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Designation: ~~E2006 – 16~~ E2006 – 22

Standard Guide for Benchmark Testing of Light Water Reactor Calculations[1]

This standard is issued under the fixed designation E2006; 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 covers general approaches for benchmarking neutron transport calculations for pressure vessel surveillance programs in light water reactor systems. A companion guide (Guide E2005) covers use of benchmark fields for testing neutron transport calculations and cross sections in well controlled environments. This guide covers experimental benchmarking of neutron fluence calculations (or calculations of other exposure parameters such as dpa) in more complex geometries relevant to reactor pressure vessel surveillance. Particular sections of the guide discuss: the use of well-characterized benchmark neutron fields to provide an indication of the accuracy of the calculational methods and nuclear data when applied to typical cases; and the use of plant specific measurements to indicate bias in individual plant calculations. Use of these two benchmark techniques will serve to limit plant-specific calculational uncertainty, and, when combined with analytical uncertainty estimates for the calculations, will provide uncertainty estimates for reactor fluences with a higher degree of confidence.

1.2 Although this guide and the companion guide, Guide E2005, are focused on power reactors, the principle of this guide is also applicable to non-power light water reactor pressure vessel surveillance programs.

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~~ safety, health, and ~~health~~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:*[2]

E170 Terminology Relating to Radiation Measurements and Dosimetry
E261 Practice for Determining Neutron Fluence, Fluence Rate, and Spectra by Radioactivation Techniques
E262 Test Method for Determining Thermal Neutron Reaction Rates and Thermal Neutron Fluence Rates by Radioactivation Techniques
E706 Master Matrix for Light-Water Reactor Pressure Vessel Surveillance Standards
E844 Guide for Sensor Set Design and Irradiation for Reactor Surveillance
E944 Guide for Application of Neutron Spectrum Adjustment Methods in Reactor Surveillance
E1006 Practice for Analysis and Interpretation of Physics Dosimetry Results from Test Reactor Experiments
E1018 Guide for Application of ASTM Evaluated Cross Section Data File

[1] This ~~test method~~ guide 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.

Current edition approved ~~June 1, 2016~~Feb. 1, 2022. Published ~~July 2016~~March 2022. Originally approved in 1999. Last previous edition approved in ~~2010~~2016 as ~~E2006 – 10.~~E2006 – 16. DOI: 10.1520/E2006-16.

[2] 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.

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

3. Terminology

3.1 *Definitions*—definitions of terms used in this guide may be found in Terminology E170.

4. Significance and Use

4.1 This guide deals with the difficult problem of benchmarking neutron transport calculations carried out to determine fluences for plant specific reactor geometries. The calculations are necessary for fluence determination in locations important for material radiation damage estimation and which are not accessible to measurement. ~~The~~ Typically, the most important application of such calculations is the estimation of fluence within the reactor vessel of operating ~~power plants~~ light water reactors (LWR) to provide accurate estimates of the irradiation embrittlement of the base and weld metal in the vessel. The benchmark procedure must not only prove that calculations give reasonable results but that their uncertainties are propagated with due regard to the sensitivities of the different input parameters used in the transport calculations. Benchmarking is achieved by building up data bases of benchmark experiments that have different influences on uncertainty propagation. For example, ~~fission spectra are the fundamental data bases which control propagation of cross section uncertainties, while such physics-dosimetry experiments as vessel wall mockups,~~ in simple vessel wall mockups where measurements are made within a simulated reactor vessel wall, ~~control error propagation associated with geometrical and methods approximations in the transport calculations.~~ the integral effect of uncertainties in iron cross sections (absorption and elastic and inelastic scattering) are dominant and have been bounded by the agreement between calculation and measurement. For more complicated integral benchmarks, other factors such as: uncertainties in the distribution of fission sources, geometry, the energy-dependent cross sections, and the angular scattering distribution for elemental components of major materials in the neutron field (such as water and iron) may all be important uncertainty contributors. This guide describes general procedures for using neutron fields with known characteristics to corroborate the calculational methodology and nuclear data used to derive neutron field information from measurements of neutron sensor response.

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4.2 The bases for benchmark field referencing are usually irradiations performed in standard neutron fields with well-known energy spectra and intensities. There are, however, less well known neutron fields that have been designed to mockup special environments, such as pressure vessel mockups in which it is possible to make dosimetry measurements inside of the steel volume of the "vessel". When such mockups are suitably characterized, they are also referred to as benchmark fields. A benchmark is that against which other things are referenced, hence the terminology "to benchmark reference" or "benchmark referencing". A variety of benchmark neutron fields, other than standard neutron fields, have been developed, or pressed into service, to improve the accuracy of neutron dosimetry measurement techniques. Some of these special benchmark experiments are discussed in this standard because they have identified needs for additional benchmarking or because they have been sufficiently documented to serve as benchmarks.

4.3 One dedicated effort to provide benchmarks whose radiation environments closely resemble those found outside the core of an operating reactor was the Nuclear Regulatory Commission's Light Water Reactor Pressure Vessel Surveillance Dosimetry Improvement Program (LWR-PV-SDIP) (**1**)[3]. This program promoted better monitoring of the radiation exposure of reactor vessels and, thereby, provided for better assessment of vessel end-of-life conditions. An objective of the LWR-PV-SDIP was to develop improved procedures for reactor surveillance and document them in a series of ASTM standards (see Matrix E706). The primary means chosen for validating LWR-PV-SDIP procedures was by benchmarking a series of experimental and analytical studies in a variety of fields (see Guide E2005).

5. Particulars of Benchmarking Transport Calculations

5.1 Benchmarking of neutron transport calculations involves several distinct steps that are detailed below.

5.1.1 Nuclear data used for transport calculations are evaluated using differential data or a combination of integral and differential data. This process results in a library of cross sections and other needed nuclear data (including fission spectra) that, in the opinion of the evaluator, gives the best fit to the available experimental and theoretical results. Some of the information used in evaluating the cross sections may be the same as that used directly for benchmarking transport calculations for LWR systems (see ~~4.1.2~~5.1.2). The cross section benchmarking itself is not addressed in this standard. It is assumed that the cross-section set is derived in this

[3] The boldface numbers given in parentheses refer to a list of references at the end of the text.

fashion to be applicable to a variety of calculational geometries and may not give the most accurate answer for LWR geometries. Thus further benchmarking in LWR geometries is required.

5.1.2 Transport calculations in LWR geometries may be benchmarked using measurements made in well-defined and well-characterized facilities that each mock-up part of an LWR-type system. These facilities have the advantage over operating plants that the dimensions and material compositions can be more accurately defined, the neutron source can be well characterized, and measurements can be made in a large number of locations that would not be accessible in actual ~~power systems. In power reactors, one is interested in the transport of neutrons from the distributed source in the fuel, through the reactor internals and water to the vessel, and through the vessel to the reactor cavity. Three mockups that together encompass this entire transport problem are described in~~ systems. ~~5.1. Modeling and calculating of neutron transport in these various geometries can be expected to identify any bias in specific parts of the calculations. Biases that can be detected include those due to modeling the irregular fuel geometry and distributed neutron source, those due to errors in the cross-sections or neutron spectra, and those due to calculational approximations.~~

5.1.2.1 In power reactors, one is interested in the transport of neutrons from the distributed source in the fuel, through the reactor internals and water to the vessel, and through the vessel to the reactor cavity. Three mockups that together encompass this entire transport problem are described in 6.1. Modeling and calculating of neutron transport in these various geometries can be expected to identify any bias in specific parts of the calculations. Biases that can be detected include those due to modeling the irregular fuel geometry and distributed neutron source, those due to errors in the cross-sections or neutron spectra, and those due to calculational approximations.

5.1.2.2 In non-power reactors, the objective is the same in that the purpose is to characterize the transport of neutrons from the distributed source in the fuel to and through the pressure vessel. However, in many non-power reactors, the geometries between the reactor core and the pressure vessel are significantly different from those represented by the mockups described in Section 6. In this case the evaluator must justify the validity of using the benchmarks discussed in Section 6. If these benchmarks cannot be justified, other benchmarks must be identified and their use justified.

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5.1.3 The benchmarking described above does not provide checks on geometries identical to actual plants and does not include bias that may exist in the definition of a specific plant model. Identification of these types of bias can only be accomplished using actual plant measurements. Benchmarking using these measurements is described in ~~5.2~~6.2 and ~~5.3~~6.3.

5.1.4 The final aspect of benchmarking is the benchmarking of the dosimetry results. This aspect is treated in Guide E2005. It is assumed that the measurements in the benchmarked facilities and in the actual operating plants are carried out using benchmarked reactions and dosimeters. This involves using reactions whose cross sections have been shown to be consistent with results in these types of neutron environments. Also, the dosimeters and measurement facilities must be of adequate quality and have measurement accuracies that have been verified (such as through round-robin testing). Periodic recalibration of laboratory measurement devices is also required using appropriate reference standards.

5.1.4.1 The selection and use of dosimeters should be according to Guide E844, and evaluation of the dosimetry results should be in accordance with Practice E261 and Test Method E262. In particular, to compare measured dosimetry results with calculated reaction rates or fluences, the following effects must be accounted for: effects of dosimetry perturbations, position or gradient corrections, gamma attenuation in counted foils, differences in counting geometry from that of calibration standards, dosimeter or reaction product burnup, effects of competing reactions in impurities and photofission or photoinduced reactions, and proper treatment of the irradiation history.

5.1.4.2 The benchmarking of the dosimetry results will also have indicated any bias that exists in the dosimetry cross sections. These cross sections are ~~essentially~~mostly independent of the transport cross sections discussed in ~~4.1.1~~5.1.1~~.~~, although some hidden correlations may be present due to evaluations taking into account integral results. Recommended dosimetry cross sections are given in Guide E1018.

5.1.5 The use of the benchmark data to determine bias in calculations and to determine best values for fluence in complex geometries is not straightforward. It often is not clear how to weight the impact of the different types of information when inconsistencies exist. Although, most calculations produce results that agree with measurements within acceptable tolerance, the cause of discrepancies within the tolerance may not be apparent from the available information. In this case, there is not universal agreement on the "best" answer, and the various approaches to use of the benchmark data can be adopted. Some of these approaches are described in Section ~~6~~7. Caution should be used if it is necessary to extrapolate beyond the limits of the benchmarks.

6. Summary of Reference Benchmarks for Transport Calculations for Reactor Pressure Vessel Surveillance Programs

6.1 *Special Benchmark Irradiation Fields:*

6.1.1 One dedicated effort to provide benchmarks whose radiation environments closely resemble those found outside the core of an operating reactor was the Nuclear Regulatory Commission's LWR-PV-SDIP (**1**). This program promoted better monitoring of the radiation exposure of reactor vessels and, thereby, provided for better assessment of vessel end-of-life conditions. In cooperation with other organizations nationally and internationally this program resulted in three benchmark configurations, VENUS (~~2~~**2-8**~~, 3, 4, 5, 6, 7, 8~~), PCA/PSF (~~9~~**9-15**~~, 10, 11, 12, 13, 14, 15~~), and NESDIP (~~16~~**16-19**~~, 17, 18, 19~~).

6.1.1.1 To serve as benchmarks, these special neutron environments had to be well characterized both experimentally and theoretically. This came to mean that differences between measurements and calculations were reconciled and that uncertainty bounds for exposure parameters were well defined. Target uncertainties were 5 % to 10 % (1σ). To achieve these objectives, benchmarked dosimetry measurements were combined with neutron transport calculations, and statistical uncertainty analysis and spectral adjustment techniques were used to establish the uncertainty bounds.

6.1.1.2 Taken together, the three benchmarks provide coverage from the fuel region to the vessel cavity. The VENUS facility was set up to measure spatial fluence distributions and neutron spectra near the fuel region and core barrel/thermal shield region. The PCA/PSF measurements looked at surveillance capsule effects and the fluence variation within the vessel itself. The NESDIP measurements overlap the PCA/PSF measurements and extend into the cavity behind the vessel. Investigations of axial streaming in the cavity were also conducted in NESDIP.

6.1.2 *The VENUS Benchmark:*

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6.1.2.1 The special benchmark field was developed at the VENUS Critical Facility CEN/SCK Laboratories, Belgium (~~2~~**2-8**~~, 3, 4, 5, 6, 7, 8~~). The facility could mock up PWR fuel geometries to investigate the fluence rate distributions in regions affected by the deviations from cylindrical symmetry. In addition, measurements on the VENUS fuel investigated the edge effects on power produced by individual pins at the outside of the fuel region and thus better established the neutron source. These data provided verification of both the flux magnitude and the azimuthal flux shape. The mock up included a simulated core barrel and thermal shield.

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6.1.2.2 There were several phases to the VENUS program. The first PV mockup configuration studies (VENUS-I) provided a link between the PCA and PSF tests and the actual environments of LWR power plants. Indeed for actual power plants, the azimuthal variation of the power distribution determined largely by complex stair-step-shaped core peripheries and by the core-boundary fuel power distributions could not be ignored, otherwise the calculations could contain undetected biases. Such biases could be further exacerbated by the use of low-leakage fuel-management schemes.

6.1.2.3 A second configuration, VENUS-2, contained a plutonium-fueled zone at the periphery of the core (to simulate burned fuel), and its objective was to investigate how much the fast neutron fluence is affected by such a core loading, and if changes in calculational modeling are necessary to account for any effects. The VENUS facility could also provide data to be used in validation of other sources asymmetries, such as those due to loading of absorber pins or dummy fuel rods in external assemblies to limit neutron leakage.

6.1.3 *The PCA/PSF Benchmark:*

6.1.3.1 The task of developing benchmark fields to meet surveillance dosimetry needs began with the construction, adjacent to the Oak Ridge National Laboratory (ORNL) Pool Critical Assembly (PCA), of a full-scale-section mockup of a pressure vessel wall in which passive and active dosimetry measurements (including neutron spectroscopy) could be made both outside and within the steel mockup (**9**, **10**, **20**). Measurement positions corresponding to the ¼, ½ , and ¾ thicknesses of the pressure vessel were provided. A simulated surveillance capsule was added to the mockup also. Extensive measurements and calculations provided sufficient characterization of the PCA benchmark experiment so that it was used for a blind test of neutron transport calculations (**9**).

6.1.3.2 The PCA benchmark also served as the critical facility for a higher fluence model of the PCA built at the Pool Side Facility (PSF) of the 30 MW Oak Ridge Research Reactor (ORR). The PSF made it possible to perform simultaneous dosimetry and metallurgical irradiations at the simulated surveillance capsule position and positions within the vessel wall. Such measurements within the vessel wall are not possible in an operating power reactor. The PSF measurements consisted of a startup experiment to confirm similarity with the PCA results, a long-term vessel wall irradiation with extensive dosimetry contained in capsules with dosimetry specimens, and three additional experiments to investigate surveillance capsule effects. The PSF irradiation facility consisting of the pressure vessel simulator was identified as the Simulated Dosimetry Measurement Facility (SDMF). The SDMF irradiations were carried out at high-flux with the Oak Ridge Reactor at 30 MW in a series of seven experiments; refer to Appendix A of reference 13 for the identification of each of these experiments and reference 15 for additional summary commentary on the SDMF Experiments 1, 2, 3 and 4.

6.1.3.3 The SDMF-1 Startup Experiment, with dosimetry in dummy surveillance capsules in place of the instrumented ones, was performed prior to the metallurgical irradiation to determine accurately the irradiation times needed to reach the target fluence. A set of calculations was performed to account for 52 different core loadings and their associated irradiation histories. Calculations were performed for each of three exposures: two surveillance capsules (SSC-1 and SSC-2) and a pressure vessel capsule. Comparisons of the ORNL-calculated end-of-life dosimeter activities with measurements indicated agreement, generally within 15 % for the first surveillance capsule, 5 % for the second capsule, and 10 % for the three locations (¼ T, ½ T, and ¾ T) in the pressure vessel capsule (**20**).

6.1.3.4 NUREG/CR-3320, ~~Vol.~~Vol 2 (**12**) provides documentation of the SDMF-1 Experiment and the results of dosimetry measurements and studies by the LWR-PV-SDIP participants. The following laboratories participated in radiometric analyses of the dosimeters: HEDL; ORNL; CEN/SCK (Mol); KFA (Julich); Harwell (England - counting for Rolls Royce Assoc. Ltd.); PTB (Federal Republic of Germany); and Petten (Netherlands). NBS (presently known as NIST) Certified Fluence Standards were supplied.

6.1.3.5 The results of the SDMF-1, SDMF-2, and SDMF-3 experiments are primarily based on radiometric sensor measurements. The SDMF-4 experiment provided benchmark referencing data for the full complement of dosimetry sensors (radiometric, solid state track recorders, helium accumulation fluence monitors, and damage monitors) which were under development and testing for PWR and BWR surveillance program applications (**15**). Therefore, the SDMF-4 measured results are particularly appropriate for benchmarking the methodology, nuclear data, and accuracy of derived neutron exposure parameter for surveillance applications.

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6.1.3.6 The later SDMF experiments were specialized geometry experiments to study the effects on dosimeter response caused by placement of the surveillance capsules in the water environment of the reactor downcomer region.

6.1.4 *The NESDIP Benchmark*—The NESTOR Shielding and Dosimetry Improvement Program (NESDIP) was started in 1982 (~~**16**~~**16-18**~~, **17**, **18**~~). NESDIP experiments have been divided into three phases, the third of which is simulation of actual commercial LWR cavity configurations in accord with cooperative interests of the NRC and US utilities and reactor vendors (**19**). The emphasis was on an internal study of the accuracy of transport theory methods, S_N and Monte Carlo methods, for predicting neutron penetration and attenuation for the radial shield and cavity region of LWRs.

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6.1.5 *Other Benchmarks*—Other benchmarks exist which may be used for comparisons for special geometries or for other reactor types. These benchmarks include those described in the benchmark referencing standard (Guide E2005). Additional benchmarks that may be applicable include the DOMPAC benchmark (**21, 22**), the OSIRIS benchmark (**23, 24**), the LR-0/VVER440 benchmark (**25, 26**), the TAPIRO source reactor benchmark (**27**), the KORPUS benchmark (**28**), the concrete benchmark (**29**), and the KUCA/KUR/UTR-KINKI benchmarks (**30, 31**~~)~~**).**

6.2 *Benchmarks at Power Reactor Facilities:*

6.2.1 In parallel with the PV mockup experiments were efforts in the Arkansas Power and Light Reactor ANO-1 to initiate ex-vessel cavity dosimetry as a supplement or replacement for vessel monitoring dosimetry in the surveillance capsule (**32**). This led to benchmarking, by LWR-PV-SDIP of cavity dosimetry in special experiments in the H.B. Robinson nuclear power reactor (**33, 34**) as well as a number of others (**35**).

6.2.2 The H.B. Robinson measurements have the advantage that simultaneous dosimetry results were obtained from a dummy surveillance capsule and from ex-vessel capsules irradiated during a single reactor cycle. Thus direct comparisons may be made with calculations on both sides of the reactor vessel.

6.3 *Specific Plant Measurements:*

6.3.1 The use of actual plant measurements to obtain fluence results is covered in Practice E1006. However, these results are seen in the benchmark context as part of the overall benchmarking process to obtain the evaluated plant specific fluence.

6.3.1.1 A large body of data, including both surveillance capsule and ex-vessel dosimetry measurements, has been obtained.

Evaluation of these data in a systematic fashion has indicated excellent self-consistency among plants of the same types (~~36~~36-38~~, 37, 38~~). ~~This~~ Further, the data indicates that ~~the~~ changes in neutron source with changes in fuel loading are being correctly handled, and that calculational bias is most probably due to systematic (not random) effects. Use of the data bases of surveillance dosimetry results can provide additional confidence in treatment of any results that appear to lie outside the normal error tolerance.

6.4 *Shielding Integral Benchmark Archive and Database (SINBAD):*

6.4.1 SINBAD (**39**) is an electronic database that includes many of the benchmarks mentioned above as well as accelerator and fusion shielding benchmarks. It represents an ongoing international effort between the OECD Nuclear Energy Agency (NEA) and ORNL Radiation Safety Information Computational Center (RSICC). Invaluable contributions to the compilation, validation, and review of the database are received from many international nuclear data experts.

7. Applications of Benchmark Results

7.1 *Comparisons of Calculations and Measurements*—Three methods can be used for comparisons of calculations and measurements. These are described in the following sections.

7.1.1 The first method is to calculate the measured dosimeter ~~disintegrations per second.~~ detector response. Use of this method involves calculations of the ~~reactions per second~~ responses from the calculated fluence rate and subsequent derivation of the ~~activity~~ total response using the irradiation history. This method enables various segments of the irradiation to be summed to get the total activity. The disadvantage of this method is that experimental results from different irradiations cannot be directly compared without using the transport calculated results. An overall comparison of calculation and experiment can be made by a suitably weighted average of the calculation/measurement (C/M) ratios.

7.1.2 The second method is to derive the average full-power reaction rate for each dosimeter using the irradiation history. These "saturated" reaction rates are independent of the length of irradiation or the time at less than full power. It is important to use a history that represents the variation of the actual fluence rate ~~of activation~~ at the dosimeter location and not just the reactor power history. Comparisons of calculated and measured reaction rates indicate possible bias in the calculation and a weighted average of the results may be used as in the method in ~~6.1.1~~7.1.1.

7.1.3 The final method is to derive a fluence rate from the average reaction rates at each location. This enables a direct comparison with the calculated fluence results. The fluence-rate may be derived from the measurements using least squares procedures. Several computer codes exist to carry out this process. See Guide E944. The use of the least squares procedures enables relations between the part of the neutron spectrum measured by the dosimeters and the part to be used to evaluate irradiation effects to be included in the weighting, in addition to measurement uncertainties. More extensive use of the least squares method to evaluate fluence is described in ~~6.2.3~~7.2.3.

7.2 *Use of Measurement Comparisons for Determination of Best-Estimate Fluence*—Depending on the confidence in measurements or calculations, several approaches can be used to develop final fluence results.

7.2.1 Once the measurements and calculations are compared, one course of action is to merely use the measurements as a test of the calculational result. The calculation would then be considered adequate if it reproduced the measurements within some tolerance. If the results are outside the tolerance, corrective action would be required. This method, while the simplest in checking methods using both benchmark and plant specific data, does not produce the best estimate result and the uncertainty in the result will be that evaluated for the calculation alone.

7.2.2 The second method is to use the plant specific measurements to renormalize the calculations. Use of this method will normally produce the best result at actual dosimetry measurement locations and at locations suitably close to the measurement locations. The plant specific measurements reflect potentially unknown deviations between actual (as-built) plant parameters and parameters used in the calculations of fluence that cannot be benchmarked in any other way. Translation of the results to locations away from measurement points can be guided by both the plant specific and special irradiation field benchmark comparisons. Fluence results benchmarked in this way will come close to best estimates using more sophisticated methods.

7.2.3 The most sophisticated method for fluence determination is to include both the calculation results and uncertainty and the measurements and uncertainty to get a best estimate result using a least squares procedures. One way to accomplish this is by use of the LEPRICON code (**40**).

7.2.3.1 In the LEPRICON procedure, benchmark experiments are first incorporated into a database of integral dosimetry

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measurements of high quality. These are measurements which, in so far as possible: have been performed in simple geometries amenable to accurate descriptions for calculational purposes; have large sensitivities to only a few differential parameters; and involve integral quantities and parameters which are highly correlated with many of those parameters used in the analyses of experiments performed in the more complex geometries of light water reactors.

7.2.3.2 The benefit of simultaneously combining heavily weighted benchmark results with those from more complicated-geometry experiments into a more self-consistent data base comes about because of the correlations induced by data sharing sensitivities to common parameters.

7.2.3.3 The data required to implement the least-squares adjustment procedure includes measured and calculated values of a dosimeter's response, sensitivities of that response to the more important differential data used in calculations, the standard deviation of each measurement along with correlations between measurements that are being combined (that is the covariances), and the covariances of the differential data among the various parameters.

7.2.3.4 It should be evident that such an undertaking is not an easy task and definition of the covariances may be difficult. For example, it was already mentioned above that the LWR benchmarks may have been used by the cross section evaluators to influence the cross section shape or magnitude; the benchmark data may be included a second time in the unfolding process. However, when a concerted effort is made to accomplish the uncertainty definition in a rigorous and well-documented manner, the result can have a significantly higher degree of certainty. Such evaluations can then be used to estimate uncertainties in similar cases without repeating the entire process.

8. Precision and Bias

NOTE 1—Measurement uncertainty is described by a precision and bias statement in this practice. Another acceptable approach is to use Type A and B uncertainty components (see ISO Guide on the Expression of Uncertainty in Measurement and Ref (**41**). This Type A/B uncertainty specification is now used in International Organization for Standardization (ISO) standards, and this approach can be expected to play a more prominent role in future uncertainty analyses.

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8.1 The benchmarking processes outlined above will serve to indicate the calculational bias and allow uncertainty estimates to be made. Typical calculational (analytic) uncertainty estimates for the fast neutron fluence rate (E > 1 MeV) are ~~15~~ 15 % to 20 % (1σ) (**9**, **11**, ~~**42**~~**42-46**~~, **43**, **44**, **45**, **46**~~) at the inside of the reactor vessel and may be as large as 30 % in the cavity. Using the benchmark results is expected to lower the uncertainty in the fast neutron fluence rate to ~~~10~~ ~10 % to 15 % at most locations in the region that is inside the pressure vessel and covers about 80 % of the active fuel height centered around the fuel mid-plane. The fast neutron fluence rate uncertainty at other locations is expected to be similar, but somewhat larger.

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8.2 Error propagation with integral detectors is complex because such detectors do not measure neutron fluence directly, and because the same measured detector responses from which a neutron fluence is derived are also used to help establish the neutron spectrum required for that fluence derivation.

8.3 The information content of uncertainty statements determines, to a large extent, the worth of the effort. A common deficiency in many statements of uncertainty is that they do not convey all the pertinent information. One pitfall is over simplification, for example, the practice of obliterating all the identifiable components of the uncertainty, by combining them into an overall uncertainty, just for the sake of simplicity.

8.4 Many "measured" dosimetry results are actually derived quantities because the observed raw data must be corrected, by a series of multiplicative adjustment factors, to compensate for other than ideal circumstances during the measurement. It is not always clear after data adjustments have been made and averages taken just how the uncertainties were taken into account. Therefore, special attention should be given to discussion of uncertainty contributions when they are comparable to or larger than the normally considered statistical uncertainties. Furthermore, benchmark procedures owe their effectiveness to strong correlations that can exist between the measurements in the benchmark and study fields. Other correlations can also exist among the measurements in each of those types of fields. It is, therefore, vital to identify those uncertainties that are correlated, between fields, among measurements, and in some cases where it may be ambiguous, those uncertainties which are uncorrelated.

9. Documentation

9.1 The procedures followed to benchmark the calculations should be extensively documented. This must include, as a minimum, the following: a description of the methods used including codes and options selected, a reference to the nuclear data used, a description of the models applied, and a listing of the benchmark data utilized.