vicinity of the piping.

This constitutes an unnecessary burden that can be alleviated by the application of ASME Code Case N-640 in the development of the proposed P-T limit curves. Implementation of the proposed P-T limit curves, as allowed by ASME Code Case N-640, does not reduce the margin of safety below acceptable limits.

10 CFR 50.12(a)(2)(v): The requested exemption provides only temporary relief from the applicable regulation. TVA has made a

good faith effort to comply with the regulation. TVA requests the exemption be granted until such time that the NRC generically approves the application of ASME Code Case N-640 for use by the nuclear industry.

Conclusion for Exemption Acceptability:

Compliance with the specified requirement of 10 CFR 50.60 would result in hardship and unusual difficulty without a compensating increase in the level of quality and safety. TVA's proposed application of ASME Code Case N-640 for SQN Units 1 and 2 allows a reduction in the lower bound fracture toughness used in ASME B&PV Code, Section XI, Appendix G, in the determination of reactor coolant system P-T limits. The proposed alternative is acceptable because the ASME code case maintains the relative margin of safety commensurate with that which existed at the time ASME B&PV Code, Section XI, Appendix G, was approved in 1974. Therefore, application of ASME Code Case N-640 for Sequoyah Units 1 and 2 will ensure an acceptable margin of safety and does not present an undue risk to the public health and safety.

**ENCLOSURE 7**

**TENNESSEE VALLEY AUTHORITY**
**SEQUOYAH NUCLEAR PLANT (SQN)**
**UNITS 1 AND 2**

**JUSTIFICATION FOR EXEMPTION ALLOWING**
**USE OF WCAP-15315**

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In accordance with 10 CFR 50.12, "Specific exemptions," TVA is requesting an exemption from the requirements of 10 CFR 50.60, "Acceptance criteria for fracture prevention measures for lightwater nuclear power reactors for normal operation." The exemption would permit the use of WCAP-15313, "Reactor Vessel Closure Head/Vessel Flange Requirements Evaluation for Operating PWR and BWR Plants," in lieu of the methodology required by 10 CFR 50, Appendix G, footnote 2 to Table 1. The WCAP demonstrates that the flange region is tolerant of assumed flaws in excess of 1/4 thickness at room temperature. In addition, since there is no known degradation mechanism for this region and the fatigue usage in this region is less than 0.1, it is concluded that flaws are unlikely to initiate in this region. Accordingly, justification exits for eliminating the reactor vessel head/flange region when determining the pressure-temperature (P-T) limits for the reactor vessel. The proposed exemption meets the criteria of 10 CFR 50.12 as discussed below.

<u>10 CFR 50.12(a) Requirements</u>

10 CFR 50.12 states that the Commission may grant an exemption from requirements contained in 10 CFR 50 provided that the following is met:

1. The requested exemption is authorized by law.

   No law exists which precludes the activities covered by this exemption request. 10 CFR 50.60(b) allows the use of alternatives to 10 CFR 50, Appendix G, when an exemption is granted by the Commission under 10 CFR 50.12.

2. The requested exemption does not present an undue risk to the public health and safety.

   The revised P-T curve limits being proposed for Sequoyah Units 1 and 2 rely on methodology described in WCAP-15313. The WCAP uses a lower stress intensity factor, $K_{IC}$ instead of $K_{Ia}$, which results in higher allowable pressures. The 10 CFR 50, Appendix G addresses the metal temperature of the closure head flange and vessel flange regions. The regulation states that the metal temperature of the closure flange regions must exceed the material unirradiated $RT_{NDT}$ by at least 120 degrees Fahrenheit (°F) for normal operation when the pressure exceeds 20 percent of the preservice hydrostatic test pressure. Implementing the $K_{IC}$ stress intensity factor without eliminating the flange requirement of 10 CFR 50, Appendix G, would place a restricted operating window in the temperature range associated with the flange/closure head (i.e., flange $RT_{NDT}$ + 120°F). In accordance with WCAP-15315, the $K_{IC}$ toughness has been shown to provide significant margin between the applied stress intensity factor and the fracture toughness of the flange/closure head. The conclusion of the WCAP analysis is the integrity of the closure head/flange is not a concern for safe plant operation and testing. Accordingly, the Sequoyah P-T limit curves are generated without flange requirements included.

3. The requested exemption is consistent with the common defense and security.

   The subject of this exemption does not affect national defense or security issues. The common defense and security are not impacted by approval of this exemption request.

4. Special circumstances are present which necessitate the request for an exemption to the regulations of 10 CFR 50.60.

   In accordance with 10 CFR 50.12(a)(2), NRC will consider granting an exemption to the regulations if special circumstances are present. This requested exemption meets the special circumstances of 10 CFR 50.12(a)(2)(ii) that states:

   "Application of the regulation in the particular circumstances would not serve the underlying purpose of the rule or is not necessary to achieve the underlying purpose of the rule;"

   The underlying purpose of 10 CFR 50.60 and 10 CFR 50, Appendix G is to protect the integrity of the reactor coolant pressure boundary. The results of the methods used in WCAP-15315 determined that significant margin exists between the applied stress intensity factor and the fracture toughness at virtually all crack depths when using the $K_{IC}$ toughness, which has been adopted by Section XI for developing Pressure-Temperature Limit Curves. Another objective of the requirements in 10 CFR 50, Appendix G is to assure that fracture margins are maintained to protect against service induced cracking due to environmental effects. Since the governing flaw is on the outside surface (the inside is in compression) where there are no environmental effects, there is even greater assurance of fracture margin. Therefore, it can be concluded that the integrity of the closure head/flange region is not a concern for SQN using the $K_{IC}$ toughness. In addition, there are no known mechanisms of degradation for this region, other than fatigue. The calculated design fatigue usage for this region is less than