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CALCULATIONAL AND DOSIMETRY METHODS FOR DETERMINING PRESSURE VESSEL NEUTRON FLUENCE

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A. INTRODUCTION

This regulatory guide has been developed to provide state-of-the-art calculations and measurement procedures that are acceptable to the NRC staff for determining pressure vessel fluence.

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. Chapter 15, "Accident Analysis," of the Standard Review Plan (NUREG-0800, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants") assumes the pressure vessel does not fail. 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." The requirements of Appendix G are imposed by 10 CFR 50.60. 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."

To satisfy the requirements of both Appendix G and 10 CFR 50.61, methods for determining the fast neutron fluence (E > 1 MeV) are necessary to estimate the fracture toughness of the pressure vessel materials. Appendix H, "Reactor Vessel Material Surveillance Program Requirements," to 10 CFR Part 50 requires the installation of surveillance capsules, including material test specimens and flux dosimeters, to provide data for material damage correlations as a function of fluence.

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, and is denoted as RT_{NDT}. The RT_{NDT} is defined in Revision 2 of Regulatory Guide 1.99, "Radiation Embrittlement of Reactor Vessel Materials," by a correlation of the fluence (E > 1 MeV), material chemistry (concentrations of Cu and Ni), initial reference temperature, and margin 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 the corresponding reference temperature is termed RT_{PTS}.

This guide is intended to ensure the accuracy and reliability of the fluence determination required by General Design Criteria (GDC) 14, 30, and 31 of Appendix A, "General Design Criteria for Nuclear Power Plants," to 10 CFR Part 50. The guide describes methods and assumptions acceptable to the NRC staff for determining the pressure vessel neutron fluence. These methods are directly applicable to the determination of RT_{NDT} and RT_{PTS}. The licensee may propose alternative methods. Alternative methods will be considered on a plant-specific basis, especially in cases involving unusual plant characteristics or factors that require different methods and assumptions. It is recognized that, when the embrittlement of a reactor vessel material is not significant and there is a large margin to the RT_{NDT} limits, more approximate methods for determining the fluence may be appropriate (Reference 1).

Compliance with this guide is not a regulatory requirement of the USNRC. However, if a licensee elects to use this guide to determine 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 are not available).
May -	Provisions that are acceptable and recommended, but are to be applied at the option of the licensee.

The information collections contained in this regulatory guide are covered by the requirements of 10 CFR Part 50, which were approved by the Office of Management and Budget, approval number 3150-0011. If a means used to impose an information collection does not display a currently valid OMB control number, the NRC may not conduct or sponsor, and a person is not required to respond to, the information collection.

B. DISCUSSION

The methods and assumptions described in this guide are for the calculation and measurement of vessel fluence for core and vessel geometrical and material configurations that are typical of current PWR and boiling water reactor (BWR) 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, the best-estimate fluence is used to calculate the shift in RT_{PTS}. Uncertainty in the shift prediction (e.g., from uncertainty in the fluence, chemistry factor, or shift correlation) has been included separately in an explicit margin term. While the $E > 1$ MeV fluence has been selected as the exposure parameter for the RT_{NDT} and RT_{PTS} correlations, the procedures described in this guide determine the damage fluence spectrum (from 0.1 to 15 MeV) and are applicable to other exposure units, such as iron displacements per atom - dpa (Ref. 2).

The determination of the 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 methods must be qualified by comparison to measurements to ensure a reliable and accurate vessel fluence determination. In this qualification, measurement-to-calculation comparisons are used to identify biases (i.e., systematic errors) in the calculations and to provide reliable estimates of the fluence uncertainties.[1] When the measurement data are of sufficient quality and quantity that they allow a reliable estimate of the calculational bias (i.e., they represent a statistically significant measurement data base), the comparisons to measurement may be used to

[1] For a discussion of the terms bias and uncertainty as used in this guide, see Sections 3.2 and 3.3 of Reference 3.

(1) determine the effect of the various modeling approximations and any calculational bias and, if appropriate, (2) modify the calculations by applying a correction to account for bias or by model adjustment or both. Typically, plant-specific data alone are not sufficient to determine a calculational bias for an individual plant. As an additional qualification, the sensitivity of the calculation to the important input and modeling parameters must be determined and combined with the uncertainties of the input and modeling parameters to provide an independent estimate of the overall calculational uncertainty.

The prediction of the vessel fluence must be made by an "absolute" fluence calculation in which the transport of the neutron flux is calculated from the core out to the vessel and cavity, rather than a simple spatial extrapolation of the fluence measurements.

The calculations of the pressure vessel fluence consist of the following steps: (1) determination of the geometrical and material input data, (2) determination of the core neutron source, (3) propagation of the neutron fluence from the core to the vessel and into the cavity, and (4) qualification of the calculational procedure. These steps are discussed in detail in Regulatory Positions 1.1 through 1.4. In Regulatory Position 2, the use of surveillance dosimetry as an in situ verification for the calculations is described. Reporting is discussed in Regulatory Position 3. The major regulatory positions are summarized in Table 1, “Summary of Regulatory Positions on Calculation and Dosimetry.”

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.

C. REGULATORY POSITION

1. NEUTRON FLUENCE CALCULATIONAL METHODS

This guide describes the application and qualification of a methodology acceptable to the NRC staff for determining the best-estimate neutron fluence experienced by materials in the beltline region of light water reactor (LWR) pressure vessels, as well as for determining the overall uncertainty associated with those best-estimate values. Both the deterministic discrete ordinates method and the statistical Monte Carlo transport method are considered acceptable for vessel fluence determination and are described in Regulatory Position 1.3.

Appendix B, "Quality Assurance Criteria for Nuclear Power Plants and Fuel Reprocessing Plants," to 10 CFR Part 50 requires that this methodology be properly qualified. Qualification includes determination of the uncertainty in the reactor vessel fluence as described in Regulatory Position 1.4. The uncertainty of the fluence must be 20% (1 σ) or less when the fluence is used to determine RT_{PTS} and RT_{NDT} for complying with 10 CFR 50.61 and Revision 2 of Regulatory Guide 1.99, "Radiation Embrittlement of Reactor Vessel Materials," respectively. It should be recognized that this 20% uncertainty value has been included in the margin term for the RT_{PTS}. Uncertainty of fluence for other applications should be determined using Regulatory Position 1.4 and included as an uncertainty allowance in the fluence estimate, as appropriate for the specific application. For example, when performing probabilistic risk assessment (PRA) evaluations such as those described in Regulatory Guide 1.154, "Format and Content of Plant-Specific Pressurized Thermal Shock Safety Analysis Reports for Pressurized Water Reactors," an explicit uncertainty term is required in addition to the best-estimate fluence values. In this case, it is acceptable to use the fluence uncertainty determined per the Regulatory Position 1.4 of this guide.

An overview of the calculational methods described in Regulatory Positions 1.1 through 1.3 is provided in Figures 1 and 2. The procedures for qualifying these methods and determining the pressure vessel fluence are summarized in Figures 3 and 4.

1.1 Input Data

1.1.1 Materials and Geometry

Detailed material and geometrical input data should be used to define the physical characteristics that determine the attenuation of the neutron flux from the core to the locations of interest on the pressure vessel. These data include material compositions, regional temperatures, and geometry of the pressure vessel, core, and internals. The geometrical input data include the dimensions and locations of the fuel assemblies, reactor internals (baffle, shroud, former plates, core support barrel, thermal shield, and neutron pads), the pressure vessel (including identification and location of all welds and plates) and cladding, and surveillance capsules. For cavity dosimetry, input data should also include the width of the reactor cavity and the material compositions of the support structure and concrete (biological) shielding, including water content, rebar, and steel. The data should, to the extent possible, be based on documented and verified as-built dimensions and plant-specific materials. The isotopic compositions of important constituent nuclides within each region should be based on as-built materials data. In the absence of plant-specific information, nominal compositions and design dimensions may be used; however, in this case conservative estimates of the variations in the compositions and dimensions should be made and accounted for in the determination of the fluence uncertainty (Regulatory Position 1.4.1). 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 in the core and inside the core barrel, as well as the presence of in-channel and downcomer voids in the case of BWRs. The in-channel and bypass water and fuel channel may be combined to determine a homogenized material region. This approximation has been evaluated in Reference 4 and results in less than a ~3% additional uncertainty in the vessel fluence.

1.1.2 Cross-Sections

The calculational method to estimate vessel fluence should use the neutron cross-sections over the energy range from ~0.1 MeV to ~15 MeV and should apply the latest version of the Evaluated Nuclear Data File (currently ENDF/B-VI). These data have been thoroughly reviewed and tested relative to experimental benchmarks.[2] Cross-section sets based on earlier or equivalent

[2]It should be noted that in many applications the ENDF/B-IV and the first three MODs of the ENDF/B-V iron cross-sections result in as much as ~20% underprediction of the vessel inner-wall fluence and ~35% underprediction of the cavity fluence (Refs. 5-7). Updated ENDF/B-VI iron cross-section data (Ref. 8) have been demonstrated to provide a more accurate determination of the flux attenuation through iron (Refs. 5, 6) and are strongly recommended. These new iron data are included in ENDF/B-VI.

nuclear-data sets that have been thoroughly benchmarked for a specific application may be used for that specific application.[3] However, when the evaluated cross-section data change, the effect of these changes on the licensee-specific methodology must be evaluated and the fluence estimates updated when the effects are significant (see reporting requirements in Regulatory Position 3).

1.1.2.1 Multigroup Libraries. Transport theory codes that are used to determine the neutron fluence, which employ a multigroup approximation, require cross-section libraries in which the basic data contained within the ENDF files has been preprocessed into a multigroup structure. The development of a multigroup library should consider the adequacy of the group structure, the energy dependence of the flux used to average the cross-sections over the individual groups, and the order of the Legendre expansion of the scattering cross-section. Sufficient details of the energy and angular dependence of the differential cross-sections (e.g., the minima in the iron total cross-section) should be included to preserve the accuracy in attenuation characteristics.

1.1.2.2 Constructing a Multigroup Library. The construction of the multigroup library involves the selection of a problem-independent, fine-multigroup, master library containing data for all required isotopes. This master library should include a sufficiently large number of groups ($\gtrsim$100) that differences between the shape of the assumed flux spectrum and the true flux have a negligible effect on the multigroup data. This library typically includes ~50-100 energy groups above ~0.1 MeV. In addition, a minimum of a P-3 Legendre expansion of the scattering cross-section must be used for typical LWR configurations (Ref. 10). Several libraries satisfying these provisions are available from RSICC, the Oak Ridge National Laboratory Radiation Safety Information Computational Center (Refs. 11-13).

The number of groups may be reduced, with little loss in accuracy, by collapsing the data in the master library over spectra that more closely approximate the true spectra. The vessel fluence calculations should be performed with a job library that has been determined by collapsing the master library. This reduction may be accomplished with a one-dimensional calculation that includes the discrete regions of the core, vessel internals, by-pass and downcomer water, pressure vessel, reactor cavity, shield, and support structures. The resulting job library should consist of multigroup cross-sections based on the region-specific isotopic compositions. This job library should include ~20 energy groups above ~0.1 MeV. The adequacy of the job library must be demonstrated by comparing calculations for a representative configuration performed with both the master and job libraries. In these comparisons, the threshold-detector cross-sections and reaction rates obtained with the fine-group structure should be preserved in the multigroup calculations. In addition to the validation of the collapsing of the master library, additional qualification of the job library is provided by the benchmarking comparisons of Regulatory Position 1.4.2.

There are several ~50-group libraries available from RSICC that were generated using light-water reactor (LWR) spectra for the group collapsing from master libraries (Refs. 14-16) and may be used for LWR application. These libraries contain microscopic cross-sections as well as some premixed macroscopic cross-sections for relevant mixtures. Because these prepackaged

[3] The ASTM E1018 standard (Ref. 9) provides additional guidance on the selection of dosimeter reaction rate cross-sections.

libraries were designed specifically for LWR pressure vessel fluence calculations, their applicability in any atypical application (for example, when an accurate thermal flux is required) should be verified prior to use.

1.2 Core Neutron Source

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

The spatial dependence of the source should be based on depletion calculations that simulate core operation or 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 based on measured data.

The core neutron source should be determined by the power distribution (which varies significantly with fuel burnup), the power level, and the fuel management scheme. The detailed state-point dependence should be accounted for (Refs. 17, 18); however, if this is not feasible, averaging over the operating power distribution is acceptable and may be obtained by either (1) averaging representative power distributions within the cycle or (2) assuming the cycle-average assembly power distribution is well approximated by the accumulated-exposure distribution at the end of the cycle.

A best-estimate power distribution may be used for reactor vessel neutron 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 gradients, and these gradients should not be neglected (Ref. 19). In the case of PWRs, the fuel pins closest to the core periphery tend to have reduced relative power while for BWRs the peripheral fuel pins may have increased relative power. The pin-wise source distribution should be used for best-estimate calculations and the peripheral-assembly pin-wise source data should be obtained from core depletion calculations. The pin-wise source distribution should represent the absolute source distribution in the assembly.

The local source should be determined as the product of the fission rate and the neutron yield. 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 changes in the isotopic fission fractions with fuel exposure (caused by Pu build-up) result in variations in the fission spectra, the number of neutrons produced per fission, and the energy released per fission. These effects tend to increase the fast neutron source per megawatt of power for high-burnup assemblies. The variations in these physics parameters with fuel exposure may be obtained from standard lattice physics depletion calculations (Refs. 20, 21). This effect is particularly important for plants that have adopted the PWR low-leakage refueling schemes (vs. out-in-in three-batch fuel management) in which once-, twice-, or thrice-burned fuel is located in the high-importance peripheral locations (Refs. 22-25). 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.

A planar-octant representation is generally acceptable for the octant-symmetric fuel-loading patterns typically employed in LWRs. However, in the case of BWRs in which the jet-pump positioning is quadrant-symmetric, a quadrant-symmetric model may be required. For determining the peak fluence, fuel-loading patterns that are not octant symmetric (e.g., as in certain low-leakage patterns) may be represented in octant geometry using the octant having the highest fluence. For evaluating dosimetry, the octant in which the dosimetry is located may be used.

When the actual planar core rectangular geometry cannot be modeled (e.g., in the case of (r,θ) discrete ordinates calculations), the core geometry may be described using an (r,θ) planar representation. To accurately represent the important peripheral assembly geometry, a θ-mesh of at least 40 angular intervals over an octant must be applied. 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 (Ref. 26). Reference 26 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. A discussion of the uncertainties in determining the core neutron source is provided in Reference 27.

Determination of the three-dimensional (3-D) fluence at the vessel using (r,z)- and (r)-geometry calculations may also be appropriate (see Regulatory Position 1.3.4). If these calculations are used to provide an axial correction factor, since the (r,z)- and the (r)- dependent fluences enter as a ratio (see Equation 4), the source specification may be less stringent if consistent sources are used in both the (r,z) and (r) calculations.

1.3 Fluence Calculation

1.3.1 Discrete Ordinates Transport Calculation

The transport of neutrons from the core to locations of interest in the pressure vessel may be determined with a two-dimensional discrete ordinates transport program (Refs. 28-30) in (r,θ) and, when appropriate, in (r,z)- and (r)- geometries.[4,5] When calculating a horizontal plane of the core/vessel geometry in which the rectangular (x,y) geometry of the core boundary and the

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[4] Additional information concerning the application of transport methods to reactor vessel surveillance is provided in ASTM Standard E 482-89 (Ref. 31)

[5] If DOT (Ref. 29) is used, the "θ-weighted" option (MODE-5 in DOT 4.3) is considered to be more accurate than the "weighted" option (MODE-3 in DOT 3.5 or 4.3) for flux extrapolation and is recommended. Also, a value of $\theta = 0.3$ is generally adequate (Refs. 4, 32, 33). Reference 34 includes a description of the directional θ-weighted scheme which may also be used.

cylindrical (r) geometry of the vessel are mixed, a more accurate description is provided by the variable (r,θ) mesh option (Ref. 29) and may be applied.

An azimuthal (θ) mesh using at least 40 intervals over an octant in (r,θ) geometry in the horizontal plane should provide an accurate representation of the spatial distribution of the material compositions and source described in Regulatory Position 1.2. The radial mesh in the core region should be ~2 intervals per inch for peripheral assemblies and may be much more coarse 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, and a radial mesh of at least ~3 intervals per inch in water and ~1.5 intervals per inch in steel should be used. Because of the relatively weak axial variation of the fluence, a coarse axial mesh of ~0.5 interval per inch may be used except near material and source interfaces, where flux gradients can be large. An S_8 fully symmetric angular quadrature must be used as a minimum for determining the fluence at the vessel. However, in reactor cavity fluence calculations 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-model representation, the calculation should be performed in two or more "bootstrap" steps rather than compromising the spatial mesh or quadrature (the number of groups used usually does not affect the storage limitations, only the execution time). In this approach, the problem volume is divided into overlapping regions. In a two-step bootstrap 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 internal-boundary condition for a second transport calculation from R'' to R (the true outer boundary of the problem). The adequacy of the overlap region must be tested (e.g., by decreasing the inner radius of the outer region) to ensure that the use of the fictitious 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. To avoid negative fluxes and improve convergence, a weighted-difference model should be used.[5] The adequacy of the spatial mesh and angular quadrature, as well as the convergence criterion, must be demonstrated by tightening the numerics until the resulting changes are negligible (Ref. 32). In discrete ordinates codes, the spatial mesh and the angular quadrature should be refined simultaneously. In some cases, these evaluations can be adequately performed with a one-dimensional model.

The application of the discrete ordinates methods described in this section is illustrated in detail in Reference 4.

1.3.2 Monte Carlo Transport Calculation

The transport of neutrons from the core to locations of interest in the pressure vessel may be determined by using Monte Carlo Transport (Refs. 4 and 35-46). The Monte Carlo method has the advantage of allowing an exact representation of the problem geometry (Refs. 43-44); i.e., the rectangular fuel assemblies, the barrel, thermal shield and vessel cylindrical regions, and the axial geometry including upper and lower core reflectors, axial dosimeter locations, and vessel circumferential weld locations. In addition, the Monte Carlo method allows a continuous (as well as a multigroup) energy description of the nuclear cross-sections and flux solution. It is noteworthy, however, that in order to obtain comparable accuracy this method typically requires substantially longer computing times than the discrete ordinates method.

The cross-section data used in the Monte Carlo calculations should satisfy the requirements of Regulatory Position 1.1.2. However, if a continuous energy cross-section library is used, the requirements concerning the Legendre expansion of the scattering cross-sections and the number of energy groups are not applicable. In addition, to ensure an accurate integration over the dosimeter cross-section, the energy bin structure used to determine dosimeter response scoring should satisfy the requirements of Regulatory Position 1.1.2 concerning the selection of the multigroup library group structure.

While the Monte Carlo programs allow a great deal of flexibility in defining problem geometry, the input is generally relatively complex, requiring the definition of a large number of surfaces and cells, and is a potential source of error. Consequently, the geometry input should be checked thoroughly (using plots, volume checks, etc.) against the model design data to validate the definition of the geometry.

In pressure vessel evaluations, the flux is required at specific dosimeter and vessel surface and internal locations. The calculated flux is edited in area and volume tally regions at these locations. Typically, the size of these tally regions is increased to increase the number of particle histories and reduce the statistical error in the flux estimate. The sensitivity of the flux estimate to the volume or area of the tally region should be determined to ensure that the spatial dependence of the flux does not introduce a bias in the flux estimate. If the size of the tally region introduces a bias into the flux estimate, the calculation should be rerun with a smaller tally region and an increased number of particle histories or the Monte Carlo prediction should be adjusted to eliminate the calculational bias. This adjustment may be made, for example, using a correction based on a deterministic calculation.

In addition to the area and volume tally regions, point detectors may be used to estimate the flux at the locations of interest. However, the use of point detectors is complicated and the following cautions should be observed. To minimize the variance in point detector edits, Monte Carlo codes may define an average-flux region surrounding the detector within which the collision estimation is determined. The neutron flux in this region is assumed to be uniform. In order to ensure that the point estimate is not biased, the flux estimate should be shown to be insensitive to the selection of the average-flux region. If the size of the average-flux region introduces a bias into the flux estimate, the calculation should be rerun with a smaller average-flux region and an increased number of particle histories. No material boundaries should be included in the average-flux region surrounding the point detector. Since the next-event particle tracking used in point detectors does not account for reflecting, periodic, or white boundary conditions, the point detector flux estimate is underpredicted when particles from these boundaries can contribute to the flux estimate. Therefore, point detectors should not be located where particles from these boundaries can contribute to the flux estimate. In order to ensure that a sufficient number of particle histories are included and that the region close to the detector is adequately sampled, the statistical edits should be reviewed thoroughly, especially if the relative error is greater than 5%.

In the Monte Carlo analysis, a calculational uncertainty is introduced as a result of the finite number N of particle histories sampled. The Monte Carlo analysis provides estimates of the mean flux, relative error (standard deviation/mean) and terms related to the higher moments, such as the variance of the variance (VOV), based on the calculated particle histories. In order to ensure that the calculation has converged and the estimated (mean) flux is valid, a reliable estimate of the sampling uncertainty is required. However, because of the finite sampling of the physical phase space, important paths in the geometry and peaks in the cross-sections may not be adequately sampled, resulting in significant errors in the estimated mean and standard deviation.

Monte Carlo programs include statistical criteria to provide assurance that the calculations have converged and that the estimated mean and relative error are valid. When performing Monte Carlo pressure vessel fluence calculations, all statistical tests provided by the Monte Carlo code should be satisfied or justification for accepting the results should be reported. The Monte Carlo code should provide sufficient statistical testing (in addition to the usual tally standard deviation) to ensure that the scoring phase space has been adequately sampled. Representative statistical tests that are used to ensure tally convergence include (e.g., Reference 43):

(1) The flux mean should not have a significant monotonic dependence (either increasing or decreasing) on N for the last half of the problem,
(2) The percent relative error of the mean flux should be less than ~10%, except for point detectors for which it should be less than ~5%,
(3) The relative error should have a $\sim 1/\sqrt{N}$ dependence for the last half of the problem,
(4) The figure of merit defined as

$$FOM = \frac{1}{R^2 T} \qquad \textbf{(Equation 1)}$$

where R is the relative error and T is the problem computing time, should not have a significant monotonic dependence on N for the last half of the problem, and
(5) The VOV should be less than ~0.10 for all tallies and decrease as ~1/N for the last half of the problem.

As a result of the strong flux attenuation between the core and the pressure vessel, very few neutrons (~less than one in a thousand) actually reach the vessel. Since the relative error (and variance) in the Monte Carlo calculation only decreases as $1/\sqrt{N}$, the number of neutrons tracked and resulting computer times required to obtain acceptable accuracy are very large. In practice, the computing time may be reduced by using variance reduction techniques that increase the number of neutrons reaching the locations of interest (such as the dosimetry capsules and vessel inner-wall) or improve the particle tracking efficiency, or both. Typically, the number of particles that contribute to the tally is increased by increasing the number of particles in the regions of the particle phase space that are judged (based on prior knowledge) to contribute significantly to the calculation. However, if this judgment is incorrect and the neutrons believed to be unimportant make a significant contribution to the tally, the Monte Carlo estimated fluence and relative error (and variance) will be erroneous. Consequently, the variance reduction methods used in performing pressure vessel fluence analyses should be qualified by comparing the Monte Carlo variance reduction predictions with estimates made without the application of the variance reduction technique. Since the calculation without variance reduction will typically involve a large number of particle histories and long computing times, this comparison can be made using a simpler representative configuration (e.g., one-dimensional geometry).

The following Monte Carlo variance reduction methods may be used in performing pressure vessel fluence analyses: (1) neutron energy cutoff, (2) source biasing, (3) geometry splitting with Russian Roulette, and (4) weight windows. Specific concerns associated with the application of these methods are described in the following.

In the application of the neutron energy cutoff technique, neutron tracking efficiency is improved by terminating tracking when the particle weight falls below a selected input energy. This method assumes that particles with energy below the cutoff do not contribute to the tally and, consequently, may underestimate or bias the tally edit. While a cutoff energy of 1-MeV can be used to calculate the E > 1-Mev fluence without introducing any error, the cutoff energy for the dosimeter response calculation depends on the specific dosimeter reaction cross-section and should be selected low enough to ensure that the calculation provides acceptable accuracy. If this method is employed, the bias introduced by the energy cutoff should be estimated by comparison with an unbiased calculation and the Monte Carlo calculation adjusted to account for the bias.

The source biasing method increases the number of source particles in the regions of phase space that are believed (based on prior knowledge) to provide the most significant contribution to the Monte Carlo tally. For example, the neutrons born in the peripheral fuel assemblies contribute most to the vessel fluence and the number of these particles can be increased (and their weight decreased proportionally) to improve the particle tracking efficiency. The space- and energy-dependent importance of the core neutrons to a specific location on the vessel can be estimated using a precalculated adjoint flux and used to determine the source biasing (e.g., Refs. 41, 42, and 44). To ensure that the biased sampling of the source is producing reliable Monte Carlo estimates, the source biasing method should be qualified by comparison with an unbiased calculation for a representative problem.

Geometrical splitting with Russian Roulette reduces the variance of the tally edit by increasing the number of neutrons that reach the vessel. This is accomplished by assigning numerical importance values to each spatial cell such that the cell importance values increase as the neutrons approach the vessel (Ref. 46). When a neutron travels toward the vessel passing from a region of lower importance I to a region of higher importance I′, it is split into I′/I neutrons. Conversely, when a neutron travels away from the vessel passing from a region of higher importance I to a region of lower importance I′, it undergoes Russian Roulette with a survival probability of I′/I. The reliability of this technique depends on the selection of the cell importances I. It is recommended that the importance in adjacent cells should not change by more than a factor of ~3, and the cell radial optical thickness should be less than ~2 mean free paths (Ref. 43). To ensure that the selection of the cell importance distribution provides reliable Monte

Carlo estimates, the geometrical splitting with Russian Roulette method should be qualified by comparison with an unbiased calculation for a representative problem.

The weight windows method may also be used to improve the efficiency of the neutron tracking in Monte Carlo pressure vessel analyses (e.g., Refs. 41, 42, and 45). In the weight windows method, the neutron weight in each space-energy cell is limited to a specific range or "weight window" by splitting and Russian Roulette. The window is defined so that the neutron weight is inversely proportional to the importance of the space-energy cell so that the scoring variance is minimized (Ref. 43). In practice, the specification of the weight windows is difficult and requires a detailed knowledge of the space-energy-dependent cell importance. Consequently, in order to ensure that the definition of the weight windows is providing reliable Monte Carlo estimates, it is important that the weight windows method be qualified by comparison with a calculation performed without variance reduction.

The application of Monte Carlo methods described in this section is illustrated in detail in Reference 4.

1.3.3 Fluence Determination

The transport calculations (both discrete ordinate and Monte Carlo) 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 precalculated by either performing forward calculations with a unit source (with the desired pin-wise source distribution) specified in the fuel assembly of interest or by performing adjoint calculations. The adjoint fluxes are used to determine the fluence at a specific (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, material, and source distribution within the assembly remain the same.

In performing calculations of surveillance capsule fluence (Regulatory Position 1.4), it should be noted that the capsule fluence is extremely sensitive to the representation of the capsule geometry and internal water region (if present), and the adequacy of the capsule representation and mesh must be demonstrated using sensitivity calculations (as described in Regulatory Position 1.4.1). The capsule fluence and spectra are sensitive to the radial location of the capsule and its proximity to material interfaces (e.g., at the vessel, thermal shield, and concrete shield in the cavity), and these should be represented accurately. The core shroud and baffle former plates can result in a 5-10% reduction in the surveillance capsule dosimeter response and should be included in the model.

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 fluence reduction), these should be modeled in detail. When the transport calculation is performed using the discrete ordinates method, a finer spatial mesh and tighter convergence criteria may be appropriate to ensure an accurate solution. These schemes may also entail a 3-D flux calculation.

To account for the neutron spectrum dependence of RT_{NDT} when the E > 1 -MeV fluence is extrapolated from the inside of the pressure vessel to the T/4- and 3T/4-vessel locations, a spectral lead factor (which accounts for the change in neutron spectrum between downcomer and vessel internal locations) must be applied to the fluence for the calculation of ΔRT_{NDT} (Ref. 47). This spectral lead factor has been included in the Equation 3 attenuation formula of Revision 2 of Regulatory Guide 1.99, and therefore is not required when this formula is used. However, when this formula is not used, the spectral lead factor must be applied to the fluence at the vessel internal locations. Displacements per atom is used as the extrapolation parameter in Regulatory Guide 1.99 and is an acceptable parameter for extrapolating the E > 1-MeV fluence to vessel internal locations and determining this lead factor.

1.3.4 Synthesis of the 3-D Fluence

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

$$\phi_g(r,\theta,z) = \phi_g(r,\theta) * L_g(r,z) \qquad \textbf{(Equation 2)}$$

where $\varphi_g(r,\theta)$ 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

$$L_g(r,z) = P(z) \qquad \textbf{(Equation 3)}$$

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

$$L_g(r,z) = \phi_g(r,z)/\phi_g(r) \qquad \textbf{(Equation 4)}$$

where $\varphi_g(r)$ and $\varphi_g(r,z)$ 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 (Refs. 4 and 48). 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 (r,z) plane. 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 3 is only applicable when (1) the axial source distribution for all important peripheral assemblies is approximately the same or is bounded by a conservative axial power shape and (2) the attenuation characteristics do not vary axially over the region of interest. In addition, since the axial flux distribution tends to flatten as it propagates from the core to the pressure vessel, for typical axial power shapes, use of Equation 3 will tend to overpredict axial flux maxima and underpredict minima. This underprediction is nonconservative and can be large near the top and bottom reflectors, as well as when minima are strongly localized as occurs in some fluence-reduction schemes. Because of the neglect of the former plates, this method may result in an additional 5-10% underprediction of the fluence.

Equation 4 is applicable when the axial source distribution and attenuation characteristics vary radially but do not vary significantly in the azimuthal (θ) direction within a given 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 3. Use of this equation may 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. Conservatism 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 4; thus, it is generally more accurate in preserving the integral properties of the three-dimensional fluence. Both Equation 3 and Equation 4 assume separability between the axial and azimuthal fluence calculations, which 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

$$\phi_g(r,\theta,z) = \sum_{i=1}^{N} a_i \phi_{gi}(r,\theta)\phi_{gi}(r,z)/\phi_{gi}(r) \qquad \textbf{(Equation 5)}$$

where the φ_{gi} are basis flux solutions, typically representing specific regions of the core and vessel geometries, 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 a calculational benchmark in Reference 28 and to an experimental benchmark in Reference 49.

1.3.5 Cavity Fluence Calculations

Accurate cavity fluence calculations are relied on to analyze dosimeters located in the reactor cavity (Ref. 50). The calculation of the neutron transport in the cavity is made difficult by (a) the strong attenuation of the E > 1 MeV fluence through the (5.5- to 10.0-inch thick) vessel and the resulting increased sensitivity to the iron inelastic-scattering cross-section and (b) the possibility of neutron streaming (i.e., strong directionally dependent) effects in the low-density materials (air and vessel insulation) in the cavity. Because of the increased sensitivity to the iron cross-sections, ENDF/B-VI cross-section data should be used for cavity fluence calculations.[2] As indicated in Regulatory Position 1.1.2, properly benchmarked alternative cross-sections may also be used. However, for cavity applications, the benchmarking must include comparisons for operating reactor cavities or simulated cavity environments.

Typically, the width of the cavity together with the close-to-midplane locations of the dosimetry capsules result in minimal cavity streaming effects. Consequently, in discrete ordinates calculations, an S_8 angular quadrature is acceptable. However, when off-midplane locations are analyzed, the adequacy of the S_8 quadrature to determine the streaming component must be demonstrated with higher-order S_n calculations. In addition, since the radial mesh in the (r,z) calculation is generally finer than the z-mesh in the cavity resulting in narrow spatial-mesh intervals, a θ-weighted difference model should be used.[5]

The cavity fluence is sensitive to both the material compositions (Regulatory Position 1.1.1) and the local geometry (e.g., the presence of detector wells) of the concrete shield, and these should be represented as accurately as possible. Benchmark measurements involving simulated reactor cavities that are recommended for methods evaluation are given in Regulatory Position 1.4.2.2. The measured energy spectrum for a typical PWR cavity is described in Reference 51. When both in-vessel and cavity dosimetry measurements are available, an additional verification of the measurements and calculations may be made by comparing the vessel inner-wall fluence determined from (1) the absolute fluence calculation, (2) the extrapolation of the in-vessel measurements, and (3) the extrapolation of the cavity measurements. Measurements performed in reactor cavities and the associated calculations are described in References 5 and 6.

1.4 Methodology Qualification and Uncertainty Estimates

The neutron transport calculational methodology must be qualified, and flux uncertainty estimates must be determined. The neutron flux undergoes several decades of attenuation before reaching the vessel, and the calculation is sensitive to the material and geometrical representation of the core and vessel internals, the neutron source, the nuclear cross-section data, and the numerical schemes used in its determination. The uncertainty estimates are used to determine the appropriate uncertainty allowance to be included in the application of the fluence estimate. While adherence to the guidelines described here will generally result in accurate fluence estimates, the overall methodology must be qualified in order to quantify uncertainties, identify any potential biases in the calculations, and provide confidence in the fluence calculations. In addition, while the methodology, including computer codes and data libraries used in the calculations, 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 uncertainty analysis (Regulatory Position 1.4.1), (2) the comparison with benchmarks and operating reactor measurements (Regulatory Position 1.4.2), and (3) the estimate of uncertainty in the calculated fluence (Regulatory Position 1.4.3). These three steps in the overall qualification procedure are discussed below and are outlined in Figure 3.

1.4.1 Analytic Uncertainty Analysis

To demonstrate the accuracy of the methodology, an analytic uncertainty analysis must be performed. This analysis includes identification of the important sources of uncertainty. For typical fluence calculations, these sources include:

- Nuclear data (transport and dosimeter reaction cross-sections and fission spectra),
- Geometry (location of internals and deviations from the nominal dimensions),
- Isotopic composition of material (density and composition of coolant water, core, core barrel, former plates, thermal shield, pressure vessel with cladding, and concrete shield),
- Neutron sources (space and energy distribution, 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 of uncertainty components is not necessarily exhaustive, and other uncertainties that are specific to a particular reactor or a particular calculational method should be considered. In typical applications, the fluence calculational uncertainty is dominated by a few easily identified uncertainty components, such as the geometry and core neutron source, which results in a substantial simplification of the uncertainty analysis.

The sensitivity of the flux to the significant component uncertainties should be determined by a series of transport sensitivity calculations in which the calculational model input data and modeling assumptions are varied and the effect on the calculated flux is determined. (A typical sensitivity would be ~10-15% decrease in vessel $E > 1$ MeV fluence per centimeter increase in vessel inside radius.) Estimates of the expected uncertainties in these input parameters must be made and combined with the corresponding fluence sensitivities to determine the total analytically determined calculation uncertainty σ_a^c (i.e., standard deviation) as indicated in Figure 3. The uncertainties should be combined in a statistical (root-sum-of-the-squares) fashion to determine the total fluence uncertainty, accounting for the correlation between uncertainties when necessary (see, e.g., Section 5.2 of Ref. 3). The known systematic errors (or biases) should be combined algebraically, recognizing the sign of each contribution, to determine the overall calculational bias B_a^c (Ref. 52).[6] The component uncertainties should be based on measurement or on the acceptable deviations included in the design specifications. The sensitivity calculations may be performed in one dimension when the model sensitivity does not require a two-dimensional representation.

If the sensitivities are determined using Monte Carlo transport calculations, the magnitude of the model variations and the number of particle histories may have to be increased to provide reliable estimates of the calculation sensitivities. If the increased model variations are large, several calculations may be required to determine the non-linear dependence of the sensitivity. Perturbation calculations using correlated sampling may also be used to determine the sensitivity to model geometry, material compositions, and cross-sections. Adjoint Monte Carlo calculations may be used to determine the sensitivity to input such as fission spectra. Alternatively, if the sensitivities can be determined with sufficient accuracy in one-dimensional geometry, the sensitivities can be determined using a one-dimensional model.

[6] The systematic errors affect the fluence prediction in a specific direction (i.e., increase or decrease the estimate). Consequently, when these errors (or biases) are applied, the signs (or directions) must be recognized and they must be combined algebraically. The independent random uncertainties have no specific direction (i.e., an increase or decrease) associated with them, and they are combined without recognizing the sign of the error.

A sensitivity analysis, in which the influence of each of these uncertainties on the calculated group fluences has been considered, is included in References 5 and 6. Since the uncertainties used in these analyses are common to many pressurized water reactors, the sensitivities (including correlations) may be used to determine initial uncertainty estimates. These variance estimates should be modified as the methods and cross-section data change or if the reactor of interest differs substantially from the reactors analyzed in these references. The referenced sensitivity analysis provides guidance for such modifications.

1.4.2 Comparisons with Benchmark Measurements and Calculations

Calculational methods must be validated by comparison with measurement and calculational benchmarks. The fluence calculation methods must be validated against (1) operating reactor measurements that provide 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 (see Regulatory Position 1.4.2.2 for a discussion of such benchmarks), and (3) the fluence calculation benchmark. The results of the validation should include comparisons of reaction rates, fluences, and group fluxes for the locations of interest (Refs. 53, 54).

The methods used to determine the plant-specific data and calculate the benchmarks must be consistent (to the extent possible) with those used to calculate the vessel fluence. That is, the same cross-sections, transport techniques, and transport code parameters that are to be used in the reactor licensing application must be employed in the calculation of the benchmark measurements and calculations.

Differences between measurements and calculations should be consistent with the combined uncertainty estimates for the measurements and calculations. (Note that the uncertainties in both the calculations and measurements will contribute to the observed measurement-to-calculation differences.) The calculated reaction rates (using the methods described in Regulatory Positions 1.1 through 1.3) typically agree with the measurements to within about 20% for in-vessel surveillance capsules and 30% for cavity dosimetry. Deviations greater than these values must be investigated and, when the cause of the deviation is determined to be an error in the calculation, the calculations must be modified.

If the calculation is modified using a least-squares statistical approach (Refs. 55-62), the measurement-to-calculation (M/C) ratios should be reviewed, prior to adjustment, in order to identify any systematic trends in the M/C data. Specific examples include systematic overprediction (or underprediction) of the dosimeter response for (1) dosimeters with low-energy thresholds, (2) dosimeters in the cavity (versus inner-wall and accelerated capsule locations), (3) dosimeters with long half-life reaction products, and (4) dosimeters monitoring specific cycles. If systematic trends, interdependence, or correlations appear in the M/C data, these should be explained and removed (prior to adjustment) or accounted for in the statistical adjustment procedure.[7] Also, since the fluence adjustment is sensitive to the input uncertainties, realistic uncertainty values should be used and documented. The dosimeter measurement uncertainty should include the response uncertainty caused by dosimeter mislocation. Additional guidance on least-squares adjustment methods is provided in Ref. 63. It should be noted, however, that the available least-squares statistical adjustment codes differ substantially in their approach and their capabilities. Consequently, the selection of the least-squares code and the details of the analysis are highly application-specific and are outside the scope of this guide.

The simulator benchmarks provide accurate measurements but typically do not provide an accurate representation of the actual plant configuration. The operating reactor measurements, on the other hand, represent the actual as-built plant configuration but, typically, include substantial measurement uncertainties. The simulator benchmarks together with the operating reactor measurements generally provide an acceptable measurement data base. The comparisons of the calculations to the measurement benchmarks should be used to estimate the calculational bias B_b^c and uncertainty σ_b^c. If the bias estimate is considered reliable, a correction equal to the bias may be applied to the calculation to determine a best-estimate fluence.[8] If the calculations or measurements are adjusted to improve the M/C agreement, using either a least-squares approach or a direct multiplicative bias, the adjustment and basis must be reported.

1.4.2.1 Operating Reactor Measurements. Well documented fluence dosimetry measurements for operating power reactors may be used for methods and data qualification.[9] Descriptions of these configurations include three-dimensional geometry, reactor operating conditions and, in some cases, both in-vessel and ex-vessel measurement data.

Comparisons of measurements and calculations 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 can be obtained when both in-vessel and cavity dosimetry are available.

These measurements should not be used to bias or adjust the fluence calculations unless a statistically significant number of measurements is available, the various dosimeter measurements are self-consistent, and a reliable estimate of the calculational bias can be determined.[8,9] Similarly, plant-specific biases should not be used unless sufficient reliable measurement data are available. As capsule and cavity measurements become available, they should be incorporated

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[7] For example, if the M/C data for the Np-237 dosimeters tends to be high, indicating a systematically low calculation or high measurement, the errors in the Np-237 calculated or measured responses are correlated. If this error is not removed prior to the adjustment, the important sources of correlated error should be identified (e.g., the Np-237 dosimeter reaction cross-section, reaction product half-life and Np-237 photo-fission correction) and included in the adjustment procedure to allow the proper treatment of this error.

[8] The adequacy of the measurement data base for determining a bias in the calculations depends on the magnitude of the bias and is therefore problem dependent. For example, if the bias is small an accurate estimate of the bias will require either (1) a large number of reasonably accurate measurements or (2) a small number of very accurate measurements. Specifically, the uncertainty in the bias should be substantially less than the bias itself. For example, if the calculated fluence is to be increased by 10% , the uncertainty in this increase should be substantially less than 10%.

[9] Additional guidance on benchmark testing is provided in the ASTM standard E2005-99 (Ref. 64) and examples of operating reactor benchmark measurements are provided in Reference 65.

into the operating reactor measurements data base, and the calculational biases and uncertainties should be updated as necessary.

1.4.2.2 Pressure Vessel Simulator Measurements. Several pressure vessel simulator benchmarks are available (Refs. 7, 49, 65-82) and may be used for methods qualification. These benchmark experiments were carried out by several laboratories, and dosimetry measurements using different techniques were compared to provide experimental results with well known and documented uncertainties. Three configurations were used in these experiments.

- **Pressure Vessel Simulator Experiments**. The Oak Ridge National Laboratory (ORNL) Pool Critical Assembly (PCA) is a full-scale-section mockup of a pressure vessel with dosimetry measurements at the inner surface of the vessel and at the T/4, T/2 and 3T/4 locations within the vessel wall (Refs. 72-75). The PCA benchmark experiment includes a simulated surveillance capsule that allows the determination of the effect of the capsule on the dosimetry measurements. This benchmark is characterized by a relatively simple geometry with generally less uncertainty in region compositions, temperatures, and source distributions than in operating power reactor benchmarks.

- **Experiments Simulating Operating Reactor Conditions**. The CEN/SK Laboratory VENUS benchmark experiment (Refs. 49, 76-79) and the ORNL Pool Side Facility (PSF) benchmark experiment (Refs. 80-82) simulate the operating reactor: (1) pin-wise power distributions in peripheral fuel assemblies, (2) three-dimensional effects caused by partial-length shield assemblies, and (3) heterogeneities caused by neutron pads attached to the core barrel.

- **Pressure Vessel Cavity Measurements.** The H.B. Robinson-2 cavity measurement (Refs. 65-67) and the NESDIP cavity simulator experiment (Refs. 7, 68-71) may be used as benchmarks to qualify cavity fluence calculational methods. Calculations of dosimetry measurements performed in an operating reactor cavity are described in References 5 and 6.

1.4.2.3 Calculational Benchmark. The vessel fluence benchmark problems provided by the NRC in NUREG/CR-6115 (Ref. 4) should be used for methods qualification. This report provides the detailed information defining the problems and corresponding numerical solutions for a set of PWR and BWR pressure vessel fluence benchmark problems. The geometry, materials, and space- and energy-dependent source are fixed by the problem specification. PWR benchmark problems have been specified for (1) a standard core loading pattern, (2) a low-leakage core loading pattern, and (3) a partial-length shield assembly core design. Since BWR fuel loading patterns are presently not being designed for vessel fluence reduction, only a single BWR problem is specified. In addition, MCNP Monte Carlo calculations have been performed for both the BWR problem and the PWR problems for the standard core and partial-length shield assembly core loadings. Comparisons of the vessel fluence determined by using both the MCNP and DORT computer codes are included.

The calculation of the benchmark problems allows a detailed assessment and verification of the numerical procedures, code implementation, and the various modeling approximations relative to state-of-the-art solutions for representative operating configurations.[10] If the differences between the benchmark problem calculation and the reference solution are substantially larger than what would be expected based on the differences in the methods approximations and nuclear data used in the two calculations, the agreement is considered unacceptable. In this case, the calculation should be reviewed and the differences between the two solutions explained. (Note that NUREG/CR-6115 includes sensitivity calculations and comparisons with Monte Carlo predictions which provide an indication of the accuracy of the methods approximations used in the NUREG/CR-6115 reference calculations.) When the cause of the deviation is determined to be an error in the calculation, the calculational method must be revised.

1.4.3 Estimate of Fluence Calculational Bias and Uncertainty

The overall fluence calculation bias B^c and uncertainty σ^c must be determined by an appropriate combination of (1) the analytic uncertainty analysis results of Regulatory Position 1.4.1 (B^c_a and σ^c_a) and (2) the results of the uncertainty analysis based on the comparisons to the operating reactor and simulator benchmark measurements of Regulatory Position 1.4.2 (B^c_b and σ^c_b). This combination may be a weighted average that accounts for the reliability of the individual estimates.[11] The bias B^c may be applied as a multiplicative correction to the calculated fluence to determine the best-estimate value, and σ^c should be used (when required) as the (1-sigma) uncertainty in the calculated best-estimate fluence.

The fluence accuracy requirements are generally application specific; however, a vessel fluence uncertainty of 20% (1 sigma) is acceptable for RT_{PTS} and RT_{NDT} determination. In the determination of RT_{NDT} and RT_{PTS}, if the overall calculation uncertainty σ^c is greater than 20%, the calculational model must be adjusted or a correction must be applied to reduce the difference between the fluence prediction and the upper 1-sigma limit to within 20%. For example, if σ^c is greater than 20% but less than 30%, the fluence φ should be determined from the calculated fluence φ_c by the relation

$$\phi = \{1 + B^c + [\sigma^c(\%) - 20]/100\}\, \phi_c \qquad \textbf{(Equation 6)}$$

If σ^c is greater than 30%, the methodology of this regulatory guide is not applicable and the application will be reviewed on an individual basis. The procedure used to determine the fluence is illustrated in Figure 4.

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[10] The calculation being used as the benchmark must be the NUREG/CR-6115 original referenced benchmark calculation, and not just a second independent calculation of the benchmark.

[11] Because the weighting of the analytic and benchmark uncertainty estimates depends on the details of the specific application that can vary widely, it is not possible to specify a practical and generically valid prescription for determining the weights. However, the following example illustrates factors that should be considered. When as-built measurements of the vessel diameter are available and reasonable estimates of the core neutron source and other input uncertainties can be determined, the analytic uncertainty estimate should be reliable and have an uncertainty of ~15% (1 sigma). Assume that there are a statistically significant number of accurate (σ < 5%) operating reactor measurements and the uncertainty estimate based on this data has an uncertainty of ~20%. The uncertainty estimate based on the vessel simulator measurements is assumed to be less certain and has an uncertainty of ~25%. Using a weighted mean in which the weight is inversely proportional to the square of the standard deviation of the estimate (i.e., σ^{-2}), the weights are $w_A = 0.5$, $w_o = 0.3$ and $w_S = 0.2$.

The fluence calculation methods of this section are summarized in Table 1, "Summary of Regulatory Positions on Calculation and Dosimetry."

2. NEUTRON FLUENCE MEASUREMENT METHODS

Dosimetry measurements provide independent estimates of specific activities and isotopic production rates that are used for validating the neutron transport calculations. Fluence is obtained from the response of passive integral detectors placed in surveillance capsules and, more recently, in the ex-vessel cavity. Procedures for performing the measurements and assessing uncertainty, including the validation of measurement methods and detector response in standard neutron fields, is described in this section.

The fluence measurement methods of this section are summarized in Table 1, "Summary of Regulatory Positions on Calculation and Dosimetry." The procedure used to qualify the measurement methods is presented in Figure 5. The guidance provided in this Regulatory Position is intended for new dosimetry and, to the extent practical, for existing dosimetry.

2.1 Measurement Procedures

Power reactor dosimetry measurement methods use passive integral detectors, which are typically activation detectors and solid-state track recorders. Most of the detectors respond to neutrons with energies above a characteristic reaction threshold. These detectors should be selected with substantial nonoverlapping 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 should provide adequate spectrum coverage. A common set of fast neutron integral detectors that may be employed in these packages is listed in Table 2. These neutron detectors are discussed in References 83 through 89. The $^{54}Fe(n,p)^{54}Mn$ and $^{58}Ni(n,p)^{58}Co$ activation reactions are useful for monitoring neutrons with energies above ~2 MeV. The $^{46}Ti(n,p)^{46}Sc$ reaction is useful for monitoring neutrons with energies above approximately 4 MeV, and the $^{63}Cu(n,\alpha)^{60}Co$ reaction is useful for measuring neutrons with energies above approximately 5 MeV. Fission monitors are important since they typically have reaction thresholds below 1 MeV. Activities determined by assaying one or more fission products from the $^{238}U(n,f)FP$ fast-neutron fission reaction are useful for monitoring neutrons with energies from approximately 1.5 to 7 MeV. Activities determined by assaying fission products from the $^{237}Np(n,f)FP$ reaction are useful for measuring neutrons with energies from approximately 0.7 to 6 MeV; however, there is a significant gamma-ray background associated with neptunium dosimetry that is due to the large quantity of naturally occurring ^{233}Pa existing in secular equilibrium with ^{237}Np. Compton scattering of the protactinium decay photons generates a large background continuum, which reduces the effective signal-to-noise ratio and lowers the accuracy of the radioassay. The $^{93}Nb(n,n')^{93m}Nb$ activation reaction is useful for monitoring neutrons with energies above approximately 1 MeV, but involves the counting of a very soft photon, which can complicate the corrections made to the measured activity to compensate for photon attenuation within the dosimeter.

In addition to the fast-neutron threshold monitors, a thermal monitor such as ^{59}Co should be included among the selected dosimeters to allow the determination of the thermal neutron fluence. The thermal fluence is needed to assess the effects of interfering activities in threshold monitors produced by low-energy activation of impurity elements. In instances where the thermal flux is relatively high, it can also be used for assessing the effects of burnout of the activated species.

Taken together with a low-energy detector such as cobalt (to estimate the thermal neutron fluence for determining interference from low-energy activations), the Table 2 dosimeter set provides satisfactory neutron energy spectrum coverage for pressure vessel dosimetry. Alternative detector sets that are used should provide equivalent spectrum coverage. Detector selection criteria and related recommendations in ASTM E 844 (Ref. 90) and E 1005 (Ref. 91) should be followed (see Table 2).

The application of the dosimeter should be evaluated relative to the expected neutron flux and the total irradiation time in order to assess the effects of dosimeter burnout, and the burnout and burn-in of reaction products. For example, when using ^{58}Ni monitors in the presence of a high thermal flux, the burnout of the nickel dosimeter as well as the ^{58}Co and ^{58m}Co reaction products should be considered. The resonant absorption of epithermal neutrons in ^{238}U and the breeding of ^{239}Pu, which can undergo thermal fission and contribute to the fission product activity, should also be considered.

Application of activation detectors involves aspects of the measurement process that must be carefully controlled and documented to obtain accurate results and establish reasonable uncertainty estimates. Where applicable, procedures in ASTM Standards E 181 (Ref. 92), E 844 (Ref. 90) and E 1005 (Ref. 91) and methods devoted to individual radiometric sensors must be used as indicated in Table 2. Specific regulatory positions associated with the dosimetry measurements are indicated in the following.

2.1.1.1 Dosimeter Nuclear and Material Proprieties. The dosimeter nuclear and material properties are important considerations in the selection and composition of the surveillance dosimetry package. The physiochemical properties of the dosimeter materials should be compatible with the prevailing service conditions. Dosimeter melting, for example, can result in the loss of monitor activity and unreliable measurements. Consequently, the melting point of the dosimeters and any associated materials should be carefully considered relative to the service environment ambient temperature and the radiation heating within the dosimeter. Although the melting point of typical radiometric monitors is sufficiently high to preclude melting, even mild heating can be problematical for certain dosimeters. For instance, diffusion losses of ^{54}Mn activity from iron dosimeters can occur when heated to temperatures greater than $700^{\circ}C$. In addition, the dosimeter should be chemically stable and corrosion resistant. Fissionable materials are especially prone to oxidation and, consequently, fission monitors should be hermetically sealed to prevent oxidation and subsequent loss of material.

The nuclear properties to be considered in dosimeter selection include the reaction cross-section, decay half-life, gamma-ray yield, and the fission yield when employing fission dosimeters. The reaction cross-section should be of sufficient magnitude to allow the dosimeter to be used as a fluence monitor, consistent with other requirements on the dosimeter mass and geometry, as discussed in following sections.

The photon yield associated with the nuclear decay should be large enough to allow an accurate radio assay, and the energy of the particular photons counted should minimize interference from neighboring spectral lines. For example, the ^{24}Na reaction product, created from the ^{27}Al(n,α) reaction, emits photons with energies of 1.369 and 2.754 MeV. Since there is less interference from the contaminant gamma rays associated with the higher-energy photon, it is preferable to count the high-energy line during the radioassay. In the case of fission product radioassay involving the ^{95}Zr-^{95}Nb fission product, three dominant spectral lines are observed at 724, 756, and 765 keV. However, the two high-energy lines are in close proximity and complicate the determination of the background subtraction. Consequently, counting the 724 keV line provides the more accurate result.

The half-life of the activated species should be long enough, relative to the elapsed time from the end of the irradiation to the time at which the dosimeters are actually counted, to permit nuclear counting to proceed with reasonable counting statistics. This time interval is often a significant consideration. The half-life should also be long enough to provide an accurate indication of the fluence during the irradiation period.

The fission product yields associated with fission dosimeters should be of sufficient magnitude to permit an accurate radioassay. The yields (and decay constants) of the fission products ^{99}Mo and ^{140}Ba-^{140}La are known with greater accuracy than most fission products; however, their half-life is generally too short for these isotopes to be useful as neutron fluence monitors in power reactor applications. Useful fission products include ^{95}Zr-^{95}Nb, ^{103}Ru, ^{144}Ce, and especially ^{137}Cs-^{137m}Ba, because of its 30-year half-life.

2.1.1.2 Material 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 ^{237}Np and ^{238}U and minute amounts of cobalt in copper and nickel dosimeters (fractional parts per million) should be determined by mass spectrography or radioactivation analysis. Tracking the principal decay half-life is also a useful technique for determining the presence of extraneous activities from impurity elements.

Dosimetric materials should be handled with care to prevent cross contamination of the dosimeters and, when necessary, the dosimeters should be cleaned of surface contaminants. For example, the handling of aluminum monitors can introduce sodium (^{24}Na) as an external surface contaminant. Neutron activation of the sodium will produce ^{24}Na activity that will interfere with the radioassay of the ^{24}Na activity produced from the desired dosimetric reaction.

Post-irradiation quality assurance may be accomplished at the National Institute for Standards and Technology (NIST) for dosimeters already in surveillance capsules by additional irradiation in a thermal neutron field or a standard fission spectrum. These additional exposures and analyses can provide data on the dosimeter masses and impurities (for example, the original cobalt-59 impurity content in copper used for the copper-63 (n, α) reaction).

2.1.1.3 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 E 844, Ref. 90). The thickness of the thermal shield should be selected based on the expected thermal flux and duty cycle. (Note that the effective cadmium cutoff energy is a function of shield thickness.) The design of the shields and their placement relative to other dosimeters should be carefully considered in order to minimize their impact on neighboring foils. For example, the use of a cadmium shield will generate a local gamma field that could affect adjacent fission foils by producing a photofission component in the measured activity.

2.1.1.4. Isotopic Mass. The mass of the dosimeters should be selected to permit the production of sufficient activity to allow an accurate radioassay. However, a dosimeter mass that is too large can lead to high activities that result in excessive dead-time losses during nuclear counting and should be avoided. Metal alloys can be used for diluting materials that have high nuclear cross-sections (e.g., dilution of cobalt in an aluminum alloy). However, stoichiometry and isotopic analysis should be well documented for dosimeters that are not of pure natural elements.

2.1.1.5. Geometry and Location. The geometric configuration of the dosimeters can have a significant effect on the dosimeter response. Most radiometric monitors are in the shape of thin circular activation foils, although other shapes are available. Dosimeters should be small enough to allow placement at the desired location and large enough to provide sufficient activity to allow an accurate measurement. The dosimeter thickness can have a significant effect on the dosimeter self-shielding (of the neutron flux) and the dosimeter response for strongly absorbing foils. In addition, increased dosimeter thickness increases the effects of scattering and photon attenuation within the dosimeter and can lead to large response corrections. When selecting fissionable dosimeters, it should be recognized that reduced foil thickness increases fission product loss from nuclear recoil out of the foil, which reduces the measured dosimeter activity.

The placement of the dosimeters can also have a significant effect on the dosimeter measurement. The surroundings of a dosimeter (e.g., adjacent dosimeters or material interface) can influence detector response. Strongly and weakly absorbing foils that respond over the same energy range should be sufficiently separated within the dosimetry package to minimize interference effects. The arrangement of the dosimeters within the package should minimize scattering effects (viz., thick dosimeters should not be stacked so that large scattering corrections are required for adjacent foils).

The location of individual dosimeters must be determined accurately and recorded, because fluence gradients in out-of-core positions are generally severe. In the pressure vessel cavity, establishing azimuthal position can be as important as the radial location. Specially designed mounting arrangements that ensure accurate radial, azimuthal, and vertical positions should be used for cavity dosimetry. Vertical wires may be used. Comprehensive and accurate detector location information should be maintained.

2.1.1.6 Solid-State Track Recorders. Solid state track recorders (SSTRs) are integral detectors that employ fission reactions. These sensors directly record the tracks of fission fragments from a thin fissionable deposit (Refs. 93 and 94). The principal advantages of these detectors are wide sensitivity ranges and a permanent measurement record. Because the application of SSTRs employs fissionable deposits in the nanogram to picogram range, details of the measurements should be well documented, and standard neutron field calibration should be performed prior to the application. ASTM Standard E 854 (Ref. 95) provides additional information concerning the use of SSTRs.

2.1.1.7 Helium Accumulation Fluence Monitors. The helium accumulation fluence monitor (HAFM) is another type of stable-product neutron monitor. These monitors rely on the generation of helium gas via (n,α) reactions that occur in a variety of materials when exposed to a neutron field. The amount of helium generated in a specific material is a function of the neutron fluence and is determined by vaporizing the sample and measuring the helium content by mass spectrometry. HAFMs are routinely used in conjunction with radiometric monitors and, in fact, several commonly used radiometric monitors also undergo (n,α) reactions and can be analyzed for helium accumulation as well as for radiometric activity. ASTM Standard E910 (Ref. 96) provides additional guidance on the use of HAFMs.

2.1.2 Detector Response Measurements

A measured response must be provided to allow comparison of the calculations and measurements. Typical measured responses include the specific activity at end-of-irradiation (given in disintegrations per second per nucleus), the measured isotopic production (for example, helium atoms per initial atom of material), and the total reactions (for example, fissions per initial atom of material). When comparing calculations with measurements, corrections should be included for detector response perturbations, interfering reactions, and, when applicable, burnup and photofission. In the calculation of end of irradiation activities (or in the conversion of end of irradiation activities into reaction rates), the power-history and the half-life of the dosimeter activation products must be included (Ref. 91). 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 reporting measurements, these corrections should be described and quantified 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 and over long periods of time.

2.1.3 Measurement Uncertainties

Regulatory Position 1.4 states that the calculations must be validated by comparison with measured benchmarks. In order to perform this validation, the uncertainty associated with the measured response must be determined for each dosimeter type. The uncertainty must be included in the documentation of the measured results.

The important uncertainties must be quantified and included in an uncertainty table that summarizes the specific components of uncertainty contributing to each detector response. Typical sources of uncertainty include (1) background counting, counting efficiency, counting statistics, and fraction of the sample counted, (2) decay constant and fission yield uncertainty, (3) uncertainty in detector weight, geometry, composition and isotopic purity, (4) power history and time of irradiation, (5) dosimeter location uncertainty, and (6) cross-section uncertainty when the fluence is determined. Additional sources are identified in Regulatory Positions 2.1.1 and 2.1.2 and in the dosimeter-specific ASTM Standards of Table 2. The entries in the table should be identified as standard deviations, upper bounds, or appropriate fractions of the correction. Each entry should be described, along with the method used to combine the entries and determine the total response uncertainty. The evaluation of the measurement uncertainty for each dosimeter is important for ensuring a meaningful comparison with the calculations.

2.2 Validation in Standard and Reference Neutron Fields

The dosimeter measurements used to benchmark the calculations must meet the requirements in Appendix B to 10 CFR Part 50 for a quality assurance program (for surveillance measurements). To ensure long-term measurement consistency and confirm measurement uncertainties, dosimetry measurements must be performed every few years[12] in well characterized neutron fields. If there are changes in dosimeter evaluation procedures, equipment, or personnel that could have a significant effect on the dosimetry, the validation must be updated. This validation may be performed using the Materials Dosimetry Reference Facility (MDRF) or using reference fission neutron sources (Refs. 97, 98). Neutron field referencing may be used as a detector response calibration (Ref. 99).

The validation is accomplished by exposing each type of detector to a certified neutron fluence in the reference neutron field and by determining the fluence using the measurement method to be validated. A calculated spectrum-averaged cross-section, generally specified along with the certified neutron fluence, must be used to derive the measurement value. If measured and certified neutron fluence agree (to within the combined uncertainty of the measured and certified fluence), the detector measurement method, including the detector cross-section, is validated. If the fluences disagree, this calculation-to-experiment (C/E) ratio represents a bias associated with the detector response measurement or the detector cross-section or both. In this case, the detector measurement methods and input parameters should be re-examined in order to eliminate the bias. If after re-examination the bias is still present, the bias may be used directly as a detector response calibration factor.

The results of the neutron field validation procedure should be reported in terms of C/E ratios for the individual detectors and should include an uncertainty table. The standard neutron field validation may be used, as appropriate, to simplify the uncertainty table called for in

[12] This measurement validation procedure does not require transport calculations.

Regulatory Position 2.1.3 by reducing or eliminating many uncertainties in the activity measurement, nuclear decay parameters, and detector cross-sections.

Aside from validating the measurement method, standard and reference neutron fields may be used for quality assurance of critical features of the detector response. Examples are activation interference by impurities, proper determination of fission product activities, and mass assay of fissionable deposits for track recorders.

Interlaboratory comparisons, such as those carried out under the NRC-sponsored LWR Surveillance Dosimetry Improvement Program (Refs. 7, 49, 72, 80, 81), have been useful in validating the quality assurance programs of various laboratories. These interlaboratory comparisons have been used to identify problems in the measurement procedures and establish consistency between the various participating laboratories.

2.3 Fluence Determination from Detector Measurements

As stated in Regulatory Position 1.4, the calculated fast-neutron fluence values must be validated by comparison with measurement benchmarks. These comparisons validate the calculational methodology by determining the bias and uncertainty in the calculated fluences, E > 1 MeV. A fast-neutron fluence, E > 1 MeV, must be obtained for each detector as the quotient of the measured reaction probability and the effective E > 1-MeV spectrum averaged cross-section (Ref. 100) based on a neutron transport calculation. These fluences and a suitably weighted average fluence must be reported together with the associated measurement uncertainty determined using the methods of Regulatory Positions 2.1.3 and 2.2 and accounting for the cross-section uncertainty. An alternative to deriving a neutron fluence from the detector responses is to directly compare specific end-of-irradiation activities, measured reaction probabilities, isotopic production, total reactions, average reaction, or measured reaction rates with results from the neutron transport calculation. The response selected should provide a reliable indication of the uncertainty and bias in the E > 1 MeV fluence calculation.

The measurement biases and uncertainties must be documented for the selected detector response. Based on the measurement-to-calculation (M/C) comparisons, an average M/C ratio should be determined as a suitably weighted average of the individual M/C values. In determining the weighting for this average, as a minimum the measurement and spectral average cross-section uncertainties should be considered. An estimate of the uncertainty in the neutron spectrum may be included as an additional contributor to the dosimeter weighting.

When justification can be provided, an individual detector may be declared suspect and given reduced weight or discarded. However, in this case the justification and procedure used must be documented.

2.4 Ex-vessel Dosimetry

As-built in-vessel surveillance capsule dosimetry 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 modifications. The ex-vessel cavity may be used as an alternative site for installing additional improved dosimetry. Recent pressure vessel benchmark experiments (Refs. 7, 67, 94) have demonstrated that the ex-vessel dosimetry can provide useful exposure information within the pressure vessel wall (Refs. 72, 81). When placed at appropriate circumferential locations, this dosimetry is a good monitor of the effectiveness of low-leakage core strategies.

3. REPORTING

When fluence determinations are required by the regulations, the licensee's documentation describing the determination of pressure vessel fluence must provide a complete description of the methods used to calculate and measure the neutron fluences. In applying the methodology of this guide, the details of the application and the results should be reported as described in this section. A topical report providing the details of the fluence methodology, as outlined in Regulatory Positions 1 and 2, should be satisfactory for compliance with this guide. Plant-specific measurements and data should be reported in the pressure vessel surveillance reports.

The specific regulatory positions on reporting are given in this section and are summarized in Table 1, “Summary of Regulatory Positions on Calculation and Dosimetry.”

3.1 Fluence Methods

The methods used to calculate the integral and multigroup fluences and fluence rates and associated methods qualification should be reported. The calculational uncertainty analysis, benchmark comparisons, and the determination of the overall fluence uncertainty should be described in detail. If the calculated fluence is adjusted, the fluence adjustment and associated uncertainty together with justification must be reported. A discussion of any deviations from the procedures provided in this regulatory guide should 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.

3.2 Multigroup Fluences

The calculated absolute multigroup neutron fluences and fluence rates at the peak wall (or other limiting) location, surveillance locations, and T/4 and 3T/4 positions within the pressure vessel should be reported. The multigroup energy boundaries should be included.

3.3 Integral Fluences

The calculated $E > 1$ MeV integral fluences and fluence rates at the vessel inner wall locations (determined as described in Regulatory Position 1.4.3), together with the uncertainties, should be reported.

3.4 Comparisons of Calculation and Measurement

If the qualification of the calculation methods is performed using fluence comparisons, the measured and calculated E > 1 MeV integral neutron fluences should be reported. If the methods qualification is performed using reaction rate comparisons (or other responses as described in Regulatory Position 2.3), the calculated and measured reaction rates (or reaction probabilities) should be reported. The spectrum-averaged reaction cross-section used to relate the fluence and reaction rate and the method for its determination should also be provided. In either case, the M/C ratios and the measurement uncertainty should be reported for the average fluence and for each detector at each measurement location. If these measurements are used to adjust the calculation, the adjustment should be described and the M/C ratios before and after adjustment should be reported.

3.5 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 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 the dosimeter thermal-neutron shield material.

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.

Except in those cases in which an applicant or licensee proposes an acceptable alternative method for complying with the specified portions of the NRC's regulations, the methods described in this guide will be used in the evaluation of applications for new licenses and for evaluating compliance with 10 CFR 50.61 and Appendix G to 10 CFR Part 50, which is imposed by 10 CFR 50.60.

TABLE 1. SUMMARY OF REGULATORY POSITIONS ON CALCULATION AND DOSIMETRY

FLUENCE CALCULATION METHODS	Regulatory Position
Fluence Determination. Absolute fluence calculations, rather than extrapolated fluence measurements, must be used for the fluence determination.	1.3
Modeling Data. The calculation modeling (geometry, materials, etc.) should be based on documented and verified plant-specific data.	1.1.1
Nuclear Data. The latest version of the Evaluated Nuclear Data File (ENDF/B) should be used for determining nuclear cross-sections. Cross-section sets based on earlier or equivalent nuclear-data sets that have been thoroughly benchmarked are also acceptable. When the recommended cross-section data change, the effect of these changes on the licensee-specific methodology must be evaluated and the fluence estimates updated when the effects are significant.	1.1.2
Cross-Section Angular Representation. In discrete ordinates transport calculations, a P_3 angular decomposition of the scattering cross-sections (at a minimum) must be employed.	1.1.2
Cross-Section Group Collapsing. The adequacy of the collapsed job library must be demonstrated by comparing calculations for a representative configuration performed with both the master library and the job library.	1.1.2
Neutron Source. The core neutron source should account for local fuel isotopics and, where appropriate, the effects of moderator density. The neutron source normalization and energy dependence must account for the fuel exposure dependence of the fission spectra, the number of neutrons produced per fission, and the energy released per fission.	1.2
End-of-Life Predictions. Predictions of the vessel end-of-life fluence should be made with a best-estimate or conservative generic power distribution. If a best estimate is used, the power distribution must be updated if changes in core loadings, surveillance measurements, or other information indicate a significant change in projected fluence values.	1.2
Spatial Representation. Discrete ordinates neutron transport calculations should incorporate a detailed radial- and azimuthal-spatial mesh of ~2 intervals per inch radially. The discrete ordinates calculations must employ (at a minimum) an S_8 quadrature and (at least) 40 intervals per octant.	1.3.1

	Regulatory Position
Multiple Transport Calculations. If the calculation is performed using two or more "bootstrap" calculations, the adequacy of the overlap regions must be demonstrated.	1.3.1
Point Estimates. If the dimensions of the tally region or the definition of the average-flux region introduce a bias in the tally edit, the Monte Carlo prediction should be adjusted to eliminate the calculational bias. The average-flux region surrounding the point location should not include material boundaries or be located near reflecting, periodic, or white boundaries.	1.3.2
Statistical Tests. The Monte Carlo estimated mean and relative error should be tested and satisfy all statistical criteria.	1.3.2
Variance Reduction. All variance reduction methods should be qualified by comparison with calculations performed without variance reduction.	1.3.2
Capsule Modeling. The capsule fluence is extremely sensitive to the geometrical representation of the capsule geometry and internal water region, and the adequacy of the capsule representation and mesh must be demonstrated.	1.3.3
Spectral Effects on RT_{NDT}. In order to account for the neutron spectrum dependence of RT_{NDT}, when it is extrapolated from the inside surface of the pressure vessel to the T/4 and 3T/4 vessel locations using the E > 1-MeV fluence, a spectral lead factor must be applied to the fluence for the calculation of ΔRT_{NDT}.	1.3.3
Cavity Calculations. In discrete ordinates transport calculations, the adequacy of the S_8 angular quadrature used in cavity transport calculations must be demonstrated.	1.3.5
Methods Qualification. The calculational methodology must be qualified by both (1) comparisons to measurement and calculational benchmarks and (2) an analytic uncertainty analysis. The methods used to calculate the benchmarks must be consistent (to the extent possible) with the methods used to calculate the vessel fluence. The overall calculational bias and uncertainty must be determined by an appropriate combination of the analytic uncertainty analysis and the uncertainty analysis based on the comparisons to the benchmarks.	1.4.1, 1.4.2, 1.4.3
Fluence Calculational Uncertainty. The vessel fluence (1 sigma) calculational uncertainty must be demonstrated to be ≤20% for RT_{PTS} and RT_{NDT} determination. In these applications, if the benchmark comparisons indicate differences greater than 20%, the calculational model must be adjusted or a correction must be applied to reduce the difference between the fluence prediction and the upper 1-sigma limit to within 20%. For other applications, the accuracy should be determined using the approach described in Regulatory Position 1.4, and an uncertainty allowance should be included in the fluence estimate as appropriate in the specific application.	1, 1.4.3

	Regulatory Position
FLUENCE MEASUREMENT METHODS	
Spectrum Coverage. The set of dosimeters should provide adequate spectrum coverage.	2.1.1
Dosimeter Nuclear and Material Properties. Use of dosimeter materials should address melting, oxidation, material purity, total and isotopic mass assay, perturbations by encapsulations and thermal shields, and accurate dosimeter positioning. Dosimeter half-life and photon yield and interference should also be evaluated.	2.1.1
Corrections. Dosimeter-response measurements should account for fluence rate variations, isotopic burnup effects, detector perturbations, self shielding, reaction interferences, and photofission.	2.1.2
Response Uncertainty. An uncertainty analysis must be performed for the response of each dosimeter.	2.1.3
Validation. Detector-response calibrations must be carried out periodically in a standard neutron field.	2.2
Fast-Neutron Fluence. The E > 1 MeV fast-neutron fluence for each measurement location must be determined using calculated spectrum-averaged cross-sections and individual detector measurements. As an alternative, the detector responses may be used to determine reaction probabilities or average reaction rates.	2.3
Measurement-to-Calculation Ratios. The M/C ratios, the standard deviation and bias between calculation and measurement, must be determined.	2.3
REPORTING PROVISIONS	
Neutron Fluence and Uncertainties	
Details of the absolute fluence calculations, associated methods qualification and fluence adjustments (if any) should be reported. Justification and a description of any deviations from the provisions of this guide should be provided.	3.1
Calculated multigroup neutron fluences and fluence rates should be reported.	3.2
The value and basis of any bias or model adjustment made to improve the measurement-to-calculation agreement must be reported.	3.2

	Regulatory Position
Calculated integral fluences and fluence rates for E > 1 MeV and their uncertainties should be reported.	3.3
Measured and calculated integral E > 1 MeV fluences or reaction rates and uncertainties for each measurement location should be reported. The M/C ratios and the spectrum averaged cross-section should also be reported.	3.4
The results of the standard field validation of the measurement method should be reported.	3.5
Specific Activities and Average Reaction Rates	
The specific activities at the end of irradiation and measured average reaction rates with uncertainties should be reported.	3.5
All corrections and adjustments to the measured quantities and their justification should be reported.	3.5

TABLE 2. THRESHOLD DETECTORS RECOMMENDED FOR PRESSURE VESSEL DOSIMETRY

	Nominal Threshold (MeV)	Applicable ASTM Standards
$^{237}Np(n,f)FP^{*}$	0.69	E 705 (Ref. 83)
$^{93}Nb(n,n')^{93m}Nb$	0.97	E 1297 (Ref. 84)
$^{238}U(n,f)FP$	1.45	E 704 (Ref. 85)
$^{58}Ni(n,p)^{58}Co$	2.05	E 264 (Ref. 86)
$^{54}Fe(n,p)^{54}Mn$	2.32	E 263 (Ref. 87)
$^{46}Ti(n,p)^{46}Sc$	3.76	E 526 (Ref. 88)
$^{63}Cu(n,\alpha)^{60}Co$	4.65	E 523 (Ref. 89)

* FP indicates fission product.

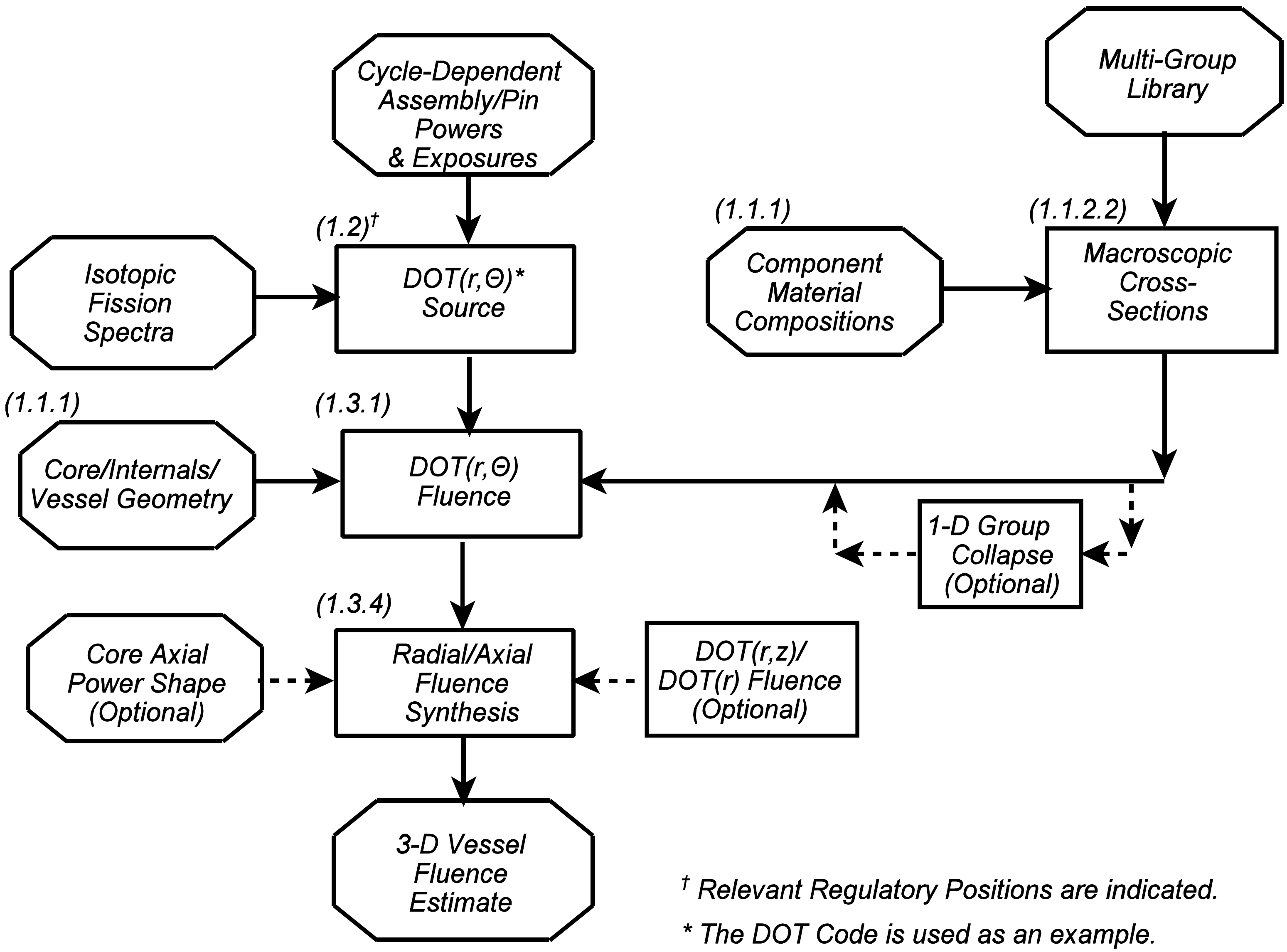

Figure 1. Discrete Ordinates Calculation Methodology

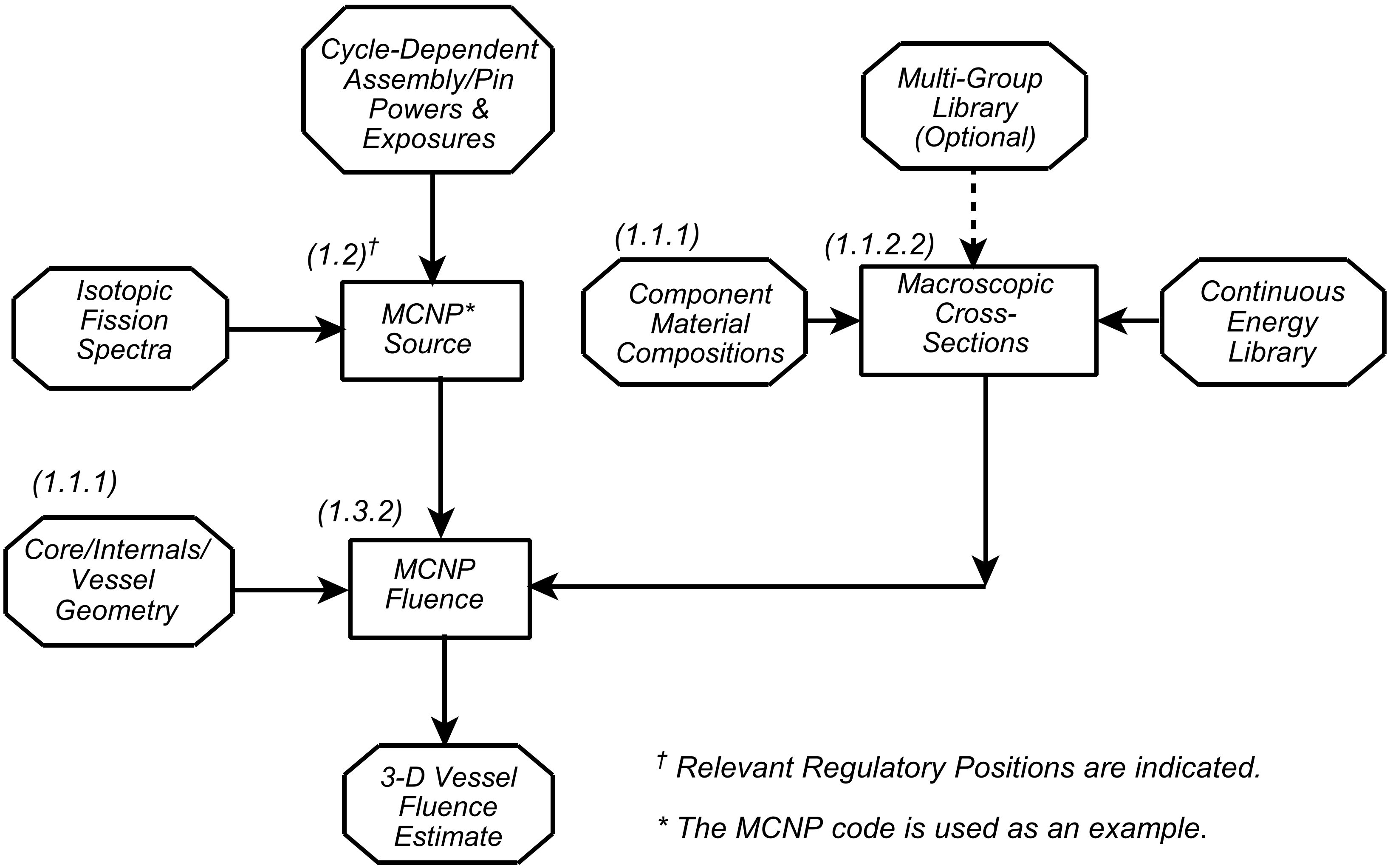

Figure 2. Monte Carlo Calculation Methodology

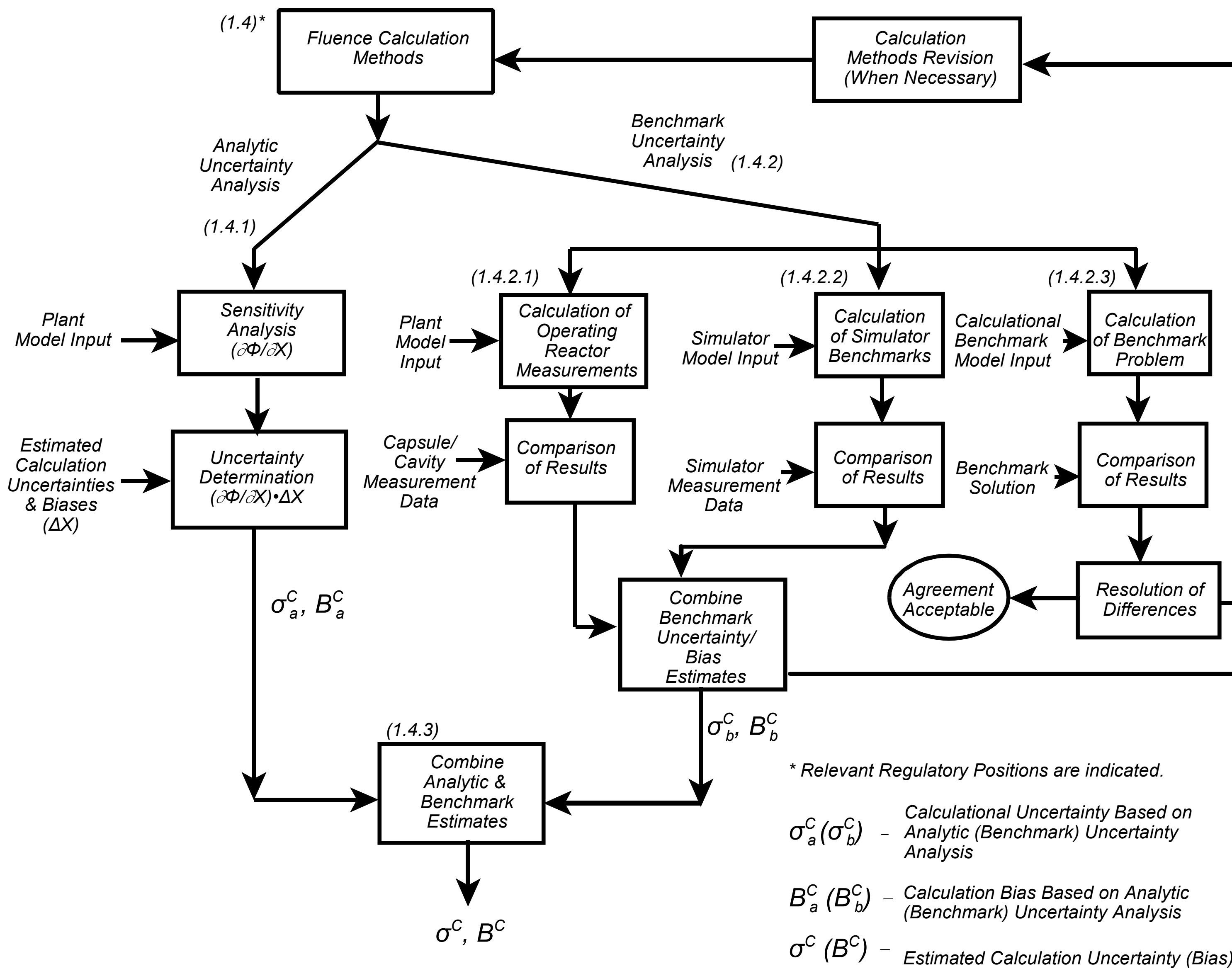

Figure 3. Calculation Methodology Qualification Procedure

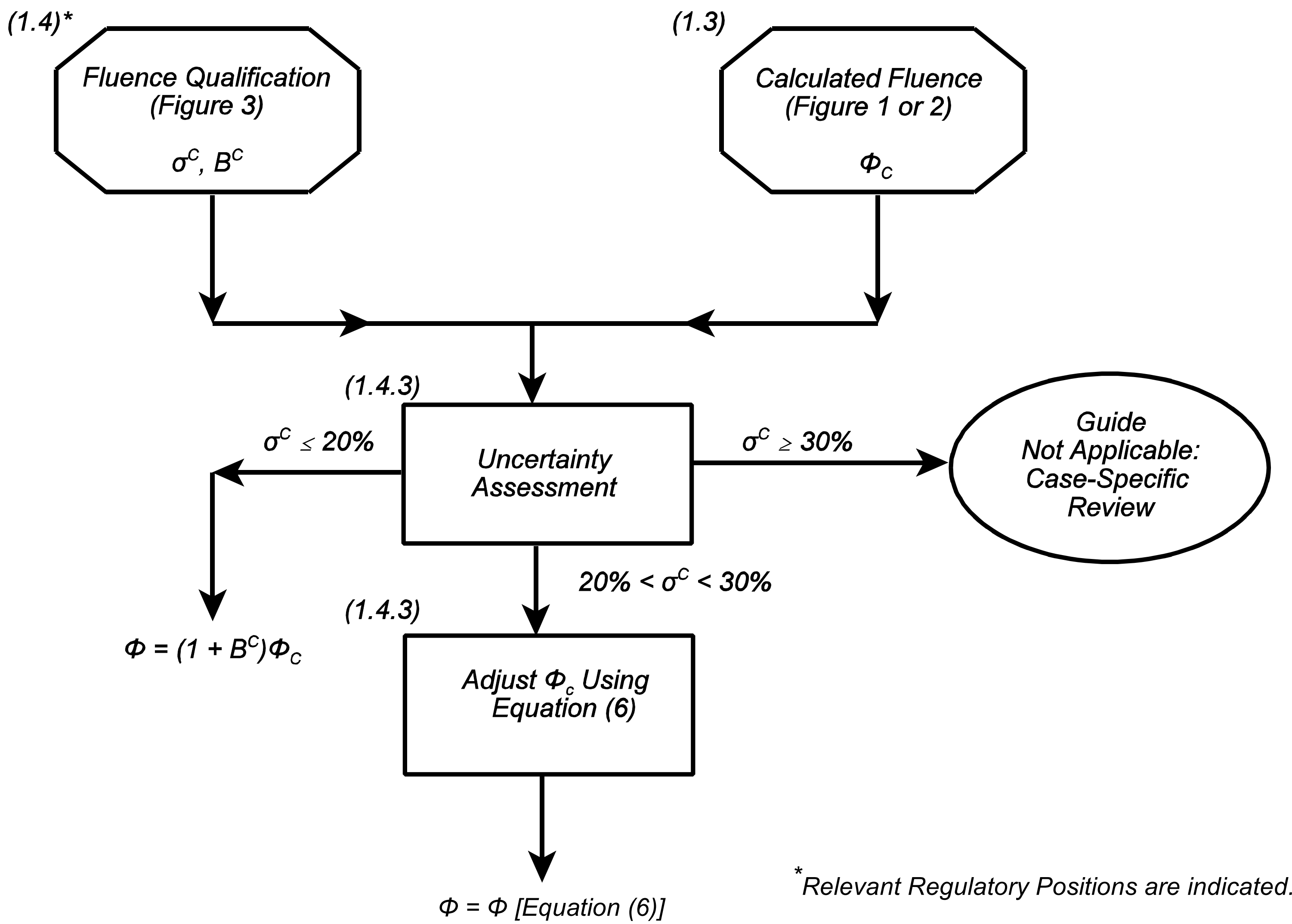

Figure 4. Uncertainty-Dependent Fluence Determination

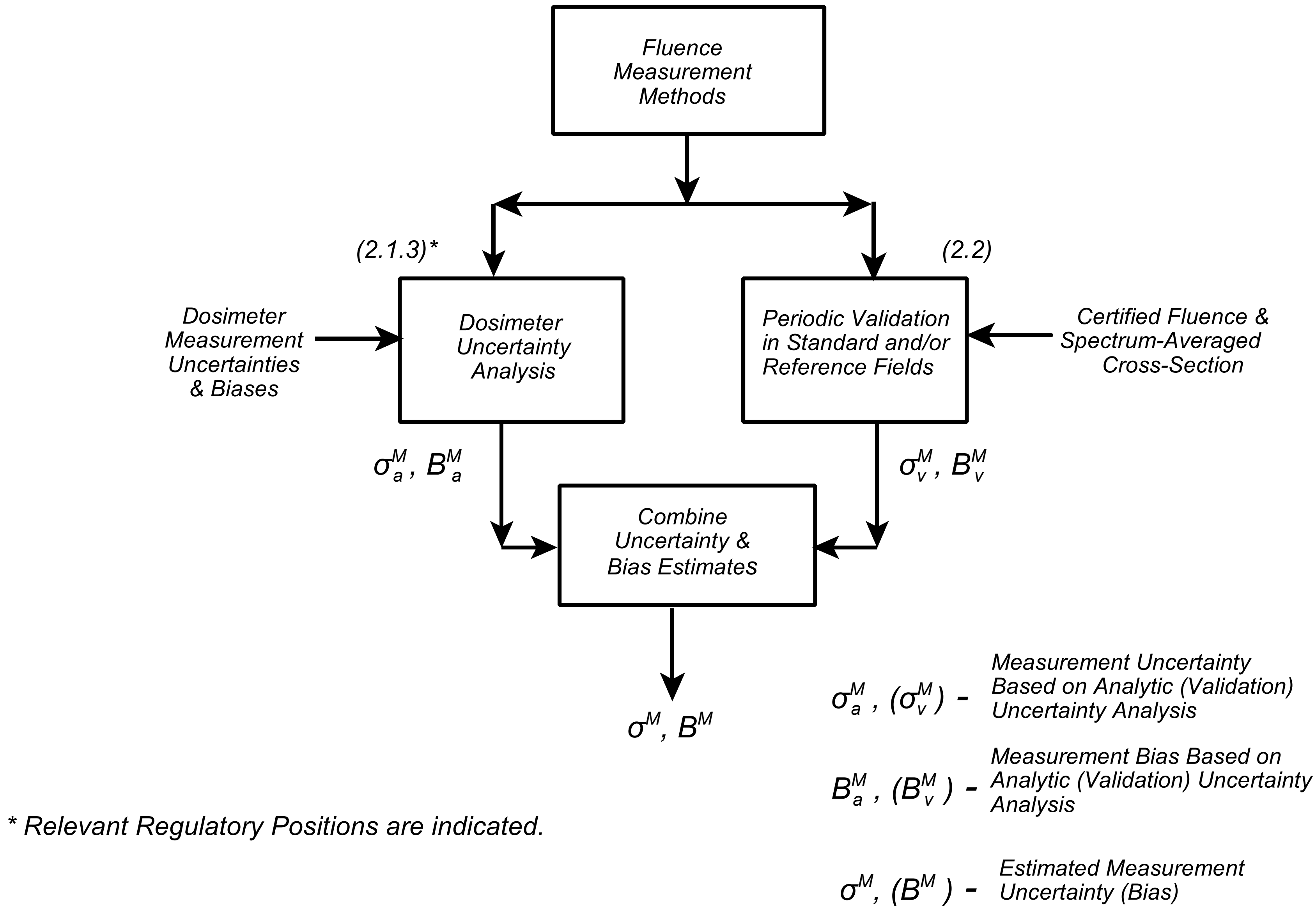

Figure 5. Measurement Qualification Procedure

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[3] Copies may be purchased at current rates from the U.S. Government Printing Office, Post Office Box 37082, Washington, DC 20013-7082 (telephone (202)512-2249 or (202)512-2171); or from the National Technical Information Service by writing NTIS at 5285 Port Royal Road, Springfield, VA 22161. Copies are available for inspection or copying for a fee from the NRC Public Document Room at 11555 Rockville Pike, Rockville, MD; the PDR's mailing address is USNRC PDR, Washington, DC 20555; telephone (301)415-4737; fax (301)415-3548.

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REGULATORY ANALYSIS

1. STATEMENT OF THE PROBLEM

The U.S. Nuclear Regulatory Commission (NRC) has promulgated regulations to ensure the structural integrity of the reactor pressure vessel for light water 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," to 10 CFR Part 50, "Domestic Licensing of Production and Utilization Facilities." These requirements are imposed through 10 CFR 50.60. 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."

To satisfy the requirements of Appendix G; 10 CFR 50.61; Criterion 14, "Reactor Coolant Pressure Boundary," Criterion 30, "Quality of Reactor Coolant Pressure Boundary," and Criterion 31, "Fracture Prevention of Reactor Coolant Pressure Boundary," of Apppendix A, "General Design Criteria for Nuclear Power Plants," to 10 CFR Part 50, methods for determining the fast neutron fluence (E > MeV) are necessary to estimate the fracture toughness of the pressure vessel materials. Appendix H, "Reactor Vessel Material Surveillance Program Requirements," to 10 CFR Part 50 requires the installation of surveillance capsules, including material test specimens and flux dosimeters, to provide data on material damage correlations as a function of fluence.

The neutron fluence is attenuated by several decades between the core and vessel. This attenuation results in a strong sensitivity of the calculated vessel fluence to the physical description of the core and vessel internals and the numerical calculation of the neutron transport, and it makes an accurate determination of the pressure vessel fluence difficult. As a result, a wide range of methods of varying reliability and accuracy have been used to determine the reactor vessel fluence. Consequently, comparisons of measured and calculated fluences have shown varying degrees of agreement, and in some cases conservatisms have been required in licensing analyses to accommodate the observed measurement-to-calculation differences.

Over the past decade, substantial improvements have been made in both the calculation and measurement of the pressure vessel fluence. These improvements have stemmed from both NRC and industry programs. These include the development and improvement of computer codes, calculational models, measurement techniques and basic cross-section data, and the systematic qualification of the fluence methods by comparison to NRC-sponsored benchmark experiments.

These calculation and measurement improvements provide increased accuracy in the fluence determinations that are an essential part of meeting the requirements of Appendices G to 10 CFR Part 50 and 10 CFR 50.61. This is especially important for plants seeking to renew their operating licenses.

The wide variation in fluence calculation methods has resulted in lengthy plant-specific reviews and made it difficult to confirm, during the review process, that the actual fluence is adequately bounded by the various calculational methods used. This calculation and dosimetry guide would provide standardized methods and procedures that would allow these reviews to be greatly simplified, and would improve confidence in the calculated fluence values.

2. OBJECTIVE

The objective of this guide is to provide state-of-the-art calculation and measurement procedures that are acceptable to the NRC staff for determining pressure vessel fluence. These procedures would yield a more accurate and reliable vessel fluence determination than the procedures that have been used in the past. The improved accuracy and realistic assessment of the uncertainty in the calculation would provide assurance that the fluence value is appropriate for use in evaluating compliance with the regulations.

3. ALTERNATIVES

The alternatives to issuing the vessel fluence calculation and dosimetry regulatory guide are as follows:

3.1 Branch Technical Position

The pressure vessel fluence methods provided by the regulatory guide could be included in a branch technical position. However, this is not considered an acceptable alternative since the branch position does not provide the same high level of input and participation, especially public, industry, and the ACRS input, that the vessel fluence analysis requires.

3.2 NUREG-Series Report

The vessel fluence procedures could be published in a NUREG-series report. However, these reports also do not receive the required input and participation from the public, industry, and the ACRS and they are not appropriate for providing regulatory guidance.

3.3 Discussions with Licensees

The detailed fluence calculational and measurement methods that are considered acceptable to the NRC staff could be provided to the licensees through individual reviews and discussions on a case-by-case basis. This alternative is basically the same as the current practice and is equivalent to taking no action. Individual licensee discussions are extremely time-consuming for both the NRC staff and the licensee, they lead to highly individual analyses and reviews, and they do not result in an established standard.

4. COSTS AND BENEFITS

4.1 Benefits

The methods described in this guide may be used for all fluence determinations used in vessel fracture toughness evaluations, including the determination of the fluence used in calculating the pressure vessel material values of RT_{NDT} specified in Appendix G to 10 CFR Part 50 imposed by 10 CFR 50.60, and the values of RT_{PTS} specified in 10 CFR 50.61. The regulatory guide would improve the accuracy and reliability of these evaluations and provide consistent and reliable uncertainty estimates by incorporating state-of-the-art methods and procedures for determining the fluence and the fluence uncertainty. The guide would also ensure the completeness of licensee vessel fluence submittals and improve the efficiency of staff reviews.

The improved fluence determination will provide fluence and uncertainty estimates that are more reliable and understandable. Thus, it will provide more reliable and accurate information for the PTS screening criteria of 10 CFR 50.61 and improve application of this rule. In this regard it is noted that, for a pressure vessel near the PTS screening criteria of 10 CFR 50.61, a 25% reduction in calculated end-of-license fluence, which is typical of existing uncertainties, will reduce the calculated vessel failure frequency by approximately a factor of three.

4.2 NRC Costs

The NRC costs for reviewing fluence-related submittals would be reduced substantially by the issuance of this guide. For estimating the costs, it is assumed that of the ~60 PWRs, half have a RT_{PTS} within ~40°F of the PTS screening criterion (or other temperature limit) at the end of license and will require a detailed review. Assuming each submittal requires a staff week and only half of the PWR owners submit revised fluence analyses, the total NRC cost is approximately 15 staff weeks. If the licensees used the methods given in the guide, this cost could be reduced to approximately 2 staff weeks.

4.3 Licensee Costs

Increased costs to the licensee would result from changes in the fluence calculation and measurement procedures. The calculational costs would be one-time costs and have been estimated in Table RA-1. The licensee costs resulting from the changes in the measurement procedures have been estimated in Table RA-2.

5. DECISION RATIONALE

It is recommended that the proposed regulatory guide be issued because (1) a high level of participation of the NRC, ACRS, industry, and the public is reflected in the guide, (2) a methodology standard would be established, and (3) inefficient use of NRC staff and licensee resources during the review process would be eliminated.

The alternatives identified above for providing acceptable fluence methods to the licensees do not provide the advantages listed above associated with issuance of this regulatory guide. In particular, while these alternative approaches may result in the same or slightly increased cost to licensees, they result in a highly inefficient use of the NRC staff resources. The alternatives to a regulatory guide are therefore judged to be unacceptable.

TABLE RA-1. ADDITIONAL LICENSEE CALCULATION COSTS

Calculation Tasks		Staff Weeks
(a)	Modifications to calculational models	+ 2
(b)	Additional calculation benchmarking and qualification	+ 8
(c)	Calculation uncertainty analysis	+ 3
(d)	Calculation documentation and reporting	+ 2
(e)	Reduced Licensing Activities*	- 4
	Total Additional Licensee Cost	+11 staff weeks

TABLE RA-2. ADDITIONAL LICENSEE MEASUREMENT COSTS

Measurement Task		Staff Weeks
(a)	Additional Quality Control	+ 1
(b)	Dosimeter Response Corrections	+ 1
(c)	Periodic Detector Calibration	+ 2
(d)	Response Uncertainty Analysis	+ 2
(e)	Additional Measurement Documentation and Reporting	+ 1
	Total Additional Licensee Cost	+ 7 staff weeks

* This estimate reflects the reduced licensee costs from not attending meetings with the NRC staff and responding to questions as a result of following the procedures in the regulatory guide.

BACKFIT ANALYSIS

The regulatory guide does not require a backfit analysis as described in 10 CFR 50.109(c) because it does not impose a new or amended provision in the Commission rules or a regulatory staff position interpreting the Commission rules that is either new or different from a previous applicable staff position. In addition, this regulatory guide does not require the modification or addition to systems, structures, components, or design of a facility or the procedures or organization required to design, construct, or operate a facility. Rather a licensee or applicant can select a preferred method for achieving compliance with license or the rules or the orders of the Commission as described in 10 CFR 50.109(a)(7). The regulatory guide provides the opportunity to use the methods described in the guide for all fluence determinations used in vessel fracture toughness evaluations, including the determination of the fluence used in calculating the pressure vessel materials' values of RT_{NDT} for use in 10 CFR Part 50, Appendix G (imposed by 10 CFR 50.60), and the values of RT_{PTS} in accordance with 10 CFR 50.61.