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SOUTHERN COMPANY
Energy to Serve Your World[SM]

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

Joseph M. Farley Nuclear Plant – Unit 2
Submittal of Revision 3 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 2 Technical Specifications, Southern Nuclear Operating Company (SNC) hereby submits Revision 3 of the FNP Unit 2 Pressure Temperature Limits Report (PTLR). Changes include revision of the surveillance capsule withdrawal schedule consistent with the revised schedule approved by NRC letter dated March 15, 2004, updating of the surveillance data credibility analysis and supplemental data sections to reflect the latest surveillance capsule analysis report (WCAP-16351-NP Rev. 0 for Capsule Y, submitted March 15, 2005) as requested by NRC letter dated January 18, 2006, and addition of pertinent reference citations. The pressure-temperature limit curves in Rev. 2 of the PTLR remained bounding after review of the Capsule Y analysis results and hence the curves are left unchanged by Rev. 3.

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

Sincerely,

Lewis Sumner

H. L. Sumner, Jr.

HLS/DWD/sdl

Enclosure: Joseph M. Farley Nuclear Plant Pressure Temperature Limits Report Unit 2 Revision 3, September 2006

cc: Southern Nuclear Operating Company
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Joseph M. Farley Nuclear Plant – Unit 2
<u>Submittal of Revision 3 of the Pressure Temperature Limits Report</u>

Enclosure

Joseph M. Farley Nuclear Plant Pressure Temperature Limits Report
Unit 2 Revision 3, September 2006

SOUTHERN COMPANY

Energy to Serve Your World®

Joseph M. Farley Nuclear Plant

Pressure Temperature Limits Report

Unit 2

Revision 3

September 2006

Table of Contents

List of Tables

List of Figures

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1.0 RCS Pressure Temperature Limits Report (PTLR)

This PTLR for Farley Nuclear Plant - Unit 2 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 (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 NRC-approved 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.

2.3 LTOP Arming Temperature (LCO – 3.4.12)

2.3.1 The LTOP system arming temperature is 325°F.

Figure 2-1

Farley Unit 2 Reactor Coolant System Heatup Limitations

(Heatup Rates up to 100°F/hr) Applicable to 33.8 EFPY (adjusted to include 60 psi ΔP at RCS temperatures ≥ 110°F and 27 psi ΔP at RCS temperatures < 110°F). Includes vessel flange requirements of 180°F and 561 psig per 10 CFR 50, Appendix G.[1]

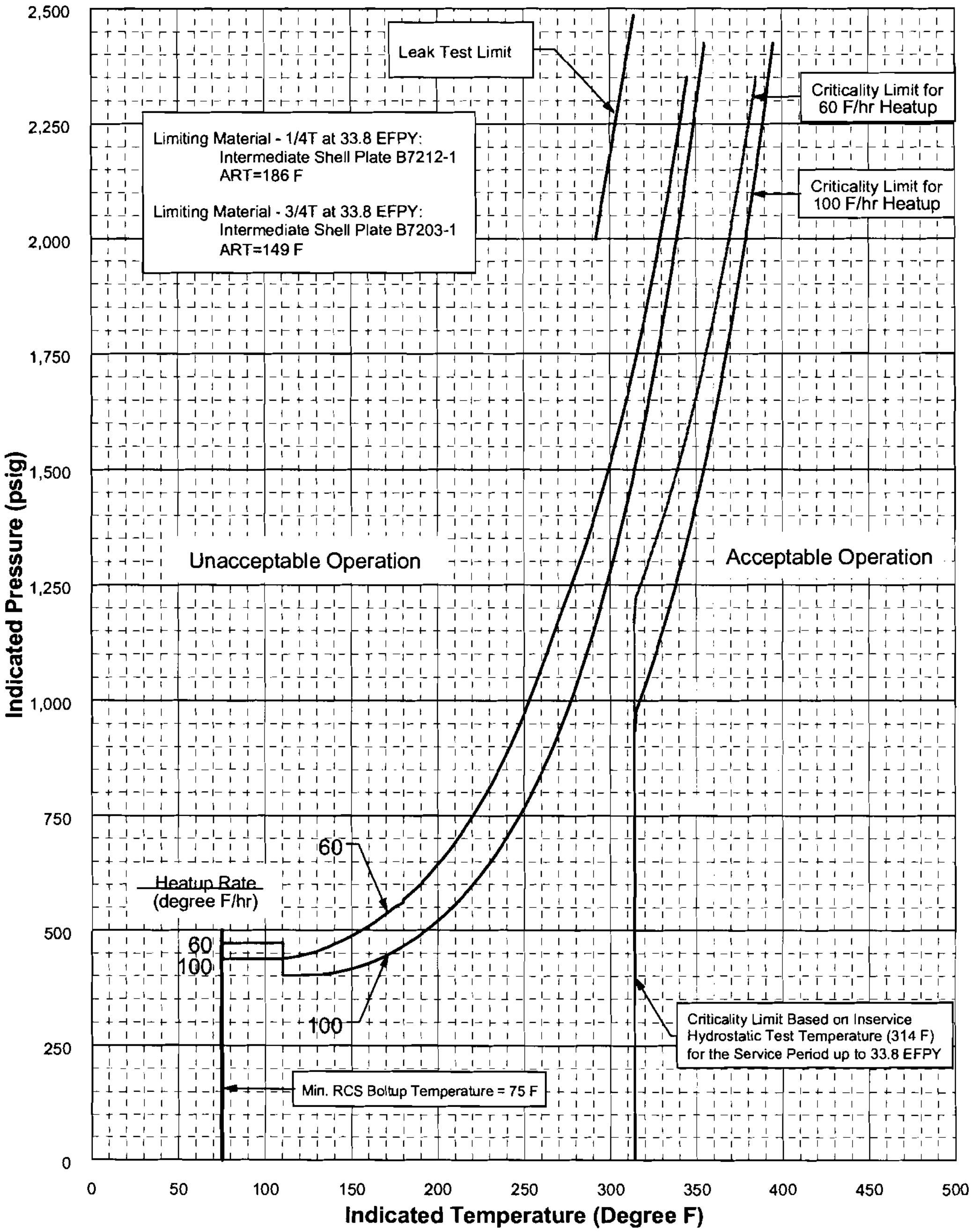

Figure 2-2

Farley Unit 2 Reactor Coolant System Cooldown Limitations

(Cooldown Rates up to 100°F/hr) Applicable to 33.8 EFPY (adjusted to include 60 psi ΔP at RCS temperatures ≥ 110°F and 27 psi ΔP at RCS temperatures < 110°F). Includes vessel flange requirements of 180°F and 561 psig per 10 CFR 50, Appendix G. [1]

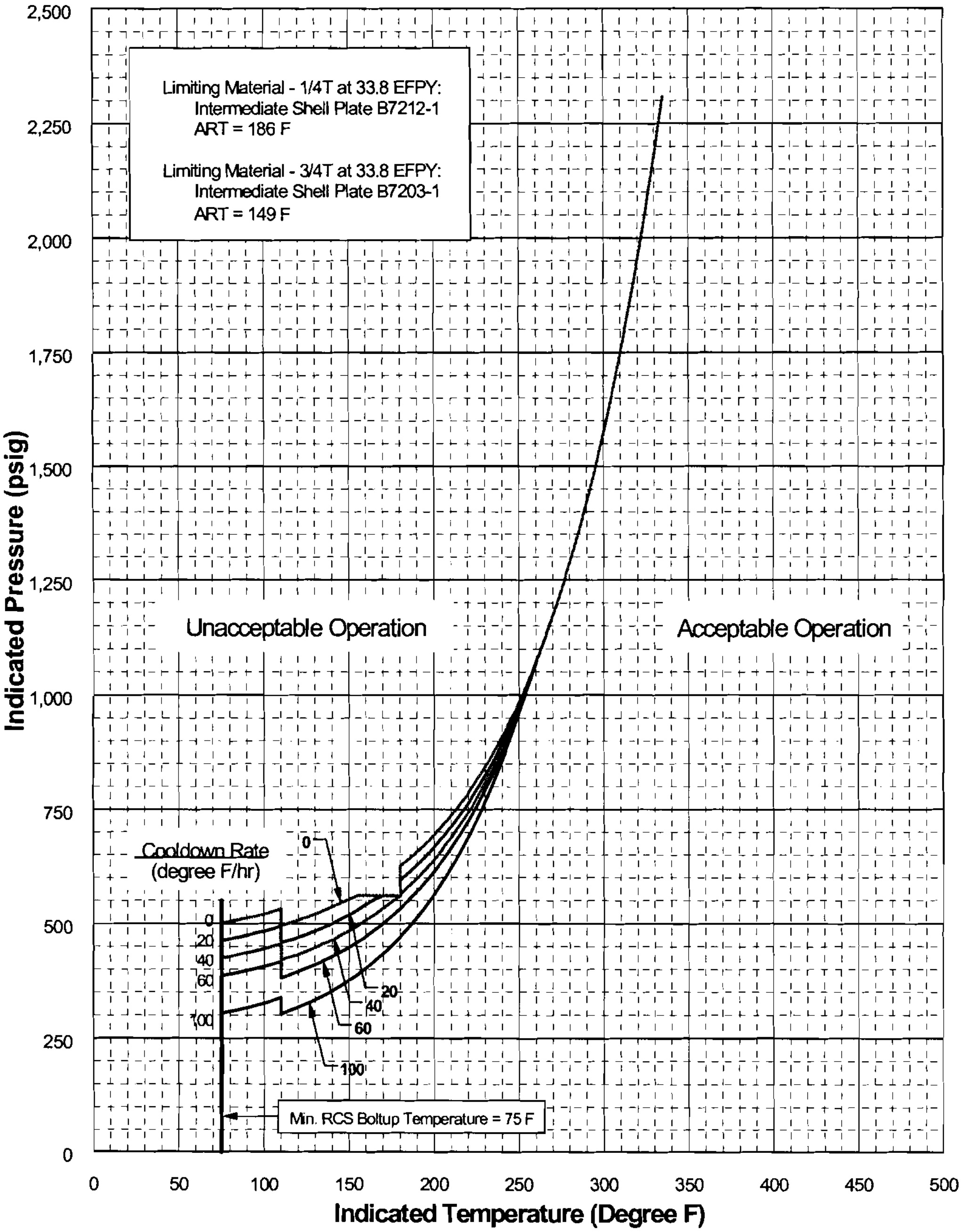

Table 2-1

Farley Unit 2 - 33.8 EFPY Heatup Curve Data Points

(adjusted to include 60 psi ΔP at RCS temperatures ≥ 110°F and 27 psi ΔP at RCS temperatures < 110°F) [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	470	314	0	75	435	314	0	292	2000
80	470	314	465	80	435	314	467	314	2485
85	470	314	454	85	435	314	451		
90	470	314	446	90	435	314	438		
95	470	314	441	95	435	314	428		
100	470	314	438	100	435	314	419		
105	470	314	437	105	435	314	413		
110	471	314	438	110	435	314	408		
110	438	314	441	110	402	314	405		
115	441	314	444	115	402	314	403		
120	444	314	449	120	402	314	402		
125	449	314	455	125	402	314	403		
130	455	314	462	130	403	314	404		
135	462	314	469	135	404	314	407		
140	469	314	478	140	407	314	411		
145	478	314	488	145	411	314	416		
150	488	314	498	150	416	314	422		
155	498	314	510	155	422	314	428		
160	510	314	522	160	428	314	436		
165	522	314	536	165	436	314	445		
170	536	314	551	170	445	314	455		
175	551	314	567	175	455	314	466		
180	561	314	584	180	466	314	478		
180	567	314	602	185	478	314	491		
185	584	314	622	190	491	314	505		
190	602	314	644	195	505	314	521		
195	622	314	667	200	521	314	538		
200	644	314	692	205	538	314	556		
205	667	314	719	210	556	314	576		
210	692	314	747	215	576	314	598		
215	719	314	778	220	598	314	621		
220	747	314	811	225	621	314	646		
225	778	314	847	230	646	314	673		
230	811	314	885	235	673	314	702		
235	847	314	926	240	702	314	733		
240	885	314	970	245	733	314	767		
245	926	314	1018	250	767	314	803		
250	970	314	1069	255	803	314	842		
255	1018	314	1119	260	842	314	883		
260	1069	314	1171	265	883	314	928		
265	1119	315	1223	270	928	315	976		
270	1171	320	1273	275	976	320	1028		
275	1223	325	1326	280	1028	325	1083		
280	1273	330	1383	285	1083	330	1143		
285	1326	335	1445	290	1143	335	1206		
290	1383	340	1510	295	1206	340	1275		
295	1445	345	1580	300	1275	345	1348		
300	1510	350	1656	305	1348	350	1426		
305	1580	355	1736	310	1426	355	1510		
310	1656	360	1822	315	1510	360	1599		
315	1736	365	1914	320	1599	365	1695		
320	1822	370	2013	325	1695	370	1798		
325	1914	375	2118	330	1798	375	1908		
330	2013	380	2231	335	1908	380	2025		
335	2118	385	2351	340	2025	385	2150		
340	2231			345	2150	390	2283		
345	2351			350	2283	395	2425		
				355	2425				

Table 2-2

Farley Unit 2 - 33.8 EFPY Cooldown Curve Data Points

(adjusted to include 60 psi ΔP at RCS temperatures ≥ 110°F and 27 psi ΔP at 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	499	75	461	75	423	75	384	75	303
80	503	80	465	80	426	80	387	80	306
85	506	85	469	85	430	85	391	85	310
90	510	90	472	90	434	90	395	90	315
95	514	95	477	95	438	95	400	95	319
100	519	100	481	100	443	100	404	100	325
105	523	105	486	105	448	105	410	105	330
110	529	110	492	110	454	110	415	110	336
110	496	110	459	110	421	110	382	110	303
115	501	115	464	115	427	115	389	115	310
120	507	120	470	120	433	120	395	120	317
125	514	125	477	125	440	125	402	125	325
130	521	130	484	130	448	130	410	130	334
135	528	135	492	135	456	135	419	135	343
140	536	140	500	140	464	140	428	140	353
145	545	145	509	145	474	145	438	145	364
150	554	150	519	150	484	150	448	150	376
155	561	155	529	155	495	155	460	155	389
160	561	160	541	160	507	160	472	160	403
165	561	165	553	165	519	165	486	165	418
170	561	170	561	170	533	170	500	170	434
175	561	175	561	175	548	175	516	175	452
180	561	180	561	180	561	180	533	180	471
180	626	180	595	180	564	185	551	185	491
185	641	185	611	185	581	190	570	190	513
190	658	190	628	190	599	195	591	195	537
195	675	195	647	195	619	200	614	200	563
200	694	200	667	200	640	205	638	205	591
205	715	205	689	205	663	210	665	210	621
210	737	210	712	210	688	215	693	215	653
215	760	215	737	215	715	220	723	220	688
220	786	220	764	220	743	225	756	225	725
225	813	225	793	225	774	230	792	230	766
230	842	230	824	230	807	235	830	235	809
235	874	235	858	235	843	240	871	240	856
240	908	240	894	240	881	245	915	245	907
245	944	245	932	245	923	250	962	250	961
250	983	250	974	250	967	255	1013	255	1019
255	1025	255	1019	255	1015	260	1068		
260	1070	260	1067	260	1066				
265	1119	265	1118						
270	1171								
275	1226								
280	1286								
285	1351								
290	1420								
295	1494								
300	1573								
305	1658								
310	1749								
315	1846								
320	1951								
325	2062								
330	2182								
335	2309								

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. Consistent with specific requirements for Farley Unit 2 associated with the granting of an exemption to Appendix H of 10 CFR 50 documented in NUREG-0117 [4], Figures 2-1 and 2-2 are based on the greater, or limiting value, of the following: (1) the actual shift in reference temperature for plate B7212-1 as determined by impact testing, or (2) the predicted shift in reference temperature for weld seam 11-923 as determined by Regulatory Guide 1.99, Revision 2. The neutron transport and dosimetry evaluation methodologies used follow the guidance and meet the requirements of Regulatory Guide 1.190, "Calculational and Dosimetry Methods for Determining Pressure Vessel Neutron Fluence"[14]. Results from the reactor vessel surveillance program will 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)
U (c)	343	3.26	1.11	6.05×10^{18}
W (c)	110	2.84	3.96	1.73×10^{19}
X (c)	287	3.39	6.43	2.98×10^{19}
Z (c)	340	2.98	13.85	4.92×10^{19} (d)
Y (c)	290	3.12	19.01	6.79×10^{19} (e)
V	107	3.57	~22 (g)	8.75×10^{19} (f)

Notes:

a) Data from Table 7-1, WCAP-16351-NP, Revision 0 [11]
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) NRC approval is required prior to changing the capsule withdrawal schedule. Reference: NRC Administrative Letter 97-04. The current schedule was submitted by SNC letter NL-04-0372, March 5, 2004 [12] and approved by NRC letter dated March 15, 2004 [13].

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 capsules removed from the Farley Unit 2 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 2 reactor vessel surveillance data and determine if the Farley Unit 2 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 2 reactor vessel consists of the following beltline region materials:

- Intermediate shell plates B7203-1 and B7212-1;
- Lower shell plates B7210-1 and B7210-2;
- Intermediate shell longitudinal weld seams 19-923 A, heat number HODA;
- Intermediate shell longitudinal weld seams 19-923 B, heat number BOLA;
- Lower shell longitudinal weld seams 20-923 A & B, heat number 83640, Linde 0091 flux, flux lot 3490; and
- Circumferential weld 11-923, heat number 5P5622, Linde 0091 flux, flux lot 1122.

Per WCAP-8956[5], the Farley Unit 2 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 2 surveillance capsule program was developed, intermediate shell plate B7212-1 was judged to be most limiting and was therefore utilized in the surveillance program.

The surveillance program weld for Farley Unit 2 was fabricated using the shielded metal arc welding process and E8018 stick electrodes, in a manner similar to that used to fabricate intermediate shell longitudinal seams 19-923 A (heat HODA) and B (heat BOLA). These electrodes were not copper-coated and do not exhibit the chemical variability found in copper-coated submerged arc weld wire. Although the surveillance weld material does not represent the limiting reactor vessel beltline weld, 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. The NRC explicitly approved the selection of the Farley Unit 2 surveillance weld material on the basis that the limiting beltline material (i.e., intermediate plate B7212-1) was included in the surveillance program and conservative methods of analysis contained in Regulatory Guide 1.99 were available to predict the radiation characteristics of the limiting beltline weld. The NRC incorporated an exemption to the requirements of Appendix H to 10 CFR Part 50 in the Farley Unit 2 Operating License, thereby approving the selected surveillance weld material based on the NRC evaluation provided in Section 5.2.1 of NUREG-0117. [4]

Although the Farley Unit 2 surveillance weld material does not meet the requirements of Criterion 1, conservative methods of analysis are available to predict the radiation characteristics of the limiting beltline weld. The limiting beltline plate material is intermediate plate B7212-1 which is more limiting than any of the reactor vessel beltline welds and is included in the reactor vessel material surveillance program. Therefore, the Farley Unit 2 reactor vessel material surveillance program provides assurance that the radiation damage to the vessel can be adequately determined and the integrity of the Farley Unit 2 reactor vessel will be ensured during normal plant operations and anticipated operational occurrences. Therefore, the Farley Unit 2 reactor vessel surveillance program meets the intent 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 WCAP-8956 [5], Alabama Power Company Joseph M. Farley Nuclear Plant Unit No. 2 Reactor Vessel Radiation Surveillance Program, dated August 1977.

Plots of Charpy energy versus temperature for the irradiated conditions are presented in the reactor vessel surveillance capsule reports for Capsules U [6], W [7], X [2], Z [10], and Y [11].

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 2 surveillance materials unambiguously. Therefore, the Farley Unit 2 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 of Regulatory Guide 1.99, Revision 2.1, was used to calculate a best-fit line for the base metal and weld data to determine if this criterion is met. Per Regulatory Position 2, the scatter of ΔRT_{NDT} values about a best-fit line should be less than 1σ (17°F for base metal and 28°F for welds).

Plate (Base Metal) Evaluation:

The scatter values obtained for the base metal (intermediate shell plate B7212-1) are shown in Table 4-1, which indicates that two of the measured ΔRT_{NDT} values are slightly lower than 1σ below the predicted value, while one value slightly exceeds the upper 1σ bound. The data show that all measured ΔRT_{NDT} values are well below the upper 1σ bound (i.e. the predicted value conservatively exceeds the measured value) except one (Capsule Y, transverse specimen).

That three measured ΔRT_{NDT} values fall slightly outside the $\pm 1\sigma$ bounds can be attributed to several factors, such as 1) the inherent uncertainty in the Charpy test data, 2) the use of a symmetric hyperbolic tangent Charpy curve fitting program vs. an asymmetric hyperbolic tangent Charpy curve fitting program or a hand-drawn curve using engineering judgement, and 3) rounding errors.

Of the 10 data points, scatter is within 17°F of the best-fit line for all but 3. Statistically, since $\pm 1\sigma$ (17°F) is expected to encompass approximately 68% of the data, for a set of 10 scatter values one could expect 3 to be outside the $\pm 1\sigma$ bounds, as observed, hence the data is statistically credible, and based on the arguments above, the plate data meets the intent of Criterion 3.

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)
Intermediate Shell Plate B7212-1 (Longitudinal)	U	0.605	0.859	105.5	90.6	0.738
	W	1.73	1.151	167.7	193.0	1.325
	X	2.98	1.289	164.8	212.4	1.662
	Z	4.92	1.399	200.1	279.9	1.957
	Y	6.79	1.458	214.2	312.3	2.126
Intermediate Shell Plate B7212-1 (Transverse)	U	0.605	0.859	124.0	106.5	0.738
	W	1.73	1.151	168.5	193.9	1.325
	X	2.98	1.289	200.1	257.9	1.662
	Z	4.92	1.399	195.8	273.9	1.957
	Y	6.79	1.458	231.0	336.8	2.126
				$\sum_{i=1}^{n}$	2257.2	15.616
	CF = Σ(FF * ΔRT_{NDT}) ÷ Σ(FF^2) = 144.5 °F					
Weld Metal	U	0.605	0.859	0.0	0.0	0.738
	W	1.73	1.151	7.0	8.1	1.325
	X	2.98	1.289	0.0	0.0	1.662
	Z	4.92	1.399	10.2	14.3	1.957
	Y	6.79	1.458	69.1	100.7	2.126
				$\sum_{i=1}^{n}$	123.1	7.808
	CF = Σ(FF * ΔRT_{NDT}) ÷ Σ(FF^2) = 15.8 °F					

NOTES:
(a) Data from Table D-1, WCAP-16351-NP, Revision 0[11]
(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)]

Intermediate Shell Plate B7212-1 Specimen Orientation	Capsule	CF (Best Fit Slope)	FF	ΔRT_{NDT} (30 ft-lb) (°F)	Best Fit ΔRT_{NDT} (°F)	Scatter of ΔRT_{NDT} (°F)	<17 °F
Longitudinal	U	144.5	0.859	105.5	124.1	18.6	No
	W	144.5	1.151	167.7	166.3	-1.4	Yes
	X	144.5	1.289	164.8	186.3	21.5	No
	Z	144.5	1.399	200.1	202.2	2.1	Yes
	Y	144.5	1.458	214.2	210.7	-3.5	Yes
Transverse	U	144.5	0.859	124.0	124.1	0.1	Yes
	W	144.5	1.151	168.5	166.3	-2.2	Yes
	X	144.5	1.289	200.1	186.3	-13.8	Yes
	Z	144.5	1.399	195.8	202.2	6.4	Yes
	Y	144.5	1.458	231.0	210.7	-20.3	No

NOTES:
(a) Data from Table D-2, WCAP-16351-NP, Revision 0 [11]

Table 4-3

Scatter of ΔRT_{NDT} Values About a Best-Fit Line
for Surveillance Weld Material [(a)]

Surveillance Material	Capsule	CF (Best Fit Slope)	FF	ΔRT_{NDT} (30 ft-lb) (°F)	Best Fit ΔRT_{NDT} (°F)	Scatter of ΔRT_{NDT} (°F)	<28 °F
Weld Metal	U	15.8	0.859	0.0	13.6	13.6	Yes
	W	15.8	1.151	7.0	18.2	11.2	Yes
	X	15.8	1.289	0.0	20.4	20.4	Yes
	Z	15.8	1.399	10.2	22.1	11.9	Yes
	Y	15.8	1.458	69.1	23.0	-46.1	No

NOTES:
(a) Data from Table D-2, WCAP-16351-NP, Revision 0 [11]

Weld Evaluation:

The scatter values obtained for the weld metal are shown in Table 4-2, which indicates that one of the five measured ΔRT_{NDT} values exceeds the $\pm 1\sigma$ scatter band of 28°F. Based on the statistical argument presented for the plate evaluation, one of five data points outside the scatter band could be expected and is acceptable, hence the surveillance weld data meets the intent of Criterion 3.

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 2 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 2 surveillance program does not include correlation monitor material. Therefore, this criterion is not applicable to Farley Unit 2.

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 2 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 2 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 2 reactor vessel beltline materials at the 1/4-T and 3/4-T locations for 33.8 EFPY.

Table 5-6 shows the calculation of the ART at 33.8 EFPY for the limiting Farley Unit 2 reactor vessel material.

Table 5-7 provides RT_{PTS} values for Farley Unit 2 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)]

			30 ft-lb Transition Temperature Shift		Upper Shelf Energy Decrease	
Material	Capsule	Fluence[(f)] ($x\ 10^{19}\ n/cm^2$, E > 1.0 MeV)	Predicted (°F) [(b)]	Measured (°F) [(c)]	Predicted (%) [(b)]	Measured (%)[(d)]
Intermediate Shell Plate B7212-1 (Longitudinal)	U	0.605	128.0	105.5	26	27.4
	W	1.73	171.5	167.7	33	21.4
	X	2.98	192.1	164.8	37	25.9
	Z	4.92	208.5	200.1	42	27.8
	Y	6.79	217.2	214.2	45	36.2
Intermediate Shell Plate B6919-1 (Transverse)	U	0.605	128.0	124.0	26	27.5
	W	1.73	171.5	168.5	33	20.5
	X	2.98	192.1	200.1	37	27.9
	Z	4.92	208.5	195.8	42	28.6
	Y	6.79	217.2	231.0	45	42.1
Surveillance Program Weld Metal	U	0.605	32.8	0.0[(e)]	17	8.3
	W	1.73	44.0	7.0	22	0.0
	X	2.98	49.2	0.0[(e)]	24	0.0
	Z	4.92	53.4	10.2	27	7.6
	Y	6.79	55.7	69.1	29	4.9
Heat Affected Zone Material	U	0.605	- - -	219.8	- - -	29.7
	W	1.73	- - -	268.8	- - -	20.3
	X	2.98	- - -	230.5	- - -	19.0
	Z	4.92	- - -	263.8	- - -	20.3
	Y	6.79	- - -	269.6	- - -	35.4

NOTES:
(a) Data from Table 5-10, WCAP-16351-NP, Revision 0 [11]
(b) Based on Reg. Guide 1.99, Rev. 2 methodology using the mean weight percent values of copper and nickel of the surveillance material.
(c) Calculated using measured Charpy data plotted using CVGRAPH, Version 5.0.2.
(d) Values are based on the definition of upper shelf energy given in ASTM E185-82.
(e) The actual measured capsule U and capsule X ΔRT_{NDT} values are -28.4°F and -15.6°F respectively. This physically should not occur, therefore for conservatism a value of zero will be reported.
(f) The fluence values presented here are the calculated values, not the best estimate values.

Table 5-2
Calculation of Chemistry Factors Using
Surveillance Capsule Data [a]

Material	Capsule	F (b)	FF (c) (x)	ΔRT_{NDT} (y)	FF x ΔRT_{NDT} (xy)	FF^2 (x^2)
Intermediate Shell Plate B7212-1 (Longitudinal)	U	0.605	0.859	105.5	90.6	0.738
	W	1.73	1.151	167.7	193.0	1.325
	X	2.98	1.289	164.8	212.4	1.662
	Z	4.92	1.399	200.1	279.9	1.957
	Y	6.79	1.458	214.2	312.3	2.126
Intermediate Shell Plate B7212-1 (Transverse)	U	0.605	0.859	124.0	106.5	0.738
	W	1.73	1.151	168.5	193.9	1.325
	X	2.98	1.289	200.1	257.9	1.662
	Z	4.92	1.399	195.8	273.9	1.957
	Y	6.79	1.458	231.0	336.8	2.126
				$\sum_{i=1}^{n}$	2257.2	15.616
	CF = $\Sigma(FF * \Delta RT_{NDT}) \div \Sigma(FF^2)$ = 144.5 °F					
Weld Metal	U	0.605	0.859	0.0 (0.0) (d)	0.0	0.738
	W	1.73	1.151	6.72 (7.0) (d)	7.73	1.325
	X	2.98	1.289	0.0 (0.0) (d)	0.0	1.662
	Z	4.92	1.399	9.79 (10.2) (d)	13.70	1.957
	Y	6.79	1.458	66.3 (69.1) (d)	96.67	2.126
				$\sum_{i=1}^{n}$	118.1	7.808
	CF = $\Sigma(FF * \Delta RT_{NDT}) \div \Sigma(FF^2)$ = 15.1 °F					

NOTES:
(a) Data from Table D-1, WCAP-16351-NP, Revision 0[11]
(b) F = Fluence (10^{19} n/cm^2, E > 1.0 MeV)
(c) FF = Fluence Factor = $F^{(0.28 - 0.1 \log f)}$
(d) ΔRT_{NDT} values from Table 4-1 (shown in parentheses) were multiplied by a ratio of 0.96 (from WCAP-14689 Rev. 4[1] Table 4, $CF_{vessel} \div CF_{surv\ weld}$ = 36.8 ÷ 38.2 = 0.96) to calculate the best fit chemistry factor (CF) as provided by Reg. Guide 1.99 Rev. 2, Position 2.1.

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
Inter. Shell Plate B7203-1	0.14	0.60	15
Inter. Shell Plate B7212-1	0.20	0.60	-10
Lower Shell Plate B7210-1	0.13	0.56	18
Lower Shell Plate B7210-2	0.14	0.57	10
Inter. Shell Longitudinal Weld Seam 19-923 A [(b)] (Heat # HODA)	0.027	0.947	-56
Inter. Shell Longitudinal Weld Seam 19-923 B [(b)] (Heat # BOLA)	0.027	0.913	-60
Surveillance Weld [(c)]	0.028	0.89	--
Circumferential Weld Seam 11-923 [(b)] (Heat # 5P5622)	0.153	0.077	-40
Lower Shell Longitudinal Weld Seams 20-923 A & B [(b)] (Heat # 83640)	0.051	0.096	-70

NOTES:
(a) From Table 2, WCAP-14689, Revision 4 [1]
(b) Best-estimate copper and nickel from CE NPSD-1039 [9]
(c) The best-estimate copper and nickel value represents the average of two chemistry measurements performed on the surveillance weld and documented in WCAP-8956 [5] and WCAP-11438 [7]. The surveillance weld is representative of intermediate shell longitudinal weld 19-923B

Table 5-4

Reactor Vessel Fluence Projections for 36 EFPY [(a,b)]

EFPY	0°	15°	15° [(c)]	30°	30° [(c)]	45°
36	4.39	2.61	2.09	1.98	1.91	1.40

NOTES:
(a) From Table 7, 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 33.8 EFPY [(a,b)]

Material	1/4-T (°F)	3/4-T (°F)
Intermediate Shell Plate B7203-1	174	149 [(c)]
Intermediate Shell Plate B7212-1	211	173
Intermediate Shell Plate B7212-1 Using S/C Data	183 [(c)]	147
Lower Shell Plate B7210-1	165	142
Lower Shell Plate B7210-2	168	143
Intermediate Shell Longitudinal Weld Seam 19-923 A (Heat # HODA)	28 [(d)]	12 [(d)]
Intermediate Shell Longitudinal Weld Seam 19-923 B (Heat # BOLA)	10 [(d)]	-9 [(d)]
Intermediate Shell Longitudinal Weld Seam 19-923 B (Heat # BOLA) Using S/C Data	-44 [(d)]	-48 [(d)]
Circumferential Weld 11-923 (Heat # 5P5622)	109	90
Lower Shell Longitudinal Weld Seams 20-923 A & B (Heat # 83640)	0 [(d)]	-19 [(d)]

NOTES:
(a) From Tables 13 & 14, WCAP-14689, Revision 4 [1]
(b) The ARTs presented here are based on the peak reactor vessel surface fluence of 4.127 x 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 1/4-T ART of 186°F and 3/4-T ART of 149°F which bounds the limiting 1/4-T and 3/4-T ARTs shown above.
(d) ARTs calculated using the peak vessel fluence of 1.32 x 10^{19} n/cm^2 (E > 1.0 MeV) at 45°

Table 5-6

Calculation of Adjusted Reference Temperature at 33.8 EFPY for the Limiting Reactor Vessel Material [(a)]

Parameter	Intermediate Shell Plate B7212-1		Intermediate Shell Plate B7203-1	
Operating Period	33.8 EFPY		33.8 EFPY	
Location	¼-T	¾-T	¼-T	¾-T
Chemistry Factor, CF (°F)	140.3	140.3	100.0	100.0
Fluence, f (10^{19} n/cm^2) [(b)]	2.573	1.00	2.573	1.00
Fluence Factor, FF	1.253	1.00	1.253	1.00
ΔRT_{NDT} = CF x FF (°F)	175.8	140.3	125.3	100.0
Initial RT_{NDT}, I (°F)	-10	-10	15	15
Margin, M (°F) [(c)]	17	17	34	34
Adjusted Reference Temperature (ART), (°F) per Regulatory Guide 1.99, Revision 2	**183**	147	174	**149**

NOTES:

(a) From Tables 13 & 14 (using surveillance capsule data), WCAP-14689, Revision 4 [1]

(b) Fluence is based on f_{surf} (10^{19} n/cm^2, E > 1.0 MeV) = 4.127 at 33.8 EFPY. The Farley Unit 2 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 B7203-1	100.0	4.39	1.38	138.0	15	34	187
Intermediate Shell Plate B7212-1	149.0	4.39	1.38	205.6	-10	34	230
Intermediate Shell Plate B7212-1 Using S/C Data	140.3	4.39	1.38	193.6	-10	17	201
Lower Shell Plate B7210-1	89.8	4.39	1.38	123.9	18	34	176
Lower Shell Plate B7210-2	98.7	4.39	1.38	136.2	10	34	180
Intermediate Shell Longitudinal Welds 19-923 A (Heat # HODA)	36.8	1.40	1.09	40.1	-56	52.6	37
Intermediate Shell Longitudinal Welds 19-923 B (Heat # BOLA)	36.8	1.40	1.09	40.1	-60	40.1	20
Intermediate Shell Longitudinal Welds 19-923 B (Heat # BOLA) Using S/C Data	8.4	1.40	1.09	9.2	-60	9.2	-42
Circumferential Weld 11-923 (Heat # 5P5622)	74.1	4.39	1.38	102.3	-40	56	118
Lower Shell Longitudinal Welds 20-923 A & B (Heat # 83640)	37.3	1.40	1.09	40.7	-70	40.7	11

<u>NOTES</u>:
(a) From Table C-2, WCAP-14689, Revision 4 [1]

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-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.

3. WCAP-14687, Joseph M. Farley Units 1 and 2 Radiation Analysis and Neutron Dosimetry Evaluation, R. L. Bencini, June 1996.

4. NUREG-0117, Supplement 5 to the Safety Evaluation Report (NUREG-75/034), Office of Nuclear Reactor Regulation, U. S. Nuclear Regulatory Commission in the matter of Alabama Power Company Joseph M. Farley Nuclear Plant Unit 2, Docket No. 50-364., March 19, 1981.

5. WCAP-8956, Alabama Power Company Joseph M. Farley Nuclear Plant Unit No. 2 Reactor Vessel Radiation Surveillance Program, J. A. Davidson, et al., August 1977.

6. WCAP-10425, Analysis of Capsule U from the Alabama Power Company Joseph M. Farley Unit 2 Reactor Vessel Radiation Surveillance Program, S. E. Yanichko, et al., October 1983.

7. WCAP-11438, Analysis of Capsule W from the Alabama Power Company Joseph M. Farley Unit 2 Reactor Vessel Radiation Surveillance Program, S. E. Yanichko, et al., April 1987.

8. WCAP-14040-NP-A, Revision 2, Methodology Used to Develop Cold Overpressure Mitigating System Setpoints and RCS Heatup and Cooldown Limit Curves, January 1996.

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-15171, Revision 1, Analysis of Capsule Z from the Alabama Power Company Joseph M. Farley Unit 2 Reactor Vessel Radiation Surveillance Program, February 2000.

11. WCAP-16351-NP, Revision 0, Analysis of Capsule Y from the Southern Nuclear Operating Company Joseph M. Farley Unit 2 Reactor Vessel Radiation Surveillance Program, February 2005.

12. SNC letter NL-04-0372, Reactor Material Surveillance Program Specimen Capsule Withdrawal Schedule Revisions – Additional Information, March 5, 2004.

13. NRC letter (SNC LC #14001) Joseph M. Farley Nuclear Plant, Units 1 and 2 Re: Specimen Capsule Withdrawal Schedule Revisions, March 15, 2004.

14. NRC Regulatory Guide 1.190, "Calculational and Dosimetry Methods for Determining Pressure Vessel Neutron Fluence," March, 2001.