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Standard Guide for Application of Neutron Transport Methods for Reactor Vessel Surveillance, E706 (IID)[1]

This standard is issued under the fixed designation E 482; 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 *Need for Neutronics Calculations*—An accurate calculation of the neutron fluence and fluence rate at several locations is essential for the analysis of integral dosimetry measurements and for predicting irradiation damage exposure parameter values in the pressure vessel. Exposure parameter values may be obtained directly from calculations or indirectly from calculations that are adjusted with dosimetry measurements; Guide E 944 and Practice E 853 define appropriate computational procedures.

1.2 *Methodology*—Neutronics calculations for application to reactor vessel surveillance encompass three essential areas: (*1*) validation of methods by comparison of calculations with dosimetry measurements in a benchmark experiment, (*2*) determination of the neutron source distribution in the reactor core, and (*3*) calculation of neutron fluence rate at the surveillance position and in the pressure vessel.

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 requirements prior to use.*

2. Referenced Documents

2.1 *ASTM Standards:*

E 170 Terminology Relating to Radiation Measurements and Dosimetry[2]

E 560 Practice for Extrapolating Reactor Vessel Surveillance Dosimetry Results, E706(IC)[2]

E 693 Practice for Characterizing Neutron Exposures in Iron and Low Alloy Steels in Terms of Displacements Per Atom (DPA), E706(ID)[2]

E 706 Master Matrix for Light-Water Reactor Pressure Vessel Surveillance Standards, E706(0)[2]

E 844 Guide for Sensor Set Design and Irradiation for Reactor Surveillance, E706(IIC)[2]

E 853 Practice for Analysis and Interpretation of Light-Water Reactor Surveillance Results, E706 (IA)[2]

E 944 Guide for Application of Neutron Spectrum Adjustment Methods in Reactor Surveillance, E706 (IIA)[2]

E 1018 Guide for Application of ASTM Evaluated Cross Section Data File, E706(IIB)[2]

E 2005 Guide for the Benchmark Testing of Reactor Dosimetry in Standard and Reference Neutron Fields E706 (IIE-1)[2]

E 2006 Guide for the Benchmark Testing of LWR Calculations E706 (IIE-2)[2]

2.2 *Nuclear Regulatory Documents:*[3]

NUREG/CR-1861 LWR Pressure Vessel Surveillance Dosimetry Improvement Program: PCA Experiments and Blind Test

NUREG/CR-3318 LWR Pressure Vessel Surveillance Dosimetry Improvement Program: PCA Experiments, Blind Test, and Physics-Dosimetry Support for the PSF Experiments

NUREG/CR-3319 LWR Pressure Vessel Surveillance Dosimetry Improvement Program: LWR Power Reactor Surveillance Physics-Dosimetry Data Base Compendium

NUREG/CR-5049 Pressure Vessel Fluence Analysis and Neutron Dosimetry

3. Significance and Use

3.1 *General*:

3.1.1 The methodology recommended in this guide specifies criteria for validating computational methods and outlines procedures applicable to pressure vessel related neutronics calculations for test and power reactors. The material presented herein is useful for validating computational methodology and for performing neutronics calculations that accompany reactor vessel surveillance dosimetry measurements (see Master Matrix E 706 and Practice E 853). Briefly, the overall methodology involves: (*1*) methods-validation calculations based on at least one well-documented benchmark problem, and (*2*) neutronics calculations for the facility of interest. The neutronics calculations on the facility of interest and on the benchmark problem should be as nearly the same as is feasible; in

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

[3] Available from Superintendent of Documents, U.S. Government Printing Office, Washington, DC 20402.

particular, the group structure and common broad-group microscopic cross sections should be preserved for both problems. The neutronics calculations involve two tasks: (*1*) determination of the neutron source distribution in the reactor core by utilizing diffusion theory (or transport theory) calculations in conjunction with reactor power distribution measurements, and (*2*) performance of a fixed fission rate neutron source (fixed-source) transport theory calculation to determine the neutron fluence rate distribution in the reactor core, through the internals and in the pressure vessel. Some neutronics modeling details for the benchmark, test reactor, or the power reactor calculation will differ; therefore, the procedures described herein are general and apply to each case. (See NUREG/CR–5049, NUREG/CR–1861, NUREG/CR–3318, and NUREG/CR–3319.)

3.1.2 It is expected that transport calculations will be performed whenever pressure vessel surveillance dosimetry data become available and that quantitative comparisons will be performed as prescribed by 3.2.2. All dosimetry data accumulated that are applicable to a particular facility should be included in the comparisons.

3.2 *Validation*—Prior to performing transport calculations for a particular facility, the computational methods must be validated by comparing results with measurements made on a benchmark experiment. Criteria for establishing a benchmark experiment for the purpose of validating neutronics methodology should include those set forth in Guides E 944 and E 2006 as well as those prescribed in 3.2.1. A discussion of the limiting accuracy of benchmark validation discrete ordinate radiation transport procedures for the LWR surveillance program is given in Reference **(1)**. Reference **(2)** provides details on the benchmark validation for a Monte Carlo radiation transport code.

3.2.1 *Requirements for Benchmarks*—In order for a particular experiment to qualify as a calculational benchmark, the following criteria are recommended:

3.2.1.1 Sufficient information must be available to accurately determine the neutron source distribution in the reactor core,

3.2.1.2 Measurements must be reported in at least two ex-core locations, well separated by steel or coolant,

3.2.1.3 Uncertainty estimates should be reported for dosimetry measurements and calculated fluences including calculated exposure parameters and calculated dosimetry activities,

3.2.1.4 Quantitative criteria, consistent with those specified in the methods validation 3.2.2, must be published and demonstrated to be achievable,

3.2.1.5 Differences between measurements and calculations should be consistent with the uncertainty estimates in 3.2.1.3,

3.2.1.6 Results for exposure parameter values of neutron fluence greater than 1 MeV and 0.1 MeV [$\phi(E > 1$ MeV and 0.1 MeV)] and of displacements per atom (dpa) in iron should be reported consistent with Practices E 693 and E 853, and

3.2.1.7 Reaction rates (preferably established relative to neutron fluence standards) must be reported for ^{237}Np(n,f) or ^{238}U(n,f), and ^{58}Ni(n,p) or ^{54}Fe(n,p); additional reactions that aid in spectral characterization, such as provided by Cu, Ti, and Co-A1, should also be included in the benchmark measurements. The ^{237}Np(n,f) reaction is an important reaction since it gives information sensitive to the same energy region as the iron dpa. Practices E 693 and E 853 and Guides E 844 and E 944 discuss this criterion.

3.2.2 *Methodology Validation*—It is essential that the neutronics methodology employed for predicting neutron fluence in a power reactor pressure vessel be validated by accurately predicting appropriate benchmark dosimetry results. In addition, the following documentation should be submitted: (*1*) convergence study results, and (*2*) estimates of variances and covariances for fluences and reaction rates arising from uncertainties in both the source and geometric modeling.

3.2.2.1 For example, model specifications for S_n methods on which convergence studies should be performed include: (*1*) group structure, (*2*) spatial mesh, and (*3*) angular quadrature. One-dimensional calculations may be performed to check the adequacy of group structure and spatial mesh. Two-dimensional calculations should be employed to check the adequacy of the angular quadrature. A P_3 cross section expansion is recommended along with an S_8 minimum quadrature.

3.2.2.2 Uncertainties that are propagated from known uncertainties in nuclear data need to be addressed in the analysis. The uncertainty analysis for discrete ordinate codes may be performed with sensitivity analysis as discussed in References **(3, 4)**. In Monte Carlo analysis the uncertainties can be treated by a perturbation analysis as discussed in Reference **(5)**. Appropriate computer programs and covariance data are available, however, and sensitivity data may be obtained as an intermediate step in determining uncertainty estimates.[4]

3.2.2.3 Effects of known uncertainties in geometry and source distribution should be evaluated based on the following test cases: (*1*) reference calculation with a time-averaged source distribution and with best estimates of the core, and pressure vessel locations, (*2*) reference case geometry with maximum and minimum expected deviations in the source distribution, and (*3*) reference case source distribution with maximum expected spatial perturbations of the core, pressure vessel, and other pertinent locations.

3.2.2.4 Measured and calculated integral parameters should be compared for all test cases. It is expected that larger uncertainties are associated with geometry and neutron source specifications than with parameters included in the convergence study. Problems associated with space, energy, and angle discretizations can be identified and corrected. Uncertainties associated with geometry specifications are inherent in the structure tolerances. Calculations based on the expected extremes provide a measure of the sensitivity of integral parameters to the selected variables. Variations in the proposed convergence and uncertainty evaluations are appropriate when the above procedures are inconsistent with the methodology to be validated. As-built data could be used to reduce the uncertainty in geometrical dimensions.

[4] Much of the nuclear covariance and sensitivity data have been incorporated into a benchmark database employed with the LEPRICON Code system. See reference **(6)**.