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Standard Guide for
Monitoring the Neutron Exposure of LWR Reactor Pressure Vessels[1]

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INTRODUCTION

Light Water Reactor (LWR) power plant safety analysis reports and subsequent neutron exposure parameter calculations for the reactor pressure vessel (RPV) wall and critical welds need to be verified using modern codes and information from surveillance dosimetry. The location of critical welds relative to the axial and azimuthal fluence rate map should be taken into account, as well as changes in fuel loading during periods when surveillance capsules are exposed and beyond to the end of the reactor's operating license. For many reactors today this interval is 60 years. In the nuclear industry, there is active consideration and evaluation of operating intervals of 80 years. Most reactor surveillance programs were designed based on the guidance of Practice E185 with an operating life of 40 years. The Practice E185 surveillance programs are designed to select and irradiate the RPV material test specimens. The dosimetry in the surveillance capsule is there primarily to measure the neutron fluence to which the capsule's material specimens have been exposed.

In addition, those programs were based on the operating assumptions in place at the time; typically annual out-in core loading patterns and base load operation at a fixed reactor power level. Reactor operations have evolved so that low-leakage core loading patterns (L^3P) are the norm as are 18 month and 24 month fuel cycles and reactor power up-ratings of up to 20 %. Many reactors have now installed flux suppression features such as natural uranium fuel rods, full or part-length hafnium or B_4C rods, or stainless steel rods to minimize the neutron exposure of critical areas of the RPV. Such developments increase the need to comprehensively monitor the RPV accrued fluence through the extended operation period.

This guide is intended to be used together with other Standards to provide best 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 coupled neutron-gamma ray transport calculations and from neutron and gamma-ray sensors located in surveillance positions on the core side of the vessel and in the reactor 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 (**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 RPV. 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 E509.)

1. Scope

1.1 This guide establishes the means and frequency of monitoring the neutron exposure of the LWR reactor pressure vessel throughout its operating life.

1.2 The physics-dosimetry relationships determined from this guide may be used to estimate reactor pressure vessel damage through the application of Practice E693 and Guide E900, using fast neutron fluence (E > 1.0 MeV and E > 0.1 MeV), displacements per atom (dpa), or damage-function-correlated exposure parameters as independent exposure variables. Supporting the application of these standards are the E853, E944, E1005, and E1018 standards, identified in 2.1.

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, health, and environmental practices and determine the applicability of regulatory limitations prior to use.*

1.4 *This international standard was developed in accordance with internationally recognized principles on standardization established in the Decision on Principles for the Development of International Standards, Guides and Recommendations issued by the World Trade Organization Technical Barriers to Trade (TBT) Committee.*

2. Referenced Documents

2.1 *ASTM Standards:*[3]

E170 Terminology Relating to Radiation Measurements and Dosimetry

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

E482 Guide for Application of Neutron Transport Methods for Reactor Vessel Surveillance

E509 Guide for In-Service Annealing of Light-Water Moderated Nuclear Reactor Vessels

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

E844 Guide for Sensor Set Design and Irradiation for Reactor Surveillance

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

E900 Guide for Predicting Radiation-Induced Transition Temperature Shift in Reactor Vessel Materials

E944 Guide for Application of Neutron Spectrum Adjustment Methods in Reactor Surveillance

E1005 Test Method for Application and Analysis of Radiometric Monitors for Reactor Vessel Surveillance

E1018 Guide for Application of ASTM Evaluated Cross Section Data File

E2005 Guide for Benchmark Testing of Reactor Dosimetry in Standard and Reference Neutron Fields

E2006 Guide for Benchmark Testing of Light Water Reactor Calculations

E2215 Practice for Evaluation of Surveillance Capsules from Light-Water Moderated Nuclear Power Reactor Vessels

2.2 *American Society of Mechanical Engineers Standard:*

Boiler and Pressure Vessel Code, Sections III and XI[4]

2.3 *Nuclear Regulatory Document:*

Code of Federal Regulations, Chapter 10, Part 50, Appendix A – "General Design Criteria for Nuclear Power Plants," Appendix G – "Fracture Toughness Requirements," and Appendix H – Reactor Vessel Material Surveillance Program Requirements"[5]

3. Terminology

3.1 Definitions for terms used in this guide are found in Terminology E170.

4. Significance and Use

4.1 *Regulatory Requirements*—The USA Code of Federal Regulations (10CFR Part 50, Appendix H) requires the implementation of a reactor vessel materials surveillance program for all operating LWRs. Other countries have similar regulations. The purpose of the program is to (*1*) monitor changes in the fracture toughness properties of ferritic materials in the reactor vessel beltline[6] resulting from exposure to neutron irradiation and the thermal environment, and (*2*) make use of the data obtained from surveillance programs to determine the conditions under which the vessel can be operated with adequate margins of safety throughout its service life. Practice E185, derived mechanical property data, and (r, θ, z) physics-dosimetry data (derived from the calculations and reactor cavity and surveillance capsule measurements **(1)** using physics-dosimetry standards) can be used together with information in Guide E900 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 guide to accomplish the necessary interpolations and extrapolations in space and time.

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[2] The boldface numbers in parentheses refer to the list of references appended to this guide.

[3] 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.

[4] Available from American Society of Mechanical Engineers (ASME), ASME International Headquarters, Two Park Ave., New York, NY 10016-5990, http://www.asme.org.

[5] Available from U.S. Government Printing Office Superintendent of Documents, 732 N. Capitol St., NW, Mail Stop: SDE, Washington, DC 20401, http://www.access.gpo.gov.

[6] Per USNRC Regulatory Issue Summary 2014-11 (**10**), the reactor vessel beltline is defined as those portions of the RPV where the accumulated neutron fluence (E > 1.0 MeV) at the end of reactor operation will exceed 10^{17} cm^{-2}. The reactor vessel extended beltline is a term commonly used to refer to materials located outside of the region opposite the active core height that are also expected to accumulate neutron fluence (E > 1.0 MeV) at the end of reactor operation exceeding 10^{17} cm^{-2}.

4.2 *Neutron Field Characterization*—The tasks required to satisfy the second part of the objective of 4.1 are complex and are summarized in Practice E853. 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 best be obtained by combining neutron transport calculations with plant measurements such as reactor cavity (ex-vessel) and surveillance capsule or RPV cladding (in-vessel) measurements, benchmark irradiations of dosimeter sensor materials, and knowledge of the spatial core power distribution, including the time dependence. Because core power distributions change with time, reactor cavity or surveillance capsule measurements 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 E482 and Guide E944 provide detailed information related to the characterization of the neutron field for BWR and PWR power plants.

4.3 *Fracture Mechanics Analysis*—Currently, operating limitations for normal heat up and cool down transients 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 one fourth of the pressure vessel thickness. In addition, the fracture mechanics analysis of accident-induced transients (Pressurized Thermal Shock, (PTS)) may involve evaluating the effect of flaws of varying depth within the vessel wall (**4**). Thus, information is required regarding the distribution of neutron exposure and the corresponding radiation damage within the pressure vessel, both in space and time (**4**). In this regard, Practice E185 provides guidelines for designing a minimum surveillance program, selecting materials, and evaluating metallurgical specimen test results for BWR and PWR power plants. Practice E2215 covers the evaluation of test specimens and dosimetry from LWR surveillance capsules.

4.4 *Neutron Spectral Effects and DPA*—Analysis of the neutron fields of operating power reactors has shown that the neutron spectral shape changes with radial depth into the pressure vessel wall (**2, 3**). The ratio of dpa/ϕt (where ϕ is the fast (E > 1.0 MeV) neutron fluence rate and t is the time that the material was exposed to an average fluence rate) changes by factors of the order of 2.0/1.0 in traversing from the inner to the outer radius. Although dpa, since it includes a more detailed modeling of the displacement phenomenon, should theoretically provide a better correlation with property degradation than fluence (E > 1.0 MeV) (**1, 19**), this topic is still controversial and the available experimental data does not provide clear guidance (**19, 20**). Thus it is recommended to calculate and report both quantities; see Practice E853 and Practice E693.

4.5 *In-Vessel Surveillance Programs:*

4.5.1 The neutron dosimetry monitors used in reactor vessel surveillance capsules provide measurements of the neutron fluence and fluence rate at single points on the core midplane within the reactor, and near the vessel wall; that is, at the surveillance capsule locations (**1**). In actual practice, the surveillance capsules may be located within the reactor at an azimuthal position that differs from that associated with the maximum neutron exposure (or that differs from the azimuthal and axial location of the assumed flaw); and at a radial position a few centimeters or more from the flaw and the pressure vessel wall (**4, 5**). Although the surveillance capsule dosimetry does provide points for normalization of the neutron physics transport calculations, it is still necessary to use analytical methods that provide an accurate representation of the spatial variation (axial, radial and azimuthal) of the neutron fluence (refer to Guide E482). It is also necessary to use other measurements to confirm the spatial distribution of RPV neutron exposure.

4.5.2 Given that surveillance capsules are located radially closer to the core than the surface of the RPV, they may be shifted azimuthally away from the peak exposure location in order to limit the magnitude of the surveillance capsule lead factor. The lead factor is defined as the ratio of the fast neutron fluence at the center of the surveillance capsule to the peak fast neutron fluence at the clad–base metal interface of the RPV. One adverse effect of this azimuthal shift away from the peak is that the surveillance capsule dosimetry does not "see" the part of the core that produces the peak exposure of the reactor vessel. As a result, the surveillance capsule is unable to monitor the effect of changes in the core power distribution that are made to reduce the peak RPV neutron exposure. Another adverse effect is that with larger lead factors, the capsules are rapidly exposed to a high neutron fluence. For example, with a lead factor of five, a surveillance capsule will receive an exposure in as little as twelve years that is equivalent to what the reactor pressure vessel peak may see in 60 years of operation. Practices E185 and E2215 suggest not exceeding twice the maximum design fluence (MDF) or twice the end-of-license fluence (EOLF). In this example, this would require withdrawing any remaining surveillance capsules after 24 years of operation. Thus, without taking other steps, the reactor would be operated for the remaining 36 years (of a 60 year life) with no dosimetry present.

4.5.3 New or replacement surveillance capsules should recognize and correct operating deficiencies by using improved capsule dosimetry. For example, for one class of PWR, the copper wire is cadmium shielded to minimize interference from trace amounts of cobalt. In about one third of the measurements the copper has become incorporated into the cadmium preventing separation and further processing. A simple solution to this problem is to use stainless steel hypodermic tubing to contain and separate the radiometric monitor wire inside the cadmium tubing. Example dimensions include: Typical radiometric monitor wire outside diameter = 0.020 in. (0.5 mm). Typical 19 gauge stainless steel tubing is 0.042 in. outside diameter by 0.027 in. inside diameter, 0.008 in. wall thickness. Typical cadmium tubing is 0.090 in. outside diameter by 0.050 in. inside diameter, 0.020 in. wall thickness.

4.5.4 Guide E844 states that radionuclides with half-lives that are short compared to the irradiation duration should not be used. For one class of BWR reactor, the surveillance capsule dosimetry is minimal; consisting of an iron wire and a copper wire (sometimes also a nickel wire). This dosimetry is not suitable for longer irradiations as the "memory" of the activation products is too short to measure the accumulated fluence. For example, for the iron (n,p) activation product, ^{54}Mn, the half-life is 312 d. For the copper (n,α) activation product, ^{60}Co, the half-life is 5.27 a. After three half-lives the remaining activity is on the same order as the counting statistics. The result is that the iron wire has "forgotten" everything that has happened more than two cycles ago and the copper wire has forgotten everything that has happened more than eight cycles ago. This assumes 24-month-long fuel cycles. The copper (n,α) reaction is induced by high energy neutrons and that at a BWR surveillance capsule position only 1 % to 3 % of the fast (E > 1.0 MeV) neutrons are of high enough energy. This limits the value of the copper wire as a neutron fluence monitor. In order to monitor the neutron exposure of the RPV other dosimetry is needed. Installation of ex-vessel neutron dosimetry is the most reasonable and cost-effective option.

4.5.5 The neutron fluence calculation on the RPV inner surface can be further verified by means of analyzing small samples of the irradiated stainless steel RPV cladding. Analyzing RPV cladding samples has been a well-established practice for over 30 years (**21-36**). During the reactor shut down periods, small samples (50 mg to 100 mg) can be machined from the RPV cladding. For retrospective dosimetry purposes the measured ^{54}Mn, ^{58}Co, and ^{93m}Nb activities are used. Because of its long half-life, ^{93m}Nb is especially useful for integrating fluence over time periods where accurate neutron transport calculations are not available. With sample locations properly selected, the fast neutron fluence distribution and its maximum on the RPV inner surface can be determined. By comparison of these data to the dosimetry data of the surveillance capsules, the lead factor at the time of measurement can also be obtained. This technique works best if the cladding material is one of the niobium-stabilized stainless steels. Type 347 with 0.7 % niobium is one example. Retrospective dosimetry has been successfully demonstrated for ordinary Type 304 stainless steel cladding with only a trace (~50 ppm) of niobium (**35**). It is important that the cladding surface is first polished to remove radioactive corrosion products before the sample is machined otherwise competing activity may compromise the sample. The tooling used to take these samples needs to be accurately located relative to reactor landmarks in order to know the actual axial and azimuthal locations of the samples. A reasonable accuracy target is ±25 mm axially and azimuthally. The effect of the sampling position error can be estimated by examining the spatial fast neutron fluence rate gradient in the vicinity of the sample point. In general, in the areas where the fast neutron fluence is the greatest, the gradient tends to be very small; approaching flat in the case of the axial distribution opposite the middle of the core. At extreme axial positions, well beyond the ends of the core, the gradient is steep. There the positioning error could lead to an estimated fluence error of ±20 %. A similar discussion applies to the azimuthal fluence rate gradients. The tooling also needs to be designed to completely retain all machined cladding chips and to prevent cross-contamination from one sample to another. Access to the full extent of azimuthal and axial clad samples is generally limited to PWRs due to the extensive structure (jet pumps, etc.) blocking general access to the RPV cladding of many BWRs. It may be possible to take a more limited set of samples from the cladding of a BWR RPV.

4.5.6 The design and manufacture of new reactor pressure vessels should consider using one of the stainless steels or Inconel alloys that contains niobium for the purpose of cladding the inner surface of the vessel. This would result in a designed-in retrospective dosimetry system that would capture neutron exposure data from reactor startup.

4.6 *Ex-Vessel Surveillance Program:*

4.6.1 Ex-vessel neutron dosimetry (EVND) has also been in wide scale application in nuclear reactors for over 30 years (**28, 29, 31, 33, 35, 37-97**). The main advantages of EVND are the relative simplicity and the relatively low cost of the dosimetry system. Removal and replacement of irradiated dosimetry takes little time. Typical installations have dosimetry that spans the active core height and continues to cover the extended beltline region of the RPV. Installation of dosimetry at multiple angles allows full octant coverage (for octant symmetric cores). Some EVND installations include multiple measurements at symmetric azimuthal angles to confirm symmetry in the azimuthal fluence rate distributions. Asymmetries may result from such things as non-symmetric core power distributions, differences in water temperatures from one loop to another, or ovality in the as-built dimensions for the reactor internals or RPV. Dosimetry capsules typically contain a full complement of radiometric monitors (refer to Guide E844) to ensure good spectral coverage and fluence integration. Typically, capsules are connected and supported by stainless steel wires or chains, which are, in turn, segmented and counted to provide axial gradient information.

4.6.2 In order to minimize measurement field perturbation, the dosimeter capsules should be made of a neutron-transparent material such as aluminum. This also serves to reduce the radiation dose rates encountered when removing and replacing dosimetry. The gradient chains or wires should be a low mass per linear foot material, again to reduce the dose rates encountered during handling of irradiated dosimetry.

4.6.3 An ex-vessel neutron dosimetry system needs to be accurately located with respect to well-known and easily verified reactor features. A reasonable accuracy target is ±25 mm axially and azimuthally. The effect of the dosimetry position error can be estimated by examining the spatial fast neutron fluence rate gradient in the vicinity of the measurement point. In general, in the areas where the fast neutron fluence is the greatest, the gradient tends to be very small; approaching flat in the case of the axial distribution opposite the middle of the core. At extreme axial positions, well beyond the ends of the core, the gradient is steep. There the positioning error could lead to an estimated fluence error of ±20 %. A similar discussion applies to the azimuthal fluence rate gradients.

4.6.4 Ideally, the ex-vessel neutron dosimetry is installed before reactor startup so that it can provide data over the operating lifetime of the reactor. It is recommended that the ex-vessel neutron dosimetry be analyzed before and after significant plant modifications that would alter the neutron exposure of the reactor vessel. Some examples include switching from low-leakage core loading patterns back to out-in loading patterns (or vice versa), performing a significant (>10 %) uprating of the plant power, adding (or removing) core flux suppression absorbers or dummy fuel rods, or modifying the reactor internals geometry. The typical dosimetry replacement interval is between one and five 18-month-long fuel cycles (or equivalent intervals for other fuel cycle lengths).

4.6.5 Periodic measurements (either RPV cladding samples or EVND) serve to confirm neutron fluence projections and help to avoid problems that result from errors in reactor-specific calculational models (**98**).

4.6.6 Calculations of neutron fields in commercial reactors show that the neutron exposure (dpa) at the inner diameter of the pressure vessel can vary by a factor of three or more as a function of azimuthal position (**2**, **3**). Dosimetry monitors in the reactor cavity outside the reactor pressure vessel are a useful tool, therefore, in determining the accuracy of the neutron field calculations at points inside the pressure vessel wall. Practice E853 recommends the use of ex-vessel reactor cavity neutron dosimetry measurements for verification of the physics transport calculations. The status of benchmark field and power reactor applications as well as studies of this approach are discussed in Refs. **1**, **18**, **19**, **37-40**, **99-112**.

5. Neutron Exposure Monitoring

5.1 *Initial Conditions:*

5.1.1 This guide assumes the existence of an analysis of record that provides projections of future neutron exposure for materials in the reactor vessel that are predicted to experience sufficient neutron radiation damage to be considered in the selection of the most limiting material with regard to radiation damage. Projections of future values of neutron exposure are then used in subsequent reactor vessel integrity evaluations to demonstrate the reactor can be operated safely during normal and off-normal conditions.

5.1.2 The operational parameters used to generate projections of future neutron exposure are frequently subject to change. A program to periodically re-assess the neutron exposure should be instituted to confirm that the neutron exposure projections used in the reactor vessel integrity evaluations remain valid. For highest accuracy, calculations of exposure should be made for all past fuel cycles and projected to the future using best estimates of future fuel management. Significant changes in the neutron exposure projections may necessitate revisions of the reactor vessel integrity evaluations. When changes in calculated exposure rates are observed, the differences should be investigated, and the basis of such differences understood.

5.2 *Means of Monitoring:*

5.2.1 Neutron exposure monitoring can be achieved by periodically performing or updating calculations to reflect actual plant operating conditions, by collection and analysis of additional reactor dosimetry measurements to validate calculated exposure projections, or both.

5.2.2 Long periods of operation without any reactor dosimetry measurements can leave undetected errors in the inputs to the neutron exposure calculation methodology. When significant changes in exposure rates are calculated, new dosimetry measurements may be required to ensure exposure estimates are within required accuracy limits. Accurate analysis to relate dosimetry measurements to exposures at critical locations requires fluence calculations for each fuel cycle that the dosimetry is irradiated and, if shorter half-life dosimeter reactions are used, may require calculations for several time intervals within a fuel cycle.

5.2.3 Guide E482 provides detailed guidance related to the calculational determination of neutron exposure for BWR and PWR power plants, and the benchmarking of those calculations. Test Method E1005 describes procedures for measuring the specific activities of radioactive nuclides produced in radiometric monitors by nuclear reactions induced during surveillance exposures for reactor vessels and support structures.

5.3 *Frequency of Monitoring:*

5.3.1 The frequency with which neutron exposure monitoring activities should be performed is dependent upon circumstances unique to each reactor. To determine an appropriate time interval for the re-assessment of neutron exposure projections, consideration should be given to the degree of consistency of actual power operation with the assumptions used in developing the neutron exposure projections, the anticipated margin remaining between current and projected neutron exposure levels, the physical constraints on the half-lives of the sensor material used in the dosimeters, and potential ancillary uses for the results of the neutron exposure calculations (for example, equipment qualification or aging management of the reactor vessel internal structures). Non-technical considerations may also be important. Over long periods of time, staff turnover may lead to challenges recovering the necessary input data, or a loss of organizational focus may occur on important issues relating to radiation damage and aging management. It is important that plans be in place to ensure that all nuclear quality assurance requirements are met, including documentation of all inputs to exposure estimates, and all calculations be carried out and reviewed by qualified personnel.

5.3.2 For example, consider a reactor that has accrued sufficient neutron exposure to place it near regulatory screening criteria limits. If continued operation is desired, such a plant may consider implementing fuel changes for the purpose of reducing reactor vessel neutron exposure. For such a reactor, a shorter monitoring interval may be appropriate to ensure safe operability in-line with analyzed conditions. By contrast, a plant with wide margins between current and projected neutron exposure, that operates with core loading patterns and operational parameters that are highly consistent with the projection assumptions, may be justified in using a longer monitoring interval.

6. Supplementary Analytical Procedures

6.1 *Basic Approach*—ASTM Practice E853 covers various aspects of the extrapolation problem. The basic approach is that a transport calculation (benchmarked per Guide E482) is to be used to supply the neutron field information at the (r, θ, z) points in the pressure vessel wall where property deterioration information will be calculated using Guide E900, or other trend curves (**4, 11-18**). The dosimetry information obtained from reactor cavity and surveillance capsule measurements and retrospective dosimetry measurements from reactor internals structures and RPV cladding is to be used to ensure that the transport calculation is valid and to adjust the transport results if needed. The adjustments are to be accomplished using the guidelines presented in Guide E944. Dosimetry from monitors in the reactor cavity and surveillance capsules can provide limits on uncertainties for the calculated neutron field at selected (r, θ, z) positions in the reactor pressure vessel wall. Time dependence of the core power distribution (due to burnup within a given cycle, or due to variations in cycle to cycle fuel loading), surveillance capsule perturbation effects, and dosimetry monitor experimental effects must be recognized as complications, and these effects must be accounted for in the calculation and adjustment methods chosen (**1-6, 11**).

6.2 *Spatial Extrapolations:*

6.2.1 *Transport Codes*—In general, three-dimensional results need to be obtained for the neutron and gamma ray fields in the region from the core to the interior of the biological shield beyond the pressure vessel. As a minimum, a three-dimensional synthesis analysis should be performed using a two-dimensional transport code. The transport calculations in that case are carried out using the following three-dimensional synthesis technique:

$$\phi(r\ ,\ \theta\ ,\ z) = \phi(r\ ,\ \theta)\frac{\phi(r\ ,\ z)}{\phi(r)} \quad (1)$$

where ϕ (r, θ, z) is the synthesized three-dimensional flux distribution, ϕ (r,θ) is the transport solution in r,θ geometry, ϕ (r,z) is the two-dimensional solution for a cylindrical reactor model, and ϕ (r) is the one-dimensional solution for a cylindrical reactor model using the same source per unit height as that used in the two-dimensional r,θ calculation.

6.2.1.1 However, for problems and regions of interest where the transport solution, that is, radiation fields, have a nonseparable three-dimensional nature (due to the core power distribution, or reactor internals structures, or away from the core midplane), the synthesis technique may reduce the accuracy of the results, thus dictating that a full three-dimensional method be used. Analysis of the extended beltline, which often includes RPV nozzles, also dictates a full three-dimensional approach. An efficient way to carry out large 3D discrete ordinates S_n transport calculations is the use of multiple processors running in parallel (**112-116**). Monte Carlo methods are also used and these are traditionally run in parallel processing computing environments. Guide E482 should be followed for the calculations and Guide E944 for measured dosimetry adjustments. If a discrete ordinates method is used, the spatial mesh should be fine enough and the order of angular quadrature should be high enough to ensure a sufficiently accurate solution in all regions of importance. The impact of the selected discretization settings on the discrete ordinates results should be assessed (**117**). Methods of ensuring that the mesh is sufficiently fine are the province of Guide E482. Similar considerations apply to tallying techniques in Monte Carlo calculations. If ex-vessel reactor cavity dosimeter measurement results are used, the modeling in the reactor cavity and external shield should be adequate to provide usable calculations for the neutron field in the reactor cavity region. This requires an attention to mesh size in the ex-vessel region and an accurate representation of the geometry and chemical makeup of the external shield. Regardless of the method chosen, the effects of relevant input parameter variations on the calculated results should be well understood. Reference parameter variation studies focused on the reactor cavity and extended beltline region are available in Ref (**118**).

6.2.1.2 *Benchmarking*—It is not the purpose of this guide to dictate the type of transport calculation to be used in the region between the core and the outer radius of the pressure vessel (or into the biological shield) or the adjustment procedures, but any such calculations or adjustment procedures should be adequately benchmarked by calculations of well defined problems (for example, PCA Blind Test (**100**), VENUS (**107**), NESDIP (**108**), BWR (**104, 105**), and PWR (**1, 37-40, 99**). For further details on benchmarking refer to Guide E2006 and Guide E944. USNRC Regulatory Guide 1.190 (**119**) also addresses benchmarking of neutron transport calculations for RPV surveillance in some detail.

6.2.2 *Power Distribution*—As discussed in Practice E853, obtain a valid, adequately time dependent, core power distribution using a diffusion calculation, or a transport calculation (**99, 100, 107**). Experimental verification of the accuracy of the results is desirable, but may be difficult to obtain. This is especially important for the pin-by-pin power distributions at the core periphery and the axial power distributions at the ends of the core. The uncertainties in the core power distribution tend to be the largest in these areas. Fuel assembly geometric features also need to be considered in the development and modeling of the core power distribution. For example, some PWR fuel assemblies use low-enrichment axial blankets and some BWR fuel bundles use several different fuel rod lengths within the bundle.

6.2.2.1 Typically, calculations are performed on a fuel cycle-by-fuel cycle basis rather than using a single power distribution that is averaged over many fuel cycles. A well-documented basis should be used for extrapolating core power distributions into the future. Extrapolations should be based on best estimate projections of future fuel cycles. One common approach is to average the three most recent core power distributions and to use that for extrapolation. The assumption being that a similar core loading strategy will continue to be used. This assumption should be revisited whenever new measurements or core designs become available.

6.2.2.2 The power distribution should include the assembly-wise and axial variation of power as well as the finer, pin-by-pin distribution in the peripheral assemblies adjacent to the reactor internals. Details of the initial ^{235}U enrichment and the cycle changes in assembly burnup should also be determined as this is needed in order to define the mix of fissioning isotopes (for example ^{235}U, ^{238}U, ^{239}Pu, ^{240}Pu, ^{241}Pu, and ^{242}Pu) in each fuel assembly. Some BWR fuel bundles use multiple ^{235}U enrichments axially within a given fuel rod.

6.2.3 *Ex-Core Regions*—Perform a transport calculation for the neutron field in all ex-core regions, using adequate modeling of the reactor geometry, and adequate modeling of the ex-vessel region. The biological shield is to be accurately modeled both in terms of geometry (ex-core detector wells, support columns, and the presence or absence of a liner plate), and materials including the biological shield composition (cement, aggregate, water content, and distribution of reinforcing steel). The water content in the biological shield will vary over time (**120**). The energy, angle, and space discretization as well as neutron balance should be checked in all regions to make sure the calculation has converged, watching in particular for spatial oscillations or ray effects in ex-vessel regions. Monte Carlo calculations should be checked to confirm that acceptable tally statistics have been achieved.

6.2.4 *Power Plant Dimensions*—In all calculations, as-built dimensions should be used. If they are unavailable, documented logic should be presented to defend the dimensions used, and the uncertainty in the final results should reflect the added uncertainty. The thickness of the reactor pressure vessel (RPV) is a key dimension in the analysis of ex-vessel neutron dosimetry. There are two ways in which the accuracy of the assumed RPV thickness may be assessed.

6.2.4.1 The ratio of the ^{237}Np fission rate to the ^{54}Fe(n,p) reaction rate in the reactor cavity may be used as a spectral index. This ratio is very sensitive to the thickness of the RPV. For example, over an RPV thickness range from 100 mm to 200 mm, the reaction rate ratio increases by nearly a factor of two. Therefore, when the calculated spectral index from a calculation with an assumed RPV thickness agrees with the measured spectral index, one can have a high degree of confidence that the assumed thickness is correct. A difference in the spectral index can also indicate how much the assumed RPV thickness is off and in which direction. The calculated spectral index needs to be determined at the same azimuthal angle as the measurement being compared.

6.2.4.2 RPV pre-service or in-service inspections are usually performed using ultrasonic testing (UT) looking for flaws in the material. Usually these are multi-angle scans. However, sometimes a zero degree (normal incidence) scan is performed. This UT scan can provide a direct measured thickness for the RPV. With sufficient advance notice, a zero degree scan can be added to a future ISI program if the spectral index assessment indicates that the design basis RPV thickness is incorrect.

6.2.5 *Dosimetry Sensor Analysis*—For analysis of any given set of reactor cavity or surveillance capsule dosimetry sensors, the integral reactions or reaction rates of the individual sensors, or both, should be calculated, using the results of the transport calculation. The measurement and analysis procedures for individual sensors should be benchmarked for each sensor type; refer to Guide E2005. If the calculated and experimental integral results (C/E ratios) agree to within the required accuracy (~5 % to 15 %, 1σ being the best attainable, see Ref **100**) expected from the benchmark calibration of the transport code, the transport calculation may be used directly to calculate the neutron field at all (r, θ, z) points in the pressure vessel wall. If the C/E ratios do not agree within acceptable accuracy limits, a physics-dosimetry adjustment code analysis should be performed as outlined in 6.2.6.

6.2.6 *Physics-Dosimetry Adjustment Code Analysis*—Guide E944 should be used to combine the transport calculation with the dosimeter results. The Guide E944 adjustment procedure should be used to indicate whether the dosimeter measurements and associated uncertainties are consistent with the transport calculation and with uncertainties implied from benchmark tests of the transport code (PCA, VENUS, NESDIP, and an appropriate Commercial BWR or PWR; see Refs **1**, **37-40**, **99**, **100**, **104**, **105**). Having established the required consistency, the adjusted results of the transport calculation may be used to calculate the best estimate neutron field at all points in the pressure vessel wall with the uncertainty estimates derived from the application of the adjustment codes.

6.2.7 *Measurement Results*—If the calculated neutron field at the measurement location is inconsistent with the experimental dosimetry results, an attempt should be made to uncover and correct errors in order to obtain consistency. Particular attention should be paid to sensor monitor correction factors such as capsule perturbation, photo-reactions, impurities, burn-in / burn-out, and other effects. Discussions of how to proceed when calculations and measurements do not agree may be found in Practice E853, especially Section 7.3.

6.3 *Time Extrapolations*—In the case where a time averaged core loading has been used to define the future neutron source term, the fluence or dpa in future years is estimated by multiplying by the expected integrated time at full power.

7. Report and Bias of Results

7.1 As a minimum, the documentation of results should include the following information:

7.1.1 A description of the analytical technique used, including a listing of pertinent input parameters that may affect the bias of the calculation. For example, if the discrete ordinates approach is used, specify or reference the source of the cross-section data, cross-section preparation procedures, energy group structure, spatial mesh, S_N order, and P_L order. Dimensions and material compositions of key structures included in the model need to be included. Some of this information may be proprietary. In that case, the source of the data used and a general description should be provided.

7.1.2 Information indicating the bias of the analytical approach in steel-water systems, including the details of benchmark calculations used to validate the procedures, and data and the bias attained in the benchmark tests.

7.1.3 The calculated total, thermal, epi-thermal (also known as epi-cadmium fluence rate), E > 0.1 MeV, and E > 1.0 MeV neutron fluence rate-fluence values, and energy spectrum at the surveillance capsule, and any ex-vessel dosimetry locations. Also report calculated values of dpa/s and dpa at the same locations.

7.1.3.1 The location of peak fluence rate-fluence points on the surface and in the interior of the vessel wall are calculated values that are required for all the above exposure and exposure rate parameters, except for the thermal and epithermal fluence rates, which generally can be best determined by