American Nuclear Society

Methods for Determining Neutron Fluence in PWR and BWR Pressure Vessel and Reactor Internals

An American National Standard

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Foreword

(Note: This forward is not part of American National Standard ANSI/ANS-19.10-200x, “Methods for Determining Neutron Fluence in PWR and BWR Pressure Vessels and Reactor Internals.”)

It is the intent of this American National Standard to provide guidance for the evaluation of pressurized- and boiling-water reactor pressure vessel and reactor internals fast neutron (E > 1.0 MeV) fluence. This standard outlines the attributes of the method, the necessary types of data, the required benchmarking of the method, and the necessary steps in performing the calculations. The method described herein requires both experimentally measured vessel dosimetry data and corresponding fast neutron fluence calculations to perform the benchmark. The fluence value to be employed is the result of calculations using a benchmarked code. This standard also allows the user to determine the existence of a bias in the calculated values and to quantify its magnitude. Likewise, the information needed for the benchmark allows for the quantification of uncertainties. The method described in this standard allows the user to calculate a best estimate value that is acceptable for use in applications related to 10 CFR 50.61 and Appendices G and H to 10 CFR Part 50. The intended applications are for American-made pressurized- and boiling-water reactors.

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Compliance with the intent of this standard can be demonstrated by meeting the following two requirements:

(1) The calculation has been validated as described in Section 6 of this standard.

(2) The validation has been based on a qualified database from measurements performed as described in Section 5 of this standard.

Suggestions for the improvement of this standard are welcome. They should be sent to the American Nuclear Society, 555 North Kensington Avenue, La Grange Park, Illinois 60526.

Working Group ANS-19.10 of the American Nuclear Society developed this standard. At the time of the standard’s completion, the following members participated in the current version:

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Contents

1.0 Introduction

Fast neutron fluence values (i.e., with energies (E) ≥ 1.0 Mega-electron Volt (MeV)) on reactor pressure vessels or reactor internals are needed to predict material properties and determine operability limits for normal operation, anticipated operational occurrences, and accident conditions in pressurized-water reactor (PWR) and boiling-water reactor (BWR) plants. These properties must be predicted as accurately as possible. Such predictions extend into many plant operating (refueling) cycles (i.e., many calendar years of operation).

This standard is intended for use by (1) those involved in the determination of material properties of irradiated reactor vessel and reactor internals (as for example the determination of pressure vessel fluence to satisfy the requirements of Title 10, Section 50.61, “Fracture Toughness Requirements for Protection Against Pressurized Thermal Shock Events,” of the *Code of Federal Regulations* (10 CFR 50.61) and Appendix G, “Fracture Toughness Requirements,” and Appendix H, “Reactor Vessel Material Surveillance Program Requirements,” to 10 CFR Part 50, “Domestic Licensing of Production and Utilization Facilities”) and (2) regulatory agencies for the evaluation of licensing actions concerning the material properties of irradiated pressure vessels and irradiated reactor internals.

2.0 Scope

This standard provides a procedure for the evaluation of the best estimate fast neutron fluence in the annular region between the core and the inside surface of the vessel, through the pressure vessel and the reactor cavity, and between the planes defined by the top and bottom of the active fuel. The neutron source in the core is assumed to be known from other methods. This evaluation employs both fast neutron flux (E ≥ 1.0 MeV) computations and measurement data from in-vessel and reactor cavity dosimetry, as appropriate. This standard applies to U.S.-designed PWR and BWR plants.

3.0 Definitions

The following definitions apply for the purposes of this standard. Other specialized terms conform to *Glossary of Terms in Nuclear Science and Technology* [1][1].

Benchmark. A well-defined set of physical experiments or mathematical constructs with results judged by a group of experts in the subject area to be sufficiently accurate for use as a calculational reference point.

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1 Numbers in brackets refer to corresponding numbers of references in Section 9.0.

Best Estimate Fluence. The most accurate value of the fluence based on all available measurements, calculated results, and adjustments, taking into account bias estimates, least-squares analyses, and engineering judgment.

Calculational Methodology. The mathematical equations, approximations, assumptions, associated parameters, and procedures that yield the calculated results. When the calculation involves more than one step, the entire sequence of steps comprises the calculational methodology.

Continuous Energy Cross-Section Data. Cross-section data that are specified in a dense point-wise manner that comprises the energy range.

Dosimeter Reaction. A neutron-induced nuclear reaction with a product nuclide having sufficient activity to be measured and related to the incident neutron fluence.

Least-Squares Adjustment Procedure. A method for combining the results of neutron transport calculations and the results of dosimetry measurements that provides an optimal estimate of the fluence by minimizing, in the least-squares sense, the measurement-to-calculation (M/C) differences.

Multigroup Cross-Section Data. Cross-section data that have been determined by averaging the continuous energy cross-section data over discrete energy intervals using specified weighting functions.

Neutron Fluence. The time-integrated neutron flux as expressed in neutrons per square centimeter.

Reactor Internals. Reactor components that are within the pressure vessel, such as the core shroud, core baffle, core barrel, thermal shield, jet pump and riser, core plate, and top guide.

Shall, Should, and **May.** The word “shall” denotes a requirement; the word “should” denotes a recommendation; and the word “may” denotes permission, neither a requirement nor a recommendation.

Solution Variance. A measure of the random statistical variation of the Monte Carlo transport solution because of a finite number of particle histories. Mathematically, it is the second central moment of the distribution about the mean value, which is used to measure the dispersion of the distribution about the mean.

4.0 Transport Theory Calculational Methods

4.1 General

The goal is to accurately determine flux or fluence distributions for the analysis of integral dosimetry measurements and for the prediction of irradiation damage to vessel internals and to the pressure vessel. In the practice suggested in this standard, a source distribution throughout the core is prepared using the results of core physics calculations; multidimensional transport theory calculations then are performed to propagate the neutrons to regions outside the core. Here, neutron transport to the pressure vessel and fast neutron vessel dosimetry measurements for commercial plants are based on average values for one or more plant operating (refueling) cycles.

This standard uses codes based on transport theory to determine multigroup three-dimensional flux distributions and to evaluate the reaction rates of dosimetry materials. Transport theory calculations should be performed using deterministic discrete ordinates (S_N) [2] or statistical Monte Carlo [3] approaches, as discussed in Sections 4.2.2 and 4.2.3, respectively. Other transport methods may be used if they are part of a benchmarked methodology. In this context, benchmarking is the process of deriving acceptable uncertainties and biases for fluence by comparing calculated to measured dosimetry values. It is assumed that the process adheres to the guidance in this standard.

4.2 Transport Calculation

4.2.1 Input Data

The following four major types of input are required:

(1) Material composition—Material composition, number densities, initial fuel cycle material composition, and coolant or moderator density are required.

(2) Geometric model—The geometric model should represent the physical configuration as closely as practical. As-built dimensions of the reactor configuration should be used when available.

(3) Cross-section data—Appropriate cross-section data should be used. Earlier cross-section sets may be used if they are part of a benchmarked methodology. Major considerations include (a) the accuracy of the data evaluation (e.g., Evaluated Nuclear Data File/B), (b) the energy group structure (when using the discrete ordinates method), (c) the order of the scattering anisotropy (i.e., P_n expansion), and (d) the method used for group-collapsing.

(4) Core neutron source—Determination of the core neutron source requires that, for calculations in cylindrical geometry, the (x, y, z) power distribution must be converted to (r, 2, z) geometry. In addition, the multigroup neutron source spectrum and the average number of neutrons produced per fission (<-bar) must be determined.

4.2.2 Discrete Ordinates (S_N) Method

To ensure an accurate representation of three-dimensional effects, three-dimensional discrete ordinates transport calculations must be used when practical. When three-dimensional calculations are not practical, a synthesis method may be used to determine the three-dimensional flux or fluence distribution. In this approach, the fluence distribution is determined by synthesizing the results of one- and two-dimensional discrete ordinates solutions [4].

4.2.3 Monte Carlo Transport Method

The geometric model used in the Monte Carlo analyses should reflect the actual physical configuration. The great flexibility in typical Monte Carlo codes allows for a very detailed representation, and this should be used to represent all the important features of the geometry under consideration.

Typically, Monte Carlo codes allow the use of either multigroup or continuous-energy cross-sections. The continuous-energy cross-sections should be used when either the neutron transport or the response function being calculated has a strong energy dependence that is not adequately represented by the multigroup library. (The continuous energy results may be edited in a multigroup format.)

In addition to the considerations in Section 4.2.1 above, the Monte Carlo model construction requires a technique to reduce the solution variance. Variance reduction techniques [3, 5] that have been validated may be used to reduce the variance in the Monte Carlo calculation. Techniques that may be used to improve the statistics at locations far from the core include (1) neutron energy cutoff, (2) source biasing, (3) geometry splitting with Russian Roulette, and (4) weight windows.

4.2.4 Adjoint Fluence Calculations

Because the reactor conditions are generally dependent on the fuel cycle, multiple transport calculations are required to track the fluence during plant operation. However, when the operating conditions that affect the transport calculation (e.g., downcomer and core bypass coolant temperatures, core mechanical design) remain constant, the multiple transport calculations may be replaced by a single adjoint calculation [6]. The adjoint is calculated for a source located at the vessel (or other) location of interest, which is taken to be proportional to the energy-dependent response cross-section. Typically, the source is taken to be unity above 1.0 MeV and zero below 1.0 MeV. When a dosimeter reaction rate is required, the source typically is taken to be equal to an energy-dependent dosimeter cross-section.

The fluence (or reaction rate response) at the location of interest is then determined for each reactor fuel cycle by integrating the cycle-specific core neutron source over the calculated adjoint function.

4.3 Validation Procedure for Neutron Fluence Calculated Values

Before performing transport calculations for a particular facility, the calculational methodology shall be validated by (1) comparing results with benchmarked calculations and measurements and (2) demonstrating that the methodology accurately determines appropriate benchmark dosimetry results.

4.4 Determination of Calculational Uncertainties

Calculational uncertainties associated with the methodology for predicting neutron fluence typically include the following:

- nuclear data (e.g., transport cross-sections, dosimeter reaction cross-sections, and fission spectra)

- geometry (e.g., locations of internals and deviations from the nominal dimensions)

- isotopic composition of material (e.g., density and composition of coolant water, vessel internals, the core barrel, thermal shielding, the shroud, jet pumps, the pressure vessel with cladding, concrete shielding)

- neutron sources (e.g., space and energy distribution, fuel burnup dependence)

- methods error (e.g., mesh density, angular expansion, convergence criteria, macroscopic group cross-sections, fluence perturbation by surveillance capsules, spatial synthesis, and cavity streaming)

These uncertainties shall be evaluated before performing transport calculations for a particular facility.

5.0 Reactor Pressure Vessel Neutron Dosimetry Measurements

Although the final product of this standard is a calculated fluence value, the code must be benchmarked to ensure that the calculated value is defensible and as close to reality as possible. Dosimetry measurements provide the means for the benchmarking. Therefore, the quality of the dosimetry is an important contributor to the calculational accuracy.

Accurate neutron dosimetry provides reasonable assurance that predictions of the reactor vessel neutron fluence at any critical location are accurate and reliable. In this regard, ratios of the measured to calculated dosimeter response are determined for each dosimeter. The measured-to-calculated (M/C) ratios are then used to assess the existence of any biasing mechanisms operative within the calculational process.

5.1 General Requirements for Reactor Pressure Vessel Neutron Metrology

Specific procedures delineated in applicable standards on neutron metrology published by the American Society for Testing and Materials (ASTM) shall be followed [7–20]. The general requirements for neutron monitors used for reactor pressure vessel dosimetry are outlined below, as are several specific requirements unique to stable-product neutron monitors:

- Types of activation detectors—The recommended set of activation detectors covering the spectral energy range from approximately 0.08 MeV to 10.0 MeV includes ^{237}Np, ^{238}U, ^{58}Ni, ^{54}Fe, ^{46}Ti, ^{63}Cu, and possibly ^{93}Nb.

- Nuclear and material properties of dosimeters—The physicochemical properties must be compatible with the prevailing service conditions; for example, the dosimeter should not melt and should be chemically stable and corrosion resistant. Basic nuclear properties to be considered when implementing fissionable-material dosimeters include activation product half-life, reaction cross-section, gamma-ray yield and fission yield.

- Dosimeter mass and isotopic composition—Dosimeters shall be of high isotopic purity and sufficient mass for adequate activation.

- Dosimeter geometry and configuration—In general, dosimeters are in the form of thin activation foils, although other shapes are available. The foil thickness is an important consideration for self-shielding during irradiation and photon absorption or fission-product loss from recoil during counting.

- Spectral coverage—Neutron dosimeters should possess adequate spectral coverage. In particular, the dosimeter should enable separate benchmarking calculations of the neutron fluence above 0.1 MeV and above 1.0 MeV. (Bias factors with their related uncertainties are discussed in ASTM E 261-03, “Standard Practice for Determining Neutron Fluence, Fluence Rate and Spectra by Radioactivation Techniques,” issued 2006 [20], and should be included in the uncertainty evaluation.)

- Selection of alternative combinations of monitors—ASTM E 844-03, “Standard Guide for Sensor Design and Irradiation for Reactor Surveillance, E 706 (IIC),” issued 2006 [14], and ASTM E 1005-03, “Standard Test Method for Application and Analysis of Radiometric Monitors for Reactor Vessel Surveillance, E 706 (IIIA)," issued 2006 [15], provide guidance on composing an appropriate dosimetry package.

- Irradiation geometry and dosimeter location—Dosimeters should be placed as close as possible to the location of interest. The dosimeter location shall be determined accurately and recorded. Structures and materials surrounding a dosimeter that can influence monitor response should be avoided when possible. When these structures or materials are present, their effect shall be assessed and included within the overall fluence determination.

- Dosimeter encapsulation—Neutron dosimeters are often placed within some form of encapsulating neutron filters or within the in-vessel surveillance capsule. The filter and capsule design should minimize perturbations to the neutron flux and spectrum. Such perturbations should be assessed and included within the overall fluence determination.

- Irradiation parameters—Exposure time, the associated power history, and the effects of dosimeter burnout shall be accurately determined.

- Dosimeter analysis—Radioassay of active species is most commonly done by direct nuclear counting with a high-resolution gamma-ray spectrometer (usually GeLi detectors, (Lithium-drifted Germanium)). When conditions preclude direct counting, one can employ radiochemical separation. In either case, a complete description of the gamma-ray spectrometer and the counting techniques employed shall be included as part of the dosimetry documentation.

5.2 Stable-Product Neutron Monitors

In addition to radiometric monitors, stable-product neutron monitors also are used for reactor fluence determinations. These devices include solid-state track recorders (SSTRs) and helium accumulation fluence monitors (HAFMs). These devices provide a permanent measurement record because of their stable responses. The provisions of ASTM E 854-03, "Standard Test Method for Application and Analysis of Solid State Track Recorder (SSTR) Monitors for Reactor Surveillance, E 706 (IIIB)," issued 2006 [17], for SSTRs and ASTM E 910-01, "Standard Test Method for Application and Analysis of Helium Accumulation Fluence Monitors for Reactor Vessel Surveillance, E 706 (IIIC)," issued 2006 [18], for HAFMs should be observed.

5.3 Detector Response Parameters

As discussed in Section 4.0, dosimeter response should be calculated and compared to the measured values described in this section. The M/C ratio can be used to validate the calculational methodology (see Section 6.0).

5.4 Uncertainty Estimates and Measurement Validation in Standard Neutron Fields

To effect a meaningful comparison between measured results and the corresponding calculated quantities, the uncertainty and bias associated with the measurement process must be carefully evaluated. Sources of uncertainty include dosimeter physical parameters, irradiation characteristics (e.g., reactor power history and decay times), nuclear data (e.g., decay constants, fission yields, nuclear cross-sections, and photon attenuation coefficients), and the nuclear counting process. Additional uncertainty sources may be present, and their presence should be investigated on a case-by-case basis.

Because dosimetry measurements are used to validate the calculational methodology, it is imperative to validate the measurement process by performing dosimetry measurements with the identical dosimeters that are exposed to certified fluences in standard neutron fields. ASTM E 2006-05, "Standard Guide for Benchmark Testing of Light Water Reactor Calculations," issued 2006 [19], discusses aspects of measurement validation in standard neutron fields. After validation in the standard neutron field, the measurement uncertainties and bias are typically insignificant compared to other uncertainties in the fluence determination.

6.0 Comparison of Calculations with Measurements

If the measurement data are of sufficient quality and quantity to allow a reliable estimate of the calculational bias and the uncertainty is within the required safety limits (i.e., they represent a statistically significant measurement database), the comparisons of calculations with measurements may be used to modify the calculation by applying a correction to account for bias, by adjusting the model, or both.

Several methods of comparison may be used in the validation process. When applying these methods, it is commonly assumed in these comparisons that the uncertainties associated with modeling, such as the spatial location of the detectors within the reactor vessel, are negligible. However, when this is not the case, the effect of these uncertainties on the comparisons shall be addressed.

6.1 Direct Comparison of Calculated Activities with Measured Sensor Activities

One method of comparison is to directly compare the calculated dosimeter-specific activities at the end of each irradiation segment with the corresponding measured dosimeter activities. This method enables various segments of the irradiation to be summed to obtain the total activity. The disadvantage is that experimental results from different irradiations cannot be compared directly without the introduction of transport theory calculations. An overall comparison of calculated and measured activities can be made by using a suitable weighted average of the M/C ratios. In determining this average, the weighting of individual sensor comparisons should include the uncertainties associated with measured activities as well as the energy spectrum coverage provided by each sensor.

6.2 Comparison of Calculated Rates with Measured Average Full-Power Reaction Rates

The second method of comparison is to derive the average full-power reaction rate for each sensor using the irradiation history of the dosimeter set. These reaction rates are independent of both the length of the irradiation and the time at less than full-power operation. The advantage of this approach is that the reaction rate comparisons permit direct comparisons of measured results from different reactors and different cycles of irradiation within the same reactor. Furthermore, comparisons of measured spectral indices (ratios of reaction rates from different sensors) provide comparisons of the energy spectra at different measurement locations. As discussed in Section 6.1, an overall M/C comparison can be made using a suitably weighted average of the reaction rate data.

6.3 Comparison of the Calculations against Measurements Using Least-Squares Methods

Another method of comparison to obtain a suitable weighting of the uncertainties in the measurements and calculations as well as the spectral coverage of the individual sensors is to apply least-squares adjustment procedures. Least-squares methods provide the capability of combining the measurement data with the neutron transport calculations, resulting in an adjusted neutron energy spectrum with associated uncertainties [21]. Estimates for key exposure parameters, such as displacements per atom (dpa) N(E ≥ 1.0 MeV) or dpa/second (dpa/s), along with their uncertainties are then obtained from the adjusted spectrum and the energy-dependent cross-sections. These exposure parameters from the adjusted spectrum can then be compared directly with the calculated exposure parameters to provide validation of the calculated results. The advantage of this method of validation is that the comparisons are performed directly for quantities of interest to pressure vessel integrity evaluations (e.g., N(E ≥ 1.0 MeV) and dpa/s).

7.0 Determination of the Best Estimate Fluence

The guidance described in Section 4.0 is intended to provide a best estimate calculational methodology. For pressure vessel work, usually the inside surface azimuthal fluence distribution is calculated. In addition, the 1/4 vessel thickness (T) and 3/4T values for the critical vessel component at the maximum azimuthal location are also calculated. For other applications, the components and the location depend on the application.

The computed value of neutron exposure is considered acceptable for safety analysis provided that both of the following are true:

- The calculation has been validated as described in Section 6.0.
- The validation was based on a qualified database from measurements performed as described in Section 5.0.

8.0 Reporting Requirements

Detailed reporting requirements depend on the application of the particular problem. The objective of the reporting requirements is to provide an auditable paper trail. At a minimum, the following should be reported in a clear and concise manner:

- Identify the particular facility, its characteristics, and the existing and projected effective full-power years of operation for which the fluence is calculated.
- Specify the calculational method, nuclear data, materials and geometry and their sources, applicable approximations, neutron source, and validation procedure.
- For validation, identify the source of the measured data or reference an existing and applicable validation.
- For vessel fluence calculations (point or axial/azimuthal distribution), report values at vessel inside surface, 1/4T and 3/4T.
- Report the uncertainty and (if applicable) the bias and/or confidence level of the calculation.

9.0 References

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10. "Standard Test Method for Measuring Fast-Neutron Reaction Rates by Radioactivation of Nickel," ASTM E 264-02, American Society for Testing and Materials, Philadelphia, 2006.

11. "Standard Test Method for Measuring Fast-Neutron Reaction Rates by Radioactivation of Iron," ASTM E 263-05, American Society for Testing and Materials, Philadelphia, 2006.

12. "Standard Test Method for Measuring Fast-Neutron Reaction Rates by Radioactivation of Titanium," ASTM E 526-02, American Society for Testing and Materials, Philadelphia, 2006.

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14. "Standard Guide for Sensor Design and Irradiation for Reactor Surveillance, E 706 (IIC)," ASTM E 844-03, American Society for Testing and Materials, Philadelphia, 2006.

15. “Standard Test Method for Application and Analysis of Radiometric Monitors for Reactor Vessel Surveillance, E 706 (IIIA),” ASTM E 1005-03, American Society for Testing and Materials, Philadelphia, 2006.

16. “Standard General Methods for Detector Calibration and Analysis of Radionuclides,” ASTM E 181-98 (reapproved 2003), American Society for Testing and Materials, Philadelphia, 2006.

17. “Standard Test Method for Application and Analysis of Solid State Track Recorder (SSTR) Monitors for Reactor Surveillance, E 706 (IIIB),” ASTM E 854-03, American Society for Testing and Materials, Philadelphia, 2006.

18. “Standard Test Method for Application and Analysis of Helium Accumulation Fluence Monitors for Reactor Vessel Surveillance, E 706 (IIIC),” ASTM E 910-01, American Society for Testing and Materials, Philadelphia, 2006.

19. “Standard Guide for Benchmark Testing of Light Water Reactor Calculations,” ASTM E 2006-05, American Society for Testing and Materials, Philadelphia, 2006.

20. “Standard Practice for Determining Neutron Fluence, Fluence Rate and Spectra by Radioactivation Techniques,” ASTM E 261-03, American Society for Testing and Materials, Philadelphia, 2006.

21 “Standard Guide for Application of Neutron Spectrum Adjustment Methods in Reactor Surveillance, E 7006 (IIA)” ASTM E 944-02, American Society for Testing and Materials, Philadelphia, 2006.