**Tennessee Valley Authority,** Sequoyah Nuclear Plant, P.O. Box 2000, Soddy Daisy, Tennessee 37384

May 15, 2018

10 CFR 50.4

ATTN: Document Control Desk
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555-0001

Sequoyah Nuclear Plant, Unit 1
Renewed Facility Operating License Nos. DPR-77
NRC Docket Nos. 50-327

Subject: **Sequoyah Unit 1 Pressure Temperature Limits Report, Revision 6**

In accordance with Sequoyah Nuclear Plant (SQN) Unit 1 Technical Specification (TS) 5.6.4.c, enclosed is the Pressure Temperature Limits Report (PTLR), Revision 6. The PTLR has been revised for license renewal actions including: an updated neutron fluence evaluation to extend pressure-temperature (P-T) limits from 32 effective full power years (EFPY) to 52 EFPY; to document the evaluation of the reactor vessel materials in the extended beltline region; and to provide a revision to the surveillance capsule withdrawal schedule. The low temperature overpressure protection system setpoints have been revised for the pressurizer power operated relief valve (PORV) digital control system upgrade. The PORV digital control system upgrade was implemented on April 27, 2018, during the Unit 1 Cycle 22 Refueling Outage. Associated administrative changes were made for report consistency.

There are no new regulatory commitments in this letter. If you have any questions, please contact Michael McBrearty, SQN Site Licensing Manager at (423) 843-7170.

Respectfully,

Anthony L. Williams
Site Vice President
Sequoyah Nuclear Plant

Enclosure
Sequoyah Unit 1 Pressure Temperature Limits Report, Revision 6

ZTK:STB
Enclosure
cc (Enclosure):
NRC Regional Administrator – Region II
NRC Senior Resident Inspector – SQN

# ENCLOSURE

# SEQUOYAH UNIT 1
# PRESSURE TEMPERATURE LIMITS REPORT, REVISION 6

# Tennessee Valley Authority
# Sequoyah Unit 1

# Pressure Temperature Limits Report
# Revision 6, September 2017

# Tennessee Valley Authority
# Sequoyah Unit 1

# Pressure Temperature Limits Report
# Revision 6, September 2017

PROJECT Sequoyah DISCIPLINE N
CONTRACT 4411 UNIT 1
DESC. RCS Pressure-Temperature Limit Report
DWG/DOC NO. PTLR-1
SHEET - OF - REV. 06
DATE 04/30/18 ECN/DCN - FILE N2N-081

EDMS, WT CA-K

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# 1.0 RCS Pressure Temperature Limits Report (PTLR)

This PTLR for Sequoyah Unit 1 has been prepared in accordance with the requirements of Technical Specification (TS) 5.6.4. Revisions to the PTLR shall be provided to the NRC after issuance.

This report affects TS 3.4.3, RCS Pressure/Temperature Limits (P/T) Limits, TS 3.4.12, Low Temperature Over Pressure Protection (LTOP) System and TS 3.5.2, ECCS - Operating.

# 2.0 RCS Pressure and Temperature Limits

The limits for TS 3.4.3 are presented in the subsections which follow and were developed using the NRC approved methodologies specified in TS 5.6.4 with exception of ASME Code Case N-640[13] (Use of $K_{Ic}$), WCAP-15984-P[14] (Elimination of the Flange Requirement), 1996 Version of Appendix G[4] and the revised fluences[15]. The operability requirements associated with LTOPS are specified in TS 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 5.6.4.

2.1 RCS Pressure/Temperature (P/T) Limits (TS 3.4.3)

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 RCS P/T limits for heatup, cooldown, inservice hydrostatic and leak testing, and criticality are specified by Figures 2-1 and 2-2. These P/T limit curves were originally documented in WCAP-15293, Revision 2[10] and were applicable to 32 Effective Full-Power Years (EFPY). The applicability of the 32 EFPY P-T limit curves for Sequoyah Unit 1 was extended to 52 EFPY as a part of the technical evaluations documented in WCAP-17539-NP, Revision 1[15].

2.1.4 Nozzle P/T limit curves were developed at 52 EFPY as a part of the evaluations contained in MCOE-LTR-16-12-NP[16] to satisfy NRC Regulatory Issue Summary (RIS 2014-11)[17]. The nozzle curves are compared to the cylindrical shell P/T limit curves (developed using Reference 9 methodology) with a revised applicability of 52 EFPY in Reference 16, which shows that the nozzle P/T limit curves are bounded by the cylindrical shell P/T limit curves.

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

The lift setpoints for the pressurizer Power Operated Relief Valves (PORVs) are presented in the subsections which follow. These lift setpoints have been developed using the NRC-approved methodologies specified in TS 5.6.4.

### 3.1 Pressurizer PORV Lift Setting Limits

The pressurizer PORV lift setpoints are specified by Figure 3-1 and Table 3-1 (Ref. 10). The limits for the LTOPS setpoints are contained in the 52 EFPY steady-state curves (Table 2-2), which are beltline conditions and are not compensated for pressure differences between the pressurizer transmitter and the reactor midplane/beltline. However, these curves are adjusted 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. 11).

Note: These setpoints include allowance for the pressure difference between the pressurizer transmitter and the reactor vessel midplane/beltline and the 50°F thermal transport effect for heat injection transients. A demonstrated accuracy calculation (Reference 12) 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 5.6.4. 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 E23[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.

## 5.0 Supplemental Data Tables

Table 5-1 contains a comparison of measured surveillance material 30 ft-lb transition temperature shifts and upper shelf energy decreases with Regulatory Guide 1.99, Revision 2[6], predictions.

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 52 EFPY for each beltline material and extended beltline material in the Sequoyah Unit 1 reactor vessel. The limiting beltline material was the Lower Shell Forging 04.

Table 5-7 provides a summary of the adjusted reference temperature (ART) values of the Sequoyah Unit 1 reactor vessel beltline materials and extended beltline materials at the 1/4T and 3/4T locations for 52 EFPY.

Table 5-8 shows the maximum ART value at 52 EFPY for each inlet and outlet nozzle in the Sequoyah Unit 1 reactor vessel.

Table 5-9 provides a summary of the limiting ART values of the Sequoyah Unit 1 inlet and outlet nozzles at 52 EFPY.

Table 5-10 provides $RT_{PTS}$ values for Sequoyah Unit 1 at 52 EFPY.

MATERIAL PROPERTY BASIS

LIMITING MATERIAL: LOWER SHELL FORGING 04
LIMITING ART VALUES AT 52 EFPY: 1/4T, 216°F
3/4T, 186°F

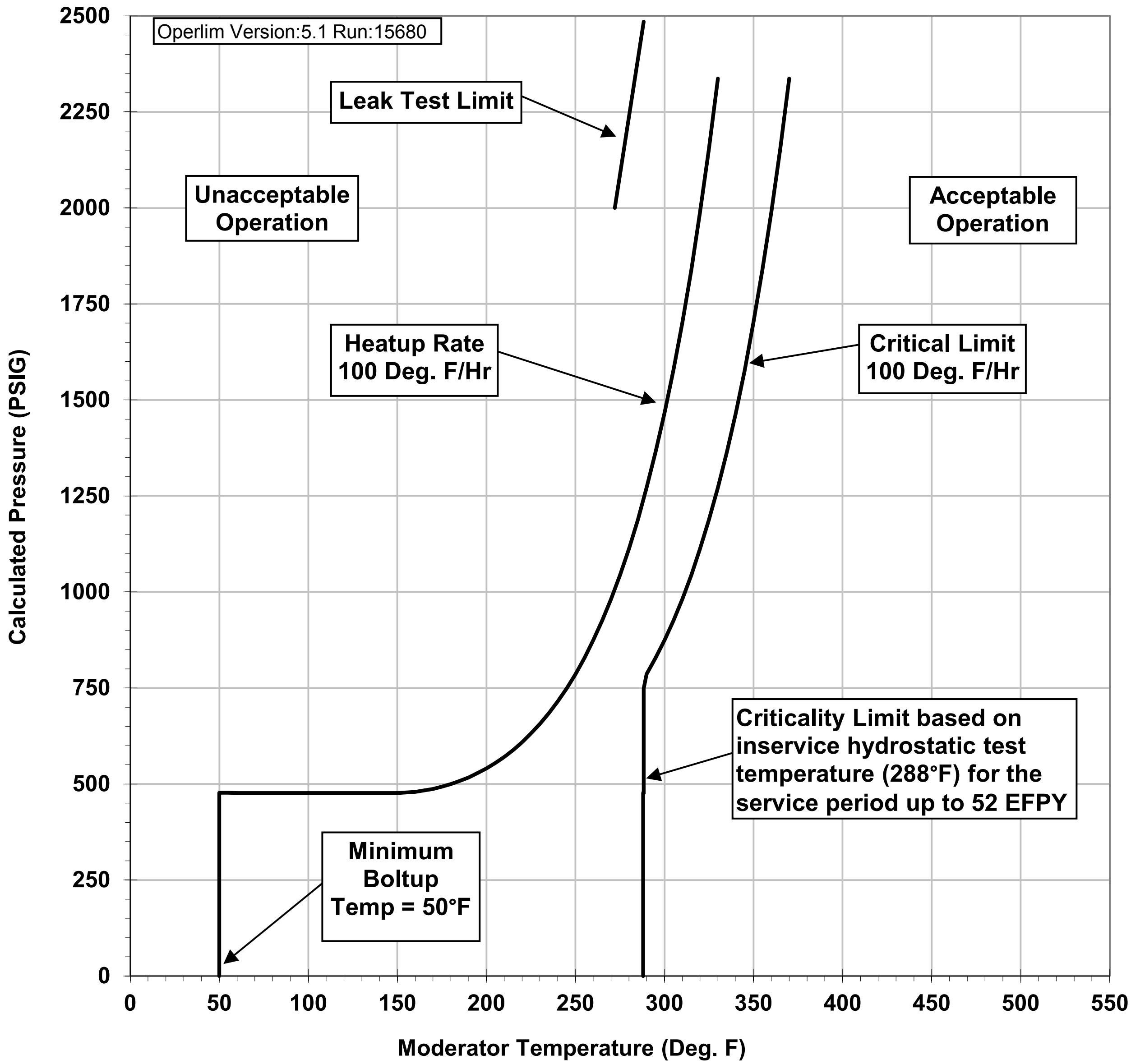


**Figure 2-1** **Sequoyah Unit 1 Reactor Coolant System Heatup Limitations (Heatup Rate of 100°F/hr) Applicable for the First 52 EFPY (w/Margins for Instrumentation Error of 10°F and 60 psig)** ***(Plotted Data provided on Table 2-1)***

MATERIAL PROPERTY BASIS

LIMITING MATERIAL: LOWER SHELL FORGING 04
LIMITING ART VALUES AT 52 EFPY: 1/4T, 216°F
3/4T, 186°F

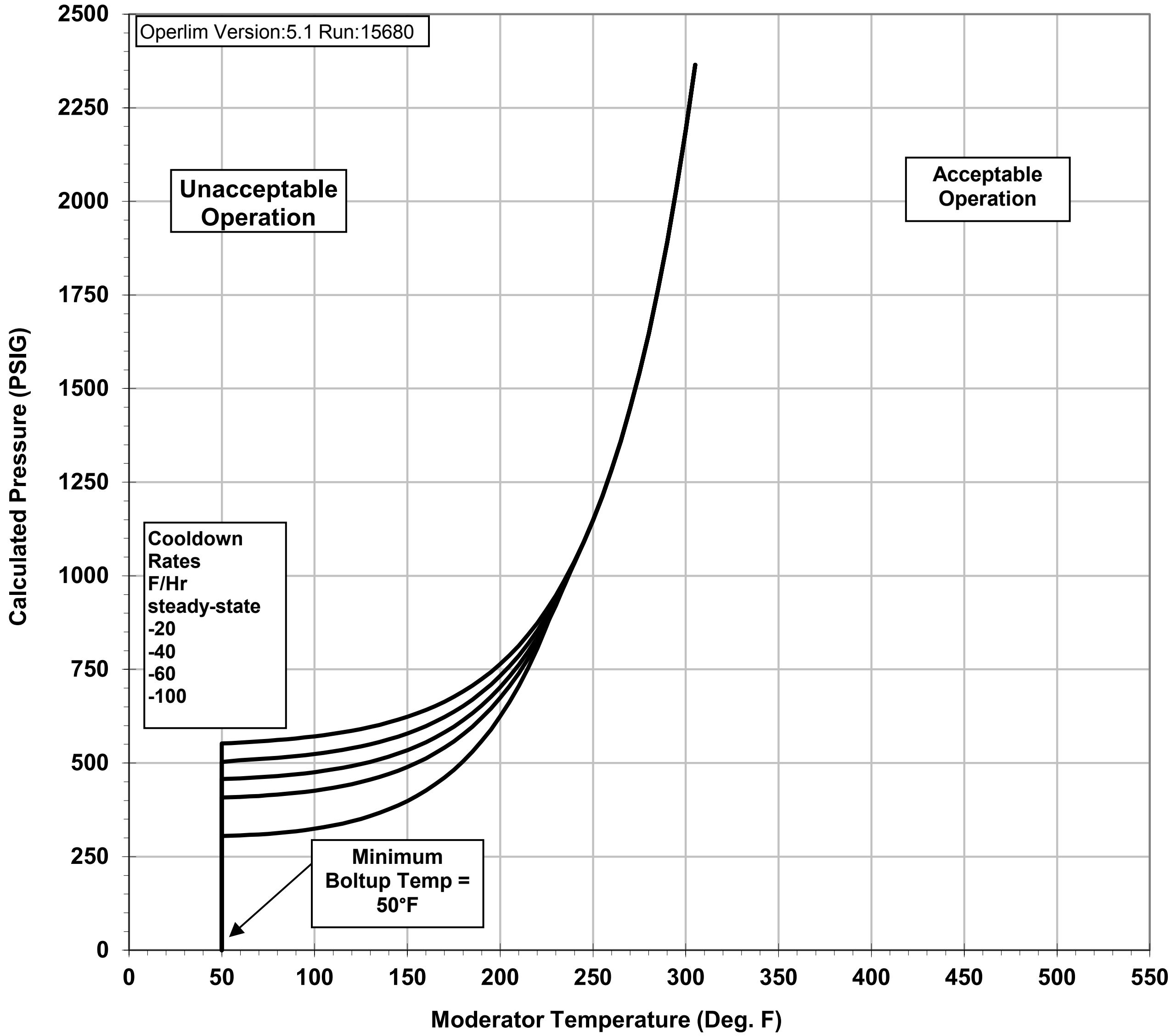


**Figure 2-2 Sequoyah Unit 1 Reactor Coolant System Cooldown Limitations (Cooldown Rates up to 100°F/hr) Applicable for the First 52 EFPY (w/Margins for Instrumentation Error of 10°F and 60 psig)** ***(Plotted Data provided on Table 2-2)***

Table 2-1
Sequoyah Unit 1 Heatup Limits at 52 EFPY
(with Uncertainties for Instrumentation Errors of 10°F and 60 psig)

| 100 Heatup | | 100 Critical Limit | | Leak Test Limit | |
|---|---|---|---|---|---|
| T | P | T | P | T | P |
| 50 | 0 | 288 | 0 | 272 | 2000 |
| 50 | 477 | 288 | 477 | 288 | 2485 |
| 55 | 477 | 288 | 477 | | |
| 60 | 477 | 288 | 477 | | |
| 65 | 477 | 288 | 477 | | |
| 70 | 477 | 288 | 478 | | |
| 75 | 477 | 288 | 478 | | |
| 80 | 477 | 288 | 480 | | |
| 85 | 477 | 288 | 481 | | |
| 90 | 477 | 288 | 483 | | |
| 95 | 477 | 288 | 485 | | |
| 100 | 477 | 288 | 487 | | |
| 105 | 477 | 288 | 490 | | |
| 110 | 477 | 288 | 493 | | |
| 115 | 477 | 288 | 497 | | |
| 120 | 477 | 288 | 500 | | |
| 125 | 477 | 288 | 505 | | |
| 130 | 477 | 288 | 508 | | |
| 135 | 477 | 288 | 515 | | |
| 140 | 477 | 288 | 517 | | |
| 145 | 477 | 288 | 527 | | |
| 150 | 477 | 288 | 528 | | |
| 155 | 478 | 288 | 541 | | |
| 160 | 480 | 288 | 541 | | |
| 165 | 483 | 288 | 555 | | |
| 170 | 487 | 288 | 557 | | |
| 175 | 493 | 288 | 571 | | |
| 180 | 500 | 288 | 575 | | |
| 185 | 508 | 288 | 589 | | |
| 190 | 517 | 288 | 609 | | |
| 195 | 528 | 288 | 631 | | |
| 200 | 541 | 288 | 656 | | |
| 205 | 555 | 288 | 684 | | |
| 210 | 571 | 288 | 714 | | |
| 215 | 589 | 288 | 748 | | |
| 220 | 609 | 290 | 786 | | |
| 225 | 631 | 295 | 828 | | |

Table 2-1 - (Continued)
Sequoyah Unit 1 Heatup Limits at 52 EFPY
(with Uncertainties for Instrumentation Errors of 10°F and 60 psig)

| 100 Heatup | | 100 Critical Limit | | |
|---|---|---|---|---|
| T | P | T | P | |
| 230 | 656 | 300 | 874 | |
| 235 | 684 | 305 | 925 | |
| 240 | 714 | 310 | 981 | |
| 245 | 748 | 315 | 1044 | |
| 250 | 786 | 320 | 1112 | |
| 255 | 828 | 325 | 1188 | |
| 260 | 874 | 330 | 1272 | |
| 265 | 925 | 335 | 1364 | |
| 270 | 981 | 340 | 1466 | |
| 275 | 1044 | 345 | 1578 | |
| 280 | 1112 | 350 | 1702 | |
| 285 | 1188 | 355 | 1838 | |
| 290 | 1272 | 360 | 1988 | |
| 295 | 1364 | 365 | 2154 | |
| 300 | 1466 | 370 | 2337 | |
| 305 | 1578 | | | |
| 310 | 1702 | | | |
| 315 | 1838 | | | |
| 320 | 1988 | | | |
| 325 | 2154 | | | |
| 330 | 2337 | | | |

Table 2-2
Sequoyah Unit 1 Cooldown Limits at 52 EFPY
(with Uncertainties for Instrumentation Errors of 10°F and 60 psig)

| 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 2-2 - (Continued)
Sequoyah Unit 1 Cooldown Limits at 52 EFPY
(with Uncertainties for Instrumentation Errors)

| 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 | | |
| 235 | 991 | 235 | 982 | 235 | 978 | | | | |
| 240 | 1038 | 240 | 1034 | | | | | | |
| 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 | | | | | | | | |

Table 3-1
Selected Setpoints[(a)],Sequoyah Unit 1

| Indicated RCS Temperature (°F) | PCV-456[(b)] Setpoint (psig) | PCV-455A[(c)] Setpoint (psig) |
|---|---|---|
| 50 | 450 | 406 |
| 73 | 450 | 406 |
| 123 | 474 | 430 |
| 148 | 498 | 455 |
| 223 | 584 | 525 |
| 277 | 715 | 626 |
| 373 | 715 | 626 |
| 500 | 2335 | 2335 |

Notes:
(a) From Reference 18 and confirmed per Reference 19.
(b) PCV-456 is PORV#2.
(c) PCV-455A is PORV#1.

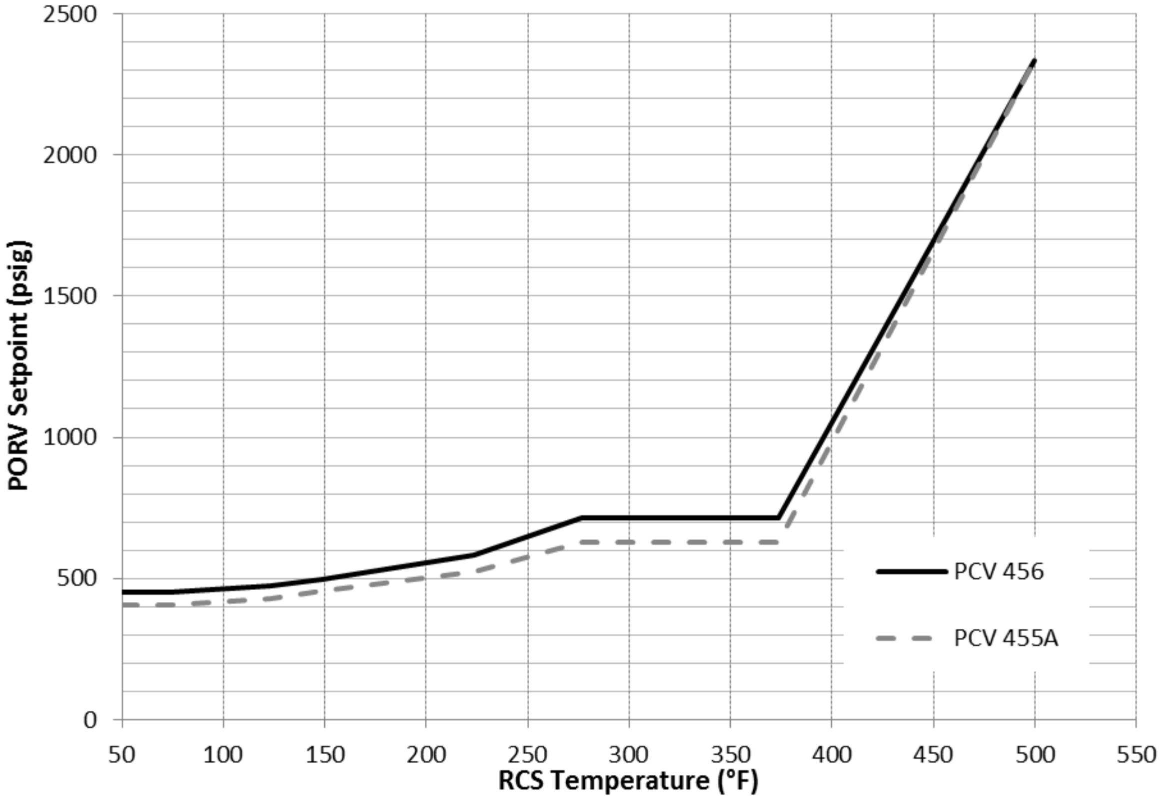


**Figure 3-1 Sequoyah Unit 1 - COMS PORV Setpoints vs. Indicated RCS Temperature [18]**

**(Includes Pressure and Temperature Instrument Uncertainties. *Plotted Data provided on Table 3-1*)**

Table 4-1
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° | 3.15 | 1.07 | $2.41 \times 10^{18}$ (c) |
| U | 140° | 3.23 | 2.85 | $6.93 \times 10^{18}$ (c) |
| X | 220° | 3.22 | 5.26 | $1.16 \times 10^{19}$ (c) |
| Y | 320° | 3.18 | 10.02 | $1.97 \times 10^{19}$ (c,d) |
| S | 4° | 0.90 | (e) | (e) |
| W | 184° | 0.90 | (e) | (e) |
| V | 176° | 0.90 | (f) | (f) |
| Z | 356° | 0.90 | (g) | (g) |

Notes:

(a) Updated in the time-limiting aging analysis (TLAA) fluence evaluation (WCAP-17539-NP Revision 1[15]).

(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 and W, which were relocated during the 1R19 outage, were found to be severely damaged during the 1R20 outage. Thus, they are no longer available to be used to satisfy the requirements of 10 CFR 50, Appendix H with consideration of license renewal to 60 total years of plant operation.

(f) Capsules V and Z are the only capsules currently remaining in the Sequoyah Unit 1 reactor vessel. Either Capsule V or Z should be withdrawn so that the capsule fluence corresponds to at least one times the 60-year EOL vessel fluence ($2.66 \times 10^{19}$ $n/cm^2$), but less than two times the 60-year EOL vessel fluence ($5.32 \times 10^{19}$ $n/cm^2$). However, neither of these two remaining capsules are predicted to experience a neutron fluence of $2.66 \times 10^{19}$ $n/cm^2$ prior to EOLE in their current locations. Therefore, TVA has elected to relocate Capsule V (or Capsule Z, which is radiologically equivalent to Capsule V, and could be alternatively selected) to a higher lead factor location, specifically the former Capsule U Location (140°), in order to achieve higher capsule fluence data. Note that Capsule V or Capsule Z could alternatively be moved to any open 40° capsule location, as they are all radiologically equivalent. Assuming Capsule V was relocated at the end of cycle 21, 22, or 23, the EFPY that corresponds to the time when the capsule experiences the peak EOLE vessel fluence value ($2.66 \times 10^{19}$ $n/cm^2$) is approximately 35.4, 36.4, or 37.3 EFPY, respectively. See Appendix B of Reference 15 for further details on capsule relocation recommendations[15].

(g) TVA has elected, for now, to leave Capsule Z in the reactor vessel at its current 356° capsule location. This capsule could be relocated to a higher lead factor location in the future if additional metallurgical data is needed in support of a potential second license renewal to 80 total years of plant operation. Note that Capsule Z would follow an equivalent subsequent withdrawal schedule to Capsule V if it is relocated to any 40° capsule location during EOC 21, 22 or 23, since Capsule Z and Capsule V are both currently at 4° effective azimuthal locations and are thus radiologically equivalent.

Table 5-1
Comparison of the Sequoyah Unit 1 Surveillance Material 30 ft-lb Transition Temperature Shifts and Upper Shelf Energy Decreases with Regulatory Guide 1.99, Revision 2, Predictions

| 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) (Heat # 980919 / 281587) | T | 0.241 | 59.85 | 67.52 | 16 | 16 |
| | U | 0.693 | 89.3 | 109.7 | 20.5 | 21 |
| | X | 1.16 | 102.6 | 145.12 | 23 | 8 |
| | Y | 1.97 | 114.95 | 129.87 | 26.5 | 23 |
| Lower Shell Forging 04 (Axial) (Heat # 980919 / 281587) | T | 0.241 | 59.85 | 50.59 | 16 | 0 |
| | U | 0.693 | 89.3 | 67.59 | 20.5 | 19 |
| | X | 1.16 | 102.6 | 103.34 | 23 | 22 |
| | Y | 1.97 | 114.95 | 133.35 | 26.5 | 19 |
| Weld Metal (Heat # 25295)[(d)] | T | 0.241 | 111.13 | 127.79 | 35 | 30 |
| | U | 0.693 | 165.82 | 144.92 | 42 | 26 |
| | X | 1.16 | 190.51 | 159.02 | 45 | 21 |
| | Y | 1.97 | 213.44 | 163.8 | 48 | 28 |
| HAZ Metal | T | 0.241 | - - | 45.48 | - - | 20 |
| | U | 0.693 | - - | 78.94 | - - | 26 |
| | X | 1.16 | - - | 95.89 | - - | 3 |
| | Y | 1.97 | - - | 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.

(d) Surveillance Weld was fabricated from weld wire type SMIT 40, Heat # 25295, Flux type SMIT 89, Lot # 1103.

Table 5-2
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) (Heat # 980919 / 281587) | T | 2.41E+18 | 0.615 | 67.52°F | 41.52°F | 0.378 |
| | U | 6.93E+18 | 0.897 | 109.70°F | 98.42°F | 0.805 |
| | X | 1.16E+19 | 1.041 | 145.12°F | 151.13°F | 1.085 |
| | Y | 1.97E+19 | 1.185 | 129.87°F | 153.92°F | 1.405 |
| Lower Shell Forging 04 (Axial) (Heat # 980919 / 281587) | T | 2.41E+18 | 0.615 | 50.59°F | 31.11°F | 0.378 |
| | U | 6.93E+18 | 0.897 | 67.59°F | 60.64°F | 0.805 |
| | X | 1.16E+19 | 1.041 | 103.34°F | 107.62°F | 1.085 |
| | Y | 1.97E+19 | 1.185 | 133.35°F | 158.04°F | 1.405 |
| | | | | SUM: | 802.39°F | 7.344 |
| | $CF_{04} = \Sigma(FF * RT_{NDT}) \div \Sigma(FF^2) = (802.39) \div (7.344) = 109.3°F$ | | | | | |
| Surveillance Weld Material[(d)] (Heat # 25295)[(e)] | T | 2.41E+18 | 0.615 | 115.01°F (127.79°F) | 70.72°F | 0.378 |
| | U | 6.93E+18 | 0.897 | 130.43°F (144.92°F) | 117.01°F | 0.805 |
| | X | 1.16E+19 | 1.041 | 143.12°F (159.02°F) | 149.05°F | 1.085 |
| | Y | 1.97E+19 | 1.185 | 147.42°F (163.80°F) | 174.72°F | 1.405 |
| | | | | SUM: | 511.50°F | 3.672 |
| | $CF_{Surv. Weld} = \Sigma(FF * RT_{NDT}) \div \Sigma(FF^2) = (511.50°F) \div (3.672) = 139.3°F$ | | | | | |

Notes:

(a) f = Calculated fluence, ($n/cm^2$, E > 1.0 MeV), updated per TLAA, WCAP-17539-NP, Revision 1[15].

(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 Table 5-10 of WCAP-15224[7].

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

(e) Surveillance Weld was fabricated from weld wire type SMIT 40, Heat # 25295, Flux type SMIT 89, Lot # 1103.

Table 5-3
Reactor Vessel Beltline and Extended Beltline Material
Unirradiated Toughness Properties for Sequoyah Unit 1

| Material Description | Chemical Composition | | | | Fracture Toughness Properties | |
|---|---|---|---|---|---|---|
| | Cu (Wt. %) | Ni (Wt. %) | P (Wt. %) | Mn (Wt. %) | Initial $RT_{NDT}$[(a)] (°F) | Initial Upper-Shelf Energy (ft-lb) |
| Reactor Vessel Beltline Materials[(b)] | | | | | | |
| Intermediate Shell Forging 05 (Heat # 980807/281489) | 0.15 | 0.86 | 0.011 | 0.70 | 40 | 79 |
| Lower Shell Forging 04 (Heat # 980919/281587) | 0.13 | 0.76 | 0.015 | 0.62 | 73 | 72 |
| Intermediate to Lower Shell Forging Circumferential Weld Seam W05 (Heat # 25295) [(c, e)] | 0.35 | 0.11 | 0.021 | 1.47 | -40 | 113 |
| Surveillance Weld [(d, e)] (Heat # 25295) | 0.39 | 0.11 | 0.021 | 1.40 | --- | --- |
| Reactor Vessel Extended Beltline Materials | | | | | | |
| Upper Shell Forging 06 (Heat # 980950/282758)[(f)] | 0.16 | 0.89 | 0.011 | 0.70 | 23 | 83 |
| Bottom Head Ring 03 (Heat # 981177/288872) [(f)] | 0.16 | 0.77 | 0.016 | 0.73 | 5 | 64 |
| Upper Shell to Intermediate Shell Circumferential Weld W06 (Heat # 25006) | 0.17[(g)] | 1.0[(g)] | 0.013[(g)] | 1.90[(g)] | 10[(h)] | 78[(h)] |
| Lower Shell to Bottom Head Ring Weld W04 (Heat # 25295)[(i)] | 0.35 | 0.11 | 0.021 | 1.47 | -40 | 113 |

Notes:

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

(b) Except for the best-estimate P and Mn weight percent values, the beltline material properties were taken from WCAP-15293, Revision 2[10]. The weight percent P and Mn values for the beltline forging materials are based on Sequoyah Unit 1 CMTR data. The weight percent P and Mn values for the beltline and surveillance weld materials were determined using Rotterdam weld certification records as well as WCAP-8233[1]

(c) 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 Cu wt. % value. The three Rotterdam test results were: 0.30, 0.25 and 0.46 % copper, as referenced from the NRC Reactor Vessel Integrity Database (RVID) and ultimately from Rotterdam Weld Certifications. The Ni wt. % value is identical to the Ni wt. % value of the surveillance weld.

(d) These copper and nickel values are best estimate values for only the surveillance weld metal and are 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[7d]), 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.

(e) 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.

(f) The chemical compositions for the extended beltline forging materials are based on Sequoyah Unit 1 certified data, except for the weight percent copper values. The maximum weight percent copper value for A508 Class 2 forging materials is conservatively applied, as described in WCAP-17539-NP, Revision 1[15].

(g) Except for the weight percent nickel, the chemical compositions were taken from a chemical analysis performed on the weld wire (heat # 25006) included in the Rotterdam weld certification records. A value of 1.0 was conservatively assumed, as described in WCAP-17539-NP, Revision 1[15].

(h) The initial $RT_{NDT}$ was determined using all available measured data for heat # 25006 as described in WCAP-17539-NP, Revision 1[15]. In absence of USE data for weld heat # 25006, weld heat # 25295 test results from the first surveillance capsule withdrawn from Sequoyah Unit 1 were used, also as described in WCAP-17539-NP, Revision 1[15].

(i) The Lower Shell to Bottom Head Ring Weld was fabricated using the same weld wire heat number and flux type as the Intermediate Shell to Lower Shell Circumferential Weld.

Table 5-4
Peak Neutron Fluence Projections on the Reactor Vessel Clad/Base Metal Interface for Sequoyah Unit 1 (x $10^{19}$ n/cm$^2$, E > 1.0 MeV)

| Reactor Vessel Material | Fluence[(a)] | |
|---|---|---|
| | 32 EFPY | 52 EFPY |
| Upper Shell Forging 06 | 0.0369 | 0.0584 |
| Intermediate Shell Forging 05 | 1.73 | 2.66 |
| Lower Shell Forging 04 | 1.73 | 2.66 |
| Bottom Head Ring 03 | 0.215 | 0.336 |
| Upper Shell to Intermediate Shell Circumferential Weld W06 | 0.0369 | 0.0584 |
| Intermediate Shell to Lower Shell Circumferential Weld W05 | 1.72 | 2.65 |
| Lower Shell to Bottom Head Ring Weld W04 | 0.215 | 0.336 |

Note:

(a) Fluence was updated per TLAA, WCAP-17539-NP, Revision 1 Tables 2-1 and 2-2[15].

Table 5-5
Sequoyah Unit 1 Calculation of the ART Values for the 1/4T Location at 52 EFPY[a]

| Material | RG 1.99 R2 Method | CF (°F) | FF | $IRT_{NDT}$[b] (°F) | $\Delta RT_{NDT}$[c] (°F) | Margin[d] (°F) | ART[e] (°F) |
|---|---|---|---|---|---|---|---|
| Reactor Vessel Beltline Materials | | | | | | | |
| Intermediate Shell Forging 05 | Position 1.1 | 115.6 | 1.1301 | 40 | 130.6 | 34.0 | 204.6 |
| Lower Shell Forging 04 | Position 1.1 | 95.0 | 1.1301 | 73 | 107.4 | 34.0 | 214.4 |
| | Position 2.1 | 109.3 | 1.1301 | 73 | 123.5 | 17.0[f] | **213.5** |
| Intermediate to Lower Shell Circumferential Weld Seam | Position 1.1 | 161.3 | 1.1291 | -40 | 182.1 | 56.0 | 198.1 |
| | Position 2.1 | 139.3 | 1.1291 | -40 | 157.3 | 28.0[f] | 145.3 |
| Reactor Vessel Extended Beltline Materials | | | | | | | |
| Upper Shell Forging 06 | Position 1.1 | 123.9 | 0.2408 | 23 | 29.8 | 29.8 | 82.7 |
| Bottom Head Ring 03 | Position 1.1 | 122.3 | 0.5722 | 5 | 70.0 | 34.0 | 109.0 |
| Upper Shell to Intermediate Shell Circumferential Weld W06 (Heat #25006) | Position 1.1 | 207.0 | 0.2408 | 10 | 49.8 | 49.8 | 109.7 |
| Lower Shell to Bottom Head Ring Weld W04 (Heat #25295) | Position 1.1 | 161.3 | 0.5722 | -40 | 92.3 | 56.0 | 108.3 |
| | Position 2.1 | 139.3 | 0.5722 | -40 | 79.7 | 28.0[f] | 67.7 |

Notes:
(a) Neutron fluence values used for all materials obtained from Table 5-4 for 52 EFPY.
(b) Initial $RT_{NDT}$ values are measured values.
(c) $\Delta RT_{NDT} = CF * FF$
(d) Margin = $2 * (\sigma_i^2 + \sigma_\Delta^2)^{1/2}$
(e) ART = Initial $RT_{NDT}$ + $\Delta RT_{NDT}$ + Margin (°F)
(f) Data deemed credible (See Reference 15), thus the reduced $\sigma_\Delta$ will be used to determine margin for the surveillance forging and surveillance weld material.

Table 5-6
Sequoyah Unit 1 Calculation of the ART Values for the 3/4T Location at 52 EFPY[(a)]

| Material | RG 1.99 R2 Method | CF (°F) | FF | $IRT_{NDT}$[(b)] (°F) | $\Delta RT_{NDT}$[(c)] (°F) | Margin[(d)] (°F) | ART[(e)] (°F) |
|---|---|---|---|---|---|---|---|
| Reactor Vessel Beltline Materials | | | | | | | |
| Intermediate Shell Forging 05 | Position 1.1 | 115.6 | 0.8481 | 40 | 98.0 | 34.0 | 172.0 |
| Lower Shell Forging 04 | Position 1.1 | 95.0 | 0.8481 | 73 | 80.6 | 34.0 | 187.6 |
| | Position 2.1 | 109.3 | 0.8481 | 73 | 92.7 | 17.0[(f)] | **182.7** |
| Intermediate to Lower Shell Circumferential Weld Seam | Position 1.1 | 161.3 | 0.8471 | -40 | 136.6 | 56.0 | 152.6 |
| | Position 2.1 | 139.3 | 0.8471 | -40 | 118.0 | 28.0[(f)] | 106.0 |
| Reactor Vessel Extended Beltline Materials | | | | | | | |
| Upper Shell Forging 06 | Position 1.1 | 123.9 | 0.1291 | 23 | 16.0 | 16.0 | 55.0 |
| Bottom Head Ring 03 | Position 1.1 | 122.3 | 0.3579 | 5 | 43.8 | 34.0 | 82.8 |
| Upper Shell to Intermediate Shell Circumferential Weld W06 (Heat #25006) | Position 1.1 | 207.0 | 0.1291 | 10 | 26.7 | 26.7 | 63.4 |
| Lower Shell to Bottom Head Ring Weld W04 (Heat #25295) | Position 1.1 | 161.3 | 0.3579 | -40 | 57.7 | 56.0 | 73.7 |
| | Position 2.1 | 139.3 | 0.3579 | -40 | 49.9 | 28.0[(f)] | 37.9 |

Notes:
(a) Neutron fluence values used for all materials obtained from Table 5-4 for 52 EFPY.
(b) Initial $RT_{NDT}$ values are measured values.
(c) $\Delta RT_{NDT} = CF * FF$
(d) Margin = $2 * (\sigma_i^2 + \sigma_\Delta^2)^{1/2}$
(e) ART = Initial $RT_{NDT}$ + $\Delta RT_{NDT}$ + Margin (°F)
(f) Data deemed credible (See Reference 15), thus the reduced $\sigma_\Delta$ will be used to determine margin for the surveillance forging and surveillance weld material.

Table 5-7
Summary of the Sequoyah Unit 1 Reactor Vessel Beltline and Extended Beltline Material ART Values

| Material | RG 1.99 R2 Method | 1/4 ART (°F) | 3/4 ART (°F) |
|---|---|---|---|
| Reactor Vessel Beltline Materials | | | |
| Intermediate Shell Forging 05 | Position 1.1 | 204.6 | 172.0 |
| Lower Shell Forging 04 | Position 1.1 | 214.4 | 187.6 |
| | Position 2.1 | **213.5** | **182.7** |
| Intermediate to Lower Shell Circumferential Weld Seam | Position 1.1 | 198.1 | 152.6 |
| | Position 2.1 | 145.3 | 106.0 |
| Reactor Vessel Extended Beltline Materials | | | |
| Upper Shell Forging 06 | Position 1.1 | 82.7 | 55.0 |
| Bottom Head Ring 03 | Position 1.1 | 109.0 | 82.8 |
| Upper Shell to Intermediate Shell Circumferential Weld W06 | Position 1.1 | 109.7 | 63.4 |
| Lower Shell to Bottom Head Ring Weld W04 | Position 1.1 | 108.3 | 73.7 |
| | Position 2.1 | 67.7 | 37.9 |

Table 5-8
Sequoyah Unit 1 Calculation of the ART Values for the Inlet and Outlet Nozzles at 52 EFPY[(a)]

| Material | Heat # | Fluence (x $10^{19}$ n/cm$^2$, E > 1.0 MeV) | $IRT_{NDT}$[(b)] (°F) | $\Delta RT_{NDT}$ (°F) | Margin[(c)] (°F) | ART[(d)] (°F) |
|---|---|---|---|---|---|---|
| Inlet Nozzle 11 | 4846 | 0.00358 | -15 | 0 | 0 | -15 |
| Inlet Nozzle 12 | 4849 | 0.00358 | -1 | 0 | 0 | -1 |
| Inlet Nozzle 13 | 4863 | 0.00358 | -22 | 0 | 0 | -22 |
| Inlet Nozzle 14 | 4865 | 0.00358 | -31 | 0 | 0 | -31 |
| Outlet Nozzle 15 | 4845 | 0.00358 | -19 | 0 | 0 | -19 |
| Outlet Nozzle 16 | 4850 | 0.00358 | -10 | 0 | 0 | -10 |
| Outlet Nozzle 17 | 4862 | 0.00358 | -24 | 0 | 0 | -24 |
| Outlet Nozzle 18 | 4864 | 0.00358 | -40 | 0 | 0 | -40 |

Notes:

(a) Neutron fluence value for the inlet nozzle to upper shell welds at 52 EFPY was used for all nozzle materials per MCOE-LTR-16-12-NP[16].

(b) Initial $RT_{NDT}$ values are measured values.

(c) Margin = 2 *$(\sigma_i^2 + \sigma_\Delta^2)^{1/2}$ = 0 for materials with measured $IRT_{NDT}$ values and fluence values less than 1 x $10^{17}$ n/cm$^2$ (E > 1.0 MeV).

(d) ART = Initial $RT_{NDT}$ + $\Delta RT_{NDT}$ + Margin (°F); $\Delta RT_{NDT}$ = 0 for fluence values less than 1 x $10^{17}$ n/cm$^2$ (E > 1.0 MeV) per Reference 17.

Table 5-9
Summary of the Sequoyah Unit 1 Nozzle Material ART Values at 52 EFPY

| Nozzle Material | Limiting ART Value (°F) |
|---|---|
| Inlet Nozzle 12 Heat # 4849 | -1 |
| Outlet Nozzle 16 Heat # 4850 | -10 |

Table 5-10
$RT_{PTS}$ Calculations for Sequoyah Unit 1 Beltline and Extended Beltline Materials at 52 EFPY[(a)]

| Material | Fluence (x $10^{19}$ n/cm$^2$, E>1.0 MeV) | FF | CF (°F) | $\Delta RT_{PTS}$[(b)] (°F) | Margin (°F) | $RT_{NDT(U)}$[(c)] (°F) | $RT_{PTS}$[(d)] (°F) |
|---|---|---|---|---|---|---|---|
| Reactor Vessel Beltline Materials | | | | | | | |
| Intermediate Shell Forging 05 | 2.66 | 1.2616 | 115.6 | 145.8 | 34.0 | 40 | 219.8 |
| Lower Shell Forging 04 | 2.66 | 1.2616 | 95.0 | 119.8 | 34.0 | 73 | 226.8 |
| Lower Shell Forging 04 (Using S/C Data) | 2.66 | 1.2616 | 109.3 | 137.9 | 17.0[(e)] | 73 | **227.9** |
| Circumferential Weld Metal | 2.65 | 1.2607 | 161.3 | 203.3 | 56.0 | -40 | 219.3 |
| Circumferential Weld Metal (Using S/C Data) | 2.65 | 1.2607 | 139.3 | 175.6 | 28.0[(e)] | -40 | 163.6 |
| Reactor Vessel Extended Beltline Materials | | | | | | | |
| Upper Shell Forging 06 | 0.0584 | 0.3180 | 123.9 | 39.4 | 34.0 | 23 | 96.4 |
| Bottom Head Ring 03 | 0.336 | 0.6997 | 122.3 | 85.6 | 34.0 | 5 | 124.6 |
| Upper Shell to Intermediate Shell Circumferential Weld W06 | 0.0584 | 0.3180 | 207.0 | 65.8 | 56.0 | 10 | 131.8 |
| Lower Shell to Bottom Head Ring Weld W04 | 0.336 | 0.6997 | 161.3 | 112.9 | 56.0 | -40 | 128.9 |
| Lower Shell to Bottom Head Ring Weld W04 (Using S/C Data) | 0.336 | 0.6997 | 139.3 | 97.5 | 28.0[(e)] | -40 | 85.5 |

Notes:

(a) Neutron fluence values used for all materials obtained from Table 5-4 for 52 EFPY.

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

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

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

(e) Data deemed credible (See Reference 15), thus the reduced $\sigma_{\Delta}$ will be used to determine margin for the surveillance forging and surveillance weld material.

## 6.0 References


1. WCAP-8233, *Tennessee Valley Authority Sequoyah Unit No. 1 Reactor Vessel Radiation Surveillance Program*, S. E. Yanichko, et. al., December 1973.

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

3. ASTM E23 *Standard Test Method Notched Bar Impact Testing of Metallic Materials*, in ASTM Standards, 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.

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

7b. WCAP-13333, *Analysis of Capsule X from the Tennessee Valley Authority Sequoyah Unit 1 Reactor Vessel Radiation Surveillance Program,* M.A. Ramirez, S. L. Anderson, L. Albertin, June 1992.

7c. SwRI Project 06-8851, *Reactor Vessel Material Surveillance Program for Sequoyah Unit No. 1: Analysis of Capsule U*, P. K. Nair, et al., October 1986.

7d. 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.

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-15293, Revision 2, *Sequoyah Unit 1 Heatup and Cooldown Limit Curves for Normal Operation and PTLR Support Documentation*, J.H. Ledger, July 2003.

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

12. Calculation SQN-IC-014, *Demonstrated Accuracy Calculation for Cold Overpressure Protection System*.

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

14. WCAP-15984-P, Revision 01, *Reactor Vessel Closure Head/Vessel Flange Requirements Evaluation for Sequoyah Units 1 and 2*, W. Bamford, et.al., April 2003.

15. WCAP-17539-NP, Revision 1, *Sequoyah Units 1 and 2 Time-Limited Aging Analysis on Reactor Vessel Integrity*, E.J. Long, May 2015.

16. MCOE-LTR-16-12-NP, Revision 0, *Pressure-Temperature Limit Curve Development for Extended Plant Operation for Sequoyah Units 1 and 2 Reactor Vessel Nozzles*, E.M. Ruminski, A.M. Carolan, August 2016.

17. U.S. NRC Regulatory Issue Summary (RIS) 2014-11, *Information on Licensing Applications for Fracture Toughness Requirements for Ferritic Reactor Coolant Pressure Boundary Components*, dated October 14, 2014.

18. LTR-SCS-16-25, Revision 0, *Cold Overpressure Mitigation System (COMS) Setpoint Analysis for Sequoyah Units 1 and 2 PORV Replacement*, R.A. Freund, B.D. Jaskiewicz, November 2016.

19. LTR-SCS-17-43, Revision 0, *Sequoyah LTOPS Setpoint Implementation for PORV Replacement Program Delay*, R.A. Freund, September 2017.