March 31, 1992

Docket Nos. 50-327
and 50-328

Senior Vice President, Nuclear Power
Tennessee Valley Authority
3B Lookout Place
1101 Market Street
Chattanooga, Tennessee 37402-2801

Dear Sir:

SUBJECT: ISSUANCE OF AMENDMENTS (TAC NOS. M79900 AND 79901) (TS 90-01)

The Commission has issued the enclosed Amendment No. 158 to Facility Operating License No. DPR-77 and Amendment No. 148 to Facility Operating License No. DPR-79 for the Sequoyah Nuclear Plant, Units 1 and 2, respectively. These amendments are in response to your application dated March 1, 1991 and superseded by letter dated September 6, 1991.

The amendment incorporates new reactor coolant system pressure-temperature limit curves that are applicable up to 16 effective full power years.

A copy of the Safety Evaluation is also enclosed. Notice of Issuance will be included in the Commission's biweekly Federal Register notice.

Sincerely,

Original signed by

David E. LaBarge, Senior Project Manager
Project Directorate II-4
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

Enclosures:
1. Amendment No. 158 to
   License No. DPR-77
2. Amendment No. 148 to
   License No. DPR-79
3. Safety Evaluation

cc w/enclosures:
See next page

NRC FILE CENTER COPY

*SEE PREVIOUS CONCURRENCE

| OFC | PDII-4/LA | PDII-4/PM | OGC* | PDII-4/D | |
|---|---|---|---|---|---|
| NAME | MSanders | DELaBarge | | FHebdon | |
| DATE | 2/10/92<br>3/31/92 | 3/31/92 | 2/12/92 | 3/31/92 | |

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1/1

9204090257 920331
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P PDR

CP-1

AMENDMENT NO. 158 FOR SEQUOYAH UNIT NO. 1 - DOCKET NO. 50-327 and
AMENDMENT NO. 148 FOR SEQUOYAH UNIT NO. 2 - DOCKET NO. 50-328
DATED: March 31, 1992

| | |
|---|---|
| Docket File | |
| NRC PDR | |
| Local PDR | |
| SQN Reading File | |
| S. Varga | 14-E-4 |
| F. Hebdon | |
| M. Sanders | |
| D. E. LaBarge | |
| B. Wilson | RII |
| W. Little | RII |
| OGC | 15-B-13 |
| D. Hagan | MNBB-3302 |
| E. Jordan | MNBB-3302 |
| G. Hill | P1-130 (8) |
| Wanda Jones | MNBB-7103 |
| J. Calvo | 14-E-4 |
| ACRS(10) | |
| GPA/PA | 2-G-5 |
| OC/LFMB | MNBB-9112 |
| S. Sheng | |

cc: Plant Service List

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

TENNESSEE VALLEY AUTHORITY

DOCKET NO. 50-327

SEQUOYAH NUCLEAR PLANT, UNIT 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 158
License No. DPR-77

1. The Nuclear Regulatory Commission (the Commission) has found that:

   A. The application for amendment by Tennessee Valley Authority (the licensee) dated March 1, 1991, and superseded September 6, 1991, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I;

   B. The facility will operate in conformity with the application, the provisions of the Act, and the rules and regulations of the Commission;

   C. There is reasonable assurance (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations;

   D. The issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public; and

   E. The issuance of this amendment is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied.

9204090261 920331
PDR ADOCK 05000327
P PDR

2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment and paragraph 2.C.(2) of Facility Operating License No. DPR-77 is hereby amended to read as follows:

   (2) Technical Specifications

   The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 158, are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

3. This license amendment is effective as of its date of issuance, to be implemented within 30 days.

FOR THE NUCLEAR REGULATORY COMMISSION

Frederick J. Hebdon, Director
Project Directorate II-4
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

Attachment:
Changes to the Technical
  Specifications

Date of Issuance: March 31, 1992

ATTACHMENT TO LICENSE AMENDMENT NO. 158

FACILITY OPERATING LICENSE NO. DPR-77

DOCKET NO. 50-327

Revise the Appendix A Technical Specifications by removing the pages identified below and inserting the enclosed pages. The revised pages are identified by the captioned amendment number and contain marginal lines indicating the area of change. Overleaf pages* are provided to maintain document completeness.

| REMOVE | INSERT |
|---|---|
| 3/4 4-24 | 3/4 4-24 |
| 3/4 4-25 | 3/4 4-25 |
| B3/4 4-6 | B3/4 4-6 |
| B3/4 4-7 | B3/4 4-7 |
| B3/4 4-8 | B3/4 4-8 |
| B3/4 4-9 | B3/4 4-9 |
| B3/4 4-10* | B3/4 4-10* |
| B3/4 4-13 | B3/4 4-13 |

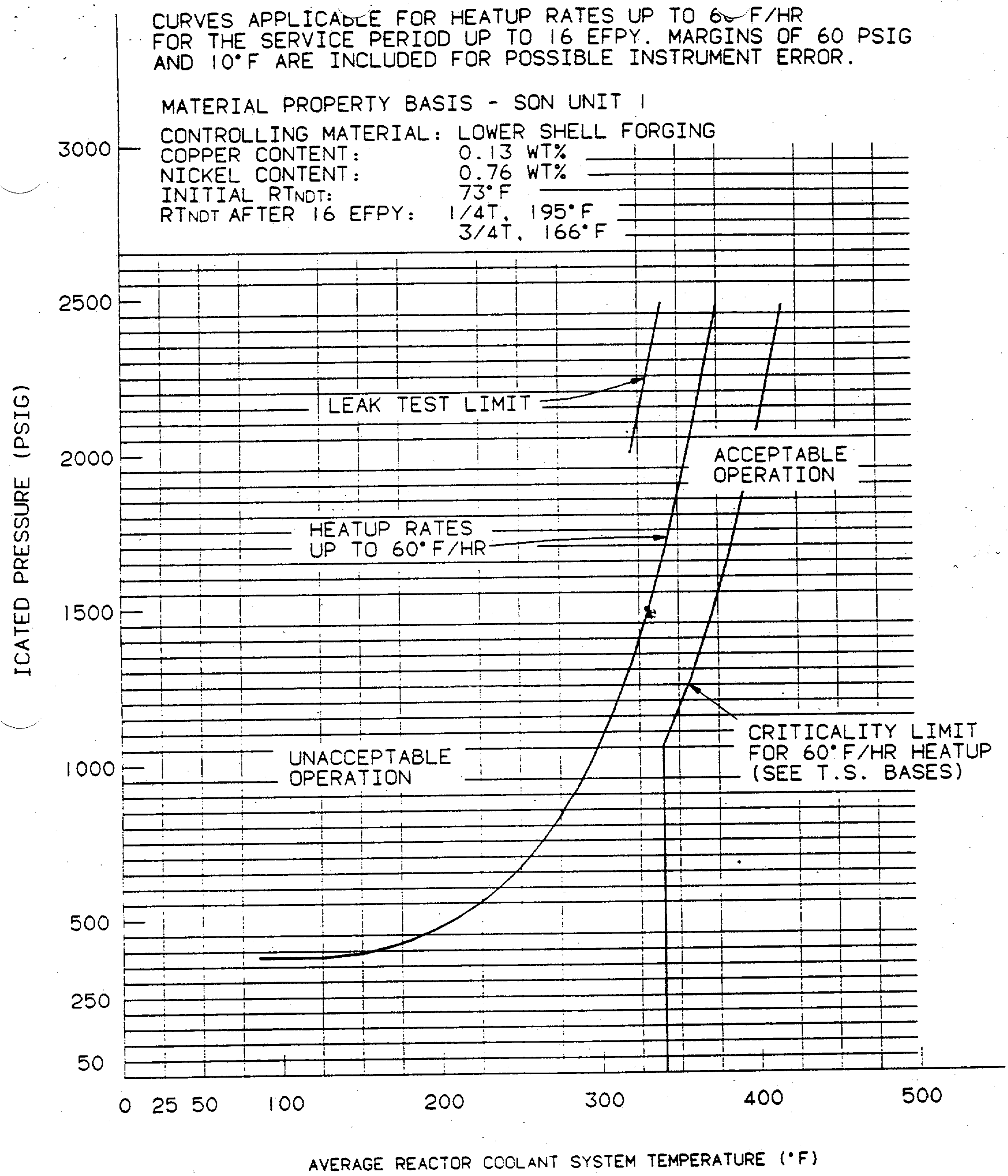


FIGURE 3.4-2 SEQUOYAH UNIT 1 REACTOR COOLANT SYSTEM HEATUP LIMITATIONS APPLICABLE UP TO 16 EFPY

Amendment No. 158

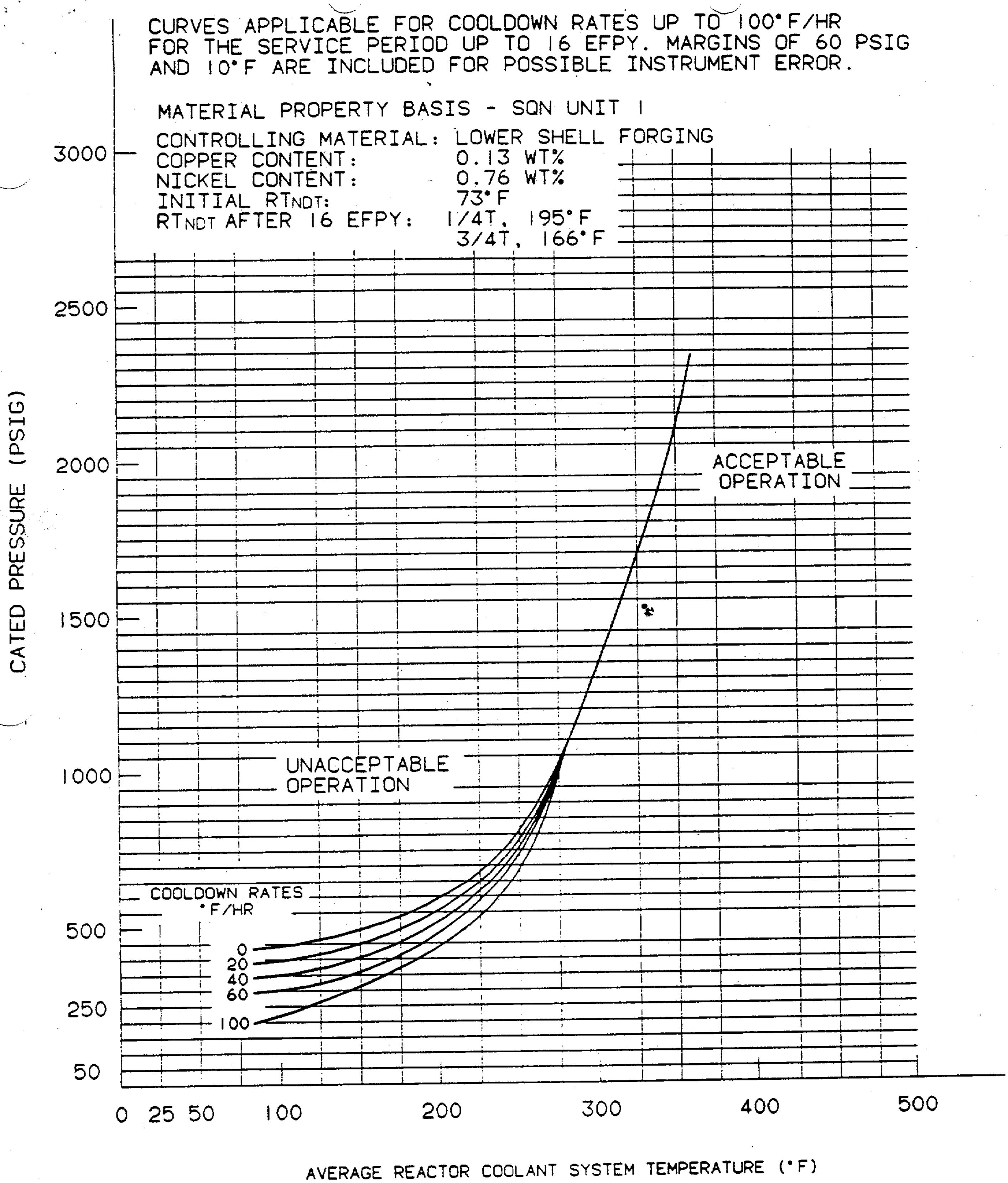


FIGURE 3.4-3 SEQUOYAH UNIT 1 REACTOR COOLANT SYSTEM COOLDOWN LIMITATIONS APPICICABLE UP TO 16 EFPY

## 3/4.4.9 PRESSURE/TEMPERATURE LIMITS

The temperature and pressure changes during heatup and cooldown are limited to be consistent with the requirements given in the ASME Boiler and Pressure Vessel Code, Section III, Appendix G.

1) The reactor coolant temperature and pressure and system heatup and cooldown rates (with the exception of the pressurizer) shall be limited in accordance with Figures 3.4-2 and 3.4-3 for the first full-power service period.

   a) Allowable combinations of pressure and temperature for specific temperature change rates are below and to the right of the limit lines shown. Limit lines for cooldown rates between those presented may be obtained by interpolation.

   b) Figures 3.4-2 and 3.4-3 define limits to assure prevention of non-ductile failure only. For normal operation, other inherent plant characteristics, e.g., pump heat addition and pressurizer heater capacity, may limit the heatup and cooldown rates that can be achieved over certain pressure-temperature ranges.

2) These limit lines shall be calculated periodically using methods provided below.

3) The secondary side of the steam generator must not be pressurized above 200 psig if the temperature of the steam generator is below 70°F.

4) The pressurizer heatup and cooldown rates shall not exceed 100°F/hr and 200°F/hr respectively. The spray shall not be used if the temperature difference between the pressurizer and the spray fluid is greater than 560°F.

5) System preservice hydrotests and in-service leak and hydrotests shall be performed at pressures in accordance with the requirements of ASME Boiler and Pressure Vessel Code, Section XI.

10 CFR 50, Appendix G, addressed metal temperature of the closure head flange and vessel regions. Appendix G states that the minimum metal temperature of the closure flange region should be at least 120 degrees Fahrenheit (F) higher than the limiting $RT_{NDT}$ for this region when the pressure exceeds 20 percent of the preservice hydrostatic test pressure (561 pounds per square inch gauge (psig) for Westinghouse Electric Corporation plants). For SQN, Unit 1, the minimum temperature of the closure flange and vessel flange regions is 90 degrees F since the limiting initial $RT_{NDT}$ for the closure head flange is -40 degrees F (see Table B 3/4.4-1). These numbers (561 psig and 90 degrees F) include a margin for instrumentation error of 10 degrees F and 60 psig. The SQN Unit 1 heat up and cooldown curves shown in Figures 3.4-2 and 3.4-3 are not impacted by this regulation.

The fracture toughness properties of the ferritic materials in the reactor vessel are determined in accordance with the NRC Standard Review Plan, and ASTM E185-82, and in accordance with additional reactor vessel requirements. These properties are then evaluated in accordance with Appendix G to 10 CFR 50 and Appendix G of the 1986 ASME Boiler and Pressure Vessel Code, Section III, Division 1 and the calculation methods described in WCAP-7924-A, "Basis for Heatup and Cooldown Limit Curves, April 1975."

Heatup and cooldown limit curves are calculated using the most limiting value of the nil-ductility reference temperature, $RT_{NDT}$, at the end of 16 effective full power years of service life. The 16 EFPY service life period is chosen such that the limiting $RT_{NDT}$ at the 1/4T location in the core region is greater than the $RT_{NDT}$ of the limiting unirradiated material. The selection of such a limiting $RT_{NDT}$ assures that all components in the Reactor Coolant System will be operated conservatively in accordance with applicable Code requirements.

The reactor vessel materials have been tested to determine their initial $RT_{NDT}$; the results of these tests are shown in Table B 3/4.4-1. Reactor operation and resultant fast neutron (E greater than 1 MEV) irradiation can cause an increase in the $RT_{NDT}$. Therefore, an adjusted reference temperature, based upon the fluence of the material in question, has been predicted using Regulatory Guide 1.99, Revision 2 and a peak surface fluence of $1.94 \times 10^{19}$ n/cm$^2$ for 16 effective full power years (Reference WCAP 12970, "Heatup and Cooldown Limit Curves for Normal Operation," June 1991). The heatup and cooldown limit curves of Figures 3.4-2 and 3.4-3 include predicted adjustments for this shift in $RT_{NDT}$ at the end of 16 EFPY, as well as adjustments for possible errors in the pressure and temperature sensing instruments.

Values of delta $RT_{NDT}$ determined in this manner may be used until the results from the material surveillance program, evaluated according to ASTM E185, are available. The first capsule was removed at the end of the first core cycle. Successive capsules will be removed in accordance with the requirements of ASTM E185-82 and 10 CFR 50, Appendix H. The heatup and cooldown curves and the low temperature overpressure protection set-points must be recalculated when the delta $RT_{NDT}$ determined from the surveillance capsule exceeds the calculated delta $RT_{NDT}$ for the equivalent capsule radiation exposure.

This page intentionally deleted.

TABLE B 3/4.4-1

SEQUOYAH NO. 1 REACTOR VESSEL TOUGHNESS DATA

| COMPONENT | Heat No. | Material Grade | Cu (%) | Ni (%) | NDT (°F) | MINIMUM 50 ft-lb/35 mil temp. Temp. (°F) | | $RT_{NDT}$ (°F) | AVERAGE UPPER SHELF ENERGY (ft-lb) | |
|---|---|---|---|---|---|---|---|---|---|---|
| | | | | | | PMWD1 | NMWD2 | | PMWD1 | NMWD2 |
| Clos. Hd. Dome | 52841-1 | A533B,Cl.1 | - | - | -40 | +14 | +34 | -26 | 104[a] | - |
| Clos. Hd. Ring | (D75600) | A508,Cl.2 | - | - | + 5 | +36 | +56* | + 5 | 125[a] | - |
| Hd. Flange | 4842 | A508,Cl.2 | - | - | -40 | -24 | -4* | -40 | 131[a] | - |
| Vessel Flange | 4866 | A508,Cl.2 | - | - | -49 | -47 | -27 | -49 | 158[a] | - |
| Inlet Nozzle | 4846 | A508,Cl.2 | - | - | -58 | +25 | +45 | -15 | 94.5[a] | - |
| Inlet Nozzle | 4949 | A508,Cl.2 | - | - | -40 | +39 | +59* | -1 | 93[a] | - |
| Inlet Nozzle | 4863 | A508,Cl.2 | - | - | -22 | +16 | +36* | -22 | 118[a] | - |
| Inlet Nozzle | 4865 | A508,Cl.2 | - | - | -67 | + 9 | +29* | -31 | 94[a] | - |
| Outlet Nozzle | 4845 | A508,Cl.2 | - | - | -49 | +21 | +41* | -19 | 94[a] | - |
| Outlet Nozzle | 4850 | A508,Cl.2 | - | - | -58 | +30 | +50* | -10 | 79.5[a] | - |
| Outlet Nozzle | 4862 | A508,Cl.2 | - | - | -58 | +16 | +36* | -24 | 103[a] | - |
| Outlet Nozzle | 4864 | A508,Cl.2 | - | - | -49 | 0 | +20 | -40 | 126[a] | - |
| Upper Shell | 4841 | A508,Cl.2 | - | - | -40 | +43 | +83 | +23 | 83[a] | 113[b] |
| Inter. Shell | 4829 | A508,Cl.2 | 0.15 | 0.86 | - 4 | +10 | +100 | +40 | 116 | 73[b,c] |
| Lower Shell | 4836 | A508,Cl.2 | 0.13 | 0.76 | + 5 | +28 | +133 | +73 | 109 | 70[b] |
| Trans. Ring | 4879 | A508,Cl.2 | - | - | + 5 | +27 | +47* | + 5 | 98[a] | - |
| Bot. Hd. Ring | 52703/2-1 | A533B,Cl.1 | - | - | -31 | +23 | +43* | -17 | 104[a] | - |
| Bot. Hd. Ring | 52703/2-2 | A533B,Cl.1 | - | - | -13 | +36 | +56* | -4 | 63[a] | |
| Bot. Hd. Ring | 52704/2 | A533B,Cl.1 | - | - | -49 | -24 | -4* | -49 | 114[a] | |
| Bot. Hd. Ring | 52703/2-2 | A533B,Cl.1 | - | - | -31 | +43 | +63* | +3 | 86[a] | |
| Bot. Hd. Ring | 52704/2 | A533B,Cl.1 | - | - | -58 | -13 | +4 | -53 | 120[a] | |
| Bot. Hd. | 52704-11 | A533B,Cl.1 | - | - | -58 | -47 | -27* | -58 | 139[a] | |
| Weld | - | Weld | 0.33 | 0.17 | -40 | - | -4 | -40 | - | 116[b] |
| HAZ | - | WELD | - | - | -22 | | +41 | -19 | - | 86[b] |

1-Parallel to Major Working Direction

2-Normal to Major Working Direction

*Estimate based on USAEC Regulatory Standard Review Plan, Section 5.3.2 MTEB

a-%Shear not reported

b-Minimum upper shelf energies

c-Minimum upper shelf energy decreased to 51 at a test temperature of 300°F. This anomaly will be reevaluated when the results of Generic task A-11 are available.

Allowable pressure -temperature relationships for various heatup and cooldown rates are calculated using methods derived from Appendix G in Section III of the ASME Boiler and Pressure Vessel Code as required by Appendix G to 10 CFR Part 50 and these methods are discussed in detail in WCAP-7924-A.

The general method for calculating heatup and cooldown limit curves is based upon the principles of the linear elastic fracture mechanics (LEFM) technology. In the calculation procedures a semi-elliptical surface defect with a depth of one quarter of the wall thickness, T, and a length of 3/2T is assumed to exist at the inside of the vessel wall as well as at the outside of the vessel wall. The dimensions of this postulated crack, referred to in Appendix G of ASME III as the reference flaw, amply exceed the current capabilities of inservice inspection techniques. Therefore, the reactor operation limit curves developed for this reference crack are conservative and provide sufficient safety margins for protection against non-ductile failure. To assure that the radiation embrittlement effects are accounted for in the calculation of the limit curves, the most limiting value of the nil ductility reference temperature, $RT_{NDT}$, is used and this includes the radiation induced shift, delta $RT_{NDT}$, corresponding to the end of the period for which heatup and cooldown curves are generated.

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

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

thermal stresses and different $K_{IR}$'s for steady-state and finite heatup rates 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 assure 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 pressure-temperature limitations for the case in which a 1/4T deep outside surface flaw is assumed. Unlike the situation at the vessel inside surface, the thermal gradients established at the outside surface during heatup produce stresses which are tensile in nature and thus tend to reinforce any pressure stresses present. These thermal stresses, of course, are dependent on both the rate of heatup and the time (or coolant temperature) along the heatup ramp. Furthermore, since the thermal stresses, at the outside are tensile and increase with increasing heatup rate, a lower bound curve cannot be defined. Rather, each heatup rate of interest 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 as follows. A composite curve is constructed 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 such that 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.

The leak test limit curve shown in Figure 3.4-2 represents the minimum temperature requirements at the leak test pressure specified by applicable codes. The leak test limit curve was determined by methods of Branch Technical Position MTEB 5-2 and 10 CFR 50, Appendix G.

The criticality limit curve shown in Figure 3.4-2 specifies pressure-temperature limits for core operation to provide additional margin during actual power production. The pressure-temperature limits for core operation (except for low power physics tests) require the reactor vessel to be at a temperature equal to or higher than the minimum temperature required for the in-service hydrostatic test, and at least 40 degrees F higher than the minimum pressure-temperature curve for heatup and cooldown. The maximum temperature for the in-service hydrostatic test for the SQN Unit 1 reactor vessel is 327 degrees F. A vertical line at 327 degrees F on the pressure-temperature curve, intersecting a curve 40 degrees F higher than the pressure-temperature limit curve, constitutes the limit for core operation for the reactor vessel.

Finally, the composite curves for the heatup rate data and the cooldown rate data are adjusted for possible errors in the pressure and temperature sensing instruments by the values indicated on the respective curves.

Although the pressurizer operates in temperature ranges above those for which there is reason for concern of non-ductile failure, operating limits are provided to assure compatibility of operation with the fatigue analysis performed in accordance with the ASME Code requirements.

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

TENNESSEE VALLEY AUTHORITY

DOCKET NO. 50-328

SEQUOYAH NUCLEAR PLANT, UNIT 2

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 148
License No. DPR-79

1. The Nuclear Regulatory Commission (the Commission) has found that:

   A. The application for amendment by Tennessee Valley Authority (the licensee) dated March 1, 1991, and superseded September 6, 1991 complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I;

   B. The facility will operate in conformity with the application, the provisions of the Act, and the rules and regulations of the Commission;

   C. There is reasonable assurance (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations;

   D. The issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public; and

   E. The issuance of this amendment is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied.

2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment and paragraph 2.C.(2) of Facility Operating License No. DPR-79 is hereby amended to read as follows:

   (2) Technical Specifications

   The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 148, are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

3. This license amendment is effective as of its date of issuance, to be implemented within 30 days.

FOR THE NUCLEAR REGULATORY COMMISSION

Frederick J. Hebdon, Director
Project Directorate II-4
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

Attachment:
Changes to the Technical
  Specifications

Date of Issuance: March 31, 1992

ATTACHMENT TO LICENSE AMENDMENT NO. 148

FACILITY OPERATING LICENSE NO. DPR-79

DOCKET NO. 50-328

Revise the Appendix A Technical Specifications by removing the pages identified below and inserting the enclosed pages. The revised pages are identified by the captioned amendment number and contain marginal lines indicating the area of change. Overleaf pages* are provided to maintain document completeness.

| REMOVE | INSERT |
|---|---|
| 3/4 4-30 | 3/4 4-30 |
| 3/4 4-31 | 3/4 4-31 |
| B3/4 4-7 | B3/4 4-7 |
| B3/4 4-8 | B3/4 4-8 |
| B3/4 4-9 | B3/4 4-9 |
| B3/4 4-10 | B3/4 4-10 |
| B3/4 4-13* | B3/4 4-13* |
| B3/4 4-14 | B3/4 4-14 |

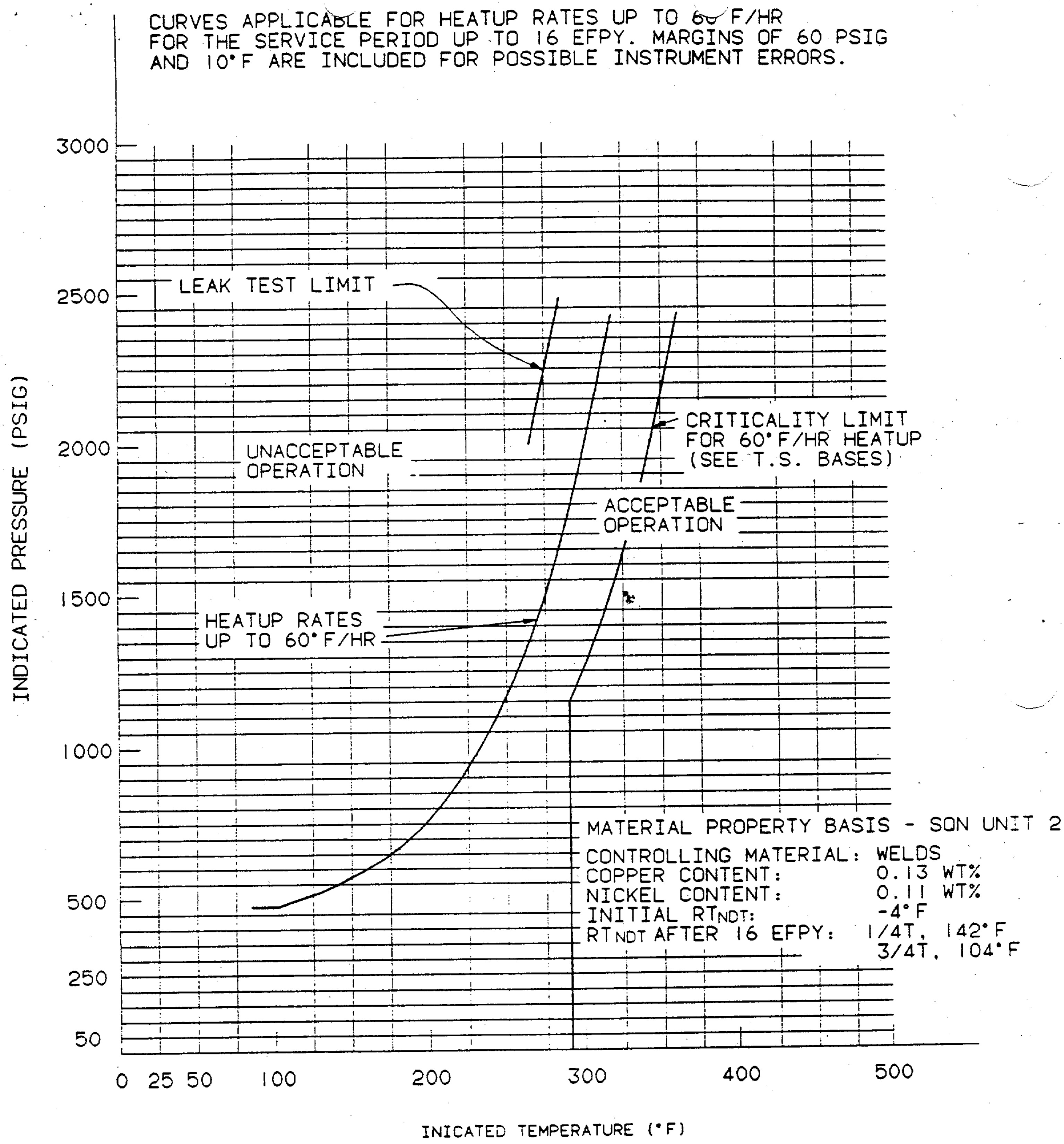


FIGURE 3.4-2 SEQUOYAH UNIT 2 REACTOR COOLANT SYSTEM HEATUP LIMITATIONS
APPLICABLE UP TO 16 EFPY

Amendment No. 148

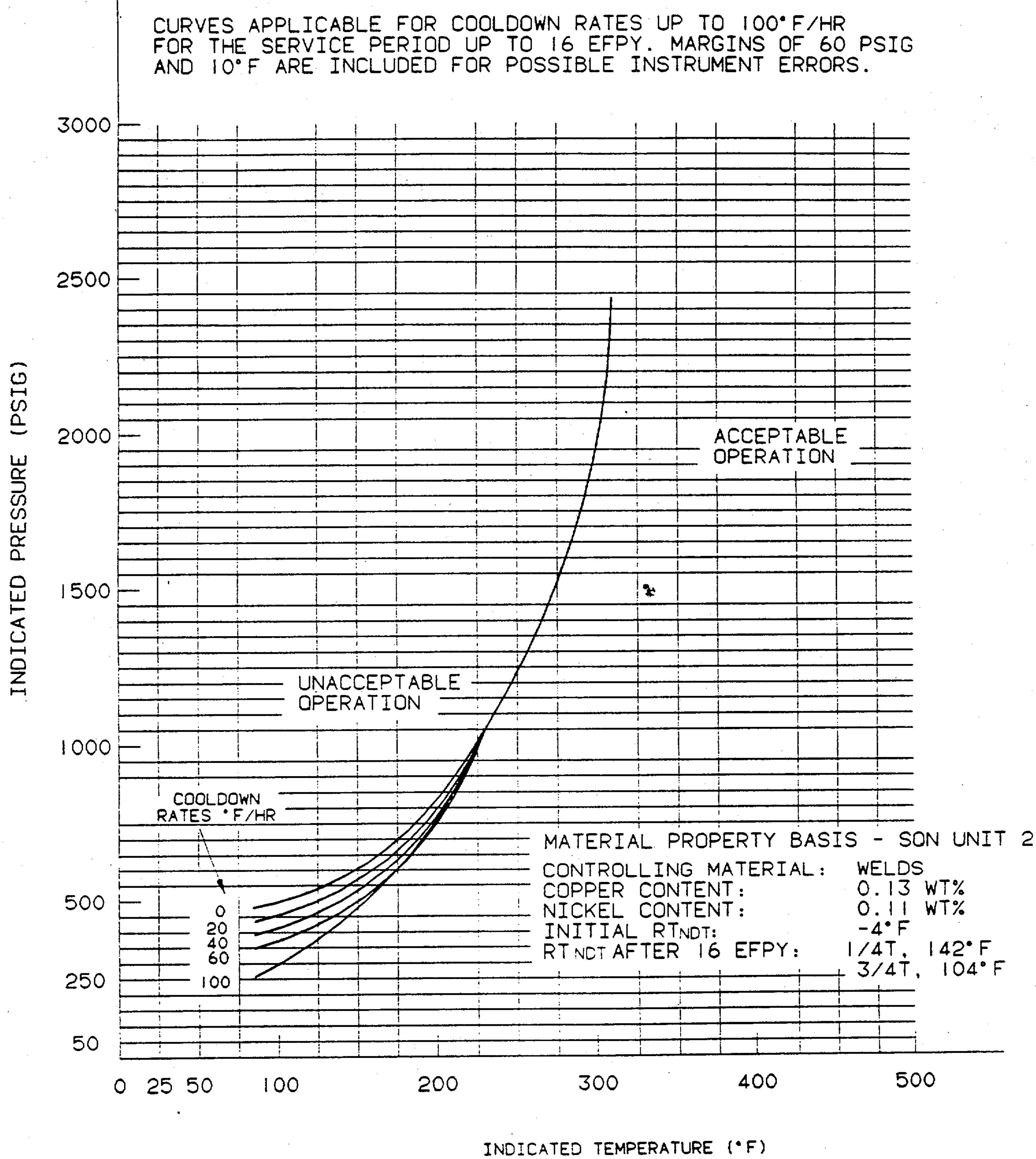


FIGURE 3.4-3 SEQUOYAH UNIT 2 REACTOR COOLANT SYSTEM COOLDOWN LIMITATIONS APPICICABLE UP TO 16 EFPY

Amendment No. 148

PRESSURE/TEMPERATURE LIMITS (Continued)

5) System preservice hydrotests and in-service leak and hydrotests shall be performed at pressures in accordance with the requirements of ASME Boiler and Pressure Vessel Code, Section XI.

10 CFR 50, Appendix G, addresses metal temperature of the closure head flange and vessel regions. Appendix G states that the minimum metal temperature of the closure flange region should be at least 120 degrees Fahrenheit (F) higher than the limiting $RT_{NDT}$ for this region when the pressure exceeds 20 percent of the preservice hydrostatic test pressure (561 pounds per square inch gauge (psig) for Westinghouse Electric Corporation plants). For SQN, Unit 2, the minimum temperature of the closure flange and vessel flange regions is 117 degrees F since the limiting initial $RT_{NDT}$ for the closure head flange is -13 degrees F (see Table B 3/4.4-1). These numbers (561 psig and 117 degrees F) include a margin for instrumentation error of 10 degrees F and 60 psig. The SQN Unit 2 heat up and cooldown curves shown in Figures 3.4-2 and 3.4-3 are not impacted by this regulation.

The fracture toughness properties of the ferritic materials in the reactor vessel are determined in accordance with the NRC Standard Review Plan, ASTM E185-82, and in accordance with additional reactor vessel requirements. These properties are then evaluated in accordance with Appendix G to 10 CFR 50 and Appendix G of the 1986 ASME Boiler and Pressure Vessel Code, Section III, Division 1 and the calculation methods described in WCAP-7924-A, "Basis for Heatup and Cooldown Limit Curves, April 1975."

Heatup and cooldown limit curves are calculated using the most limiting value of the nil-ductility reference temperature, $RT_{NDT}$, at the end of 16 effective full power years of service life. The 16 EFPY service life period is chosen such that the limiting $RT_{NDT}$ at the 1/4T location in the core region is greater than the $RT_{NDT}$ of the limiting unirradiated material. The selection of such a limiting $RT_{NDT}$ assures that all components in the Reactor Coolant System will be operated conservatively in accordance with applicable Code requirements.

The reactor vessel materials have been tested to determine their initial $RT_{NDT}$; the results of these tests are shown in Table B 3/4.4-1. Reactor operation and resultant fast neutron (E greater than 1 MEV) irradiation can cause an increase in the $RT_{NDT}$. Therefore, an adjusted reference temperature, based upon the fluence of the material in question, has been predicted using Regulatory Guide 1.99, Revision 2 and a peak surface fluence of $0.864 \times 10^{19}$ n/cm² for 16 effective full power years (Reference WCAP 12971, "Heatup and Cooldown Limit Curves for Normal Operation," June 1991. The heatup and cooldown limit curves of Figures 3.4-2 and 3.4-3 include predicted adjustments for this shift in $RT_{NDT}$ at the end of 16 EFPY, as well as adjustments for possible errors in the pressure and temperature sensing instruments.

PRESSURE/TEMPERATURE LIMITS (Continued)

Values of $\Delta RT_{NDT}$ determined in this manner may be used until the results from the material surveillance program, evaluated according to ASTM E185, are available. The first capsule was removed at the end of the first core cycle. Successive capsules will be removed in accordance with the requirements of ASTM E185-82 and 10 CFR 50, Appendix H. The heatup and cooldown curves and the low temperature overpressure protection setpoints must be recalculated when the $\Delta RT_{NDT}$ determined from the surveillance capsule exceeds the calculated $\Delta RT_{NDT}$ for the equivalent capsule radiation exposure.

Allowable pressure-temperature relationships for various heatup and cooldown rates are calculated using methods derived from Appendix G in Section III of the ASME Boiler and Pressure Vessel Code as required by Appendix G to 10 CFR Part 50 and these methods are discussed in detail in WCAP-7924-A.

The general method for calculating heatup and cooldown limit curves is based upon the principles of the linear elastic fracture mechanics (LEFM) technology. In the calculation procedures a semi-elliptical surface defect with a depth of one-quarter of the wall thickness, T, and a length of 3/2T is assumed to exist at the inside of the vessel wall as well as at the outside of the vessel wall. The dimensions of this postulated crack, referred to in Appendix G of ASME III as the reference flaw, amply exceed the current capabilities of inservice inspection techniques. Therefore, the reactor operation limit curves developed for this reference crack are conservative and provide sufficient safety margins for protection against non-ductile failure. To assure that the radiation embrittlement effects are accounted for in the calculation of the limit curves, the most limiting value of the nil ductility reference temperature, $RT_{NDT}$, is used and this includes the radiation induced shift, $\Delta RT_{NDT}$, corresponding to the end of the period for which heatup and cooldown curves are generated.

TABLE B 3/4.4-1

SEQUOYAH-UNIT 2 REACTOR VESSEL TOUGHNESS DATA

| COMPONENT | HEAT NO. | MATERIAL GRADE | CU % | Ni % | NDTT °F | MINIMUM 50 FT-LB/35 MIL TEMP °F PMWD[1] | NMWD[2] | $RT_{NDT}$ °F | AVERAGE UPPER SHELF FTLB PMWD[1] | NMWD[2] |
|---|---|---|---|---|---|---|---|---|---|---|
| CL Hd. Dome | 52899-1 | A533BCL1 | - | - | -13 | 28 | 48* | -12 | 75[a] | |
| CL Hd. Ring | - | A508CL2 | - | - | 5 | 34 | 54* | 5 | 125.5[a] | |
| Hd Flange | 4890 | A508CL2 | - | - | -13 | <-67* | <-67 | -13 | 141.[a] | |
| Vessel Flange | 4832 | A508CL2 | - | - | -22 | -47 | -27 | -22 | 155.5[a] | |
| Inlet Nozzle | 4868 | A508CL2 | - | - | -22 | 41 | 61* | 1 | 79[a] | |
| Inlet Nozzle | 4872 | A508CL2 | - | - | -22 | 12 | 32* | -22 | 108[a] | |
| Inlet Nozzle | 4877 | A508CL2 | - | - | -31 | 1 | 21* | -31 | 113[a] | |
| Inlet Nozzle | 4886 | A508CL2 | - | - | -31 | -52 | -32* | 28 | 138[a] | |
| Outlet Nozzle | 4867 | A508CL2 | - | - | -31 | 19 | 39* | -21 | 85[a] | |
| Outlet Nozzle | 4873 | A508CL2 | - | - | -22 | 21 | 41* | -19 | 76[a] | |
| Outlet Nozzle | 4878 | A508CL2 | - | - | -40 | -6 | 14* | -40 | 105[a] | |
| Outlet Nozzle | 4887 | A508CL2 | - | - | -22 | -11 | 9* | -22 | 143.5[a] | |
| Upper Shell | 4885 | A508CL2 | - | - | 5 | 25 | 45* | 5 | 104[a] | |
| Inter Shell | 4853 | A508CL2 | 0.13 | 0.74 | -22 | 19 | 70 | 10 | 138 | 93 |
| Lower Shell | 4994 | A508CL2 | 0.14 | 0.76 | -40 | 8 | 38 | -22 | 140.5 | 100 |
| Trans. Ring | 4879 | A508CL2 | - | - | 5 | 27 | 47* | 5 | 98[a] | |
| Bot. Hd. Rim | 52835-1B | A533BCL1 | - | - | -4 | 48 | 68* | 8 | 81[a] | |
| Bot. Hd. Rim | 52835-1B | A533BCL1 | - | - | -22 | 25 | 45* | -15 | 81[a] | |
| Bot. Hd. Rim | 52899-2 | A533BCL1 | - | - | -13 | 39 | 59* | -1 | 62[a] | |
| Bot. Hd. | 5297-1 | A533BCL1 | - | - | -31 | 14 | 34* | -26 | 99.5[a] | |
| Weld | - | Weld | 0.13 | 0.11 | -4 | - | 14 | -4 | - | 101 |
| HAZ | - | HAZ | - | - | -13 | - | 17 | -13 | - | 120 |

[1] Paralled to Major Working Direction
[2] Normal to Major Working Direction
* Estimate based on USAEC Regulatory Standard Review Plan, Section 5.3.2 MTEB 5-2
[a] % Shear not reported

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PRESSURE/TEMPERATURE LIMITS (Continued)

The above procedures are needed because there is no direct control on temperature at the 1/4T location; 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 assures conservative operation of the system for the entire cooldown period.

HEATUP

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 vessel wall. The thermal gradients during heatup produce compressive stresses at the inside of the wall that alleviate the tensile stresses produced by internal pressure. The metal temperature at the crack tip lags the coolant temperature; therefore, the $K_{IR}$ for the 1/4T crack during heatup is lower than the $K_{IR}$ 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 such that the effects of compressive thermal stresses and different $K_{IR}$'s for steady-state and finite heatup rates 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 assure 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 pressure-temperature limitations for the case in which a 1/4T deep outside surface flaw is assumed. Unlike the situation at the vessel inside surface, the thermal gradients established at the outside surface during heatup produce stresses which are tensile in nature and thus tend to reinforce any pressure stresses present. These thermal stresses, of course, are dependent on both the rate of heatup and the time (or coolant temperature) along the heatup ramp. Furthermore, since the thermal stresses, at the outside are tensile and increase with increasing heatup rate, a lower bound curve cannot be defined. Rather, each heatup rate of interest must be analyzed on an individual basis.

PRESSURE/TEMPERATURE LIMITS (Continued)

Following the generation of pressure-temperature curves for both the steady-state and finite heatup rate situations, the final limit curves are produced as follows. A composite curve is constructed 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 such that 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.

The leak test limit curve shown in Figure 3.4-2 represents the minimum temperature requirements at the leak test pressure specified by applicable codes. The leak test limit curve was determined by methods of Branch Technical Position MTEB 5-2 and 10 CFR 50, Appendix G.

The criticality limit curve shown in Figure 3.4-2 specifies pressure-temperature limits for core operation to provide additional margin during actual power production. The pressure-temperature limits for core operation (except for low power physics tests) require the reactor vessel to be at a temperature equal to or higher than the minimum temperature required for the in-service hydrostatic test, and at least 40 degrees F higher than the minimum pressure-temperature curve for heatup and cooldown. The maximum temperature for the in-service hydrostatic test for the SQN Unit 2 reactor vessel is 274 degrees F. A vertical line at 274 degrees F on the pressure-temperature curve, intersecting a curve 40 degrees F higher than the pressure-temperature limit curve, constitutes the limit for core operation for the reactor vessel.

Finally, the composite curves for the heatup rate data and the cooldown rate data are adjusted for possible errors in the pressure and temperature sensing instruments by the values indicated on the respective curves.

Although the pressurizer operates in temperature ranges above those for which there is reason for concern of non-ductile failure, operating limits are provided to assure compatibility of operation with the fatigue analysis performed in accordance with the ASME Code requirements.

3/4.4.10 STRUCTURAL INTEGRITY

The inservice inspection and testing programs for ASME Code Class 1, 2 and 3 components ensure that the structural integrity and operational readiness of these components will be maintained at an acceptable level throughout the life of the plant. These programs are in accordance with Section XI of the ASME Boiler and Pressure Vessel Code and applicable Addenda as required by 10 CFR Part 50.55a(g) except where specific written relief has been granted by the Commission pursuant to 10 CFR Part 50.55a (g) (6) (i).

Components of the reactor coolant system were designed prior to the issuance of Section XI of the ASME Boiler and Pressure Vessel Code. These components will be tested to the extent practical within the limitations of the original plant design, geometry and materials of construction.

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

ENCLOSURE 3

# SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

# RELATED TO AMENDMENT NO.158 TO FACILITY OPERATING LICENSE NO. DPR-77

# AND AMENDMENT NO. 148 TO FACILITY OPERATING LICENSE NO. DPR-79

# TENNESSEE VALLEY AUTHORITY

# SEQUOYAH NUCLEAR PLANT, UNITS 1 AND 2

# DOCKET NOS. 50-327 AND 50-328

## 1.0 INTRODUCTION

By letter dated March 1, 1991, as superseded September 6, 1991, the Tennessee Valley Authority (the licensee) submitted a request for changes to the Sequoyah Nuclear Plant, Units 1 and 2 Technical Specifications (TS). The requested changes would revise the pressure/temperature (P/T) limits in the Sequoyah Units 1 and 2 Technical Specifications, Section 3.4. The proposed P/T limits are valid for 16 effective full power years (EFPY) and were developed using Regulatory Guide (RG) 1.99, Revision 2. Generic Letter 88-11, "NRC Position on Radiation Embrittlement of Reactor Vessel Materials and Its Effect on Plant Operations," recommends that RG 1.99, Revision 2, be used in calculating P/T limits, unless the use of different methods can be justified. The P/T limits provide for the operation of the reactor coolant system during heatup, cooldown, criticality, and hydrotest.

To evaluate the P/T limits, the staff uses the following NRC regulations and guidance: Appendices G and H of 10 CFR Part 50; the American Society of Testing Materials (ASTM) Standards and the American Society of Mechanical Engineeers (ASME) Codes, which are referenced in Appendices G and H; 10 CFR 50.36(c)(2); RG 1.99, Revision 2; Standard Review Plan (SRP) Section 5.3.2; and Generic Letter 88-11.

Each licensee authorized to operate a nuclear power reactor is required by 10 CFR 50.36 to provide Technical Specifications for the operation of the plant. In particular, 10 CFR 50.36(c)(2) requires that limiting conditions of operation be included in the Technical Specifications. The P/T limits are among the limiting conditions of operation in the Technical Specifications for all commercial nuclear plants in the United States. Appendices G and H of 10 CFR Part 50 describe specific requirements for fracture toughness and reactor vessel material surveillance that must be considered in setting P/T limits. An acceptable method for constructing the P/T limits is described in SRP Section 5.3.2.

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Appendix G of 10 CFR Part 50 specifies fracture toughness and testing requirements for reactor vessel materials in accordance with the ASME Code and, in particular, that the beltline materials in the surveillance capsules be tested in accordance with Appendix H of 10 CFR Part 50. Appendix H, in turn, refers to ASTM Standards. These tests define the extent of vessel embrittlement at the time of capsule withdrawal in terms of the increase in reference temperature. Appendix G also requires the licensee to predict the effects of neutron irradiation on vessel embrittlement by calculating the adjusted reference temperature (ART) and Charpy upper shelf energy (USE). Generic Letter 88-11 requested that licensees use the methods in Regulatory Guide 1.99, Revision 2, to predict the effect of neutron irradiation on reactor vessel materials. This guide defines the ART as the sum of unirradiated reference temperature, the increase in reference temperature resulting from neutron irradiation, and a margin to account for uncertainties in the prediction method.

Appendix H of 10 CFR Part 50 requires the licensee to establish a surveillance program to periodically withdraw surveillance capsules from the reactor vessel. Appendix H refers to the ASTM Standards which, in turn, require that the capsules be installed in the vessel before startup and that they contain test specimens made from plate, weld, and heat-affected-zone (HAZ) materials of the reactor beltline.

## 2.0 EVALUATION

The staff evaluated the effect of neutron irradiation embrittlement on each beltline material in the Sequoyah 1 and 2 reactor vessels. The amount of irradiation embrittlement was calculated in accordance with RG 1.99, Revision 2. The staff has determined that the material with the highest ART at 16 EFPY in Sequoyah 1 was the lower shell forging with 0.13% copper (CU), 0.76% nickel (Ni), and an initial RT(ndt) of 73°F; the material with the highest ART at 16 EFPY for Sequoyah 2 was the weld between the intermediate and lower shell forgings with 0.13% copper (Cu), 0.11% nickel (Ni), and an initial RT(ndt) of -4°F.

So far, the licensee has removed two surveillance capsules from each unit. The results from capsules T and U of Sequoyah 1 were published in Westinghouse report WCAP-10340 and Southwest Research Institute Report SwRI 06-8651. The results from capsules T and U of Sequoyah 2 were published in Westinghouse Report WCAP-10509 and Southwest Research Institute Report SwRI 17-8851. All surveillance capsules contained Charpy impact specimens and tensile specimens made from base metal, weld metal, and HAZ metal.

For the limiting beltline material in Sequoyah 1, the lower shell forging, the staff calculated the ART to be 194.4°F at 1/4T (T = reactor vessel beltline thickness) and 165.6°F for 3/4T at 16 EFPY. The staff used a neutron fluence of 1.16E19 n/sq.cm at 1/4T and 4.11E18 n/sq.cm at 3/4T. The ART was determined by the least squares extrapolation method using the surveillance data from Sequoyah 1. The least squares method is described in Section 2.1 of RG 1.99, Revision 2.

For the limiting beltline material in Sequoyah 2, the weld between the intermediate and lower shell forgings, the staff calculated the ART to be 141.5°F at 1/4T and 103.1°F for 3/4T at 16 EFPY. In this case, the staff used a neutron fluence of 5.15 E18 n/sq.cm at 1/4T and 1.83E18 n/sq.cm at 3/4T. The ART was determined from the Sequoyah 2 surveillance data in the same way.

The licensee used the method in RG 1.99, Revision 2, to calculate an ART of 1.95°F at 16 EFPY at 1/4T for the same limiting lower shell forging for Sequoyah 1, and an ART of 1.42°F at 16 EFPY at 1/4T for the same limiting weld between the intermediate and lower shell forgings for Sequoyah 2. Since the licensee's ARTs of 195°F and 142°F are almost the same as the staff's ARTs of 194.4°F and 141.5°F, they are acceptable. Substituting the ARTs of 194.4 °F and 141.5°F into equations in SRP 5.3.2, the staff verified that the proposed P/T limits for heatup, cooldown, and hydrotest meet the beltline material requirements in Appendix G of 10 CFR Part 50.

In addition to beltline materials, Appendix G of 10 CFR Part 50 also imposes P/T limits based on the reference temperature for the reactor vessel closure flange materials. Section IV.2 of Appendix G states that when the pressure exceeds 20% of the preservice system hydrostatic test pressure, the temperature of the closure flange regions highly stressed by the bolt preload must exceed the reference temperature of the material in those regions by at least 120°F for normal operation and by 90°F for hydrostatic pressure tests and leak tests. Based on the flange reference temperature of -49°F for Sequoyah 1 and -13°F for Sequoyah 2, the staff has determined that the proposed P/T limits satisfy Section IV.2 of Appendix G.

Section IV.B of Appendix G requires that the predicted Charpy USE at end of life be above 50 ft-lb. For Sequoyah 1, the measured Charpy USE is 58 ft-lb for the lower shell forging, based on data from surveillance capsule U withdrawn at 2.92 EFPY. This is a 19.4% reduction from the unirradiated value of 72 ft-lb. Using the method in RG 1.99, Revision 2, the staff calculated that the Charpy USE of the lower shell forging at the end of life will be 55.2 ft-lb. Data from capsule T was discounted because of change in USE was found there. For Sequoyah 2, the material with the lowest initial USE is the intermediate shell forging with 93 ft-lb. Using the Figure 2 of RG 1.99, Revision 2, the staff calculated that the EOL USE at 1/4T will be 67.9 ft-lb. This number has not been adjusted by surveillance data because applying capsule data would give higher EOL USEs. Since both numbers are greater than 50 ft-lb, they are acceptable.

The staff concludes that the proposed P/T limits for the reactor coolant system for heatup, cooldown, leak test, and criticality are valid through 16

EFPY because the limits conform to the requirements of Appendices G and H of 10 CFR Part 50. The proposed P/T limits also satisfy Generic Letter 88-11 because the method in RG 1.99, Revision 2, was used to calculate the ARTs. Hence, the proposed P/T limits are acceptable for use in the Sequoyah Units 1 and 2 Technical Specifications.

## 3.0 STATE CONSULTATION

In accordance with the Commission's regulations, the Tennessee State official was notified of the proposed issuance of the amendment. The State official had no comments.

## 4.0 ENVIRONMENTAL CONSIDERATION

The amendment changes a requirement with respect to installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20. The NRC staff has determined that the amendment involves no significant increase in the amounts, and no significant change in the types, of any effluents that may be released offsite, and that there is no significant increase in individual or cumulative occupational radiation exposure. The Commission has previously issued a proposed finding that the amendment involves no significant hazards consideration, and there has been no public comment on such finding (56 FR 13669 and 56 FR 49928). Accordingly, the amendment meets the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Pursuant to 10 CFR 51.22(b) no environmental impact statement or environmental assessment need be prepared in connection with the issuance of the amendment.

## 5.0 CONCLUSION

The Commission has concluded, based on the considerations discussed above, that: (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, (2) such activities will be conducted in compliance with the Commission's regulations, and (3) the issuance of the amendment will not be inimical to the common defense and security or to the health and safety of the public.

Principal Contributor: S. Sheng

Date: March 31, 1992