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L-2011-029
10 CFR 50.90

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

Re: Turkey Point Units 3 and 4
Docket Nos. 50-250 and 50-251
Response to NRC Request for Additional Information Regarding
Extended Power Uprate License Amendment Request No. 205 and
Reactor Materials Issues – Round 1

References:

(1) M. Kiley (FPL) to U.S. Nuclear Regulatory Commission (L-2010-113), "License Amendment Request No. 205: Extended Power Uprate (EPU)," (TAC Nos. ME4907 and ME4908), Accession No. ML103560169, October 21, 2010.
(2) Email from J. Paige (NRC) to S. Franzone (FPL), "EPU Acceptance Review Question Re: Equivalent Margin Analysis," Accession No. ML103070063, November 1, 2010.
(3) M. Kiley (FPL) to U.S. Nuclear Regulatory Commission (L-2010-268), "Response to NRC Request for Additional Information (RAI) Regarding Extended Power Uprate (EPU) License Amendment Request (LAR) No. 205 and Equivalent Margin Analysis (EMA)," November 12, 2010.
(4) M. Kiley (FPL) to U.S. Nuclear Regulatory Commission (L-2010-303), Supplemental Response to NRC Request for Additional Information (RAI) Regarding Extended Power Uprate (EPU) License Amendment Request (LAR) No. 205 and Equivalent Margin Analysis (EMA), Accession No. ML103610321, December 21, 2010.
(5) Email from J. Paige (NRC) to T. Abbatiello (FPL), "Turkey Point EPU - Vessels and Internals Integrity (CVIB) Requests for Additional Information - Round 1", Accession No. ML110420241, February 11, 2011

By letter L-2010-113 dated October 21, 2010 [Reference 1], Florida Power and Light Company (FPL) requested to amend Renewed Facility Operating Licenses DPR-31 and DPR-41 and revise the Turkey Point Units 3 and 4 Technical Specifications (TS). The proposed amendment will increase each unit's licensed core power level from 2300 megawatts thermal (MWt) to 2644 MWt and revise the Renewed Facility Operating Licenses and TS to support operation at this increased core thermal power level. This represents an approximate increase of 15% and is therefore considered an extended power uprate (EPU).

By email from the U.S. Nuclear Regulatory Commission (NRC) Project Manager (PM) dated November 1, 2010 [Reference 2], additional information regarding the Equivalent Margin Analysis (EMA) was requested by the NRC staff in the Vessels and Internals Integrity (CVIB) to support their acceptance review of the EPU License Amendment Request (LAR) [Reference 1]. FPL provided its responses to the NRC request by letters L-2010-268 and L-2010-303 dated November 12, 2010 and December 21, 2010, respectively [References 3 and 4]. The responses included AREVA NP Inc proprietary copies of Turkey Point EMA Reconciliation Report and ANP-2312P, Rev 3, "Low Upper-Shelf Toughness Fracture Mechanics Analysis of Reactor Vessels of Turkey Point Units 3 and 4 For Extended Life Through 48 Effective Full Power Years," January 2010.

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By email from the NRC PM dated February 11, 2011 [Reference 5], additional information regarding reactor materials issues was requested by the NRC staff in CVIB to support their review of Reference 1. The Request for Additional Information (RAI) consisted of six (6) questions regarding reactor vessel materials issues. These six RAI questions and the applicable FPL responses are documented in the Attachment to this letter.

In accordance with 10 CFR 50.91(b)(1), a copy of this letter is being forwarded to the State Designee of Florida.

This submittal does not alter the significant hazards consideration or environmental assessment previously submitted by FPL letter L-2010-113 [Reference 1].

This submittal contains no new commitments and no revisions to existing commitments.

Should you have any questions regarding this submittal, please contact Mr. Robert J. Tomonto, Licensing Manager, at (305) 246-7327.

I declare under penalty of perjury that the foregoing is true and correct.

Executed on March 9th, 2011.

Very truly yours,

Michael Kiley
Site Vice President
Turkey Point Nuclear Plant

Attachment

cc: USNRC Regional Administrator, Region II
USNRC Project Manager, Turkey Point Nuclear Plant
USNRC Resident Inspector, Turkey Point Nuclear Plant
Mr. W. A. Passetti, Florida Department of Health

Turkey Point Units 3 and 4

RESPONSE TO NRC RAI REGARDING EPU LAR NO. 205
AND CVIB REACTOR MATERIALS ISSUES – ROUND 1

ATTACHMENT

Response to Request for Additional Information

The following information is provided by Florida Power and Light Company (FPL) in response to the U. S. Nuclear Regulatory Commission's (NRC) Request for Additional Information (RAI). This information was requested to support License Amendment Request (LAR) 205, Extended Power Uprate (EPU), for Turkey Point Nuclear Plant (PTN) Units 3 and 4 that was submitted to the NRC by FPL via letter (L-2010-113) dated October 21, 2010 [Reference 1].

In an email dated November 1, 2010 [Reference 2], additional information regarding the PTN Equivalent Margin Analysis (EMA) was requested by the NRC's Vessels and Internals Integrity Branch (CVIB) to support their acceptance review of the EPU LAR. FPL provided responses to the NRC request by letters L-2010-268 and L-2010-303 dated November 12, 2010 and December 21, 2010, respectively [References 3 and 4]. The responses included AREVA NP Inc proprietary copies of Turkey Point EMA Reconciliation Report and ANP-2312P, Rev 3, "Low Upper-Shelf Toughness Fracture Mechanics Analysis of Reactor Vessels of Turkey Point Units 3 and 4 for Extended Life Through 48 Effective Full Power Years," January 2010 [References 5 and 6].

In an email dated February 11, 2011 [Reference 7], the NRC staff requested additional information regarding FPL's request to implement the Extended Power Uprate. The RAI consisted of six (6) questions from CVIB regarding reactor materials issues. These six RAI questions and the applicable FPL responses are documented below.

CVIB-1.1 The revised surveillance capsule withdrawal schedule for Turkey Point, Units 3 and 4 allows the last capsule, X4, to be withdrawn between 31.4 and 47.8 effective full power years (EFPY). This schedule does meet the recommendation of American Society for Testing and Materials (ASTM) E 185-82, "Standard Practice for Conducting Surveillance Tests for Light-Water Cooled Nuclear Power Reactor Vessels," that for a reactor with five surveillance capsules installed, the last capsule should be withdrawn at a fluence greater than once but less than twice the peak end-of-life (EOL) vessel fluence. However, the staff requests the licensee provide a single estimated EFPY value at which the capsule will be withdrawn rather than a range, or commit to providing this value later.

Surveillance Capsule X_4 will be withdrawn when it reaches a fluence that is approximately equivalent to the 80-year (67 EFPY) peak reactor vessel fluence of 8.14×10^{19} n/cm2 (E >1.0MeV). Therefore, accounting for EPU conditions, Capsule X_4 will be withdrawn at the vessel refueling date that is nearest to 35.8 EFPY. This withdrawal date of 35.8 EFPY will be specified in the Turkey Point Units 3 and 4 surveillance capsule withdrawal schedule. It should be noted that the withdrawal fluence is consistent with the fluence and the intent of the "Coordinated U. S. PWR Reactor Vessel Surveillance Program." However, the withdrawal EFPY recommended in it differs from the withdrawal EFPY listed above because the report did not consider the effects of EPU.

CVIB-1.2 The revised equivalent margins analysis (EMA) forwarded by letter dated December 21, 2010 (Reference 1), stated that the low-upper shelf fracture mechanics evaluation is performed according to the acceptance criteria and evaluation procedures contained in Appendix K to Section XI of the American Society of Mechanical Engineers Boiler & Pressure Vessel Code (ASME Code),

and references the ASME Code, Section XI, 1998 Edition through 2000 Addenda. Title 10 of Code of Federal Regulations (10 CFR) Part 50, Appendix G, IV.A.1.a, requires that such analyses use the latest edition and addenda of the ASME Code incorporated by reference into 10 CFR 50.55a(b)(2) at the time the analysis is submitted. The latest edition of the ASME Code, Section XI (Division 1) incorporated by reference into 10 CFR 50.55a at the time of the submittal is the 2004 edition. The staff therefore requests the licensee reconcile the differences between the 1998 through 2000 Addenda, and 2004 editions of the ASME Code, Section XI, specifically as the differences affect the low-upper shelf toughness evaluation.

With respect to a low upper-shelf toughness evaluation of reactor vessel steels,. there are minor differences between the 1998 Edition through 2000 Addendum and the 2004 Edition of the ASME Code. The two areas of the ASME Code which affect the low-upper shelf toughness evaluation performed for the Turkey Point vessels are the material properties in Section II, Part D, and the acceptance criteria and evaluation procedures of Section XI, Appendix K.

The material properties obtained from the ASME Code for use in the Turkey Point low-upper shelf toughness evaluation are listed in the following tables for the two versions of the Code.

1998 Edition through 2000 Addendum

	Base Metal Material			Cladding Material	
Temp	Young's Modulus	Coeff. of Thermal Expansion	ASME Yield Strength	Young's Modulus	Coeff. of Thermal Expansion
(F)	(ksi)	(in/in/F)	(ksi)	(ksi)	(in/in/F)
70	27800	6.40E-06	50.0	28300	8.50E-06
200	27100	6.70E-06	47.0	27600	8.90E-06
300	26700	6.90E-06	45.5	27000	9.20E-06
400	26100	7.10E-06	44.2	26500	9.50E-06
500	25700	7.30E-06	43.2	25800	9.70E-06
600	25200	7.40E-06	42.1	25300	9.80E-06

2004 Edition

	Base Metal Material			Cladding Material	
Temp	Young's Modulus	Coeff. of Thermal Expansion	ASME Yield Strength	Young's Modulus	Coeff. of Thermal Expansion
(F)	(ksi)	(in/in/F)	(ksi)	(ksi)	(in/in/F)
70	27800	6.40E-06	50.0	28300	8.50E-06
200	27100	6.70E-06	47.0	27500	8.90E-06
300	26700	6.90E-06	45.5	27000	9.20E-06
400	26200	7.10E-06	44.2	26400	9.50E-06
500	25700	7.30E-06	43.2	25900	9.70E-06
600	25100	7.40E-06	42.1	25300	9.80E-06

The 2004 changes to Code material properties, highlighted in gray, are less than 0.5% and only affect the Young's modulus. Using the values in the 2004 Edition would not significantly affect the results of the Turkey Point low-upper shelf toughness evaluation.

Regarding Appendix K to Section XI, the only difference between the two versions of the Code is the addition of SI Units in the 2004 Edition. This change would have no effect on the results of the Turkey Point low-upper shelf toughness evaluation.

CVIB-1.3 In Section 7 of Reference 1, the licensee indicated that the applied J-integral was calculated using the following equation:

$$J_{applied}(a) = 1000K^2_{Itotal}(a)(1-v^2)/E$$

This is essentially the same as the ASME Code, Section XI K-5210 equation:

$$J = 1000(K'_I)^2/E'$$

where:

$E' = E/(1-v^2)$
K'_I = stress intensity factor adjusted for small scale yielding.

Article K-5000, subparagraph K-5210 of the ASME Code, Section XI, Appendix K (2004 Edition), provides an adjustment of the effective flaw depth for small-scale yielding as follows:

$$a_e = a + [1/(6\pi)](K_I/\sigma_y)^2$$

where:

a = actual flaw depth,
a_e = effective flaw depth,
K_I = applied stress intensity,
σ_y = yield strength.

Paragraph K-5210 further states that the stress intensity factor for small scale yielding, K'_I, shall be calculated by substituting a_e for a.

The licensee did not discuss whether the effects of small scale yielding were included in the K_{Itotal} term. The staff therefore requests that the licensee discuss how the effects of small scale yielding were accounted for in the K_{Itotal} term.

Small-scale yielding is addressed in Appendix K to Section XI through use of a plastic zone correction to the postulated flaw depth, such that the effective flaw depth is expressed as

$$a_e = a + [1/(6\pi)](K_I/\sigma_y)^2 \qquad \text{Equation [1]}$$

This effective flaw depth is explicitly cited in Section 4 of ANP-2312P [Reference 6] for the prescriptive Appendix K flaw evaluation procedure for Levels A and B Service Loadings. Article K-5210 of Appendix K presents an overall procedure for calculating applied J-integrals for Levels C and D Service Loadings. The evaluation for Levels C and D Service Loadings requires plant specific transient analysis to determine pressure and thermal loads and stress intensity factors as a function of time. The PCRIT computer code used by AREVA to determine time-varying stress intensity factors has a built-in feature to calculate the effective flaw depth described by Equation [1]. This option of the code was used in the Turkey Point low-upper shelf toughness evaluation for Levels C and D Service Loadings.

CVIB-1.4 Provide the basis, such as a report or calculation, for the pressure-temperature (P-T) limits for Turkey Point, Units 3 and 4 that are given in proposed revised Technical Specification Figures 3.4-2 and 3.4-3. If the report or calculation does not contain the following items, then the following items should be provided separately:

a. Provide a tabulation of the thermal stress intensity factors (K_{It}) used to generate the heatup and cooldown curves for each coolant temperature for heatup and cooldown.

Per the agreement reached during the telephone conference call on February 3, 2011 involving the NRC, FPL, and Westinghouse, the thermal stress intensity factors are being provided for only the most limiting heatup and cooldown rates (100°F/hr). The K_{It} values for the 100°F/hr heatup rate are presented in Table 1. The K_{It} values for the 100°F/hr cooldown rate are presented in Table 2.

Table 1
K_{It} Values for 100°F/hr Heatup Curve for 48 EFPY

Water Temp. (°F)	1/4T Thermal Stress Intensity Factor (ksi√in.)	3/4T Thermal Stress Intensity Factor (ksi√in.)
70	-0.9847	0.4966
75	-2.3616	1.4564
80	-3.4847	2.3644
85	-4.5262	3.1774
90	-5.3926	3.8779
95	-6.1705	4.4891
100	-6.8263	5.0171
105	-7.4130	5.4784
110	-7.9115	5.8789
115	-8.3582	6.2297
120	-8.7400	6.5358
125	-9.0835	6.8051
130	-9.3790	7.0415
135	-9.6465	7.2507
140	-9.8782	7.4356
145	-10.0895	7.6004
150	-10.2739	7.7473
155	-10.4437	7.8792
160	-10.5931	7.9979
165	-10.7322	8.1055
170	-10.8556	8.2032

Water Temp. (^{o}F)	1/4T Thermal Stress Intensity Factor (ksi√in.)	3/4T Thermal Stress Intensity Factor (ksi√in.)
175	-10.9718	8.2929
180	-11.0761	8.3751
185	-11.1754	8.4513
190	-11.2654	8.5221
195	-11.3520	8.5884
200	-11.4315	8.6505
205	-11.5088	8.7094
210	-11.5803	8.7653
215	-11.6507	8.8187
220	-11.7164	8.8698
225	-11.7816	8.9191
230	-11.8430	8.9667
235	-11.9043	9.0130
240	-11.9626	9.0581
245	-12.0210	9.1021
250	-12.0769	9.1452
255	-12.1332	9.1876
260	-12.1874	9.2293
265	-12.2422	9.2705
270	-12.2952	9.3112
275	-12.3488	9.3515
280	-12.4009	9.3914
285	-12.4537	9.4311
290	-12.5052	9.4705
295	-12.5575	9.5097
300	-12.6085	9.5487
305	-12.6604	9.5877
310	-12.7112	9.6265
315	-12.7628	9.6652
320	-12.8134	9.7039
325	-12.8649	9.7425
330	-12.9155	9.7811
335	-12.9668	9.8196
340	-13.0174	9.8582
345	-13.0688	9.8967
350	-13.1194	9.9353
355	-13.1708	9.9739
360	-13.2215	10.0125
365	-13.2730	10.0512
370	-13.3239	10.0899

Water Temp. (°F)	1/4T Thermal Stress Intensity Factor (ksi√in.)	3/4T Thermal Stress Intensity Factor (ksi√in.)
375	-13.3754	10.1286
380	-13.4264	10.1674
385	-13.4781	10.2062
390	-13.5292	10.2451
395	-13.5810	10.2840
400	-13.6323	10.3230
405	-13.6843	10.3621
410	-13.7358	10.4012
415	-13.7879	10.4404
420	-13.8396	10.4796
425	-13.8918	10.5189
430	-13.9437	10.5583
435	-13.9961	10.5977
440	-14.0482	10.6373
445	-14.1008	10.6768
450	-14.1531	10.7165
455	-14.2059	10.7562
460	-14.2584	10.7960
465	-14.3114	10.8359
470	-14.3640	10.8758
475	-14.4172	10.9159
480	-14.4701	10.9559
485	-14.5234	10.9961
490	-14.5765	11.0363
495	-14.6301	11.0767
500	-14.6833	11.1170
505	-14.7371	11.1575
510	-14.7906	11.1980
515	-14.8445	11.2387
520	-14.8982	11.2793
525	-14.9524	11.3201
530	-15.0063	11.3609
535	-15.0606	11.4019
540	-15.1147	11.4428
545	-15.1692	11.4839
550	-15.2236	11.5250

Table 2
K_{It} Values for 100°F/hr Cooldown Curve for 48 EFPY

Water Temp. (°F)	1/4T Thermal Stress Intensity Factor (ksi√in.)
545	0.9598
540	2.4111
535	3.7032
530	4.9429
525	6.0336
520	7.0381
515	7.9218
510	8.7274
505	9.4366
500	10.0798
495	10.6456
490	11.1567
485	11.6052
480	12.0089
475	12.3616
470	12.6778
465	12.9524
460	13.1973
455	13.4083
450	13.5953
445	13.7546
440	13.8945
435	14.0118
430	14.1135
425	14.1966
420	14.2673
415	14.3228
410	14.3684
405	14.4015
400	14.4269
395	14.4419
390	14.4509
385	14.4514
380	14.4472
375	14.4360

Water Temp. (°F)	1/4T Thermal Stress Intensity Factor (ksi√in.)
370	14.4212
365	14.4007
360	14.3774
355	14.3493
350	14.3193
345	14.2852
340	14.2498
335	14.2110
330	14.1712
325	14.1287
320	14.0855
315	14.0401
310	13.9943
305	13.9465
300	13.8986
295	13.8490
290	13.7995
285	13.7486
280	13.6979
275	13.6459
270	13.5943
265	13.5416
260	13.4892
255	13.4360
250	13.3831
245	13.3295
240	13.2764
235	13.2225
230	13.1691
225	13.1151
220	13.0616
215	13.0075
210	12.9540
205	12.9000
200	12.8465
195	12.7925
190	12.7391
185	12.6852

Water Temp. (°F)	1/4T Thermal Stress Intensity Factor (ksi√in.)
180	12.6319
175	12.5782
170	12.5250
165	12.4715
160	12.4185
155	12.3651
150	12.3123
145	12.2591
140	12.2065
135	12.1536
130	12.1012
125	12.0485
120	11.9963
115	11.9438
110	11.8918
105	11.8396
100	11.7879
95	11.7359
90	11.6844
85	11.6326
80	11.5814
75	11.5298
70	11.4788

b. **Provide a tabulation or graph of the temperature differential from the coolant to the crack tip used to generate the P-T limits, and describe the methodology used to determine this differential, unless Figure G-2214-1 and Figure G-2214-2 of the ASME Code, Section XI, Appendix G, were used to determine the temperature differential.**

Per the agreement reached during the telephone conference call on February 3, 2011 involving the NRC, FPL, and Westinghouse, the coolant and crack tip temperatures will be provided only for the most limiting heatup and cooldown rates (100°F/hr). The temperature values for the 100°F/hr heatup rate are presented in Table 3. The temperature values for the 100°F/hr cooldown rate are presented in Table 4.

Regarding the methodology used in calculating temperature differential, the temperatures are calculated using the one-dimensional transient heat conduction equation that is contained in Section 2.6.1 of WCAP-14040-A, Revision 4 [Reference 8]. A through-wall temperature distribution was calculated for each time step during each cooldown or heatup ramp of interest. These methods are incorporated into the OPERLIM computer code.

Table 3
Temperature Values for 100°F/hr Heatup Curve for 48 EFPY

Water Temperature (°F)	1/4T Crack Tip Temperature (°F)	3/4T Crack Tip Temperature (°F)
70	66.156	65.070
75	68.992	65.451
80	72.216	66.385
85	75.670	67.874
90	79.389	69.826
95	83.228	72.190
100	87.263	74.904
105	91.390	77.918
110	95.655	81.191
115	99.995	84.686
120	104.432	88.373
125	108.927	92.223
130	113.491	96.214
135	118.101	100.326
140	122.760	104.543
145	127.455	108.849
150	132.183	113.232
155	136.940	117.681
160	141.721	122.187
165	146.524	126.742
170	151.344	131.340
175	156.181	135.974
180	161.029	140.639
185	165.891	145.331
190	170.761	150.046
195	175.642	154.782
200	180.528	159.534
205	185.422	164.302
210	190.320	169.082
215	195.224	173.873
220	200.131	178.674
225	205.043	183.484
230	209.957	188.300
235	214.874	193.122
240	219.793	197.950

Water Temperature (°F)	1/4T Crack Tip Temperature (°F)	3/4T Crack Tip Temperature (°F)
245	224.714	202.782
250	229.637	207.619
255	234.561	212.458
260	239.486	217.300
265	244.413	222.145
270	249.340	226.992
275	254.268	231.841
280	259.197	236.692
285	264.126	241.543
290	269.056	246.396
295	273.986	251.250
300	278.916	256.104
305	283.847	260.959
310	288.778	265.815
315	293.709	270.671
320	298.640	275.528
325	303.572	280.384
330	308.503	285.241
335	313.435	290.098
340	318.366	294.956
345	323.298	299.813
350	328.229	304.670
355	333.161	309.528
360	338.092	314.385
365	343.023	319.242
370	347.955	324.099
375	352.886	328.956
380	357.817	333.813
385	362.749	338.670
390	367.680	343.526
395	372.611	348.383
400	377.542	353.239
405	382.472	358.095
410	387.403	362.951
415	392.334	367.806
420	397.264	372.662
425	402.195	377.517
430	407.125	382.372
435	412.055	387.227
440	416.985	392.081
445	421.915	396.935

Water Temperature (°F)	1/4T Crack Tip Temperature (°F)	3/4T Crack Tip Temperature (°F)
450	426.845	401.789
455	431.775	406.643
460	436.704	411.497
465	441.634	416.350
470	446.563	421.203
475	451.492	426.056
480	456.422	430.909
485	461.351	435.761
490	466.279	440.613
495	471.208	445.465
500	476.137	450.316
505	481.065	455.168
510	485.994	460.019
515	490.922	464.869
520	495.850	469.720
525	500.778	474.570
530	505.706	479.420
535	510.633	484.270
540	515.561	489.119
545	520.489	493.968
550	525.416	498.817

Table 4
Temperature Values for 100°F/hr Cooldown Curve for 48 EFPY

Water Temperature (°F)	1/4T Crack Tip Temperature (°F)
545	549.091
540	546.642
535	543.653
530	540.452
525	536.970
520	533.372
515	529.588
510	525.700
505	521.678
500	517.562
495	513.345
490	509.045
485	504.668
480	500.220
475	495.712
470	491.144
465	486.529
460	481.864
455	477.161
450	472.417
445	467.643
440	462.836
435	458.003
430	453.144
425	448.264
420	443.363
415	438.445
410	433.510
405	428.561
400	423.598
395	418.624
390	413.639
385	408.645
380	403.642
375	398.631
370	393.613
365	388.590
360	383.561

Water Temperature (°F)	1/4T Crack Tip Temperature (°F)
355	378.527
350	373.488
345	368.446
340	363.400
335	358.350
330	353.298
325	348.244
320	343.187
315	338.129
310	333.068
305	328.006
300	322.943
295	317.879
290	312.813
285	307.746
280	302.679
275	297.611
270	292.543
265	287.473
260	282.404
255	277.334
250	272.263
245	267.193
240	262.122
235	257.051
230	251.979
225	246.908
220	241.837
215	236.765
210	231.693
205	226.622
200	221.550
195	216.478
190	211.407
185	206.335
180	201.264
175	196.192
170	191.121
165	186.050
160	180.979
155	175.907

Water Temperature (°F)	1/4T Crack Tip Temperature (°F)
150	170.836
145	165.765
140	160.695
135	155.624
130	150.553
125	145.483
120	140.413
115	135.342
110	130.272
105	125.202
100	120.132
95	115.062
90	109.993
85	104.923
80	99.854
75	94.785
70	89.716

c. **Provide the numerical temperature versus pressure values corresponding to the heatup and cooldown curves, and the hydrotest curve, in Technical Specification Figures 3.4-2 and 3.4-3.**

The numerical temperature versus pressure values for the heatup curves and the hydrotest curve are presented in Table 5. The numerical temperature versus pressure values for the cooldown curves are presented in Table 6.

Table 5
Data Points for Heatup P-T Limit Curves Applicable to 48 EFPY with Flange, without Temperature and Pressure Uncertainties, and Using Combined Methodology[(a)]

Leak Test Limit		60°F/hr Heatup		60°F/hr Criticality		100°F/hr Heatup		100°F/hr Criticality	
T (°F)	P (psig)	T (°F)	P (psig)	T (°F)	P (psig)	T (°F)	P (psig)	T (°F)	P (psig)
238	2000	70	0	*262*	*0*	70	0	*262*	*0*
238	2000	70	587	*262*	*621*	70	552	*262*	*621*
262	*2485*	75	587	*262*	*621*	75	552	*262*	*621*
262	*2485*	80	587	*262*	*621*	80	552	*262*	*621*
		85	587	*262*	*621*	85	552	*262*	*621*
		90	587	*262*	*621*	90	552	*262*	*621*
		95	587	*262*	*621*	95	552	*262*	*621*
		100	588	*262*	*621*	100	552	*262*	*621*
		105	591	*262*	*621*	105	552	*262*	*621*

Leak Test Limit		60°F/hr Heatup		60°F/hr Criticality		100°F/hr Heatup		100°F/hr Criticality	
T (°F)	P (psig)	T (°F)	P (psig)	T (°F)	P (psig)	T (°F)	P (psig)	T (°F)	P (psig)
		110	596	*262*	*621*	110	552	*262*	*621*
		115	603	*262*	*621*	115	552	*262*	*621*
		120	612	*262*	*1018*	120	553	*262*	*920*
		120	612	*262*	*1023*	120	553	*262*	*921*
		120	612	*262*	*1029*	120	553	*262*	*921*
		125	621	*262*	*1037*	125	557	*262*	*923*
		130	632	*262*	*1046*	130	561	265	946
		135	645	*262*	*1056*	135	567	270	988
		140	658	*262*	*1067*	140	575	275	1033
		145	673	*262*	*1079*	145	583	280	1081
		150	690	*262*	*1093*	150	594	285	1133
		155	708	*262*	*1108*	155	605	290	1188
		160	727	*262*	*1124*	160	618	295	1248
		165	748	*262*	*1141*	165	632	300	1313
		170	770	*262*	*1137*	170	648	305	1382
		175	795	265	1167	175	666	310	1456
		180	821	270	1222	180	685	315	1536
		185	849	275	1280	185	705	320	1621
		190	880	280	1343	190	728	325	1713
		195	912	285	1410	195	752	*330*	*1801*
		200	947	290	1483	200	779	*335*	*1880*
		205	985	295	1560	205	808	*340*	*1964*
		210	1026	300	1644	210	838	*345*	*2055*
		215	1070	305	1734	215	872	*350*	*2153*
		220	1117	310	1830	220	908	*355*	*2258*
		225	1167	315	1934	225	946	*360*	*2371*
		230	1222	320	2045	230	988		
		235	1280	*325*	*2159*	235	1033		
		240	1343	*330*	*2255*	240	1081		
		245	1410	*335*	*2348*	245	1133		
		250	1483	*340*	*2443*	250	1188		
		255	1560			255	1248		
		260	1644			260	1313		
		265	1734			265	1382		
		270	1830			270	1456		
		275	1934			275	1536		
		280	2045			280	1621		
		285	*2159*			285	1713		
		290	*2255*			*290*	*1801*		
		295	*2348*			*295*	*1880*		

Leak Test Limit		**60°F/hr Heatup**		**60°F/hr Criticality**		**100°F/hr Heatup**		**100°F/hr Criticality**	
T (°F)	**P (psig)**	**T (°F)**	**P (psig)**	**T (°F)**	**P (psig)**	**T (°F)**	**P (psig)**	**T (°F)**	**P (psig)**
		300	*2443*			*300*	*1964*		
						305	*2055*		
						310	*2153*		
						315	*2258*		
						320	*2371*		

(a) Pressure values in *italics* resulted from the use of circumferential flaw methodology. All other pressure values resulted from the use of axial flaw methodology.

Table 6

Data Points for Cooldown P-T Limit Curves Applicable to 48 EFPY with Flange, without Temperature and Pressure Uncertainties, and Using Combined Methodology[(a)]

Steady State		**20°F/hr.**		**40°F/hr.**		**60°F/hr.**		**100°F/hr.**	
T (°F)	**P (psig)**	**T (°F)**	**P (psig)**	**T (°F)**	**P (psig)**	**T (°F)**	**P (psig)**	**T (°F)**	**P (psig)**
70	0	70	0	70	0	70	0	70	0
70	602	70	565	70	527	70	488	70	409
75	609	75	572	75	534	75	495	75	417
80	616	80	579	80	541	80	503	80	425
85	621	85	586	85	549	85	511	85	434
90	621	90	595	90	558	90	520	90	444
95	621	95	604	95	567	95	530	95	455
100	621	100	613	100	577	100	541	100	466
105	621	105	621	105	588	105	552	105	479
110	621	110	621	110	600	110	564	110	493
115	621	115	621	115	612	115	577	115	507
120	621	120	621	120	621	120	592	120	523
120	621	120	621	120	626	125	607	125	540
120	694	120	660	125	640	130	624	130	559
125	707	125	674	130	656	135	641	135	579
130	721	130	689	135	673	140	661	140	600
135	737	135	705	140	691	145	682	145	624
140	753	140	722	145	711	150	704	150	649
145	771	145	741	150	732	155	728	155	676
150	790	150	761	155	755	160	754	160	705
155	810	155	783	160	780	165	782	165	737
160	833	160	806	165	806	170	812	170	771

Steady State		**20°F/hr.**		**40°F/hr.**		**60°F/hr.**		**100°F/hr.**	
T (°F)	**P (psig)**	**T (°F)**	**P (psig)**	**T (°F)**	**P (psig)**	**T (°F)**	**P (psig)**	**T (°F)**	**P (psig)**
165	856	165	831	170	835	175	845	175	808
170	882	170	858	175	865	180	880	180	847
175	909	175	887	180	898	185	918	185	890
180	939	180	918	185	934	190	958	190	936
185	970	185	952	190	972	195	1002	*195*	*970*
190	1005	190	988	195	1014	200	1049	*200*	*999*
195	1041	195	1027	200	1058	205	1100	*205*	*1030*
200	1081	200	1068	205	1106	210	1155	*210*	*1064*
205	1123	205	1113	210	1157	215	1212	*215*	*1100*
210	1169	210	1162	215	1212	220	1270	*220*	*1140*
215	1218	215	1214	220	1270	*225*	*1324*	*225*	*1182*
220	1270	220	1270	225	1327	*230*	*1365*	*230*	*1228*
225	1327	225	1327	230	1388	*235*	*1409*	*235*	*1278*
230	1388	230	1388	235	1453	*240*	*1456*	*240*	*1332*
235	1453	235	1453	*240*	*1520*	*245*	*1507*	*245*	*1389*
240	1524	240	1524	*245*	*1568*	*250*	*1562*	*250*	*1452*
245	1599	245	1599	*250*	*1620*	*255*	*1621*	*255*	*1519*
250	1681	*250*	*1680*	*255*	*1676*	*260*	*1685*	*260*	*1592*
255	1768	*255*	*1732*	*260*	*1736*	*265*	*1754*	*265*	*1670*
260	*1843*	*260*	*1789*	*265*	*1801*	*270*	*1828*	*270*	*1755*
265	*1901*	*265*	*1850*	*270*	*1870*	*275*	*1908*	*275*	*1846*
270	*1962*	*270*	*1915*	*275*	*1945*	*280*	*1994*	*280*	*1944*
275	*2029*	*275*	*1985*	*280*	*2026*	*285*	*2087*	*285*	*2050*
280	*2100*	*280*	*2061*	*285*	*2113*	*290*	*2186*	*290*	*2164*
285	*2177*	*285*	*2143*	*290*	*2206*	*295*	*2294*	*295*	*2287*
290	*2259*	*290*	*2230*	*295*	*2307*	*300*	*2409*	*300*	*2409*
295	*2348*	*295*	*2325*	*300*	*2415*				
300	*2443*	*300*	*2426*						
305	*2545*								

(a) Pressure values in *italics* resulted from the use of circumferential flaw methodology. All other pressure values resulted from the use of axial flaw methodology.

d. **The P-T curves provided in EPU Licensing Report Figures 2.1.2-1 and 2.1.2-2 and TS Figures 3.4-2 and 3.4-3 do not indicate whether there is any pressure difference between the reactor vessel (RV) pressure and pressure at the measurement location. If such a pressure difference exists, provide the correction factors used to correct between the actual reactor vessel (RV) pressure and the indicated pressure at the measurement location.**

   The P-T limit curves do not include margin for the pressure difference between the RV pressure and the pressure at the measurement location. The PTN Overpressure Mitigation System (OMS) power-operated relief valve (PORV) setpoint, which prevents the P-T limits from being exceeded, does however account for a pressure differential of 57.4 psi (see Licensing Report (LR) Section 2.8.4.3.2.3) between the pressure measurement location and the RV. Since the OMS setpoint includes the impact of the pressure differential, it is not necessary to include this impact in the P-T limit curves.

e. **In the technical specification (TS) bases markups provided with the EPU application, the licensee provided a revised Table B 3/4.4-1 that shows the closure flange RTNDT has been changed from 44 °F to -50 °F. Therefore, the staff requests the licensee provide the basis for changing the highest RTNDT of the closure flange region that is highly stressed by bolt preload from 44 °F to -50 °F.**

   The closure head for each Unit was replaced. The initial RT_{NDT} values of the new closure heads are -50°F. Therefore, the P-T limit curves were developed based on the limiting initial RT_{NDT} in the flange region, which pertains to the Unit 4 vessel flange initial RT_{NDT} of -1°F.

f. **The EPU Licensing Report Figures and the marked up TS bases 3/4.4.9 indicate that the revised P-T limits are based on the KIa methodology of the 1996 Edition of ASME Code, Section XI, Appendix G. Since 1996 is an addenda rather than an edition of the ASME Code, the staff requests the licensee confirm that the revised P-T limits are based on the 1995 Edition through 1996 Addenda of the ASME Code, Section XI, Appendix G, and requests the licensee modify the TS bases accordingly.**

   The TS bases have been modified accordingly to cite that the P-T limit curves were developed based on the 1995 Edition through 1996 Addenda version of the ASME Code, Section XI, Appendix G. See attached Figure 1 for marked up TS bases pages 70, 71, and 75.

g. **Provide the following information related to the determination of the adjusted reference temperature (ART) for the limiting RV beltline materials:**

   1. **supporting data for, and the calculation of, the chemistry factors for those reactor vessel (RV) materials that have surveillance data;**

      Tables 7 and 8 provide this information.

Table 7
Calculation of Chemistry Factors using Turkey Point Unit 3 Surveillance Capsule Data

Material	Capsule	Fluence[(a)] (n/cm^2, E>1.0 MeV)	FF	ΔRT_{NDT} (°F)	FF* ΔRT_{NDT} (°F)	FF^2
Intermediate Shell Forging 123P461VA1	T_3	0.599 x 10^{19}	0.856	11.48[(c)]	9.83	0.734
	S_3	1.272 x 10^{19}	1.067	2.83[(c)]	3.02	1.138
				Sum:	12.85	1.872
	$CF_{123P461VA1} = \Sigma(FF * \Delta RT_{NDT}) \div \Sigma(FF^2) = (12.85) \div (1.872) =$ **6.9°F**					
Lower Shell Forging 123S266VA1	S_3	1.272 x 10^{19}	1.067	48.55[(c)]	51.80	1.138
	V_3	1.223 x 10^{19}	1.056	42.68[(c)]	45.08	1.115
	X_3	2.897 x 10^{19}	1.282	72.44[(c)]	92.89	1.644
				Sum:	189.77	3.898
	$CF_{123S266VA1} = \Sigma(FF * \Delta RT_{NDT}) \div \Sigma(FF^2) = (189.77) \div (3.898) =$ **48.7°F**					
Weld Metal Heat #71249	Davis Besse	2.956 x 10^{19} [(f)]	1.287	188.1[(d, e)] (215[(f)])	242.1	1.657
	T_3	0.599 x 10^{19}	0.856	141.4[(e)] (163.87[(c)])	121.1	0.734
	V_3	1.223 x 10^{19}	1.056	156.0[(e)] (180.77[(c)])	164.8	1.115
	T_4	0.649 x 10^{19}	0.879	182.1[(e)] (211[(b)])	160.0	0.772
	X_3	2.897 x 10^{19}	1.282	164.9[(e)] (191.06[(c)])	211.4	1.644
				Sum:	899.5	5.923
	$CF_{71249} = \Sigma(FF * \Delta RT_{NDT}) \div \Sigma(FF^2) = (899.5) \div (5.923) =$ **151.9°F**					

Notes:
(a) Capsule fluence values were updated as part of the EPU, unless otherwise noted.
(b) Values taken from WCAP-15092, Revision 3 [Reference 9].
(c) Values taken from WCAP-15916 [Reference 10].
(d) A 9°F correction factor was used in the calculation of this value to account for the difference in operating temperatures between Turkey Point and Davis Besse.
(e) Final ΔRT_{NDT} value has been adjusted using the ratio procedure. For the Davis Besse capsule, the ratio is 0.833. For the Turkey Point Units 3 and 4 capsules, the ratio is 0.863.
(f) Values taken from WCAP-15885, Revision 0 [Reference 11].

Table 8
Calculation of Chemistry Factors Using Turkey Point Unit 4 Surveillance Capsule Data

Material	Capsule	Fluence[a] (n/cm^2, E>1.0 MeV)	FF	ΔRT_{NDT} (°F)	FF* ΔRT_{NDT} (°F)	FF^2
Intermediate Shell Forging 123P481VA1	S_4	1.29	1.071	35[c]	37.5	1.147
				Sum:	(e)	(e)
	$CF_{123P481VA1}$ = N/A					
Lower Shell Forging 122S180VA1	T_4	0.649	0.879	12[c]	10.6	0.772
	S_4	1.29	1.071	0[c]	0	1.147
				Sum:	10.6	1.919
	$CF_{122S180VA1} = \Sigma(FF * \Delta RT_{NDT}) \div \Sigma(FF^2) = (10.6) \div (1.919) =$ **5.5°F**					
Weld Metal Heat #71249	Davis Besse	2.956[g]	1.287	188.1[d, f] (215[g])	242.1	1.657
	T_3	0.599	0.856	141.4[f] (163.87[b])	121.1	0.734
	V_3	1.223	1.056	156.0[f] (180.77[b])	164.8	1.115
	T_4	0.649	0.879	182.1[f] (211[c])	160.0	0.772
	X_3	2.897	1.282	164.9[f] (191.06[b])	211.4	1.644
				Sum:	899.5	5.923
	$CF_{71249} = \Sigma(FF * \Delta RT_{NDT}) \div \Sigma(FF^2) = (899.5) \div (5.923) =$ **151.9°F**					

Notes:
(a) Capsule fluence values were updated as part of the EPU, unless otherwise noted.
(b) Values taken from WCAP-15916 [Reference 10].
(c) Values taken from WCAP-15092, Revision 3 [Reference 9].
(d) A 9°F correction factor was used in the calculation of this value to account for the difference in operating temperatures between Turkey Point and Davis Besse.
(e) In order to apply this calculation, there must be at least two data points for the material.
(f) Final ΔRT_{NDT} value has been adjusted using the ratio procedure. For the Davis Besse capsule, the ratio is 0.833. For the Turkey Point Units 3 and 4 capsules, the ratio is 0.863.
(g) Values taken from WCAP-15885, Revision 0 [Reference 11].

2. **the copper and nickel values for the surveillance materials;**

   This information is provided in Table 9.

Table 9
Copper and Nickel Values for Surveillance Weld Metal Heat # 71249

Plant	Cu Wt. %	Ni Wt. %
Turkey Point Units 3 and 4	0.31	0.57
Davis Besse	0.33	0.57

3. the credibility evaluation of the surveillance data; and

Introduction

Regulatory Guide (RG) 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. Positions 2.1 and 2.2 of RG 1.99, Revision 2, describe the method for calculating the adjusted reference temperature and Charpy upper shelf energy of reactor vessel beltline materials using surveillance capsule data. The methods of Positions 2.1 and 2.2 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 Turkey Point Unit 3 reactor vessel and two from the Turkey Point Unit 4 reactor vessel. This capsule data must be shown to be credible. In accordance with the discussion of RG 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 document the information provided by FPL in regard to the Turkey Point Units 3 and 4 reactor vessel surveillance data for each of the credibility requirements of RG 1.99, Revision 2.

Evaluation

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

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

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

The forging materials and weld metal contained in the capsules are representative of all of the materials in the Turkey Point Units 3 and 4 reactor vessel beltline regions. Therefore, this criterion is met.

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 USE unambiguously.*

No surveillance capsule data has been analyzed since the time that Capsule X_3 was analyzed in WCAP-15916 [Reference 10]. Based on the plots contained in WCAP-15916, this criterion is met.

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 USE if the upper shelf can be clearly determined, following the definition given in ASTM E185-82.*

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

The Turkey Point Unit 3 intermediate shell and lower shell forgings and surveillance weld material will be evaluated for credibility. The Turkey Point Unit 4 lower shell forging and surveillance weld material will be evaluated for credibility. Since the plants have an integrated surveillance program, the surveillance weld material evaluation will be identical between plants and thus applicable to both plants. The weld is made from weld wire heat 71249; Turkey Point Units 3 and 4 have a sister plant that shares the same weld wire heat and thus, utilize data from a sister plant (Davis Besse). The method of RG 1.99, Revision 2 will be followed for determining credibility of the weld as well as the forging material.

Credibility Assessment

The chemistry factors for the Turkey Point Units 3 and 4 surveillance forging and weld material contained in the surveillance program were calculated in accordance with RG 1.99, Revision 2, Position 2.1 and presented in Tables 7 and 8 of this letter. A new fitted chemistry factor for the Turkey Point Units 3 and 4 weld material will be calculated only for the purposes of this credibility evaluation. For this evaluation, the adjustment for chemistry differences between the beltline weld and surveillance weld will not be taken into account. The fitted chemistry factor calculation for the weld material is shown in Table 10. The scatter of ΔRT_{NDT} values about the functional form of a best-fit line drawn as described in Regulatory Position 2.1 is presented in Tables 11 and 12 for Turkey Point Units 3 and 4, respectively.

Table 10
Calculated CF for Turkey Point Units 3 and 4 Weld Heat # 71249 Using Turkey Point Units 3 and 4 and Davis Besse Surveillance Capsule Data

Material	Capsule	Capsule f ($x10^{19}$ n/cm^2)	FF	Measured ΔRT_{NDT} (°F)	Adjusted ΔRT_{NDT} (°F)	FF * ΔRT_{NDT} (°F)	FF^2
Surveillance Weld Metal Heat #71249	Davis Besse	2.956	1.287	215	224	288.3	1.657
	T_3	0.599	0.856	163.9	163.9	140.4	0.734
	V_3	1.223	1.056	180.8	180.8	190.9	1.115
	T_4	0.649	0.879	211	211	185.4	0.772
	X_3	2.897	1.282	191.1	191.1	245.0	1.644
					Sum:	1050.0	5.923
	$CF_{71249}= \Sigma(FF * \Delta RT_{NDT}) \div \Sigma(FF^2) = (1050.0) \div (5.923) =$ **177.3°F**						

Table 11
Turkey Point Unit 3 Surveillance Capsule Data Scatter about the Best-Fit Line

Material	Capsule	CF ($Slope_{best\ fit}$) (°F)	Capsule f ($x10^{19}$ n/cm^2)	FF	Adjusted ΔRT_{NDT} (°F)	Predicted ΔRT_{NDT} (°F)	Scatter ΔRT_{NDT}[(a)] (°F)	<17°F (Base Metal) <28°F (Weld)
IS Forging 123P461VA1	T_3	6.9	0.599	0.856	11.5	5.9	5.6	Yes
	S_3	6.9	1.272	1.067	2.8	7.3	4.5	Yes
LS Forging 123S266VA1	S_3	48.7	1.272	1.067	48.6	51.9	3.4	Yes
	V_3	48.7	1.223	1.056	42.7	51.4	8.7	Yes
	X_3	48.7	2.897	1.282	72.4	62.4	10.0	Yes
Surveillance Weld Metal Heat #71249	Davis Besse	177.3	2.956	1.287	224	228.2	4.2	Yes
	T_3	177.3	0.599	0.856	163.9	151.9	12.0	Yes
	V_3	177.3	1.223	1.056	180.8	187.2	6.5	Yes
	T_4	177.3	0.649	0.879	211	155.8	55.2	No
	X_3	177.3	2.897	1.282	191.1	227.4	36.3	No

Note:
(a) For the ΔRT_{NDT} scatter, absolute values are listed.

Turkey Point Unit 3

Table 11 indicates that zero of the surveillance data points fall outside the $\pm 1\sigma$ of 17°F scatter band for base metals. Therefore, the intermediate shell forging and lower shell forging data is deemed "credible" per the third criterion. Table 11 indicates that two of the five surveillance data points fall outside the $\pm 1\sigma$ of 28°F scatter band for surveillance weld materials. Therefore the surveillance weld data is deemed "not credible" per the third criterion.

Table 12
Turkey Point Unit 4 Surveillance Capsule Data Scatter about the Best-Fit Line

Material	Capsule	CF ($Slope_{best\ fit}$) (°F)	Capsule f ($x10^{19}$ n/cm^2)	FF	Adjusted ΔRT_{NDT} (°F)	Predicted ΔRT_{NDT} (°F)	Scatter ΔRT_{NDT}[(a)] (°F)	<17°F (Base Metal) <28°F (Weld)
IS Forging 123P481VA1	S_4	N/A	1.29	1.071	35	N/A	N/A	N/A
LS Forging 122S180VA1	T_4	5.5	0.649	0.879	12	4.8	7.2	Yes
	S_4	5.5	1.29	1.071	0	5.9	5.9	Yes
Surveillance Weld Metal Heat # 71249	Davis Besse	177.3	2.956	1.287	224	228.2	4.2	Yes
	T_3	177.3	0.599	0.856	163.9	151.9	12.0	Yes
	V_3	177.3	1.223	1.056	180.8	187.2	6.5	Yes
	T_4	177.3	0.649	0.879	211	155.8	55.2	No
	X_3	177.3	2.897	1.282	191.1	227.4	36.3	No

Note:
(a) For the ΔRT_{NDT} scatter, absolute values are listed.

Turkey Point Unit 4

Table 12 indicates that zero of the surveillance data points fall outside the $\pm 1\sigma$ of 17°F scatter band for base metals. Therefore, the lower shell forging data is deemed "credible" per the third criterion. Table 12 indicates that two of the five surveillance data points fall outside the $\pm 1\sigma$ of 28°F scatter band for surveillance weld materials. Therefore the surveillance weld data is deemed "not credible" per the third criterion.

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

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

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

The Turkey Point Unit 3 surveillance program does contain correlation monitor material. This evaluation will be re-evaluated using the updated surveillance capsule fluence values. NUREG/CR-6413, ORNL/TM-13133 [Reference 12], contains a plot of Residual vs. Fast Fluence for the correlation monitor material (Figure 10 of the NUREG Report). The Figure shows a 2σ uncertainty of 50°F. The data used for this plot is contained in Table 15 in the NUREG Report. However, the data in the NUREG report has not been considered for the recalculated fluence values as documented herein. Thus, Table 13 below presents an updated calculation of Residual vs. Fast Fluence.

Table 13
Calculation of Residual vs. Fast Fluence

Capsule	Fluence ($x10^{19}$ n/cm^2)	Fluence Factor (FF)	Measured Shift (°F)	RG 1.99 Shift (CF*FF[b]) (°F)	Residual (Measured – RG Shift)
S_3	1.272	1.067	106.7[a]	106.7	0
T_3	0.5990	0.856	86.66[a]	85.6	1.1
V_3	1.223	1.056	100.32	105.6	5.3

Notes:

(a) USE T@30 values taken from WCAP-15916 [Reference 10].
(b) Per NUREG/CR-6413, ORNL/TM-13133, the Cu and Ni values for the correlation monitor material are 0.20 and 0.18, respectively. This equates to a chemistry factor of 100°F from RG 1.99, Revision 2.

Table 13 shows a 2σ uncertainty of less than 50°F, which is the allowable scatter in NUREG/CR-6413, ORNL/TM-13133. Hence, this criterion is met.

<u>Conclusion</u>

Based on the preceding responses to all five criteria of RG 1.99, Rev 2, Section B, the Turkey Point Unit 3 intermediate shell forging and lower shell forging surveillance data is deemed "credible," but the weld data is deemed "not credible." The Turkey Point Unit 4 lower shell forging surveillance data is deemed "credible," but the weld data is deemed "not credible."

4. **whether the ratio procedure of Regulatory Guide 1.99, Rev. 2, Position 2.1 was used.**

   The ratio procedure in RG 1.99, Revision 2, Position 2.1, was used in the chemistry factor (CF) calculations. As described in footnote (e) in Table 7 above, certain ratios were applied for the Turkey Point Units 3 and 4 weld metal and the Davis Besse weld metal. The calculations of these ratios are detailed below.

   Turkey Point Units 3 and 4 Reactor Vessel Beltline Weld Heat # 71249

   Cu Wt. % = 0.23

   Ni Wt. % = 0.59

   $CF_{\text{Beltline Weld}}$ = 167.6 (using Table 1 of RG, Revision 2)

   Turkey Point Units 3 and 4 Surveillance Weld Heat # 71249

   Cu Wt. % = 0.31

   Ni Wt. % = 0.57

   $CF_{\text{Surveillance Weld}}$ = 194.1 (using Table 1 of RG 1.99, Revision 2)

   Thus, the ratio for the Turkey Point Units 3 and 4 surveillance weld is as follows:

   Ratio = $CF_{\text{Beltline Weld}} / CF_{\text{Surveillance Weld}}$ = 167.6°F/194.1°F = **0.863**

   Davis Besse Surveillance Weld Heat # 71249

   Cu Wt. % = 0.33

   Ni Wt. % = 0.57

   $CF_{\text{Surveillance Weld}}$ = 201.3 (using Table 1 of RG 1.99, Revision 2)

   Thus, the ratio for the Davis Besse surveillance weld is as follows:

   Ratio = $CF_{\text{Beltline Weld}} / CF_{\text{Surveillance Weld}}$ = 167.6°F/201.3°F = **0.833**

CVIB-1.5 Reference 2, Section 2.1.4.2.5 concludes that the new EPU environmental conditions (chemistry, temperature, and neutron fluence) will not introduce any new aging effects on the RVI components during 60 years of operation, nor will the EPU change the manner in which the component aging will be managed by the aging management program credited in the topical report WCAP-14577, Rev. 1-A, "License Renewal Evaluation: Aging Management of Reactor Internals," and accepted by the NRC in the Safety Evaluation Report (SER). The susceptibility of the Turkey Point, Units 3 and 4 RVI components to these aging effects was also assessed for license renewal as documented in the License Renewal Application (LRA) for Turkey Point Units 3 and 4 and the associated SER, NUREG-1759.

Although the licensee stated that there will be no new aging effects, Reference 2 does not address whether particular RVI components will become susceptible to additional aging effects due to higher neutron fluences, temperatures, or stresses introduced by the EPU. The staff therefore requests the following information:

a. **Describe the method of determining if additional RVI components become susceptible to the aging effects of 1) cracking due to stress corrosion cracking (SCC), irradiation assisted stress corrosion cracking (IASCC), or primary water stress corrosion cracking (PWSCC); 2) reduction of fracture toughness due to irradiation embrittlement (IE); 3) loss of material due to wear; 4) loss of mechanical closure integrity due to IASCC, IE, irradiation creep, or stress relaxation (SR); and 5) loss of preload due to SR, or dimensional change due to void swelling. The discussion should address whether a detailed fluence and temperature map was used, and whether stresses in individual components were reevaluated.**

a1a: SCC is a synergistic degradation mechanism requiring stress, environment, and a susceptible material. Eliminate any of the required three and SCC will not occur. As identified in License Renewal Application (LRA) Table 3.2-1 all internals components have already been identified as requiring aging management to control SCC. The Turkey Point chemistry controls program maintains rigorous control of reactor coolant chemistry; the increase in temperature or stress due to the EPU therefore will not increase the susceptibility to SCC for the extended license period.

a1b: For IASCC to have a potential to occur both sufficient fluence and stress are required; temperature is not included in current industry standard thresholds for evaluating IASCC. In accordance with WCAP-14577 [Reference 13], $1x10^{21}$ n/cm^2 (E >0.1MeV) and 30 ksi stress are threshold values used to screen in susceptibility to IASCC. For IASCC, the following components were not previously identified in LRA Table 3.2-1 as being susceptible: radial keys and clevis inserts, upper core plate alignment pins, core barrel outlet nozzle diffusers, upper support plates and columns, secondary core support, upper core plate, head/vessel alignment pins, guide tubes and guide pins, internals holddown spring, bottom mounted instrumentation (BMI) columns and upper instrumentation columns. An updated fluence map has shown that fluences exceeding $1x10^{21}$ n/cm^2 (E>0.1MeV) extends from the upper core plate down to 9.5" below the lower core plate. The following table shows that two components not currently identified in LRA Table 3.2-1 as requiring aging management for IASCC (upper core plate and portions of the BMI columns) are within this region. The operating stresses in the BMI columns are well below the threshold for IASCC. WCAP-14577 did not originally identify the upper core plate as a component with a fluence greater than $1x10^{21}$ n/cm^2 (E>0.1MeV). However, the updated fluence calculations indicate that the upper core plate fluence at 60 years will exceed this threshold. The higher fluence results from a combination of the plant uprating and updated calculation methodologies.

	Fluence	Stress (Pm+Pb+Q)
radial keys and clevis inserts	$< 1x10^{21}$ n/cm^2 (E>0.1MeV)	
upper core plate alignment pins	$< 1x10^{21}$ n/cm^2 (E>0.1MeV)	
core barrel outlet nozzles	$< 1x10^{21}$ n/cm^2 (E>0.1MeV)	
diffusers	$< 1x10^{21}$ n/cm^2 (E>0.1MeV)	
upper support plates	$< 1x10^{21}$ n/cm^2 (E>0.1MeV)	
upper support columns	$< 1x10^{21}$ n/cm^2 (E>0.1MeV)	
upper core plate	$> 1x10^{21}$ n/cm^2 (E >0.1MeV)	> 30 ksi
secondary core support	$< 1x10^{21}$ n/cm^2 (E>0.1MeV)	
head/vessel alignment pins	$< 1x10^{21}$ n/cm^2 (E>0.1MeV)	
guide tubes	$< 1x10^{21}$ n/cm^2 (E>0.1MeV)	
guide pins	$< 1x10^{21}$ n/cm^2 (E>0.1MeV)	
internals holddown spring	$< 1x10^{21}$ n/cm^2 (E>0.1MeV)	
BMI columns	$> 1x10^{21}$ n/cm^2 (E>0.1MeV)	< 30 ksi
upper instrumentation columns	$< 1x10^{21}$ n/cm^2 (E>0.1MeV)	
* Guide pins were replaced in 2007 & 2008		
**Upper portions closest to the lower core support plate		

a1c: Similar to SCC, PWSCC is a synergistic degradation mechanism requiring stress, environment, and a susceptible material. For nickel-base materials that are susceptible to PWSCC, all internals components made of nickel-base materials have already been identified in LRA Table 3.2-1 as requiring aging management to control PWSCC. The minimal temperature increase due to the EPU, which is the primary influence on PWSCC, is not expected to increase significantly the susceptibility of nickel-base materials during the extended license period.

a2: For IE to have a potential to occur sufficient fluence is required; stress and temperature do not influence IE. In accordance with WCAP-14577, $1x10^{21}$ n/cm^2 (E>0.1MeV) is the threshold value used to screen in susceptibility to IASCC. For IE the following components were not previously identified in LRA Table 3.2-1 as being susceptible: radial keys and clevis inserts, upper core plate alignment pins, core barrel outlet nozzle diffusers, upper support plates and columns, head/vessel alignment pins, guide tubes and guide pins, internals holddown spring, BMI columns and upper instrumentation columns. An updated fluence map has shown that fluences exceeding $1x10^{21}$ n/cm^2 (E >0.1MeV) extends from the upper core plate down to 9.5" below the lower core plate. As discussed in the response to CVIB-1.5a1b, previous estimates of the upper core plate fluence at 60 years have been below this threshold. The following table shows that two components (upper core plate and portions of the BMI columns) exceed the fluence threshold used in WCAP-14577 to identify components with potential IE concerns.

	Fluence
radial keys and clevis inserts	< $1x10^{21}$ n/cm^2 (E>0.1MeV)
upper core plate alignment pins	< $1x10^{21}$ n/cm^2 (E>0.1MeV)
core barrel outlet nozzles	< $1x10^{21}$ n/cm^2 (E>0.1MeV)
diffusers	< $1x10^{21}$ n/cm^2 (E>0.1MeV)
upper support plates	< $1x10^{21}$ n/cm^2 (E>0.1MeV)
upper support columns	< $1x10^{21}$ n/cm^2 (E>0.1MeV)
upper core plate	> $1x10^{21}$ n/cm^2 (E >0.1MeV)
secondary core support	< $1x10^{21}$ n/cm^2 (E>0.1MeV)
head/vessel alignment pins	< $1x10^{21}$ n/cm^2 (E>0.1MeV)
guide tubes	< $1x10^{21}$ n/cm^2 (E>0.1MeV)
guide pins	< $1x10^{21}$ n/cm^2 (E>0.1MeV)
internals holddown spring	< $1x10^{21}$ n/cm^2 (E>0.1MeV)
BMI columns	> $1x10^{21}$ n/cm^2 (E>0.1MeV)
upper instrumentation columns	< $1x10^{21}$ n/cm^2 (E>0.1MeV)
* Guide pins were replaced in 2007 & 2008	
**Upper portions closest to the lower core support plate	

a3: Loss of material due to wear is a flow dependent phenomenon. Calculations completed for the EPU determined that the EPU will result in a minimal increase in the expected best estimate flow in the reactor coolant system of 0.2%. This was evaluated and it was concluded that this minor increase in flow will not result in any new RVI components being susceptible to the loss of material due to wear during the extended license period.

a4: Loss of mechanical closure integrity, including loss of preload, applies to core support bolting. All RVI bolting has already been identified in LRA Table 3.2-1 as being susceptible to loss of mechanical closure integrity. There are no chemistry changes due to the EPU and changes in stress or temperature are not expected to change how bolting is managed during the period of extended license.

a5: Besides core support bolting the holddown spring would be the only other RVI component susceptible to loss of preload and it is identified as such in LRA Table 3.2-1. Westinghouse evaluated the performance of the holddown spring with respect to the EPU. It was determined that the reactor internals would remain seated and stable for the EPU conditions for the extended license period.

With respect to void swelling, joint industry testing has been conducted since publication of WCAP-14577. Based upon this testing, the industry is currently using $1.3x10^{22}$ n/cm^2 (E>1.0MeV), as published in MRP-175 [Reference 14], as a threshold for void swelling. While some internals components will exceed this value there have been no indications from the different bolt removal programs that there are any discernable effects attributed to swelling. Turkey Point will continue to participate and follow up industry efforts to investigate swelling effects of the core components.

b. **Confirm whether the design projections of gamma heating bound the projected amount of gamma heating of the RVI under EPU conditions. Discuss the acceptability of the effects of gamma heating on the RVI components under EPU conditions.**

Gamma heating rates for the RVI under EPU conditions were explicitly determined and compared with design values as part of the EPU Program. The heating rates calculated at EPU conditions were all less than the design heating rates for the RVI. Thus, there is no impact on the RVI with respect to gamma heating rates under EPU conditions.

c. **Clarify whether any additional RVI components were determined to be susceptible to the aging effects listed in part "a" of this question as a result of EPU, compared to those listed as susceptible to these mechanisms in Table 3.2-1 of the LRA for Turkey Point, Units 3 and 4.**

Compared to components listed as susceptible to the mechanisms of Table 3.2-1 of the LRA, the upper core plate may be susceptible to IASCC and the upper core plate and portions of the BMI columns may be susceptible to IE.

CVIB-1.6 Several aging effects identified for RVI in the LRA for Turkey Point, Units 3 and 4, are not evaluated in the EPU evaluation of RVI materials. The SER related to the Turkey Point, Units 3 and 4 LRA, NUREG-1759, concurred with the aging effects requiring management for the RVI. The staff requests the licensee provide an evaluation of the effects of EPU on the following aging effects requiring management, or explain why the aging effect did not require reevaluation.

a. **LRA Section 3.2.5.2.3 stated that loss of material due to mechanical wear is an aging effect requiring management for the period of extended operation. Loss of material due to wear can occur on the lower core plate fuel pins, core barrel flanges, guide tubes and guide pins, upper core plate alignment pins, and radial keys and clevis inserts.**

Loss of material due to wear is a flow dependent phenomenon. Calculations completed for the EPU determined that the EPU will result in a minimal increase in the best estimate flow in the reactor coolant system of 0.2%. It was concluded that this minor change would have a negligible impact on the wear of the lower core plate fuel pins, core barrel flanges, guide tubes and guide pins, upper core plate alignment pins, and radial keys and clevis inserts.

b. **The LRA indicates loss of mechanical closure integrity due to SCC and SR is an aging effect for upper support column, guide tube, and clevis insert bolting. For baffle-former bolting and barrel-former bolting, loss of mechanical closure integrity can be caused by IASCC, IE, irradiation creep, and irradiation-assisted SR.**

All RVI bolting has already been identified in LRA Table 3.2-1 as being susceptible to loss of mechanical closure integrity. With respect to SCC, there are no chemistry changes due to the EPU and changes in stress or temperature are not expected to change how bolting is managed during the period of extended license. With respect to SR, the minimal changes in temperature and fluence due to the EPU are not expected to change how bolting is managed during the period of extended license.

With respect to baffle-former and barrel-former bolting, these bolts receive the highest internals fluence which is well above known industry thresholds for fluence induced aging mechanisms such as IASCC, IE, irradiation creep, and irradiation-assisted SR (e.g. MRP-175 or WCAP-14577). The minimal increases in temperature and fluence due to the EPU are not expected to change management of such bolting.

c. **The LRA indicates loss of preload due to SR can occur for the RVI hold-down spring.**

Westinghouse evaluated the performance of the holddown spring with respect to the EPU, considering the effects of SR during the extended license period (60 years). It was determined that there will be no significant impact on the loss of preload during the extended license period (60 years) and the reactor internals will remain seated and stable for the EPU conditions.

References

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9. WCAP-15092, Revision 3, "Turkey Point Units 3 and 4 WOG Reactor Vessel 60-Year Evaluation Minigroup Heatup and Cooldown Limit Curves for Normal Operation," T. J. Laubham and J. H. Ledger, May 2000.
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11. WCAP-15885, Revision 0, "R. E. Ginna Heatup and Cooldown Limit Curves for Normal Operation," T. J. Laubham, July 2002.
12. ORNL/TM-13133; NUREG/CR-6413, "Analysis of the Irradiation Data for A302B and A533B Correlation Monitor Materials," J. A. Wang, Oak Ridge National Laboratory, Oak Ridge, TN, April 1996
13. WCAP-14577, Revision 1-A, "License Renewal Evaluation: Aging Management for Reactor Internals," March 2001.
14. MRP-175, Revision 0, "Materials Reliability Program: PWR Internals Aging Degradation Mechanism Screening and Threshold Values," December 2005.

Figure 1: Modified P-T Limits TS Bases

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3/4.4.9 (Cont'd)

During heatup, the thermal gradients through the reactor vessel wall produce thermal stresses which are compressive at the reactor vessel inside surface and which are tensile at the reactor vessel outside surface. Since reactor vessel internal pressure always produces tensile stresses at both the inside and outside surface locations, the total applied stress is greatest at the outside surface location. However, since neutron irradiation damage is larger at the inside surface location when compared to the outside surface, the inside surface flaw may be more limiting. Consequently for the heatup analysis both the inside and outside surface flaw locations must be analyzed for the specific pressure and thermal loadings to determine which is more limiting.

During cooldown, the thermal gradients through the reactor vessel wall produce thermal stresses which are tensile at the reactor vessel inside surface and which are compressive at the reactor vessel outside surface. Since reactor vessel internal pressure always produces tensile stresses at both the inside and outside surface locations, the total applied stress is greatest at the inside surface location. Since the neutron irradiation damage is also greatest at the inside surface location, the inside surface flaw is the limiting location. Consequently, only the inside surface flaw must be evaluated for the cooldown analysis.

The temperature and pressure changes during heatup and cooldown are limited to be consistent with the requirements given in the 1995 Edition through 1996 Addenda of the ASME Boiler and Pressure Vessel Code, Section ~~III~~ XI, Appendix G:

1. The reactor coolant temperature and pressure and system heatup and cooldown rates (with the exception of the pressurizer) shall be limited in accordance with Figures ~~3.4-2 to 3.4-4~~ 3.4-2 and 3.4-3 for the service period specified thereon:

   a. Allowable combinations of pressure and temperature for specific temperature change rates are below and to the right of the limit lines shown. Limit lines for cooldown rates between those presented may be obtained by interpolation; and

   b. Figures ~~3.4-2 to 3.4-4~~ 3.4-2 and 3.4-3 define limits to assure prevention of non-ductile failure only. For normal operation, other inherent plant characteristics, e.g., pump heat addition and pressurizer heater capacity, may limit the heatup and cooldown rates that can be achieved over certain pressure-temperature ranges.

2. These limit lines shall be calculated periodically using methods provided below.

3. The secondary side of the steam generator must not be pressurized above 200 psig if the temperature of the steam generator is below 70°F.

Figure 1: Modified P-T Limits TS Bases (continued)

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

3/4.4.9 (Cont'd)

4. The pressurizer heatup and cooldown rates shall not exceed 100°F/h and 200°F/h, respectively. The spray shall not be used if the temperature difference between the pressurizer and the spray fluid is greater than 320°F, and

5. System preservice hydrotests and inservice leak and hydrotests shall be performed at pressures in accordance with the requirements of ASME Boiler and Pressure Vessel Code, Section XI.

The fracture toughness properties of the ferritic materials in the reactor vessel are determined in accordance with the NRC Standard Review Plan, the version of the ASTM E185 standard required by 10 CFR 50, Appendix H, and in accordance with additional reactor vessel requirements. *

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1995

The properties are then evaluated in accordance with Appendix G of the ~~1983~~ Edition of Section ~~III~~ of the ASME Boiler and Pressure Vessel Code and the additional requirements of 10 CFR 50, Appendix G and the calculation methods described in Westinghouse Report ~~GTSD-A-1.12, Procedure for Developing Heatup and Cooldown Curves~~.

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through 1996 Addenda

Heatup and cooldown limit curves are calculated using the most limiting value of the nil-ductility reference temperature, ~~RTNDT~~, at ~~the end of 19~~ effective full power years (EFPY) of service life. The ~~19~~ EFPY service life period is chosen such that the limiting ~~RTNDT~~, at the 1/4T location in the core region is greater than the ~~RTNDT~~, of the limiting unirradiated material. The selection of such a limiting ~~RTNDT~~ assures that all components in the Reactor Coolant System will be operated conservatively in accordance with applicable Code requirements.

heatup rates of 60 and

RT_{NDT}

3.4-2 and 3.4-3,

The heatup and cooldown limit curves, Figures ~~3.4-2, 3.4-3 and 3.4-4~~ are composite curves prepared by determining the most conservative case with either the inside or outside wall controlling, for ~~any heatup rate up to~~ 100 degrees F per hour and cooldown rates ~~of up to~~ 100 degrees F per hour. The heatup and cooldown curves were prepared based upon the most limiting value of predicted adjusted reference temperature at the end of the applicable service period (~~19 EFPY~~).

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20,40,60, and

The reactor vessel materials have been tested to determine their initial ~~RTNDT~~; the results of these tests are shown in Tables B 3/4.4-1 and B 3/4.4-2. Reactor operation and resultant fast neutron (E greater than 1 MeV) irradiation can cause an increase in the ~~RTNDT~~. Therefore, an adjusted reference temperature, based upon the fluence and chemistry factors of the material has been predicted using Regulatory Guide 1.99, Revision 2, dated May 1988, Radiation Embrittlement of Reactor Vessel Materials. The heatup and cooldown limit curves of Figures ~~3.4-2, 3.4-3, and 3.4-4~~ include predicted adjustments for this shift in ~~RTNDT~~ at the end of the applicable service period.

3.4-2 and 3.4-3

RT_{NDT}

* Topical Report BAW-2308, Revision 2-A is the source for the initial weld materials properties for Linde 80 welds.

Figure 1: Modified P-T Limits TS Bases (continued)

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TECHNICAL SPECIFICATION BASES

Section XI of the 1995 Edition through 1996 Addenda

3/4.4.9 (Cont'd)

Allowable pressure-temperature relationships for various heatup and cooldown rates are calculated using methods derived from Appendix G in ~~Section III~~ of the ASME Boiler and Pressure Vessel Code as required by Appendix G to 10 CFR Part 50 and Westinghouse Report ~~GTSD-A-1.12, Procedure for Developing Heatup and Cooldown Curves.~~

Appendix G of the 1995 Edition through 1996 Addenda of ASME Section XI

The general method for calculating heatup and cooldown limit curves is based upon the principles of the linear elastic fracture mechanics (LEFM) technology. In the calculation procedures a semi-elliptical surface defect with a depth of one-quarter of the wall thickness, T, and a length of 3/2T is assumed to exist at the inside of the vessel wall as well as at the outside of the vessel wall. The dimensions of this postulated crack, referred to in ~~Appendix G of ASME Section III~~ as the reference flaw, amply exceed the current capabilities of inservice inspection techniques. Therefore, the reactor operation limit curves developed for this reference crack are conservative and provide sufficient safety margins for protection against nonductile failure. To assure that the radiation embrittlement effects are accounted for in the calculation of the limit curves, the most limiting value of the nil-ductility reference temperature, ~~RTNDT~~, is used and this includes the radiation-induced shift, ~~ΔRTNDT~~, corresponding to the end of the period for which heatup and cooldown curves are generated.

ΔRT_{NDT} K_I RT_{NDT}

The ASME approach for calculating the allowable limit curves for various heatup and cooldown rates specifies that the total stress intensity factor, ~~KI~~, for the combined thermal and pressure stresses at any time during heatup or cooldown cannot be greater than the reference stress intensity factor, ~~KIR~~, for the metal temperature at that time. ~~KIR~~ is obtained from the reference fracture toughness curve, defined in Appendix G to the ASME Code. The ~~KIR~~ curve is given by the equation:

K_{Ia}

K_{Ia} $\cancel{K_{IR}} = 26.78 + 1.223 \exp[0.0145(T - RT_{NDT} + 160)]$ RT_{NDT} (1)

Where: ~~KIR~~ is the reference stress intensity factor as a function of the metal temperature T and the metal nil-ductility reference temperature ~~RTNDT~~. Thus, the governing equation for the heatup-cooldown analysis is defined in Appendix G of the ASME Code as follows:

$< K_{Ia}$

$C\,K_{IM} + K_{IT} \cancel{\leq K_{IR}}$ (2)

Where: K_{IM} = the stress intensity factor caused by membrane (pressure) stress,

K_{IT} = the stress intensity factor caused by the thermal gradients,

K_{Ia} $\cancel{K_{IR}}$ = constant provided by the Code as a function of temperature relative to the RT_{NDT} of the material,

C = 2.0 for level A and B service limits, and

C = 1.5 for inservice hydrostatic and leak test operations.

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