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

This standard is issued under the fixed designation E900; 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 values of reference transition temperature ~~adjustments~~ shift (*TTS*) for irradiated ~~light-water cooled power reactor pressure vessel materials based on~~ pressure vessel materials. The method is based on the *TTS*~~Charpy V-notch 30-ft·lbf (41-J) data. Radiation damage calculative procedures have~~ exhibited by Charpy V-notch data at 41-J (30-ft·lbf) obtained from surveillance programs conducted in several countries for commercial pressurized (PWR) and boiling (BWR) light-water cooled (LWR) power reactors. An embrittlement correlation has been developed from a statistical analysis of ~~an irradiated material database that was available as of May~~the large surveillance database consisting of radiation-induced ~~*2000.*~~*TTS* and related information compiled and analyzed by Subcommittee E10.02. The details of the database and analysis are described in a separate report **(1)**.[2,3] ~~The~~This embrittlement correlation ~~used in this guide~~ was developed using the ~~following variables: copper and nickel contents,~~ variables copper, nickel, phosphorus, manganese, irradiation temperature, ~~and neutron fluence. The form of the model was based on current understanding for two mechanisms of embrittlement: stable~~neutron fluence, and product form. Data ranges and conditions for these variables are listed in 1.1.1~~matrix damage (SMD)~~. Section 1.1.2 ~~and copper-rich precipitation (CRP); saturation of copper effects (for different weld~~lists the materials included in the database and the domains of exposure variables that may influence ~~*materials)*~~*TTS* ~~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:~~but are not used in the embrittlement correlation.

1.1.1 *The range of material and irradiation conditions in the database for variables used in the embrittlement correlation:*

1.1.1.1 Copper content up to 0.4 %.

1.1.1.2 Nickel content up to 1.7 %.

1.1.1.3 Phosphorus content up to 0.03 %.

1.1.1.4 Manganese content within the range from 0.55 to 2.0 %.

1.1.1.5 Irradiation temperature within the range from 255 to 300°C (491 to 572°F).

1.1.1.6 Neutron fluence within the range from 1×10^{21} n/m² to 2×10^{24} n/m² (E> 1 MeV).

1.1.1.7 A categorical variable describing the product form (that is, weld, plate, forging).

1.1.2 ~~*Materials:*~~ *The range of material and irradiation conditions in the database for variables not included in the embrittlement correlation:*

1.1.2.1 A533 Type B Class 1 and 2, ~~A302~~A302 Grade B, ~~A302~~A302 Grade B (modified), ~~A508~~and A508 Class 2 and 3. Also, European and Japanese steel grades that are equivalent to these ASTM Grades.

1.1.2.2 Submerged arc welds, shielded arc welds, and electroslag welds ~~for materials~~ having compositions consistent with those of the welds used to join the base materials described in ~~1.1.1.1~~1.1.2.1.

1.1.2.3 Neutron fluence rate within the range from 3×10^{12} n/m²/s to 5×10^{16} n/m²/s (E > 1 MeV).

[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.~~boldface numbers in parentheses refer to a list of references at the end of this standard.

[3] To inform the ~~*Charpy*~~*TTS* ~~Embrittlement Correlations—Status of Combined~~prediction of Section 5 ~~Mechanistic and Statistical Bases for U.S. Pressure Vessel Steels (MRP-45), PWR Materials Reliability Program (PWRMRP),~~of this guide, the E10.02 Subcommittee decided to limit the data considered to Charpy shift ~~EPRI, Palo~~values (ΔT_{41J} ~~Alto, CA, 2001, 1000705.~~) measured from irradiations conducted in PWRs and BWRs. A database of 1,878 Charpy *TTS* measurements was compiled from surveillance reports on operating and decommissioned light water reactors of Western design from 13 countries (Brazil, Belgium, France, Germany, Italy, Japan, Mexico, The Netherlands, South Korea, Sweden, Switzerland, Taiwan, and the United States), and from the technical literature. For each data record, the following information had to be available: fluence, fluence rate, irradiation temperature, and % content of Cu, Ni, P, and Mn. Reports and technical papers documenting the results of research programs conducted in material test reactors were also reviewed. Data from these sources was included in the database for information, but was not used in the development of the *TTS* prediction of Section 5 of this guide.

1.1.2.4 Neutron energy spectra within the range expected at the reactor vessel region adjacent to the core of commercial PWRs and BWRs (greater than approximately 500MW electric).

1.1.2.5 Irradiation exposure times of up to 25 years in boiling water reactors and 31 years in pressurized water reactors.

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 It is the responsibility of the user to show that the conditions of interest in their application of this guide are addressed adequately by the technical information on which the guide is based. It should be noted that the conditions quantified by the database are not distributed evenly over the range of materials and irradiation conditions described in 1.1, and that some combination of variables, particularly at the extremes of the data range are under-represented. Particular attention is warranted when the guide is applied to conditions near the extremes of the data range used to develop the *TTS* equation and when the application involves a region of the data space where data is sparse. Although the embrittlement correlation developed for this guide was based on statistical analysis of a large database, prudence is required for applications that involve variable values beyond the ranges specified in 1.1. Due to strong correlations with other exposure variables within the database (that is, fluence), and due to the uneven distribution of data within the database (for example, the irradiation temperature and flux range of PWR and BWR data show almost no overlap) neither neutron fluence rate nor irradiation time sufficiently improved the accuracy of the predictions to merit their use in the embrittlement correlation in this guide. Future versions of this guide may incorporate the effect of neutron fluence rate or irradiation time, or both, on *TTS,* as such effects are described in (**2**). The irradiated material database, the technical basis for the method of adjusting the reference temperature is developing the embrittlement correlation, and issues involved in its application, are discussed in a separate report. report (**1**). That report describes the nine different *TTS* equations considered in the development of this guide, some of which were developed using more limited datasets (for example, national program data (**3,4**)). If the material variables or exposure conditions of a particular application fall within the range of one of these alternate correlations, it may provide more suitable guidance.

1.3 This guide is Part IIF of Master Matrixexpected to be used in E706 which coordinates coordination with several standards used for addressing irradiation surveillance of light-water reactor vessel materials. MethodsMethod of determining the applicable fluence for use in this guide are addressed in Master Matrix Guides E706E482, Practices E560 (IC) and Guide E944 (IIA), , and Test Method E1005 (IIIA). . The overall application of these separate guides and practices is described in Practice E853 (IA). .

1.4 The values givenstated in customary U.S. SI units are to be regarded as the standard. The SI values given in parentheses are for information only.mathematical conversions to U.S. Customary units that are provided for information only and are not considered standard.

1.5 This standard guide does not define how the *shiftTTS* in transition temperature should be used to determine the final adjusted reference temperature. (Thattemperature, which would typically include consideration of the initial starting point, transition temperature before irradiation, the predicted *shift,TTS,* and the uncertaintyuncertainties in the shift estimation method.) 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:*[4]

A302 Specification for Pressure Vessel Plates, Alloy Steel, Manganese-Molybdenum and Manganese-Molybdenum-Nickel

A508 Specification for Quenched and Tempered Vacuum-Treated Carbon and Alloy Steel Forgings for Pressure Vessels

A533 Specification for Pressure Vessel Plates, Alloy Steel, Quenched and Tempered, Manganese-Molybdenum and Manganese-Molybdenum-Nickel

E185 Practice for Design of Surveillance Programs for Light-Water Moderated Nuclear Power Reactor Vessels

E560E482 Practice for Extrapolating Reactor Vessel Surveillance Dosimetry Results, E 706(IC)Guide for Application of Neutron Transport Methods for Reactor Vessel Surveillance, E706 (IID) (Withdrawn 2009)

E693 Practice for Characterizing Neutron Exposures in Iron and Low Alloy Steels in Terms of Displacements Per Atom (DPA), E 706(ID)

E706 Master Matrix for Light-Water Reactor Pressure Vessel Surveillance Standards, E 706(0) (Withdrawn 2011)[5]

E853 Practice for Analysis and Interpretation of Light-Water Reactor Surveillance Results

[4] For referenced ASTM standards, visit the ASTM website, www.astm.org, or contact ASTM Customer Service at service@astm.org. For *Annual Book of ASTM Standards* volume information, refer to the standard's Document Summary page on the ASTM website.

E944 Guide for Application of Neutron Spectrum Adjustment Methods in Reactor Surveillance, E 706 (IIA)
E1005 Test Method for Application and Analysis of Radiometric Monitors for Reactor Vessel Surveillance, E 706 (IIIA)
E2215 Practice for Evaluation of Surveillance Capsules from Light-Water Moderated Nuclear Power Reactor Vessels

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.

3.1.1 *best-estimate chemical composition*—the best-estimate chemical composition (copper [Cu] and [Cu], nickel [Ni], in wt %)phosphorus [P], and manganese [Mn] in %) 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,added, 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.

3.1.1.1 *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.

3.1.1.2 *Discussion*—

In all cases where engineering judgment wasis used to select a best estimate copper or nickelcopper, nickel, phosphorus, or manganese 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.

3.1.2 *fluence* (Φ)— the in this guide the term "*fluence*" refers to the fast (E > 1MeV) neutron fluence, that is, the number of neutrons per square centimeter with energy greater than 1.0 MeV at the location of interest.

3.1.3 *Gfluence rate*(Φ)—a in this guide the term "*fluence functionrate* term in the transition temperature shift equation." refers to the fast (E > 1MeV) neutron fluence rate, that is, the number of neutrons per square centimeter per unit time with energy greater than 1.0 MeV at the location of interest. This is also referred to as fast neutron flux.

3.1.4 *SMD—SRM*—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.Standard Reference Material. Also known as correlation monitor material.

3.1.5 *T*—$_c$—irradiation temperature at full power, in °F, and is°C, given by the estimated time-weighted average (based on the mean temperature over each fuel cycle) cold leg temperature for PWRs pressurized water reactors (PWRs) and recirculation temperature for BWRs.boiling water reactors (BWRs).

3.1.6 *TTS*—the predicted mean value of the transition temperature shift from predicted by the embrittlement 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 nonductilenon-ductile 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 temperaturetemperatures as the neutron damage accumulates. The present practice is to base that adjustment on the *increaseTTS* in transition temperature produced by neutron irradiation as measured at the Charpy V-notch 30-ft·lbf (41-J)41-J (30-ft·lbf) energy level. To establish pressure temperature operating limits during the operating life of the plant, a prediction of *adjustmentTTS* in transition temperature must be made.

4.1.1 In the absence of surveillance data for a given reactor material (see Practice E185 and E2215), the use of calculative procedures will be are necessary to make the prediction. Even when credible surveillance data are available, it will usually be