Designation: E 900 – 87 (Reapproved 1994)

Standard Guide for Predicting Neutron Radiation Damage to Reactor Vessel Materials, E706 (IIF)[1]

This standard is issued under the fixed designation E 900; the number immediately following the designation indicates the year of original adoption or, in the case of revision, the year of last revision. A number in parentheses indicates the year of last reapproval. A superscript epsilon (ϵ) indicates an editorial change since the last revision or reapproval.

1. Scope

1.1 This guide presents a method for predicting reference transition temperature adjustments for irradiated light-water cooled power reactor pressure vessel materials based on Charpy V-notch 41-J (30-ft·lbf) data. Radiation damage calculative procedures have been developed from a statistical analysis of an irradiated material data base that was available as of June 1982, and checked against readily available data up to August 1983. In the procedure, a chemistry factor given in tabular form as a function of copper and nickel contents, is multiplied by a fluence factor read from a graph or calculated from a formula. A difference between this guide and the earlier edition is the addition of nickel content in the chemistry factor. This guide is applicable for the following specific materials, range of irradiation temperature, neutron fluence, and fluence rate:

1.1.1 *Materials*:

1.1.1.1 A 533 Type B Class 1 and 2, A302 Grade B, A302 Grade B (modified), A508 Class 2 and 3.

1.1.1.2 Submerged arc welds, shielded arc welds, and electroslag welds for materials in 1.1.1.1.

1.1.1.3 Weld heat-affected zones of the materials in 1.1.1.1 and 1.1.1.2.

1.1.2 Copper contents within the range from 0.01 to 0.40 weight %.

1.1.3 Nickel content within the range from 0 to 1.2 weight %.

1.1.4 Irradiation exposure temperature within the range from 530 to 590°F (277 to 310°C).

1.1.5 Neutron fluence within the range from 1 by 10^{17} to 1 by 10^{20} n/cm^2(E > 1 MeV).

1.1.6 Neutron fluence rate and energy spectra within the range expected at the reactor vessel core beltline region of light-water cooled reactors.

1.2 The basis for the method of adjusting the reference temperature is a report describing the basis for Regulatory Guide 1.99.[2] The report is based on the reactor vessel surveillance data and analyses described by Guthrie[3] and Odetle and Lombrozo[4]; the extent of that data base is indicated by the dashed lines in Tables 1 and 2.

1.3 This guide is Part IIF of Master Matrix E 706 which coordinates several standards used for irradiation surveillance of light-water reactor vessel materials. Methods of determining the applicable fluence for use in this guide are addressed in Master Matrix E 706, Practices E 560 (IC) and E944 (IIA), and Method E 1005 (IIIA). The overall application of these separate guides and practices is described in Practice E 853 (IA).

1.4 The values given in inch-pound units are to be regarded as the standard. The values given in parentheses are for information only.

1.5 *This standard does not purport to address all of the safety concerns, if any, associated with its use. It is the responsibility of the user of this standard to establish appropriate safety and health practices and determine the applicability of regulatory limitations prior to use.*

2. Referenced Documents

2.1 *ASTM Standards:*

E 185 Practice for Conducting Surveillance Tests for Light-Water Cooled Nuclear Power Reactor Vessels, E706 (IF)[5]

E 560 Practice for Extrapolating Reactor Vessel Surveillance Dosimetry Results, E706 (IC)[5]

E 693 Practice for Characterizing Neutron Exposures in Iron and Low-Alloy Steels in Terms of Displacements per Atom (DPA), E706 (ID)[5]

E 706 Master Matrix for Light-Water Reactor Pressure Vessel Surveillance Standards[5]

E 853 Practice for Analysis and Interpretation of Light-Water Reactor Surveillance Results, E706 (IA)[5]

[1] This guide is under the jurisdiction of ASTM Committee E-10 on Nuclear Technology and Applicationsand is the direct responsibility of Subcommittee E10.02on Nuclear Materials, Components, and Environmental Effects.

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[2] Randall, P. N., "Basis for Revision 2 of U.S. NRC Regulatory Guide 1.99," *Radiation Embrittlement of Nuclear Reactor Pressure Vessel Steels*, ASTM STP 909, 1986, pp. 149–162.

[3] Guthrie, G. L., "Charpy Trend Curves Based on 177 Data Points," LWR Pressure Vessel Surveillance Dosimetry Improvement Program, Quarterly Progress Report April 1983 through June 1983, Hanford Engineering Development Laboratory, NUREG/CR-3391, Vol 2, HEDL-TME 83-22.

[4] Odette, G. R., and Lombrozo, P. M., "Physical Based Regression Correlations of Embrittlement Data From Reactor Pressure Vessel Surveillance Programs," EPRI NP-3319 Final Report, January 1984, Prepared for Electric Power Research Institute.

[5] *Annual Book of ASTM Standards*, Vol 12.02.

TABLE 1 Chemistry Factor for Welds, °F[A]

Copper, Weight, %	Nickel, Weight, %						
	0	0.20	0.40	0.60	0.80	1.00	1.20
0	20	20	20	20	20	20	20
0.01	20	20	20	20	20	20	20
0.02	21	26	27	27	27	27	27
0.03	22	35	41	41	41	41	41
0.04	24	43	54	54	54	54	54
0.05	26	49	67	68	68	68	68
0.06	29	52	77	82	82	82	82
0.07	32	55	85	95	95	95	95
0.08	36	58	90	106	108	108	108
0.09	40	61	94	115	122	122	122
0.10	44	65	97	122	133	135	135
0.11	49	68	101	130	144	148	148
0.12	52	72	103	135	153	161	161
0.13	58	76	106	139	162	172	176
0.14	61	79	109	142	168	182	188
0.15	66	84	112	146	175	191	200
0.16	70	88	115	149	178	199	211
0.17	75	92	119	151	184	207	221
0.18	79	95	122	154	187	214	230
0.19	83	100	126	157	191	220	238
0.20	88	104	129	160	194	223	245
0.21	92	108	133	164	197	229	252
0.22	97	112	137	167	200	232	257
0.23	101	117	140	169	203	236	263
0.24	105	121	144	173	206	239	268
0.25	110	126	148	176	209	243	272
0.26	113	130	151	180	212	246	276
0.27	119	134	155	184	216	249	280
0.28	122	138	160	187	218	251	284
0.29	128	142	164	191	222	254	287
0.30	131	146	167	194	225	257	290
0.31	136	151	172	198	228	260	293
0.32	140	155	175	202	231	263	296
0.33	144	160	180	205	234	266	299
0.34	149	164	184	209	238	269	302
0.35	153	168	187	212	241	272	305
0.36	158	172	191	216	245	275	308
0.37	162	177	196	220	248	278	311
0.38	166	182	200	223	250	281	314
0.39	171	185	203	227	254	285	317
0.40	175	189	207	231	257	288	320

[A] t°C = t°F/1.8.

E 944 Guide for Application of Neutron Spectrum Adjustment Methods in Reactor Surveillance (IIA)[5]

E 1005 Test Method for Application and Analysis of Radiometric Monitors for Reactor Vessel Surveillance, E706 (IIIA)[5]

3. Significance and Use

3.1 Operation of commercial power reactors must conform to pressure-temperature limits during heatup and cooldown to prevent over-pressurization at temperatures that might cause nonductile behavior in the presence of a flaw. Radiation damage to the reactor vessel beltline region is compensated for by adjusting the pressure-temperature limits to higher temperature as the neutron damage accumulates. The present practice is to base that adjustment on the increase in transition temperature produced by neutron irradiation as measured at the Charpy V-notch 41-J (30-ft·lbf) energy level. To establish pressure-temperature operating limits during the operating life of the plant, a prediction of adjustment in transition temperature must be made.

3.1.1 In the absence of surveillance data for a given reactor (see Practice E 185), the use of calculative procedures will be necessary to make the prediction. Even when credible surveillance data are available, it will usually be necessary to extrapolate the data to obtain an adjustment in transition temperature for a specific time in the plant operating life. The fluence function presented herein has been developed for those purposes.

3.2 Research has established that certain elements, notably copper and nickel, cause a variation in radiation sensitivity of steels. The importance of other suspect elements remains a subject of additional research. Copper and nickel are the parameters used in developing the radiation damaged calculative procedures.

3.3 Only power reactor surveillance data were used in the derivation of these procedures. The measure of fluence used in the procedure is n/cm^2 (E > 1 MeV). Differences in the neutron fluence rate and in neutron energy spectra experienced in power reactors and test reactors have not been considered in the development of these procedures because the technology is