ENCLOSURE 6

Joseph M. Farley Nuclear Plant

Unit 1

Pressure Temperature Limits Report

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Pressure Temperature Limits Report

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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 (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. These limits have been developed using the NRC-approved methodologies specified in TS 6.9.1.15 with the exception that low temperature overpressure protection (LTOP) is provided by the RHR relief valves (RHRRVs) in lieu of the PORVs. Therefore, the operability requirements associated with LTOP will be retained in TS LCO 3.4.10.3. 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 60°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.

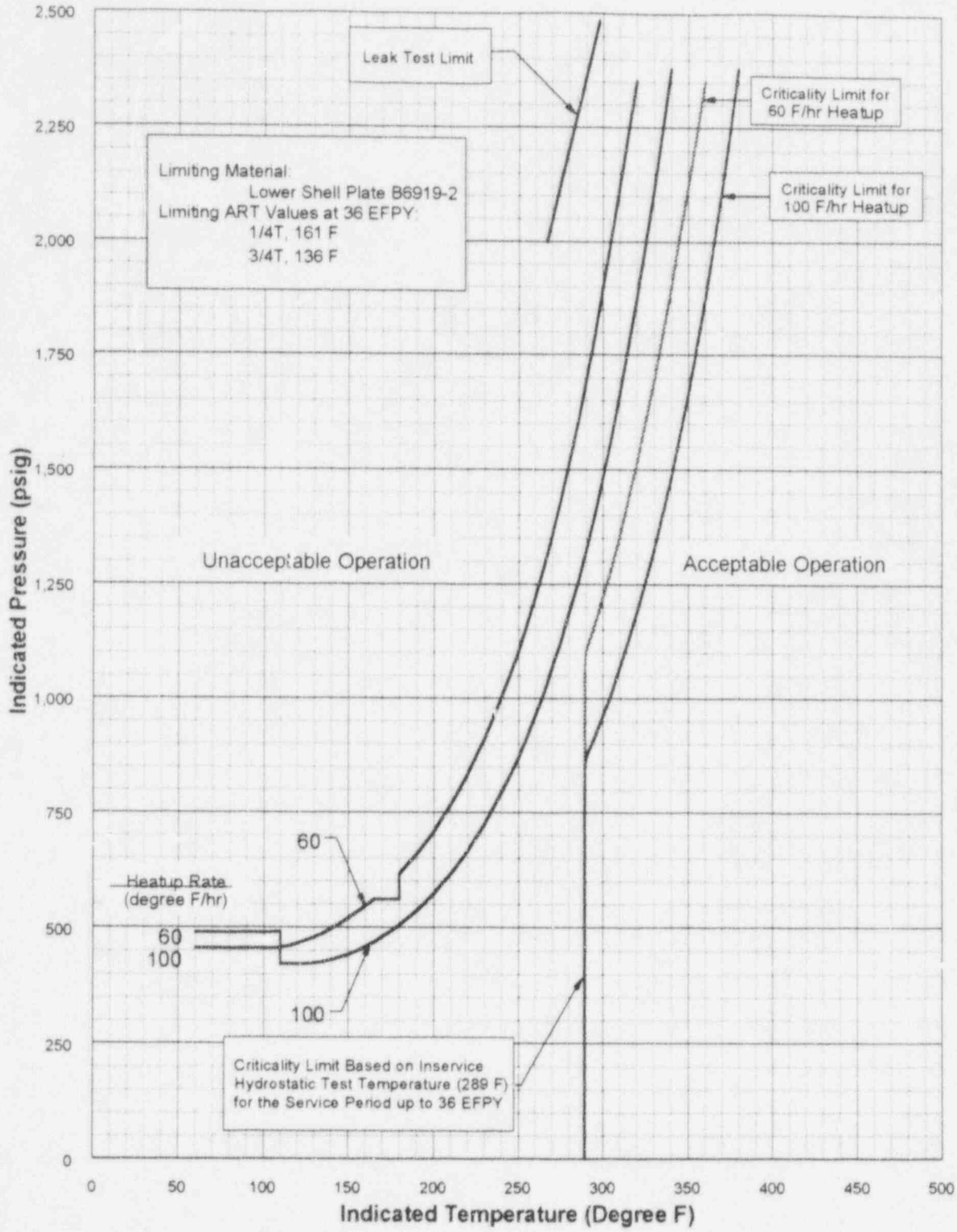

Figure 2-1

Farley Unit 1 Reactor Coolant System Heatup Limitations (Heatup Rates up to 100°F/hr) Applicable to 36 EFPY (Without Margins for Instrument Errors). Includes vessel flange requirements of 180°F and 561 psig per 10 CFR 50, Appendix G. [1]

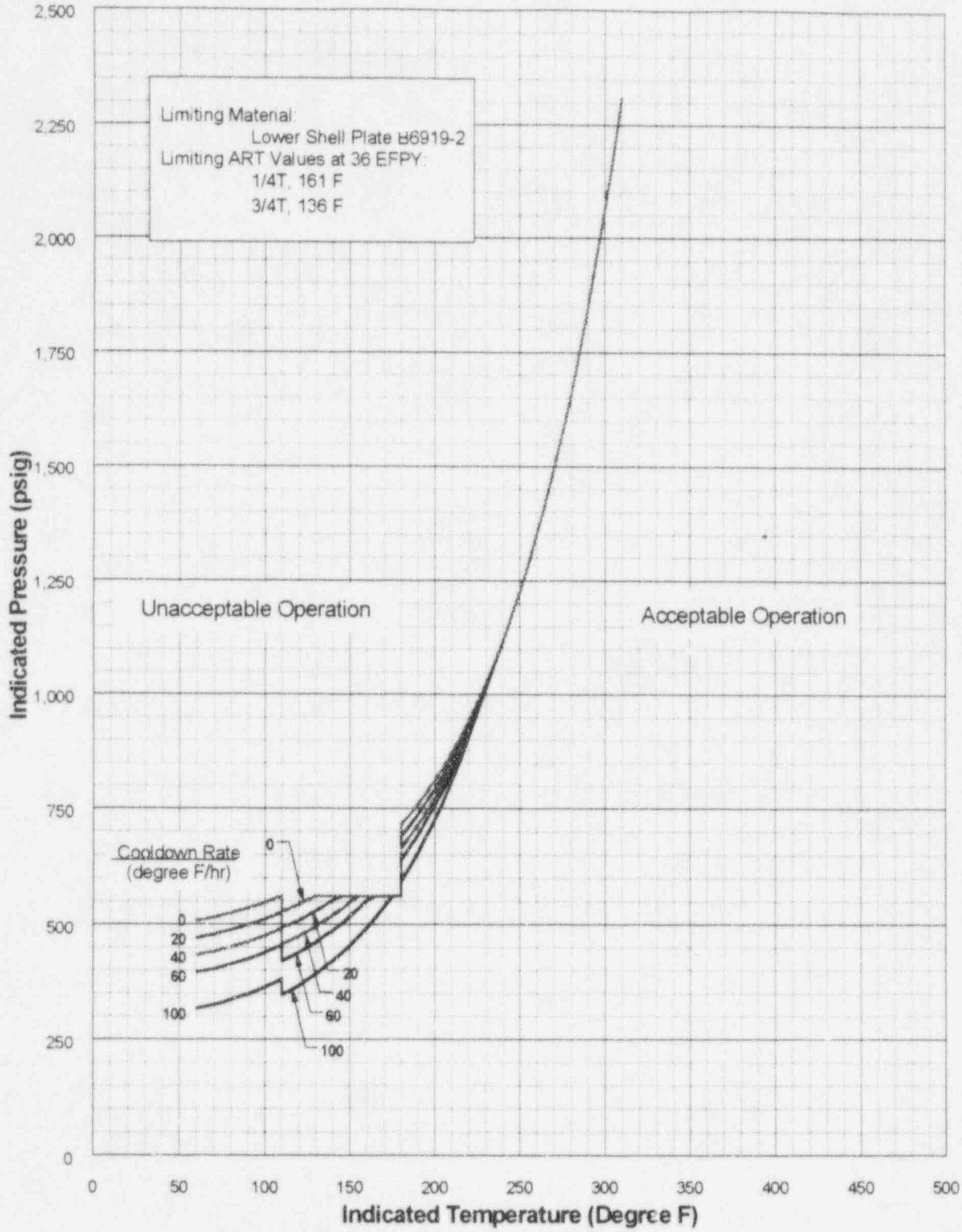

Figure 2-2

Farley Unit 1 Reactor Coolant System Cooldown Limitations (Cooldown Rates up to 100°F/hr) Applicable to 36 EFPY (Without Margins for Instrument Errors). 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
60	490	289	0	60	457	289	0	267	2000
65	490	289	480	65	457	289	481	298	2485
70	490	289	469	70	457	289	466		
75	490	289	462	75	457	289	454		
80	490	289	457	80	457	289	444		
85	490	289	455	85	457	289	436		
90	490	289	456	90	457	289	430		
95	490	289	458	95	457	289	426		
100	490	289	461	100	457	289	423		
105	491	289	466	105	457	289	422		
110	493	289	473	110	457	289	423		
110	458	289	480	110	422	289	424		
115	461	289	488	115	422	289	427		
120	466	289	498	120	422	289	432		
125	473	289	509	125	423	289	437		
130	480	289	521	130	424	289	443		
135	488	289	534	135	427	289	451		
140	498	289	547	140	432	289	460		
145	509	289	563	145	437	289	469		
150	521	289	579	150	443	289	481		
155	534	289	597	155	451	289	493		
160	547	289	617	160	460	289	506		
165	561	289	637	165	469	289	521		
170	561	289	660	170	481	289	537		
175	561	289	684	175	493	289	555		
180	561	289	710	180	506	289	574		
180	617	289	738	185	521	289	595		
185	637	289	768	190	537	289	617		
190	660	289	800	195	555	289	641		
195	684	289	835	200	574	289	668		
200	710	289	873	205	595	289	696		
205	738	289	913	210	617	289	726		
210	768	289	956	215	641	289	759		
215	800	289	1002	220	668	289	794		
220	835	289	1051	225	696	289	832		
225	873	290	1105	230	726	290	872		
230	913	295	1162	235	759	295	916		
235	956	300	1223	240	794	300	963		
240	1002	305	1288	245	832	305	1013		
245	1051	310	1358	250	872	310	1068		
250	1105	315	1434	255	916	315	1125		
255	1162	320	1515	260	963	320	1188		
260	1223	325	1601	265	1013	325	1254		
265	1288	330	1694	270	1068	330	1326		
270	1358	335	1793	275	1125	335	1402		
275	1434	340	1899	280	1188	340	1484		
280	1515	345	2012	285	1254	345	1572		
285	1601	350	2120	290	1326	350	1666		
290	1694	355	2233	295	1402	355	1767		
295	1793	360	2354	300	1484	360	1874		
300	1899			305	1572	365	1988		
305	2012			310	1666	370	2111		
310	2120			315	1767	375	2241		
315	2233			320	1874	380	2380		
320	2354			325	1988				
				330	2111				
				335	2241				
				340	2380				

Table 2-1

Farley Unit 1 36 EFPY Heatup Curve Data Points (Without Margins for Instrument Errors) [1]

0 °F		20 °F		40 °F		60 °F		100 °F	
T	P	T	P	T	P	T	P	T	P
60	508	60	471	60	434	60	396	60	317
65	512	65	475	65	438	65	400	65	321
70	516	70	479	70	442	70	404	70	326
75	521	75	484	75	447	75	409	75	331
80	525	80	489	80	452	80	414	80	337
85	531	85	494	85	457	85	420	85	343
90	536	90	500	90	463	90	426	90	349
95	542	95	506	95	470	95	433	95	357
100	549	100	513	100	477	100	440	100	365
105	556	105	520	105	484	105	448	105	373
110	561	110	528	110	492	110	456	110	382
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	813	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 36 EFPY Cooldown Curve Data Points (Without Margins for Instrument Errors)[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 [c] (n/cm^2)
Y [d]	343	3.33	1.13	5.80×10^{18}
U [d]	107	3.34	3.02	1.69×10^{19}
X [d]	287	3.38	6.12	2.95×10^{19}
W [d,e]	110	3.13	12.43	3.82×10^{19}
V	290	3.11	Standby	--
Z	340	3.11	Standby	--

NOTES:

(a) WCAP-14689 [1]

(b) Effective Full Power Years (EFPY) from plant startup

(c) Fluence recalculated using methodology contained in WCAP-14040-NP-A, Revision 2.[4]

(d) Plant-specific evaluation

(e) Final capsule withdrawal required by ASTM E-185-82.

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.

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, therefore, 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. 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	0.580	0.848	85	72.0	0.718
	U	1.69	1.14	105	120.2	1.31
	X	2.95	1.29	135	173.7	1.66
	W	3.82	1.35	155	208.7	1.81
Lower Shell Plate B6919-1 (Transverse)	Y	0.580	0.848	55	46.6	0.718
	U	1.69	1.14	90	103.0	1.31
	X	2.95	1.29	105	135.1	1.66
	W	3.82	1.35	145	195.2	1.81
	$\sum$		9.25	875	1054.5	10.99
Weld Metal	Y	0.580	0.848	80	67.8	0.718
	U	1.69	1.14	80	91.6	1.31
	X	2.95	1.29	100	129.0	1.66
	W	3.82	1.35	95	127.9	1.81
	$\sum$		4.62	355	415.9	5.50

NOTES:

(a) WCAP-14689 [1]

(b) F = Fluence (10^{19} n/cm^2, E > 1.0 MeV)

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

Per the 27th Edition of the CRC Standard Mathematical Tables (page 497), for a straight line fit by the method of least squares, the values b_0 and b_1 are obtained by solving the normal equations

$$nb_0 + b_1 \sum x_i = \sum y_i$$

and

$$b_0 \sum x_i + b_1 \sum x_i^2 = \sum x_i y_i$$

These equations can be re-written as follows:

$$\sum_{i=1}^{n} y_i = an + b \sum_{i=1}^{n} x_i$$

and

$$\sum_{i=1}^{n} x_i y_i = a \sum_{i=1}^{n} x_i + b \sum_{i=1}^{n} x_i^2$$

Lower Shell Plate B6919-1:

Based on the data provided in Table 4-1, these equations become:

$$875 = 8a + 9.25b$$

and

$$1054.5 = 9.25a + 10.99b$$

Thus, b=145.2 and a=-58.5, and the equation of the straight line which provides the best fit in the sense of least squares is:

$$Y' = 145.2\,(X) - 58.5$$

The error in predicting a value of Y corresponding to a given X value is e = Y - Y'.

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	0.848	85	64.6	20.4
	1.14	105	107.0	-2.0
	1.29	135	128.8	6.2
	1.35	155	137.5	17.5
Transverse	0.848	55	64.6	-9.6
	1.14	90	107.0	-17.0
	1.29	105	128.8	-23.8
	1.35	145	137.5	7.5

NOTES:
(a) WCAP-14689 [1]

The scatter of ΔRT_{NDT} values about a best-fit line drawn, as described in Regulatory Position 2.1, should be less than 17°F for base metal. However, even if the fluence range is large, the scatter should not exceed twice this value (i.e., 34°F). As shown above, the error is within 34°F of the best-fit line. Therefore, Criterion 3 is met for the Farley Unit 1 surveillance plate material.

Weld Metal:
Based on the data provided in Table 4-1, the equation becomes:

$$355 = 4a + 4.62b$$

and

$$415.9 = 4.62a + 5.5b$$

Thus, b=35.6 and a=47.625, and the equation of the straight line which provides the best fit in the sense of least squares is:

$$Y' = 35.6\,(X) + 47.625$$

The error in predicting a value of Y corresponding to a given X value is e = Y - Y'.

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	0.848	80	77.8	2.2
	1.14	80	88.2	-8.2
	1.29	100	93.5	6.5
	1.35	95	95.7	-0.7

NOTES:
(a) WCAP-14689[1]

The scatter of ΔRT_{NDT} values about a best-fit line drawn, 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 reactor vessel 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 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 generation of the heatup and cooldown curves and 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 36 EFPY.

Table 5-6 shows the calculation of the ART at 36 EFPY for the limiting Farley Unit 1 reactor vessel material (lower shell plate B6919-2).

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 SHIFT AND UPPER SHELF ENERGY DECREASE 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	0.580	83	85	20	9
	U	1.69	112	105	26.5	21
	X	2.95	126	135	30	19
	W	3.82	132	155	32	22
Plate B6919-1 (Transverse)	Y	0.580	83	55	20	0
	U	1.69	112	90	26.5	9
	X	2.95	126	105	30	11
	W	3.82	132	145	32	16
Weld Metal	Y	0.580	101	80	34	13
	U	1.69	136	80	44	28
	X	2.95	153	100	49	23
	W	3.82	160	95	52	26
HAZ Metal	Y	0.580	--	60	--	11
	U	1.69	--	120	--	26
	X	2.95	--	125	--	19
	W	3.82	--	110	--	14

NOTES:
(a) WCAP-14689 [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	0.580	0.848	85	72.0	0.718
	U	1.69	1.14	105	120.2	1.31
	X	2.95	1.29	135	173.7	1.66
	W	3.82	1.35	155	208.7	1.81
Lower Shell Plate B6919-1 (Transverse)	Y	0.580	0.848	55	46.6	0.718
	U	1.69	1.14	90	103.0	1.31
	X	2.95	1.29	105	135.1	1.66
	W	3.82	1.35	145	195.2	1.81
				Sum:	1054.5	10.99
	Chemistry Factor = Σ (FF * ΔRT_{NDT}) ÷ Σ (FF^2) = 95.9					
Weld Metal	Y	0.580	0.848	80	67.8	0.718
	U	1.69	1.14	80	91.6	1.31
	X	2.95	1.29	100	128.7	1.66
	W	3.82	1.35	95	127.9	1.81
				Sum:	415.9	5.50
	Chemistry Factor = Σ (FF * ΔRT_{NDT}) ÷ Σ (FF^2) = 75.7					

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

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
Surveillance Plate [b]	0.14	0.55	15
Lower Shell Plate B6919-2	0.14	0.56	5
Intermediate Shell Longitudinal Weld Seams 19-894 A & B [c] (Heat # 33A277)	0.24	0.17	-56
Surveillance Weld [d]	0.24	0.17	-56
Circumferential Weld Seam 11-894 [c] (Heat # 6329637)	0.21	0.11	-56
Lower Shell Longitudinal Weld Seams 20-894 A & B [c] (Heat # 90099)	0.20	0.20	-56

NOTES:
(a) WCAP-14689 [1]
(b) The surveillance plate is representative of lower shell plate B6919-1
(c) Generic Letter 92-01, Revision 1, Supplement 1 response
(d) The surveillance weld is representative of intermediate shell longitudinal welds 19-894 A & B

Table 5-4

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

EFPY	0°	15°	15° [c]	30°	30° [c]	45°
36	3.97	2.45	1.96	1.83	1.77	1.23

NOTES:
(a) WCAP-14689 [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 36 EFPY [a, b]

Material	1/4-T (°F)	3/4-T (°F)
Intermediate Shell Plate B6903-2	147	124
Intermediate Shell Plate B6903-3	146	125
Lower Shell Plate B6919-1	171	146
Lower Shell Plate B6919-1 Using S/C Data [c]	151	127
Lower Shell Plate B6919-2	161 [d]	136 [d]
Intermediate Shell Longitudinal Weld Seams 19-894 A & B (Heat # 33A277)	120 [e]	89 [e]
Intermediate Shell Longitudinal Weld Seams 19-894 A & B (Heat # 33A277) Using S/C Data [c]	58 [e]	39 [e]
Circumferential Weld 11-894 (Heat # 6329637)	135	110
Lower Shell Longitudinal Weld Seams 20-894 A & B (Heat # 90099)	106 [e]	80 [e]

NOTES:

(a) WCAP-14689 [1]

(b) The ARTs presented here are based on the peak reactor vessel surface fluence of 3.97×10^{19} n/cm^2 (E > 1.0 MeV) unless otherwise noted.

(c) Based on surveillance capsule data contained in WCAP-14196 [2]

(d) ART values used to generate heatup and cooldown curves.

(e) ARTs calculated using the peak vessel fluence of 1.23×10^{19} n/cm^2 (E > 1.0 MeV) at 45°

Table 5-6

CALCULATION OF ADJUSTED REFERENCE TEMPERATURE AT 36 EFPY FOR THE LIMITING REACTOR VESSEL MATERIAL - LOWER SHELL PLATE B6919-2 [a]

Parameter		
Operating Period	36 EFPY	
Location	1/4-T	3/4-T
Chemistry Factor, CF (°F)	98.2	98.2
Fluence, f (10^{19} n/cm^2) [b]	2.48	0.962
Fluence Factor, FF	1.24	0.989
ΔRT_{NDT} = CF x FF (°F)	122	97
Initial RT_{NDT}, I (°F)	5	5
Margin, M (°F) [c]	34	34
Adjusted Reference Temperature (ART), (°F) per Regulatory Guide 1.99, Revision 2	161	136

NOTES:

(a) WCAP-14689 [1]

(b) Fluence is based on f_{surf} (10^{19} n/cm^2, E > 1.0 MeV) = 3.97 at 36 EFPY. The Farley Unit 1 reactor vessel wall thickness is 7.875 inches in the beltline region.

(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 the ΔRT_{NDT} term, σ_Δ, is 17°F for the plate, except that σ_Δ need not exceed 0.5 times the mean value of ΔRT_{NDT}. In accordance with Regulatory Guide 1.99, Revision 2, Position 2.1, values of σ_Δ may be cut in half when based on credible surveillance data.

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	3.97	1.35	123.3	0	34	157
Intermediate Shell Plate B6903-3	82.2	3.97	1.35	111.3	10	34	155
Lower Shell Plate B6919-1	97.8	3.97	1.35	132.5	15	17	164
Lower Shell Plate B6919-1 Using S/C Data	95.9	3.97	1.35	129.9	15	34	179
Lower Shell Plate B6919-2	98.2	3.97	1.35	133.0	5	34	172
Intermediate Shell Longitudinal Welds 19-894 A & B (Heat # 33A277)	118.6	1.23	1.06	125.4	-56	66	135
Intermediate Shell Longitudinal Welds 19-894 A & B (Heat # 33A277) Using S/C Data	75.7	1.23	1.06	80.1	-56	44	68
Circumferential Weld 11-894 (Heat # 6329637)	100.8	3.97	1.35	136.5	-56	66	147
Lower Shell Longitudinal Welds 20-894 A & B (Heat # 90099)	104.0	1.23	1.06	110.0	-56	66	120

NOTES:
(a) WCAP-14689 [1]

6.0 References

1. WCAP-14689, Revision 1, Farley Units 1 and 2 Heatup and Cooldown Limit Curves for Normal Operation and PTLR Support Documentation, E. Terek, April 1997.
2. WCAP-14196, Analysis of Capsule W from the Alabama Power Company Farley Unit 1 Reactor Vessel Radiation Surveillance Program, P. A. Peters, et al., February 1995.
3. WCAP-14687, Joseph M. Farley Units 1 and 2 Radiation Analysis and Neutron Dosimetry Evaluation, R. L. Bencini, June 1996.
4. WCAP-14040-NP-A, Revision 2, Methodology Used to Develop Cold Overpressure Mitigating System Setpoints and RCS Heatup and Cooldown Limit Curves, January 1996.
5. WCAP-8810, Southern Alabama Power Company Joseph M. Farley Nuclear Plant Unit No. 1 Reactor Vessel Radiation Surveillance Program, J. A. Davidson, et al., December 1976
6. WCAP-9717, Analysis of Capsule Y from the Alabama Power Company Farley Unit No. 1 Reactor Vessel Radiation Surveillance Program, S. E. Yanichko, et al., June 1980.
7. WCAP-10474, Analysis of Capsule U from the Alabama Power Company Joseph M. Farley Unit 1 Reactor Vessel Radiation Surveillance Program, R. S. Boggs, et al., February 1984.