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   A. The application for amendment by Tennessee Valley Authority (the licensee) dated March 1, 1991, and superseded September 6, 1991, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and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

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

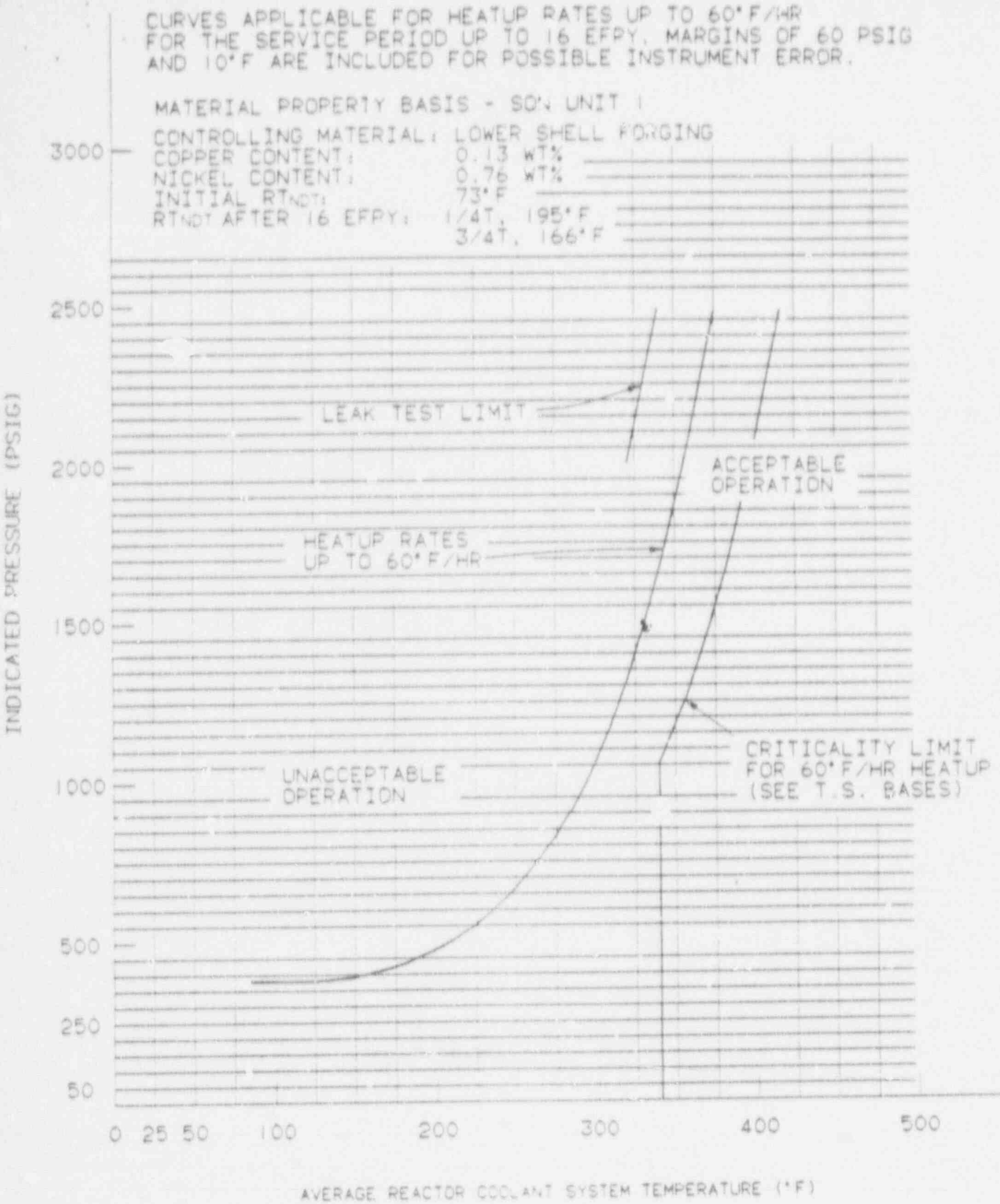

FIGURE 3.4-2 SEQUOYAH UNIT 1 REACTOR COOLANT SYSTEM HEATUP LIMITATIONS APPLICABLE UP TO 16 EFPY

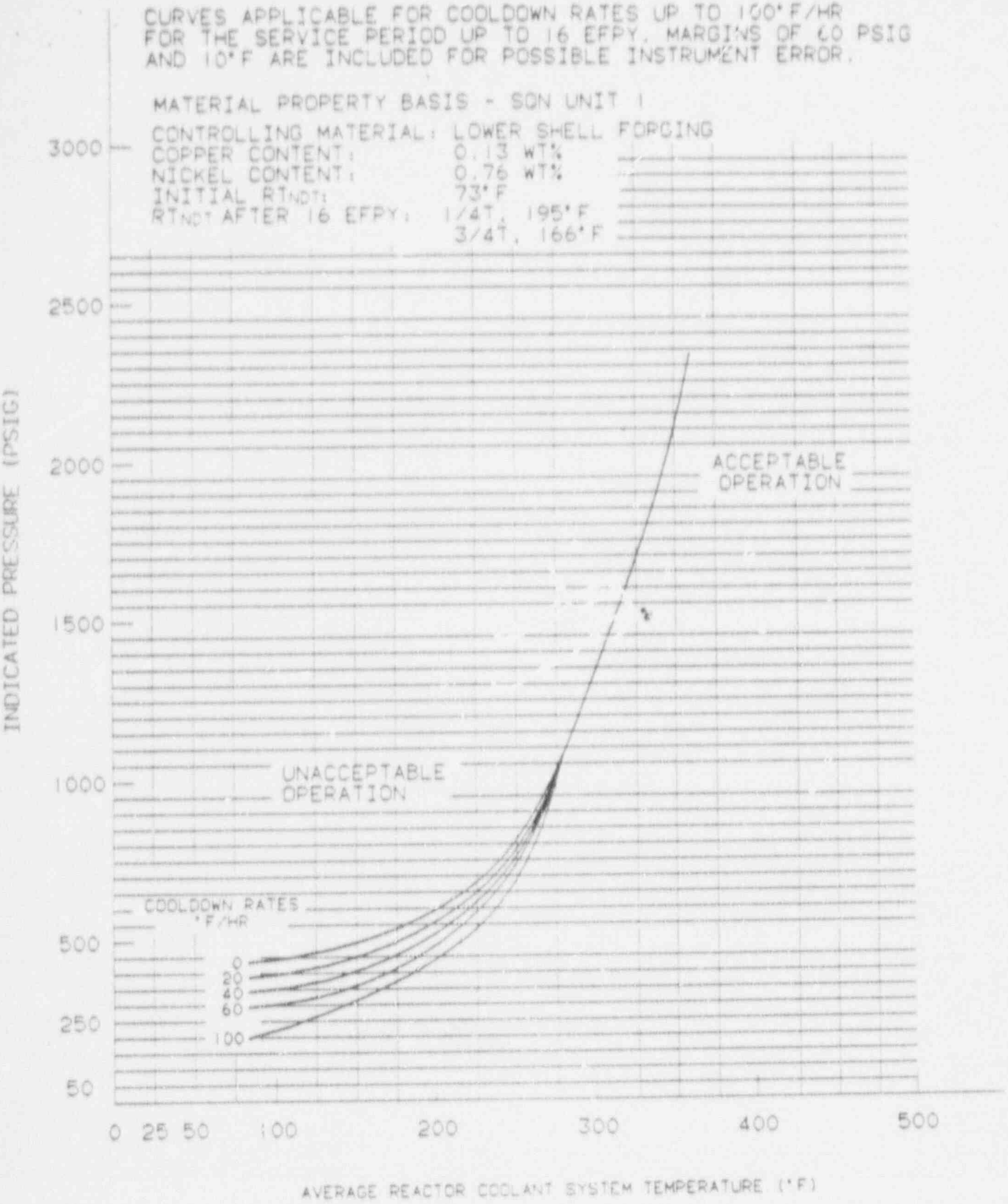

FIGURE 3.4-3 SEQUOYAH UNIT 1 REACTOR COOLANT SYSTEM COOLDOWN LIMITATIONS APPICICABLE UP TO 16 EFPY

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 1.94×10^{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.

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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) PMWD1	NMWD2	RT_{NDT} (°F)	AVERAGE UPPER SHELF ENERGY (ft-lb) PMWD1	NMWD2
Clos. Hd. Dome	52841-1	A533B,Cl.1	-	-	-40	+14	+34	-26	104[a]	-
C os. Hd. Ring	(B75600)	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:

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

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

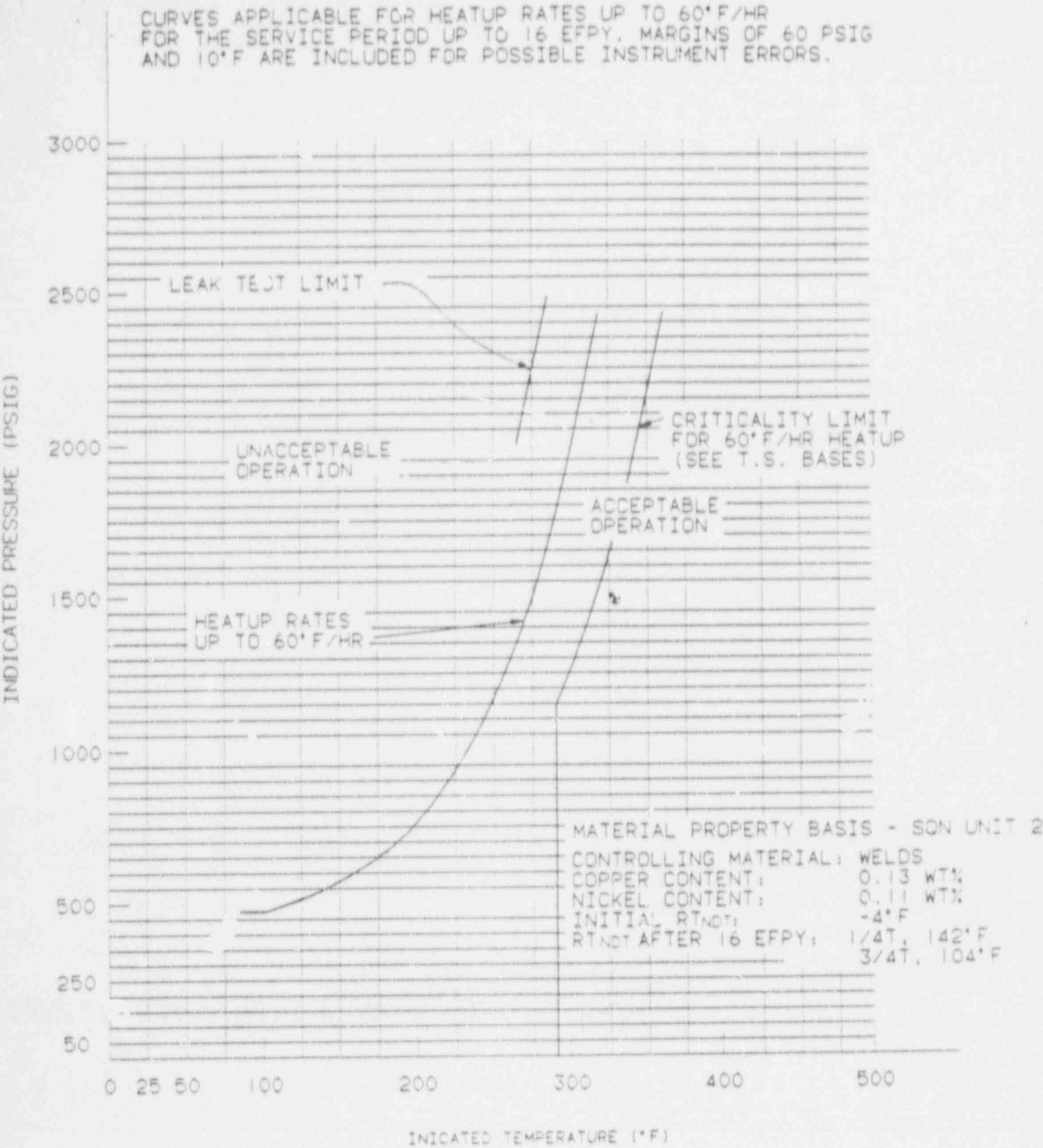

FIGURE 3.4-2 SEQUOYAH UNIT 2 REACTOR COOLANT SYSTEM HEATUP LIMITATIONS APPLICABLE UP TO 16 EFPY

Amendment No. 148

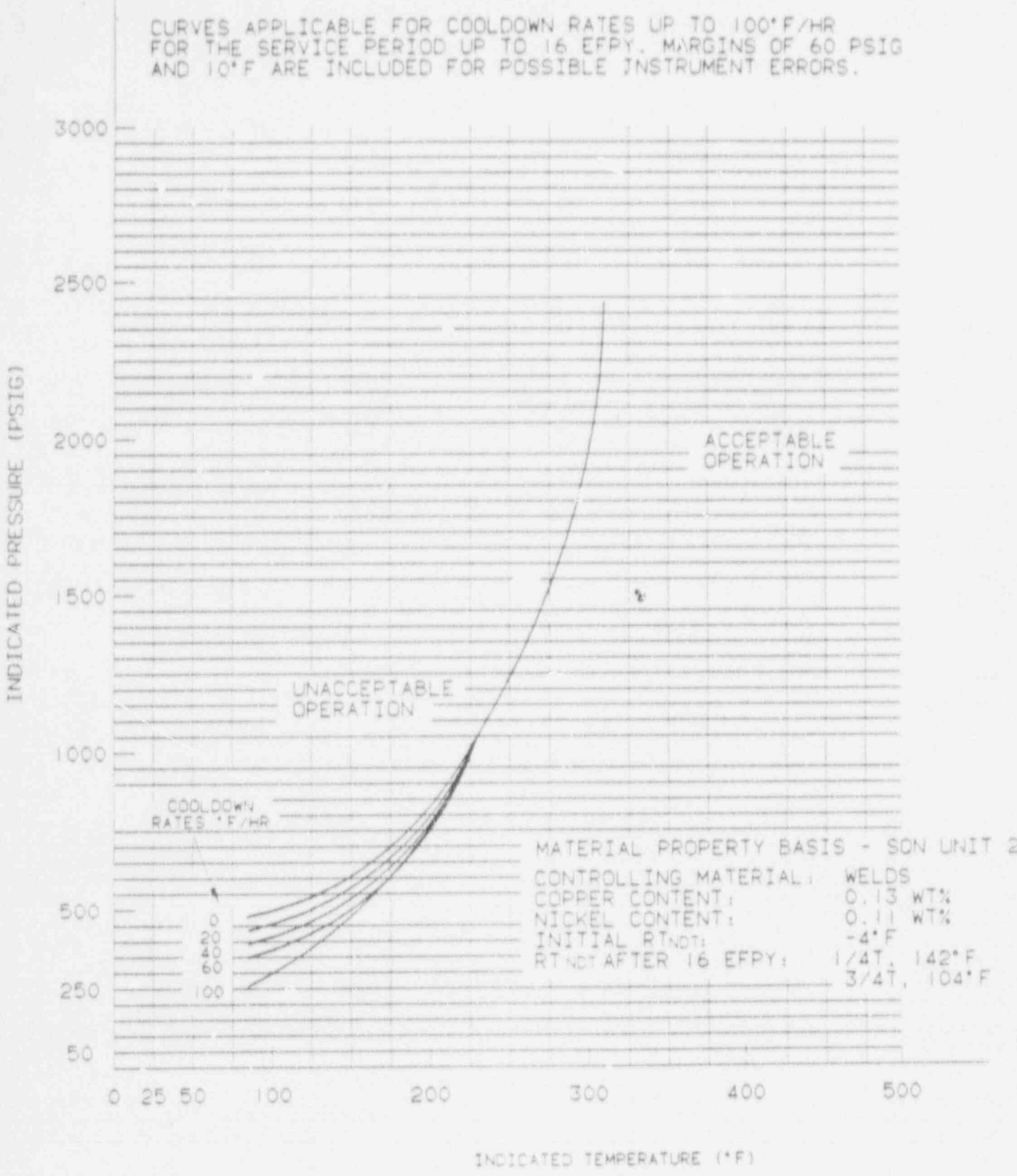

FIGURE 3.4-3 SEQUOYAH UNIT 2 REACTOR COOLANT SYSTEM COOLDOWN LIMITATIONS APPICICABLE UP TO 16 EFPY

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 0.864×10^{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 ΔRT_{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 ΔRT_{NDT} determined from the surveillance capsule exceeds the calculated ΔRT_{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 dimenstions 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, ΔRT_{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 [illegible] GHNE[illegible] ATA

COMPONENT	HEAT NO.	MATERIAL GRADE	CU %	Ni %	NDTT °F	50 F[illegible] 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

This page intentionally deleted.

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.