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U. S. Nuclear Regulatory Commission
ATTN: Document Control Desk
Washington, D. C. 20555-0001

Joseph M. Farley Nuclear Plant - Unit 1
Submittal of Revision 2 of the Pressure Temperature Limits Report

Ladies and Gentlemen:

In accordance with Section 5.6.6 of the Joseph M. Farley Nuclear Plant (FNP) Unit 1 Technical Specifications, Southern Nuclear Operating Company (SNC) hereby submits Revision 2 of the FNP Unit 1 Pressure Temperature Limits Report (PTLR). Changes include revision of the surveillance capsule withdrawal schedule (as approved by NRC letter dated March 15, 2004), replacement of the heatup and cooldown limit curves with new curves extending the service period to 29.0 effective full power years (EFPY), and updating of the surveillance data credibility analysis and supplemental data sections to reflect the results of the latest surveillance capsule analysis report (WCAP-16221-NP, Rev. 0, submitted April 16, 2004).

This letter contains no NRC commitments. If you have any questions, please advise.

Sincerely,

L. M. Stinson

LMS/DWD/sdl

Enclosure: Joseph M. Farley Nuclear Plant Unit 1 Pressure Temperature Limits Report, Revision 2

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A001

Joseph M. Farley Nuclear Plant

Unit 1

Pressure Temperature Limits Report

REVISION 2

APPROVED FOR ISSUE

10-28-04

OPERATIONS MANAGER 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) 5.6.6. Revisions to the PTLR shall be provided to the NRC after issuance.

This report affects TS 3.4.3, RCS Pressure/Temperature Limits (P/T) Limits. All TS requirements associated with low temperature overpressure protection (LTOP) are contained in TS 3.4.12, RCS Overpressure Protection Systems.

2.0 Operating Limits

The limits for TS 3.4.3 are presented in the subsection which follows and were developed using the methodologies specified in TS 5.6.6. The operability requirements associated with LTOP are specified in TS LCO 3.4.12 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 5.6.6. 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 – 3.4.3)

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.

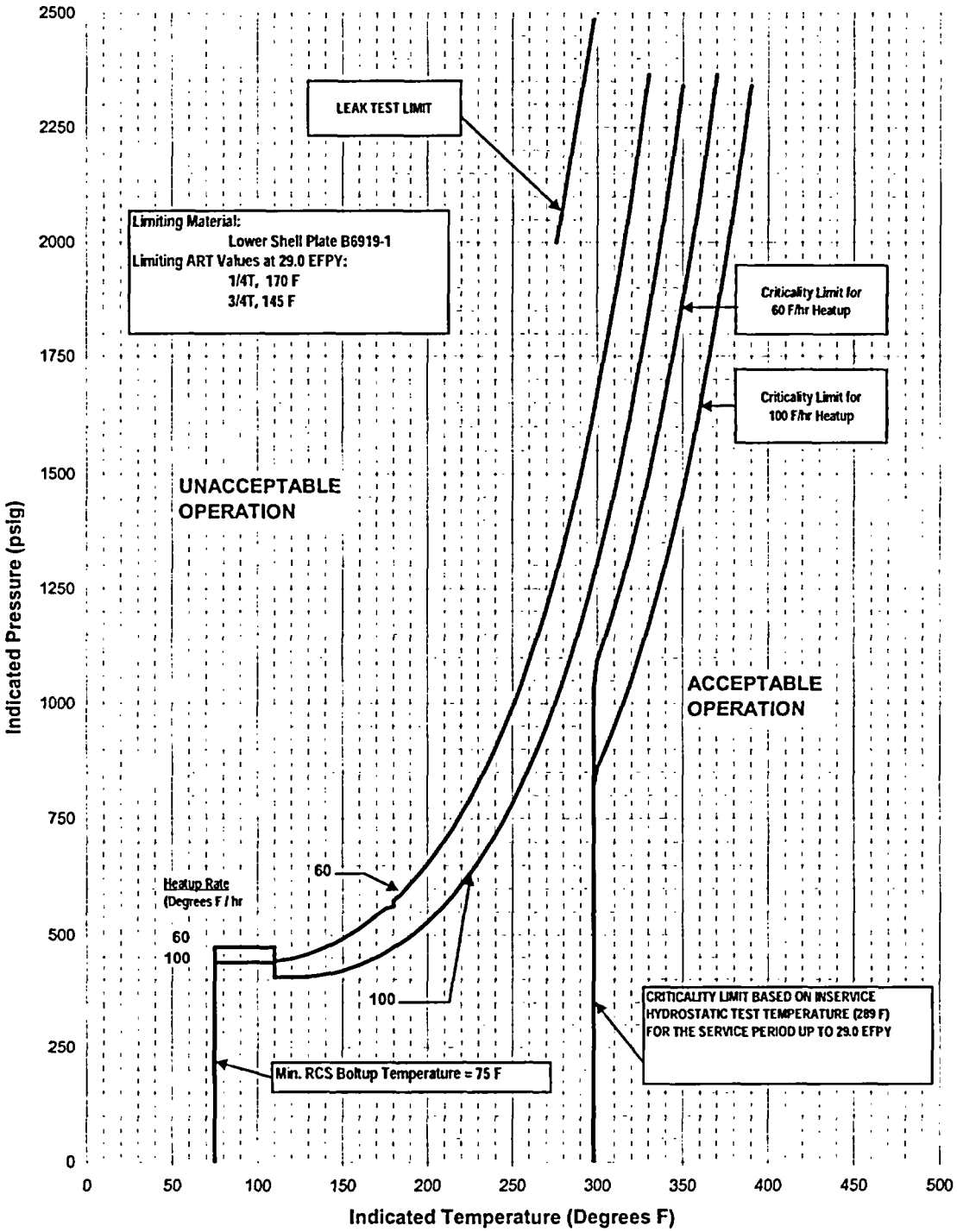

Figure 2-1

Farley Unit 1 Reactor Coolant System Heatup Limitations (Heatup Rates up to 100°F/hr) Applicable to 29.0 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]

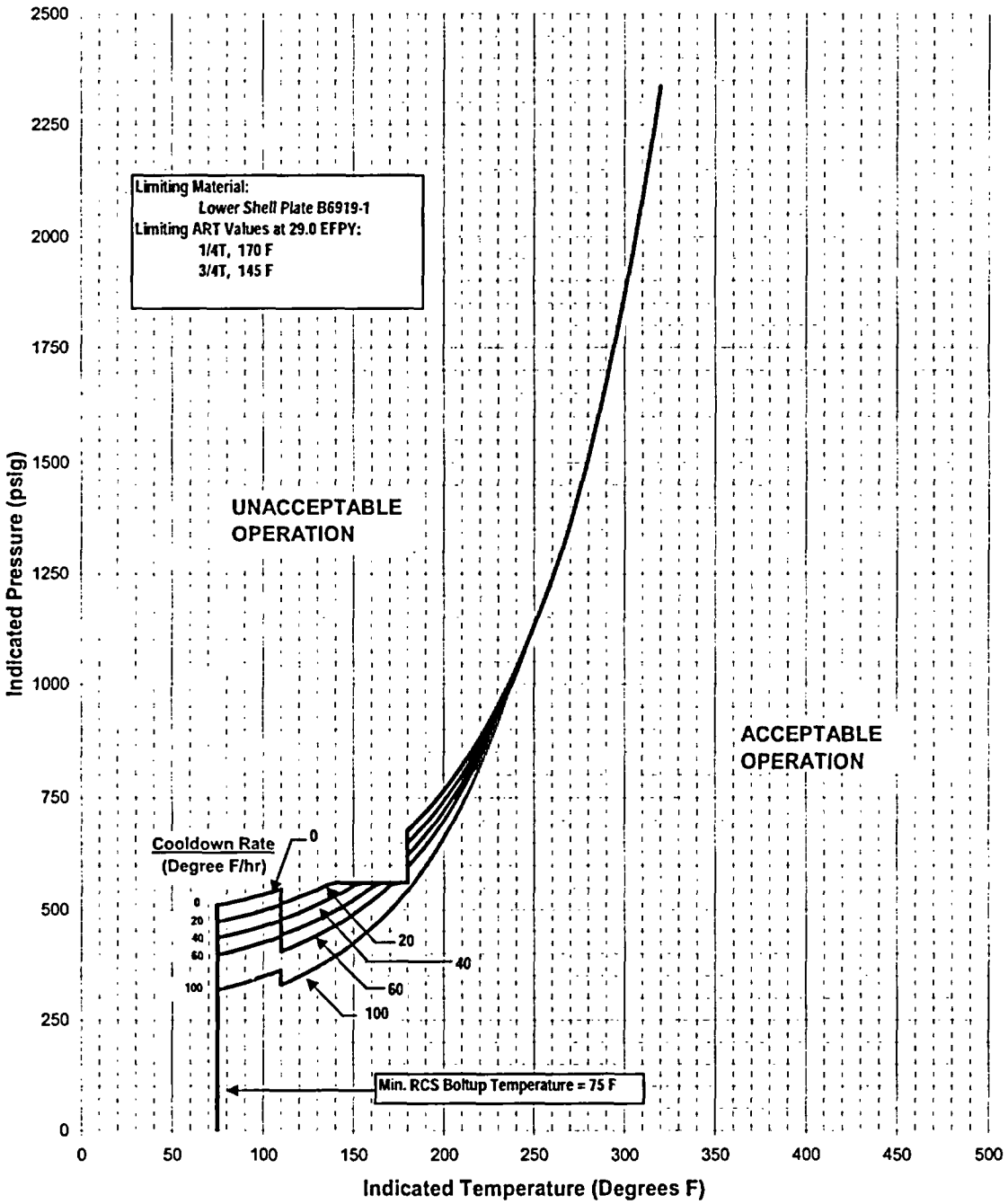

Figure 2-2

Farley Unit 1 Reactor Coolant System Cooldown Limitations (Cooldown Rates up to 100°F/hr) Applicable to 29.0 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 Limit	
T	P	T	P	T	P	T	P	T	P
75	473	298	0	75	438	298	0	276	2000
80	473	298	467	80	438	298	469	298	2485
85	473	298	456	85	438	298	454		
90	473	298	448	90	438	298	441		
95	473	298	443	95	438	298	430		
100	473	298	441	100	438	298	422		
105	473	298	440	105	438	298	415		
110	474	298	441	110	438	298	411		
110	441	298	444	110	405	298	407		
115	444	298	447	115	405	298	406		
120	447	298	452	120	405	298	405		
125	452	298	458	125	405	298	406		
130	458	298	465	130	406	298	408		
135	465	298	473	135	408	298	411		
140	473	298	482	140	411	298	415		
145	482	298	492	145	415	298	420		
150	492	298	503	150	420	298	426		
155	503	298	515	155	426	298	433		
160	515	298	528	160	433	298	441		
165	528	298	542	165	441	298	450		
170	542	298	557	170	450	298	460		
175	557	298	574	175	460	298	472		
180	561	298	591	180	472	298	484		
180	574	298	611	185	484	298	497		
185	591	298	631	190	497	298	512		
190	611	298	653	195	512	298	528		
195	631	298	677	200	528	298	546		
200	653	298	703	205	546	298	565		
205	677	298	730	210	565	298	585		
210	703	298	760	215	585	298	608		
215	730	298	791	220	608	298	632		
220	760	298	826	225	632	298	657		
225	791	298	862	230	657	298	685		
230	826	298	902	235	685	298	715		
235	862	298	944	240	715	298	747		
240	902	298	989	245	747	298	782		
245	944	298	1038	250	782	298	819		
250	989	300	1091	255	819	300	859		
255	1038	305	1147	260	859	305	902		
260	1091	310	1207	265	902	310	948		
265	1147	315	1271	270	948	315	998		
270	1207	320	1341	275	998	320	1051		
275	1271	325	1415	280	1051	325	1108		
280	1341	330	1494	285	1108	330	1169		
285	1415	335	1579	290	1169	335	1234		
290	1494	340	1670	295	1234	340	1305		
295	1579	345	1768	300	1305	345	1380		
300	1670	350	1872	305	1380	350	1460		
305	1768	355	1983	310	1460	355	1546		
310	1872	360	2102	315	1546	360	1639		
315	1983	365	2229	320	1639	365	1737		
320	2102	370	2364	325	1737	370	1843		
325	2229			330	1843	375	1956		
330	2364			335	1956	380	2076		
				340	2076	385	2204		
				345	2204	390	2341		
				350	2341				

Table 2-1

Farley Unit 1 29.0 EFPY Heatup Curve Data Points (adjusted to include 60 psi ΔP at RCS temperatures ≥ 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	511	75	474	75	436	75	398	75	318
80	515	80	478	80	440	80	402	80	323
85	520	85	483	85	445	85	407	85	329
90	524	90	488	90	450	90	412	90	334
95	530	95	493	95	456	95	418	95	341
100	535	100	499	100	462	100	424	100	347
105	541	105	505	105	469	105	431	105	355
110	548	110	512	110	476	110	439	110	363
110	515	110	479	110	443	110	406	110	330
115	522	115	486	115	450	115	413	115	338
120	530	120	494	120	458	120	422	120	348
125	538	125	503	125	467	125	431	125	358
130	546	130	512	130	477	130	441	130	369
135	556	135	522	135	487	135	452	135	381
140	561	140	532	140	498	140	464	140	394
145	561	145	543	145	510	145	476	145	408
150	561	150	556	150	523	150	490	150	423
155	561	155	561	155	537	155	505	155	440
160	561	160	561	160	552	160	520	160	458
165	561	165	561	165	561	165	537	165	477
170	561	170	561	170	561	170	556	170	497
175	561	175	561	175	561	175	561	175	520
180	561	180	561	180	561	180	561	180	544
180	676	180	651	180	624	180	597	185	570
185	698	185	672	185	645	185	620	190	598
190	719	190	694	190	669	190	644	195	629
195	741	195	717	195	694	195	671	200	661
200	765	200	743	200	721	200	700	205	696
205	791	205	770	205	750	205	731	210	734
210	819	210	799	210	781	210	764	215	775
215	849	215	831	215	815	215	800	220	819
220	880	220	865	220	851	220	838	225	867
225	915	225	902	225	890	225	880	230	918
230	952	230	941	230	932	230	925	235	973
235	991	235	983	235	977	235	973	240	1024
240	1034	240	1028	240	1025	240	1024		
245	1080	245	1077	245	1077				
250	1129								
255	1182								
260	1238								
265	1299								
270	1364								
275	1434								
280	1509								
285	1589								
290	1675								
295	1768								
300	1867								
305	1972								
310	2086								
315	2206								
320	2336								

Table 2-2

Farley Unit 1 29.0 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	3.24	1.15	6.12×10^{18}
U [c]	107	3.34	3.08	1.73×10^{19}
X [c]	287	3.35	6.11	3.06×10^{19}
W [c]	110	3.01	12.43	4.75×10^{19} [d]
V [c]	290	3.04	20.16	7.14×10^{19} [e]
Z	340	3.04	~24	8.44×10^{19} [f][g]

NOTES:

(a) WCAP-16221-NP, Revision 0 [10]

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

(c) Plant-specific evaluation

(d) This fluence is not less than once or greater than twice the peak EOL fluence for the initial 40-year license term.

(e) This fluence is not less than once or greater than twice the peak EOL fluence for a 20-year license renewal term to 60 years.

(f) This projected fluence is not less than once or greater than twice the peak EOL fluence for an additional 20-year license renewal term to 80 years.

(g) This projected fluence was obtained using WCAP-16221-NP, Rev. 0[10], Table 6-2, "Calculated Azimuthal Variation of Maximum Exposure Rates and Integrated Exposures at the Reactor Vessel Clad/Base Metal Interface." This table projects neutron fluences at various EFPY. The tabulated fluence values for neutrons with E>1.0 MeV were found at 0^0 azimuth (the peak fluence position) for EFPY values above and below the desired withdrawal EFPY. The fluence value corresponding to the desired withdrawal EFPY was then determined by interpolation from the tabulated values and multiplied by the capsule lead factor to yield the projected capsule fluence value listed above

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 five 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], W[2]and V[10].

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 [d]

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	0.612	0.862	64.83	55.883	0.744
	U	1.73	1.151	110.34	127.001	1.324
	X	3.06	1.295	129.71	167.974	1.678
	W	4.75	1.392	145.57	202.633	1.938
	V	7.14	1.466	178.01	260.963	2.149
Lower Shell Plate B6919-1 (Transverse)	Y	0.612	0.862	70.45	60.728	0.744
	U	1.73	1.151	100.51	115.687	1.324
	X	3.06	1.295	110.72	143.382	1.678
	W	4.75	1.392	150.54	209.552	1.938
	V	7.14	1.466	161.87	237.301	2.149
				SUM:	1581.104	15.666
	$CF_{B6919-1} = \Sigma(FF * \Delta RT_{NDT}) + \Sigma(FF^2) = (1581.104) + (15.666) = 100.9°F$					
Weld Metal	Y	0.612	0.862	72.92	62.857	0.744
	U	1.73	1.151	81.13	93.381	1.324
	X	3.06	1.295	93.19	120.681	1.678
	W	4.75	1.392	104.17	145.005	1.938
	V	7.14	1.466	123.29	180.743	2.149
				SUM:	602.667	7.833
	$CF_{Surv.\ Weld} = \Sigma(FF * \Delta RT_{NDT}) + \Sigma(FF^2) = (602.667) + (7.833) = 76.9°F$					

NOTES:

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

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

(c) ΔRT_{NDT} values are the measured 30 ft-lb shift values from Appendix C of WCAP-16221, Revision 0 [10]

(d) Data from Table D-1, WCAP-16221, Revision 0 [10]

Table 4-2

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

Lower Shell Plate B6919-1 Orientation	Capsule	CF (Best Fit Slope)	FF	ΔRT_{NDT} (30 ft-lb) (°F)	Best Fit ΔRT_{NDT} (°F)	Scatter of ΔRT_{NDT} (°F)
Longitudinal	Y	100.9	0.862	64.83	86.98	-22.2
	U	100.9	1.151	110.34	116.14	-5.8
	X	100.9	1.295	129.71	130.67	-1.0
	W	100.9	1.392	145.57	140.45	5.1
	V	100.9	1.466	178.01	147.92	30.1
Transverse	Y	100.9	0.862	70.45	86.98	-16.5
	U	100.9	1.151	100.51	116.14	-15.6
	X	100.9	1.295	110.72	130.67	-20.0
	W	100.9	1.392	150.54	140.45	10.1
	V	100.9	1.466	161.87	147.92	14.0

NOTES:
(a) Data from Table D-2, WCAP-16221-NP, Revision 0 [10]

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 three of the data points are not within 17°F of the best-fit line. Therefore, Criterion 3 is not met for the Farley Unit 1 surveillance plate material. Since not all of the data is 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	Capsule	CF (Best Fit Slope)	FF	ΔRT_{NDT} (30 ft-lb) (°F)	Best Fit ΔRT_{NDT} (°F)	Scatter of ΔRT_{NDT} (°F)
Weld Metal	Y	76.9	0.862	72.92	66.29	6.6
	U	76.9	1.151	81.13	88.51	-7.4
	X	76.9	1.295	93.19	99.59	-6.4
	W	76.9	1.392	104.17	107.04	-2.9
	V	76.9	1.466	123.29	112.74	10.6

NOTES:
(a) Data from Table D-2, WCAP-16221-NP, Revision 0 [10]

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 29.0 EFPY.

Table 5-6 shows the calculation of the ART at 29.0 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]

			30 ft-lb Transition Temperature Shift		Upper Shelf Energy Decrease	
Material	Capsule	Fluence[d] (x 10^{19} n/cm^2, E > 1.0 MeV)	Predicted (°F) [a]	Measured (°F) [b]	Predicted (%) [a]	Measured (%)[c]
Lower Shell	Y	0.612	84.30	64.83	20	8.6
Plate B6919-1	U	1.73	112.57	110.34	25.5	22.9
(Longitudinal)	X	3.06	126.65	129.71	29	18.6
	W	4.75	136.14	145.57	33	22.1
	V	7.14	143.37	178.01	35	22.1
Lower Shell	Y	0.612	84.30	70.45	20	0
Plate B6919-1	U	1.73	112.57	100.51	25.5	8.9
(Transverse)	X	3.06	126.65	110.72	29	11.1
	W	4.75	136.14	150.54	33	15.6
	V	7.14	143.37	161.87	35	20
Surveillance	Y	0.612	67.32	72.92	25	12.1
Program	U	1.73	89.89	81.13	32	29.5
Weld Metal	X	3.06	101.14	93.19	36	22.8
	W	4.75	108.72	104.17	40	26.2
	V	7.14	114.49	123.29	42.5	26.2
Heat Affected Zone	Y	0.612	---	32.55	---	10.3
Material	U	1.73	---	160.37	---	25.8
	X	3.06	---	136.91	---	21.9
	W	4.75	---	126.04	---	14.2
	V	7.14	---	174.39	---	16.8

NOTES:

(a) Data from Table 5-10, WCAP-16221-NP, Revision 0 [10]

Table 5-2

CALCULATION OF CHEMISTRY FACTORS USING SURVEILLANCE CAPSULE DATA[d]

Material	Capsule	Capsule f[a]	FF[b]	ΔRT_{NDT}[c]	$FF*\Delta RT_{NDT}$	FF^2
Lower Shell	Y	0.612	0.862	64.83	55.883	0.744
Plate B6919-1	U	1.73	1.151	110.34	127.001	1.324
(Longitudinal)	X	3.06	1.295	129.71	167.974	1.678
	W	4.75	1.392	145.57	202.633	1.938
	V	7.14	1.466	178.01	260.963	2.149
Lower Shell	Y	0.612	0.862	70.45	60.728	0.744
Plate B6919-1	U	1.73	1.151	100.51	115.687	1.324
(Transverse)	X	3.06	1.295	110.72	143.382	1.678
	W	4.75	1.392	150.54	209.552	1.938
	V	7.14	1.466	161.87	237.301	2.149
				SUM:	1581.104	15.666
	$CF_{B6919-1} = \Sigma(FF * RT_{NDT}) + \Sigma(FF^2) = (1581.104) + (15.666) = 100.9°F$					
Surveillance Weld	Y	0.612	0.862	118.13 (72.92)[c]	101.828	0.744
Material	U	1.73	1.151	131.43 (81.13)[c]	151.276	1.324
	X	3.06	1.295	150.97 (93.19)[c]	195.506	1.678
	W	4.75	1.392	168.76 (104.17)[c]	234.914	1.938
	V	7.14	1.466	199.73 (123.29)[c]	292.804	2.149
				SUM:	976.328	7.833
	$CF_{Surv.\ Weld} = \Sigma(FF * RT_{NDT}) + \Sigma(FF^2) = (976.328) + (7.833) = 124.6°F$					

NOTES:

(a) f = fluence (x 10^{19} n/cm^2, E > 1.0 MeV) from WCAP-16221-NP[10]

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

(c) ΔRT_{NDT} values from Table 4-1 (shown in parentheses) were multiplied by a ratio factor of 1.62 ($CF_{vessel} + CF_{surv\ weld} = 126.2 + 78.1 = 1.62$), to calculate the best fit chemistry factor (CF) as provided by Reg. Guide 1.99 Rev. 2, Position 2.1.

(d) Based on Table D-1, WCAP-16221-NP, Revision 0[10]

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°	30°	45°
36	4.14	2.41	1.73	1.24

NOTES:
(a) From Table 6-2 of WCAP-16221-NP Rev. 0 [10]
(b) Fluence in 10^{19} n/cm^2 (E > 1.0 MeV)

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 29.0 EFPY [a, b]

Material	1/4-T (°F)	3/4-T (°F)
Intermediate Shell Plate B6903-2	143	120
Intermediate Shell Plate B6903-3	143	121
Lower Shell Plate B6919-1	166	141
Lower Shell Plate B6919-1 Using S/C Data	170 [c]	144[c]
Lower Shell Plate B6919-2	157	131
Intermediate Shell Longitudinal Weld Seams 19-894 A & B (Heat # 33A277)	120 [d]	88 [d]
Intermediate Shell Longitudinal Weld Seams 19-894 A & B (Heat # 33A277) Using S/C Data	97 [d]	65 [d]
Circumferential Weld 11-894 (Heat # 6329637)	128	103
Lower Shell Longitudinal Weld Seams 20-894 A & B (Heat # 90099)	90 [d]	67 [d]

NOTES:

(a) From Westinghouse letter ALA-04-170[11].

(b) The ARTs presented here are based on the peak reactor vessel surface fluence of 3.34×10^{19} n/cm^2 (E > 1.0 MeV) from WCAP-16221-NP, Revision 0[10], Table 6-2 unless otherwise noted.

(c) Limiting 1/4T and 3/4T ART values. The P/T limit curves are based on ART values of 170°F and 145°F, originally documented in WCAP-14689, Revision 4 [1] for 31.9 EFPY using a peak fluence of 3.868×10^{19} n/cm^2 (E > 1.0 MeV) and a chemistry factor (CF) of 97.8°F (per Reg. Guide 1.99 Rev. 2, Position 1.1). However, subsequent to the analysis of Capsule V (ref. WCAP-16221-NP, Revision 0 [10]) the CF using Reg. Guide 1.99 Rev. 2, Position 2.1 increased from 93.3°F to 100.9°F, making the Position 2.1 ART higher than the Position 1.1 ART used by WCAP-14689. Thus, in order to maintain the 1/4T ART value at 170°F (the value used in generating the P/T limit curves, Figs. 2-1 and 2-2), the service period of the P/T curves was reduced to 29.0 EFPY. This reduction was calculated using the updated fluence value cited in (b) above. Since maintaining a 3/4T ART of 145°F would cause a lesser reduction in the service period (to 30 EFPY), the 1/4T location is more limiting. Therefore, while the 145°F 3/4T ART value used in generation of the P/T limit curves is still cited in Figs. 2-1and 2-2, the updated 3/4T ART value of 144°F at 29.0 EFPY is tabulated.

(d) ARTs calculated using the peak vessel fluence of 1.01×10^{19} n/cm^2 (E > 1.0 MeV) at 45°, interpolated from WCAP-16221-NP, Revision 0[10], Table 6-2.

Table 5-6

CALCULATION OF ADJUSTED REFERENCE TEMPERATURE AT 29.0 EFPY FOR THE LIMITING REACTOR VESSEL MATERIAL - LOWER SHELL PLATE B6919-1 [(a)]

Parameter	Value	
Operating Period	29.0 EFPY	
Location	1/4-T	3/4-T
Chemistry Factor, CF (°F)	100.9	100.9
Fluence, f (10^{19} n/cm^2) [(b)]	2.080	0.809
Fluence Factor, FF	1.199	0.941
ΔRT_{NDT} = CF x FF (°F)	121.0	94.9
Initial RT_{NDT}, I (°F)	15	15
Margin, M (°F)	34	34
Adjusted Reference Temperature (ART), (°F) per Regulatory Guide 1.99, Revision 2	170[(c)]	144[(c)]

NOTES:
(a) From Westinghouse letter ALA-04-170[11].
(b) Fluence is based on f_{surf} (10^{19} n/cm^2, E > 1.0 MeV) = 3.34, from WCAP-16221-NP, Revision 0[10], Table 6-2. The Farley Unit 1 reactor vessel wall thickness is 7.875 inches in the beltline region.
(c) Refer to note (c) of Table 5-5.

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	4.14	1.36	123.8	0	34	158
Intermediate Shell Plate B6903-3	82.2	4.14	1.36	111.8	10	34	156
Lower Shell Plate B6919-1	97.8	4.14	1.36	133.0	15	34	182
Lower Shell Plate B6919-1 Using S/C Data	100.9	4.14	1.36	137.2	15	34	186[b]
Lower Shell Plate B6919-2	98.2	4.14	1.36	133.6	5	34	173
Intermediate Shell Longitudinal Welds 19-894 A & B (Heat # 33A277)	126.3	1.24	1.06	133.9	-56	66	144
Intermediate Shell Longitudinal Welds 19-894 A & B (Heat # 33A277) Using S/C Data	124.6	1.24	1.06	132.1	-56	44	120
Circumferential Weld 11-894 (Heat # 6329637)	98.4	4.14	1.36	133.8	-56	66	144
Lower Shell Longitudinal Welds 20-894 A & B (Heat # 90099)	91.4	1.24	1.06	96.9	-56	66	107

NOTES:

(a) From Westinghouse letter ALA-04-170[11].

(b) This RT_{PTS} value was calculated using the CF from the surveillance data and a full σ_Δ margin of 17°F, since this surveillance data is not credible.

6.0 References

1. WCAP-14689, Revision 4, Farley Units 1 and 2 Heatup and Cooldown Limit Curves for Normal Operation and PTLR Support Documentation, E. Terek, April 1998.
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.
8. WCAP-12471, Analysis of Capsule X from the Alabama Power Company Joseph M. Farley Unit 2 Reactor Vessel Radiation Surveillance Program, E. Terek, et al., December 1989.
9. CE NPSD-1039, Revision 2, Best Estimate Copper and Nickel Values in CE Fabricated Reactor Vessel Welds, Combustion Engineering Owners Group, June 1997.
10. WCAP-16221-NP, Revision 0, Analysis of Capsule V from the Southern Nuclear Operating Company, Joseph M. Farley Unit 1 Reactor Vessel Radiation Surveillance Program, K.G. Knight, et. al., March 2004.
11. Westinghouse letter ALA-04-170, Review of Revision 2 Draft of Pressure-Temperature Limits Report, E.C. Arnold to L. M. Stinson, October 25, 2004.