WOLF CREEK GENERATING STATION – UNIT 1

PRESSURE AND TEMPERATURE LIMITS REPORT

Revision 0

PRESSURE AND TEMPERATURE LIMITS REPORT

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1.0 Reactor Coolant System (RCS) Pressure and Temperature Limits Report (PTLR)

This PTLR for WCGS has been prepared in accordance with the requirements of Technical Specification (TS) 5.6.6. The TS addressed in this report are listed below:

LCO 3.4.3 RCS Pressure and Temperature (P/T) Limits

LCO 3.4.12 Low Temperature Overpressure Protection (LTOP) Systems

2.0 Operating Limits

The parameter limits for the specifications listed in Section 1.0 are presented in the following subsections. The limits were developed using a methodology that is in accordance with the NRC-approved methodology specified in Specification 5.6.6 (Ref. 1). NRC approval of this methodology was received in Amendment No. [ ], (Ref. 2). NRC approval of the PTLR was received in Reference 3.

The revised P/T Limit curves account for a requirement of 10 CFR 50, Appendix G, that the temperature of the closure head flange and vessel flange regions must be at least 120°F higher than the limiting RT_{NDT} for these regions when the pressure exceeds 20% of the preservice hydrostatic test pressure (3106 psig).

2.1 RCS Pressure and Temperature (P/T) Limits (LCO 3.4.3)

2.1.1 The RCS temperature rate-of-change limits are (Ref. 2)

a. A maximum heatup of 100°F in any 1-hour period.

b. A maximum cooldown of 100°F in any 1-hour period.

c. A maximum temperature change of 10°F in any 1-hour period during inservice hydrostatic and leak testing operations above the heatup and cooldown limit curves.

2.1.2 The RCS P/T limits for heatup, cooldown, inservice hydrostatic and leak testing, and criticality are specified by Figures 2.1-1 and 2.1-2.

2.2 Low Temperature Overpressure Protection (LTOP)System Setpoints (LCO 3.4.12)

The power-operated relief valves (PORVs) shall each have lift settings in accordance with Figure 2.2-1. The LTOP System (Cold Overpressure Mitigation System/PORVs) arming temperature is 368°F. These lift setpoints have been developed using the NRC approved methodologies specified in Technical Specification 5.6.6.

2.2 (continued)

The maximum allowed PORV setpoint for LTOP is derived by analysis which models the performance of the LTOP assuming limiting mass and heat input transients with incorporation of 10% relaxation in accordance with ASME Code Case N-514. Operation with a PORV setpoint less than or equal to the maximum setpoint ensures that Appendix G criteria will not be violated with consideration for: (1) process and instrumentation uncertainties; (2) single failure. To ensure mass and heat input transients more severe than those assumed cannot occur, it is required to lockout both Safety Injection pumps and one centrifugal charging pump (one centrifugal charging pump and the normal charging pump are operational) while in MODES 4, 5, and 6 with the reactor vessel head installed and limit heat input of starting a reactor coolant pump if secondary temperature is more than 50°F above reactor coolant temperature.

MATERIAL PROPERTY BASIS

LIMITING MATERIAL: LOWER SHELL PLATE R2508-3
LIMITING ART VALUES AT 20 EFPY: 1/4T, 90°F
3/4T, 80°F

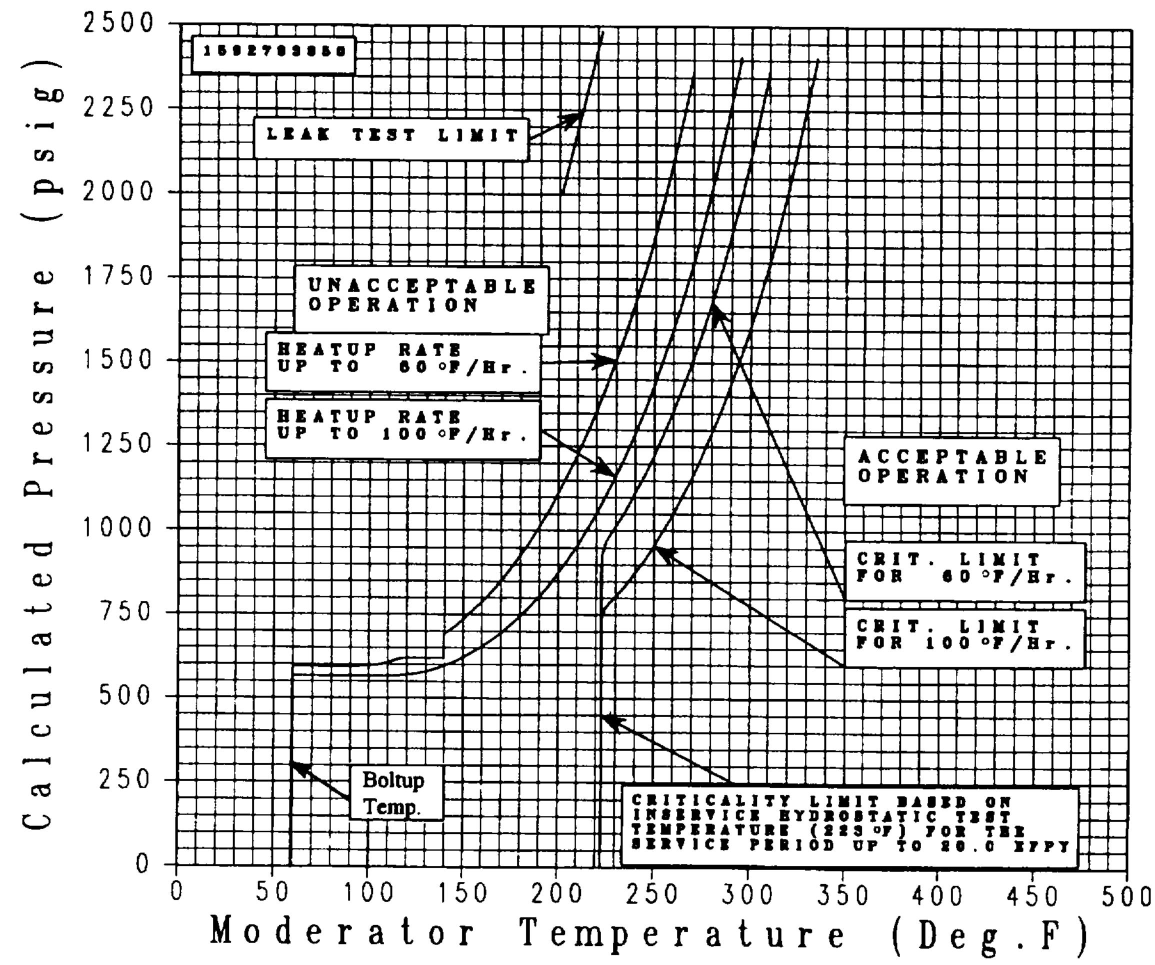

FIGURE 2.1-1 Wolf Creek Reactor Coolant System Heatup Limitations (Heatup Rates of 60 and 100°F/hr) Applicable to 20 EFPY (Without Margins for Instrumentation Uncertainty)

TABLE 2.1-1 Wolf Creek Heatup Limits at 20 EFPY Without Margins for Instrumentation Uncertainty									
60°F/hr		60°F/hr Crit. Limit		100°F/hr		100°F/hr Crit. Limit		Leak Test Limit	
Temp. (°F)	Press. (psig)	Temp. (°F)	Press. (psig)	Temp. (°F)	Press. (psig)	Temp. (°F)	Press. (psig)	Temp. (°F)	Press. (psig)
60	0	223	0	60	0	223	0	201	2000
60	595	223	605	60	567	223	567	223	2485
65	596	223	604	65	567	223	567		
85	596	223	598	85	567	223	567		
85	596	223	596	90	567	223	567		
90	596	223	596	95	567	223	567		
95	596	223	600	100	567	223	567		
100	596	223	606	105	567	223	567		
105	600	223	615	110	567	223	567		
110	606	223	626	115	567	223	569		
115	615	223	639	120	569	223	573		
120	621	223	654	125	573	223	579		
125	621	223	672	130	579	223	587		
130	621	223	691	135	587	223	596		
135	621	223	712	140	596	223	608		
140	621	223	736	145	608	223	622		
140	691	223	761	150	622	223	637		
145	712	223	789	155	637	223	655		
150	736	223	820	160	655	223	674		
155	761	223	852	165	674	223	696		
160	789	223	888	170	696	223	719		
165	820	223	926	175	719	223	745		
170	852	225	967	180	745	225	774		
175	888	230	1012	185	774	230	805		
180	926	235	1060	190	805	235	838		
185	967	240	1111	195	838	240	875		
190	1012	245	1166	200	875	245	914		
195	1060	250	1225	205	914	250	956		
200	1111	255	1289	210	956	255	1002		
205	1166	260	1357	215	1002	260	1052		
210	1225	265	1430	220	1052	265	1105		
215	1289	270	1509	225	1105	270	1162		
220	1357	275	1593	230	1162	275	1223		
225	1430	280	1683	235	1223	280	1289		
230	1509	285	1779	240	1289	285	1360		
235	1593	290	1882	245	1360	290	1436		
240	1683	295	1992	250	1436	295	1517		
245	1779	300	2110	255	1517	300	1605		
250	1882	305	2235	260	1605	305	1698		
255	1992	310	2369	265	1698	310	1798		
260	2110			270	1798	315	1905		
265	2235			275	1905	320	2019		
270	2369			280	2019	325	2141		
				285	2141	330	2271		
				290	2271	335	2409		
				295	2409				

MATERIAL PROPERTY BASIS

LIMITING MATERIAL: LOWER SHELL PLATE R2508-3
LIMITING ART VALUES AT 20 EFPY: 1/4T, 90°F
3/4T, 80°F

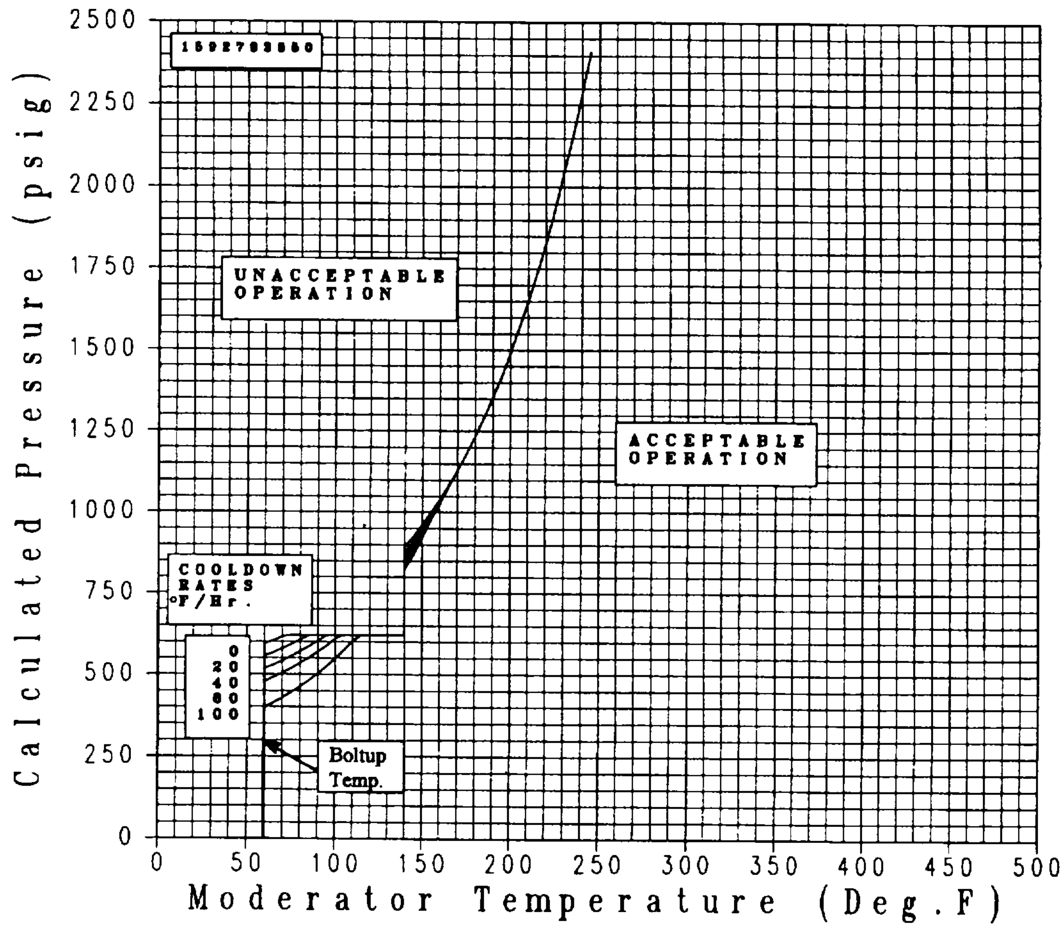

Figure 2.1-2 Wolf Creek Reactor Coolant System Cooldown Limitations (Cooldown Rates of 0, 20, 40, 60 and 100°F/hr) Applicable to 20 EFPY (Without Margins for Instrumentation Uncertainty)

TABLE 2.1-2 Wolf Creek Cooldown Limits at 20 EFPY Without Margins for Instrumentation Uncertainty									
Steady State		20°F/hr		40°F/hr		60°F/hr		100°F/hr	
Temp. (°F)	Press. (psig)	Temp. (°F)	Press. (psig)	Temp. (°F)	Press. (psig)	Temp. (°F)	Press. (psig)	Temp. (°F)	Press. (psig)
60	0	60	0	60	0	60	0	60	0
60	595	60	557	60	518	60	480	60	401
65	605	65	568	65	530	65	492	65	414
70	616	70	579	70	542	70	505	70	429
75	621	75	592	75	556	75	519	75	445
80	621	80	606	80	570	80	534	80	463
85	621	85	620	85	586	85	551	85	482
90	621	90	621	90	602	90	569	90	502
95	621	95	621	95	620	95	588	95	525
100	621	100	621	100	621	100	609	100	549
105	621	105	621	105	621	105	621	105	574
110	621	110	621	110	621	110	621	110	602
115	621	115	621	115	621	115	621	115	621
120	621	120	621	120	621	120	621	120	621
125	621	125	621	125	621	125	621	125	621
130	621	130	621	130	621	130	621	130	621
135	621	135	621	135	621	135	621	135	621
140	621	140	621	140	621	140	621	140	621
140	894	140	876	140	859	140	844	140	822
145	927	145	911	145	897	145	885	145	870
150	962	150	948	150	937	150	929	150	920
155	1000	155	989	155	981	155	975	155	975
160	1040	160	1032	160	1028	160	1026	160	1034
165	1084	165	1079	165	1078	165	1080		
170	1130	170	1129						
175	1181								
180	1234								
185	1292								
190	1354								
195	1421								
200	1492								
205	1569								
210	1651								
215	1739								
220	1833								
225	1934								
230	2042								
235	2158								
240	2281								
245	2413								

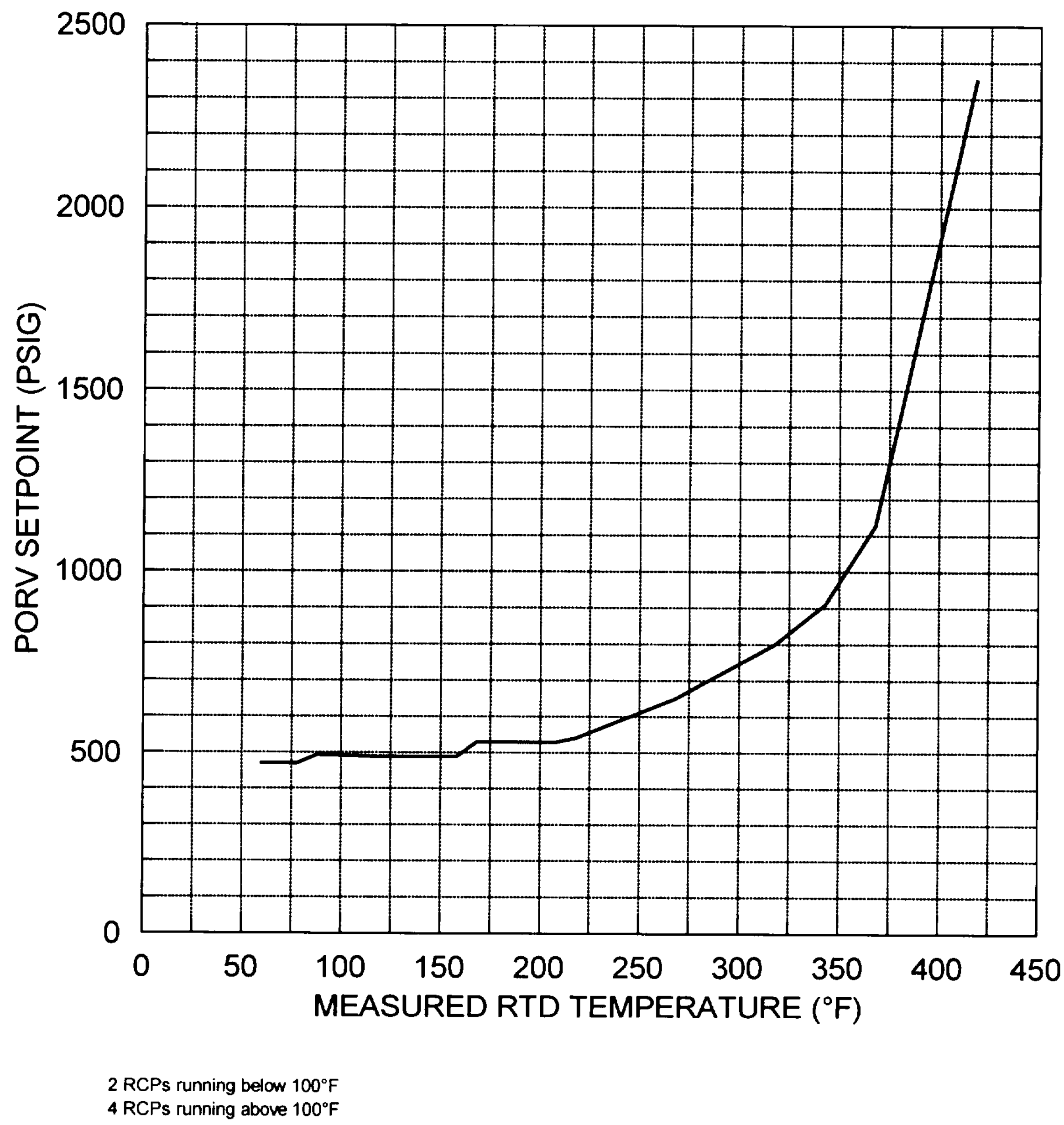

FIGURE 2.2-1 Maximum Allowed PORV Setpoint for the Cold Overpressure Mitigation System

TABLE 2.2-1 Data Points for Maximum Allowed PORV Setpoints	
Temperature (°F)	**Pressure (psig)**
60	469
78	469
88	493
118	488
158	488
168	529
218	540
268	650
318	800
343	910
368	1127
418	2350

3.0 Reactor Vessel Material Surveillance Program

The reactor vessel material surveillance program is in compliance with Appendix H to 10 CFR 50, entitled "Reactor Vessel Material Surveillance Program Requirements" and Section 5.3 of the WCGS Updated Safety Analysis Report. The withdrawal schedule is presented in USAR Table 5.3-11. The surveillance capsule reports are as follows:

1. WCAP-11553, August 1987, "Analysis of Capsule U from the Wolf Creek Nuclear Operating Corporation Wolf Creek Reactor Vessel Radiation Surveillance Programs."

2. WCAP-13365, Revision 1, April 1993, "Analysis of Capsule Y from the Wolf Creek Nuclear Operating Corporation Wolf Creek Reactor Vessel Radiation Surveillance Programs."

3. WCAP-15078, Revision 1, August 1998, "Analysis of Capsule V from the Wolf Creek Nuclear Operating Corporation Wolf Creek Reactor Vessel Radiation Surveillance Programs."

4.0 Reactor Vessel Surveillance Data Credibility

Regulatory Guide 1.99, Revision 2, describes general procedures acceptable to the NRC staff for calculating the effects of neutron radiation embrittlement of the low-alloy steels currently used for light-water-cooled reactor vessels. Position C.2 of Regulatory Guide 1.99, Revision 2, describes the method for calculating the adjusted reference temperature and Charpy upper-shelf energy of reactor vessel beltline materials using surveillance capsule data. The methods of Position C.2 can only be applied when two or more credible surveillance data sets become available from the reactor in question.

To date there has been three surveillance capsules removed from the Wolf Creek Unit 1 reactor vessel. To use these surveillance data sets, they must be shown to be credible. In accordance with the discussion of Regulatory Guide 1.99, Revision 2, there are five requirements that must be met for the surveillance data to be judged credible.

The purpose of this evaluation is to apply the credibility requirements of Regulatory Guide 1.99, Revision 2, to the Wolf Creek Unit 1 reactor vessel surveillance data and determine if the Wolf Creek Unit 1 surveillance data is credible.

Criterion 1: Materials in the capsules should be those judged most likely to be controlling with regard to radiation embrittlement.

The beltline region of the reactor vessel is defined in Appendix G to 10 CFR Part 50, "Fracture Toughness Requirements," as follows:

> "the reactor vessel (shell material including welds, heat affected zones, and plates or forgings) that directly surrounds the effective height of the active core and adjacent regions of the reactor vessel that are predicted to experience sufficient neutron radiation damage to be considered in the selection of the most limiting material with regard to radiation damage."

The Wolf Creek Unit 1 reactor vessel consists of the following beltline region materials:

- Intermediate shell plate R2005-1,
- Intermediate shell plate R2005-2,
- Intermediate shell plate R2005-3,
- Lower shell plate R2508-1,
- Lower shell plate R2508-2,
- Lower shell plate R2508-3, and
- All vessel beltline weld seams were fabricated with weld wire heat number 90146. The intermediate to lower shell circumferential weld seam 101-171 was fabricated with Flux Type 124 Lot Number 1061. The intermediate and lower shell longitudinal weld seams 101-124A, B & C and 101-142A, B & C were fabricated with Flux Type 0091 Lot Number 0842. The surveillance weld metal was fabricated with weld wire heat number 90146, Flux Type 124 Lot Number 1061. Per Regulatory Guide 1.99, Revision 2, "weight-percent copper" and "weight percent nickel" are the best-estimate values for the material, which will normally be the mean of the measured values for a plate or forging or for weld samples made with the weld wire heat number that matches the critical vessel weld. The surveillance weld metal was made with the same weld wire heat as all of the vessel beltline weld seams and is therefore representative of all of the beltline weld seams.

The Wolf Creek Unit 1 surveillance program utilizes longitudinal and traverse test specimens from lower shell plate R2508-3. The surveillance weld metal was fabricated with weld wire heat number 90146, Flux Type 124 Lot Number 1061.

The Wolf Creek Unit 1 surveillance program was based on ASTM E185-79. When the surveillance program material was selected it was believed that copper and phosphorus were the elements most important to embrittlement of reactor vessel steels. Lower shell plate R2508-3 had the highest initial RT_{NDT} and the lowest initial USE of all plate materials in the beltline region. In addition, lower shell plate R2508-3 had approximately the same copper and phosphorous content as the other beltline plate materials. Hence, based on the highest initial RT_{NDT} and lowest initial upper shelf energy, lower shell plate R2508-3 was chosen for the surveillance program.

Per Regulatory Guide 1.99, Revision 2, "weight-percent copper" and weight percent nickel" are the best-estimate values for the material, which will normally be the mean of the measured values for a plate or forging or for weld samples made with the weld wire heat number that matches the critical vessel weld. Since the surveillance weld metal was made with the same weld wire heat as all of the vessel beltline weld seams, it is representative of the limiting beltline weld metal.

Therefore, the Wolf Creek Unit 1 surveillance material meets the intent of this criteria.

Criterion 2: Scatter in the plots of Charpy energy versus temperature for the irradiated and unirradiated conditions should be small enough to permit the determination of the 30 ft-lb temperature and upper shelf energy unambiguously.

Plots of Charpy energy versus temperature for the unirradiated and irradiated condition are presented in WCAP-15078, Revision 1, September 1998, "Analysis of Capsule V from the Wolf Creek Nuclear Operating Corporation Wolf Creek Reactor Vessel Radiation Surveillance Program."

Based on engineering judgement, the scatter in the data presented in these plots is small enough to permit the determination of the 30 ft-lb temperature and the upper shelf energy of the Wolf Creek Unit 1 surveillance materials unambiguously. Hence, the Wolf Creek Unit 1 surveillance program meets this criterion.

Criterion 3: When there are two or more sets of surveillance data from one reactor, the scatter of ΔRT_{NDT} values about a best-fit line drawn as described in Regulatory Position 2.1 normally should be less than 28°F for welds and 17°F for base metal. Even if the fluence range is large (two or more orders of magnitude), the scatter should not exceed twice those values. Even if the data fail this criterion for use in shift calculations, they may be credible for determining decrease in upper shelf energy if the upper shelf can be clearly determined, following the definition given in ASTM E185-82.

The functional form of the least squares method as described in Regulatory Position 2.1 will be utilized to determine a best-fit line for this data and to determine if the scatter of the ΔRT_{NDT} values about this line is less than 28°F for welds and less than 17°F for the plate.

Following is the calculation of the best-fit line as described in Regulatory Position 2.1 of Regulatory Guide 1.99, Revision 2.

NOTE: Since the calculated vessel fluence values are greater than the best estimate vessel fluence values, the calculated fluence values will be used for all calculations.

Table 4.0-1
Wolf Creek Surveillance Capsule Data

Material	Capsule	f[1]	FF[2]	ΔRT_{NDT}	FFxΔRT_{NDT}	FF^2
Lower Shell Plate R2508-3 (Longitudinal)	U	0.3429	0.705	36.46°F	25.70°F	0.50
	Y	1.308	1.075	16.03°F	17.23°F	1.16
	V	2.528	1.249	52.03°F	64.99°F	1.56
Lower Shell Plate R2508-3 (Transverse)	U	0.3429	0.705	23.79°F	16.77°F	0.50
	Y	1.308	1.075	35.39°F	38.04°F	1.16
	V	2.528	1.249	54.53°F	68.11°F	1.56
				SUM	230.84°F	6.44
	$CF_{R2508\text{-}3} = \Sigma(FF^* RT_{NDT}) \div \Sigma(FF^2) = (230.84°F) \div (6.44) = 35.8°F$					
Weld Metal[3]	U	0.3429	0.705	27.21°F	19.18°F	0.50
	Y	1.308	1.075	45.09°F	48.47°F	1.16
	V	2.528	1.249	46.33°F	57.87°F	1.56
				SUM	125.52°F	3.22
	$CF_{weld} = \Sigma(FF^* RT_{NDT}) \div \Sigma(FF^2) = (125.52) \div (3.22) = 39.0°F$					

(1) f = Calculated Fluence (10^{19} n/cm^2, E > 1.0 MeV). These values were re-evaluated as part of the capsule V analysis. (See Section 6 of WCAP-15078, Revision 1.)

(2) FF = Fluence Factor = $f^{(0.28 - 0.1 * \log f)}$

(3) ΔRT_{NDT} values do not include the adjustment ratio procedure of Regulatory Guide 1.99, Revision 2, Position 2.1, since this calculation is based on the actual surveillance weld metal measured shift values.

The scatter of ΔRT_{NDT} values about the functional form of a best fit line drawn as described in Regulatory Position 2.1 is present in Table 4.0-2.

Table 4.0-2
Wolf Creek Surveillance Capsule Data

Material	Capsule	CF	FF	Best Estimate ΔRT_{NDT}[a]	Measured ΔRT_{NDT}[b]	Change in ΔRT_{NDT}
Lower Shell Plate R2508-3 (Longitudinal)	U	35.8°F	0.705	25.24	36.46°F	-11.2
	Y	35.8°F	1.075	38.49	16.03°F	22.46
	V	35.8°F	1.249	44.71	52.03°F	-7.32
Lower Shell Plate R2508-3 (Transverse)	U	35.8°F	0.705	25.24	23.79°F	1.45
	Y	35.8°F	1.075	38.49	35.39°F	3.10
	V	35.8°F	1.249	44.71	54.53°F	-9.82
Surveillance Program Weld Metal	U	39.0°F	0.705	27.50	27.21°F	0.29
	Y	39.0°F	1.075	41.93	45.09°F	-3.16
	V	39.0°F	1.249	48.71	46.33°F	2.38

(a) Best estimate ΔRT_{NDT} = CF * FF. Where the CF used when comparing best-estimate ΔRT_{NDT} values to measured ΔRT_{NDT} values for the credibility analysis were calculated based on the measure surveillance data.

(b) Calculated using measured Charpy data plotted using CVGRAPH 4.1.

The scatter of ΔRT_{NDT} values about the functional form of a best-fit line drawn as described in Regulatory Position 2.1 is less than 17°F for all but one plate material data point. One out of six values for the plate material would be expected to be out of the 1 σ range.

However, the capsule Y longitudinal data point is out on the low side (i.e., the prediction is 22.5°F higher than the measured value) and is less than 2 σ out. The scatter of ΔRT_{NDT} values about the functional form of a best-fit line drawn as described in Regulatory Position 2.1 is less than 28°F for the weld metal. Therefore, this criteria is met for the surveillance data of the Wolf Creek Unit 1 surveillance program material.

Criterion 4: The irradiation temperature of the Charpy specimens in the capsule should match the vessel wall temperature at the cladding/base metal interface within +/- 25°F.

The capsule specimens are located in the reactor between the neutron pads and the vessel wall and are positioned opposite the center of the core. The test capsules are in baskets attached to the neutron pads. The location of the specimens with respect to the reactor vessel beltline provides assurance that the reactor vessel wall and the specimens experience equivalent operating conditions such that the temperatures will not differ by more than 25°F. Hence this criteria is met.

Criterion 5: The surveillance data for the correlation monitor material in the capsule should fall within the scatter band of the data base for that material.

The Wolf Creek Unit 1 surveillance program does not contain correlation monitor material. Therefore, this criterion is not applicable to the Wolf Creek Unit 1 surveillance program.

5.0 Supplemental Data Tables

Table 5.0-1 Comparison of Wolf Creek Surveillance Material 30-ft-lb Transition Temperature Shifts and Upper Shelf Energy Decreases with Regulatory Guide 1.99, Revision 2, Predictions.

Table 5.0-2 Calculation of Chemical Factors Using Surveillance Capsule Data.

Table 5.0-3 Provides the unirradiated reactor vessel toughness data. The bolt-up temperature is also included in this Table.

Table 5.0-4 Provides a summary of the peak pressure vessel neutron fluence values at 20 EFPY used for the calculation of ART values.

Table 5.0-5 Provides a summary of the adjusted reference temperature (ARTs) for reactor vessel beltline materials at the ¼-T and ¾-T locations for 20 EFPY.

Table 5.0-6 Shows the calculation of the ART at 20 EFPY for the limiting reactor vessel material (lower shell plate R-2508-3).

Table 5.0-7 Provides RT_{PTS} values for 35 EFPY.

6.0 References

1. Technical Specification 5.6.6, "Reactor Coolant System (RCS) PRESSURE AND TEMPERATURE LIMITS REPORT (PTLR)."

2. License Amendment No. [ ], dated [ ], from [ ] to [ ].

3. NRC letter dated [ ], [title of letter]

4. WCAP-14040-NP-A, "Methodology Used to Develop Cold Overpressure Mitigating System Setpoints and RCS Heatup and Cooldown Limit Curves," January 1996.

Table 5.0-1
Comparison of Wolf Creek Surveillance Material 30 ft-lb Transition Temperature Shifts and Upper Shelf Energy Decreases with Regulatory Guide 1.99, Revision 2, Predictions

Materials	Capsule	Fluence (n/cm2, E>1.0 MeV)	30 ft-lb Transition Temperature Shift		Upper Shelf Energy Decrease	
			Predicted (°F) [(a)]	Measured (°F) [(b)]	Predicted (%) [(a)]	Measured (%) [(c)]
Lower Shell Plate R2508-3 (Longitudinal)	U	3.429×10^{18}	40.9	36.46	14.5	2
	Y	1.308×10^{19}	62.4	16.03	20.0	11
	V	2.528×10^{19}	72.4	52.03	24.0	13
Lower Shell Plate R2508-3 (Transverse)	U	3.429×10^{18}	40.9	23.79	14.5	0
	Y	1.308×10^{19}	62.4	35.39	20.0	0
	V	2.528×10^{19}	72.4	54.53	24.0	6
Weld Metal	U	3.429×10^{18}	30.7	27.21	16.5	8
	Y	1.308×10^{19}	46.8	45.09	2.5	6
	V	2.528×10^{19}	54.3	46.33	26.5	11
HAZ Metal	U	3.429×10^{18}	--	58.41	--	13
	Y	1.308×10^{19}	--	12.98	--	0
	V	2.528×10^{19}	--	55.91	--	0

(a) Based on Regulatory Guide 1.99, Revision 2, methodology using the mean weight percent values of copper and nickel of the surveillance material.

(b) Calculated using measured Charpy data plotted using CVGRAPH, Version 4.1.

(c) Values are based on the definition of upper shelf energy given in ASTM E185-82.

Table 5.0-2 Calculation of Chemistry Factors Using Surveillance Capsule Data						
Material	Capsule	f[1]	FF[2]	ΔRT_{NDT}	$FF \times \Delta RT_{NDT}$	FF^2
Lower Shell Plate R2508-3 (Longitudinal)	U	0.3429	0.705	36.46°F	25.70°F	0.50
	Y	1.308	1.075	16.03°F	17.23°F	1.16
	V	2.528	1.249	52.03°F	64.99°F	1.56
Lower Shell Plate R2508-3 (Transverse)	U	0.3429	0.705	23.79°F	16.77°F	0.50
	Y	1.308	1.075	35.39°F	38.04°F	1.16
	V	2.528	1.249	54.53°F	68.11°F	1.56
				SUM	230.84°F	6.44
	$CF_{R2508-3} = \Sigma(FF^* RT_{NDT}) \div \Sigma(FF^2) = (230.84°F) \div (6.44) = 35.8°F$					
Weld Metal[3]	U	0.3429	0.705	27.21°F	19.18°F	0.50
	Y	1.308	1.075	45.09°F	48.47°F	1.16
	V	2.528	1.249	46.33°F	57.87°F	1.56
				SUM	125.52°F	3.22
	$CF_{weld} = \Sigma(FF^* RT_{NDT}) \div \Sigma(FF^2) = (125.52) \div (3.22) = 39.0°F$					

(1) f = Calculated Fluence (10^{19} n/cm^2, E > 1.0 MeV). These values were re-evaluated as part of the capsule V analysis. (See Section 6 of WCAP 15078, Revision 1.)

(2) FF = Fluence Factor = $f^{(0.28 - 0.1 * \log f)}$

(3) ΔRT_{NDT} values do not include the adjustment ratio procedure of Regulatory Guide 1.99, Revision 2, Position 2.1, since this calculation is based on the actual surveillance weld metal measured shift values.

TABLE 5.0-3 Reactor Vessel Beltline Material Unirradiated Toughness Properties			
Material Description	**Cu (%)**	**Ni(%)**	**Initial RT_{NDT}[a]**
Closure Head Flange R2504-1	- -	0.66	20°F[c]
Vessel Flange R2501-1	- -	0.70	20°F[c]
Intermediate Shell Plate R2005-1	0.04	0.66	-20°F
Intermediate Shell Plate R2005-2	0.04	0.64	-20°F
Intermediate Shell Plate R2005-3	0.05	0.63	-20°F
Lower Shell Plate R2508-1	0.09	0.67	0°F
Lower Shell Plate R2508-2	0.06	0.64	10°F
Lower Shell Plate R2508-3	0.09	0.58	40°F
Intermediate and Lower Shell Longitudinal Weld Seams[b]	0.04	0.08	-50°F
Intermediate to Lower Shell Circumferential Weld Seam [b]	0.04	0.08	-50°F
Surveillance Program Weld Metal[b]	0.07	0.10	- -

(a) The initial RT_{NDT} values for the plates and welds are based on measured data.

(b) All welds, including the surveillance weld, were fabricated with weld wire heat number 90146. The intermediate to lower shell circumferential weld seam 101-171 was fabricated with Flux Type 124 Lot Number 1061. The intermediate shell longitudinal weld seams 101-124A, B & C and lower shell longitudinal weld seams 101-142A, B, & C were fabricated with Flux Type 0091 Lot 0842. The surveillance weld metal was fabricated with weld wire heat number 90146, Flux Type 124 Lot number 1061. Per Regulatory Guide 1.99, Revision 2, "weight percent copper " and "weight percent nickel" are the best-estimate values for the material, which will normally be the mean of the measured values for a plate or forging or for weld samples made with the weld wire heat number that matches the critical vessel weld." The surveillance weld metal was made with the same weld wire heat as all of the vessel beltline weld seam and is therefore representative of all of the beltline weld seams.

(c) These values are used for considering requirements for the heatup/cooldown curves. Per the methodology given in WCAP-14040-NP-A (Ref. 4), the minimum boltup temperature is 60°F.

TABLE 5.0-4
Summary of the Peak Pressure Vessel Neutron Fluence Values at 20 EFPY used for the Calculation of ART Values (n/cm^2, E > 1.0 MeV)

Material	Surface	¼-T	¾-T
Intermediate Shell Plate R2005-1	1.265×10^{19}	7.54×10^{18}	2.68×10^{18}
Intermediate Shell Plate R2005-2	1.265×10^{19}	7.54×10^{18}	2.68×10^{18}
Intermediate Shell Plate R2005-3	1.265×10^{19}	7.54×10^{18}	2.68×10^{18}
Lower Shell Plate R2508-1	1.265×10^{19}	7.54×10^{18}	2.68×10^{18}
Lower Shell Plate R2508-2	1.265×10^{19}	7.54×10^{18}	2.68×10^{18}
Lower Shell Plate R2508-3	1.265×10^{19}	7.54×10^{18}	2.68×10^{18}
Intermediate & Lower Shell Longitudinal Weld Seam 101-124A & 101-142A (90° Azimuth)	6.82×10^{18}	4.06×10^{18}	1.44×10^{18}
Intermediate & Lower Shell Longitudinal Weld Seam 101-124B,C & 101-142B,C (210° & 330° Azimuth)	1.265×10^{19}	7.54×10^{18}	2.68×10^{18}
Intermediate to Lower Shell Girth Weld Seam 101-171	1.265×10^{19}	7.54×10^{18}	2.68×10^{18}

TABLE 5.0-5

Summary of Adjusted Reference Temperatures (ARTs) for the Reactor Vessel Beltline Materials at the ¼-T and ¾-T Locations for 20 EFPY

Material	20 EFPY ART[(a)]		
	RG 1.99 Rev. 2 Method	¼-T (°F)	¾-T (°F)
Intermediate Shell Plate R2005-1	Position 1-1	28°F	13°F
Intermediate Shell Plate R2005-2	Position 1-1	28°F	13°F
Intermediate Shell Plate R2005-3	Position 1-1	37°F	20°F
Lower Shell Plate R2508-1	Position 1-1	87°F	71°F
Lower Shell Plate R2508-2	Position 1-1	78°F	57°F
Lower Shell Plate R2508-3	Position 1-1	127°F	111°F
	Position 2-1	90°F[(b)]	80°F[(b)]
Intermediate & Lower Shell Longitudinal Weld Seam 101-124A & 101-142A (90° Azimuth)	Position 1-1	-3°F	-19°F
	Position 2-1	-8°F	-22°F
Intermediate & Lower Shell Longitudinal Weld Seam 101-124B,C & 101-142B,C (210° & 330° Azimuth)	Position 1-1	8°F	-10°F
	Position 2-1	2°F	-14°F
Intermediate to Lower Shell Girth Weld Seam 101-171	Position 1-1	8°F	-10°F
	Position 2-1	2°F	-14°F

(a) ART = Initial RT_{NDT} + ΔRT_{NDT} + Margin (°F)

(b) These ART values are used to generate the heatup and cooldown curves.

TABLE 5.0-6

Calculation of Adjusted Reference Temperature at 20 EFPY for the Limiting Wolf Creek Reactor Vessel Material (Lower Shell Plate R 2508-3)

Parameter	**ART Value**	
Location	**¼-T**	**¾-T**
Chemistry Factor, CF (°F)	35.8	35.8
Fluence ÷ 10^{19} n/cm^2 (E > 1.0 MeV), f[(a)]	0.754	0.268
Fluence Factor, FF[(b)]	0.92	0.64
ΔRT_{NDT} = CF x FF, (°F)	32.9	22.9
Initial RT_{NDT}, I (°F)	40	40
Margin, M (°F)[(c)]	17	17
ART = I + (CF x FF) + M (°F) per Regulatory Guide 1.99, Rev. 2	90	80

(a) Fluence, f, is based upon f_{surf} (10^{19} n/cm^2, E > 1.0 MeV) = 1.265 at 20 EFPY. The Wolf Creek reactor vessel wall thickness is 8.625 inches at the beltline region.

(b) Fluence Factor (FF) per Regulatory Guide 1.99, Revision 2, is defined as $FF = f^{(0.28 - 0.10 \log f)}$.

(c) 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 0°F since the initial RT_{NDT} is a measured value. The standard deviation for ΔRT_{NDT} term σ_Δ, is 17°F for the plate, except that σ_Δ need not exceed the 0.5 times the mean value of ΔRT_{NDT}.

TABLE 5.0-7

RT_{PTS} Calculations for Wolf Creek Beltline Region Materials at 35 EFPY[(d)]

Material	Fluence (n/cm^2, E >1.0 MeV)	FF	CF (°F)	ΔRT_{PTS}[(c)] (°F)	Margin (°F)	$RT_{NDT(U)}$[(a)] (°F)	RT_{PTS}[(b)] (°F)
Intermediate Shell Plate R2005-1	2.18E+19	1.21	26.0	31.5	31.5	-20	43
Intermediate Shell Plate R2005-2	2.18E+19	1.21	26.0	31.5	31.5	-20	43
Intermediate Shell Plate R2005-3	2.18E+19	1.21	31.0	37.5	34.0	-20	52
Lower Shell Plate R2508-1	2.18E+19	1.21	58.0	70.2	34.0	0	104
Lower Shell Plate R2508-2	2.18E+19	1.21	37.0	44.8	34.0	10	89
Lower Shell Plate R2508-3	2.18E+19	1.21	58.0	70.2	34.0	40	143
Using S/C Data	2.18E+19	1.21	35.8	43.3	17	40	100
Inter. and Lower Shell Long. Weld Seams101-124A & 101-142A (90° Azimuth)	1.15E+19	1.04	31.6	32.9	32.9	-50	16
Using S/C Data	1.15E+19	1.04	28.3	29.4	28.0	-50	7
Inter. and Lower Shell Long. Weld Seams101-124A & 101-142B/C (210° & 330° Azimuth)	2.18E+19	1.21	31.6	38.2	38.2	-50	26
Using S/C Data	2.18E+19	1.21	28.3	34.2	28.0	-50	12
Intermediate to Lower Shell Circumferential Weld Seam 101-171	2.18E+19	1.21	31.6	38.2	38.2	-50	26
Using S/C Data	2.18E+19	1.21	28.3	34.2	28.0	-50	12

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

(b) $RT_{PTS} = RT_{NDT(U)} + \text{Margin} + \Delta RT_{PTS}$

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

(d) Projected no. of EFPY at the EOL.