HEATUP AND COOLDOWN LIMIT CURVES
FOR THE
BRAIDWOOD UNIT 1 REACTOR VESSEL
(CAPSULE U)

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HEATUP AND COOLDOWN LIMIT CURVES FOR NORMAL OPERATION

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 Braidwood Unit 1 of the reactor vessels 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:

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

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:

$$C\, K_{IM} + K_{IT} \leq K_{IR} \qquad (2)$$

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 inside to the outside, and the pressure limit must at all times be based on analysis 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 -10°F occurs in the vessel flange of Braidwood Unit 1, so the minimum allowable temperature of this region is 110°F. These limits are shown in figures 1 and 2 whenever applicable.

4. HEATUP AND COOLDOWN LIMIT CURVES

Limit curves for normal heatup and cooldown of the primary Reactor Coolant System have been calculated using the methods discussed in section 3.0. Figure 1 is the heatup curve for 100°F/hr and applicable for the first 32 EFPY with margins for possible instrumentation errors. Figure 2 is the cooldown curve up to 100°F/hr and applicable for the first 32 EFPY with margins for possible instrumentation errors.

Allowable combinations of temperature and pressure for specific temperature change rates are below and to the right of the limit lines shown in figures 1 and 2. This is in addition to other criteria which must be met before the reactor is made critical.

The leak limit curve shown in figure 1 represents 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.

Figures 1 and 2 define limits for ensuring prevention of nonductile failure for the Braidwood Unit 1 Primary Reactor Coolant System.

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:

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

Initial RT_{NDT} is the reference temperature for the unirradiated material as defined in paragraph NB-2331 of Section III of the ASME Boiler and Pressure Vessel Code. 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:

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

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.

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

where x (in inches) is the depth into the vessel wall measured from the vessel inner (wetted) surface. 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. All materials in the beltline region of Braidwood Unit 1 were considered for the limiting material. RT_{NDT} at 1/4T and 3/4T are summarized in table 2. From table 2, it can be seen that the limiting material is intermediate to lower shell weld for heatup and cooldown curves applicable up to 32 EFPY. A sample calculation for RT_{NDT} is shown in Table 3.

TABLE 1

BRAIDWOOD UNIT 1 REACTOR VESSEL TOUGHNESS TABLE (Unirradiated)

Component	Cu (%)	Ni (%)	I [(a)] (°F)
Closure head flange (b)	-	-	-20
Vessel flange (b)	-	-	-10
Intermediate Shell Forging 49D383/49C344-1-1	.05	.73	-30
Lower Shell Forging 49D867/49C813-1-1	.03	.73	-20
Circumferential Weld WF562	.04 (c)	.67 (c)	40

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

b. To be used for considering flange requirements for heatup/cooldown curves[4]

c. These values from Weld Certification Test Report WF562 are higher than the values of 0.03% Cu and 0.67% Ni used in the PTS submittal [5]

TABLE 2

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

	32 EFPY RT_{NDT} AT	
Component	1/4T (°F)	3/4T (°F)
Intermediate Shell	40	25
Lower Shell	27	15
Intermediate to Lower Shell Weld	159*	135*

* These RT_{NDT} numbers used to generate heatup and cooldown curves applicable up to 32 EFPY

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TABLE 3

CALCULATION OF ADJUSTED REFERENCE TEMPERATURES FOR LIMITING BRAIDWOOD UNIT 1 REACTOR VESSEL MATERIAL - CIRCUMFERENTIAL WELD

	Regulatory Guide 1.99 - Revision 2	
	32 EFPY	
Parameter	1/4 T	3/4 T
Chemistry Factor, CF (°F)	54.00	54.00
Fluence, f (10^{19} n/cm^2)(a)	1.82	.66
Fluence Factor, ff	1.16	.88

****

ΔRT_{NDT} = CF x ff (°F)	62.6	47.5
Initial RT_{NDT}, I (°F)	40.0	40.0
Margin, M (°F) (b)	56	47.5

****

Revision 2 to Regulatory Guide 1.99

Adjusted Reference Temperature, ART = Initial RT_{NDT} + ΔRT_{NDT} + Margin	159	135

****

(a) Fluence, f, is based upon f_{surf} (10^{19} n/cm^2, E>1 Mev) = 3.03 at 32 EFPY. The Braidwood Unit 1 reactor vessel wall thickness is 8.5 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}, ($\sigma\Delta$) is 28°F for the weld, except that $\sigma\Delta$ need not exceed 0.50 times the mean value of ΔRT_{NDT}.

MATERIAL PROPERTY BASIS

CONTROLLING MATERIAL: CIRCUMFERENTIAL WELD

INITIAL RT_{NDT}: 40°F

ART AFTER 32 EFPY: 1/4T, 159°F

3/4T, 135°F

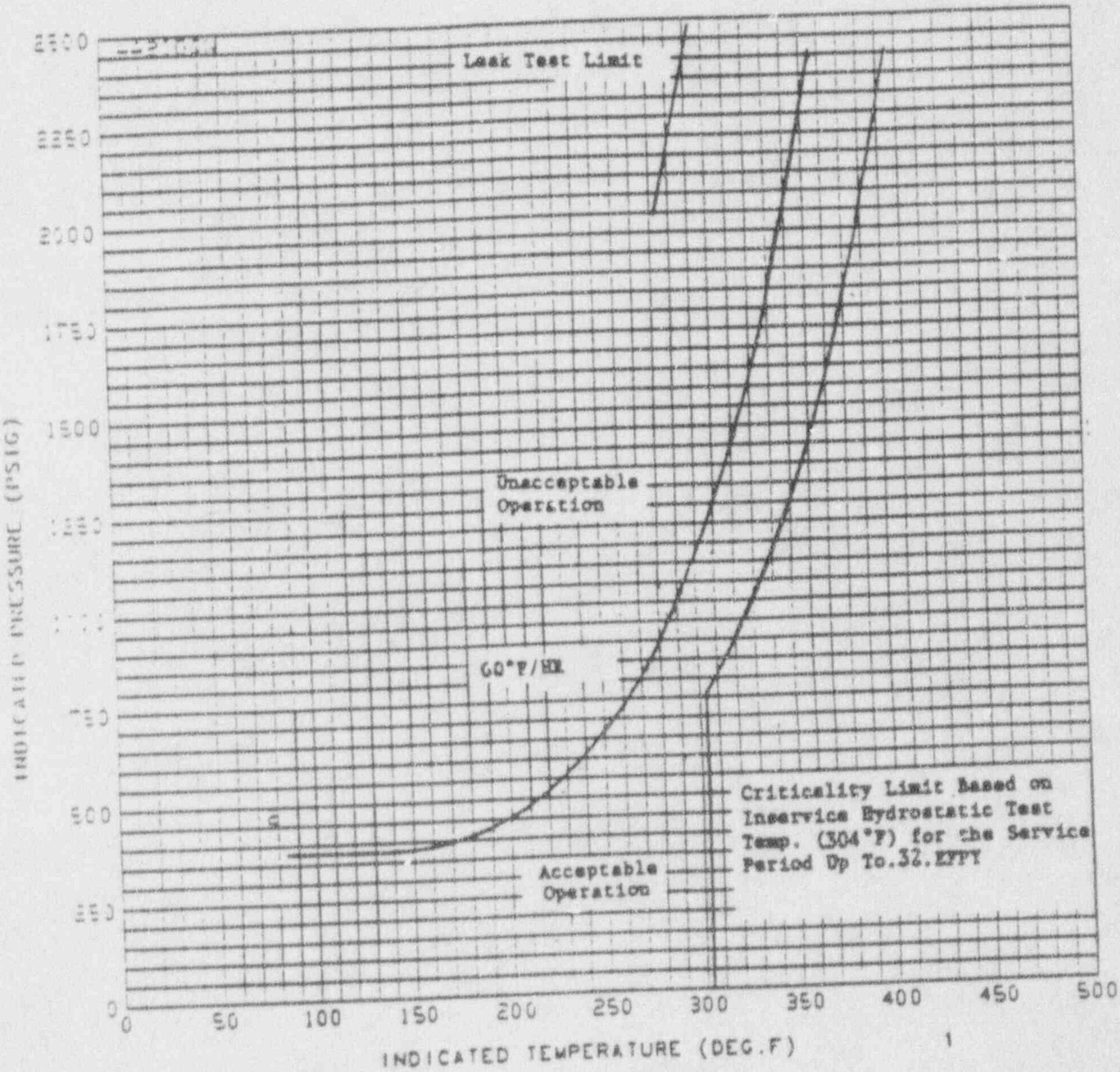

Figure 1. Braidwood Unit 1 Reactor Coolant System Heatup Limitations (Heat up rate up to 100°F/hr) Applicable for the First 32 EFPY (With Margins 10°F and 60 psig For Instrumentation Errors)

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MATERIAL PROPERTY BASIS

CONTROLLING MATERIAL: CIRCUMFERENTIAL WELD

INITIAL RT_{NDT}: 40°F

ART AFTER 32 EFPY: 1/4T, 159°F

3/4T, 135°F

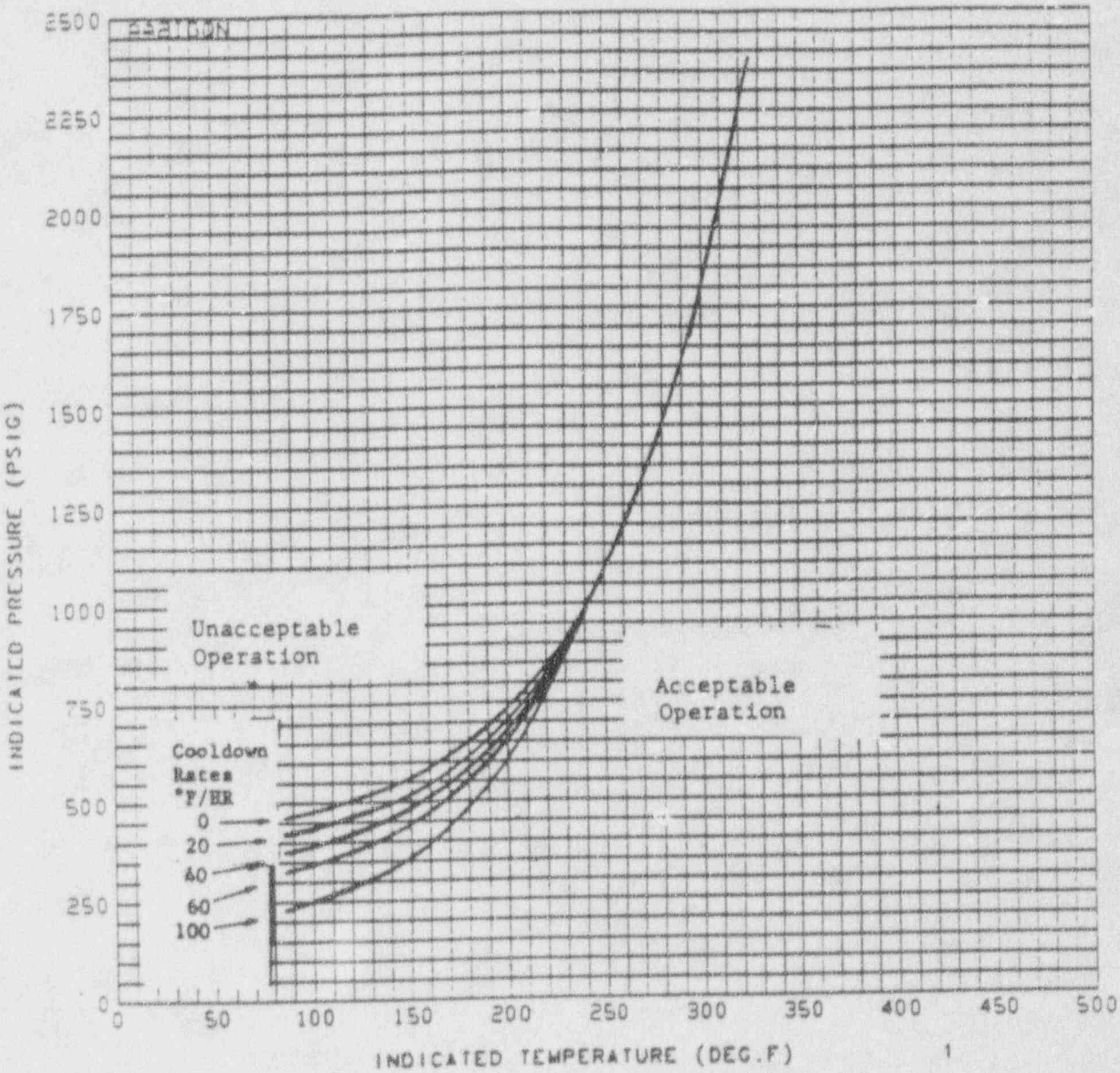

Figure 2. Braidwood Unit 1 Reactor Coolant System Cooldown (Cooldown rates up to 100°F/hr) Limitations Applicable for the First 32 EFPY (With Margins 10°F and 60 psig For Instrumentation Errors)

7. 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.

5 PTS Submittal for Braidwood Unit 1 Reactor Vessel (Section 5.3, Braidwood FSAR, Amendment 47, Commonwealth Edison Company, April 1986)

ATTACHMENT A

DATA POINTS FOR HEATUP AND COOLDOWN CURVES

(With Margins 10°F and 60 psig for Instrumentation Errors)

CCE COOLDOWN CURVES REG GUIDE 1 99, REV 2 WITH MARGIN

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

IRRADIATION PERIOD = 32.000 EFP YEARS
FLAW DEPTH = ADWIN 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	461.32	18	170.000	613.78	34	250.000	1082.12
2	90.000	466.05	19	175.000	629.96	35	255.000	1133.14
3	95.000	471.14	20	180.000	647.26	36	260.000	1187.91
4	100.000	476.61	21	185.000	665.99	37	265.000	1246.63
5	105.000	482.49	22	190.000	685.94	38	270.000	1309.50
6	110.000	488.71	23	195.000	707.60	39	275.000	1377.38
7	115.000	495.51	24	200.000	730.79	40	280.000	1449.90
8	120.000	502.82	25	205.000	755.68	41	285.000	1527.75
9	125.000	510.68	26	210.000	782.33	42	290.000	1611.13
10	130.000	519.13	27	215.000	811.23	43	295.000	1700.50
11	135.000	528.21	28	220.000	842.06	44	300.000	1796.26
12	140.000	537.85	29	225.000	875.32	45	305.000	1898.70
13	145.000	548.35	30	230.000	911.02	46	310.000	2008.37
14	150.000	559.64	31	235.000	949.34	47	315.000	2125.57
15	155.000	571.78	32	240.000	990.46	48	320.000	2260.92
16	160.000	584.68	33	245.000	1034.65	49	325.000	2384.70
17	165.000	598.70						

CCE COOLDOWN CURVES REG. GUIDE 1.99, REV 2 WITH MARGIN 11/02/90

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

IRRADIATION PERIOD = 32.000 EFP YEARS
FLAW DEPTH = 1/4T

	INDICATED TEMPERATURE (DEG.F)	INDICATED PRESSURE (PSI)		INDICATED TEMPERATURE (DEG.F)	INDICATED PRESSURE (PSI)		INDICATED TEMPERATURE (DEG.F)	INDICATED PRESSURE (PSI)
1	85.000	417.06	13	145.000	508.26	24	200.000	702.87
2	90.000	421.92	14	150.000	520.20	25	205.000	729.67
3	95.000	427.18	15	155.000	533.07	26	210.000	758.37
4	100.000	432.85	16	160.000	546.81	27	215.000	789.21
5	105.000	438.88	17	165.000	561.74	28	220.000	822.53
6	110.000	445.47	18	170.000	577.79	29	225.000	858.27
7	115.000	452.58	19	175.000	594.94	30	230.000	896.65
8	120.000	460.23	20	180.000	613.53	31	235.000	937.91
9	125.000	468.49	21	185.000	633.52	32	240.000	982.28
10	130.000	477.37	22	190.000	654.93	33	245.000	1029.95
11	135.000	486.84	23	195.000	678.10	34	250.000	1081.21
12	140.000	497.14						

11/02/90

CCE COOLDOWN CURVES REG GUIDE 1.99,REV 2 WITH MARGIN

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

IRRADIATION PERIOD = 32.000 EFP YEARS
FLAW DEPTH = AOWIN T

	INDICATED TEMPERATURE (DEG.F)	INDICATED PRESSURE (PSI)
1	85.000	371.82
2	90.000	376.85
3	95.000	382.33
4	100.000	388.15
5	105.000	394.54
6	110.000	401.43
7	115.000	408.89
8	120.000	416.93
9	125.000	425.63
10	130.000	435.00
11	135.000	445.06
12	140.000	455.96
13	145.000	467.75
14	150.000	480.45
15	155.000	494.05
16	160.000	508.81
17	165.000	524.74
18	170.000	541.7[illegible]
19	175.000	560.26
20	160.000	580.15
21	185.000	601.48
22	190.000	624.57
23	195.000	649.32
24	200.000	676.08
25	205.000	704.80
26	210.000	735.64
27	215.000	769.07
28	220.000	804.87
28	225.000	843.39
30	230.000	884.[illegible]9
31	235.000	929.38
32	240.000	977.31
33	245.000	1028.92

11/02/90

CCE COOLDOWN CURVES REG. GUIDE 1.99,REV 2 WITH MARGIN

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

IRRADIATION PERIOD = 32.000 EFP YEARS
FLAW DEPTH = AOWIN T

	INDICATED TEMPERATURE (DEG.F)	INDICATED PRESSURE (PSI)
1	85.000	325.55
2	90.000	330.78
3	95.000	336.49
4	100.000	342.65
5	105.000	349.35
6	110.000	356.53
7	115.000	364.40
8	120.000	372.88
9	125.000	382.08
10	130.000	391.94
11	135.000	402.70
12	140.000	414.29
13	145.000	426.85
14	150.000	440.39
15	155.000	454.94
16	160.000	470.71
17	165.000	487.66
18	170.000	506.02
19	175.000	525.87
20	180.000	547.11
21	185.000	570.18
22	190.000	594.88
23	195.000	621.69
24	200.000	650.39
25	205.000	681.47
26	210.000	714.85
27	215.000	750.78
28	220.000	789.43
29	225.000	831.09
30	230.000	875.91
31	235.000	924.19
32	240.000	976.14
33	245.000	1032.09

CCE COOLDOWN CURVES REG GUIDE 1 99, REV 2 WITH MARGIN

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

IRRADIATION PERIOD = 32.000 EFP YEARS
FLAW DEPTH = ADWIN 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	229.44	12	140.000	329.63	23	195.000	570.35
2	90.000	235.37	13	145.000	344.03	24	200.000	603.98
3	95.000	241.67	14	150.000	358.52	25	205.000	640.31
4	100.000	248.50	15	155.000	376.39	26	210.000	679.60
5	105.000	255.97	16	160.000	394.52	27	215.000	721.86
6	110.000	264.07	17	165.000	414.25	28	220.000	767.40
7	115.000	272.88	18	170.000	435.54	29	225.000	816.48
8	120.000	282.45	19	175.000	458.50	30	230.000	869.32
9	125.000	292.88	20	180.000	483.36	31	235.000	926.10
10	130.000	304.16	21	185.000	510.14	32	240.000	987.46
11	135.000	316.38	22	190.000	538.01			

CCE60 F/HR HEATUP CURVE REG GUIDE 1 99,REV 2 WITH MARGIN 11/02/90

THE FOLLOWING DATA WERE CALCULATEDFOR THE INSERVICE HYDROSTATIC LEAK TEST

MINIMUM INSERVICE LEAK TEST TEMPERATURE (32.000 EFPY)

PRESSURE (PSI)	TEMPERATURE (DEG F)
2000	283
2485	304

PRESSURE (PSI)	PRESSURE STRESS (PSI)	1 5 KIM (PSI SQ RT IN)
2000	21694	91656
2485	26802	114239

11/02/90

CCE60 F/HR HEATUP CURVE REG GUIDE 1 99, REV 2 WITH MARGIN

COMPOSITE CURVE PLOTTED FOR HEATUP PROFILE 2 HEATUP RATE(S) (DEG F/HR) = 100.0

IRRADIATION PERIOD = 32.000 EFP YEARS
FLAW DEPTH = (1-ADWIN)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	~~464.82~~	20	180.000	412.14	39	275.000	850.82
2	90.000	~~466.60~~	21	185.000	421.33	40	280.000	896.31
3	95.000	~~441.03~~	22	190.000	431.66	41	285.000	945.19
4	100.000	~~426.82~~	23	195.000	443.10	42	290.000	997.64
5	105.000	~~414.85~~ 378.44	24	200.000	455.86	43	295.000	1053.93
6	110.000	~~401.30~~	25	205.000	469.92	44	300.000	1114.35
7	115.000	~~398.62~~	26	210.000	485.32	45	305.000	1179.04
8	120.000	~~389.93~~	27	215.000	502.04	46	310.000	1248.48
9	125.000	~~388.07~~	28	220.000	520.37	47	315.000	1322.82
10	130.000	~~381.54~~	29	225.000	540.16	48	320.000	1402.45
11	135.000	~~378.36~~	30	230.000	561.75	49	325.000	1487.71
12	140.000	378.44	31	235.000	584.98	50	330.000	1578.97
13	145.000	378.77	32	240.000	610.25	51	335.000	1676.50
14	150.000	380.23	33	245.000	637.38	52	340.000	1780.78
15	155.000	382.85	34	250.000	666.82	53	345.000	1892.26
16	160.000	386.46	35	255.000	698.37	54	350.000	2011.19
17	165.000	391.26	36	260.000	732.33	55	355.000	2138.00
18	170.000	397.12	37	265.000	769.10	56	360.000	2273.07
19	175.000	404.09	38	270.000	808.50	57	365.000	2417.06

June 8,1992
In reply refer to
CHRON No.

187340

To: K.L. Kofron
Braidwood Station Manager

Subject: Braidwood Unit 1
Heatup and Cooldown Curves

Reference: 1. Westinghouse letter CCE-90-317 dated Nov.19, 1990
2. NED letter from E.D. Swartz to K.L. Kofron dated Nov.27, 1990
3. Westinghouse letter CCE-92-193 dated June 2, 1992

As a result of the NRC issuing revision 2 to Regulatory Guide 1.99, the current Tech Spec heatup-cooldown curves for Braidwood Unit 1 were determined by the Nuclear Engineering Department/Westinghouse to be applicable to 4.5 EFPY. Revised heatup-cooldown curves, applicable to 32 EFPY, were therefore prepared and submitted by NED/Westinghouse to Braidwood Station per references 1 & 2.

The Nuclear Engineering Department-PWR Systems Design Group, had therefore requested Westinghouse to reanalyze the Unit 1 Low Temperature Overpressure Protection Setpoints (LTOPS) as a result of the necessity to revise the heatup-cooldown curves. Westinghouse has recently completed this effort (Ref. 3) and has provided new LTOP setpoints to be submitted together with the revised heatup-cooldown curves for Tech Spec revision. NED has reviewed this LTOP analysis and has no comments.

Should you have any question or comments, please call Bob Waninski at x-7387, Downers Grove.

R.E. Waninski 6/8/92
R.E. Waninski
PWR System Design Engineer

F.G. Lentine 6/12/92
E.D. Swartz
for PWR System Design Supervisor

REW/
Attachment
cc: ~~L.S. Dworakowski~~ D.M. Lauzauiz
T.W. Simpkin
P.J. Wicyk
NEDCC

See also NTS items starting with: 456-130-89.4.9.1-0100 through 10

Westinghouse Electric Corporation
Energy Systems
Box 355
Pittsburgh Pennsylvania 15230-0355

CCE-92-193
June 2, 1992

Mr. Robert E. Waninski
Commonwealth Edison Company
1400 Opus Place, Suite 500
Downers Grove, Illinois 60515

Commonwealth Edison Company
Braidwood Unit 1
LTOPS Setpoint Reanalysis

Dear Mr. Waninski:

Attached for your information are the following documents related to the revision of the pressure-temperature limits resulting from the analysis of capsule U from the Braidwood Unit 1 reactor vessel:

1. A report documenting the development of the Lower Temperature Overpressure Protection System (LTOPS) setpoints required to meet the revised pressure-temperature limits.

2. A markup of the RCS Cold Overpressure Mitigation System section of the PLS document showing the Braidwood Unit 1 setpoints.

3. Replacement figures for the Braidwood Units 1 and 2 Technical Specifications sections 3.4.9.1 and 3.4.9.3. No text revisions are required.

Westinghouse has instituted a customer feedback program on engineering efforts in order to enable us to track our performance against an objective of providing the highest quality products and services. As a valued customer, your participation in this survey is most important to us. Your completion and return of the enclosed quality survey is appreciated.

If you have any questions, please do not hesitate to contact me.

Sincerely yours,

WESTINGHOUSE ELECTRIC CORPORATION

Janet M. Cuniciello for

B. S. Humphries, Manager
Commonwealth Edison Projects
Domestic Customer Projects

JMB/cem
Attachment

1218d:50:JMB:06/02/1992-1

cc: S. Bishop
D. Skoza
G. Masters

1218d:50:JMB:06/02/1992-2

Low Temperature Overpressure Protection System

Capsule U Setpoint Development

Commonwealth Edison Company

Braidwood Unit 1

1.0 Introduction

The setpoints provided for the Low Temperature Overpressure Protection System (LTOPS) are selected such that the pressure peaks resulting from design basis overpressure events are limited to values less than those specified by Appendix G of 10 CFR 50. Appendix G provides the fracture toughness requirements for reactor vessels under specified operating conditions, and USNRC Regulator Guide 1.99, "Radiation Embrittlement of Reactor Vessel Materials" specifies the procedure acceptable to the NRC staff for calculating the pressure limits required by Appendix G. Revision 2 of this regulatory guide is currently applicable.

The analysis of capsule U, obtained from the Braidwood Unit 1 reactor vessel, resulted in a revision to the current pressure-temperature limits. As a result of these new limits, Westinghouse was instructed by Commonwealth Edison Co. to develop new LTOPS setpoints.

Setpoints for the Braidwood units LTOPS were initially generated in 1982, again in 1985 to implement the results of the LTOPS algorithm design review which removed the pressure instrument uncertainty and imposed an 800 psig PORV piping limit, and in 1989 to update the setpoints for revised Appendix G limits. Both the 1985 and the 1989 effort used the mass and heat injection over-pressures developed in 1982. Commonwealth Edison has implemented no plant design changes that would require their recalculation, so that the reported over-pressures continue to be applicable.

2.0 Summary of Results

The setpoint program specified in Table 1, and shown graphically in Figure 1, is based on a steady-state pressure-temperature limit corresponding to a reactor vessel exposure of 32 Effective Full Power Years (EFPY). In addition, the setpoint function accounts for a 50 °F thermal transport effect, a 27 °F temperature streaming and instrumentation uncertainty, and the 800 psig PORV piping limit. Consistant with current Westinghouse practice, nominal pressure values are assumed; i.e., pressure instrument uncertainty is not included in the analysis.

The setpoints have been selected to prevent opening both pressurizer PORVs at the same time, while providing reactor coolant pump No. 1 seal protection. Simultaneous opening of both PORVs would result in a large underpressure that would place the RCP No. 1 seal at risk. Setpoints could not be selected above the residual heat removal system (RHRS) safety valve setpressure value of 495 psig (450 psig plus 10% accumulation).

Table 1 Braidwood Unit 1 (CCE) LTOPS Setpoints

Water Temperature (°F)	P_{set1} (psig)	P_{set2} (psig)
70.0	455.0	490.0
150.0	455.0	490.0
180.0	485.0	515.0
200.0	515.0	545.0
220.0	550.0	590.0
240.0	600.0	640.0
280.0	680.0	740.0
320.0	650.0	725.0
450.0	2350.0	2350.0

2.1 LTOPS Enable/Disable Temperature

The figure also shows the Westinghouse recommended LTOPS enable (during cooldown) or disable (during heatup) temperature. The recommendation is based on the intersection of the pressurizer PORVs normal operational setpoint of 2335 psig and the pressure-temperature limit specified by Appendix G. This provides pressure relief capability throughout the full range of temperatures spanned by heatup and cooldown operations. For Braidwood Unit 1, at 32 EFPY, the intersection (hence, the enable temperature recommendation) is at approximately 310 °F. Since the pressure-temperature limits have been calculated for a reactor vessel exposure of 32 EFPY, the enable temperature recommendation is conservative throughout the life of the plant, assuming the continued applicability of the current algorithm used to generate the pressure-temperature limits.

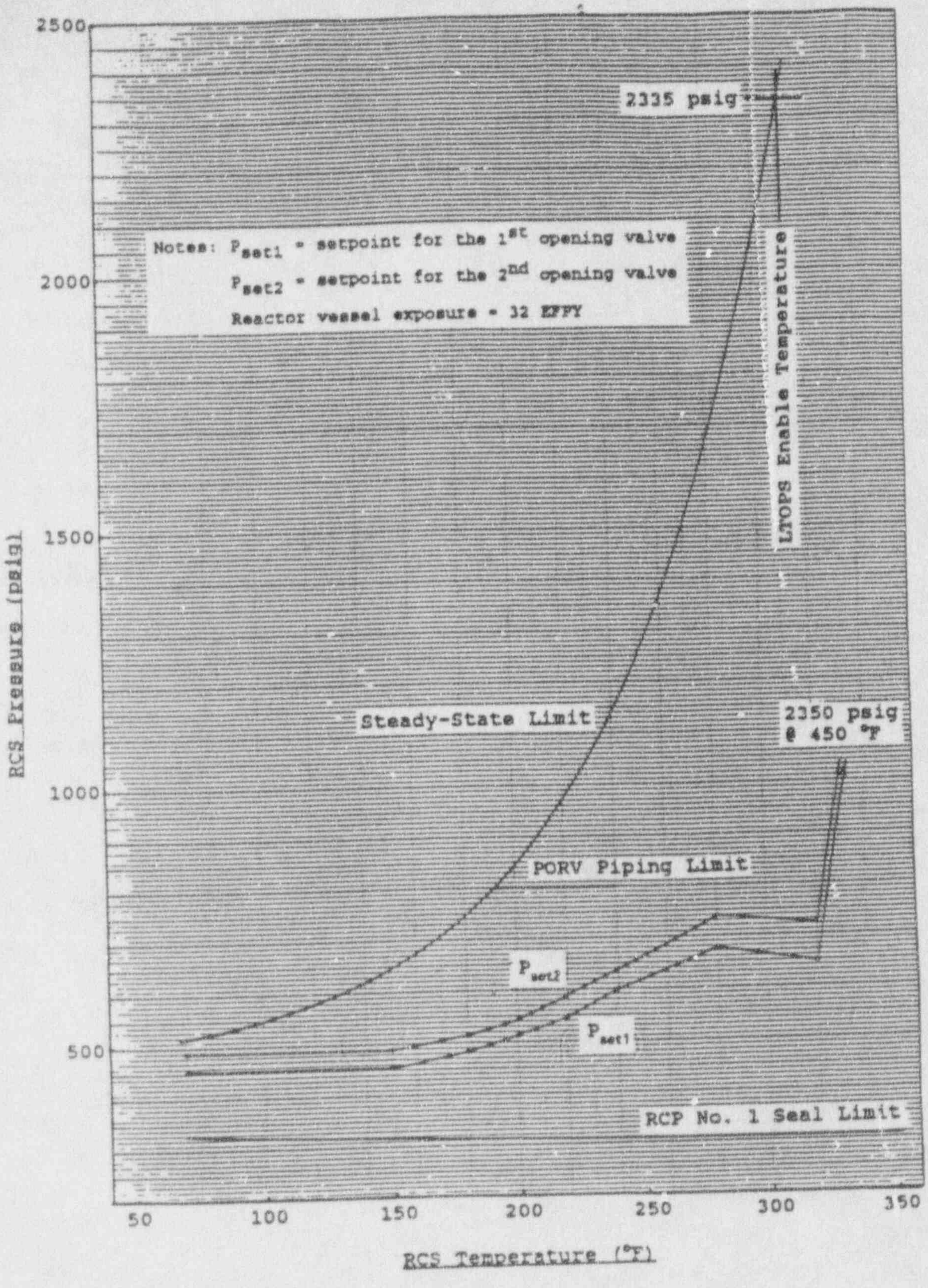

Figure 1 Braidwood Unit 1 LTOPS Setpoint Program

3.0 Analytical Basis

The standard initial condition assumed for a low temperature overpressure analysis is that the reactor coolant system is water solid, and that the transient results from either a mass injection or a heat input event. The mass injection transient is based on the operation of a single centrifugal charging pump, and is initiated by a spurious loss of letdown concurrent with a failure of the charging flow controls to the maximum flow allowed by the reactor coolant system pressure. The heat injection event results from the start of a reactor coolant pump assuming that the shell side of the steam generator is 50 °F warmer than the water contained in the reactor coolant system. This represents the maximum temperature asymmetry that could develop during a cooldown maneuver, following the shutdown of the reactor coolant pump, and the continued cooling by means of the residual heat removal system. Isothermal conditions are assumed to exist between the primary and secondary sides of the steam generators, so that when an RCP is started, the steam generator's secondary side acts as a heat source for the NSSS. The setpoint selection is based on the most restrictive of either the mass injection or the heat input cases, without credit for the residual heat removal system's safety valves.

3.1 Pressure Instrument Uncertainty

Revision 2 of USNRC regulatory guide 1.99 generally served to reduce the available operating margin between the Appendix G limits and the minimum pressure required to start the reactor coolant pumps, thus worsening the inevitable decrease in margin as reactor vessel exposure increases. For a number of older U.S. plants, this has the potential of precluding reactor coolant pump starts until some relatively high RCS temperature.

In order to restore some of this lost margin, Westinghouse took the position, following a 1985 design review of the LTOPS setpoint algorithm, that the use of nominal pressure values is acceptable, i.e., pressure instrument uncertainty is not included in the setpoint development. The justification of this position is discussed in the following paragraphs.

- The pressure-temperature limits are developed from an algorithm that includes an unusual degree of conservatism; perhaps a factor of two or more. Exceeding these limits by the pressure instrument uncertainty will not significantly increase the probability of brittle vessel failure and is not viewed by Westinghouse as a serious violation. A number of overpressure events experienced in the past have exceeded the limits by several times the pressure instrument uncertainty without adversely effecting the integrety of the pressure vessel.

- The LTOPS is considered to be a mitigation system, rather than a protection system, and is viewed as supplemental to the residual heat removal system safety valves.

- The analysis assumes a failure of the RHR/RCS isolation valve auto-closure interlock, resulting in an instantaneous closure of the valve. Simultaneous to the valve closure, a failure of the charging control system increases charging flow to the maximum allowed by the RCS pressure, thus creating a worst case mismatch between charging and letdown. In reality, the isolation valve would take several seconds to close, and it is highly improbable that a charging control systems failure would occur at the same time. The margin that these assumptions provide the calculation should be sufficient to compensate the nonconservatism of neglecting the pressure instrument uncertainty.

Because of the conservatism inherent in the calculation of both the pressure-temperature limits and the LTOPS setpoints, the additional conservatism of pressure instrument uncertainty is not believed to be required. If an instrument develops a calibration error, or drifts out of alignment in the most unfavorable direction, a limit might be exceeded. However, the deviation from the nominal limits will be, at most, the instrument uncertainty. As discussed above, exceeding the Appendix G limit by the pressure instrument uncertainty is not considered serious by Westinghouse.

3.2 Pressure-Temperature Limits

Setpoint selection is based on preventing overpressure transients from exceeding the steady-state pressure-temperature limit. The steady-state limit provides the greatest operational flexibility, and has been accepted by the NRC with the justification that most overpressure transients occur during isothermal conditions.

Westinghouse practice, for all LTOPS analyses is to select setpoints such that the peak overpressure will not exceed the most limiting of either Appendix G or 800 psig. The 800 psig limit results from an analysis of water hammer effects on relief valve piping for certain classes of rapidly opening valves (e.g., solenoid operated valves) under water solid conditions. These valves have typical stroke times of two seconds or less, and, because of their characteristic curves (C_v vs. stem position), are effectively full open or full closed in a few tenths of a second; thus setting the stage for a water hammer.

The flow through air operated valves is much less sensitive to stem position in the early stages of the opening stroke or in the latter stages of the closing stroke. This, in addition to the relatively slow opening times of air operated valves, will much reduce water hammer forces, if not effectively eliminate them. An evaluation of water hammer forces on the piping of air operated valves has not been performed by Westinghouse. The practice has been to assume the conservative position of taking the worst case results (the solenoid valve analysis) and applying them to all LTOPS setpoint evaluations, regardless of the type of relief valve designed into the plant.

Precautions, Limitations, and Setpoints Document Markup

RCS Cold Overpressure Mitigation System Section (Page 31)

Capsule U LTOPS Setpoint Revisions

Commonwealth Edison Company

Braidwood Unit 1

4. RCS Cold Overpressure Mitigation System

a. Power Operated Relief Valve
Cold Overpressure Actuation

1) Comparators (PB-406C, PB-407C) PCV-456, PCV-455A — 0 psi to open, -20 psi to close

2) Pressure Limit Function Generator (TY-413P, TY-413M) PCV-456, PCV-455A — See Table below.

PCV 456 (TY-413P)		PCV 455A (TY-413M)	
Auctioneered Low RCS Temperature (°F)	RCS Pressure (psig)	Auctioneered Low RCS Temperature (°F)	RCS Pressure (psig)
70	~~515~~ 490	70	~~470~~ 455
~~100~~ 150	~~515~~ 490	~~100~~ 150	~~470~~ 455
~~160~~ 180	~~530~~ 515	~~200~~ 180	~~465~~ 465
~~180~~ 200	~~540~~ 545	~~220~~ 200	~~460~~ 515
220	~~525~~ 590	~~235~~ 220	~~460~~ 550
~~250~~ 240	~~600~~ 640	~~270~~ 240	~~530~~ 600
~~270~~ 280	~~700~~ 740	280	~~600~~ 680
~~300~~ 320	~~700~~ 725	~~300~~ 320	~~600~~ 650
450	2350	450	2350

* b. Low Temperature Alarm for COMS Arming (TB-413K, TB-413L) — 350°F

c. Overpressure Alarms (PB-406D, PB-407D) — -20 psi(3)

*The COMS arming temperature is conservative with respect to the value recommended in the setpoint report (310°F). The arming temperature can be changed to the recommended value or left as specified above.

Technical Specification Replacement Figures

Sections 3.4.9.1 and 3.4.9.3

Capsule U LTOPS Setpoint Revisions

Commonwealth Edison Company

Braidwood Unit 1

Technical Specification LCO 3.4.9.1:

Attached Figures 1 and 2 replace Technical Specification Figures 3.4-2a and 3.4-3a, respectively.

Technical Specification LCO 3.4.9.3:

Attached Figure 3 replaces Technical Specification Figure 3.4-4. It is suggested that Figure 3.4-4a be deleted from the technical specifications. The Westinghouse version of the LCO (Amendment 33) makes no reference to this figure.

The replacement figure is applicable to the lowest reading RTD (consistant with the current tech. spec. figure), and includes a 27 °F correction for temperature streaming and instrument uncertainty. The setpoints were selected to prevent overpressures from exceeding the 800 psig PORV piping limit, thus the setpoint reduction after about 227 °F reflects the increasingly large overpressures with temperature due to heat injection events.

MATERIAL PROPERTY BASIS

CONTROLLING MATERIAL: CIRCUMFERENTIAL WELD

INITIAL RT_{NDT}: 40°F

ART AFTER 32 EFPY: 1/4T, 159°F

3/4T, 135°F

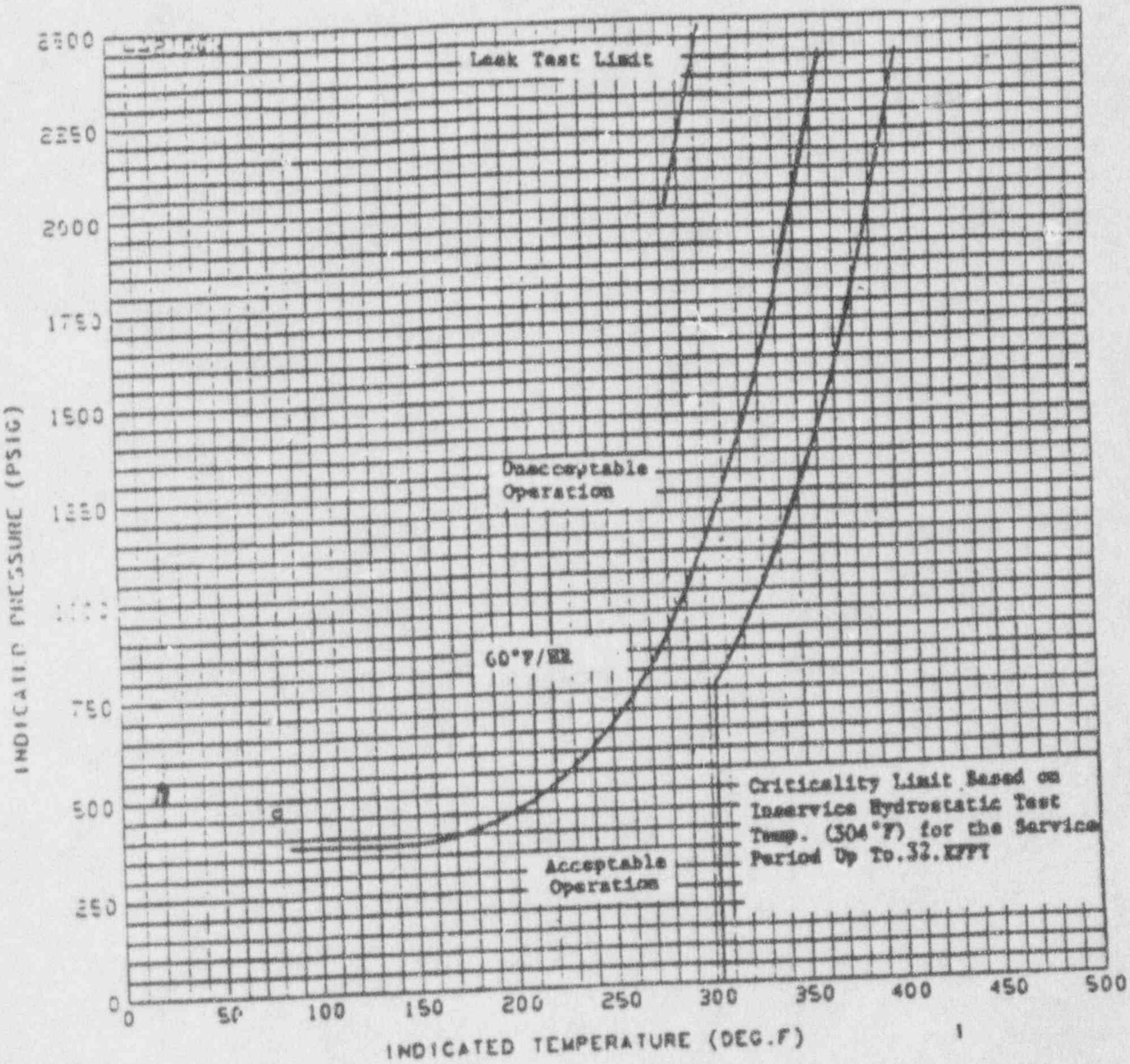

Figure 1. Braidwood Unit 1 Reactor Coolant System Heatup Limitations (Heat up rate up to 100°F/hr) Applicable for the First 32 EFPY (With Margins 10°F and 60 psig For Instrumentation Errors)

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MATERIAL PROPERTY BASIS

CONTROLLING MATERIAL: CIRCUMFERENTIAL WELD

INITIAL RT_{NDT}: 40°F

ART AFTER 32 EFPY: 1/4T, 159°F

3/4T, 135°F

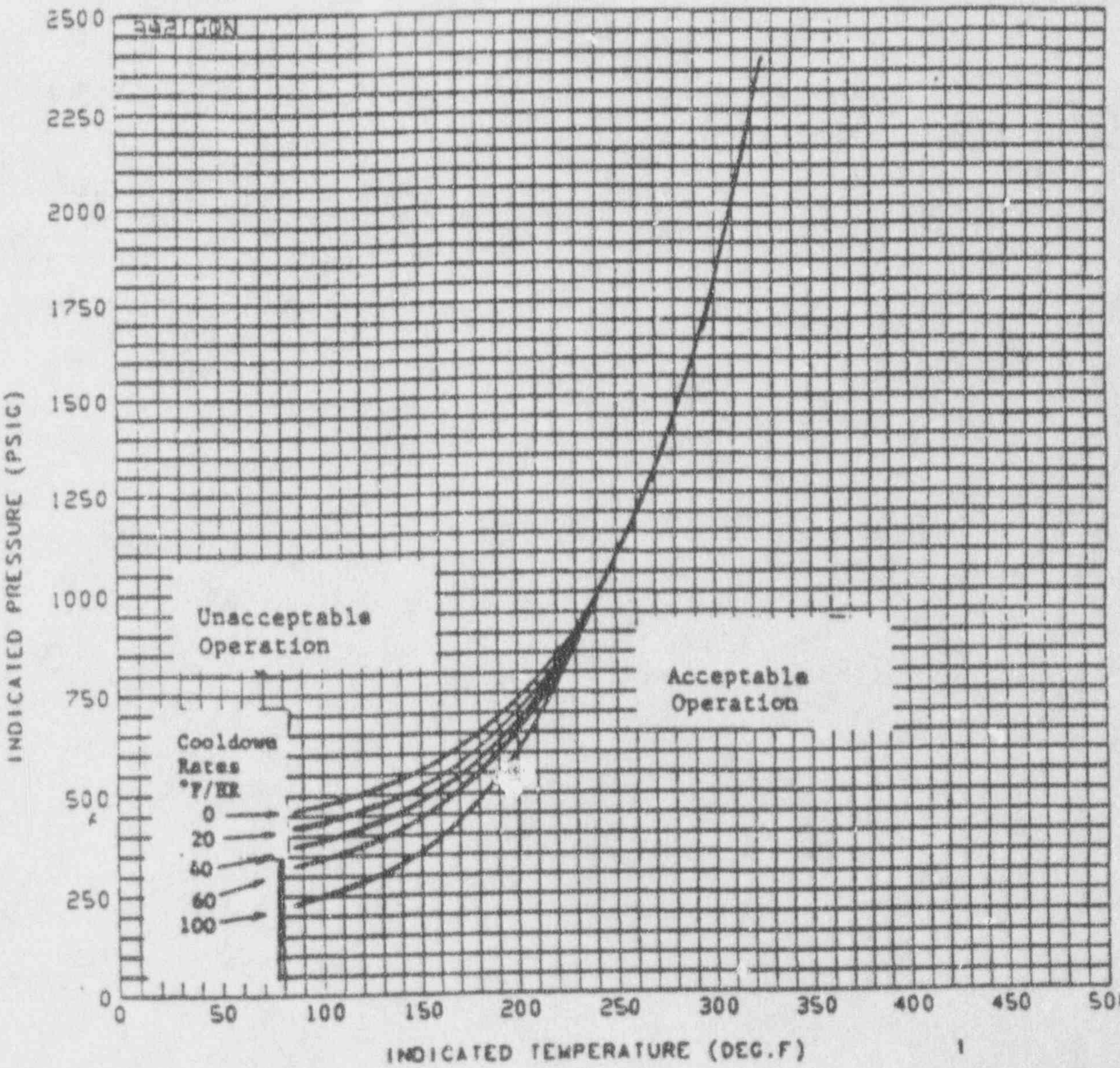
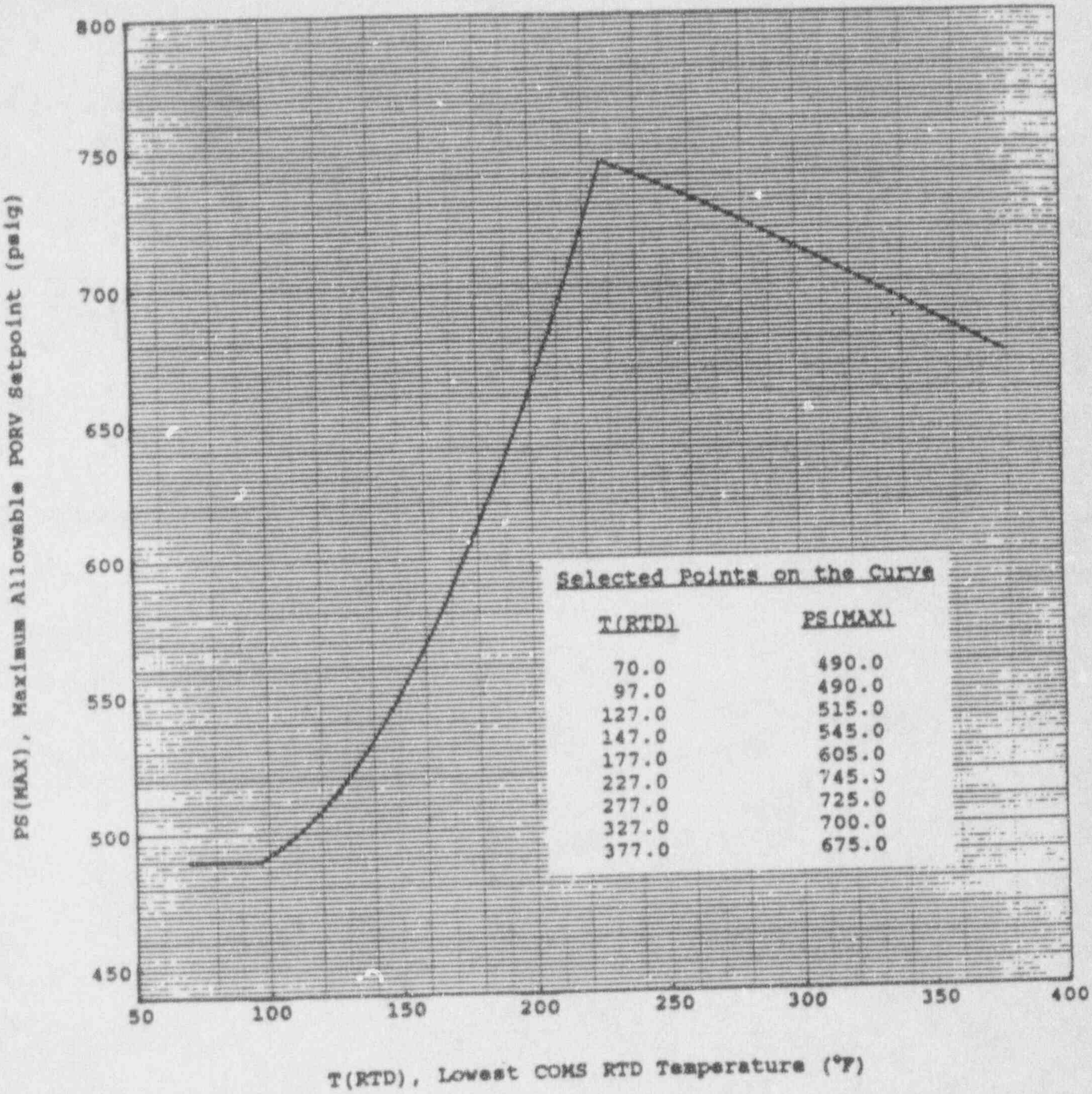


Figure 2. Braidwood Unit 1 Reactor Coolant System Cooldown (Cooldown rates up to 100°F/hr) Limitations Applicable for the First 32 EFPY (With Margins 10°F and 60 psig For Instrumentation Errors)



Selected Points on the Curve

T(RTD)	PS(MAX)
70.0	490.0
97.0	490.0
127.0	515.0
147.0	545.0
177.0	605.0
227.0	745.0
277.0	725.0
327.0	700.0
377.0	675.0

Figure 3 Nominal PORV Pressure Relief Setpoint vs. RCS Temperature for the Cold Overpressure Protection System Applicable to 32 EFPY (Unit 1)

Jan.29,1992
In reply refer to
CHRON No.

179488

To: T. Morello
Braidwood Tech Staff

Subject: Westinghouse Report CCE-91-126
"Heatup/Cooldown Final Report (Capsule U)"

Per our recent conversation, the subject report was issued by Westinghouse as part of their contract with CECo to analyze data from the reactor vessel specimen capsules.

Using physical data from the capsule, Westinghouse has calculated a new Adjusted Reference Temperature for the limiting vessel material - circumferential weld (Table 3). New heatup and cooldown curves were also prepared using this value (Figure 1 & 2). The results have been determined to be conservative to the existing analysis and as result, the current Tech Spec heatup/cooldown curves remain applicable to Reg.Guide 1.99 rev.2 for 16 EFPY.

Should you have any question or comments, please call Bob Waninski at x-7387, Downers Grove.

1/29/92
R.E. Waninski
PWR System Design Engineer

1/29/92
E.D. Swartz
PWR System Design Supervisor

Westinghouse
Electric Corporation

Energy Systems

Box 355
Pittsburgh Pennsylvania 15230 0355

CCE-91-126
February 19, 1991

Mr. G. Gerzen
Commonwealth Edison Company
Technical Center
1319 South First Avenue
Maywood, Illinois 60153

RECEIVED
FEB 26 1991
BRAIDWOOD STATION

Commonwealth Edison Company
Braidwood Unit 2
Heatup/Cooldown Final Report (Capsule U)

Dear Mr. Gerzen:

Attached is the final report on *"Heatup and Cooldown Curves for the Braidwood Unit 2 Reactor Vessel (Capsule U)."* Comments from Commonwealth Edison Company have been incorporated in the final report.

Also enclosed is a postage paid *"YOU BE THE JUDGE"* form. It would be very much appreciated if you could take a few minutes to fill out the form and return to Westinghouse Electric Corporation.

If you have any questions, please do not hesitate to contact me.

Sincerely yours,

WESTINGHOUSE ELECTRIC CORPORATION

G. P. Toth, Manager
Commonwealth Edison Company
Domestic Customer Projects

NKRay:SAP/cem

cc: K. Kofron
M. Lohmann
M. Sears
E. Swartz
W. Feimster
R. Waninski

0050DCP:50:SAP:021991-1