NEI

NUCLEAR ENERGY INSTITUTE

PDR
per
W. Jones

April 12, 1994

Rules Review and Directives Branch
Division of Freedom of Information and
Public Services
Office of Administration
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

SUBJECT: Recommended Redraft of DG-1025, "Calculational and Dosimetry Methods for Determining Pressure Vessel Fluence"

On January 28, 1994, NUMARC submitted comments on DG-1025, "Calculational and Dosimetry Methods for Determining Pressure Vessel Fluence." In the cover letter we said the regulatory guide is difficult to understand, lacks continuity on how to apply its guidance and needs to be made more user friendly. In support of this concern, we developed the enclosed redraft of DG-1025. It incorporates our comments, provides additional editorial clarifications, and hopefully clearly reflects what we understood the NRC staff's intent to be on this rather difficult, very technical topic. We urge the staff to give careful consideration to this material in establishing the final guidance document.

In our view, it remains absolutely essential that demonstration of the DG methodology be performed on an actual plant prior to issuance of the final guide. This is necessary to confirm the DG's technical adequacy and the estimate resources required to follow the guidance. The demonstration is also important because as we noted in our original Comment #6, the industry is very concerned with the general concept of using analytical calculational uncertainty analysis to independently estimate the biases and uncertainty in the calculated fluence.

We propose that the NRC staff and industry meet to discuss our original comments and this submittal. Because this issue is extremely technical and has never had a regulatory guide or other industry standard developed that outlined acceptable analytical methods, additional public interactions are warranted.

DF03

1776 I STREET, NW SUITE 400 WASHINGTON, DC 20006-3708 PHONE 202.739.8000 FAX 202.785.4019

RES-1B-1

Kurt Cozens of the NEI staff should be contacted to address any questions concerning this submittal or for establishing an acceptable meeting date.

Sincerely,

William H. Rasin
Vice President & Director
Technical

WHR/KOC/rs
Enclosure

c: Lawrence Shao, NRC
Mike Mayfield, NRC
Al Taboada, NRC

DRAFT REGULATORY GUIDE DG-1025
SUGGESTED RE-WRITE

CALCULATIONAL AND DOSIMETRY METHODS FOR DETERMINING PRESSURE VESSEL NEUTRON FLUENCE

A. INTRODUCTION

The U.S. Nuclear Regulatory Commission (NRC) has promulgated regulations that ensure the structural integrity of the reactor pressure vessel for light-water-cooled power reactors. Specific fracture toughness requirements for normal operation and for anticipated operational occurrences for power reactors are set forth in Appendix G, "Fracture Toughness Requirements," of 10 CFR Part 50, "Domestic Licensing of Production and Utilization Facilities." Additionally, in response to concerns over potential pressurized thermal shock (PTS) events in pressurized water reactors (PWRs), the NRC issued 10 CFR 50.61, "Fracture Toughness Requirements for Protection Against Pressurized Thermal Shock Events." In the situation where the RT_{PTS}screening limits have been exceeded, RG 1.154, "Format and Content of Plant-Specific Pressurized Thermal Shock Safety Analysis Reports for Pressurized Water Reactors" has been provided as an acceptable guide to determine if the reactor pressure vessel is safe to operate.

For PWRs that must satisfy the requirements of Appendix G, 10 CFR 50.61, and RG 1.154, methods for accurately determining the best estimate fast neutron fluence (E > 1.0 MeV) are required in order to estimate the fracture toughness of the pressure vessel materials. Appendix H, "Reactor Vessel Material Surveillance Program Requirements," of 10 CFR Part 50 requires the installation of surveillance capsules, including both material test specimens and neutron dosimetry, in operating power reactors to provide data on material damage correlations as a function of neutron fluence; and, to provide measured neutron dosimetry data for validation of calculations.

The fracture toughness of pressure vessel materials is related to a parameter called the material's "reference temperature for nil-ductility transition," or simply "reference temperature" denoted as RT_{NDT}. The material $RT_{NDT}$is determined from a correlation of the fast neutron fluence (E > 1.0 MeV), material chemistry (concentrations of Cu and Ni), and unirradiated reference temperature. A margin is also included in the RT_{NDT} determination to account for uncertainties in the correlation and input values. In 10 CFR 50.61, evaluation of the reference temperature based on the best estimate of the fast neutron fluence at the end of the license period is required, and, in this instance, the corresponding reference temperature is termed RT_{PTS}.

This guide describes methods and assumptions acceptable to the NRC staff for determining the best estimate fast neutron fluence for use in RT_{NDT} and RT_{PTS} determinations in 10 CFR 50.61, Appendix G, and RG 1.154 for PWRs. Because the best estimate fast neutron fluence is required only for PWRs per 10 CFR 50.61, this guide is focused toward PWR applications. It is understood that the methodology presented here is beyond the accuracy required for BWRs. Therefore, although this guide may be applied to BWRs, it is not expected that BWRs would benefit from such application and this Regulatory Guide is considered optional for BWRs. Cases of unusual plant characteristics or factors not covered in this guide that require different methods and assumptions will be considered on a plant-specific basis.

Compliance with this guide is not a regulatory requirement of the USNRC. However, if a licensee elects to use the methods described in this guide to determine best estimate pressure vessel neutron fluence, implementation of the guide would not be satisfied unless the licensee complies with certain specific provisions identified in the Regulatory Position of the guide. The use of the following terms is explained to clarify compliance with these regulatory positions.

Must -	Necessary provisions if implementation is to be satisfied.
Should -	Provisions that are expected to be complied with unless it is not possible because of specific circumstances (for example, data needed to meet the position requirement are not available).
May -	Provisions that are acceptable and recommended, but are to be applied at the option of the licensee.

This draft regulatory guide contains voluntary information collections that are subject to the Paperwork Reduction Act of 1980 (44 U.S.C. 3501 et seq.) This guide has been submitted to the Office of Management and Budget for review and approval of the paperwork requirements.

The public reporting burden for this collection of information over and above the burden previously required for this activity is estimated to be an average of ___[1] hours per respondent, including the time for reviewing instructions, searching existing data sources, gathering and maintaining the data needed, and completing and reviewing the collection of information. Send comments regarding this burden estimate or any other aspect of this collection of information, including suggestions for further reducing the reporting burden, to the Information and Records Management Branch (MNBB-7741), U.S. Nuclear Regulatory Commission, Washington, DC 20555; and to the Desk Officer, Office of Information and Regulatory Affairs, NEOB-3019, (3150-0011), Office of Management and Budget, Washington, DC 20503.

B. DISCUSSION

The methods and assumptions described in this guide are applicable to the calculation and measurement of vessel fluence for core and vessel geometrical and material configurations that are typical of current PWR and BWR power reactor designs. This guide does not address the determination of surveillance specimen material properties or the correlation between material properties and neutron fluence. The methodology presented is intended as a best-estimate, rather than a bounding or conservative, fluence determination. For example, in the RT_{PTS} correlation called for in 10 CFR 50.61 for PWRs, the best estimate fluence is used to calculate the shift. Uncertainty in the shift prediction (e.g., from uncertainty in the fluence, chemistry factor, or shift correlation) is treated separately in an explicit margin term. While the $E > 1.0$ MeV fluence has been selected as the exposure parameter for use in the RT_{NDT} and RT_{PTS} correlations, the procedures described in this guide are also acceptable for the determination of the neutron spectrum from 0.1 to 15 MeV; and, are generally applicable to the calculation of other exposure units, such as displacements per atom (dpa)

The determination of the best-estimate pressure vessel fluence is based on both calculations and measurements; the fluence prediction is made with a calculation and the measurements are used to qualify the calculational methodology. Because of the importance and the difficulty of these calculations, the method's qualification by comparison to measurement must be made to ensure a reliable and accurate vessel fluence determination. In the qualification procedure, calculation-to-measurement comparisons are used to identify biases in the calculations. When the measurement data are of sufficient quantity and quality (i.e., they represent a statistically significant measurement

[1] The cost of implementation of this Regulatory Guide will be quite sensitive to its final form. The demonstration NUREG report suggested in the industry comments to this guide would serve to demonstrate the actual implementation cost for an acceptable methodology qualification and application.

base), the comparisons to measurement may be used to (1) determine the effect of the various modeling approximations and inherent calculational bias; and, if appropriate, (2) modify the calculations by explicit application of a bias or by model adjustment or both. The prediction of the vessel fluence must be made with a methodology that has been qualified by comparison to measurements in similar geometries. The prediction should account for any significant systematic calculational bias as indicated by the qualification. The methods qualification must include a sensitivity study of the important input and modeling parameters to determine the contribution to the overall calculational uncertainty.[2]

The calculations of the pressure vessel fluence consist of (1) methods qualification and (2) methods application. These steps are discussed in detail in Regulatory Positions 1.1 and 1.2. In Regulatory Position 2, the use of surveillance dosimetry as an in situ verification of the calculations is described. Reporting is discussed in Regulatory Position 3.

As an indication of current practice, selected codes and cross-section libraries are listed in the references; however, it is the responsibility of the licensee to demonstrate their acceptability in a specific application. Additional material related to the determination of pressure vessel fluence and material damage, but considered outside the scope of this guide, is contained in other regulatory guides and ASTM and ANSI/ANS standards.

C. REGULATORY POSITION

1. NEUTRON FLUENCE CALCULATIONAL METHODS QUALIFICATION AND APPLICATION

This guide describes the qualification and application of methodologies acceptable to the staff for determination of the best-estimate neutron fluence experienced by materials comprising the beltline region of Light Water Reactor (LWR) pressure vessels; and for determining the overall uncertainty associated with those best-estimate values.

The methodology for determining the best-estimate reactor vessel fluence to be used in the evaluation of RT_{PTS} or RT_{NDT} must be qualified to predict the vessel fluence

[2] In this guide, the term "uncertainty" is defined as an estimate of potential inaccuracies in a measured or derived quantity based on an explicit evaluation and combination of all sources of error. The total uncertainty value is made up of a random component (sometimes referred to as "precision") and a bias that is a non-random or systematic difference between the fluence estimate and the true value.

to within a standard deviation of 20% or less. In these evaluations, the inclusion of this uncertainty explicitly as an additional margin term is not required. The determination of an uncertainty of 20% or less merely serves to demonstrate the adequacy of the best-estimate fluence.

When performing Probabilistic Risk Assessment (PRA) evaluations such as those described in Regulatory Guide 1.154, an explicit uncertainty term is required in addition to the best-estimate fluence values. In this case, it is beneficial and permissible to use the actual fluence uncertainty determined per this guide.

Section 1.1 provides guidance on how to qualify calculational methods for use in vessel fluence evaluations. Using the descrete ordinates method as an example, an acceptable approach to the application of a qualified method is described in Section 1.2. Informational Figure 1 through 3 provide a flow diagram of the qualification and application procedures.

1.1 Methods Qualification

While adherence to the neutron transport calculation guidelines described in Section 1.2 will generally result in accurate fluence estimates, the overall methodology must be qualified per this section in order to quantify uncertainties, identify any potential biases in the calculations, and provide confidence in the fluence calculations. While the methodology (including computer codes and data libraries) may have been found to be acceptable in previous applications, the qualification ensures that the licensee's implementation of the methodology is valid.

The methods qualification consists of three parts: (1) the analytic sensitivity study, (2) the comparison with benchmark and plant specific measurements, and (3) the estimate of fluence calculational uncertainty. These three phases of the overall qualification procedure are discussed in Sections 1.1.1 through 1.1.3.

1.1.1 Analytic Sensitivity Study

An analytic sensitivity study must be performed to demonstrate the precision of the methodology. This study includes identification of the important sources of uncertainty. For typical fluence calculations, these sources include:

- Nuclear data (cross-sections and fission spectrum)

- Geometry (locations of components and deviations from nominal dimensions)

- Isotopic composition of material (density and composition of coolant water, core barrel, thermal shield, pressure vessel with cladding, and concrete shield)

- Neutron sources (core leakage, space and energy distribution, and burnup dependence)

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

This list is not necessarily exhaustive, other uncertainties that are specific to a particular reactor or a particular calculational method should be considered.[3]

A series of sensitivity calculations in which the calculational model input data and modeling assumptions are varied and the numerical effect on the calculated fluence is determined should be performed to establish the effect of the significant component uncertainties.[4] Estimates of the expected uncertainties in these input parameters must be made and combined with the corresponding fluence sensitivities to determine the expected impact on the total fluence uncertainty (i.e., standard deviation). The independent random uncertainties should be combined in a statistical or root-mean-square fashion, and any systematic errors (or biases) should be combined algebraically, recognizing the sign of each contribution. The component uncertainties should be based on measurement or on the acceptable tolerances included in the design specifications. The sensitivity calculations may be performed in one dimension when the model sensitivity does not involve a detailed two-dimensional representation.

A sample sensitivity analysis, in which the influence of various component uncertainties on the calculated group fluences has been considered, is included in References 1 and 2. Since the uncertainties used in these analyses are common to many pressurized water reactors, the sensitivities from References 1 and 2 (including correlations) may be used as initial uncertainty estimates. These variance estimates

[3] In typical applications, the fluence uncertainty is dominated by a few uncertainty components, such as geometry, that are usually easily identified and substantially simplify the uncertainty analysis.

[4] A typical sensitivity would be ~10-15% decrease in vessel $E > 1.0$ MeV fluence per centimeter increase in vessel inside radius.

should be modified as additional experience is obtained or if the reactor of interest differs substantially from the benchmark reactor. The referenced benchmark sensitivity analysis provides guidance for such modifications.

1.1.2 Comparisons with Benchmark and Plant-Specific Measurements

The fluence calculational methods should be validated against (1) a power reactor benchmark that provides in-vessel surveillance capsule dosimetry or ex-vessel cavity measurements or both, (2) a pressure vessel simulator benchmark that provides measurements at the inner surface and at the T/4 and 3T/4 positions within the vessel, and, optionally, (3) available fluence calculational benchmarks. The results of the validation should include comparisons of reaction rates, fluences, and group fluxes for the locations of interest (References 30, 31)[5]. Any adjustments made to the calculations must be justified and documented.

The comparisons of the calculations to plant-specific measurements and to the benchmarks must be used to estimate the calculational bias and precision. When a bias is applied to the calculation to determine a best-estimate fluence, the justification and basis for the bias must be documented.

1.1.2.1 Operating Plant Measurements.

Comparisons of measurement with calculation must be performed for the specific reactor being analyzed or for reactors of similar design. Plant specific measurements have the advantage of including the as-built materials and geometry and the actual reactor operating conditions. An especially accurate determination of the fluence attenuation through the vessel (e.g., to the T/4 and 3T/4 locations) can be obtained when both in-vessel and cavity dosimetry are available.

Alternatively, several documented sets of fluence measurements for operating power reactors may be used for methods and data qualification.[6] Descriptions of these configurations include three-dimensional geometry and reactor operating conditions; and, in some cases, the measurements include both in-vessel and ex-vessel data.

[5] The reference numbers used in this suggested re-draft of DG-1025 are the same as used in the original version published by the NRC staff for public comment.

[6] Examples of operating reactor benchmark measurements are provided in reference 32.

1.1.2.2 Simulator Benchmark Measurements

Several pressure vessel simulator benchmarks (References 3, 27, 32, 59-61) may be used for methods qualification. These benchmark experiments were carried out in three types of configurations:

1.1.2.2.1 Pressure Vessel Simulator Mockup Experiments

Pressure vessel simulator mockups of the vessel in the vicinity of test reactors (Reference 32) are unique in that they provide benchmarked dosimetry measurements at the inner surface of the vessel and at locations within the vessel wall (e.g., T/4 and 3T/4). These benchmarks are further characterized by relatively simple geometries with generally less uncertainty in region compositions, temperatures, and source distributions than in operating power reactor benchmarks.

1.1.2.2.2 Power Reactor Mockup Experiments

Reactor benchmarks that have been used to evaluate pin-wise power distributions in peripheral fuel assemblies, to investigate three-dimensional effects caused by partial-length shield assemblies, and to validate the modeling of heterogeneities caused by neutron pads attached to the core barrel (References 27, 59).

1.1.2.2.3 Cavity Mockup Experiments

Benchmark measurements involving simulated reactor cavities are described in References 3 and 32. Measurements performed in power reactor cavities are described in References 1 and 2.

1.1.2.3 Calculational Benchmarks.

The two-dimensional vessel fluence problem provided by the NRC (Reference 33) and the one-dimensional shielding calculational benchmarks proposed by the American Nuclear Society Benchmark Committee (Reference 34) may be used for methods qualification. In these benchmarks, the geometry, materials, and the space- and energy-dependent source are fixed by the problem specification. The calculation of these problems provides a test of the cross-sections and other input to the transport solution, such as spatial mesh, quadrature, and convergence criteria. The methods used for these calculational benchmarks must be consistent (to the extent possible) with those used in actual operating power reactor fluence calculations. That is, the same cross-sections, transport techniques, and transport code input parameters that are used in the licensing application must be employed in the calculational benchmarks.

1.1.3 Estimate of Fluence Calculational Uncertainty

The overall uncertainty in the calculated best-estimate fluence must be determined by an appropriate combination of (1) the analytical sensitivity study (Regulatory Position 1.1.1) and (2) the uncertainty estimate based on the comparisons to plant-specific measurements and benchmarks (Regulatory Position 1.1.2). For the determination of RT_{PTS} or RT_{NDT}, the qualified methodology should produce a best estimate fluence with an associated uncertainty less than or equal to 20% (1 sigma). For PTS applications, if the benchmark comparisons indicate an uncertainty greater than 20%, the best-estimate vessel fluence must be modified to account for this larger potential error.

1.2 Methods Application

The discrete ordinates method is generally used in the United States for the calculation of pressure vessel fluence. It is recognized that alternative methods such as monte carlo are available, and analyses performed with these methods will be reviewed on a case by case basis.

In this section, the application of the discrete ordinates methodology to the calculation of pressure vessel fluence is discussed. The section includes a limited discussion of the discrete ordinates code input for guidance. In a plant specific application, the various modeling assumptions and other input parameters used in the calculation must be consistent with the results of the methods qualification performed per Section 1.1.

1.2.1 Transport Calculation[7]

The transport of neutrons from the core to locations of interest in the pressure vessel and reactor cavity should be calculated with a two-dimensional or three-dimensional discrete ordinates transport program (References 22-24). If a two-dimensional transport solution is used, an (r,θ) geometry should be implemented. When

[7] Additional information concerning the application of transport methods to reactor vessel surveillance is provided in ASTM Standard E482-89 (Ref. 21).

appropriate, (r,z) and (r) geometries should also be implemented for determination of the axial variation of the neutron fields. Synthesis of the one- and two-dimensional solutions into a three- dimensional fluence representation is discussed in Regulatory Position 1.2.2. If a three-dimensional transport solution is used, an (r,θ,z) geometry should be implemented.

When modeling the core/vessel geometry in which the rectangular (x,y) geometry of the core boundary and the cylindrical (r) geometry of the vessel are mixed, a more accurate description is provided by the variable (r,θ) mesh option (Reference 23) and may be applied.

1.2.1.1 Geometry Input Data

Detailed geometrical input data should be used to define the transport model used to determine the attenuation of the neutron flux from the core to the locations of interest on the pressure vessel. The geometrical input data include the dimensions and locations of the fuel assemblies, reactor internals (baffle, core support barrel, thermal shield, and neutron pads), pressure vessel (including identification and location of all welds and plates), cladding, and surveillance capsules. For applications including the use of reactor cavity dosimetry, input data should also include the width of the reactor cavity and the material compositions of the support structure and biological shield.

The geometry input data should, to the extent possible, be based on documented and verified as-built plant-specific dimensions. In the absence of plant-specific information, nominal design dimensions and tolerances may be used; and the dimensional tolerances should be accounted for in the determination of the fluence uncertainty (Regulatory Position 1.1.3).

1.2.1.2 Material Input Data

The isotopic compositions of important constituent nuclides within each geometric region should also, to the extent possible, be based on available as-built materials data. In the absence of plant-specific information, "generic" compositions may be used; however, in this case conservative estimates of the variations in the compositions should be investigated as a part of the qualification of the methodology per Section 1.1. The determination of the concentrations of the major isotopes responsible for the fluence attenuation (e.g., iron and water) should be emphasized. The water number densities should be based on plant full-power operating temperatures and pressures, as well as standard steam tables. The data should account for axial and radial variations in water density caused by temperature differences and the presence of in-channel and downcomer voids in the case of BWR's.

1.2.1.3 Cross-Section Input Data

The calculational method to estimate vessel damage fluence must use the neutron cross-sections included in the methods qualification per Section 1.1. Cross-sections should apply over the energy range from ~0.1 MeV to ~15 MeV.

There are several ~50 group libraries available from the Radiation Shielding Information Center (RSIC) that were generated using LWR spectra for group collapsing from "master" libraries (References 8, 9) and may be used for LWR application. These libraries contain microscopic as well as some premixed macroscopic cross-sections for relevant isotopes and materials. Because these prepackaged libraries were designed specifically for LWR pressure vessel fluence calculations, their applicability in any atypical application should be verified prior to use.[8]

As an alternative to the use of available broad group cross-section libraries, input cross-sections may be generated directly from fine group "master" libraries using the procedures described in References 8 and 9.

1.2.1.4 Core Neutron Source Input Data

The determination of the fixed neutron source for the pressure vessel fluence calculations should entail specification of the temporal, spatial, and energy dependence together with the absolute source normalization.

The spatial dependence of the source should be from depletion calculations which represent core operation or from measured data. The depletion calculations can be performed in either two or three dimensions. Three-dimensional calculations will provide the source in both the radial and axial directions. If two-dimensional calculations are used, the axial effects should be from measured data.

The core neutron source should be determined from the power distribution, which varies significantly with fuel burnup, power level, and the fuel management scheme. The detailed state-point dependence should be accounted for (References 10,11); if this is not

---

[8] Libraries based on the latest version of the Evaluated Nuclear Data File (ENDF/B-VI) are now available. These data have been thoroughly reviewed and tested relative to experimental benchmarks and will be used by the staff as a standard of comparison for results obtained with other cross-section sets.

feasible, an approximate averaging over the operating power distribution is acceptable and may be obtained by (1) averaging representative power distributions within the cycle or (2) assuming the cycle-average assembly power distribution is well approximated by the accumulated burnup distribution at the end of the cycle. If a representative average power distribution is used, care should be taken to assure that enough state points through the cycle are examined to provide an accurate average. At a minimum, it is recommended that one or more MOC (middle of cycle) state points be included.

A best-estimate power distribution may be used for reactor vessel neutron damage fluence calculations. However, this best-estimate power distribution must be updated if changes in core loadings, surveillance measurements, or other information indicate a significant change in projected fluence values. This updating may be avoided by using a conservative "generic" average power distribution, provided no measured distribution yields higher power levels for the important peripheral assemblies.

The peripheral assemblies, which contribute the most to the vessel fluence, have strong radial power (source) gradients. These gradients should be accounted for to prevent an overprediction of the fluence. The pin-wise source distribution should be used for best-estimate calculations and the peripheral assembly pin-wise source data should be obtained from the two- or three- dimensional depletion calculations. Where possible as-built rather than design depletion calculations should be used. The pin-wise source distribution should represent the absolute power distribution in the assembly.

The energy dependence of the source (i.e., the spectrum) and the normalization of the source to the number of neutrons per megawatt must account for the fact that there is a change in the overall fission energy spectra due to the variations of isotopic composition of the fuel as a result of burn-in of transuranics such as Pu. These effects tend to increase the fast neutron source per megawatt of power for high-burnup assemblies. The variations in these physics parameters (fission spectra, number of neutrons per fission, and energy release per fission) with fuel burnup may be obtained from standard lattice physics depletion calculations (References 13,14). This effect should also be accounted for in plants that have adopted the PWR low-leakage refueling schemes in which once, twice, or thrice burned fuel is located in the high importance peripheral locations (References 15-18). The harder spectrum in the BWR fuel regions having a high void fraction will have a similar effect on the isotopic fission fractions and on the neutron source normalization and spectrum.

The horizontal core geometry may be described using an (r,θ) representation of the nominal plane. A planar octant representation is acceptable for the octant symmetric fuel loading patterns typically employed in LWR's. Fuel loading patterns that are not octant symmetric may be represented in octant geometry using the octant having the highest fluence. To accurately represent the important peripheral assembly geometry, a θ-mesh of at least 40 angular intervals must be applied. Generally, a model consisting of 40-80 angular intervals provides sufficient detail in the geometry. The r,θ representation should reproduce the true physical assembly area to within ~0.5% and the pin-wise source gradients to within ~10%. The assignment of the (x,y) pin-wise powers to the individual (r,θ) mesh intervals should be made on a fractional area or equivalent basis (References 19, 20). Reference 20 is particularly useful if the radial mesh is a function of θ.

The overall source normalization should be performed with respect to the (r,θ) source so that differences between the core area in the (r,θ) representation and the true core area do not bias the fluence predictions.

1.2.1.5 Transport Code Input Parameters

Determination of the three-dimensional fluence at the vessel using (r,Z) and (r) geometry calculations may also be appropriate. If these calculations are used to provide an axial correction factor, the source specification may be less stringent if consistent sources are used.

The azimuthal (θ) mesh must provide an accurate representation of the spatial distribution of the material compositions and neutron source. The radial mesh in the core region should be ~2 intervals per inch for peripheral assemblies, and may be much coarser for assemblies more than approximately two assembly pitches removed from the core-reflector interface. In excore regions, a spatial mesh that ensures the flux in any group changes by less than a factor of ~2 between adjacent intervals should be applied. Generally, a radial mesh of ~3 intervals per inch in water and ~1.5 intervals per inch in steel provides an adequate representation. Because of the relatively weak axial variation of the fluence, a coarse axial mesh of ~0.5 intervals per inch may be used except near material and source interfaces, where flux gradients can be large.

An S_8 fully symmetric angular quadrature should be used as a minimum for determining the fluence at the vessel. However, for applications in the reactor cavity a higher order quadrature may be needed, depending on the width of the cavity and the axial location at which the fluence is being calculated.

Where computer storage limitations prevent the implementation of these mesh densities in a single run, the calculation should be performed in two or more "bootstrap" steps rather than compromise the spatial mesh or quadrature (the number of energy groups used usually does not affect the storage limitations, only the execution time). In the "bootstrap" approach, the problem volume is sub-divided into a series of overlapping regions. In a two-step calculation, for example, a transport calculation is performed for the cylinder defined by $0 < r < R'$ with a fictitious vacuum boundary condition applied at R'. From this initial calculation, a boundary source is determined at the radius $R'' = R' - \Delta$ and is subsequently applied as the left boundary condition for a second transport calculation for the cylinder defined as $R'' < r < R$, where R represents the true outer boundary of the problem. The adequacy of the overlap region (Δ) must be tested (e.g., by decreasing the inner radius of the outer region) to ensure that the use of the fictious vacuum boundary condition at R' has not unduly affected the boundary source at R" or the results at the vessel.

A point-wise flux convergence criterion of < 0.001 should be used; and a sufficient number of iterations should be allowed within each group to ensure convergence. The adequacy of the spatial mesh and angular quadrature, as well as the convergence, must be demonstrated by tightening the numerics until the resulting changes are negligible. In discrete ordinates codes, the spatial mesh and angular quadrature should be refined simultaneously. In many cases, these evaluations can be adequately performed with a one-dimensional model.

In performing calculations of surveillance capsule fluence it should be noted that the capsule fluence is extremely sensitive to the geometric representation of the capsule geometry and internal water region (if applicable), and the adequacy of the capsule modeling and mesh spacing must be demonstrated using sensitivity calculations per section 1.1.1. In addition, the capsule fluence and spectra are sensitive to the radial location of the capsule and its proximity to material interfaces and these should be represented accurately.

The transport calculations may be performed in either the forward or adjoint modes. When several transport calculations are needed for a specific geometry, assembly importance factors may be recalculated by either performing calculations with a unit source specified in the assembly of interest or by performing adjoint calculations. The adjoint fluxes are used to determine the fluence at a specified (field) location, while the forward fluxes from the unit source calculations determine the fluence at all locations in the problem. Once calculated, these factors contain the required information from the transport solution, and by weighting the assembly importance factors with the source distribution of interest, the vessel (or capsule) fluence may be determined without additional transport calculations, assuming the in-vessel geometry and material remain the same.

When fluence reduction schemes have introduced strong axial or azimuthal heterogeneities into the source (e.g., an axially zoned replacement of fuel with stainless steel for localized fluence reduction), a finer spatial mesh and tighter convergence criteria may be appropriate to ensure an accurate solution. These schemes may also entail a three-dimensional flux calculation (Regulatory Position 1.2.2).

1.2.2 Synthesis of the Three-Dimensional Fluence

When three-dimensional calculations are not performed, a 3D fluence representation may be constructed by synthesizing calculations of lower dimensions using the expression

$$\phi_g(r,\theta,z) = \phi_g(r,\theta)*L_g(r,z) \qquad \text{(Equation 1)}$$

where $\phi_g(r,\theta)$ is the group-g transport solution in (r,θ) geometry for a representative plane and $L_g(r,z)$ is a group-dependent axial shape factor. Two simple methods available for determining $L_g(r,z)$ are defined by the expressions

$$L_g(r,z) = P(z) \qquad \text{(Equation 2)}$$

where P(z) is the peripheral assembly axial power distribution, and

$$L_g(r,z) = [\phi_g(r,z)]/[\phi_g(r)] \qquad \text{(Equation 3)}$$

where $\phi_g(r)$ and $\phi_g(r,z)$ are one- and two- dimensional group-g flux solutions, respectively, for a cylindrical representation of the geometry that preserves the important axial source and attenuation characteristics (Reference 26). The (r,z) plane should correspond to the azimuthal location of interest (e.g., peak vessel fluence or dosimetry locations) or a conservatively θ-averaged geometry. The source per unit height for both the (r,θ) and (r) models should be identical, and the true axial source density should be used in the (r,z) model.

Equation 2 is applicable when (a) the axial source distribution for all important peripheral assemblies is approximately the same or is bounded by a conservative axial power shape and (b) the attenuation characteristics do not vary axially over the region of interest. In addition, since the axial fluence distribution tends to flatten as it propagates from the core through the pressure vessel, this approximation will tend to over predict axial fluence maxima and under predict minima. This nonconservative underprediction could be large adjacent to the top and bottom of the core zone, as well as if minima are strongly localized as occurs in some fluence reduction schemes.

Equation 3 is applicable when the axial source distribution and attenuation characteristics vary radically, but do not vary significantly in the azimuthal (θ) direction within a given radial annulus. For example, this approximation is not appropriate when strong axial fuel enrichment variations are present only in selected peripheral assemblies.

In summary, an (r,θ) geometry fluence calculation and a knowledge of the peripheral assembly axial power distribution are needed when using Equation 2. It will result in fluence overpredictions near the midplane at relatively large distances from the core, e.g., in the cavity, and underpredictions at axial locations beyond the beltline at relatively large radial distances from the core. Conservatisms may be included in the latter case by using the peak axial power for all elevations.

Both radial and axial fluence calculations are needed when using Equation 3; thus, it is generally more accurate in preserving the integral properties of the three-dimensional fluence. Both Equations 2 and 3 assume separability between axial and azimuthal fluence calculations, which, in general, is only approximately true.

When these simple synthesis techniques are not applicable, multichannel synthesis methods may be used. In the multichannel synthesis calculation, the fluence is represented as the sum

$$\phi(r,\theta,z) = \sum_{i=1}^{N} \frac{a_i \phi_{gi}(r,\theta)\phi_{gi}(r,z)}{\phi_{gi}(r)} \qquad \text{(Equation 4)}$$

where the ϕ_{gi} are basis flux solutions, typically representing specific regions of the core/vessel geometry, and the weighting coefficients a_i are determined to provide an optimum prediction of the vessel fluence. It should be emphasized, however, that the accuracy of this method is sensitive to the selection of the basis functions, especially at region interfaces; and three-dimensional calculations should be considered where strong axial or azimuthal heterogeneities exist. This synthesis technique has been applied to an experimental benchmark and the results have been reported in Reference 27.

1.2.3 Cavity Fluence Calculations

Fluence calculations in the reactor cavity are relied on to analyze dosimetry located external to the reactor vessel wall. The comparison of the calculations to these measurements ensures that the best estimate of the fluence is appropriate for vessel embrittlement evaluations. The calculation of the neutron transport in the cavity region is strongly influenced by the attenuation in the iron of the 5.5 - 10.0 inch thick pressure vessel wall; and, for locations above and below the central region of the reactor core, by potential neutron streaming effects in the low density materials (air and vessel insulation).

Because of the increased sensitivity to the transport in iron, the qualification of cross-section sets used for calculations in the reactor cavity must include comparisons with measurements obtained in actual power reactor cavities or in simulated cavity environments provided by benchmark facilities. Likewise, when off-beltline locations are being analyzed (Note: Streaming is possible in the beltline region), sensitivity studies on the angular quadrature must be performed to assure that the angular definition is sufficient to compute the streaming component at the locations of interest. In addition to the input cross-sections and angular quadrature, the cavity fluence calculation is also sensitive to both the material compositions and the local geometry of the primary biological shield (e.g., the presence of detector wells). The shield composition and local geometry must be represented as accurately as possible. Any modeling approximations must be verified with analytical sensitivity studies and benchmarks.

1.2.4 Comparison of Best Estimate Fluence with Plant Specific Measurements

For plant specific applications, the best estimate fluence calculation must be compared with all available plant specific measurements from either the surveillance capsule or reactor cavity locations. The measured fluence is expected to agree with the calculated fluence with an accuracy consistent with the measurement and calculation uncertainties, and generally to less than or equal to 20% for in-vessel and less than or equal to 30% for cavity dosimetry. If this is not the case after checking and eliminating errors and systematic bias, it must be suspected that an unknown error or bias is present and further information will be required to identify the source of this error or bias.

2. NEUTRON FLUENCE MEASUREMENT METHODS

Dosimetry measurements provide independent values of specific activities or isotopic production rates that must be used to benchmark calculations. The dosimetry measurements may be used with spectrum averaged cross-sections to estimate the measured neutron fluence. The measurement predictions are obtained from the response of passive integral detectors placed in surveillance capsules and, more recently, in the ex-vessel cavity. Procedures for performing these measurements to obtain accurate data with a complete uncertainty assessment are described in Sections 2.1 through 2.5. In addition, standard neutron field validation and sites for placing updated dosimetry are described.

2.1 Measurement Procedures

Measurement methods used in power reactor dosimetry include passive integral detectors, which typically include activation detectors and solid state track recorders. The most frequently used detectors respond to neutrons with energies above a characteristic reaction threshold. These detectors should be selected with substantial non-overlapping energy regions (i.e., with well separated thresholds) to provide coarse spectrum information as well as an estimate of the neutron fluence.

2.1.1 Specification and Application of Dosimeters

Neutron dosimetry for pressure vessel surveillance may consist of as-built packages of threshold dosimeters placed in surveillance capsules during reactor construction. The selected dosimeter set must provide adequate thresholds for separately benchmarking calculations above 1.0 MeV and 0.1 MeV as noted by ASTM Standards. For example, E705 for Np-237 (Reference 46), E704 for Uranium-238 (Reference 47), E264 for Nickel-58 (Reference 48), E263 for Iron-54 (Reference 49), E526 for Titanium-46 (Reference 50), E523 for Copper-63 (Reference 51), E1297 for Niobium-93 (Reference 62), et cetera.

Application of activation detectors involves measurement elements that must be carefully controlled and documented to establish accurate results and reasonable uncertainty estimates. Where applicable, procedures in ASTM Standards E181 (Reference 45), E844 (Reference 43), and E1005 (Reference 44) and other ASTM procedures devoted to individual radiometric sensors must be used. Specific regulatory positions associated with the dosimetry measurements are indicated in the following subsections.

2.1.1.1 Isotopic Composition.

The dosimeter materials should be pure enough to ensure there is no significant error in the response of the dosimeter from extraneous activities. Cursory specifications of materials regarding impurities are often unreliable. Specifically, fissile residuals in Np-237 and U-238 and minute amounts of cobalt in copper (fractional parts per million) should be determined by mass spectrography or radioactivation analysis.

2.1.1.2 Encapsulation.

The detector capsule design must take into account possible activation interference and neutron spectrum perturbation. Thermal neutron shields that eliminate interference from thermal neutron reactions in some detectors must be designed to accommodate radiation heating and should be placed apart from low energy detectors (see ASTM Standard E844, Reference 43).

2.1.1.3 Isotopic Mass.

Stoichiometry and isotopic analysis should be well documented for dosimeters that are not of pure natural elements.

2.1.1.4 Location.

The location of individual dosimeters must be determined accurately and recorded, because fluence gradients in out-of-core positions are generally severe. Also, the surroundings of a dosimeter (e.g., adjacent dosimeters or material interface) can influence detector response. In the pressure vessel cavity, establishing azimuthal position can be as important as the radial location. Specially designed mounting arrangements, including vertical gradient wires, should be used for cavity dosimetry. Comprehensive and accurate detector location information should be maintained.

2.1.1.5 Solid State Track Recorders.

In addition to activation detectors, integral detectors employing fission reactions make use of solid state track recorders (SSTRs). These sensors directly record fission fragments from a thin fissionable deposit (Reference 52). Advantages of these detectors are wide sensitivity ranges, a permanent measurement record, and convenient application of fission reaction dosimetry in remote and hostile environments. Because the application is new and employs fissionable deposits in the nanogram to picogram range, details of the measurements should be well documented, and standard neutron field calibration prior to application should be performed. ASTM Standard E854 (Reference 53) provides additional information concerning the use of SSTRs.

2.1.2 Detector Response Measurements

In order to provide a means of comparing calculated dosimeter responses with measurements, an appropriate measured response parameter must be documented, such as: the measured specific activity at end-of-irradiation (given in disintegrations per second per nucleus and divided by the fission yield where appropriate), the measured isotopic production (for example, helium atoms per initial atom of material), the measured total reactions (for example, fissions per initial atom of material), and the measured average reaction rate (for example, disintegrations per second per nucleus). Directly measured quantities (such as the measured specific activity at end-of-irradiation) have the advantage of not involving irradiation related corrections that are somewhat uncertain and may be subject to re-evaluation and, therefore, are the preferred quantities for reporting.

When comparing calculations with measurements, corrections must be included for time-history, detector response perturbations, interfering reactions, and, when applicable, burnup and photofission. In order to allow an accurate treatment of time-history effects, the half-life of the dosimeter activation products should be considered (Reference 44). Photofission corrections can vary considerably (from 2-15%) depending upon the location of the dosimetry and the type of reactor. Fission yields should be those specified in the relevant ASTM Standards, the ENDF library, or the validated job library. In situ neutron field perturbations (e.g., by the surveillance capsule and detector encapsulation) must be accounted for if they are not an integral part of the neutron transport calculation.

When documenting measurements, these corrections should be described along with any other effects that have a significant impact on the measurements. This is especially important because pressure vessel surveillance dosimetry often involves comparison of measurements carried out by different organizations over long periods of time.

2.1.3 Uncertainty Estimates

Regulatory position 1.1.2 states that the calculations must be validated by comparison with measured benchmarks. In order to validate the calculations with comparisons to measurement benchmarks, evaluations must be performed to estimate the bias and uncertainty associated with the measured response for each dosimeter type. The bias and uncertainty must be included in the documentation of the measured results.

There are several different methods that are acceptable for estimating the measured biases and uncertainties. Whatever method is used, the specific components used to determine the systematic deviations (biases) and standard deviations (uncertainties) for each dosimeter type should be separately identified. The bias and uncertainty values should be noted as upper, lower, or best estimates that are either added to the measurements, or multiplied by the measurements. Each component of the bias and uncertainty methodology should be described as part of the documentation explaining how the component values are combined to obtain the bias and uncertainty for each dosimeter type. This evaluation of the measurement biases and uncertainties for each dosimeter provides a means of ensuring a reliable benchmark of the calculations.

2.2 Validation in Standard and Reference Neutron Fields

The dosimeter measurements used to benchmark the calculations must meet the requirements of a quality assurance program (required for surveillance measurements by 10 CFR 50). The program must ensure long term measurement consistency and confirm that the measurements are reliable. Inter-laboratory comparisons, such as those carried out under the NRC sponsored LWR Surveillance Dosimetry Improvement Program (LWRSDIP) have been useful in validating the quality assurance programs of various laboratories. These inter-laboratory comparisons have identified problems in some measurement procedures that were previously unrecognized. They have also served to establish consistency among the measurements from various laboratories. It is important that both (a) the random deviations, or the standard deviation in the measurements (uncertainties), and (b) the systematic deviations (biases) be evaluated to ensure reliable measurements.

A primary consideration in the validation of measurements is the calibration of the instrumentation and measurement systems to accurately account for the dosimetry activity levels. Calibration of the instrumentation and measurement systems must be carried out using either suitable reference standards (such as radioactive sources available from

National Institute of Standards and Technology (NIST) or other laboratories) or standards irradiated to known fluences in reference fields (such as the Materials Dosimetry Reference Facility). Either calibration method can determine biases and uncertainties in the measurements from any laboratory. The laboratory uncertainties combined with those of the calibration standard should result in an overall measurement uncertainty that is less than 20% (see References 32, 57, and 61). Furthermore, the measurement uncertainties should be less than the overall uncertainties determined by benchmarking the calculations to the measurements.

2.3 Fluence Determination from Detector Measurements

As stated in Regulatory Position 1.1.2, the calculated fluence values must be validated by comparison with measurement benchmarks. These comparisons serve to validate the calculational methodology by providing a means of determining the bias and uncertainty in the calculated fluences (E > 1.0 MeV). The benchmark comparisons may be performed in several different ways. Two of these are:

(1) Based on the calculated fluence spectrum at the measurement location, calculate the reaction rate for each detector. These calculated reaction rates should be compared with reaction rates derived from the measured activity or total reaction values using an appropriate flux history for the irradiation. This comparison produces calculation/measurement (C/E) ratios for each dosimeter which should be unity within the combined uncertainties of the calculation and measurement.

(2) Calculations may be performed to determine unsaturated activities (i.e., dps/mg at end-of-irradiation), total reactions (as measured by solid state track recorders), or other measured quantities. These calculated values may be directly compared with the respective measured responses to produce C/E values.

Whatever approach is used to determine the bias and uncertainty in the calculated results from the benchmark comparisons, the calculated biases and uncertainties must be documented along with the measurement biases and uncertainties. Based on the comparisons, an average C/E ratio should be determined as a suitably weighted average of the individual C/E values. In determining the weighting for this average, as a minimum the measurement and spectral average cross-section uncertainties should be considered. Estimated uncertainty in the neutron spectrum can optionally be included as an additional contributor to the dosimeter weighting. Known biases in the spectrum calculation or the cross-sections can be corrected if necessary.

Occasionally individual detector measurements give spurious results. If a result is suspect (typically ones which fall more than three standard deviations from the expected value) it may be disregarded or given reduced weight in the average.

The measured fluence is defined as the calculated fluence divided by the average C/E ratio. Other exposure parameters such as dpa are defined similarly, but the C/E ratios may be slightly different if spectral effects are taken into account.

2.4 Sites for Updated Dosimetry

As-built dosimetry in surveillance capsules cannot easily be updated. Furthermore, the dosimetry irradiated with metallurgical specimens is only available at infrequent intervals. However, additional and upgraded dosimetry is important for understanding and following vessel exposures, especially for low-leakage core configurations. The ex-vessel cavity may be used as an alternative site for installing additional improved dosimetry. Recent pressure vessel benchmark experiments (References 3, 55, 56) have demonstrated that the ex-vessel dosimetry can provide useful exposure information within the pressure vessel wall (References 32, 57) and, when placed at appropriate circumferential locations, is a good monitor of low-leakage core strategies.

3. REPORTING

When fluence determinations are required by the regulations, the report shall be per the applicable regulation. If no reporting requirements are provided in the regulation, an acceptable report would, as a minimum, include the following:

3.1 Neutron Fluences and Uncertainties

3.1.1 Fluence Methods

The methods used to calculate the integral and multi-group fluences and fluence rates and associated methods qualification must be documented. A discussion of any deviations from the procedures provided in this regulatory guide must be included. The source of the cross-section data, the numerical methods (e.g., quadrature, mesh, and convergence criteria), and the treatment of special effects (e.g., fuel burnup, axial effects, and pin power distributions) should be described in detail. This general methods documentation may be provided in a single topical report that may be referenced for plant specific submittals. Any modifications to the methodology described in the topical report must also be discussed in individual plant specific submittals.

3.1.2 Calculational Adjustments

The value and basis for any adjustments to the calculated vessel fluences based on comparisons to plant specific measurement should be described in detail.

3.1.3 Integral Fluences

The best estimate (E > 1.0 Mev), and (E > 0.1 MeV) integral fluences and fluence rates together with the uncertainties at the vessel inner wall locations must be reported. The uncertainties for each measurement and vessel inner wall location must also be reported. The results of the measurement validation should also be described. If thermal neutron fluence rate measurements have been performed, these should be reported together with the uncertainty and thermal cover material.

3.2 Specific Activities and Average Reaction Rates

The specific activities at the end-of-irradiation and the measured average reaction rates should be reported together with the associated uncertainty tables and the power time history. For each dosimeter, provide the reaction type, dosimeter material and form (wire, foil, etc.), weight-percent and isotopic-percent of the target material, fission yields, and half-lives. The corrections for the detector response perturbations, interfering reactions, and photofission should also be described. The calculated reaction rates corresponding to these measurements should be reported together with the C/E ratios and a composite uncertainty derived from the measurement and calculation uncertainties.

D. IMPLEMENTATION

The purpose of this section is to provide information to applicants and licensees regarding the NRC staff's plans for using this regulatory guide.

This draft regulatory guide has been released to encourage public participation in its development. Except in those cases in which an applicant proposes an acceptable alternative method for complying with specified portions of the Commission's regulations, the methods to be described in the active guide reflecting public comments will be used in the evaluation of applications for new licenses and for evaluating compliance with 10 CFR 50.60 and 50.61.

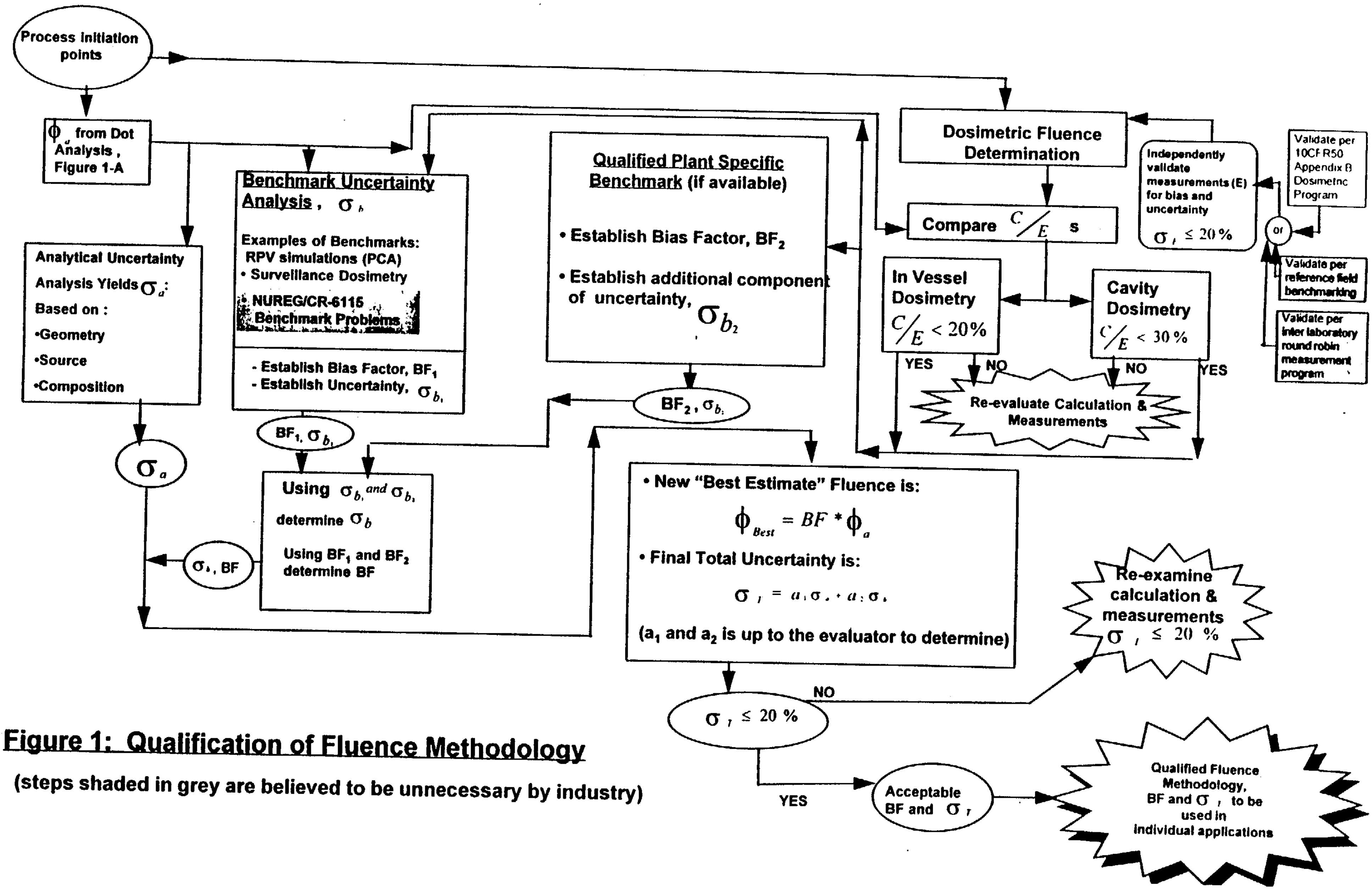

Figure 1: Qualification of Fluence Methodology

(steps shaded in grey are believed to be unnecessary by industry)

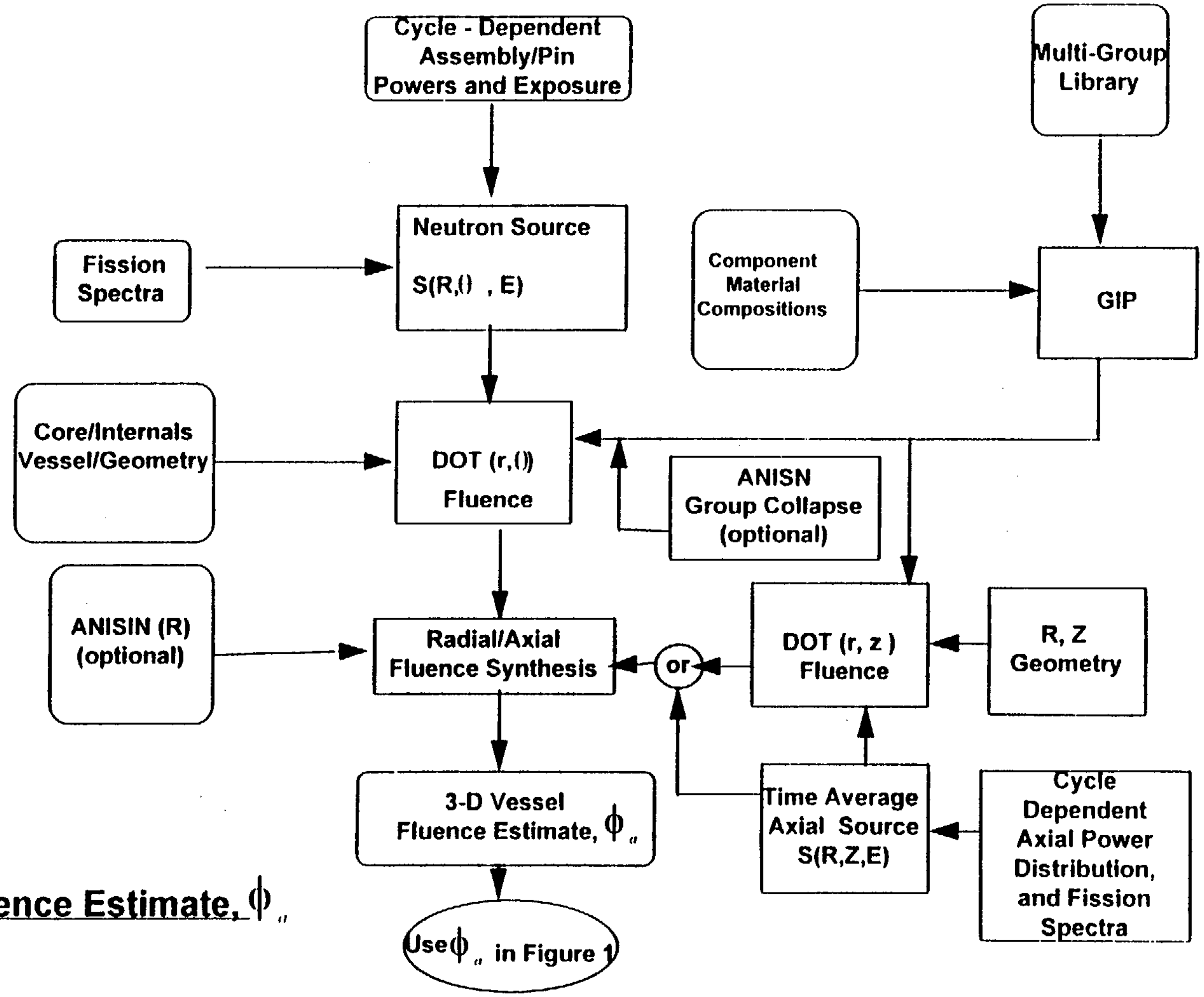

Figure 1-A: Fluence Estimate, $\phi_{\prime\prime}$

Figure 2: Use of Qualified Fluence Methodology in Specific Application

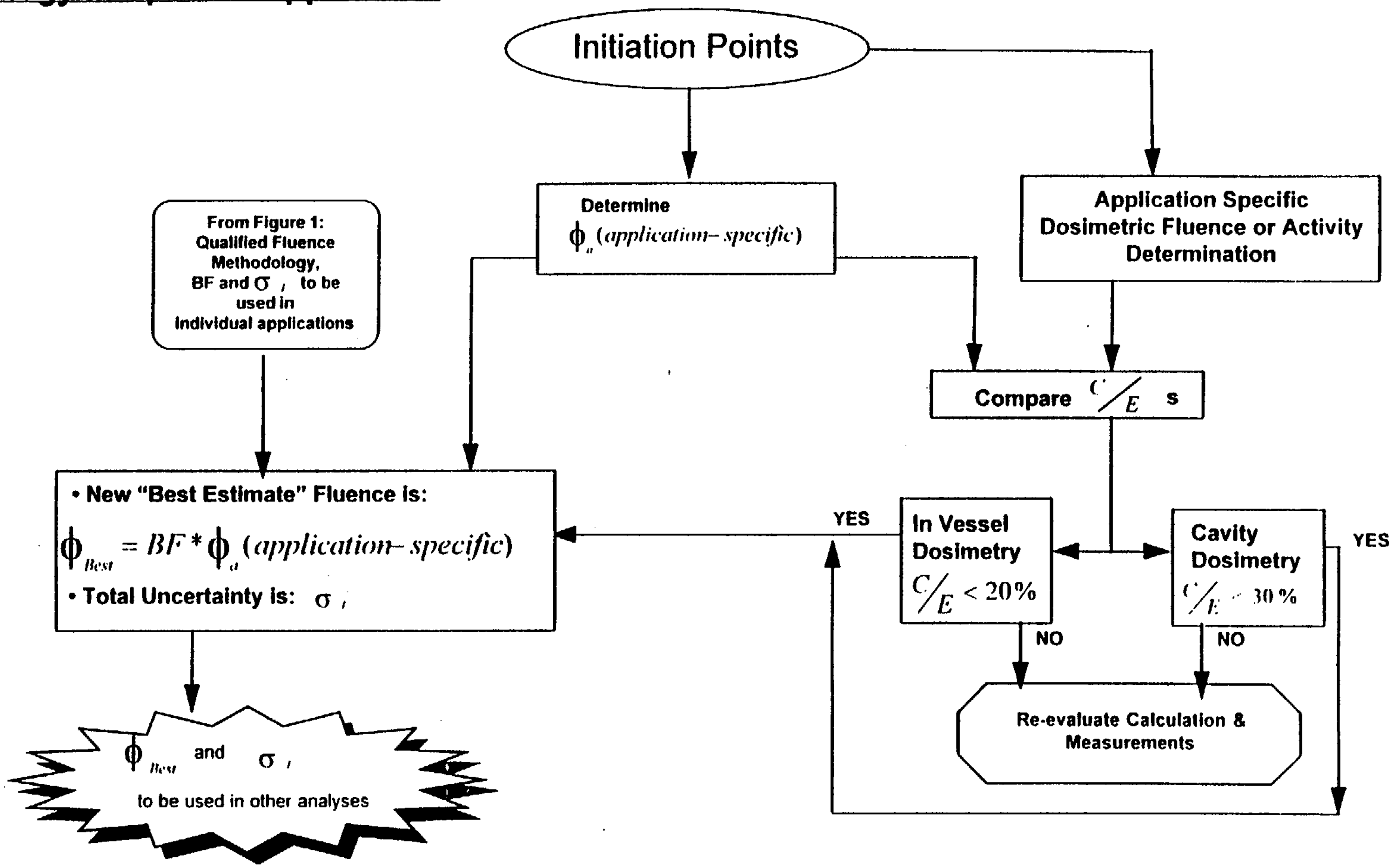

Figure 3: Use of qualified fluence methodology in specific application in cases where there are known plant specific biases that are not present in the benchmarks