U.S. NUCLEAR REGULATORY COMMISSION
OFFICE OF NUCLEAR REGULATORY RESEARCH

DRAFT REGULATORY GUIDE

September 1993
Division 1
Task DG-1025

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DRAFT REGULATORY GUIDE DG-1025

CALCULATIONAL AND DOSIMETRY METHODS FOR DETERMINING PRESSURE VESSEL NEUTRON FLUENCE

FOR COMMENT

This regulatory guide is being issued in draft form to involve the public in the early stages of the development of a regulatory position in this area. It has not received complete staff review and does not represent an official NRC staff position.

Public comments are being solicited on the draft guide (including any implementation schedule) and its associated regulatory analysis or value/impact statement. Comments should be accompanied by appropriate supporting data. Written comments may be submitted to the Regulatory Publications Branch, DFIPS, Office of Administration, U.S. Nuclear Regulatory Commission, Washington, DC 20555. Copies of comments received may be examined at the NRC Public Document Room, 2120 L Street NW., Washington, DC. Comments will be most helpful if received by December 17, 1993.

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A. INTRODUCTION

The U.S. Nuclear Regulatory Commission (NRC) has promulgated regulations that ensure the structural integrity of the reactor pressure vessel for light-water-cooled power reactors. Specific fracture toughness requirements for normal operation and for anticipated operational occurrences for power reactors are set forth in Appendix G, "Fracture Toughness Requirements," of 10 CFR Part 50, "Domestic Licensing of Production and Utilization Facilities." Additionally, in response to concerns over potential pressurized thermal shock (PTS) events in pressurized water reactors (PWRs), the NRC issued 10 CFR 50.61, "Fracture Toughness Requirements for Protection Against Pressurized Thermal Shock Events."

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," of 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 fracture toughness of pressure vessel materials is related to a parameter called the material's "reference nil-ductility temperature," or simply reference temperature, and is denoted as RT_{NDT}. The RT_{NDT} is defined by a correlation of the fast neutron 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 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}. Cases of unusual plant characteristics or factors that require different methods and assumptions will be considered on a plant-specific basis.

Compliance with this guide is not a regulatory requirement of the USNRC. However, if a licensee elects to use the final version of 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.

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

The public reporting burden for this collection of information over and above the burden previously required for this activity is estimated to be an average of 880 hours per respondent, including the time for reviewing instructions, searching existing data sources, gathering and maintaining the data needed, and completing and reviewing the collection of information. Send comments regarding this burden estimate or any other aspect of this collection of information, including suggestions for further reducing the reporting burden, to the Information and Records Management Branch (MNBB-7741), U.S. Nuclear Regulatory Commission, Washington DC 20555; and to the Desk Officer, Office of Information and Regulatory Affairs, NEOB-3019, (3150-0011), Office of Management and Budget, Washington, DC 20503.

B. DISCUSSION

The methods and assumptions described in this guide are for the calculation and measurement of vessel fluence for core and vessel geometrical and material configurations that are typical of current PWR and BWR power reactor designs. This guide does not address the determination of surveillance specimen material properties or the correlation between material properties and neutron fluence. The methodology presented is intended as a best-estimate, rather than a bounding or conservative, fluence determination. When required, for example, in the RT_{PTS} correlation called for in 10 CFR 50.61, an uncertainty margin should be included separately. 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 entire damage fluence spectrum (from 0.1 to 15 MeV) and are generally applicable to other exposure units, such as displacements per atom (dpa).

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 method's qualification by comparison to measurement must be made to ensure a reliable and accurate vessel fluence determination. In this qualification, calculation-to-measurement comparisons are used to identify biases in the calculations and to provide reliable estimates of the fluence uncertainties.[1] When the measurement data are of sufficient quality and quantity, the comparisons to measurement may be used to (1) determine the effect of the various modeling approximations and any calculational bias and, if appropriate, (2) modify the calculations by application of a bias or by model adjustment or both. As an additional methods qualification, the sensitivity of the calculation to the important input and modeling parameters must be determined and used to provide an independent estimate of the overall calculational uncertainty. The prediction of the vessel fluence must be made by an absolute fluence calculation 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.

As an indication of current practice, selected codes and cross-section libraries are listed in the references; however, it is the responsibility of

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[1]In this guide, the term "uncertainty" refers to the random difference between the estimated fluence (based on either calculation or measurement) and the true fluence value. The term "bias" refers to the nonrandom or systematic difference between the fluence estimate and the true value.

the licensee to demonstrate their acceptability in a specific application. Additional material related to the determination of pressure vessel fluence and material damage, but considered outside the scope of this guide, is contained in other regulatory guides and ASTM and ANSI/ANS Standards.

C. REGULATORY POSITION

1. NEUTRON FLUENCE CALCULATIONAL METHODS

The calculational methodology for estimating reactor vessel fluence should agree with available (benchmark quality) measurements to within ~20% (1 sigma)[2] for the determination of RT_{PTS} as described in 10 CFR 50.61. For other applications, the accuracy should be determined using the approach described in Regulatory Position 1.4 as an uncertainty allowance included in the fluence estimate, as appropriate for the specific application.

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 descriptions of the pressure vessel, core, and internals; material compositions; regional temperatures; and geometry. The geometrical input data include the dimensions and locations of the fuel assemblies, reactor internals (baffle, 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 shielding, including water content, rebar, and steel. The data should be based on documented and verified as-built plant-specific dimensions and materials. The isotopic compositions of important constituent nuclides within each region should be based on measured data. In the absence of plant-specific information, "generic" compositions and

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[2] It is recognized that a lower fluence calculational uncertainty may be needed as the vessel approaches its projected end of life.

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 1.4.3). The determination of the concentrations of the major isotopes responsible for the fluence attenuation (e.g., iron and water) should be emphasized. The water number densities should be based on plant full-power operating temperatures and pressures, as well as standard steam tables. The data should account for axial and radial variations in water density caused by temperature differences and the presence of in-channel and downcomer voids in the case of BWRs.

1.1.2 Cross-Sections

The calculational method to estimate vessel damage fluence should use the neutron cross-sections over the energy range from ~0.1 MeV to ~15 MeV and apply the latest version of the Evaluated Nuclear Data File (ENDF/B-VI). These data have been thoroughly reviewed and tested relative to experimental benchmarks.[3] Cross-section sets based on earlier or equivalent nuclear data sets that have been thoroughly benchmarked for a specific application may be used for that specific application. However, when the reevaluated cross-section data change, the effect of these changes on the licensee-specific methodology must be evaluated and the fluence estimates updated as appropriate.

Since the discrete ordinates transport codes that are used to determine the neutron fluence employ a multigroup approximation, the basic data contained within the ENDF files must be 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 angular 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

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[3]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 ~20% underprediction of the vessel inner-wall fluence and ~35% underprediction of the cavity fluence (Refs. 1-3). Updated ENDF/B-V iron cross-section data (Ref. 4) have been demonstrated to provide a more accurate determination of the flux attenuation through iron (Refs. 2, 3) and are strongly recommended. These new iron data are included in ENDF/B version VI.

total cross-section) should be included to preserve the prescribed accuracy in attenuation characteristics.

The construction of the multigroup library involves the selection of a problem-independent multigroup "master" library containing data for all required isotopes. This library should include a sufficiently large number of groups (~100-200) so 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 should be used for typical LWR configurations. Several libraries satisfying these provisions are available from RSIC -- the Oak Ridge National Laboratory Radiation Shielding Information Center (Refs. 5, 6, 7).

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. 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 macroscopic multigroup (<50) cross-sections based on the region-specific isotopic compositions. This library should include ~20 energy groups above ~0.1 MeV. It is the responsibility of the licensee to demonstrate the adequacy of the "job" library, and this may be accomplished 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.

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

1.2 Core Neutron Source

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

The spatial dependence of the source should be from core-follow calculations or from measured data. Core-follow calculations should be performed with a three-dimensional (3-D) coarse mesh simulator code and provide the relative power over large rectangular nodes (typically ~20 x 20 x 30 cm in the x, y, and z dimensions, respectively). Plant process computers provide similar power distribution data obtained from in-core instrumentation.

The core neutron source should be determined by the power distribution, which varies significantly with fuel burnup, power level, and the fuel management scheme. The detailed state-point dependence should be accounted for (Refs. 10, 11); however, if this is not feasible, an approximate averaging over the operating power distribution is acceptable and may be obtained by (1) averaging representative power distributions within the cycle or (2) assuming the cycle-average assembly power distribution is well approximated by the accumulated exposure distribution at the end of the cycle.

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

The peripheral assemblies, which contribute the most to the vessel fluence, have strong radial power (source) gradients, and these gradients should not be neglected since this will result in an overprediction of the fluence if the average assembly power is used (Ref. 12). The pin-wise source distribution should be used for best-estimate calculations and the peripheral assembly pin-wise source data should be obtained from quarter-core fine mesh calculations in which each fuel-pin cell is explicitly represented. The pin-wise power distribution and the nodal power should be combined to define the absolute power distribution in the assembly.

The energy dependence of the source (i.e., the spectrum) and the normalization of the source to the number of neutrons per megawatt must account for the fact that changes in the isotopic fission fractions with fuel exposure result in variations in the energy-dependent 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. 13, 14). This effect should also be accounted for in 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. 15-18). The harder spectrum in the BWR fuel regions having a high void fraction will have a similar effect on the isotopic fission fractions and on the neutron source normalization and spectrum.

The horizontal core geometry may be described using an (r,θ) representation of the nominal plane. A planar octant representation is acceptable for the octant symmetric fuel-loading patterns typically employed in LWRs. Fuel-loading patterns that are not octant symmetric may be represented in octant geometry using the octant having the highest fluence. To accurately represent the important peripheral assembly geometry, a θ-mesh of 40-80 angular intervals 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 (Refs. 19, 20). Reference 20 is particularly useful if the radial mesh is a function of θ).

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

Determination of the 3-D fluence at the vessel using (r,z) and (r) geometry calculations may also be appropriate, (see Regulatory Position 1.3.2). If these calculations are used to provide an axial correction factor, the source specification may be less stringent if consistent sources are used.

1.3 Fluence Calculation

1.3.1 Transport Calculation[4]

The transport of neutrons from the core to locations of interest in the pressure vessel should be determined with a two-dimensional discrete ordinates[5] transport program (Refs. 22-24) in (r,θ) and, when appropriate, in (r,z) and (r) geometries.[6] When calculating a horizontal plane of the core/vessel geometry in which the rectangular (x,y) geometry of the core boundary and the cylindrical (r) geometry of the vessel are mixed, a more accurate description is provided by the variable (r,θ) mesh option (Ref. 23) and may be applied.

The azimuthal (θ) mesh using 40-80 intervals over an octant in (r,θ) geometry in the horizontal plane must 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 coarser for assemblies more than approximately two assembly pitches removed from the core-reflector interface. In excore regions, a spatial mesh that ensures the flux in any group changes by less than a factor of ~2 between adjacent intervals should be applied, and 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 should 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,

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

[5]The discrete ordinates transport method is generally used for the calculation of pressure vessel fluence. It is recognized, however, that alternative methods such as Monte Carlo are available, and analyses performed with these methods will be reviewed on a case-by-case basis.

[6]If DOT (Ref. 23) 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, the default θ = 0.9 value is adequate.

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, 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 over-lapping 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 applied as the left-hand 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 $\lesssim 0.001$ 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.[6] The adequacy of the spatial mesh and angular quadrature, as well as the convergence, must be demonstrated by tightening the numerics until the resulting changes are negligible. In discrete ordinates codes, the spatial mesh and the angular quadrature should be refined simultaneously. In many cases, these evaluations can be adequately performed with a one-dimensional model.

In performing calculations of surveillance capsule fluence (Regulatory Position 1.4), it should be noted that the capsule fluence is extremely sensitive to the geometrical 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. In addition, 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. To account for the neutron spectrum dependence of RT_{NDT} when the fluence is extrapolated from the surveillance capsule or 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) should be applied for the calculation of ΔRT_{NDT} (Ref. 25).

The transport calculations may be performed in either the forward or adjoint modes. When several transport calculations are needed for a specific geometry, assembly importance factors may be precalculated by either performing calculations with a unit source specified in the assembly of interest or by performing adjoint calculations. The adjoint fluxes 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 weighing the assembly importance factors with the source distribution of interest, the vessel (or capsule) fluence may be determined without additional transport calculations, assuming the in-vessel geometry and material remain the same.

When fluence reduction schemes have introduced strong axial or azimuthal heterogeneities into the source (e.g., an axially zoned replacement of fuel with stainless steel for fluence reduction), 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 (Regulatory Position 1.3.2).

1.3.2 <u>Synthesis of the 3-D Fluence</u>

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

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

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

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

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

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

where $\phi_g(r)$ and $\phi_g(r,z)$ are one- and two-dimensional group-g flux solutions, respectively, for a cylindrical representation of the geometry that preserves the important axial source and attenuation characteristics (Ref. 26). The (r,z) plane should correspond to the azimuthal location of interest (e.g., peak vessel fluence or dosimetry locations) or a conservatively θ-averaged (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 2 is applicable when (a) the axial source distribution for all important peripheral assemblies is approximately the same or is bounded by a conservative axial power shape and (b) the attenuation characteristics do not vary axially over the region of interest. In addition, since the axial fluence distribution tends to flatten as it propagates from the core through the pressure vessel, this approximation will tend to overpredict axial fluence maxima and underpredict minima. This nonconservative underprediction could be large near the top and bottom reflectors, as well as if minima are strongly localized as occurs in some fluence reduction schemes.

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

In summary, an (r,θ) geometry fluence calculation and a knowledge of the peripheral assembly axial power distribution are needed when using Equation 2. It will result in fluence overpredictions near the midplane at relatively large distances from the core, e.g.,in the cavity, and underpredictions at axial locations beyond the beltline at relatively large radial distances from the core. 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 3; thus, it is generally more accurate in preserving the integral properties of the three-dimensional fluence. Both Equation 2 and Equation 3 assume separability between the axial and azimuthal fluence calculations, which is generally 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

$$\phi_g(r,\theta,z) = \sum_{i=1}^{N} a_i\phi_{gi}(r,\theta)\phi_{gi}(r,z)/\phi_{gi}(r) \quad \textbf{(Equation 4)}$$

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

1.3.3 Cavity Fluence Calculations

Accurate cavity fluence calculations are relied on to analyze dosimetry located in the reactor cavity and to determine the fluence accumulated by the reactor support structures (Ref. 28). 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 vessel and the resulting increased sensitivity to the iron inelastic cross-section and (b) the possibility of neutron streaming (i.e., strong directionally dependent) effects in the cavity. Because of this increased sensitivity to the iron cross-sections, the most accurate ENDF/B-VI cross-section data must be used for cavity fluence calculations.[3] Typically, the width of the cavity together with the close-to-beltline locations of the dosimetry capsules result in minimal cavity streaming effects, and an S_8 angular quadrature is acceptable. However, when off-beltline locations or narrow cavities are analyzed, the adequacy of the S_8 quadrature 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.[6]

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 described in Reference 3. The measured energy spectrum for a typical PWR cavity is given in Reference 29. 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 the extrapolation of the two measurements. Measurements performed in reactor cavities are described in References 1 and 2.

1.4 Qualification

The neutron transport calculation must be qualified, and fluence uncertainty estimates must be determined. The neutron fluence 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, 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, (2) the comparison with benchmarks and plant-specific data, and (3) the estimate of uncertainty in calculated fluence.

1.4.1 Calculational 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 (cross-sections and fission spectrum)

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

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

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

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

This list is not necessarily exhaustive, and other uncertainties that are specific to a particular reactor or a particular calculational method should be considered. In typical applications, the fluence uncertainty is dominated by a few uncertainty components, such as the geometry, which are usually easily identified and substantially simplify the uncertainty analysis.

The effect of the significant component uncertainties should be determined by a series of sensitivity calculations in which the calculational model input data and modeling assumptions are varied and the numerical effect on the calculated fluence is determined. (A typical sensitivity would be ~10-15% decrease in vessel >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 expected total fluence uncertainty (i.e., standard deviation). The independent random uncertainties should be combined in a statistical or root-mean-square fashion, and the systematic errors (or biases) should be combined algebraically, recognizing the sign of each contribution. The component uncertainties should be based on measurement or on the acceptable deviations included in the design specifications. The sensitivity calculations may be performed in one dimension when the model sensitivity does not involve a detailed two-dimensional representation.

A sensitivity analysis, in which the influence of each of these uncertainties on the calculated group fluences has been considered, is included in References 1 and 2 for several power reactor benchmarks. Since the uncertainties used in these analyses are common to many pressurized water reactors, the uncertainties (including correlations) may be used as initial uncertainty estimates. These variance estimates should be modified as additional experience is obtained or if the reactor of interest differs substantially from the benchmark reactor. The referenced benchmark sensitivity analysis provides guidance for such modifications.

1.4.2 Comparisons with Benchmark and Plant-Specific Data

Calculational methods must be validated by comparison with measurement and calculational benchmarks. The fluence calculational methods should be validated against (1) a power reactor benchmark that provides in-vessel surveillance capsule dosimetry or ex-vessel cavity measurements or both, (2) a pressure vessel simulator benchmark that provides measurements at the inner surface and at the T/4 and 3T/4 positions within the vessel (see Regulatory Position 2.5 for a discussion of such benchmarks), and (3) available fluence calculational benchmarks. The results of the validation should include comparisons of reaction rates, fluences, and group fluxes for the locations of interest (Refs. 30, 31). Any adjustments made to the calculations should be justified and reported.

1.4.2.1 Surveillance Capsule Measurements. The vessel surveillance capsules provide the most important source of fluence measurement data, and calculations should be performed for capsule measurements of the specific reactor or reactors of similar design. These measurements have the advantage of including the as-built materials and geometry and the actual operating conditions. In calculating the capsule dosimeter reaction rates, the latest dosimeter cross-sections and a detailed modeling of the capsule materials and geometry should be employed.

1.4.2.2 Operating Reactor Fluence Measurements. Several fluence measurements for operating reactors may be used for methods and data qualification (examples are provided in Reference 32). These measurements include the three-dimensional geometry and operating conditions, and in some cases, include both in-vessel and ex-vessel measurements. An especially accurate determination of the fluence attenuation through the vessel (e.g., to the T/4 and 3T/4 locations) can be obtained when both in-vessel and cavity dosimetry are available.

1.4.2.3 Calculational Benchmarks. The two-dimensional vessel fluence standard problem provided by the NRC (Ref. 33) and the one-dimensional shielding calculational benchmarks proposed by the American Nuclear Society Benchmark Committee (Ref. 34) may be used for methods qualification. In these benchmarks, the geometry, materials, space, and energy-dependent source are fixed by the problem specification. The calculation of these problems provides a detailed test of the cross-sections and various aspects of the transport calculations, such as the spatial mesh, quadrature, and convergence criteria.

The methods used to calculate the benchmarks and plant-specific data must be consistent (to the extent possible) with those used in the vessel fluence calculations. 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 reference calculations.

Differences between measurements and calculations should be consistent with the combined uncertainty estimates for the measurements and calculations. The calculated and measured reaction rates (using the methods described in Regulatory Positions 1.1 through 1.3) typically agree with the measurements to within about 20% for surveillance capsules and 30% for cavity dosimetry. Deviations outside these uncertainty limits must be investigated and the calculations or measurements modified when the cause of the deviation is identified (Refs. 35-42).

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

1.4.3 Estimate of Fluence Calculational Uncertainty

The overall fluence calculation uncertainty must be determined by an appropriate combination of (1) the analytic uncertainty analysis (Regulatory Position 1.4.1) and (2) the uncertainty estimate based on the comparisons to benchmarks (Regulatory Position 1.4.2). The fluence accuracy requirements are generally application specific; however, a vessel fluence uncertainty of 20% (1 sigma) is acceptable for RT_{PTS} determination. For PTS applications, if the benchmark comparisons indicate an uncertainty greater than ~20%, the calculational model must be adjusted or a bias applied to bring the agreement within this range. The uncertainty determined may be combined with the calculated best-estimate fluence to provide a high-probability upper bound on the vessel fluence. For example, assuming the uncertainty is distributed normally, the 95% probability upper tolerance fluence limit is the best-estimate value plus 1.65 sigma.

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

2. NEUTRON FLUENCE MEASUREMENT METHODS

Dosimetry measurements provide an independent estimate of the neutron fluence to confirm neutron transport calculations. This 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 these measurements to obtain a complete analysis, a reliable uncertainty assessment, and proper documentation are described in this section. Standard neutron field validation and sites for placing updated dosimetry are also described.

The fluence measurement provisions of this section are summarized in Section B of Table 1, Summary of Regulatory Positions on Calculation and Dosimetry.

2.1 Measurement Procedures

Measurement methods in power reactor dosimetry include passive integral detectors, which are typically activation detectors, and solid state track recorders. The most frequently used detectors respond to neutrons with energies above a characteristic reaction threshold. These detectors should be selected with substantial 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 must provide adequate spectrum coverage. A common set of fast neutron integral detectors that may be employed in these packages is listed in Table 2. Taken together with a low-energy detector such as cobalt (to measure the thermal neutron fluence and determine interference from low-energy activations), the 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 E844 (Ref. 43) and E1005 (Ref. 44) should be followed (see Table 2).

Application of activation detectors involves measurement elements that must be carefully controlled and documented to establish reasonable uncertainty estimates. Where applicable, procedures in ASTM Standards E181 (Ref. 45) and E1005 (Ref. 44) 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 Isotopic Composition. The dosimeter materials should be pure enough to ensure there is no significant error in the response of the dosimeter from extraneous activities. Cursory specifications of materials regarding impurities are often unreliable. Specifically, fissile residuals in Np-237 and U-238 and minute amounts of cobalt in copper (fractional parts per million) should be determined by mass spectrography or radioactivation analysis.

2.1.1.2 Encapsulation. The detector capsule's designs must taken into account possible activation interference and neutron spectrum perturbation. Thermal neutron shields that eliminate interference from thermal neutron reactions in some detectors must be designed to accommodate radiation heating and should be placed apart from low-energy detectors (see ASTM Standard E844, Ref. 43).

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

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

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

2.1.2 Detector Response Measurements

In order to allow comparison of the dosimetry measurements with the neutron transport calculation, two detector response parameters must be reported: the measured reaction probability (disintegrations per nucleus) and the measured average reaction rate (disintegrations per second, per nucleus). Corrections should be included for time-history, detector response perturbations, interfering reactions, and, when applicable, burnup and photofission. In order to allow an accurate treatment of the time-history effects, the half-lives of the dosimeter activation products should be considered (Ref. 44). Photofission corrections can vary considerably (from 2 to 15%) depending upon the location of the surveillance capsule and type of reactor. Fission yields should be those specified in the relevant ASTM standards. 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 along with any other effects that have a significant impact on the measurements. This is especially important because pressure vessel surveillance dosimetry often involves comparison of measurements carried out by different organizations over long periods of time.

The "specific activity at end-of-irradiation" (given in disintegrations per second, per nucleus, and divided by the fission yield where appropriate) is an additional detector response parameter that should be reported. This directly measured quantity has the advantage of not involving irradiation-related corrections that are somewhat uncertain and are often subject to re-evaluation.

2.1.3 Uncertainty Estimates

There must be at least one uncertainty table that states the component uncertainties contributing to each detector response. The entries in the table should be stated as standard deviations, upper bounds, or appropriate fractions of the correction. Each entry should be described, along with the method by which the entries are combined to obtain a total uncertainty. This accounting of uncertainties provides a reliable estimate of the measurement accuracy and helps establish the extent to which the analysis of the measurement is complete.

2.2 Validation in Standard and Reference Neutron Fields

To ensure long-term measurement consistency and confirm measurement uncertainties, dosimetry measurements must be performed periodically in one or more well-characterized neutron fields. The Materials Dosimetry Reference Facility (MDRF) and fission neutron sources may be used for this purpose (Ref. 54). Neutron field referencing may be used as a detector response calibration.

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, 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 calibration factor. Alternatively, when the bias factor is not too large it may be taken as an indicator of the overall accuracy of the dosimetry measurement system for the detector involved.

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

2.3 Fluence Determination from Detector Measurements

A fast neutron fluence (E > 1 MeV) should be obtained for each detector as the quotient of the measured reaction probability and a calculated spectrum-averaged cross-section (truncated at 1 MeV) from the neutron transport calculation. These measured fluences and a suitably weighted average fluence must be reported along with an uncertainty that is an appropriate combination of the entries in the uncertainty table (described in Regulatory Position 2.1.3) combined with whatever cross-section uncertainty is available from the calculation. An alternative to deriving a neutron fluence from the detector responses is to directly compare measured reaction probabilities or average reaction rates with results from the neutron transport calculation.

Whatever alternative is taken, the C/E ratios[7] for these quantities must be reported along with the measurement uncertainties. When the C/E ratios differ from unity by less than the assigned uncertainties, they validate the calculation to within the accuracy of the measurements. If the C/E ratios differ from unity by more than ~20% for in-vessel dosimetry or ~30% for cavity dosimetry, the measurement and calculation must be reexamined (see Regulatory Position 1.4.2). If an individual detector is declared suspect, it may with

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[7]It should be noted that these C/E ratios differ from those discussed in the previous section in that they refer to a comparison of in-vessel fluences rather than calibration fluences.

justification be disregarded or given reduced weight. This procedure must be documented.

2.4 Sites for Updated Dosimetry

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

2.5 Experimental Benchmarks for Validating Calculations

A series of pressure vessel benchmark experiments were performed in the 1980s (Ref. 58) to test and validate particular aspects of neutron transport calculations and improve pressure vessel surveillance dosimetry measurements (Refs. 3, 27, 32, 59-61). Specific designs were incorporated into the experimental facilities to address calculational problems. Fluence transport calculations were carried out by several laboratories, and dosimetry measurements using different techniques were compared to provide experimental results with well-known and documented uncertainties. These benchmarks may be used to define calculational uncertainties and to test nuclear data, transport calculation methods, and the simulation of three-dimensional geometry.

The benchmark experiments were carried out in three types of configurations:

2.5.1 Pressure Vessel Simulator Mockup Experiments

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

2.5.2 Power Reactor Mockup Experiments

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

2.5.3 Cavity Mockup Experiments

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

3. REPORTING

This guide does not specifically contain reporting requirements. However, 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. The specific regulatory positions on reporting are given in this section and are summarized in Table 1.

3.1 Neutron Fluences and Uncertainties

3.1.1 Fluence Methods

The methods used to calculate the integral and multi-group fluences and fluence rates and associated methods qualification should 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.

3.1.2 Multigroup Fluences

The calculated absolute multigroup neutron fluence and fluence rates and their uncertainties at the peak wall (or other limiting) location and at the T/4 and 3T/4 positions within the pressure vessel should be reported. The multigroup energy boundaries should be included. The value and basis for any adjustments made to the calculated vessel fluences based on comparisons to measurement should be described in detail.

3.1.3 Integral Fluences

The calculated [E > 1 MeV] and [E > 0.1 MeV] integral fluences and fluence rates, together with the uncertainties at the vessel inner wall locations, should be reported. The measured [E > 1 MeV] integral neutron fluences and uncertainties for each measurement location should be reported. The results of the measurement validation should also be described. If thermal neutron fluence rate measurements have been performed, these should be reported together with the uncertainty and thermal shield material.

3.2 Specific Activities and Average Reaction Rates

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

D. IMPLEMENTATION

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

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

TABLE 1. SUMMARY OF REGULATORY POSITIONS ON CALCULATION AND DOSIMETRY

	Regulatory Position
FLUENCE CALCULATION METHODS	
Fluence Determination. Absolute fluence calculations, rather than extrapolated fluence measurements, should 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 material reaction 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 should be evaluated and the fluence estimates updated.	1.1.2
Cross-Section Angular Representation. A P_3 angular decomposition of the scattering cross-sections (at a minimum) should be employed.	1.1.2
Cross-Section Group Collapsing. The adequacy of the collapsed job library should be demonstrated. This may be accomplished 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, 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 should be updated if changes in core loadings, surveillance measurements, or other information indicate a significant change in projected fluence values.	1.2
Spatial Representation. Neutron transport calculations should incorporate a detailed radial and azimuthal spatial mesh of ~2 intervals per inch radially and 40-80 intervals per octant. The discrete ordinates calculations must employ (at a minimum) an S_8 quadrature.	1.3.1

TABLE 1.
(Continued)

	Regulatory Position
Multiple Transport Calculations. If the calculation is performed using two or more "bootstrap" calculations, the adequacy of the overlap regions should be demonstrated.	1.3.1
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 should be demonstrated.	1.3.1
Spectral Effects on RT_{NDT}. In order to account for the neutron spectrum dependence of RT_{NDT}, when the fluence is extrapolated from the surveillance capsule or from the inside of the pressure vessel to the T/4 and 3T/4 vessel locations, a spectral lead factor must be applied for the calculation of ΔRT_{NDT}.	1.3.1
Cavity Calculations. The adequacy of the S_n angular quadrature used in cavity transport calculations must be demonstrated.	1.3.3
Methods Qualification. The calculational methodology should be qualified by both (1) comparisons to measurement and calculational benchmarks and (2) an analytic uncertainty analysis. The methods used to calculate the benchmarks should be consistent with the methods used to calculate the vessel fluence. The overall calculational uncertainty should be determined by an appropriate combination of the analytic uncertainty analysis and the uncertainty estimate based on the comparisons to the benchmarks.	1.4.2, 2.5
Fluence Calculational Uncertainty. The vessel fluence (1 sigma) calculational uncertainty must be demonstrated to be <20% for RT_{PTS} determination. If the benchmark comparisons indicate an uncertainty greater than ~20%, the calculational model must be adjusted or a bias applied to bring the agreement within this range. For non-PTS 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., 1.3.2, 1.4.3
FLUENCE MEASUREMENT METHODS	
Spectrum Coverage. The set of dosimeters must provide adequate spectrum coverage.	2.1.1
Isotopic Composition. Use of dosimeter materials should address material purity, total and isotopic mass assay, perturbations by encapsulations and thermal shields, and accurate dosimeter positioning.	2.1.1

TABLE 1.
(Continued)

	Regulatory Position
Corrections. Dosimeter response measurements should account for fluence rate variations, isotopic burnup effects, detector perturbations, self shielding, reaction interferences, and photo fission.	2.1.2
Response Uncertainty. An uncertainty analysis should be performed for the response of each dosimeter.	2.1.3
Validation. Detector response calibrations should be carried out periodically in a standard neutron field.	2.2
Calculation-To-Experiment Ratios. The C/E ratios, the standard deviation and bias between calculation and measurement should be determined.	2.3
Fast Neutron Fluence. The [E > 1 MeV] fast neutron fluence for each measurement location should be determined using calculated spectrum-averaged cross-sections and individual detector measurements.	2.3
REPORTING PROVISIONS	
Neutron Fluence and Uncertainties	
Details of the absolute fluence calculations and associated methods qualification should be reported, and the justification and description for any deviations from the provisions of this guide should be provided.	3.1
Calculated multigroup neutron fluences, fluence rates, and their uncertainties should be reported.	3.1
Calculated integral fluences and fluence rates for E > 1 MeV and E > 0.1 MeV and their uncertainties should be reported.	3.1
Measured integral E > 1 MeV fluences and uncertainties for each measurement location should be reported.	3.1
The value and basis of any bias or model adjustment made to improve the calculation-to-measurement agreement should be reported.	3.1
The results of the standard field validation of the measurement method should be reported.	3.1

TABLE 1.
(Continued)

	Regulatory Position
<u>Specific Activities and Average Reaction Rates</u>	
Measured specific activities at the end of irradiation and derived average reaction rates with uncertainties should be reported.	3.2
The corresponding calculated reaction rates and C/E ratios with uncertainties should be reported.	3.2
All corrections and adjustments to the measured quantities and their justification should be reported.	3.2

TABLE 2. THRESHOLD DETECTORS RECOMMENDED FOR PRESSURE VESSEL DOSIMETRY

	Nominal Threshold (MeV)	Applicable ASTM Standards
Neptunium-237	0.6	E705 (Ref. 46)
Uranium-238	1.5	E704 (Ref. 47)
Nickel-58	2.1	E264 (Ref. 48)
Iron-54	2.3	E263 (Ref. 49)
Titanium-46	3.8	E526 (Ref. 50)
Copper-63	4.7	E523 (Ref. 51)

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[1]Copies are available for inspection or copying for a fee from the NRC Public Document Room at 2120 L Street NW., Washington, DC; the PDR's mailing address is Mail Stop LL-6, Washington, DC 20555; telephone (202)634-3273; fax (202)634-3343. 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.

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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," of 10 CFR Part 50, "Domestic Licensing of Production and Utilization Facilities." Additionally, in response to concerns over potential pressurized thermal shock (PTS) events in pressurized water reactors (PWRs), the NRC issued 10 CFR 50.61, "Fracture Toughness Requirements for Protection Against Pressurized Thermal Shock Events."

To satisfy the requirements of both Appendix G and 10 CFR 50.61, 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," of 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 calculation-to-measurement 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 improvements include the development and improvement of computer codes and calculational models, revision of basic cross-section data, improved measurement techniques, 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 10 CFR Part 50, Appendices G and H, 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 ensure that the actual fluence is adequately bounded by the calculational methods. The 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 is generally employed at the present time. 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 requirements of 10 CFR Part 50, Appendices G and H, and 10 CFR 50.61.

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 as wide a consensus, especially from the public and the ACRS, which 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 consensus from industry and the public, 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. 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 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 materials' values of RT_{NDT} for use in 10 CFR Part 50, Appendix G, and the values of RT_{PTS} in accordance with 10 CFR 50.61. The regulatory guide would improve the accuracy and reliability of these evaluations and ensure consistency with the present regulatory position on vessel fluence uncertainty 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.

While it is recognized that the improved fluence determination may be used by licensees to relax conservatisms in the present fluence calculations, it is expected the improved fluence estimates will result in enhanced safety. 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 1. The licensee costs resulting from the changes in the measurement procedures have been estimated in Table 2.

5. DECISION RATIONALE

It is recommended that the proposed regulatory guide be issued because (1) a wide consensus of the NRC, ACRS, industry, and the public would be provided, (2) a methodology standard would be established, and (3) inefficient use of the NRC staff, vendor and licensee resources would be eliminated.

The alternatives identified above for providing acceptable fluence methods to the licensees do not provide the level of review and wide consensus required for the vessel fluence determination used to ensure adequate pressure vessel fracture toughness. In addition, while these 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 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 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.

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