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Standard Guide for Predicting ~~Neutron Radiation Damage to~~ Radiation-Induced Transition Temperature Shift in Reactor Vessel Materials, E706 (IIF)[1]

[1] This guide is under the jurisdiction of ASTM Committee E10 on Nuclear Technology and Applications and is the direct responsibility of Subcommittee E10.02 on ~~Nuclear Materials, Components,~~ Behavior and ~~Environmental Effects.~~ Use of Nuclear Structural Materials.

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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)~~ 30-ft·lbf (41-J) data. Radiation damage calculative procedures have been developed from a statistical analysis of an irradiated material database that was available as of ~~June 1982, and checked against readily available data up to August 1983. In the procedure, a chemistry factor given~~ May 2000.[2] The embrittlement correlation used in ~~tabular form as a function of~~ this guide was developed using the following variables: copper and nickel contents, ~~is multiplied by a fluence factor read from a graph or calculated from a formula. A difference between this guide~~ irradiation temperature, and ~~the earlier edition is the addition~~ neutron fluence. The form of ~~nickel content in~~ the model was based on current understanding for two mechanisms of embrittlement: stable matr~~y~~ fix damage (SMD) and copper-rich precipitation (CRP); saturation of copper effects (for different weld materials) was included. This guide is applicable for the following specific materials, copper, nickel, and phosphorus contents, range of irradiation temperature, ~~neutron fluence,~~ and neutron fluence based on the overall database:

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~~ 0 to ~~0.40 weight %.~~ 0.50 wt %.

1.1.3 Nickel content within the range from 0 to 1.~~2~~3 wt %.

1.1.4 Phosphorus content wi~~g~~thin the range 0 to 0.025 wt %.

1.1.~~4~~5 Irradiation exposure temperature within the range from ~~530~~ 500 to ~~590°F (277~~ 570°F (260 to ~~310~~ 299°C).

1.1.~~5~~6 Neutron fluence within the range from 1 ~~by~~ $\times$ $10^{\sout{17}16}$ ~~to 1 by~~ to 8 $\times$ $10^{\sout{20}19}$ n/cm^2 (E > 1 MeV).

~~1.1.6 Neutron fluence rate and~~

1.1.7 Neutron energy spectra within the range expected at the reactor vessel core beltline region of light~~-~~ water cooled reactors and fluence rate within the range from 2 $\times$ 10^8 to 1 $\times$ 10^{12} n/cm^{2}s (E > 1 MeV).

1.2 The basis for the method of adjusting the reference temperature is discussed in 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~~ separate report.[4] ~~and Odetle and Lombrozo[5]; 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 Guide E 944 (IIA), and Test 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~~ customary U.S. units are to be regarded as the standard. The SI values given in parentheses are for information only.

1.5 This standard guide does not define how the shift in transition temperature should be used to determine the final adjusted reference temperature. (That would typically include consideration of the initial starting point, the predicted shift, and the uncertainty in the shift estimation method.)

~~[2] Randall, P. N., "Basis for Revision 2 of U.S. NRC Regulatory Guide 1.99," *Radiation Embrittlement of Nuclear Reactor Pressure Vessel Steels,*~~

[2] The Charpy surveillance data were originally obtained from the Oak Ridge National Laboratory Power Reactor-Embrittlement Database (PR-EDB) and subsequently updated by ASTM ~~STP 909, 1986, pp. 149–162.~~ Subcommittee E10.02, May 2000.

1.6 *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)[4]

E 693 Practice for Characterizing Neutron Exposures in Iron and Low-Alloy Steels in Terms of Displacements per Atom (DPA), E706 (ID)[4]

E 706 Master Matrix for Light-Water Reactor Pressure Vessel Surveillance Standards[4]

E 853 Practice for Analysis and Interpretation of Light-Water Reactor Surveillance Results, E706 (IA)[4]

E 944 Guide for Application of Neutron Spectrum Adjustment Methods in Reactor Surveillance (IIA)[4]

E 1005 Test Method for Application and Analysis of Radiometric Monitors for Reactor Vessel Surveillance, E706 (IIIA)[4]

3. Terminology

3.1 *Definitions of Terms Specific to This Standard:*

3.1.1 *A, B*—material fitting coefficients that are a function of material type.

3.1.2 *best-estimate chemical composition*—the best-estimate chemical composition (copper [Cu] and nickel [Ni], in wt %) may be established using one of the following methods: (*1*) Use a simple mean for a small set of uniformly distributed data; that is, sum the measurements and divide by the number of measurements; (*2*) Use a weighting process for a non-uniformly distributed data set, especially when the number of measurements from one source are much greater in terms of material volume analyzed. For a plate, a unique sample could be a set of test specimens taken from one corner of the plate. For a weldment, a unique sample would be a set of test specimens taken from a unique weld deposit made with a specific electrode heat. A simple mean is calculated for test specimens comprising each unique sample, the sample means are then summed, and the sum is divided by the number of unique samples to get the sample weighted mean; (*3*) Use an alternative weighting scheme when other factors have a significant influence and a physical model can be established. For the preceding, the best estimate for the sample should be used if evaluating surveillance data from that sample.

3.1.2.1 *Discussion*—For cases where no chemical analysis measurements are available for a heat of material, the upper limiting values given in the material specifications to which the vessel was built may be used. Alternately, generic mean values for the class of material may be used.

3.1.2.2 *Discussion*—In all cases where engineering judgment was used to select a best estimate copper or nickel content, the rationale shall be documented which formed the basis for the selection.

3.1.3 *CRP*—the copper rich precipitate term of the transition temperature shift equation and is based on the knowledge of copper-enriched clustering that occurs in RPV steels.

3.1.4 *F(Cu)*—a copper term in the transition temperature shift equation that is a function of the measured copper content and material, and is subject to a saturation level at a high copper content.

3.1.5 *fluence* (Φ)—the number of neutrons per square centimeter with energy greater than 1.0 MeV at the location of interest.

3.1.6 *G* (Φ)—a fluence function term in the transition temperature shift equation.

3.1.7 *SMD*—the stable matrix damage term of the transition temperature shift equation and is based on an assumed understanding of matrix damage mechanisms in RPV steels.

3.1.8 T_c—irradiation temperature at full power, in °F, and is the estimated time-weighted average (based on the mean temperature over each fuel cycle) cold leg temperature for PWRs and recirculation temperature for BWRs.

3.1.9 *TTS*—the predicted mean value of the transition temperature shift from the correlation.

4. Significance and Use

~~3~~4.1 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)~~ 30-ft·lbf (41-J) 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.

~~[4] Guthrie, G. L., "Charpy Trend Curves Based on 177 Data Points," LWR~~

[4] *Charpy Embrittlement Correlations—Status of Combined Mechanistic and Statistical Bases for U.S. 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.*~~ *Steels (MRP-45), PWR Materials Reliability Program (PWRMRP)*, EPRI, Palo Alto, CA, 2001, 1000705.

~~[5] Odette, G. R., and Lombrozo, P. M., "Physical Based Regression Correlations~~

[5] *Annual Book of* ~~*Embrittlement Data From Reactor Pressure Vessel Surveillance Programs," EPRI NP-3319 Final Report, January 1984, Prepared for Electric Power Research Institute.*~~ *ASTM Standards*, Vol 12.02.

~~3~~4.1.1 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~~ embrittlement correlation presented herein has been developed for those purposes.

~~3~~4.2 Research has established that certain elements, notably copper and nickel, cause a variation in radiation sensitivity of steels. The importance of other ~~suspect elements~~ elements, such as phosphorus (P), remains a subject of additional research. Copper and nickel are the key chemistry parameters used in developing the ~~radiation damaged~~ calculative procedures described here.

~~3~~4.3 Only power reactor surveillance data were used in the derivation of these procedures. The measure of fast neutron 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~~ applied in ~~the development of~~ these ~~procedures because the technology is not yet available to provide quantitative~~ procedures. The manner in which these factors ~~for calculation.~~

~~4.~~ were considered is addressed elsewhere.[3]

5. Calculative Procedure~~s~~ for Transition Temperature ~~Adjustment~~

~~4.1 This~~ Shift

5.1 This guide presents a calculative procedure~~s~~ for estimating the ~~"shift," the~~ transition temperature ~~elevation~~ shift caused by neutron ~~radiation, in three parts.~~ radiation. The ~~first part is to calculate the mean or best estimate value~~ form of ~~shift at~~ the ~~vessel inner surface as the product of a chemistry factor and a fluence factor.~~ correlation involves two major embrittlement terms. The ~~second part is to calculate the reduction~~ form of ~~shift due to~~ the ~~attenuation of neutron energy as a function of depth in the vessel wall. The third part~~ terms is ~~to choose the margin to be added to the best estimate value of shift at depth to cover uncertainties in the calculative procedures as well as uncertainties in the values of copper, nickel,~~ mechanistically guided, and ~~fluence that were entered in~~ the ~~formulas. Each part of~~ terms represent the ~~calculation (best estimate shift, attenuation,~~ hardening contribution from small microstructural defects and ~~margin) shall be separately identified in the report.~~

~~4.1.1~~ clusters created during irradiation.

5.1.1 *Mean Transition Temperature Shift ~~at Vessel Inner Surface~~*:

~~4~~5.1.1.1 The mean value of ~~shift, at the vessel inside wall surface, ΔCv 30M Surface (°F), should be~~ TTS, in °F, is calculated as follows:

$$\cancel{\Delta C_{vTTS = SMD + CRP}} \quad (1)$$

$$TTS = SMD + CRP \quad (1)$$

where

$$SMD = A \exp[20730/(T_c + 460)](\Phi)^{0.5076} \quad (2)$$

$$CRP = B[1 + 2.106 Ni^{1.173}]F(Cu)G(\Phi) \quad (3)$$

and

$$A = 6.70 \times 10^{-18}$$

$$\cancel{B = \begin{pmatrix} 234, \text{ welds; } 30_m \\ 208, \text{ Combustion Engineering plates;} \\ 156, \text{ other plates} \end{pmatrix}}$$

$$B = \begin{pmatrix} 234, \text{ welds;} \\ 128, \text{ forgings;} \\ 208, \text{ Combustion Engineering plates;} \\ 156, \text{ other plates} \end{pmatrix}$$

$$\cancel{G(\Phi) = \frac{1}{2} + \frac{1}{2} \text{Surface} = [CF] f^{(0.28 - 0.10 \log f)}}$$

$$G(\Phi) = \frac{1}{2} + \frac{1}{2} \tanh\left[\frac{\log(\Phi) - 18.24}{1.052}\right]$$

$$F(Cu) = \begin{pmatrix} 0, Cu \leq 0.072 \text{ wt \%} \\ (Cu - 0.072)^{0.577}, Cu > 0.072 \text{ wt \%} \end{pmatrix}$$

~~4~~subject to

$$\cancel{Cu_{max} = (0.1\ .305 \text{ wt \%, for other welds})}$$

$$Cu_{max} = \begin{pmatrix} 0.25 \text{ wt \%, for welds with Linde 80 or Linde 0091 flux;} \\ 0.305 \text{ wt \%, for other welds} \end{pmatrix}$$

6. Attenuation Through the Vessel Wall

6.1 The effective fluence should be used to calculate the transition temperature shift for all locations within the vessel wall, rather than the actual fluence (E > 1 MeV) for those locations.

6.2 To calculate the shift at some location within the vessel wall away from the inside surface, it is necessary to account for the change in neutron spectrum. Due to these changes in neutron energy spectrum, the use of neutron fluence (E > 1 MeV) may give a non-conservative estimate of the neutron damage attenuation within the vessel wall. The preferred exposure parameter for accommodating this change is displacements per atom (*dpa*). Since ~~*CF*,~~ *dpa* can be calculated through the vessel wall thickness following Practice E 693 during normal surveillance program evaluations, each plant could have the calculated *dpa* as a function of depth into the vessel wall. The calculated *dpa* can be used to obtain the effective vessel wall fluence for use with the embrittlement correlation in this guide as shown below:

$$\cancel{(\Phi)_x = (\Phi)_{IS}\,[dppa_{IS}]} \quad (4)$$

$$(\Phi)_x = (\Phi)_{IS}\,[dpa_x/dpa_{IS}] \quad (4)$$

where $(\Phi)_x$ and dpa_x are the effective vessel wall fluence (units of n/c~~k~~m^2 for E > 1.0 MeV) and calculated *dpa*, respectively, at any distance *x* (in inch units) into the vessel wall from the inside surface (*IS*); and $(\Phi)_{IS}$ and dpa_{IS} are the calculated fluence and *dpa*, respectively, at the inside surface of the pressure vessel wall.

6.3 Alternately, the following exponential attenuation formula may be used:[6]

$$(\Phi)_x = (\Phi)_{IS}\exp(-0.24\,x) \quad (5)$$

where $(\Phi)_x$ is ~~given in Table 1~~ the effective vessel wall fluence (units of n/cm^2 for ~~w~~ E > 1.0 MeV) at any distance *x* (in inch units) into the vessel wall from the inside surface (*IS*); and ~~Table 2~~ $(\Phi)_{IS}$ is the calculated fluence at the inside surface of the pressure vessel wall. A recent review has confirmed that this formula provides a reasonable estimation of through wall attenuation.[7]

6.4 Other forms of attenuation of embrittlement through the wall of the reactor vessel can be used if they can be technically justified.

7. Evaluation of Uncertainty

7.1 The following guidance on uncertainty is provided for ~~b~~ use in applying the predicted transition temperature shift to determine an adjusted reference temperature.

7.2 The procedure outlined in 5.1.1.1 provides an estimate of the irradiated Charpy transition temperature shift based on analysis of reactor pressure vessel surveillance data. When this procedure is applied to the original database, the standard error of the correlation is 22.0°F (12.2°C). Principal contributors to this error include the uncertainty in the input parameters (Cu, Ni, fluence and ~~f~~ irradiation temper~~g~~ature uncertain~~g~~ties). The remainder of the error is attributed to uncertainties in the Charpy shift determinations and model errors.

7.3 The standard error describes how well the model describes the Charpy transition temperature shift. A significant portion of this error may be attributed to the actual shift measurement. The statistical regression averages the material response over the entire database. In this case, the model will describe mean material behavior more accurately than it describes an individual measurement of shift. The 22.0°F (12.2°C) standard error of the original analysis provides an upper limit on the error for the mean behavior of an individual surveillance material.

7.4 In the application of the model, the uncertainty in the model will depend on the uncertainty in the input parameters. If the uncertainties in the input parameters for the specific application are similar to those in the surveillance capsule database, it is conservative to assume that the standard error of the original analysis applies to the application. Adjustments may be required if the uncertainties in the input data significantly exceed the uncertainties in the surveillance capsule database.

7.5 Although phosphorus has been identified as a ~~fluence (E > 1 MeV)~~ potential embrittling agent in reactor pressure vessel steels, it has not been included in this procedure. Independent analysis has demonstrated that the *P* effect cannot be unambiguously identified in the surveillance capsule database. Phosphorus levels in the surveillance capsule database ranged from 0.003 to 0.024 wt % with a mean of 0.012 wt % and a standard deviation of 0.004 wt %; there are no apparent material type differences even between welds and base metals. Within the range of the database, additional adjustments for phosphorus uncertainty generally are not required; a simple uncertainty analysis for the effect of ~~1 × 10^{19}~~ *P* revealed an estimated one sigma uncertainty of 5°F (2.8°C), which when appropriately combined with other uncertainties makes a difference of 1 to 2°F (0.6 to 1.1°C) in the global uncertainty.

7.6 The neutron fluence rate is not included in the model. Although the surveillance capsule database includes neutron fluence rates ranging from 2 × 10^8 n/cm^2. ~~Linear interpolation is permitted between given values. In Tables 1 and 2, "percent copper" and "percent nickel" are the best-estimate values for the material, which will normally be the mean of the measured values for a plate or forging or for weld samples made with the weld wire heat number that matches the weld being evaluated. In the case where measured values are not available, the least restrictive of the following approaches may be used: (*a*) the upper limiting values of copper and nickel given in the material specifications~~s to ~~which the vessel was built; (*b*) conservative estimates (mean plus one standard deviation) based on generic data, where justification must be provided; or (*c*) assume 0.35 % copper and 1.0 % nickel.~~

~~4.1.1.3 The fluence, "*f*," is the calculated value of the neutron fluence at the inner surface of the vessel at the location of the postulated defect,~~ 1 × 10^{12} n/cm^2~~(E > 1 MeV) divided by 10^{19}~~.

~~4.1.1.4 The fluence factor, $f^{0.28-0.10\log f}$, is determined by calculation or from Fig. 1.~~

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[6] ~~*Annual Book*~~*Radiation Embrittlement* *of* ~~*ASTM Standards*~~ *Reactor Vessel Materials*, ~~Vol 12.02.~~ Regulatory Guide 1.99, Revision 2, U.S. Nuclear Regulatory Commission, Washington, D.C., May 1988.

[7] *Materials Reliability Program: Attenuation in U.S. RPV Steels (MRP-56)*, EPRI, Palo Alto, CA: 2002. 1006584.

~~4.1.2 *Attenuation Through Vessel Wall*:~~

~~4.1.2.1 To calculate shift at some location within~~-s, the ~~vessel wall, it is necessary to account for the attenuation~~ preponderance of ~~neutron fluence through~~ the ~~wall. Traditionally, the basis for this calculation has been the reduction in number of neutrons of energy greater than 1 MeV. However, more recent analysis indicates that this may not be conservative, because other changes in the neutron energy spectrum with depth should also be considered. Some have recommended the use of displacements per atom (dpa) for this purpose (see Practice E 693). It is the responsibility of the user to ensure that the attenuation modeling used reflects the state of the art.~~

~~4.1.3 *Data Scatter*:~~

~~4.1.3.1 Data scatter about the mean was measured by calculating the residual, which is the measured value of shift for each line of~~ data ~~minus the value predicted by Eq 1. Fig.~~ lies between 3×10^9 n/cm^2-s and 2 ~~is a plot of~~ $\times 10^{11}$ n/cm^2-s. Within the ~~residual versus the calculated value~~ limitations of ~~shift for~~ the ~~data base analyzed.~~[3]

~~4.1.3.2 The measure of scatter[4] was~~ surveillance capsule database and a ~~value~~ thorough review of ~~one standard deviation (σ) of 22°F (12°C), for both welds and base metal. For the data base analyzed in Footnote 3, the values of one standard deviation were 28°F (16°C) for welds and 17°F (9°C) for base metal. To use any of these values in predicting an upper bound or design value, the user should be aware that the scatter is not uniform over the range of~~ mechanistic understanding, a neutron fluence ~~values, copper contents, and nickel contents. In general, the value of the standard deviation is most applicable~~ rate effect could not be unambiguously identified. No additional adjustments to ~~the fluence region within which the data~~ uncertainty are ~~concentrated.~~

~~4.1.4 *Application*:~~

~~4.1.4.1 When surveillance data~~ required for ~~the~~ typical reactor ~~in question are available, 4.1.1 may be used to supplement that data and improve its application to the reactor in question. This normally would be done by combining the~~ pressure vessel beltline neutron fluence ~~factor from the equation in 4.1.1 with the best fit value for the chemistry factor in that equation, obtained from the~~ rates.

7.7 If future surveillance data ~~by a least squares fitting procedure, for example. The result can be used to calculate shift at fluence values of interest in the reactor vessel, provided the surveillance material really represents the controlling material in the vessel. The surveillance data can be considered credible when:~~

~~*(a)* (*a*) The change in the yield strength is consistent with the shift in the Charpy curve,~~

~~*(b)* (*b*) The shift for the correlation monitor material in the capsule falls within the scatter band of the data for~~ provide unambiguous evidence that ~~material, and~~

~~*(c)* (*c*) The shift for~~ the ~~surveillance material(s) is consistent with the normal trends~~ effects of ~~similar materials and with previous surveillance data from the same reactor.~~

~~4.1.4.2 The amount of adjustment (margin) to be applied to the mean shift prediction curve is to be justified by the user. It~~ additional mechanistic phenomenon should ~~reflect the following:~~

~~*(a)* (*a*) Uncertainty in the chemistry factor already applied (see 4.1.1) and the neutron fluence,~~

~~*(b)* (*b*) The neutron fluence range and shift magnitude (see 4.1.3), and~~

~~*(c)* (*c*) The uncertainty in the surveillance data~~ be considered (for example, phosphorus, fluence ratte, etc.) a revision of the ~~Charpy data, consistency with normal trends for the surveillance and correlation monitor materials, and consistency between yield strength changes and shift~~ calculative procedure presented in ~~the Charpy curve).~~

~~4.1.4.3 If no surveillance data are available, prediction shall~~ Section 5 may be ~~based upon the mean curve with an upward adjustment for scatter. The margin to be applied is for the user to justify (see 4.1.4.2). In the example shown~~ warranted. Alternately, additional adjustments in ~~Fig. 2,~~[2] the ~~suggested margin is twice the standard deviation except at low values of calculated shift where the margin added need not exceed 100 % of the calculated value.~~ prediction uncertainty may be appropriate.