Designation: E 900 – 02

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

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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 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 May 2000.[2] The embrittlement correlation used in this guide was developed using the following variables: copper and nickel contents, irradiation temperature, and neutron fluence. The form of the model was based on current understanding for two mechanisms of embrittlement: stable matrix 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, 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.

1.1.2 Copper contents within the range from 0 to 0.50 wt %.

1.1.3 Nickel content within the range from 0 to 1.3 wt %.

1.1.4 Phosphorus content within the range 0 to 0.025 wt %.

1.1.5 Irradiation exposure temperature within the range from 500 to 570°F (260 to 299°C).

1.1.6 Neutron fluence within the range from 1×10^{16} to 8×10^{19} n/cm^2 (E > 1 MeV).

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 2×10^{8} to 1×10^{12} n/cm^{2}s (E > 1 MeV).

1.2 The basis for the method of adjusting the reference temperature is discussed in a separate report.[3]

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 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.)

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)[4]

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]

[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 Behavior and Use of Nuclear Structural Materials.

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[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 Subcommittee E10.02, May 2000.

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

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

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

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 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.

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

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

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 neutron energy spectra experienced in power reactors and test reactors have not been applied in these procedures. The manner in which these factors were considered is addressed elsewhere.[3]

5. Calculative Procedure for Transition Temperature Shift

5.1 This guide presents a calculative procedure for estimating the transition temperature shift caused by neutron radiation. The form of the correlation involves two major embrittlement terms. The form of the terms is mechanistically guided, and the terms represent the hardening contribution from small microstructural defects and clusters created during irradiation.

5.1.1 *Mean Transition Temperature Shift*:

5.1.1.1 The mean value of TTS, in °F, is calculated as follows:

$$TTS = SMD + CRP \tag{1}$$

where

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

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

and

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