



Designation: E 560 – 01

Standard Practice for Extrapolating Reactor Vessel Surveillance Dosimetry Results, E 706(IC)¹

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INTRODUCTION

LWR power plant safety analysis report and subsequent neutron exposure parameter calculations for the PV wall and critical welds need to be verified using modern codes and information from surveillance dosimetry. That is, the location of critical welds relative to the axial and azimuthal flux map should be taken into account, as well as changes in fuel loading during periods when surveillance capsules are exposed.

This practice is intended to be used together with other E 706 LWR Matrix Standards to provide estimates of the neutron exposure and exposure rate (together with uncertainties) at positions at the inner diameter and within the pressure vessel wall of a light water reactor. Also provided will be estimates of gamma-ray exposure and exposure rates to interpret dosimetry sensor photo-reaction and other gamma-ray induced effects. Information used to make these estimates is obtained from neutron-gamma transport calculations and from neutron and other sensor monitors located in surveillance positions on the core side of the vessel and in the cavity outside the vessel wall **(1)**². Benchmark field irradiations of similar monitors also provide valuable information used in the verification of the accuracy of the calculations (a type of cross section covariance and dosimetry monitor counting calibration) **(1)**.

Knowledge of the time-dependent relationship between exposure parameters at surveillance locations and selected (r , θ , z) locations within the pressure vessel wall is required to allow determination of the time dependent radiation damage to the pressure vessel. The time dependency must be known to allow proper accounting for complications due to burn-up, as well as, changes in core loading configurations **(2-5)**. An estimate of the uncertainty in the neutron exposure parameter values at selected (r , θ , z) points in the vessel wall **(1)** is also needed to place an upper bound on the allowable operating lifetime of the reactor vessel without remedial action **(6-9)**. (See Guide E 509).

1. Scope

1.1 This practice covers analytical and analytical-experimental approaches that can be used to determine the variation in neutron exposure (fluence $E > 1.0$ MeV, dpa, etc.) and exposure rate and energy spectrum between surveillance locations and points in the pressure vessel wall. Procedures for

reporting² the results of these analyses with assigned uncertainties are also suggested. This practice deals with the physics-dosimetry aspects of surveillance programs and must be used in conjunction with other Matrix E 706 standards to provide extrapolations based on metallurgical damage correlations.

1.2 The physics-dosimetry relationships determined from this practice may be used to estimate pressure vessel damage through application of Practice E 693 and Guide E 900 standards, using fluence ($E > 1.0$ MeV), dpa, or damage function derived exposure parameters as independent exposure variables. Supporting the application of these standards is E 944, E 1018, E 1005, and E 854 standards, identified in 2.1.

¹ This practice is under the jurisdiction of ASTM Committee E10 on Nuclear Technology and Applications and is the direct responsibility of Subcommittee E10.05 on Nuclear Radiation Metrology.

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² The boldface numbers in parentheses refer to the list of references appended to this practice.

1.3 *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:

- C 859 Terminology Relating to Nuclear Materials³
- E 170 Terminology Relating to Radiation Measurements and Dosimetry⁴
- E 184 Practice for Effects of High-Energy Neutron Radiation on the Mechanical Properties of Metallic Materials, E706 (IB)^{4,5}
- E 185 Practice for Conducting Surveillance Tests for Light Water-Cooled Nuclear Power Reactor Vessels, E706 (IF)^{4,6}
- E 482 Guide for Application of Neutron Transport Methods for Reactor Vessel Surveillance, E706 (IID)^{4,6}
- E 509 Guide for In-Service Annealing of Light-Water Cooled Nuclear Reactor Vessels⁴
- E 636 Practice for Conducting Supplemental Surveillance Tests for Nuclear Power Reactor Vessels, E706 (IH)^{4,6}
- E 693 Practice for Characterizing Neutron Exposures in Iron and Low Alloy Steels in Terms of Displacements Per Atom (DPA), E706 (ID)^{4,6}
- E 706 Master Matrix for Light-Water Reactor Pressure Vessel Surveillance Standards, E706 (O)⁴
- E 844 Guide for Sensor Set Design and Irradiation for Reactor Surveillance, E706 (IIC)^{4,6}
- E 853 Practice for Analysis and Interpretation of Light-Water Reactor Surveillance Results, E706 (IA)^{4,6}
- E 854 Test Method for Application and Analysis of Solid State Track Recorder (SSTR) Monitors for Reactor Surveillance, E706 (IIIB)^{4,6}
- E 900 Guide for Predicting Neutron Radiation Damage to Reactor Vessel Materials, E706 (IIF)^{4,6}
- E 910 Test Method for Application and Analysis of Helium Accumulation Fluence Monitors for Reactor Vessel Surveillance, E706 (IIIC)^{4,6}
- E 944 Guide for Application of Spectrum Adjustment Methods in Reactor Surveillance E706 (IIA)⁴
- E 1005 Test Method for Application and Analysis of Radiometric Monitors for Reactor Vessel Surveillance, E706 (IIIA)⁴
- E 1006 Practice for Analysis and Interpretation of Physics Dosimetry Results for Test Reactors, E706 (II)⁴
- E 1018 Guide for Application of the ASTM-evaluated Cross Section Data Files, E706 (IIB)⁴
- E 1035 Practice for Determining Radiation Exposure for Nuclear Reactor Vessel Support Structures, E706 (IG)⁴
- E 2005 Guide for the Benchmark Testing of Reactor Do-

simetry in Standard and Reference Neutron Fields, E706 (IIE-1)^{2,3}

E 2006 Guide for the Benchmark Testing of LWR Calculations, E706 (IIE-2)^{2,3}

2.2 *American Society of Mechanical Engineers Standard: Boiler and Pressure Vessel Code, Sections III and XI*⁷

2.3 *Nuclear Regulatory Document:*

Code of Federal Regulations, Chapter 10, Part 50, Appendixes G and H⁸

3. Significance and Use

3.1 *Regulatory Requirements*—The Code of Federal Regulations (10CFR Part 50, Appendix H) requires the implementation of a reactor vessel materials surveillance program for all operating LWR's (10). The purpose of the program is to (1) monitor changes in the fracture toughness properties of ferritic materials in the reactor vessel beltline region resulting from exposure to neutron irradiation and the thermal environment, and (2) make use of the data obtained from the surveillance program to determine the conditions under which the vessel can be operated with adequate margins of safety throughout its service life. Practice E 185, derived mechanical property data, and (r, θ, z) physics-dosimetry data (derived from the calculations and cavity and surveillance capsule measurements (1) using Matrix E 706 physics-dosimetry standards) can be used together with information in Guide E 900 and Refs. 4, 11-18 to provide a relation between property degradation and neutron exposure, commonly called a "trend curve." To obtain this trend curve at all points in the pressure vessel wall requires that the selected trend curve be used together with the appropriate (r, θ, z) neutron field information derived by use of this practice to accomplish the necessary interpolations and extrapolations in space and time.

3.2 *Neutron Field Characterization*—The tasks required to satisfy the second part of the objective of 3.1 are complex and are summarized in Practice E 853. In doing this, it is necessary to describe the neutron field at selected (r, θ, z) points within the pressure vessel wall. The description can be either time dependent or time averaged over the reactor service period of interest. This description can only be obtained by combining neutron transport calculations with cavity and surveillance capsule measurements, benchmark irradiations of dosimeter sensor materials, and a knowledge of the core power distribution, including either the time dependence, or time averaged. Because core power distribution may change with time, the cavity or surveillance capsule measurement obtained early in plant life may not be representative of long-term reactor operation. Therefore, a simple normalization of neutron transport calculations to dosimetry data from a given capsule is unlikely to give a satisfactory solution to the problem over the full reactor lifetime. Guide E 482 and Guide E 944 standards provide detailed information related to the characterization of the neutron field for BWR and PWR power plants.

3.3 *Fracture Mechanics Analysis*—Currently, operating limitations for normal heat up and cool down transient imposed on the reactor pressure vessel are based on the fracture mechanics techniques outlined in the ASME Boiler and Pressure Vessel Code. This code requires the assumption of the presence of a surface flaw of depth equal to the thickness of the

³ *Annual Book of ASTM Standards*, Vol 12.01.

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

⁵ For standards that are in the draft stage and have not received an ASTM designation, see Section 5 as well as Figs. 1 and 2 of Matrix E706.

⁶ The reference in parentheses refers to Section 5 as well as Figs. 1 and 2 of Matrix E706.