Joseph M. Farley Nuclear Plant

Unit 1

Pressure Temperature Limits Report

APPROVED FOR ISSUE

OPERATIONS MANAGER 5-8-98 DATE

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1.0 RCS Pressure Temperature Limits Report (PTLR)

This PTLR for Farley Nuclear Plant - Unit 1 has been prepared in accordance with the requirement of Technical Specification (TS) 6.9.1.15. Revisions to the PTLR shall be provided to the NRC after issuance.

This report affects TS 3.4.10.1, RCS Pressure/Temperature Limits (P/T) Limits. All TS requirements associated with low temperature overpressure protection (LTOP) are contained in TS 3.4.10.3, RCS Overpressure Protection Systems.

2.0 Operating Limits

The limits for TS 3.4.10.1 are presented in the subsection which follows and were developed using the methodologies specified in TS 6.9.1.15. The operability requirements associated with LTOP are specified in TS LCO 3.4.10.3 and were determined to adequately protect the RCS against brittle fracture in the event of an LTOP transient in accordance with the methodology specified in TS 6.9.1.15. The limitation on the number of operating reactor coolant pumps (RCPs) is necessary to assure operation consistent with the pressure corrections incorporated in the P/T limits for flow losses associated with the RCPs.

2.1 RCS Pressure/Temperature (P/T) Limits (LCO - 3.4.10.1)

2.1.1 The minimum boltup temperature is 75°F.

2.1.2 The RCS temperature rate-of-change limits are:

a. A maximum heatup of 100°F in any one hour period.
b. A maximum cooldown of 100°F in any one hour period.
c. A maximum temperature change of less than or equal to 10°F in any one hour period during inservice hydrostatic and leak testing operations above the heatup and cooldown limit curves.

2.1.3 The RCS P/T limits for heatup and cooldown are specified by Figures 2-1 and 2-2, respectively.

2.2 RCP Operation Limits

2.2.1 The number of operating RCPs is limited to one at RCS temperatures less than 110°F with the exception that a second pump may be started for the purpose of maintaining continuous flow while taking the operating pump out of service.

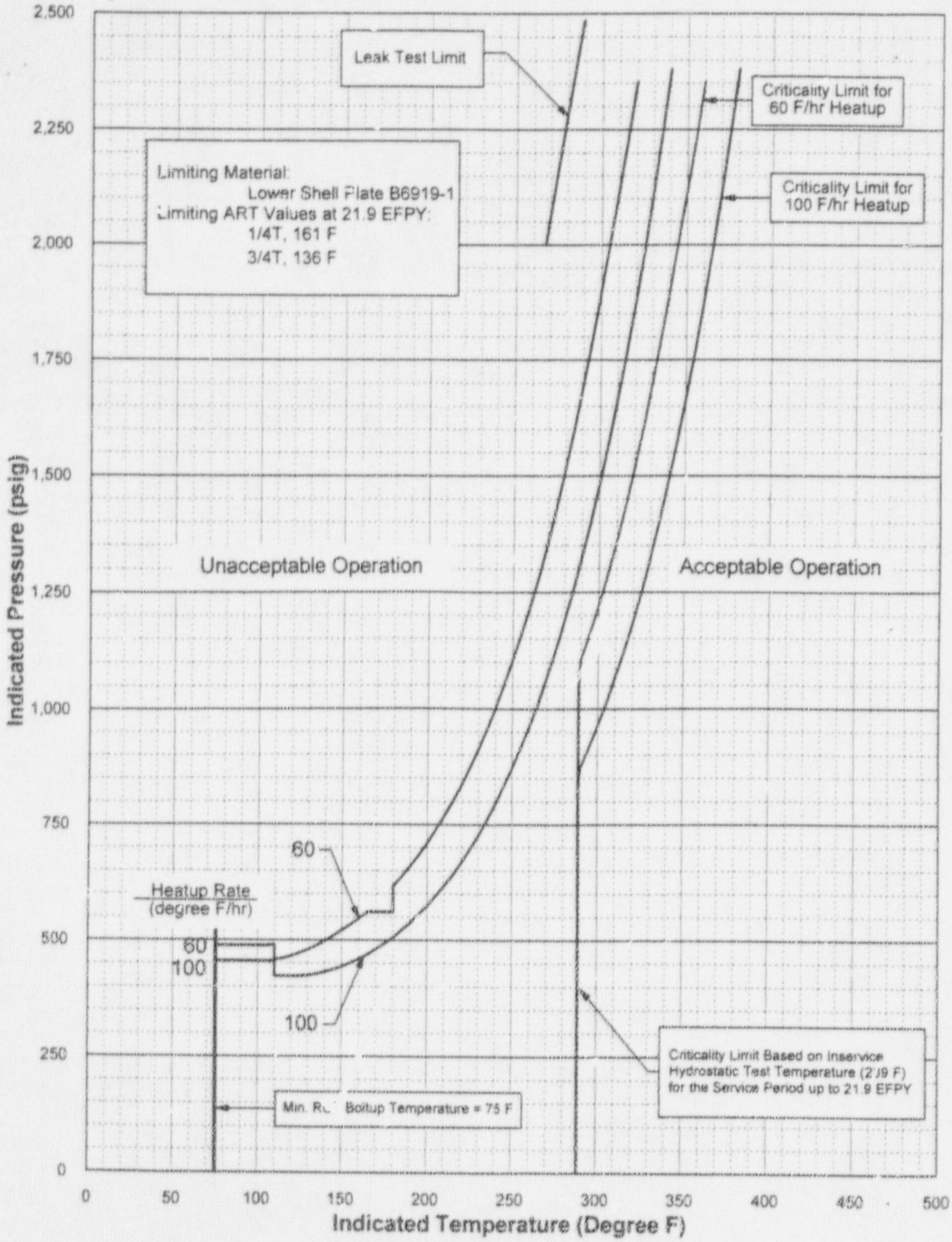

Figure 2-1

Farley Unit 1 Reactor Coolant System Heatup Limitations (Heatup Rates up to 100°F/hr) Applicable to 21.9 EFPY (adjusted to include 60 psi ΔP at RCS temperatures $\geq$ 110°F and 27 psi ΔP for RCS temperatures < 110°F). Includes vessel flange requirements of 180°F and 561 psig per 10 CFR 50, Appendix G. [1]

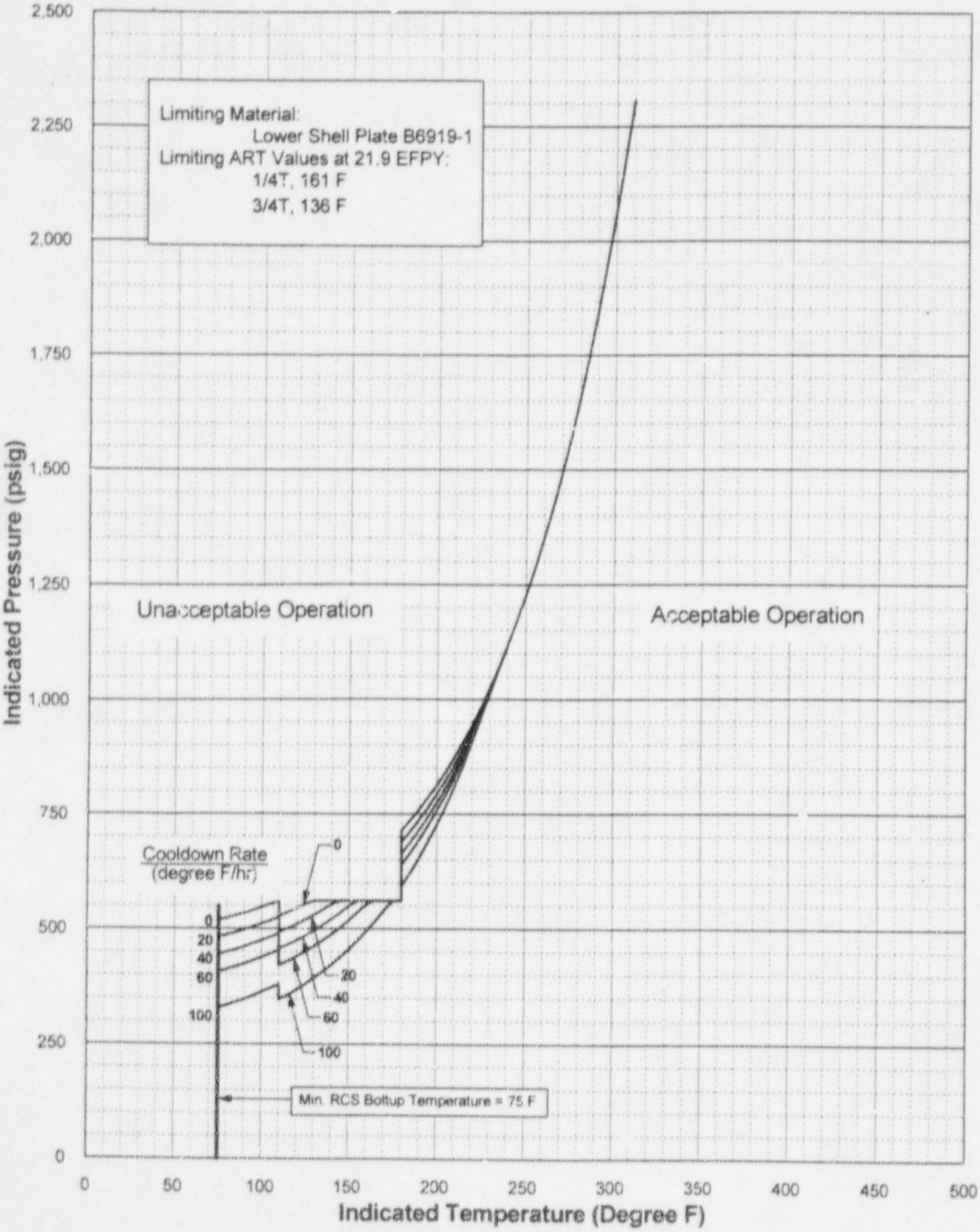

Figure 2-2

Farley Unit 1 Reactor Coolant System Cooldown Limitations (Cooldown Rates up to 100°F/hr) Applicable to 21.9 EFPY (adjusted to include 60 psi ΔP at RCS temperatures ≥ 110°F and 27 psi ΔP for RCS temperatures < 110°F). Includes vessel flange requirements of 180°F and 561 psig per 10 CFR 50, Appendix G.[1]

60 °F		60 °F Criticality Limit		100 °F		100 °F Criticality Limit		Leak Test	
T	P	T	P	T	P	T	P	T	P
75	488	289	0	75	455	289	0	267	2000
80	488	289	480	80	455	289	481	289	2485
85	488	289	469	85	455	289	466		
90	488	289	462	90	455	289	454		
95	488	289	457	95	455	289	444		
100	488	289	455	100	455	289	436		
105	489	289	456	105	455	289	430		
110	491	289	458	110	455	289	426		
110	458	289	461	110	422	289	423		
115	461	289	466	115	422	289	422		
120	466	289	473	120	422	289	423		
125	473	289	480	125	423	289	424		
130	480	289	488	130	424	289	427		
135	488	289	498	135	427	289	432		
140	498	289	509	140	432	289	437		
145	509	289	521	145	437	289	443		
150	521	289	534	150	443	289	451		
155	534	289	547	155	451	289	460		
160	547	289	563	160	460	289	469		
165	561	289	579	165	469	289	481		
170	561	289	597	170	481	289	493		
175	561	289	617	175	493	289	506		
180	561	289	637	180	506	289	521		
180	617	289	660	185	521	289	537		
185	637	289	684	190	537	289	555		
190	660	289	710	195	555	289	574		
195	684	289	738	200	574	289	595		
200	710	289	768	205	595	289	617		
205	738	289	800	210	617	289	641		
210	768	289	835	215	641	289	668		
215	800	289	873	220	668	289	696		
220	835	289	913	225	696	289	726		
225	873	289	956	230	726	289	759		
230	913	289	1002	235	759	289	794		
235	956	289	1051	240	794	289	832		
240	1002	290	1105	245	832	290	872		
245	1051	295	1162	250	872	295	916		
250	1105	300	1223	255	916	300	963		
255	1162	305	1288	260	963	305	1013		
260	1223	310	1358	265	1013	310	1068		
265	1288	315	1434	270	1068	315	1125		
270	1358	320	1515	275	1125	320	1188		
275	1434	325	1601	280	1188	325	1254		
280	1515	330	1694	285	1254	330	1326		
285	1601	335	1793	290	1326	335	1402		
290	1694	340	1899	295	1402	340	1484		
295	1793	345	2012	300	1484	345	1572		
300	1899	350	2120	305	1572	350	1666		
305	2012	355	2233	310	1666	355	1767		
310	2120	360	2354	315	1767	360	1874		
315	2233			320	1874	365	1988		
320	2354			325	1988	370	2111		
				330	2111	375	2241		
				335	2241	380	2380		
				340	2380				

Table 2-1

Farley Unit 1 21.9 EFPY Heatup Curve Data Points (adjusted to include 60 psi ΔP at RCS temperatures $\geq$ 110°F and 27 psi ΔP for RCS temperatures < 110°F)[1]

0 °F		20 °F		40 °F		60 °F		100 °F	
T	P	T	P	T	P	T	P	T	P
75	519	75	482	75	445	75	407	75	329
80	523	80	487	80	450	80	412	80	335
85	529	85	492	85	455	85	418	85	341
90	534	90	498	90	461	90	424	90	347
95	540	95	504	95	468	95	431	95	355
100	547	100	511	100	475	100	438	100	363
105	554	105	518	105	482	105	446	105	371
110	559	110	526	110	490	110	454	110	380
110	528	110	493	110	457	110	421	110	347
115	536	115	501	115	466	115	430	115	357
120	545	120	510	120	475	120	440	120	368
125	554	125	520	125	485	125	451	125	380
130	561	130	530	130	496	130	462	130	393
135	561	135	541	135	508	135	475	135	407
140	561	140	553	140	521	140	488	140	422
145	561	145	561	145	534	145	502	145	438
150	561	150	561	150	549	150	518	150	455
155	561	155	561	155	561	155	535	155	474
160	561	160	561	160	561	160	553	160	494
165	561	165	561	165	561	165	561	165	516
170	561	170	561	170	561	170	561	170	540
175	561	175	561	175	561	175	561	175	561
180	561	180	561	180	561	180	561	180	561
180	715	180	689	180	664	180	640	180	593
185	737	185	713	185	689	185	666	185	623
190	760	190	738	190	716	190	694	190	655
195	786	195	765	195	744	195	725	195	690
200	313	200	794	200	775	200	758	200	727
205	842	205	825	205	808	205	793	205	767
210	874	210	858	210	844	210	831	210	811
215	908	215	894	215	882	215	872	215	857
220	944	220	933	220	923	220	916	220	908
225	983	225	974	225	968	225	963	225	962
230	1025	230	1019	230	1015	230	1014	230	1020
235	1070	235	1067	235	1066	235	1069		
240	1119	240	1118						
245	1171								
250	1226								
255	1286								
260	1351								
265	1420								
270	1494								
275	1573								
280	1658								
285	1749								
290	1846								
295	1951								
300	2062								
305	2182								
310	2309								

Table 2-2

Farley Unit 1 21.9 EFPY Cooldown Curve Data Points (adjusted to include 60 psi ΔP at RCS temperatures ≥ 110°F and 27 psi ΔP for RCS temperatures < 110°F) [1]

3.0 Reactor Vessel Material Surveillance Program

The reactor vessel material surveillance program is in compliance with 10 CFR 50, Appendix H, and is described in Section 5.4.3.6 of the Farley FSAR. The removal schedule is provided in Table 3-1. The results of these examinations shall be used to update Figures 2-1 and 2-2 if the results indicate that the adjusted reference temperature (ART) for the limiting beltline material exceeds the ART used to generate the P/T limits shown in Figures 2-1 and 2-2 for the specified fluence period.

Table 3-1

SURVEILLANCE CAPSULE WITHDRAWAL SCHEDULE [a]

Capsule	Capsule Location (Degree)	Lead Factor	Removal EFPY [b]	Fluence (n/cm^2)
Y [c]	343	3.11	1.13	6.42×10^{18}
U [c]	107	3.18	3.02	1.81×10^{19}
X [c]	287	3.30	6.12	3.24×10^{19}
W [c]	110	3.02	12.43	5.17×10^{19}
V	290	3.02	Standby	--
Z	340	3.02	Standby	--

NOTES:
(a) WCAP-14689, Revision 4 [1]
(b) Effective Full Power Years (EFPY) from plant startup
(c) Plant-specific evaluation

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 methodology 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 have been four surveillance capsules removed from the Farley Unit 1 reactor vessel. 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 Farley Unit 1 reactor vessel surveillance data and determine if the Farley 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 50, Fracture Toughness Requirements, December 19, 1995, to be:

> 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 Farley Unit 1 reactor vessel consists of the following beltline region materials:

- Intermediate shell plates B6903-2 and B6903-3;
- Lower shell plates B6919-1 and B6919-2;
- Intermediate shell longitudinal weld seams 19-894 A & B, heat number 33A277, Linde 1092 flux, flux lot 3889;
- Lower shell longitudinal weld seams 20-894 A & B, heat number 90099, Linde 0091 flux, flux lot 3977; and
- Circumferential weld 11-894, heat number 6329637, Linde 0091 flux, flux lot 3999.

Per WCAP-8810[5], the Unit 1 surveillance program was based on ASTM E185-73, Standard Recommended Practice for Surveillance Tests for Nuclear Reactor Vessels. Per Section 4.1 of ASTM E185-73, the base metal and weld metal to be included in the program should represent the material that may limit the operation of the reactor during its lifetime. The test material should be selected on the basis of initial transition temperature, upper shelf energy level, and estimated increase in transition temperature considering chemical composition (copper and phosphorus) and neutron fluence.

Therefore, at the time the Farley Unit 1 surveillance capsule program was developed, lower shell plate B6919-1 was judged to be most limiting based on the above recommendations and was utilized in the surveillance program.

The surveillance program weld for Farley Unit 1 was fabricated using the same heat of weld wire used to fabricate the middle shell axial seams 19-894 A & B (heat 33A277). The results of mechanical property tests performed on the surveillance weld are considered to be representative of the property changes expected in the reactor vessel beltline seams.

Therefore, the materials selected for use in the Farley Unit 1 surveillance program were those judged to be most likely controlling with regard to radiation embrittlement according to the accepted methodology at the time the surveillance program was developed. Based on the above, the Farley Unit 1 surveillance program meets the requirements of Criterion 1.

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 condition are presented in the Unit 1 reactor vessel surveillance program description contained in WCAP-8810[5].

Plots of Charpy energy versus temperature for the irradiated conditions are presented in the reactor vessel surveillance capsule reports for capsules Y [6], U [7], X [8], and W [2].

Based on engineering judgment, the scatter in the data presented in these plots is small enough to determine the 30 ft-lb temperature and upper shelf energy of the Farley Unit 1 surveillance materials unambiguously. Therefore, the Farley Unit 1 surveillance program meets the requirements of Criterion 2.

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 least squares method, as described in Regulatory Position 2.1, will be utilized in determining a best-fit line for this data to determine if this criterion is met.

[Continued on the following page]

Table 4-1

SURVEILLANCE CAPSULE DATA CALCULATION OF BEST-FIT LINE AS DESCRIBED IN POSITION 2.1 OF REGULATORY GUIDE 1.99, REVISION 2 [a]

Material	Capsule	F [b]	FF [c] (x)	ΔRT_{NDT} (y)	FF x ΔRT_{NDT} (xy)	FF^2 (x^2)
Lower Shell Plate B6919-1 (Longitudinal)	Y	0.642	0.876	85	74.5	0.767
	U	1.81	1.16	105	121.8	1.35
	X	3.24	1.31	135	176.9	1.72
	W	5.17	1.41	155	218.6	1.99
Lower Shell Plate B6919-1 (Transverse)	Y	0.642	0.876	55	48.2	0.767
	U	1.81	1.16	90	104.4	1.35
	X	3.24	1.31	105	137.6	1.72
	W	5.17	1.41	145	204.5	1.99
				$\sum_{i=1}^{n}$	1086.5	11.65
	$CF = \Sigma(FF * \Delta RT_{NDT}) \div \Sigma(FF^2) = 93.3°F$					
Weld Metal	Y	0.642	0.876	80	70.1	0.767
	U	1.81	1.16	80	92.8	1.35
	X	3.24	1.31	100	131.0	1.72
	W	5.17	1.41	95	134.0	1.99
				$\sum_{i=1}^{n}$	427.9	5.83
	$CF = \Sigma(FF * \Delta RT_{NDT}) \div \Sigma(FF^2) = 73.4°F$					

NOTES:
(a) WCAP-14689, Revision 4 [1]
(b) F = Fluence (10^{19} n/cm^2, E > 1.0 MeV)
(c) FF = Fluence Factor = $F^{(0.28 - 0.1 \log f)}$

Table 4-2

SCATTER OF ΔRT_{NDT} VALUES ABOUT A BEST-FIT LINE FOR SURVEILLANCE PLATE MATERIAL [a]

Lower Shell Plate B6919-1 Orientation	FF	ΔRT_{NDT} (30 ft-lb) (°F)	Best Fit ΔRT_{NDT} (°F)	Scatter of ΔRT_{NDT} (°F)
Longitudinal	0.876	85	81.7	-3.3
	1.16	105	108.2	3.2
	1.31	135	122.2	-12.8
	1.41	155	131.6	-23.4
Transverse	0.876	55	81.7	26.7
	1.16	90	108.2	18.2
	1.31	105	122.2	17.2
	1.41	145	131.6	-13.4

NOTES:
(a) WCAP-14689, Revision 4 [1]

The scatter of ΔRT_{NDT} values about a best-fit line drawn with the y-intercept equal to zero, as described in Regulatory Position 2.1, should be less than 17°F for base metal. As shown above, the scatter of four of the data points are not within 17°F of the best-fit line. Therefore, this criteria is not met for the Farley Unit 1 surveillance plate material. Since all of the data is not within 17°F of the best fit line, SNC has chosen to use the CF from this surveillance data along with a σ_{Δ} of 17°F when predicting the Farley Unit 1 vessel properties.

Table 4-3

SCATTER OF ΔRT_{NDT} VALUES ABOUT A BEST-FIT LINE FOR SURVEILLANCE WELD MATERIAL [a]

Material	FF	ΔRT_{NDT} (30 ft-lb) (°F)	Best Fit ΔRT_{NDT} (°F)	Scatter of ΔRT_{NDT} (°F)
Weld Metal	0.876	80	64.3	-15.7
	1.16	80	85.1	5.1
	1.31	100	96.2	-3.8
	1.41	95	103.5	8.5

NOTES:
(a) WCAP-14689, Revision 4 [1]

The scatter of ΔRT_{NDT} values about a best-fit line drawn with the y-intercept equal to zero, as described in Regulatory Position 2.1, is less than 28°F as shown above. Therefore, Criterion 3 is met for the Farley Unit 1 surveillance weld 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 Farley Unit 1 capsule specimens are located in the reactor between the neutron shielding pads and the vessel wall and are positioned opposite the center of the core. The test capsules are in guide tubes attached to the neutron shielding 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 and will not differ by more than 25°F. Therefore, the Farley surveillance program meets the requirements of Criterion 4.

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 Farley Unit 1 surveillance program does not include correlation monitor material. Therefore, Criterion 5 is not applicable to Farley Unit 1.

CONCLUSION:
Based on the preceding responses to the criteria of Regulatory Guide 1.99, Revision 2, Section B, and the application of engineering judgment, the Farley Unit 1 surveillance plate material data is not credible and the Farley Unit 1 surveillance weld data is credible.

5.0 Supplemental Data Tables

Table 5-1 contains a comparison of measured surveillance material 30 ft-lb transition temperature shifts and upper shelf energy decreases with Regulatory Guide 1.99, Revision 2, predictions.

Table 5-2 shows the calculation of the surveillance material chemistry factors using surveillance capsule data.

Table 5-3 provides the unirradiated Farley Unit 1 reactor vessel toughness data.

Table 5-4 provides a summary of the fluences used in the PTS evaluation.

Table 5-5 provides a summary of the adjusted reference temperatures (ARTs) of the Farley Unit 1 reactor vessel beltline materials at the 1/4-T and 3/4-T locations for 21.9 EFPY.

Table 5-6 shows the calculation of the ART at 21.9 EFPY for the limiting Farley Unit 1 reactor vessel material (lower shell plate B6919-1).

Table 5-7 provides RT_{PTS} values for Farley Unit 1 for 36 EFPY.

Table 5-1

COMPARISON OF SURVEILLANCE MATERIAL 30 FT-LB TRANSITION TEMPERATURE SHIFTS AND UPPER SHELF ENERGY DECREASES WITH REGULATORY GUIDE 1.99, REVISION 2, PREDICTIONS [a]

Material	Capsule	Fluence ($x 10^{19}$ n/cm^2, E > 1.0 MeV)	30 ft-lb Transition Temperature Shift		Upper Shelf Energy Decrease	
			Predicted (°F)	Measured (°F)	Predicted (%)	Measured (%)
Plate B6919-1 (Longitudinal)	Y	0.642	85.7	85	21	9
	U	1.81	113.7	105	27	21
	X	3.24	128.0	135	31	19
	W	5.17	137.8	155	34	22
Plate B6919-1 (Transverse)	Y	0.642	85.7	55	21	0
	U	1.81	113.7	90	27	9
	X	3.24	128.0	105	31	11
	W	5.17	137.8	145	34	16
Weld Metal	Y	0.642	68.4	80	25	13
	U	1.81	90.8	80	33	28
	X	3.24	102.2	100	38	23
	W	5.17	110.0	95	42	26
HAZ Metal	Y	0.642	--	60	--	11
	U	1.81	--	120	--	26
	X	3.24	--	125	--	19
	W	5.17	--	110	--	14

NOTES:
(a) WCAP-14689, Revision 4 [1]

Table 5-2

CALCULATION OF CHEMISTRY FACTORS USING SURVEILLANCE CAPSULE DATA [a]

Material	Capsule	f [b]	FF [c]	ΔRT_{NDT}	FF * ΔRT_{NDT}	FF^2
Lower Shell Plate B6919-1 (Longitudinal)	Y	0.642	0.876	85	74.5	0.767
	U	1.81	1.16	105	121.8	1.35
	X	3.24	1.31	135	176.9	1.72
	W	5.17	1.41	155	218.6	1.99
Lower Shell Plate B6919-1 (Transverse)	Y	0.642	0.876	55	48.2	0.767
	U	1.81	1.16	90	104.4	1.35
	X	3.24	1.31	105	137.6	1.72
	W	5.17	1.41	145	204.5	1.99
				Sum:	1086.5	11.65
	Chemistry Factor (CF) = Σ (FF * ΔRT_{NDT}) ÷ Σ (FF^2) = 93.3°F					
Weld Metal [d]	Y	0.642	0.876	129.6	113.5	0.767
	U	1.81	1.16	129.6	150.3	1.35
	X	3.24	1.31	162.0	212.2	1.72
	W	5.17	1.41	153.9	217.0	1.99
				Sum:	693.0	5.83
	Chemistry Factor (CF) = Σ (FF * ΔRT_{NDT}) ÷ Σ (FF^2) = 118.9°F					

NOTES:

(a) WCAP-14689, Revision 4 [1]

(b) f = fluence (x 10^{19} n/cm^2, E > 1.0 MeV)

(c) FF = fluence factor = $f^{(0.28 - 0.1 \log f)}$

(d) ΔRT_{NDT} values were multiplied by a ratio factor of 1.62 (CF_{vessel} ÷ $CF_{surv\ weld}$ = 126.2 ÷ 78.1 = 1.62)

Table 5-3

REACTOR VESSEL TOUGHNESS TABLE (UNIRRADIATED) [a]

Beltline Material	Cu Weight %	Ni Weight %	IRT_{NDT} (°F)
Closure Head Flange	--	--	60
Vessel Flange	--	--	60
Intermediate Shell Plate B6903-2	0.13	0.60	0
Intermediate Shell Plate B6903-3	0.12	0.56	10
Lower Shell Plate B6919-1	0.14	0.55	15
Lower Shell Plate B6919-2	0.14	0.56	5
Intermediate Shell Longitudinal Weld Seams 19-894 A & B [b] (Heat # 33A277)	0.258	0.165	-56
Surveillance Weld [c]	0.14	0.19	--
Circumferential Weld Seam 11-894 [b] (Heat # 6329637)	0.205	0.105	-56
Lower Shell Longitudinal Weld Seams 20-894 A & B [b] (Heat # 90099)	0.197	0.060	-56

NOTES:
(a) WCAP-14689, Revision 4 [1]
(b) Best-estimate copper and nickel from CE NPSD-1039 [9]
(c) The surveillance weld is representative of intermediate shell longitudinal welds 19-894 A & B. Best-estimate copper and nickel values represent a single chemical analysis documented in WCAP-8810 [5]

Table 5-4

REACTOR VESSEL FLUENCE PROJECTIONS FOR 36 EFPY [a, b]

EFPY	0°	15°	15° [c]	30°	30° [c]	45°
36	4.34	2.68	2.14	2.01	1.93	1.35

NOTES:
(a) WCAP-14689, Revision 4 [1]
(b) Fluence in 10^{19} n/cm^2 (E > 1.0 MeV)
(c) Indicates location in octants with a 26° neutron pad span.

Table 5-5

SUMMARY OF ADJUSTED REFERENCE TEMPERATURES (ARTs) FOR REACTOR VESSEL BELTLINE MATERIALS AT THE 1/4-T AND 3/4-T LOCATIONS FOR 21.9 EFPY [a, b]

Material	1/4-T (°F)	3/4-T (°F)
Intermediate Shell Plate B6903-2	138	114
Intermediate Shell Plate B6903-3	138	115
Lower Shell Plate B6919-1	161 [c]	135 [c]
Lower Shell Plate B6919-1 Using S/C Data	156	131
Lower Shell Plate B6919-2	151	126
Intermediate Shell Longitudinal Weld Seams 19-894 A & B (Heat # 33A277)	113 [d]	82 [d]
Intermediate Shell Longitudinal Weld Seams 19-894 A & B (Heat # 33A277) Using S/C Data	85 [d]	56 [d]
Circumferential Weld 11-894 (Heat # 6329637)	123	97
Lower Shell Longitudinal Weld Seams 20-894 A & B (Heat # 90099)	85 [d]	62 [d]

NOTES:

(a) WCAP-14689, Revision 4 [1]

(b) The ARTs presented here are based on the peak reactor vessel surface fluence of 2.718×10^{19} n/cm^2 (E > 1.0 MeV) unless otherwise noted.

(c) Limiting 1/4-T and 3/4-T ART values. The P/T limit curves are those previously generated based on a 1/4T ART of 161°F and a 3/4-T ART value of 136°F which bounds the limiting 1/4-T and 3/4-T ARTs shown above.

(d) ARTs calculated using the peak vessel fluence of 0.8307×10^{19} n/cm^2 (E > 1.0 MeV) at 45°

Table 5-6

CALCULATION OF ADJUSTED REFERENCE TEMPERATURE AT 21.9 EFPY FOR THE LIMITING REACTOR VESSEL MATERIAL - LOWER SHELL PLATE B6919-1 [a]

Parameter		
Operating Period	21.9 EFPY	
Location	1/4-T	3/4-T
Chemistry Factor, CF (°F)	97.8	97.8
Fluence, f (10^{19} n/cm^2) [b]	1.695	0.659
Fluence Factor, FF	1.145	0.883
ΔRT_{NDT} = CF x FF (°F)	112.0	86.4
Initial RT_{NDT}, I (°F)	15	15
Margin, M (°F)	34	34
Adjusted Reference Temperature (ART), (°F) per Regulatory Guide 1.99, Revision 2	161	135

NOTES:
(a) WCAP-14689, Revision 4 [1]
(b) Fluence is based on f_{surf} (10^{19} n/cm^2, E > 1.0 MeV) = 2.718 at 21.9 EFPY. The Farley Unit 1 reactor vessel wall thickness is 7.875 inches in the beltline region.

Table 5-7

PRESSURIZED THERMAL SHOCK (RT_{PTS}) VALUES FOR 36 EFPY [a]

Material	CF	Surface Fluence (10^{19} n/cm^2, E > 1.0 MeV)	FF	ΔRT_{NDT} (CF x FF) (°F)	I (°F)	M (°F)	RT_{PTS} (°F)
Intermediate Shell Plate B6903-2	91.0	4.34	1.374	125.0	0	34	159
Intermediate Shell Plate B6903-3	82.2	4.34	1.374	112.9	10	34	157
Lower Shell Plate B6919-1	97.8	4.34	1.374	134.4	15	34	183
Lower Shell Plate B6919-1 Using S/C Data	93.3	4.34	1.374	128.2	15	34 [b]	177
Lower Shell Plate B6919-2	98.2	4.34	1.374	134.9	5	34	174
Intermediate Shell Longitudinal Welds 19-894 A & B (Heat # 33A277)	126.2	1.35	1.083	136.7	-56	66	147
Intermediate Shell Longitudinal Welds 19-894 A & B (Heat # 33A277) Using S/C Data	118.9	1.35	1.083	128.8	-56	44	117
Circumferential Weld 11-894 (Heat # 6329637)	98.4	4.34	1.374	135.2	-56	66	145
Lower Shell Longitudinal Welds 20-894 A & B (Heat # 90099)	91.4	1.35	1.083	99.0	-56	66	109

NOTES:
(a) WCAP-14689, Revision 4 [1]
(b) σ_Δ = 17°F since the plate surveillance data did not meet credibility criteria

9 References

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