WCAP-12971

QA Record

APPROVED
This approv does not relieve the Contractor from of his responsibility for the ctness of design, details c sions.
Lett No. N8553
Date: July 28, 1991
TENNESSEE VALLEY AUTHORITY
SQEP (N) BY P.G. Trudel

PROJECT Sequoyah DISCIPLINE N
CONTRACT 91NNP-86305B UNIT 2
DESC. RCS Heatup/Cooldown Curves
DWG/DOC NO. WCAP-12971
SHEET - OF - REV 00
DATE 7/28/91 ECN/FCR - FILE N2N-048

HEATUP AND COOLDOWN LIMIT CURVES FOR NORMAL OPERATION

SEQUOYAH UNIT 2

J. M. Chicots
N. K. Ray

June 1991

Work Performed Under Shop Order TJAP-N004

Prepared by Westinghouse Electric Corporation
for the Tennessee Valley Authority

Approved by: T. A. Meyer
T. A. Meyer, Manager
Structural Reliability & Plant Life Optimization

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

TABLE OF CONTENTS

LIST OF ILLUSTRATIONS

LIST OF TABLES

1. INTRODUCTION

Heatup and cooldown limit curves are calculated using the most limiting value of RT_{NDT} (reference nil-ductility temperature) for the reactor vessel. The most limiting RT_{NDT} of the material in the core region of the reactor vessel is determined by using the preservice reactor vessel material fracture toughness properties and estimating the radiation-induced ΔRT_{NDT}. RT_{NDT} is designated as the higher of either the drop weight nil-ductility transition temperature (NDTT) or the temperature at which the material exhibits at least 50 ft-lb of impact energy and 35-mil lateral expansion (normal to the major working direction) minus 60°F.

RT_{NDT} increases as the material is exposed to fast-neutron radiation. Therefore, to find the most limiting RT_{NDT} at any time period in the reactor's life, ΔRT_{NDT} due to the radiation exposure associated with that time period must be added to the original unirradiated RT_{NDT}. The extent of the shift in RT_{NDT} is enhanced by certain chemical elements (such as copper and nickel) present in reactor vessel steels. The Nuclear Regulatory Commission (NRC) has published a method for predicting radiation embrittlement in Regulatory Guide 1.99 Rev. 2 (Radiation Embrittlement of Reactor Vessel Materials)[1]. Regulatory Guide 1.99, Revision 2 is used for the calculation of RT_{NDT} values at 1/4T and 3/4T locations (T is the thickness of the vessel at the beltline region).

2. FRACTURE TOUGHNESS PROPERTIES

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

3. CRITERIA FOR ALLOWABLE PRESSURE-TEMPERATURE RELATIONSHIPS

The ASME approach for calculating the allowable limit curves for various heatup and cooldown rates specifies that the total stress intensity factor, K_I, for the combined thermal and pressure stresses at any time during heatup or cooldown cannot be greater than the reference stress intensity factor, K_{IR}, for the metal temperature at that time. K_{IR} is obtained from the reference fracture toughness curve, defined in Appendix G to the ASME Code[3]. The K_{IR} curve is given by the following equation:

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

where

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

Therefore, the governing equation for the heatup-cooldown analysis is defined in Appendix G of the ASME Code[3] as follows:

$$C\, K_{IM} + K_{IT} \le K_{IR} \quad (2)$$

where

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

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

K_{IR} = function of temperature relative to the RT_{NDT} of the material

C = 2.0 for Level A and Level B service limits

C = 1.5 for hydrostatic and leak test conditions during which the reactor core is not critical

At any time during the heatup or cooldown transient, K_{IR} is determined by the metal temperature at the tip of the postulated flaw, the appropriate value for RT_{NDT}, and the reference fracture toughness curve. The thermal stresses resulting from the temperature gradients through the vessel wall are calculated and then the corresponding (thermal) stress intensity factors, K_{IT}, for the reference flaw are computed. From equation 2, the pressure stress intensity factors are obtained and, from these, the allowable pressures are calculated.

For the calculation of the allowable pressure versus coolant temperature during cooldown, the reference flaw of Appendix G to the ASME Code is assumed to exist at the inside of the vessel wall. During cooldown, the controlling location of the flaw is always at the inside of the wall because the thermal gradients produce tensile stresses at the inside, which increase with increasing cooldown rates. Allowable pressure-temperature relations are generated for both steady-state and finite cooldown rate situations. From these relations, composite limit curves are constructed for each cooldown rate of interest.

The use of the composite curve in the cooldown analysis is necessary because control of the cooldown procedure is based on the measurement of reactor coolant temperature, whereas the limiting pressure is actually dependent on the material temperature at the tip of the assumed flaw.

During cooldown, the 1/4 T vessel location is at a higher temperature than the fluid adjacent to the vessel ID. This condition, of course, is not true for the steady-state situation. It follows that, at any given reactor coolant temperature, the ΔT developed during cooldown results in a higher value of K_{IR} at the 1/4 T location for finite cooldown rates than for steady-state operation. Furthermore, if conditions exist so that the increase in K_{IR} exceeds K_{IT}, the calculated allowable pressure during cooldown will be greater than the steady-state value.

The above procedures are needed because there is no direct control on temperature at the 1/4 T location and, therefore, allowable pressures may unknowingly be violated if the rate of cooling is decreased at various

intervals along a cooldown ramp. The use of the composite curve eliminates this problem and ensures conservative operation of the system for the entire cooldown period.

Three separate calculations are required to determine the limit curves for finite heatup rates. As is done in the cooldown analysis, allowable pressure-temperature relationships are developed for steady-state conditions as well as finite heatup rate conditions assuming the presence of a 1/4 T defect at the inside of the wall that alleviate the tensile stresses produced by internal pressure. The metal temperature at the crack tip lags the coolant temperature; therefore, the K_{IR} for the 1/4 T crack during heatup is lower than the K_{IR} for the 1/4 T crack during steady-state conditions at the same coolant temperature. During heatup, especially at the end of the transient, conditions may exist so that the effects of compressive thermal stresses and lower K_{IR}'s do not offset each other, and the pressure-temperature curve based on steady-state conditions no longer represents a lower bound of all similar curves for finite heatup rates when the 1/4 T flaw is considered. Therefore, both cases have to be analyzed in order to ensure that at any coolant temperature the lower value of the allowable pressure calculated for steady-state and finite heatup rates is obtained.

The second portion of the heatup analysis concerns the calculation of the pressure-temperature limitations for the case in which a 1/4 T deep outside surface flaw is assumed. Unlike the situation at the vessel inside surface, the thermal gradients established at the outside surface during heatup produce stresses which are tensile in nature and therefore tend to reinforce any pressure stresses present. These thermal stresses are dependent on both the rate of heatup and the time (or coolant temperature) along the heatup ramp. Since the thermal stresses at the outside are tensile and increase with increasing heatup rates, each heatup rate must be analyzed on an individual basis.

Following the generation of pressure-temperature curves for both the steady state and finite heatup rate situations, the final limit curves are produced by constructing a composite curve based on a point-by-point comparison of the steady-state and finite heatup rate data. At any given temperature, the

allowable pressure is taken to be the lesser of the three values taken from the curves under consideration. The use of the composite curve is necessary to set conservative heatup limitations because it is possible for conditions to exist wherein, over the course of the heatup ramp, the controlling condition switches from the insi to the outside, and the pressure limit must at all times be based on analy s of the most critical criterion.

Finally, the 1983 Amendment to 10CFR50[4] has a rule which addresses the metal temperature of the closure head flange and vessel flange regions. This rule states that the metal temperature of the closure flange regions must exceed the material RT_{NDT} by at least 120°F for normal operation when the pressure exceeds 20 percent of the preservice hydrostatic test pressure.

Table 1 indicates that the initial RT_{NDT} of -13°F occurs in the closure head flange of Sequoy h Unit 2, so the minimum allowable temperature of this region is 107°F. These limits are shown in Figu 1 and 2 whenever applicable.

4. HEATUP AND COOLDOWN LIMIT CURVES

Limit curves for normal heatup and cooldown of the primary reactor pressure vessel have been calculated using the methods discussed in Section 3. Figures 1, 2 and 3 contain the heatup curves for 20, 40 and 60°F/hr, respectively. Figure 4 contains the cooldown curves up to 100°F/hr. Figures 1 through 4 are applicable for the first 16 EFPY of operation. No margins were included in the development of heatup and cooldown curves to allow for possible instrumentation errors.

Allowable combinations of temperature and pressure for specific temperature change rates for heatup operation are below and to the right of the limit lines shown in Figures 1 through 3. This is in addition to other criteria which must be met before the reactor is made critical.

The leak limit curve shown in Figures 1, 2 and 3 represent minimum temperature requirements at the leak test pressure specified by applicable codes[2,3].

The leak test limit curve was determined by methods of References 2 and 4.

The criticality limit curve shown in Figures 1, 2 and 3, specifies pressure-temperature limits for core operation to provide additional margin during actual power production as specified in Reference 4. The pressure-temperature limits for core operation (except for low power physics tests) are that the reactor vessel must be at a temperature equal to or higher than the minimum temperature required for the inservice hydrostatic test, and at least 40°F higher than the minimum pressure-temperature curve for heatup and cooldown calculated as described in Section 3. The maximum temperature for the inservice hydrostatic test for the Sequoyah Unit 2 reactor vessel is 274°F. A vertical line at 274°F on the pressure-temperature curve, intersecting a curve 40°F higher than the pressure-temperature limit curve, constitutes the limit for core operation for the reactor vessel.

Figures 1 through 4 define limits for ensuring prevention of nonductile failure for the Sequoyah Unit 2 reactor vessel.

5. ADJUSTED REFERENCE TEMPERATURE

From Regulatory Guide 1.99 Rev. 2 [1] the adjusted reference temperature (ART) for each material in the beltline is given by the following expression:

$$ART = \text{Initial } RT_{NDT} + \Delta RT_{NDT} + \text{Margin} \qquad (3)$$

Initial RT_{NDT} is the reference temperature for the unirradiated material as defined in paragraph N2331 of Section III of the ASME Boiler and Pressure Vessel Code. If measured values of initial RT_{NDT} for the material in question are not available, generic mean values for that class of material may be used if there are sufficient test results to establish a mean and standard deviation for the class.

ΔRT_{NDT} is the mean value of the adjustment in reference temperature caused by irradiation and should be calculated as follows:

$$\Delta RT_{NDT} = [CF] f^{(0.28-0.10 \log f)} \tag{4}$$

To calculate ΔRT_{NDT} at any depth (e.g., at 1/4T or 3/4T), the following formula must first be used to attenuate the fluence at the specific depth.

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

where x (in inches) is the depth into the vessel wall measured from the vessel clad/base metal interface. The resultant fluence is then put into equation (4) to calculate ΔRT_{NDT} at the specific depth.

CF (°F) is the chemistry factor, obtained from Reference 1. In addtion, the chemistry factor is also calculated using surveillance capsule data. A sample of this calculation is shown in Table 2.

All materials in the beltline region of Sequoyah Unit 2 were considered for the limiting material. RT_{NDT} at 1/4T and 3/4T are summarized in Table 3. From Table 3, it can be seen that the limiting material is the weld for heatup and cooldown curves applicable up to 16 EFPY. A sample calculation for the RT_{NDT} for 16 EFPY is shown in Table 4.

TABLE 1

SEQUOYAH UNIT 2 REACTOR VESSEL TOUGHNESS TABLE (Unirradiated)

Material Description	CU (%)	NI (%)	I-RTNDT (a) (°F)
Closure Head Flange (b)	--	--	-13
Vessel Flange (b)	--	--	-22
Intermediate Shell Forging	0.13	0.74	10
Lower Shell Forging	0.14	0.76	-22
Welds	0.13	0.11	-4

a. The initial RT_{NDT} (I) values for the forgings and welds are measured values.

b. These items are to be used for considering flange requirements for heatup/cooldown curves[4].

TABLE 2

CALCULATION OF CHEMISTRY FACTOR BASED ON SURVEILLANCE CAPSULE DATA

Material Description	Capsule	Fluence	FF	DRTNDT	FF*DRTNDT	$(FF)^2$
Forging 05 (axial)	T	0.220	0.592	25	14.811	0.351
	U	0.643	0.876	62	54.327	0.768
Forging 05 (tangential)	T	0.220	0.592	60	35.546	0.351
	U	0.643	0.876	93	81.490	0.768
					186.173	2.238

$$\text{Chemistry Factor} = \frac{186.173}{2.238} = 83.205$$

Weld Metal	T	0.220	0.592	80	47.394	0.351
	U	0.643	0.876	130	113.911	0.768
					161.305	1.119

$$\text{Chemistry Factor} = \frac{161.305}{1.119} = 144.182$$

TABLE 3

SUMMARY OF ADJUSTED REFERENCE TEMPERATURE (ART) AT 1/4T and 3/4T LOCATION

	16 EFPY	
	RT_{NDT} at	
Component	1/4T (°F)	3/4T (°F)
Intermediate Shell Forging	(95)	(73)
Lower Shell Forging	97	70
Welds	(142)*	(104)*

RT_{NDT} numbers within ( ) are based on the chemistry factor calculated using capsule data.

* These RT_{NDT} numbers are used to generate heatup and cooldown curves applicable up to 16 EFPY.

TABLE 4

CALCULATION OF ADJUSTED REFERENCE TEMPERATURES FOR LIMITING SEQUOYAH UNIT 2 REACTOR VESSEL MATERIAL - WELD

Parameter	Regulatory Guide 1.99 - Revision 2, 16 EFPY, 1/4 T	Regulatory Guide 1.99 - Revision 2, 16 EFPY, 3/4 T
Chemistry Factor, CF (°F)	68 (144)	68 (144)
Fluence, f (10^{19} n/cm^2)[(a)]	.5206	.1889
Fluence Factor, ff	.818	.556
ΔRT_{NDT} = CF x ff (°F)	56 (118)	38 (80)
Initial RT_{NDT}, I (°F)	-4	-4
Margin, M (°F) [(b)]	56 (28)	56 (28)
Revision 2 to Regulatory Guide 1.99		
Adjusted Reference Temperature, ART = Initial RT_{NDT} + ΔRT_{NDT} + Margin	108 (142)	90 (104)

(a) Fluence, f, is based upon f_{surf} (10^{19} n/cm^2, E>1 Mev) = 0.8644 at 16 EFPY. The Sequoyah Unit 2 reactor vessel wall thickness is 8.608 inches at the beltline region.

(b) Margin is calculated as, $M = 2\,[\,\sigma_I^2 + \sigma_\Delta^2]^{0.5}$. The standard deviation for the initial RT_{NDT} margin term, σ_I, is assumed to be 0°F since the initial RT_{NDT} is a measured value. The standard deviation for ΔRT_{NDT} term, σ_Δ, is 28°F for the weld, except that σ_Δ need not exceed 0.5 times the mean value of ΔRT_{NDT}. σ_Δ, is 14°F for the weld (cut in half) when surveillance data is used.

The numbers within ( ) are calculated using surveillance capsule data.

MATERIAL PROPERTY BASIS

LIMITING ART AFTER 16 EFPY: 1/4T, 142°F
3/4T, 104°F

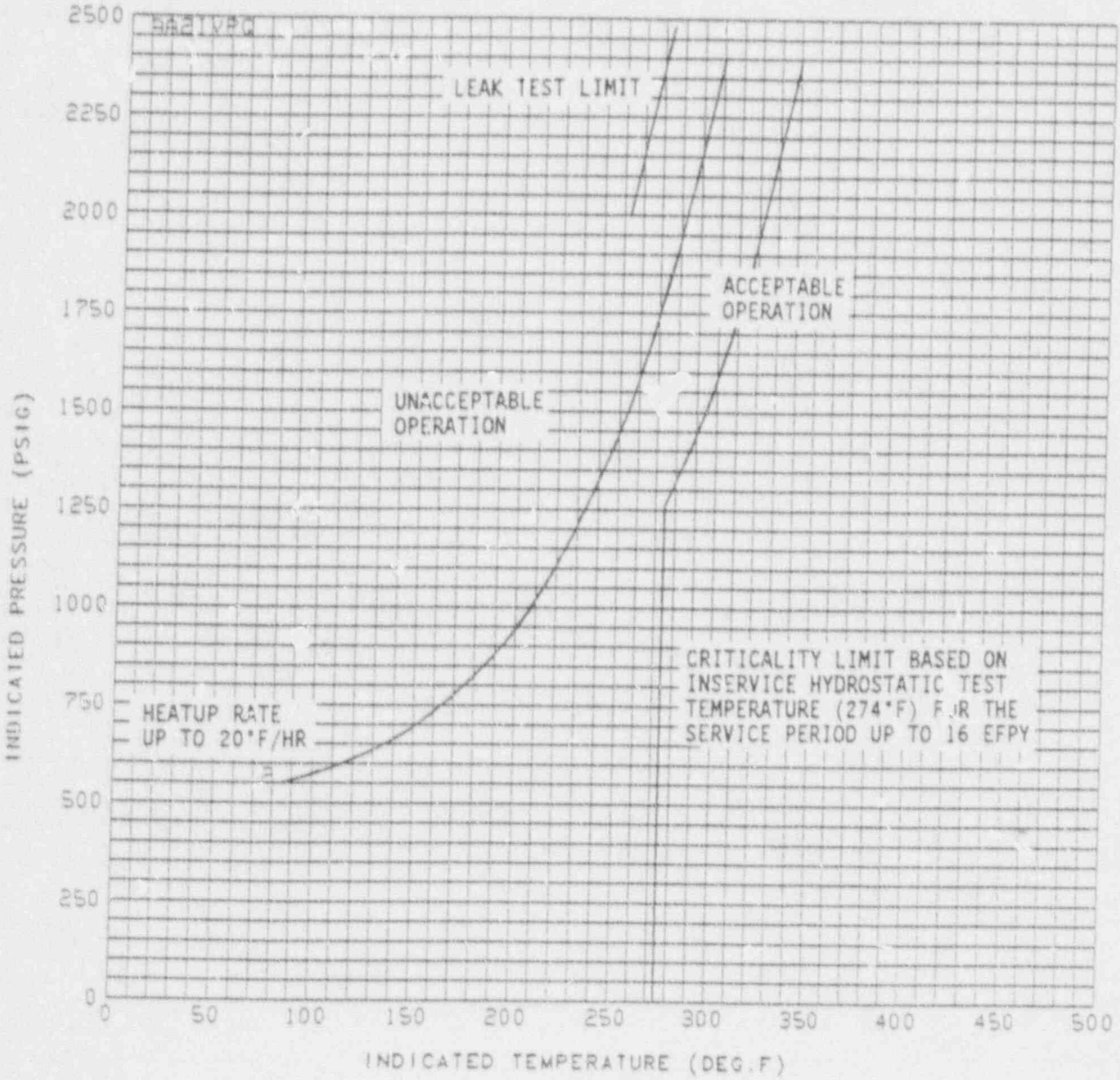

Figure 1. Sequoyah Unit 2 Reactor Coolant System Heatup Limitations (Heat up rates up to 20°F/hr) Applicable for the First 16 EFPY (Without Margins for Instrumentation Errors)

MATERIAL PROPERTY BASIS

LIMITING ART AFTER 16 EFPY: 1, 142°F
3/4T, 104°F

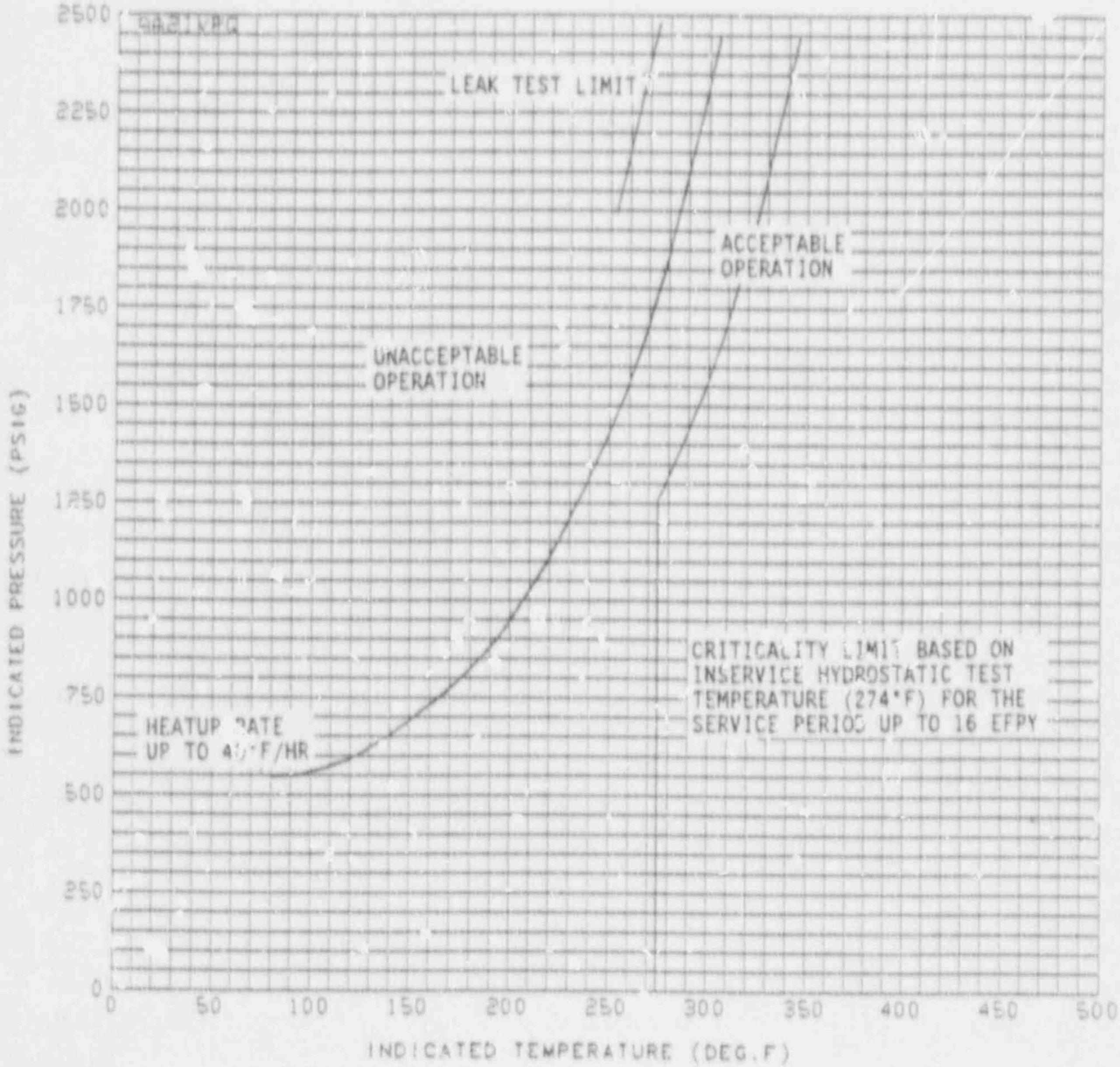

Figure 2. Sequoyah Unit 2 Reactor Coolant System Heatup Limitations (Heat up rates up to 40°F/hr) Applicable for the First 16 EFPY (Without Margins for Instrumentation Errors)

MATERIAL PROPERTY BASIS

LIMITING ART AFTER 16 EFPY: 1/4T, 142°F
3/4T, 104°F

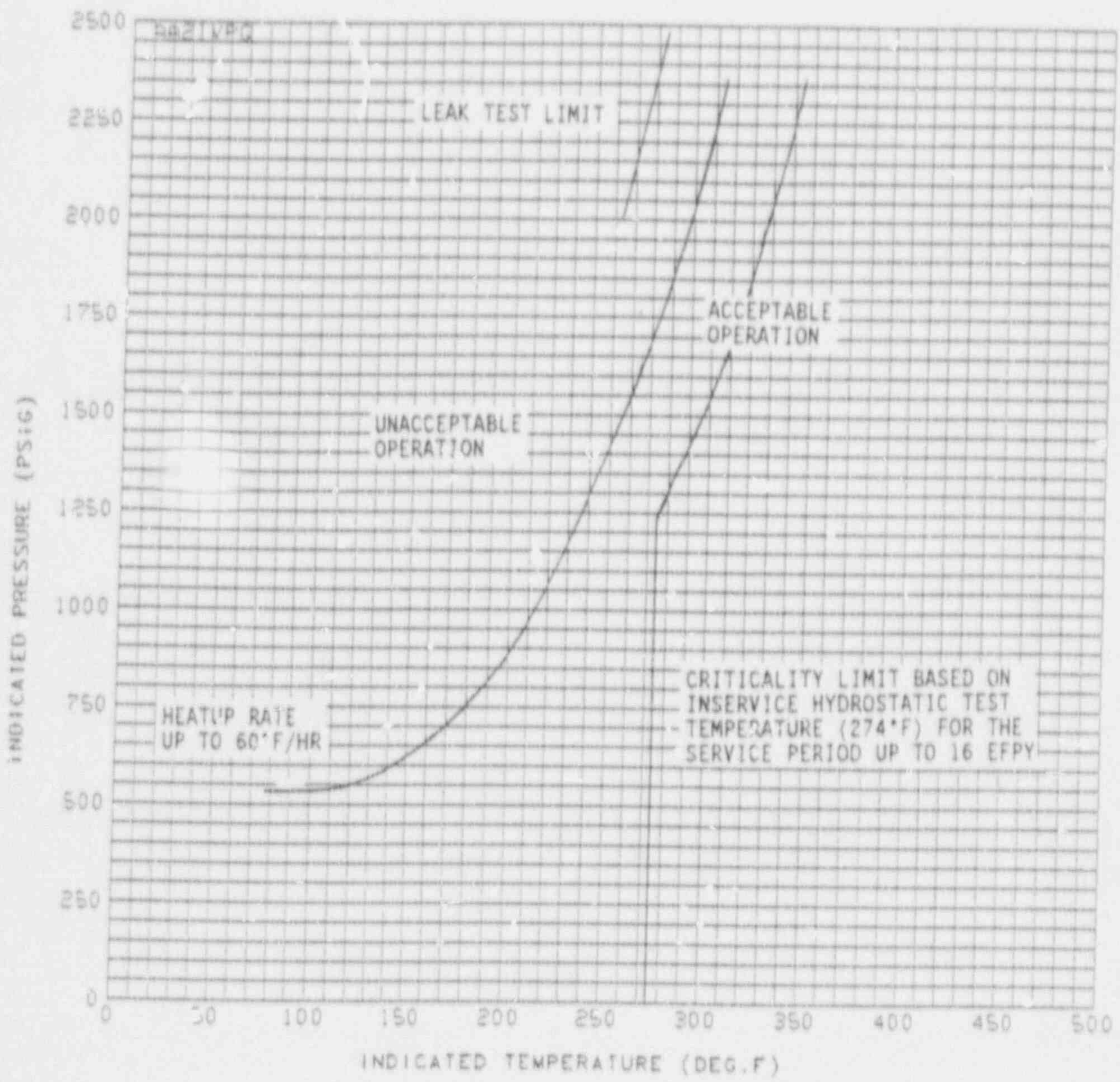

Figure 3. Sequoyah Unit 2 Reactor Coolant System Heatup Limitations (Heat up rates up to 60°F/hr) Applicable for the First 16 EFPY (Without Margins for Instrumentation Errors)

MATERIAL PROPERTY BASIS

LIMITING ART AFTER 16 EFPY: 1/4T, 142°F
3/4T, 104°F

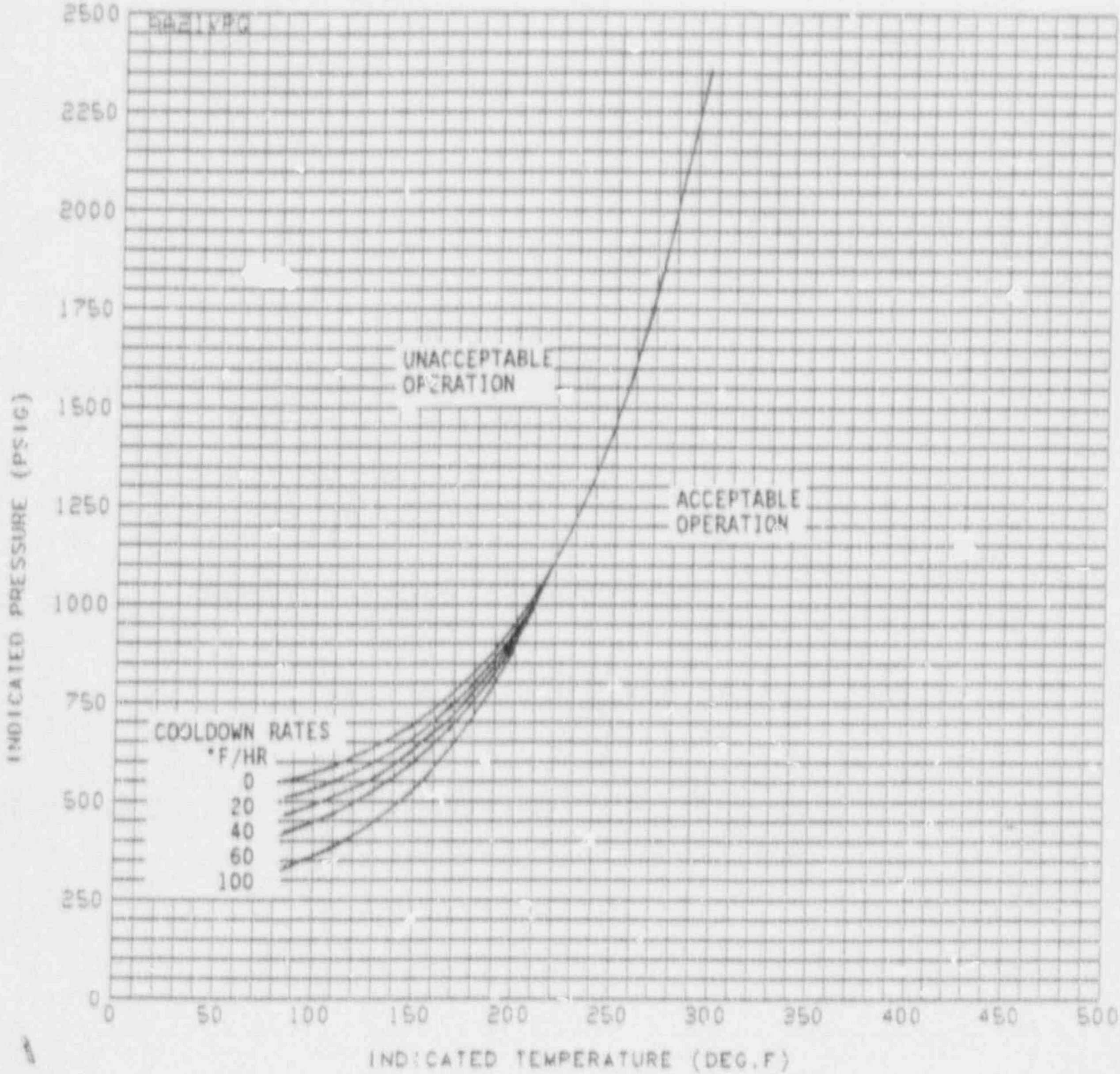

Figure 4. Sequoyah Unit 2 Reactor Coolant System Cooldown (Cooldown rates up to 100°F/hr) Limitations Applicable for the First 16 EFPY (Without Margins for Instrumentation Errors)

6. REFERENCES

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

2 "Fracture Toughness Requirements," Branch Technical Position MTEB 5-2, Chapter 5.3.2 in Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants, LWR Edition, NUREG-0800, 1981.

3 ASME Boiler and Pressure Vessel Code, Section III, Division 1 - Appendixes, "Rules for Construction of Nuclear Power Plant Components, Appendix G, Protection Against Nonductile Failure," pp. 558-563, 1986 Edition, American Society of Mechanical Engineers, New York, 1986.

4 Code of Federal Regulations, 10CFR50, Appendix G, "Fracture Toughness Requirements," U.S. Nuclear Regulatory Commission, Washington, D.C., Federal Register, Vol. 48 No. 104, May 27, 1983.

ATTACHMENT 1

DATA POINTS FOR HEATUP AND COOLDOWN CURVES

(Without Margins for Instrumentation Errors)

TEN 20 DEG-F/HR HEATUP CURVE REG. GUIDE 1.99,REV.2 W/O MARGIN 05/14/91

THE FOLLOWING DA A WERE CALCULATEDFOR THE INSERVICE HYDROSTATIC LEAK TEST.

MINIMUM INSERVICE LEAK TEST TEMPERATURE (16.000 EFPY)

PRESSURE (PSI)	TEMPERATURE (DEG.F)
2000	253
2485	274

PRESSURE (PSI)	PRESSURE STRESS (PSI)	1.5 K1M (PSI SQ.RT.IN.)
2000	21145	89012
2485	26273	111558

TEN 20 DEG-'/HR HEATUP CURVE REG. GUIDE 1.99,REV.2 W/O MARGIN

05/14/91

COMPOSITE CURVE PLOTTED FOR HEATUP PROFILE 2   HEATUP RATE(S) (DEG.F/HR) = 20.0

IRRADIATION PERIOD = 16.000 EFP YEARS
FLAW DEPTH = (1-AOWIN)T

	INDICATED TEMPERATURE (DEG.F)	INDICATED PRESSURE (PSI)		INDICATED TEMPERATURE (DEG.F)	INDICATED PRESSURE (PSI)		INDICATED TEMPERATURE (DEG.F)	INDICATED PRESSURE (PSI)
1	85.000	550.75	16	160.000	733.02	31	235.000	1264.36
2	90.000	557.74	17	165.000	753.57	32	240.000	1320.25
3	95.000	565.26	18	170.000	775.81	33	245.000	1380.62
4	100.000	573.34	19	175.000	799.55	34	250.000	1445.06
5	105.000	582.03	20	180.000	825.23	35	255.000	1514.16
6	110.000	591.36	21	185.000	852.69	36	260.000	1588.31
7	115.000	601.28	22	190.000	882.32	37	265.000	1667.78
8	120.000	612.08	23	195.000	914.08	38	270.000	1752.85
9	125.000	623.69	24	200.000	948.15	39	275.000	1843.93
10	130.000	636.15	25	205.000	984.74	40	280.000	1941.52
11	135.000	649.44	26	210.000	1024.29	41	285.000	2045.91
12	140.000	663.86	27	215.000	1066.60	42	290.000	2157.54
13	145.000	679.37	28	220.000	1112.02	43	295.000	2276.62
14	150.000	695.86	29	225.000	1160.79	44	300.000	2403.82
15	155.000	713.78	30	230.000	1212.04			

TEN 40 DEG-F/HR HEATUP CURVE REG. GUIDE 1.99,REV.2 W/O MARGIN 05/14/91

COMPOSITE CURVE PLOTTED FOR HEATUP PROFILE 2 HEATUP RATE(S) (DEG.F/HR) = 40.0

IRRADIATION PERIOD = 16.000 EFP YEARS
FLAW DEPTH = (1-AOWIN)T

	INDICATED TEMPERATURE (DEG.F)	INDICATED PRESSURE (PSI)		INDICATED TEMPERATURE (DEG.F)	INDICATED PRESSURE (PSI)		INDICATED TEMPERATURE (DEG.F)	INDICATED PRESSURE (PSI)
1	85.000	550.75	16	160.000	733.02	31	235.000	1263.10
2	90.000	552.06	17	165.000	753.57	32	240.000	1315.05
3	95.000	553.02	18	170.000	775.81	33	245.000	1371.09
4	100.000	556.71	19	175.000	799.55	34	250.000	1431.22
5	105.000	562.86	20	180.000	825.23	35	255.000	1495.46
6	110.000	570.91	21	185.000	852.69	36	260.000	1564.35
7	115.000	580.84	22	190.000	882.32	37	265.000	1638.21
8	120.000	592.28	23	195.000	914.06	38	270.000	1717.28
9	125.000	605.23	24	200.000	948.15	39	275.000	1801.90
10	130.000	619.70	25	205.000	984.74	40	280.000	1892.43
11	135.000	635.70	26	210.000	1024.29	41	285.000	1989.39
12	140.000	652.00	27	215.000	1066.60	42	290.000	2093.05
13	145.000	672.03	28	220.000	1112.02	43	295.000	2203.85
14	150.000	692.43	29	225.000	1160.79	44	300.000	2322.16
15	155.000	713.79	30	230.000	1213.19	45	305.000	2448.28

TEN 60 DEG-F/HR HEATUP CURVE REG. GUIDE 1.99,REV.2 W/O MARGIN

05/14/91

COMPOSITE CURVE PLOTTED FOR HEATUP PROFILE 2 HEATUP RATE(S) (DEG.F/HR) = 60.0

IRRADIATION PERIOD = 16.000 EFP YEARS
FLAW DEPTH = (1-AOWIN)T

	INDICATED TEMPERATURE (DEG.F)	INDICATED PRESSURE (PSI)		INDICATED TEMPERATURE (DEG.F)	INDICATED PRESSURE (PSI)		INDICATED TEMPERATURE (DEG.F)	INDICATED PRESSURE (PSI)
1	85.000	~~549.04~~	16	160.000	661.34	31	235.000	1231.27
2	90.000	~~541.43~~ 535.06	17	165.000	682.64	32	240.000	1295.75
3	95.000	~~536.90~~	18	170.000	705.60	33	245.000	1364.56
4	100.000	535.06	19	175.000	730.64	34	250.000	1421.80
5	105.000	525.82	20	180.000	757.54	35	255.000	1481.70
6	110.000	528.65	21	185.000	786.73	36	260.000	1545.85
7	115.000	543.57	22	190.000	818.10	37	265.000	1614.57
8	120.000	550.10	23	195.000	851.84	38	270.000	1688.13
9	125.000	558.52	24	200.000	888.15	39	275.000	1766.82
10	130.000	56F.49	25	205.000	927.41	40	280.000	1851.24
11	135.000	580.10	26	210.000	969.47	41	285.000	1941.25
12	140.000	593.20	27	215.000	1014.70	42	290.000	2037.75
13	145.000	607.79	28	220.000	1063.22	43	295.000	2140.72
14	150.000	624.04	29	225.000	1115.36	44	300.000	2250.85
15	155.000	641.92	30	230.000	1171.30	45	305.000	2368.27

TEN COOLDOWN CURVES REG. GUIDE 1.99,REV 2 W/O MARGIN

05/14/91

THE FOLLOWING DATA WERE PLOTTED FOR COOLDOWN PROFILE 1 (STEADY-STATE COOLDOWN)

IRRADIATION PERIOD = 16.000 EFP YEARS
FLAW DEPTH = AOWIN I

	INDICATED TEMPERATURE (DEG.F)	INDICATED PRESSURE (PSI)		INDICATED TEMPERATURE (DEG.F)	INDICATED PRESSURE (PSI)		INDICATED TEMPERATURE (DEG.F)	INDICATED PRESSURE (PSI)
1	85.000	550.75						
2	90.000	557.74	16	160.000	733.02	30	230.000	1213.18
3	95.000	565.26	17	165.000	753.57	31	235.000	1269.25
4	100.000	573.34	18	170.000	775.81	32	240.000	1329.78
5	105.000	582.03	19	175.000	799.55	33	245.000	1394.60
6	110.000	591.36	20	180.000	825.23	34	250.000	1463.92
7	115.000	601.28	21	185.000	852.69	35	255.000	1538.60
8	120.000	612.08	22	190.000	882.32	36	260.000	1618.50
9	125.000	623.69	23	195.000	914.08	37	265.000	1704.06
10	130.000	636.15	24	200.000	948.15	38	270.000	1795.74
11	135.000	649.44	25	205.000	984.74	39	275.000	1894.09
12	140.000	663.86	26	210.000	1024.29	40	280.000	1999.28
13	145.000	679.37	27	215.000	1066.60	41	285.000	2111.74
14	150.000	695.86	28	220.000	1112.02	42	290.000	2232.06
15	155.000	713.79	29	225.000	1160.79	43	295.000	2360.10

TEN COOLDOWN CURVES REG. GUIDE 1.99,REV.2 W/O MARGIN

05/14/91

THE FOLLOWING DATA WERE PLOTTED FOR COOLDOWN PROFILE 2 (20 DEG-F / HR COOLDOWN)

IRRADIATION PERIOD = 16.000 EFP YEARS
FLAW DEPTH = AOWIN T

	INDICATED TEMPERATURE (DEG.F)	INDICATED PRESSURE (PSI)		INDICATED TEMPERATURE (DEG.F)	INDICATED PRESSURE (PSI)		INDICATED TEMPERATURE (DEG.F)	INDICATED PRESSURE (PSI)
1	85.000	508.95	11	135.000	613.10	20	180.000	800.74
2	90.000	516.25	12	140.000	628.42	21	185.000	830.34
3	95.000	524.13	13	145.000	644.78	22	190.000	862.05
4	100.000	532.60	14	150.000	662.53	23	195.000	896.11
5	105.000	541.74	15	155.000	681.63	24	200.000	932.70
6	110.000	551.46	16	160.000	702.03	25	205.000	972.31
7	115.000	562.07	17	165.000	724.18	26	210.000	1014.68
8	120.000	573.47	18	170.000	747.78	27	215.000	1060.24
9	125.000	585.76	19	175.000	773.41	28	220.000	1109.14
10	130.000	598.85						

TEN COOLDOWN CURVES REG. GUIDE 1.99,REV.2 W/O MARGIN

05/14/91

THE FOLLOWING DATA WERE PLOTTED FOR COOLDOWN PROFILE 3 (40 DEG-F / HR COOLDOWN)

IRRADIATION PERIOD = 16.000 EFP YEARS
FLAW DEPTH = AOWIN T

	INDICATED TEMPERATURE (DEG.F)	INDICATED PRESSURE (PSI)
1	85.000	466.40
2	90.000	474.05
3	95.000	482.33
4	100.000	491.25
5	105.000	500.81
6	110.000	511.19
7	115.000	522.42
8	120.000	534.51
9	125.000	547.45
10	130.000	561.50
11	135.000	576.67
12	140.000	592.99
13	145.000	610.49
14	150.000	629.44
15	155.000	649.74
16	160.000	671.74
17	165.000	695.29
18	170.000	720.81
19	175.000	748.12
20	180.000	777.68
21	185.000	809.37
22	190.000	843.42
23	195.000	880.29
24	200.000	919.77
25	205.000	962.28
26	210.000	1007.96
27	215.000	1057.14
28	220.000	1109.92

TEN COOLDOWN CURVES REG. GUIDE 1.99,REV.2 W/O MARGIN

05/14/91

THE FOLLOWING DATA WERE PLOTTED FOR COOLDOWN PROFILE 4 (60 DEG-F/HR COOLDOWN)

IRRADIATION PERIOD = 16.000 EFP YEARS
FLAW DEPTH = AOWIN T

	INDICATED TEMPERATURE (DEG.F)	INDICATED PRESSURE (PSI)		INDICATED TEMPERATURE (DEG.F)	INDICATED PRESSURE (PSI)		INDICATED TEMPERATURE (DEG.F)	INDICATED PRESSURE (PSI)
1	85.000	423.05	10	130.000	524.00	19	175.000	724.28
2	90.000	431.10	11	135.000	540.20	20	180.000	755.94
3	95.000	439.84	12	140.000	557.54	21	185.000	790.08
4	100.000	449.19	13	145.000	576.39	22	190.000	827.00
5	105.000	459.41	14	150.000	596.57	23	195.000	866.63
6	110.000	470.42	15	155.000	618.49	24	200.000	909.27
7	115.000	482.35	16	160.000	642.07	25	205.000	955.18
8	120.000	495.20	17	165.000	667.43	26	210.000	1004.58
9	125.000	509.02	18	170.000	694.69	27	215.000	1057.70

TEN COOLDOWN CURVES REG. GUIDE 1.99,REV.2 W/O MARGIN

05/14/91

THE FOLLOWING DATA WERE PLOTTED FOR COOLDOWN PROFILE 5 (100 DEG-F/HR COOLDOWN)

IRRADIATION PERIOD = 16.000 EFP YEARS
FLAW DEPTH = AOWIN I

	INDICATED TEMPERATURE (DEG.F)	INDICATED PRESSURE (PSI)		INDICATED TEMPERATURE (DEG.F)	INDICATED PRESSURE (PSI)		INDICATED TEMPERATURE (DEG.F)	INDICATED PRESSURE (PSI)
1	85.000	333.83	10	130.000	448.70	19	175.000	681.24
2	90.000	342.86	11	135.000	467.35	20	180.000	718.27
3	95.000	352.70	12	140.000	487.47	21	185.000	758.25
4	100.000	363.35	13	145.000	509.17	22	190.000	801.31
5	105.000	374.88	14	150.000	532.67	23	195.000	847.78
6	110.000	387.40	15	155.000	557.99	24	200.000	897.8[illegible]
7	115.000	401.00	16	160.000	585.40	25	205.000	951.78
8	120.000	415.63	17	165.000	614.92	26	210.000	1009.85
9	12[illegible].000	431.57	18	170.000	646.72			

ENCLOSURE 5

PRESSURIZED THERMAL SHOCK ASSESSMENT

SEQUOYAH NUCLEAR PLANT UNITS 1 AND 2 - PRESSURIZED THERMAL SHOCK SCREENING CRITERIA

1. PURPOSE

The purpose of this calculation is to verify that Sequoyah Nuclear Plant Units 1 and 2 reactor pressure vessels meet the pressurized thermal shock (PTS) screening criteria of 10 CFR 50.61 dated May 15, 1991.

2. INTRODUCTION

Pressurized thermal shock events are system transients in a pressurized water reactor (PWR) that can cause severe overcooling followed by immediate repressurization. Internal stresses caused by rapid cooling of the reactor vessel inside surface combined with the pressure stresses increase the potential for fracture if an initiating flaw is present in low toughness material. This material may exist in the reactor vessel beltline, adjacent to the core, where neutron radiation gradually embrittles the material during operation of the plant. The degree of embrittlement depends upon the chemical composition of the steel, especially the copper and nickel contents. The pressurized thermal shock rule (PTS rule) establishes a screening criterion which limits the amount of embrittlement of a reactor beltline beyond which the plant cannot operate without justification based on a plant specific analysis.

The toughness of reactor vessel materials is characterized by a reference temperature for nil ductility transition (RT_{NDT}), which is determined by destructive tests of material specimens. RT_{NDT} is designated as the higher of either the drop weight nil-ductility transition temperature (NDTT) or the temperature at which the material exhibits at least 50 ft-lb of impact energy and 35-mil lateral expansion (normal to the major working direction) minus $60^{o}F$.

The pressurized thermal shock rule, 10 CFR 50.61, adopted May 15, 1991, changes the procedure for establishing the limiting level of embrittlement. This screening criterion is given in terms of RT_{NDT}, but called RT_{PTS} to distinguish it from other procedures for calculating RT_{NDT}.

2. REFERENCE TEMPERATURE FOR PRESSURIZED THERMAL SHOCK

The pressurized thermal shock (PTS) screening criterion is $270^{o}F$ for plates, forgings, and axial weld materials, or $300^{o}F$ for circumferential weld materials.

From 10 CFR 50.61 dated May 15, 1991, RT_{PTS} is determined for each material in the reactor vessel beltline from:

$$RT_{PTS} = I + M + \Delta RT_{PTS} \qquad \text{(Eqn. 1)}$$

where I = the initial reference temperature (RT_{NDT}) of the unirradiated material as defined in paragraph N2331 of Section III of the ASME Boiler and Pressure Vessel Code.

M = the margin to be added to covered uncertainties in the values of initial RT_{NDT}, copper and nickel contents, fluence, and calculational procedures. In equation 1, M is 66°F for welds and 48°F for base metals if generic values of I are used, and M is 56°F for welds and 34°F for base metal if measured values of I are used.

ΔRT_{PTS} = the mean value of the adjustment in reference temperature caused by irradiation and is calculated as follows:

$$\Delta RT_{PTS} = (CF) f^{(0.28 - 0.10 \log f)} \qquad \text{(Eqn. 2)}$$

where CF(°F) = the chemistry factor, a function of copper and nickel content, given in tables 1 and 2 of 10 CFR 50.61.

f = the best estimate of neutron fluence, in units of 10^{19} n/cm^2 (E greater then 1 MeV), at the clad-base-metal interface at the location where the material in question receives the highest fluence for the period of service in question.

ff = fluence factor = $f^{(0.28 - 0.10 \log f)}$

3. FRACTURE TOUGHNESS PROPERTIES

The fracture-toughness properties of the beltline materials were determined in accordance with the NRC Regulatory Standard Review Plan (Ref. 2). The pre-irradiation fracture-toughness properties of Sequoyah Units 1 and 2 are given in Tables 3 and 4. These data are the same as presented in references 3 and 4 on heatup and cooldown limit curves.

4. CALCULATIONS

TABLE 1[a]

Sequoyah Unit 1

Material Description	$CF^{b,c}$	$\Delta RT(PTS)$ $(CF\times ff)^{d}$	Initial RT_{PTS}	Margin	RT_{PTS}	Screening Criterion Limit
Forging 05	116	157	40	34	231	270
Forging 04	95	128	73	34	235	270
	[100]	133	73	[17]	223	270
Welds	158	213	-40	56	229	300
	[173]	230	-40	[28]	218	300

a. All units in °F.

b. Chemistry Factor obtained from tables 1 and 2 of 10 CFR 50.61

c. Values in [ ] were determined from credible surveillance data as reported in heatup and cooldown limit calculations (Ref. 3, page 9).

d. Unit 1 fluence at 32 EFPY is taken as 2 times the 16 EFPY surface fluence. f_{surf} (10^{19} n/cm^2, E>1 MeV) = 1.94 at 16 EFPY (Ref. 3, page 10). ff(32 EFPY) = 1.35.

TABLE 2[a]

Sequoyah Unit 2

Material Description	$CF^{b,c}$	$\Delta RT(PTS)$ $(CF\times ff)^{d}$	Initial RT_{PTS}	Margin	RT_{PTS}	Screening Criterion Limit
Forging 05	95	108	10	34	152	270
	[83]	95	10	[17]	122	270
Forging 04	104	119	-22	34	131	270
Welds	68	76	-4	56	128	300
	[144]	164	-4	[28]	188	300

a. All units in °F.

b. Chemistry Factor obtained from tables 1 and 2 of 10 CFR 50.61

c. Values in [ ] were determined from credible surveillance data as reported in heatup and cooldown limit calculations (Ref. 4, page 9).

d. Unit 2 fluence at 32 EFPY is taken as 2 times the 16 EFPY surface fluence. f_{surf} (10^{19} n/cm^2, E>1 MeV) = 0.8644 at 16 EFPY (Ref. 4, page 10). ff(32 EFPY) = 1.14.

5. **CONCLUSIONS**

For each of the materials in the vessel beltline regions of both Sequoyah Units 1 and 2, the calculated value of the RT_{PTS} is less than the screening criteria limits. No further action is required.

6. **REFERENCES**

1. 10 CFR 50.61, "Fracture Toughness Requirements for Protection Against Pressurized Thermal Shock Events," Federal Register, Vol. 56 No. 94, May 15, 1991.

2. "Fracture Toughness Requirements," Branch Technical Position MTEB 5-2, Chapter 5.3.2 in Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants, LWR Edition, NUREG-0800, 1981.

3. WCAP-12970, "Heatup and Cooldown Limit Curves for Normal Operation," Sequoyah Nuclear Plant Unit 1, June 1991.

4. WCAP-12971, "Heatup and Cooldown Limit Curves for Normal Operation," Sequoyah Nuclear Plant Unit 2, June 1991.

5. Code of Federal Regulations, 10 CFR 50, Appendix G, "Fracture Toughness Requirements," U.S. Nuclear Regulatory Commission, Washington D.C., Federal Register, Vol. 48 No. 104, May 27, 1983.

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

Prepared by Paul V. Guthrie, Jr. \ 8/1/91
Reviewed by Patrick A. Walker \ 8/1/91
Approved by (for) [signature] \ 8-1-91

TABLE 3

SEQUOYAH UNIT 1 REACTOR VESSEL TOUGHNESS TABLE (Unirradiated)

Material Description	CU (%)	NI (%)	Initial RT_{NDT} (a) (°F)
Intermediate Shell (Forging 05)	0.15	0.86	+40
Lower Shell (Forging 04)	0.13	0.76	+73
Welds	0.33	0.17	-40

a. The initial RT_{NDT} values for the forgings are measured values.

TABLE 4

SEQUOYAH UNIT 2 REACTOR VESSEL TOUGHNESS TABLE (Unirradiated)

Material Description	CU (%)	NI (%)	Initial RT_{NDT} (a) (°F)
Intermediate Shell (Forging 05)	0.13	0.74	+10
Lower Shell (Forging 04)	0.14	0.76	-22
Welds	0.13	0.11	-4

a. The initial RT_{NDT} values for the forgings are measured values.