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NEUTRON DOSIMETRY FOR RADIATION DAMAGE IN FISSION AND FUSION REACTORS

by

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NEUTRON DOSIMETRY FOR RADIATION DAMAGE IN FISSION AND FUSION REACTORS

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The properties of materials subjected to the intense neutron radiation fields characteristic of fission power reactors or proposed fusion energy devices is a field of extensive current research. These investigations seek important information relevant to the safety and economics of nuclear energy. In high-level radiation environments, neutron metrology is accomplished predominantly with passive techniques which require detailed knowledge about many nuclear reactions. The quality of neutron dosimetry has increased noticeably during the past decade owing to the availability of new data and evaluations for both integral and differential cross sections, better quantitative understanding of radioactive decay processes, improvements in radiation detection technology, and the development of reliable spectrum unfolding procedures. However, there are problems caused by the persistence of serious integral-differential discrepancies for several important reactions. There is a need to further develop the data base for exothermic and low-threshold reactions needed in thermal and fast-fission reactor dosimetry, and for high-threshold reactions needed in fusion-energy dosimetry. The unsatisfied data requirements for fission reactor dosimetry appear to be relatively modest and well defined, while the needs for fusion are extensive and less well defined because of the immature state of fusion technology. These various data requirements are examined with the goal of providing suggestions for continued dosimetry-related nuclear data research.

[Neutron dosimetry, radiation damage, fission, fusion, reactors, radioactivity, cross sections.]

Introduction

The widely-publicized incident at the Three-Mile-Island plant has taught us that the costs inflicted by a malfunctioning nuclear reactor are very high. The expense of bringing the facility back on line and of purchasing power from other sources during the interim period must be added to the inestimable effect of shaken public confidence in nuclear energy. Electric utilities must have reactors which will operate safely and reliably for many years. Extensive research and development is needed to meet this requirement because of the existence of complex engineering problems, unprecedented materials science phenomena and uncertainties resulting from incomplete quantitative understanding of fundamental atomic and nuclear processes.

An understanding of the scale of nuclear reactor parameters is necessary before one can truly appreciate the magnitude of this issue. Some statistics on operating power reactors may help to provide perspective: The core power density for a commercial light-water reactor (LWR) is ~30-100 KW/liter. This is ~60-200 times larger than for a 100 W light bulb. Most units under construction in the U.S. will produce >1000 MWe. Operating pressures of >1000 psi for a boiling-water reactor (BWR) and >2000 psi for a pressurized-water reactor (PWR) must be withstood by the pressure vessel (PV). Operating temperatures of several hundred degrees centigrade are typical. PVs are enormous: ~10-15 meters high by ~5-7 meters diameter with steel walls ~20 centimeters thick. Commercial nuclear plants are productive (~2 x 10^8 U.S. dollars of electricity per year) but very capital intensive (~2 x 10^9 U.S. dollars to build a plant). Structural components of reactors experience radiation damage in addition to conventional wear produced by high temperatures and pressures. In one full-power year, the interior of the PV receives a neutron dose $\lesssim 10^{18}$ n/cm^2 (>1 MeV). Measurable property changes have been observed for PV steel at fluences >10^{19} n/cm^2 (>1 MeV), but this depends on the spectrum and the operating temperature history as well as on material properties. Ten nuclear plants in the U.S.A. are >10 years old. The issue of reactor lifespan is of a paramount social and economic importance.

The scope of this paper is very broad; it encompasses operating problems for thermal power reactors (e.g. LWRs), developmental problems for fast breeder reactors and conceptual problems for controlled fusion reactors. This review is an overview of the field. The primary goal is to aquaint nuclear data people with some nuclear data needs for reactor dosimetry. The motivations for dosimetry transcend damage problems and include other important topics (fuel burnup, shielding, etc.). There have been major advances in data development and neutron metrology during the last decade. This is evident from the dramatic change in accuracy expectations for fast reactor metrology (see Fig. 1). Published references on the subject are extensive. This paper will not attempt to credit all sources of material used in the review, but will list mostly those references considered useful to readers who might wish to pursue certain aspects of the subject in more detail.

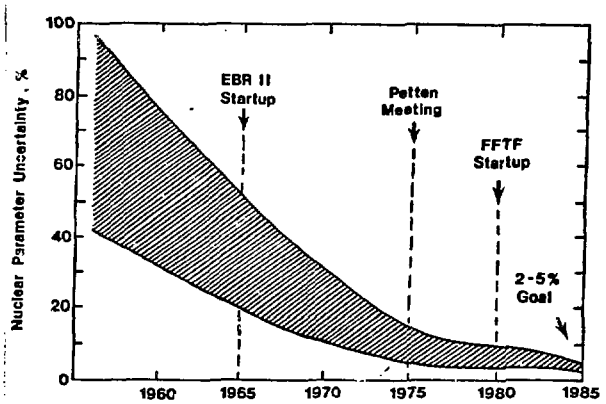


Fig. 1. Chronology of accuracy expectations for U.S. fast reactor neutron metrology [after McElroy et al., Ref. 1].

Fundamentals of Neutron Dosimetry

Concepts

There is no simple relationship between materials damage and the neutron dose. Nevertheless, a review of the field should begin with a consideration of the fundamentals of neutron metrology. Awareness of the characteristics of reactor neutron spectra is needed in order to understand damage phenomena. Figure 2 provides qualitative insight. Since wide dynamic ranges of energy and flux are encountered, logarithmic scales

and the parameter lethargy u, defined as $\ell n(E/E_0)$, are often used when discussing reactor spectra. Neutron flux, represented by $\phi(E,t)$ or $\phi(t)$, refers to neutron intensity in units of n/cm^2/sec/MeV or n/cm^2/sec (if all energies are considered). Fluence refers to the time integrals of these quantities and is expressed in dose units. Reactor people usually refer to (nv) or (nvt) which are the energy differential flux and fluence, respectively, for neutron density n and velocity v (e.g. Ref. 3). Often the variable t is dropped and $\phi(E)$ or ϕ is used when referring to the neutron spectrum or its energy integral. The function $\phi(u)$ is equivalent to $E\phi(E)$. Group fluxes or fluences refer to total neutron intensity in an energy interval $\Delta E\phi(E)$, or lethargy interval $\Delta u\phi(u)$. An energy dependent process is represented by $\omega(E)$. The response R of ω in spectrum ϕ is the product $\omega\phi$. The observable I is an integral quantity

$$I = \int_0^\infty R(E)dE = \int_0^\infty \omega(E)\phi(E)dE \quad . \qquad (1)$$

The function ω may be a reaction differential cross section σ in which case I is a reaction rate. I is then related to the spectrum average cross section $\bar{\sigma}$ ($I = \Phi\bar{\sigma}$). If ω is a damage function, then I is the observed damage rate associated with integrated flux Φ.

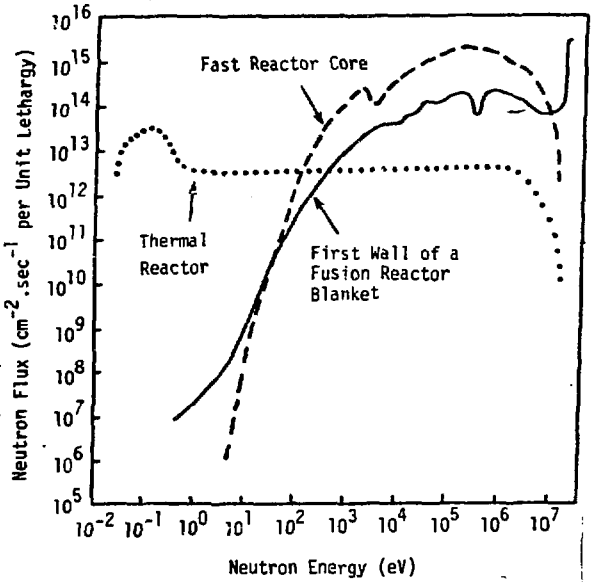


Fig. 2. Characteristic reactor neutron spectra [after Qaim, Ref. 2].

Dosimeter cross sections σ can be categorized as threshold or nonthreshold, activation or nonactivation, and fission or nonfission. Threshold reactions respond to flux above the threshold energy E_t, and the truncated average cross section $\bar{\sigma}_t$ is defined by

$$\bar{\sigma}_t = \int_{E_t}^\infty \sigma(E)\phi(E)dE \Big/ \int_{E_t}^\infty \phi(E)dE \quad . \qquad (2)$$

The 90% response range is the energy interval which contributes 90% to I in Eq. (1). Figure 3 shows 90% response ranges for several reactions in a spectrum like that which will be used for fusion material testing.

Non-threshold reactions may respond mainly to the lower energies of the spectrum (e.g. the 1/v and resonance region for capture reactions) or to most of the spectrum (e.g. U-235 or Pu-239 fission). Flat response reactions such as Pu-239 fission are insensitive to the spectrum shape above ~10 keV and are ideal flux integrators for fast-neutron spectra. This feature is used in calibrating fast reactors relative to absolute Cf-252 sources. Neutron metrology in the lower-energy regions of reactor spectra is hindered by resonance phenomena. Dosimetry differential cross sections there are not well known and uncertainties in flux depression factors (because of resonances in the total cross section) obscure the true reaction rates. The need to improve this state of affairs is motivation for further nuclear data research in the resonance region.

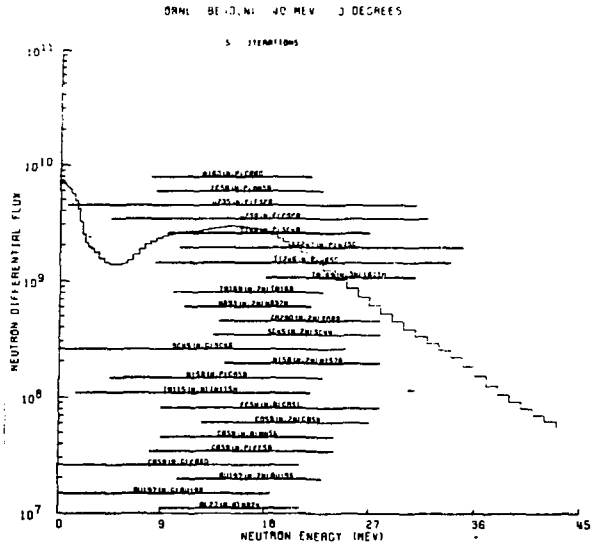


Fig. 3. Response ranges for several dosimetry reactions in a d + Be neutron spectrum [after Greenwood, Ref. 4].

Threshold reactions are useful spectral indicators. Two fundamental problems plague fission reactor dosimetry: One is a lack of suitable threshold reactions for the region 1 keV to 0.5 MeV. It is possible to shift the response range of nonthreshold reactions to higher energies by encapsulating monitor foils in Cd or B. It is not possible to shift the response to lower energies. The second problem is that above ~2 MeV all fission-based spectra have similar shapes and produce similar truncated responses for most threshold reactions (e.g. Fig. 4). Various threshold reactions provide redundant information about spectra because the response ranges overlap to a considerable degree in the MeV region. Another problem affects dosimetry for fusion reactor applications: Response ranges for threshold reactions are much broader for these spectra then they are for fission spectra. This affects the reliability of spectrum unfolding procedures.

Activation reactions are more widely used for dosimetry than nonactivation reactions. Long-half-life activation reactions (≳months) are essential for power reactor dosimetry since access is generally possible only during complete shutdown for refueling. Test reactors and critical assemblies are more accessible and shorter half-life monitors (≳minutes) are also useful for them (e.g. Ref. 5).

Fission reactions are very important for dosimetry because the cross sections are relatively well known and these processes can be used in active (fission chamber) measurements at low power and passive (fission products) applications at high power. Fission products offer a wide range of decay half lives which can be used for dosimetry purposes. Fission is beset by problems such as burn-in (growth of parasitic fissionable isotopes by capture) at high power and photofission. These effects complicate the interpretation of measured integral data (see Table I and Refs. 6-8).

Photofission corrections can be estimated with the aid of graded gamma-ray shields, but burn-in effects cannot be avoided. A subtle problem for threshold fission reactions is that the thresholds are poorly defined and the cross sections for the so-called "subthreshold" region are not well known.

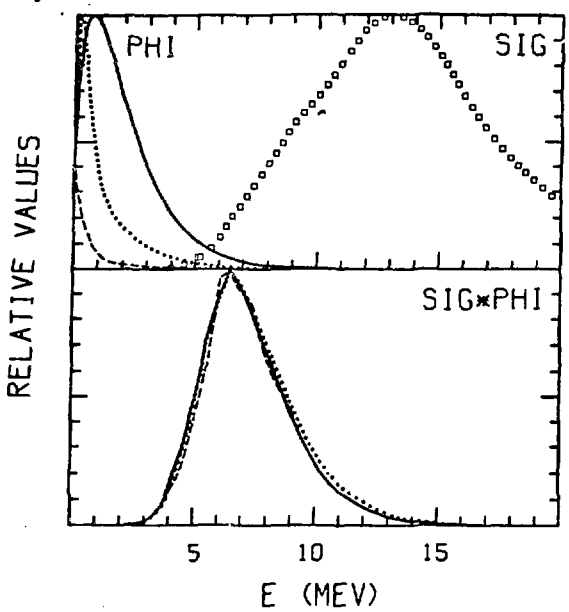


Fig. 4. Response function shapes for three different fission-driven spectra are nearly the same for a high threshold reaction.

Table I. Relative importance of photo fission at inner surface of pressure vessels. Cd is used to cover foil packets [after Bowman et al., Ref. 8].

	PWR			BWR		
Detector	Cd gammas	Reactor gammas	Reactor Neutrons	Cd gammas	Reactor gammas	Reactor Neutrons
U-235	3.80(-5)	1.08(-2)	1	6.09(-5)	2.91(-2)	1
U-238	1.90(-3)	0.476	1	2.28(-3)	0.962	1
Th-232	5.06(-3)	0.968	1	6.03(-3)	1.95	1

Spectrum Unfolding

Measurement of a series of spectrum sensitive observables I_i provides information which can be used to estimate the spectrum shape. It is necessary to solve the problem

$$I_i = \sum_{j=1}^{m} \phi_{ij}\, \sigma_{ij}\, \Delta E_j \qquad (i = 1, \ldots n) \quad . \qquad (3)$$

This is a version of the Fredholm integral problem of the first kind. Usually $m > n$ so no unique solution for ϕ exists. What is sought is a solution consistent with a preconceived form which is often based upon reactor Physics calculations. Various schemes for unfolding spectra from measured reaction rates and assumed differential cross sections have been developed. These have been discussed by Oster (Ref. 9) and Zijp et al. (Ref. 10) among others. The IAEA sponsored a critical comparison of several commonly used spectrum unfolding codes such as SAND-II, Crystal Ball, RFSP-Julich, etc. (see Refs. 9-11). The conclusion from this effort was that these various codes yield rather similar results provided that: i) the requisite trial spectrum does not differ substantially from the final solution, ii) the uncertainties in the integral quantities I_i are moderate, iii) the differential cross sections σ_i are realistic, and iv) the dosimeter response functions cover the spectrum well. As indicated above, it is generally not possible to cover spectra as well as desired with the response functions of available dosimeters. The energy region $\gtrsim$1 MeV is manageable for fission spectra. The lower limit for accurate spectrum unfolding is ~5 MeV for fusion-like spectra. Measurements $\lesssim$1 keV are complicated by resonance phenomena. Stallman and Kam (Ref. 12) have reported success in the use of linear programming techniques to generate artificial response functions with desired properties from combinations of the natural response functions of several dosimeter reactions. This "window function" method was conceived to deal with the region $\lesssim$1 keV. The gap from ~1 keV to ~0.5 MeV is problematic for fission spectrum dosimetry as indicated above.

Since spectrum unfolding methods do not usually yield unique results, it is reasonable to ask how one can deduce the most likely spectrum representation and estimate its uncertainty from available integral data and evaluated differential dosimetry cross sections. F. Perey addressed this problem and developed a least-squares procedure which answers this question in a rigorous manner (Ref. 13). A variation of this approach can also be used for performing unbiased evaluations of cross section data. The method uses techniques of matrix algebra and covariance matrices must be provided for the trial spectrum, for the differential cross sections, and for the integral reaction rates. This requirement is both a source of strength and of weakness in this approach. The strength lies in the fact that all uncertainties in the unfolding procedure are properly considered and the unfolded spectrum is the best estimate (in the least-squares sense) which the available information can provide. The weakness is that it is very difficult to provide realistic covariance matrix elements (especially off diagonal elements representing cross correlation effects). Use of inadequate matrix elements can thwart the process and lead to unreasonable results. It is generally accepted that the Perey formalism is a logical way to proceed - for the long run. Steps are being taken to implement it (e.g. inclusion of covariance matrices in ENDF/B-V). Experience gained over the next few years should establish whether it is a practical approach. In the meantime, it is likely that many other methods will continue to be used.

Neutron Damage Phenomenology

Determination of neutron flux (or fluence) and spectral shapes is only part of damage analysis. A fundamental understanding of microscopic and macroscopic damage phenomena is required to extrapolate results from damage studies in test facilities to the environment of power reactors.

The basic mechanisms of neutron damage have been discussed in several papers (e.g. Refs. 2, 14-19). Neutrons with energies of a few eV can rupture chemical bonds and participate in a few exothermic reactions, such as $^{58}Ni(n,\gamma)^{59}Ni(n,\alpha)^{56}Fe$ which produces gas in reactor structural materials. Above 40 eV, scattered neutrons can impart sufficient energy to Fe atoms in steel to displace them from the lattice. At ~1 MeV, neutrons can impart ~1000 times more recoil energy to Fe atoms than is needed to eject them from the lattice. Spectra of these recoil or primary knock-on atoms (PKA) can be calculated from a knowledge of neutron cross sections. The PKA propagate through the lattice producing more displacements until the available energy is exhausted. High-energy neutrons produce copious displacements. The term displacements per lattice atom (DPA) is used and DPA cross sections are calculated using models (e.g. the model of Kinchin-Pease, of Thompson-Wright and of Linhard). Analysis of this cascade process requires methods from atomic, molecular, and solid-state Physics. DPA cross sections determined

using various methods agree reasonably well for most important reactor materials.

DPA cross sections increase with neutron energy and may be >1000 barns in the MeV region (see Figs. 5 and 6B). During long periods of service it is very likely that most of the atoms of components located in high flux regions of a power reactor will experience displacement. Fortunately, most displacements are not permanent and much of the damage is eliminated by rearrangement of the lattice - especially at elevated temperatures (annealing). However, there are neutron-induced processes which hinder the annealing of displacement damage. All transmutation reactions produce some displacement by recoil, but the more important effect is a weakening of the lattice by introduction of foreign atoms. Even more serious are the hydrogen- and helium-producing reactions which become important at higher energies. The presence of gas in the lattice encourages the growth of voids produced by displaced atoms. This promotes swelling, creep and embrittlement of the material (Fig. 6C). Fracture resistance at elevated temperatures is reduced (Fig. 6A). The

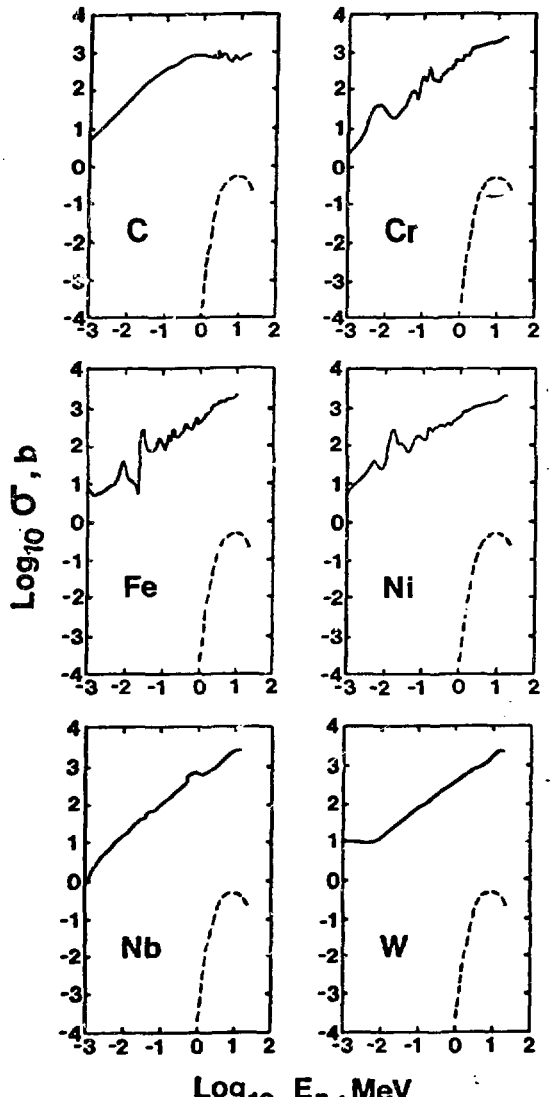


Fig. 5. Typical atomic displacement cross sections. Dashed line shows the $^{59}Ni(n,p)^{59}Co$ cross section for comparison. [after Zijp et al., Ref. 16].

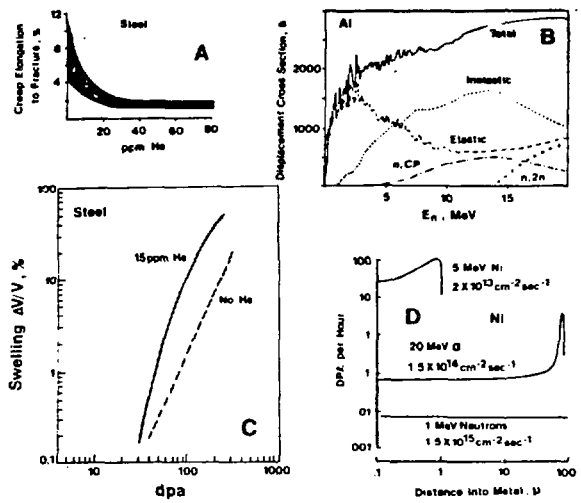


Fig. 6. Several aspects of damage phenomenology: A - Fracture resistance of steel is reduced by He [after Ullmaier et al., Refs. 20-21], B - Contributions to neutron-induced atomic displacements from various processes [after Doran et al., Ref. 19], C - Swelling vs. atomic displacement is affected by He [after Johnston et al., Ref. 22], D - Radiation damage produced by ion beams occurs near the surface whereas neutrons produce a more uniform effect throughout the volume [after Ullmaier and Schilling, Ref. 20].

presence of helium is more serious than hydrogen because hydrogen migrates out of the lattice at elevated temperatures while helium is more likely to be trapped. In reactor PVs where temperatures are lower, hydrogen production may be a significant problem.

The DPA parameter appears to be more useful for damage correlation than other common indicators (e.g. fluence >0.1 MeV or >1 MeV). However, the connection between DPA and macroscopic damage is neither simple nor well understood. Considerable research is being devoted to this problem. Gas production data files have been developed within the ENDF system. A library of DPA cross sections has been published in Europe (Ref. 16). Many irradiation experiments have been performed on reactor materials to study macroscopic damage. Ion beams readily produce atomic displacements, but the observed damage consists mainly of blistering and erosion near the surface owing to range limitations (Fig. 6D). Neutron irradiations under accelerated conditions in the core of test fast reactors such as EBR-II (U.S.A.) provide results which are not easily related to what might be expected in other environments (e.g. the PV of a LWR or the first-wall of a fusion reactor). This area of research will continue to be of paramount importance in the future.

Figures 5 and 6 show that effective thresholds for atomic displacements are relatively low, and the important damage response region for fission reactors (~50 keV to ~5 MeV) is only partially covered by activation dosimetry reactions. Effort is being directed toward developing the $^{103}Rh(n,n')^{103m}Rh$ and $^{93}Nb(n,n')^{93m}Nb$ reactions as activation dosimeters because they have low thresholds (~40 keV and ~30 KeV respectively). The response functions resemble those for DPA cross sections (Fig. 7). ^{103}Rh is better known, but a short half life (~56m) limits its application to low-power research reactors. ^{93}Nb is not well developed but has potential for power reactor applications because of a long half life (~13.6y). Further development of the data base for these reactions is an important task for the nuclear data community.

Damage Dosimetry Requirements and Programs

A review of all dosimetry-related requirements is beyond the scope of this paper. Here, an attempt will

be made to indicate some of the current damage dosimetry interests for the LWR industry, for fast-reactor development and for controlled fusion technology.

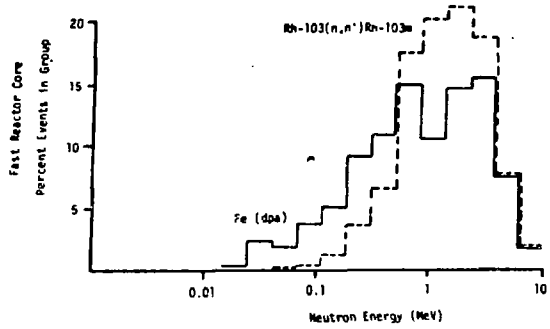


Fig. 7. Comparison of the response of Fe DPA and $^{103}Rh(n,n')^{103m}Rh$ in a fast reactor spectrum [after Sanders, Ref. 23].

LWR Industry

A major concern is longevity of the PV. The PV must last for the 40-year design life of the reactor. It has been proposed that in-situ annealing of the PV be used to remove some damage effects in welded joints of older units where the concentration of Cu and S is known to be high. Apparently, no attempt has been made yet to attempt this difficult task. Regulatory quidelines for PV surveillance (PVS) were established without a detailed understanding of damage phenomena. They may be excessively conservative, yet the results of periodic mechanical tests on specimens of irradiated PV steel must conform to these guidelines if the unit is to remain in operation. This is an important economic consideration. Dosimetry practices at most power reactors are of pre-1970 vintage and a factor of two improvement in dosimetry accuracy could be achieved using current fast-reactor techniques. Improvement of cross section data for the resonance region is required in order to better estimate resonance self-shielding in dosimeter packets. Furthermore, more accurate cross section data for some long-half-life, threshold activation monitors such as $^{54}Fe(n,p)^{54}Mn$, $^{63}Cu(n,\alpha)^{60}Co$ and $^{93}Nb(n,n')^{93m}Nb$ are required since studies have shown the MeV region to be important for PV damage effects (Fig. 8). Considerable effort is being devoted to improving computational methods (see Table II) for flux determination in LWRs. Benchmark measurement programs such as the one at Oak Ridge (U.S.), have been undertaken to test computational and experimental dosimetry methods. Other research programs are underway at selected controlled-environment power reactors (e.g. Browns Ferry-III in the U.S.). Additional material on LWR dosimetry can be found in Refs. 26-31.

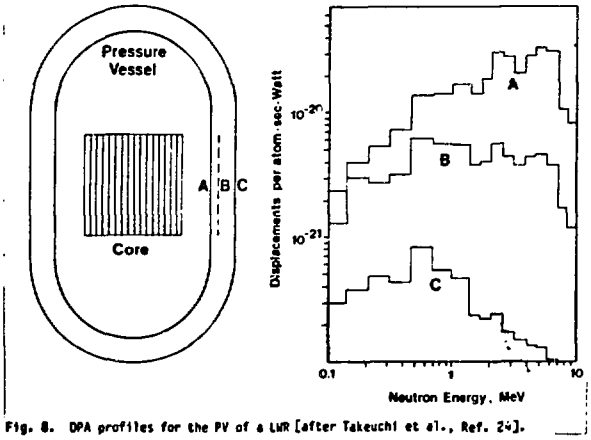


Fig. 8. DPA profiles for the PV of a LWR [after Takeuchi et al., Ref. 24].

Table II. Calculated fluxes for a PWR and BWR [after Simmons, Ref. 25].

	PWR			BWR		
Location	Flux* >0.414 eV	Flux >100 keV	Flux >1 MeV	Flux* >0.414 eV	Flux >100 keV	Flux >1 MeV
Core center	2.59(14)	1.32(14)	6.14(13)	3.08(14)	1.58(14)	7.73(13)
Core boundary	5.35(13)	2.74(13)	1.39(13)	9.33(13)	4.74(13)	2.16(13)
Pressure vessel inner surface	8.03(9)	4.56(9)	2.52(9)	6.55(10)	3.47(10)	1.77(10)
Pressure vessel outer surface	2.65(9)	1.47(9)	3.57(8)	1.38(10)	6.82(9)	9.35(8)

*Cadmium cutoff

Fast-reactor Development

The development of modern dosimetry techniques is being sponsored mostly under the auspices of fast reactor programs. Much of the support for cross section determination comes from this source as well. Engineering studies have been in progress for several years at various benchmark and critical facilities and at materials testing facilities (e.g. FBR-II in the U.S.). Damage research in the next decade will be concentrated at several new facilities such as FFTF (U.S.), PFR (U.K.) and Phenix and Super-Phenix (France). Core damage phenomena (swelling, creep, etc.) are undoubtably a very great concern for fast reactors since high fuel burnups (long fuel cycles) are important for efficiency. Alteration of core configurations by swelling and creep can affect efficiency and might lead to safety problems since fast-reactors are not designed to operate in the most reactive configuration. Improvements in dosimetry methodology are required in all facets of the field to meet the accuracy goals of Fig. 1. These primarily involve refinements of existing techniques and increases in accuracy of the data base <10 MeV for neutron reactions which have already been studied.

Controlled Fusion Technology

The design of fusion reactors is speculative since breakeven (feasibility) of controlled fusion has not been demonstrated. Most materials damage studies are formulated under the assumption that a demonstration facility will be a Tokamak; however, fission-fusion hybrids, laser fusion machines, and ion beam devices are being considered as alternatives. Damage to the "first wall" of the blanket is of primary concern. It is anticipated that the first wall will experience a power loading of ~1-3 MW/m^2 due primarily to fast neutrons ($\geq 10^{14}$ $n/cm^2/sec$). Approximately 75% of the neutrons will have energies >100 keV and 80% will be below 13.5 MeV. The average energy for this degraded "14-MeV" spectrum is considerably higher than in a fast reactor spectrum. Although the atomic displacement rate is expected to be similar to the core of a fast reactor, damage rates an order of magnitude greater are anticipated because of larger gas production rates. A major goal is the design of radiation-resistant materials for the first wall. Materials testing for fusion applications will take place at fission test reactors (e.g. FFTF), but this will have to be supplemented by measurements at higher energies. Programs are now being conducted at 14-MeV facilities (e. g. the RTNS at Livermore in the U.S.). The U.S. intends to build a high-energy test facility (FMIT) based on the d + Li reaction. It will provide a spectrum similar to the one shown in Fig. 3 with total neutron output ~10^{16} n/sec. Dosimetry for the region 10 - 15 MeV will involve some extrapolation of the existing data base for fission reactors. Proper utilization of a d + Li test facility will require extension of dosimetry techniques to ~40 MeV. The opening of many uninvestigated nuclear reaction channels will necessitate expansion of our quantitative knowledge of nuclear processes far beyond current limits. Since the current level of research support for fusion energy nuclear data development is low,

it is difficult to speculate on how this might be accomplished. The reader is referred to the proceedings of a 1977 symposium at Brookhaven, U.S.A., and other selected references for more information on this subject (Refs. 32-36).

Dosimetry Techniques

Dosimetry techniques are categorized as active or passive, nuclear reaction or direct damage, activation or nonactivation, high flux or low flux, differential or integral, and flux measurement or fluence measurement. A requirement for most methods is accurate characterization of dosimetry materials (e.g. Ref. 37). Accurate assay of fissionable materials is crucial for other areas of fission reactor development as well. A comparison of fission standard materials used in several laboratories in the U.S. and Europe has been carried out under the auspices of the Interlaboratory Reaction Rate (ILRR) program (see Ref. 38). Nuclear data people should not overlook the fact that dosimetry materials must survive the high temperature environments of power reactors where some materials will melt or vaporize and could be lost without proper encapsulation (e.g. see Refs. 6 and 27).

Active Dosimetry

This term applies to on-line measurements with electronic instrumentation. An obvious advantage of active dosimetry is the ability to detect rate variations. Disadvantages include the possibility of instrument failure, flux level limitations (for most methods) and complexity. Discussion of time-of-flight techniques will be avoided here, though they are useful for measuring the thick-target spectra which are employed in fusion research.

Fission chambers are widely used. They are compact and the cross sections for several fission reactions are well known. Measurement procedures are not difficult. Flux limitations ($\lesssim 10^{10}$ n/cm^2/sec) restrict these detectors to moderately low-power applications. Fission detectors provide only integral reaction rate data (e.g. see Ref. 39).

Differential spectrometry is only possible at quite low fluence levels ($\lesssim 10^7$ n/cm^2/sec). Proportional counters based upon n-p scattering or the ^{3}He(n,p)^{3}H reaction, and solid state detectors based upon both these reactions and the ^{6}Li(n,t)^{4}He reaction as well are used (e.g Refs. 40-44). Differential measurements require great care and experience, but are worth the effort since they provide spectral information in the region from 10 keV to $\lesssim$5 MeV which is difficult to investigate by other means. Figure 9 shows the quality of data which can be obtained from such measurements. Accuracies of ~5-10% for $0.01 < E_n < 2$ MeV and ~10-15% for $E_n > 2$ MeV are possible using differential techniques. Discrepancies have been observed in measurements involving the ^{6}Li(n,t)^{4}He reaction which lead to underestimation of neutron fluence ~40-80 keV and ~200-300 keV. Inexact knowledge of the cross section may be responsible.

Self-powered neutron detectors (SPND) are used for measurements at full power in LWRs. The SPND probe is charged with rhodium. Neutrons are captured by ^{103}Rh and energetic beta rays are emitted during the decay of ^{104}Rh. These betas penetrate the insulation to the sheath of the probe; electrons then flow up from ground through a leadwire to neutralize the emitter. This generates a measurable current. These devices are in commercial use, but suffer from several problems including the effects of Rh depletion (it converts to Pd) and radiation degradation of the insulation (see Ref. 45).

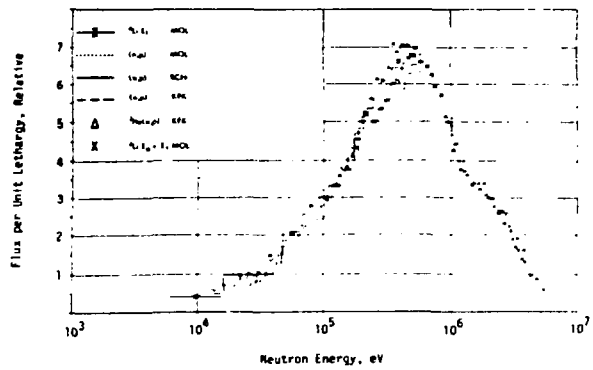


Fig. 9. Some differential spectrometry results for the Mol Sigma-Sigma field [after Fabry et al., reproduced from Ref. 41 with permission from the Am. Nucl. Soc.].

Passive Dosimeters

Passive dosimeters offer many advantages for research and for routine dosimetry in power reactors. They can be used at flux levels covering a dynamic range of over ten decades. They can be made compact and require no attention during irradiation. Activation-detector dosimetry is more widely used than any other method and it offers great versatility. The development of instruments for high resolution photon spectroscopy (Li and Ge diodes) has revolutionized activation counting. Older methods involving the use of NaI (Tl) gamma-ray detectors and beta detectors are on the decline. Attention will be directed in this paper toward less-well-known dosimetry techniques, several of which were introduced recently.

Solid-State Track Recorders (SSTR) and Nuclear Research Emulsion (NRE) techniques have been used for many years. They offer several features which insure that they will continue to be useful for LWR, FBR and Fusion dosimetry applications. These recorders measure dose and are not rate sensitive. Once exposed, they provide a permanent record of the irradiation. SSTR recorders provide an enormous sensitivity range. Manual and automated scanning techniques have been developed and track densities in the range 1 to 10^8 tracks/cm^2 can be handled. By properly selecting the recorder materials it is possible to discriminate between various events (e.g. alpha particles and fission fragments). SSTRs can be doped with various sensor materials including fissionable isotopes and (more recently) alpha emitters. NREs have been in use longer than SSTRs. NRE recorders offer the possibility of measuring the angular dependence of neutron flux. No other reactor dosimetry technique offers this feature. NRE dosimetry is limited to fluences $\lesssim 3 \times 10^9$n/cm^2. The reader is referred to Refs. 5 and 46 for more information on this subject.

One of the most promising techniques to be developed during recent years is the helium-accumulation fluence monitor (HAFM). High sensitivity mass spectrometry and isotopic dilution methods are used to detect helium produced by (n,α) reactions in sealed dosimeter capsules. The detection range for the apparatus is from ~10^9 to ~10^{18} He atoms. Measurements can be made with an accuracy < 2% if >10^{11} He atoms are present. Initially, only the ^{6}Li(n,t)^{4}He and ^{10}B(n,α)^{7}Li reactions were employed, but the method is being extended to other (n,α) reactions (e.g. to F, Fe, Ni, etc.). All early capsules were made of vanadium because of its low (n,α) cross section and its ability to contain He. Techniques involving other encapsulation materials (e.g. Au) and the measurement of He trapped in bare solid

wires are under current investigation. One of the most exciting features of HAFM dosimetry is the possibility for measurement of He buildup which is closely related to observed macroscopic damage. At present, only one group (at Rockwell International Corp., California, U.S.A.) has a laboratory equipped for these measurements. Exploitation of this promising nonactivation technique should be more widespread (see Ref. 47). Of interest to nuclear data people is the requirement for accurate differential (n,α) cross section data which this method imposes.

Effort is being devoted to develop so-called "direct damage" sensors because they possess response ranges closely related to the atomic displacement cross sections. Research on the alteration of physical properties of quartz by neutrons has been conducted at Mol (Belgium). The effect of neutrons on p-i-n diodes has also been considered for dosimetry purposes. Neither approach is widely used at this time. The French have developed the graphite damage monitor (GAMIN) and (more recently) a variation involving tungsten. The change in resistance of precision graphite (or tungsten) resistors following neutron irradiation is measured. Information about the sample temperature during irradiation is also needed. Damage dosimetry information is deduced from measured resistance changes and from a knowledge of the fission-neutron equivalent fluence above ~1 MeV deduced from $^{58}Ni(n,p)^{58}Co$ activation monitors included in the dosimetry package. DPA cross sections for major components of steel (Fe, Ni and Cr) are bracketed by those for carbon and tungsten (See Fig. 5). The French method therefore provides upper and lower estimates for structural materials damage (Refs. 15, 48-50).

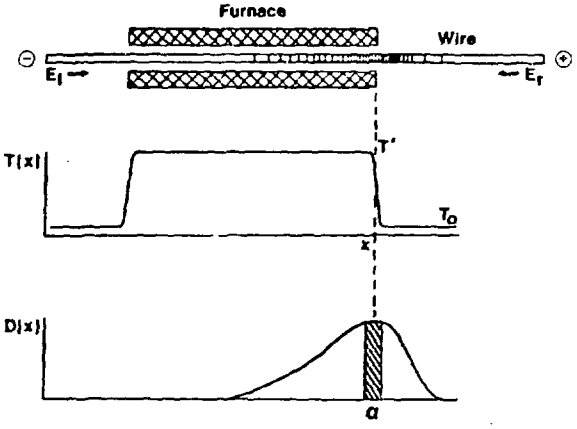
Another direct damage technique is based on the change in the thermocouple properties of neutron irradiated wires (Fig. 10). This technique is being investigated at Mol. One version involves measurements on a uniform wire which has received various neutron doses as a function of position along the wire. The irradiated wire is inserted into a furnace with a sharp gradient. Net currents are produced as damaged portions of wire pass through the region with a large temperature gradient. The sharper the gradient, the better the resolution for measuring damage versus position. Interpretation is complicated by the fact that some types of damage are annealed out of the wire while it is in the furnace. This fact can also be put to good use if the apparatus is properly calibrated (see Ref. 51).

Dosimetry Cross Section Data Development

This section considers methods for improving the dosimetry cross section data base. Progress toward this goal is being achieved as a result of work in four distinct areas: i) differential measurements, ii) theoretical calculations, iii) integral measurements in benchmark fields, and iv) evaluations. The accuracy of the cross section data required for most fission reactor applications has improved remarkably over the last decade. Each of these distinct activities can claim a share of the credit for this progress.

Differential Measurements

Techniques and problems associated with differential dosimetry cross section measurements at energies >100 keV were reviewed in 1976 (Ref. 52). It was concluded that an accuracy goal of ~5% for the energy range 0.1-20 MeV (except for a gap from 10-14 MeV) is feasible for most activation reactions, but that it will be quite a while before this goal is achieved. The future is more uncertain for nonactivation reactions and for energies <0.1 MeV and >20 MeV. It is necessary to direct measurement effort toward energy regions which are important for major applications. This fact is illustrated in Fig. 11. For threshold reactions the first eight MeV above threshold are important in fission reactor applications. The sensitivity of integral quantities to differential data merits detailed investigation so that relevant differential measurement programs can be undertaken. Smith (Ref. 53) and Mannhart (Ref. 54) have addressed this problem. Several integral-differential discrepancies were eliminated during the last decade as a result of various independent broad-energy-scope differential measurement programs with accuracies ≤10%. It is anticipated that carefully conducted differential measurements will continue to serve this purpose in the future.



$$E = E_f - E_i = F[D(x), T', T_o, \cdots] \neq 0 \text{ if } D(x) > 0$$

Fig. 10. Basic concept of thermocouple neutron dosimetry [after Mathieu et al., Ref. 51].

Theoretical Calculations

In fission reactor applications, model calculations are important for interpolation between experimental data points (especially from ~10-14 MeV) and for extrapolations from 15 to 20 MeV and to threshold. As a general rule, computations for energies from 0.1 to 20 MeV cannot be trusted to much better than ~20% unless they are guided by experimental data. The role of theory in high-energy dosimetry data development (20-40 MeV) will be much greater than for lower energies. It is too early to speculate on how successful this approach will be (see Refs. 32-33). The energy region <1 keV should also be a fertile one for theoretical work since differential measurements are difficult in this region.

Integral Benchmark Fields

The quality of integral measurements is not always superior to corresponding differential ones. A survey of integral measurements for the Cf-252 neutron field indicates that there are sizeable differences between reported spectrum average cross sections for several important reactions (e.g. for (n,p) reactions on Ti-46, 47, 48). Furthermore, integral studies have not been particularly successful in distinguishing between shape and normalization effects in differential cross sections (e.g. Fig. 12). One is led to the conclusion that it will not be possible to develop the cross section data base for dosimetry applications to stated accuracy goals (e.g. Fig. 1) by means of benchmark integral studies alone as has been implied by some members of the integral community (see Fig. 13). Nevertheless, the importance of research at well-characterized integral facilities must not be discounted. These facilities are especially useful for development of metrology and computational techniques

required for development of nuclear energy sources. Good quality spectrum average cross section measurements have been very helpful in pointing out deficiences in differential data (e.g. the (n,p) reaction on Ti-48). Interlaboratory cooperation (e.g. the ILRR and IAEA sponsored programs) has been valuable in upgrading the overall quality of integral data. The topic of benchmark fields has been reviewed by Grundl and Eisenhauer (Ref. 3), so only a few items of interest will be mentioned here.

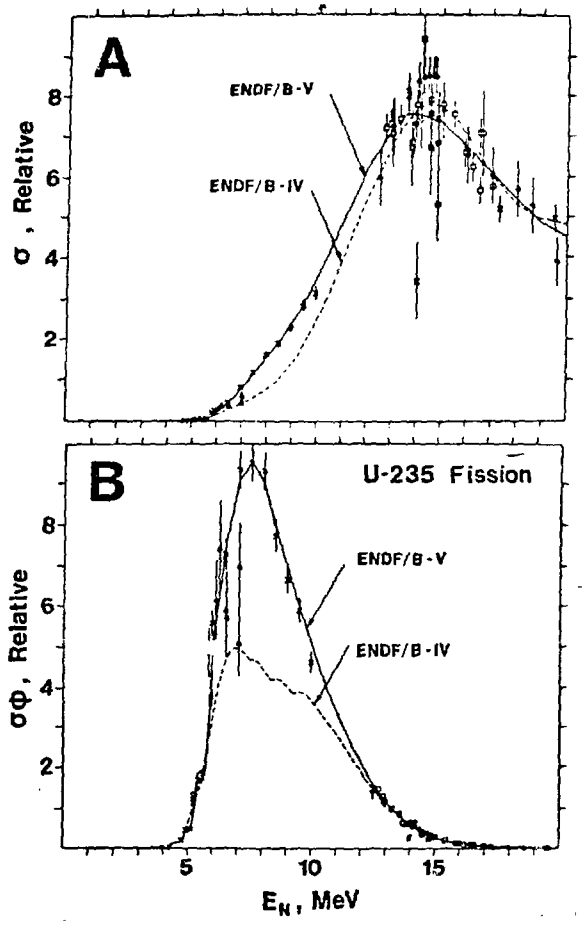


Fig. 11. Cross sections for $^{48}Ti(n,p)^{48}Sc$: A - normal, B - weighted by a fission spectrum.

Table III provides information about integral facilities which are generally accepted as benchmark fields for nuclear data development. Figure 14 shows the approximate spectral shapes for most of these fields. The redundancy of several of these fields is apparent. The available detector response ranges are limited. However, this is an advantage for investigating threshold reactions since the truncated average cross sections for several of these fields are quite similar. It is sometimes possible to compare the results of measurements at various facilities without regard to absolute fluence normalization by considering reaction rate ratios.

The best known field is that of Cf-252 spontaneous fission. This field has the advantage that absolute source strengths can be measured readily. The spectrum is not well known <0.25 MeV or >8 MeV. None of the benchmark fields in Table III are of much value for investigations above ~10 MeV except for d + Be.

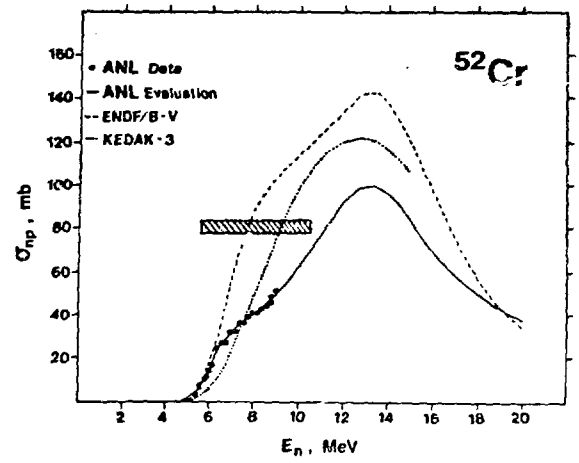


Fig. 12. Integral measurements in a fission spectrum do not sense much difference between the ANL and Kedak-3 evaluations. Experimental data support the ANL evaluation. Dashed bar shows the fission spectrum 50% response range.

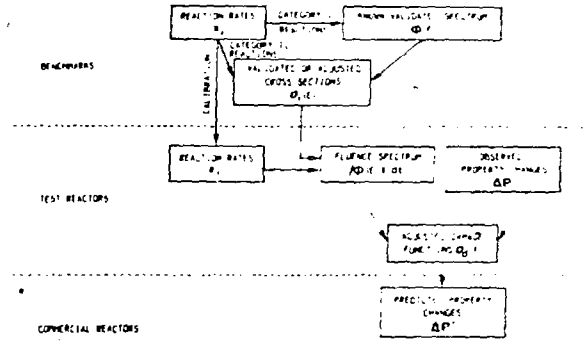


Fig. 13 The integral approach to dosimetry data development [after Fabry et al., Ref. 55].

Table III. Benchmark fields for dosimetry cross section development [after Grundl and Vlasov, Refs. 3 and 56].

Benchmark Neutron Field	Average Energy	Energy Range for Data Testing	Available Fluxes	Status of Group-flux Spectrum Characterization
^{252}Cf spontaneous fission neutrons	2.13 MeV	0.1 to ~10 MeV	~(1-3)x10^9 n/sec	±13%, E<0.25 MeV ±2%, 0.2<E<8 MeV ±5%, 8<E<12 MeV ±20%, 12<E<15 MeV
^{235}U pure thermal fission neutrons	1.97 MeV	0.1 to ~10 MeV	≤1x10^{13} n/cm^2/sec	±15%, E<0.25 MeV ±2-5%, 0.25<E<8 MeV ±5%, 8<E<12 MeV ±10%, 12<E<15 MeV
Near 1/E (ISNF/CV)	0.75 MeV	0.4 eV to 0.1 MeV	~5.6x10^8 n/cm^2/sec	±5% over range
Sigma-Sigma	0.76 MeV	0.01 to ~10 MeV	~7x10^8 n/cm^2/sec	±15%, E<0.1 keV ±5%, 0.1 keV <E<2 MeV ±5%, E>2 MeV
ISNF	~1 MeV	0.008 to ~10 MeV	~0.8x10^9 n/cm^2/sec	±5%, E < ~2 MeV ~±2-5%, 2<E<12 MeV
Big Ten	0.58	0.01 to ~10 MeV	10^9-10^{11} n/cm^2/sec	±5%, 0.05<E<2 MeV ±5%, E>2 MeV
CFRMF	0.76	0.01 to ~10 MeV	<10^{11} n/cm^2/sec	±15%, E<0.01 MeV ±5%, 0.01<E<2 MeV ±5-10%, E>2 MeV
FAST	1.5	0.01 to ~10 MeV	<10^{12} n/cm^2/sec	±5-20%, 0.01<E<10 MeV
Fast/6	1.5	0.01 to ~10 MeV	~1x10^{13} n/cm^2/sec	±5-20%, 0.01<E<10 MeV
Accelerator d + Be	~0.4 Ed	1 to ~30 MeV	<1 x 10^{12} n/cm^2/sec	±10-30%, 1<E<30 MeV

The field from pure thermal neutron fission of U-235 is less well known, but knowledge of the spectrum of this field is extremely important since it is used not only for cross section validation, but also in reactor transport calculations. The Cf-252 and U-235

spectra have been evaluated in a very important study by Grundl and Eisenhauer (Ref. 57). The results of their work appear to have been confirmed by most integral tests during the last four years although there has been some dissent (e.g. Ref. 58). Various representations of the U-235 fission spectrum which have been used during the last few years differ considerably from each other above ~7 MeV (see Fig. 15). The recent ENDF/B-V evaluation (a Watts function) does not differ seriously from the NBS evaluation of Grundl and Eisenhauer. Uncertainty in the shape of the U-235 fission spectrum has led to considerable confusion in the area of differential-integral data comparisons-especially for high threshold reactions. For the same reason, fission reactor spectra are poorly known at high energies. Fission spectrum measurements are useless for testing high threshold differential data, but good quality differential cross section determinations for these reactions may eventually help to define fission spectra above 10 MeV to suitable accuracy.

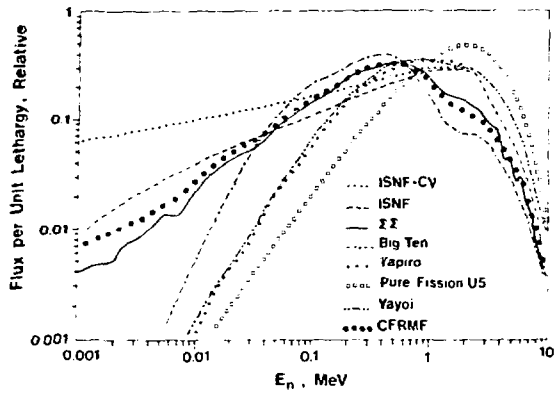


Fig. 14. Approximate spectral shapes for several integral benchmark fields [most of this material comes from Fabry, Ref. 55].

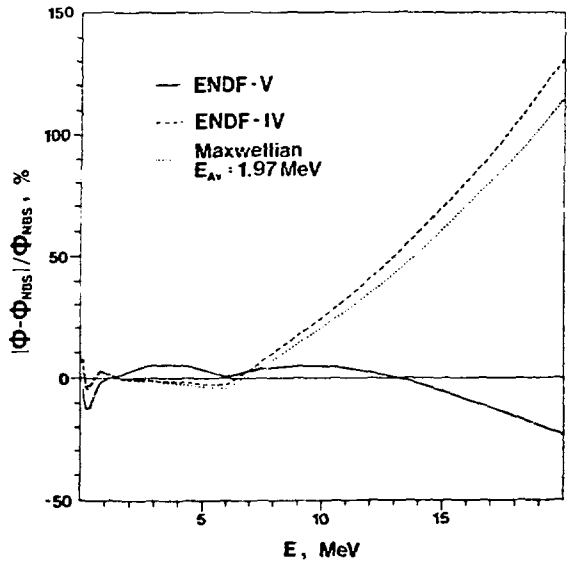


Fig. 15. Several representations of the U-235 fission spectrum plotted relative to the NBS evaluation from Ref. 57.

Nearly pure U-235 spectra can be obtained by using small enriched uranium converters in reactor thermal column cavities; the thermal neutrons are excluded by Cd shields (Fig. 16A). Tailored cavity spectra can be generated by using scatterers of carbon, boron or cadmium (Fig. 16B). The geometries and scattering cross sections for these cavities are well known so their spectra can be calculated reliably using transport methods. Other reactor benchmarks do not possess these advantages. The geometries are more complex and scattering in Fe and U produce uncertainties because of inaccurate knowledge of the cross sections. Uncertainties in uranium inelastic scattering have been a major source of difficulty in characterizing these fields (e.g. see the effect of changes in U-238 inelastic scattering on CFRMF in Figs. 17 and 18).

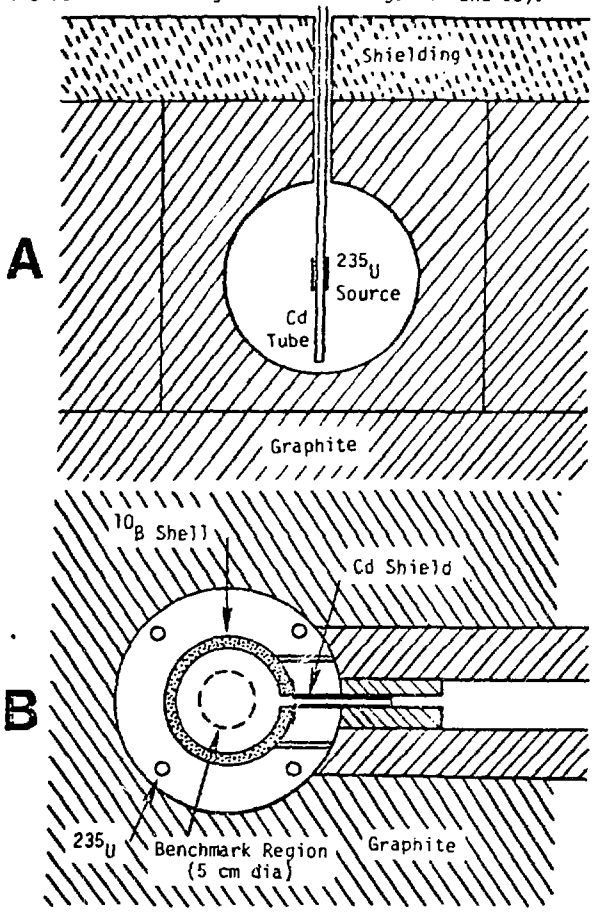


Fig. 16. Representative cavity benchmark fields: A - The U-235 field at Hal, B - the NBS ISNF [after Grundl and Eisenhauer, Ref. 3].

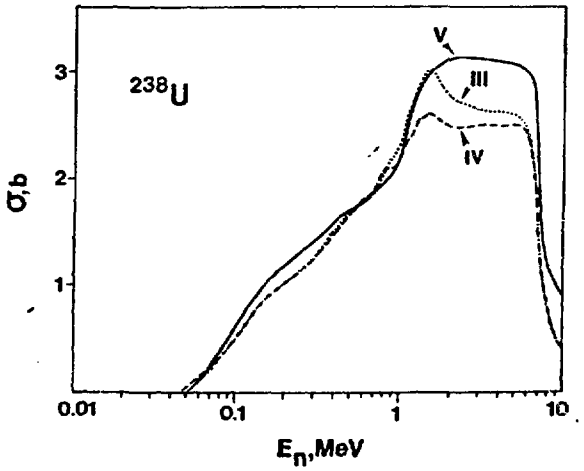


Fig. 17. Three generations of ENDF evaluations for U-238 inelastic scattering [Ref. 59].

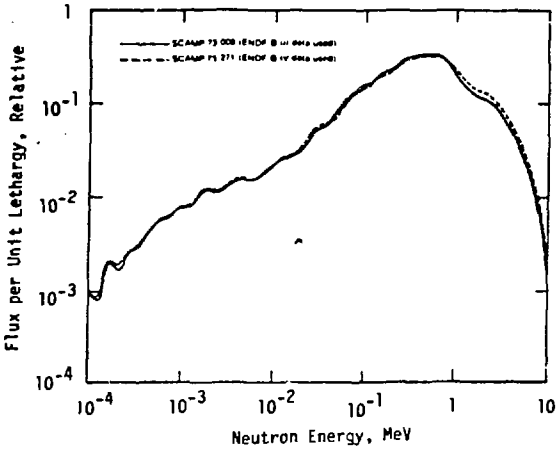


Fig. 18. Change in the CFRMF neutron spectrum produced by converting from ENDF/B-III to IV cross sections. Most of the effect is due to U-238 inelastic scattering [after Rogers, Ref. 38].

Data testing in the integral fields produced by bombarding thick Be targets with deuterons is underway on a limited scale for fusion applications. The opportunity to use pulsed beams and time-of-flight methods to measure spectrum shapes is an important feature of this work. This effort should be encouraged so that the methodology can be developed, however the results will continue to be affected for the forseable future by the totally inadequate knowledge of differential cross sections above 20 MeV (Refs. 32, 60).

Evaluations

The widespread availability of comprehensive differential data evaluations such as those provided by ENDF (U.S.A.) and KEDAK (F. R. Germany) has eliminated several discrepancies which could be traced to use of inconsistent differential cross sections. Evaluations of integral cross sections are also useful, but far less work has been done in this area. Continued evaluation effort should be encouraged. It is hard to find people who are willing and able to perform these evaluations. More international collaboration is recommended and cooperation between theorists and experimentalists is essential.

Some Remarks on the Status of Cross Section Data for Dosimetry

Four categories of nuclear data are of interest for dosimetry applications. These are: i) radioactive decay data, ii) fission mass yield data, iii) differential cross section data, and iv) integral cross section data. There are many lists of reactions considered important as dosimeters. The compiled list of all separate lists is intractable. Table IV represents an attempt to produce a list of those reactions which appear most often on other lists. This is still a very long list and the ENDF/B Dosimetry File is a small subset (39%). The number of reactions which are thought to be adequately known for applications (Category I or I Candidate) is smaller still (18%). Reactions with thresholds >10 MeV are of marginal interest for fission applications, yet they constitute ~25% of the list. If one accepts this list as the goal for nuclear data development, then <20% of the needed work has been accomplished. Actually the situation is worse than this when the need to advance to ~40 MeV is considered.

The list of requisite decay data for dosimeter applications is also impressive (Table V). If one arbitrarily assumes that decay half lives should be

Table IV. Reactions useful for neutron dosimetry. Approximate effective threshold energies E_t (MeV) are shown where applicable.

Reaction		E_t	Reaction		E_t	Reaction		E_t
Li-6(n,total He)	*	-	Fe(n,total He)		1	In-115(n,G)In-116m	*	-
B-10(n,total He)	*	-	Co-59(n,G)Co-60	*IC	-	I-127(n,G)I-128		-
F-19(n,2n)F-18		10.9	Co-59(n,A)Mn-56	*IC	5	I-127(n,2n)I-126	*	9.3
Na-23(n,G)Na-24	*	-	Co-59(n,2n)Co-58	*	10.6	La-139(n,G)La-140		-
Na-23(n,2n)Na-22		13	Co-59(n,3n)Co-57		19.1	Dy-164(n,G)Dy-165		-
Mg-24(n,p)Na-24		5	Co-59(n,4n)Co-56		30.5	Tm-169(n,2n)Tm-168		8
Al-27(n,total He)		4.9	Co-59(n,p)Fe-59		2.5	Tm-169(n,3n)Tm-167		14.9
Al-27(n,p)Mg-27	*	2.3	Ni-58(n,p)Co-58	*IC	0.3	Tm-169(n,4n)Tm-166		23.6
Al-27(n,A)Na-24	*I	4.9	Ni-58(n,2n)Ni-57	*I	12.5	Lu-175(n,2n)Lu-174		7.7
P-31(n,p)Si-31		1.5	Ni-58(n,3n)Ni-56		22.9	Lu-176(n,2n)Lu-175		14
S-32(n,p)P-32	*IC	1.5	Ni-60(n,p)Co-60	*	3.3	Lu-175(n,G)Lu-176m		-
S(n,total He)		1.5	Ni-60(n,2n)Ni-59		11.6	Lu-176(n,G)Lu-177m		-
Cl-35(n,G)Cl-36		-	Ni(n,total He)		0.5	Ta-181(n,G)Ta-182		-
Cl-35(n,A)P-32		3	Cu-63(n,G)Cu-64	*	-	W-186(n,G)W-187		-
A-40(n,G)A-41		-	Cu-63(n,A)Co-60	*	3.0	Au-197(n,2n)Au-196	*I	-
Sc-45(n,G)Sc-46	*	-	Cu-63(n,2n)Cu-62		10.9	Au-197(n,3n)Au-195		8.5
Sc-45(n,2n)Sc-44m		11.9	Cu-65(n,2n)Cu-64	*	10.1	Au-197(n,4n)Au-194		16
Ti-46(n,2n)Ti-45		13.2	Cu(n,total He)		2	Au-197(n,4n)Au-194		25
Ti-46(n,p)Sc-46	*	2.5	Zn-64(n,p)Cu-64		1	Au-197(n,5n)Au-193		30.1
Ti-47(n,p)Sc-47	*	1	Y-89(n,2n)Y-88		11.5	Hg-199(n,n')Hg-199m		0.53
Ti-47(n,n-p)Sc-46	*	10.8	Y-89(n,3n)Y-87		21.2	Pa-232(n,f F P	*	0.6
Ti-48(n,p)Sc-48	*	4.5	Zr-90(n,p)Y-90m		4.5	Pa-232(n,f [Pa-233]	*	-
Ti-48(n,n-p)Sc-47	*	11.7	Zr-90(n,2n)Zr-89		12	U-238(n,f F P		1
Cr-50(n,G)Cr-51		-	Zr-90(n,3n)Zr-88		21.3	U-235(n,f F P	*IC	-
V-51(n,G)V-52		-	Nb-93(n,n')Nb-93m		0.03	U-236(n,f F P		-
Mn-55(n,G)Mn-56		-	Nb-93 n,2n)Nb-92m		9	U-238(n,f [Np-239]	*IC	-
Mn-55(n,2n)Mn-54	*	10.5	Rh-103(n,n')Rh-103m	IC	0.04	U-238(n,2n)U-237		6
Fe-54(n,A)Cr-51		1	Ag-107(n,2n)Ag-106m		9.7	Th-232(n,f F P	*I	0.5
Fe-54(n,p)Mn-54	*IC	0.8	Ag-107(n,3n)Ag-105		17.5	Np-237(n,f F P	*I	0.1
Fe-56(n,p)Mn-56	*I	3.7	Ag-109(n,G)Ag-110m		-	Pu-239(n,f F P	*I	-
Fe-58(n,G)Fe-59	*	-	In-115(n,n')In-115m	*IC	0.34	Am-241(n,f)F P		-

*-ENDF/B Dosimetry File / I-Category I / IC-Category I Candidate / [...]-Useful Product

Table V. Decay data required for dosimetry applications: L - (Ref. 61), H - (Helmer and Greenwood, Ref. 38). Half life uncertainties 20.5% and principle gamma-ray branch uncertainties 21% are underlined. Bracketed isotopes are fission products.

Activity		$t_{1/2}$		E_G, MeV	I_G	
F-18	L	109.72m	(0.05%)	0.511 (Beta +)	1.94	(0.2%)
Na-22	L	2.602y	(0.08%)	1.275	0.9993	(0.02%)
Na-24	H	15.00h	(0.1%)	1.3686	0.99993	(0.002%)
Mg-27	H	9.462m	(0.1%)	0.8438	0.717	(0.7%)
Si-31	L	2.62h	(0.4%)	1.2662	0.0007	(?)
P-32	L	14.282d	(0.04%)	Beta -	<0.0001	
Cl-36	L	3.00(5)y	(0.7%)	Beta -, EC	---	
A-41	L	1.827h	(0.4%)	1.2936	0.9916	(0.02%)
Sc-44m	L	2.442d	(0.2%)	0.2712	0.866	(0.2%)
Ti-45	L	3.08h	(0.3%)	0.7196	1.54(-3)	(7.8%)
Sc-46	H	83.9d	(0.4%)	0.8893	0.99984	(0.002%)
Sc-47	H	3.40d	(0.9%)	0.1594	0.690	(3.6%)
Sc-48	H	43.8h	(0.2%)	0.9834	0.99987	(0.002%)
Cr-51	H	27.700	(0.04%)	0.3201	0.0983	(1.4%)
V-52	H	3.75m	(0.3%)	1.4341	1.00	(1%)
Mn-54	H	312.5d	(0.2%)	0.8348	0.9997	(0.02%)
Mn-56	L	2.5785h	(0.02%)	0.8468	0.9887	(0.04%)
Ni-56	L	6.10d	(0.3%)	0.1584	0.988	(0.1%)
Co-57	L	271.65d	(0.05%)	0.1221	0.856	(0.5%)
Ni-57	L	35.99h	(0.3%)	1.3776	0.776	(1%)
Co-58	H	70.85d	(0.2%)	0.8108	0.9944	(0.01%)
Fe-59	H	44.6d	(0.2%)	1.0992	0.561	(1.8%)
Ni-59	L	7.5(4)y	(17%)	EC	---	
Co-60	H	5.271y	(0.04%)	1.3325	0.99980	(0.003%)
Cu-62	L	9.73m	(0.2%)	0.511 (Beta +)	1.956	(<0.5%)
Cu-64	H	12.702h	(0.03%)	0.511 (Beta +)	0.38	(5.3%)
Y-87	L	80.3h	(0.4%)	0.4848	0.92	(1.1%)
Y-88	L	106.61d	(0.02%)	1.8360	0.9934	(0.07%)
Zr-88	L	83.4d	(0.4%)	0.3937	0.973	(0.1%)
Zr-89	L	78.43h	(0.1%)	0.9092	0.9901	(0.04%)
Y-90m	L	3.19h	(0.3%)	0.4795	0.91	(4.3%)
Nb-92m	L	10.14d	(0.3%)	0.9344	0.992	(0.2%)
Nb-93m	L	13.6y	(2.2%)	IC	---	
[Zr-95]	H	64.1d	(0.5%)	0.7567	0.546	(0.9%)
[Zr-97]	H	16.88h	(0.4%)	0.7433	0.929	(0.3%)
Rh-103m	L	56.116m	(0.02%)	IC	<0.00068	
[Ru-103]	H	39.43d	(0.3%)	0.4971	0.89	(1.1%)
Ag-105	L	41.29d	(0.2%)	0.443	0.12	(8.3%
Rh-106m	L	130m	(1.5%)	0.512	0.864	(0.5%)
[Rh-106]	H	30.0s	(0.7%)	0.512	0.205	(1%)
[Ru-106]	L	368.2d	(0.3%)	Beta -	---	
Ag-110m	L	252.2d	(0.1%)	0.6577	0.944	(0.1%)
In-115m	H	4.486h	(0.09%)	0.3362	0.459	(0.2%)
In-116m	H	54.2m	(0.2%)	1.2935	0.848	(0.6%)
I-126	L	13.02d	(0.5%)	0.3887	0.35	(8.6%)
I-128	L	24.99m	(0.08%)	0.4429	0.16	(12.5%)
[Te-132]	H	77.9h	(0.6%)	0.2282	0.89	(5.6%)
[Cs-137]	H	30.03y	(0.5%)	0.6616	0.853	(0.5%)
[La-140]	H	40.26h	(0.05%)	1.5962	0.9540	(0.08%)
[Ba-140]	H	12.789d	(0.05%)	0.5374	0.244	(1.2%)
[Ce-141]	H	32.50d	(0.2%)	0.1454	0.49	(4.1%)
[Ce-143]	H	33.0h	(0.6%)	0.2933	0.47	(8.5%)
[Pr-144]	H	17.28m	(0.3%)	0.6965	0.01342	(1%)
[Ce-144]	H	284.4d	(0.1%)	0.1335	0.110	(1.8%)
Dy-165	L	2.334h	(0.3%)	0.095	0.036	(11.1%)
Tm-166	L	7.70h	(1%)	2.0527	0.2	(10%)
Tm-167	L	9.25d	(0.2%)	0.2078	0.41	(2.4%)
Tm-168	L	93.1d	(0.1%)	0.198	0.50	(6%)
Lu-174	L	3.31y	(1.5%)	1.242	0.06	(?)
Lu-176m	L	3.664h	(0.2%)	0.088	0.087	(10.3%)
Lu-177m	L	160.1d	(0.1%)	0.105	0.12	(8.3%)
Ta-182	L	115.0d	(0.2%)	1.1213	0.351	(1.4%)
W-187	L	23.85h	(0.3%)	0.4796	0.21	(?)
Au-193	L	17.5h	(1.1%)	0.256	0.07	(?)
Au-194	L	39.5h	(1.3%)	0.328	0.01	(9.8%)
Au-195	L	182.9d	(0.3%)	0.099	0.11	(?)
Au-196	H	6.1d	(1.6%)	0.3557	0.877	(2.3%)
Hg-199m	L	42.6m	(0.5%)	0.158	0.523	(1%)
Pa-233	L	26.95d	(0.2%)	0.3119	0.37	(5.4%)
U-237	L	6.752d	(0.3%)	IC	---	
Np-239	L	2.355	(0.2%)	0.2776	0.143	(1.4%)

known to better than 0.5%, and the yield of the principle decay gamma ray should be known to better than 1%, then it is clear that much work remains to be done in this area too.

Knowledge of mass yields for fission reactions is important for passive applications of fission dosimeter reactions. The subject has been reviewed in articles by Maeck et al., Gilliam et al. and Kellogg et al. (Ref. 62). Fission mass yields are important for reactor application other than dosimetry and they have been investigated in both integral and differential experiments. The accuracies of fission yield results are no better than the accuracies for relevant decay data (see Table V). Mass yield accuracy requests are ~2% while the accuracy of the data base is presently ~5-10%. Clearly, much work is needed in this area to meet the stated objectives.

The available literature was surveyed to see how many of the ENDF/B Dosimetry File reactions (Table IV) have been investigated in the benchmark fields listed in Table III. The results of this survey appear in Table VI. Notice the limited data available for (n,2n) reactions. It is also evident that there has been only limited data testing in the NBS ISNF and ISNF/CV fields. These facilities are suitable for testing capture data and should be used for this purpose.

Table VI. An inventory of spectrum average cross section measurements in benchmark fields.

Reaction	Cf-252	U-235	1/E (ISNF/CV)	Sigma-Sigma[b]	ISNF	Big-Ju	CFRMF	YAYOI	Tapiro	Accelerator n-Be
Li-6(n,total He)						•				
B-10(n, total He)						•	•			
Na-23(n,G)Na-24	•									
Al-27(n,p)Mg-27	•	•		•		•	•	•	•	
Al-27(n,A)Na-24	•	•		•		•	•		•	•
Sc-45(n,G)Sc-46				•		•	•			•
Ti-46(n,p)Sc-46[a]	•	•		•		•	•	•		•
Ti-47(n,n-p)Sc-46[a]	•	•		•		•	•	•		•
Ti-47(n,p)Sc-47[a]	•	•		•		•	•	•		•
Ti-48(n,n-p)Sc-47[a]	•	•		•		•	•	•		•
Ti-48(n,p)Sc-48	•	•		•		•	•	•		•
Mn-55(n,2n)Mn-54	•	•								
Fe-54(n,p)Mn-54	•	•				•	•	•	•	•
Fe-56(n,p)Mn-56	•	•		•		•		•		•
Fe-58(n,G)Fe-59						•	•			
Co-59(n,G)Co-60	•					•	•		•	•
Co-59(n,2n)Co-58	•	•								•
Co-59(n,A)Mn-56	•	•						•	•	•
Ni-58(n,2n)Ni-57		•						•		•
Ni-58(n,p)Co-58	•	•		•		•	•	•	•	•
Ni-60(n,p)Co-60	•									•
Cu-63(n,G)Cu-64	•	•		•		•	•			
Cu-63(n,A)Co-60		•						•	•	
Cu-65(n,2n)Cu-64										
In-115(n,n')In-115m	•	•		•		•	•		•	•
In-115(n,G)In-116	•	•		•		•	•			
I-127(n,2n)I-126		•								
Au-197(n,G)Au-198	•	•		•		•	•		•	•
Th-232(n,f)F.P.	•	•		•				•		
Th-232(n,G)[Pa-233]		•								
U-235(n,f)F.P.	•	•		•	•	•	•		•	•
U-238(n,f)F.P.	•	•		•	•	•	•	•	•	•
U-238(n,G)[Np-239]		•		•	•	•	•			•
Np-237(n,f)F.P.	•	•		•		•	•			
Pu-239(n,f)F.P.	•	•		•	•	•	•			

[a] Activation measurements usually produce cross sections for Ti(n,2)Sc-46 or Ti(n,2')Sc-47 rather than individual components.

[b] Quite similar Sigma-Sigma type facilities are found at Mol (Belgium), NESCS (U.K.) and ITN (Romania). This is a composite list for these laboratories.

A comparison of integral cross sections with those calculated using differential data provides an overall indication of the status of the available data base. Table VII provides information on the reactions in the ENDF/B Dosimetry File. Values in this table are based entirely on ENDF/B-V. It is seen that ~29% of the reactions satisfy dosimetry goals, ~34% are known to not satisfy dosimetry goals, and a lack of integral data makes it impossible to reach a conclusion for ~37% of these reactions.

Table VII. Comparison of integral and differential cross sections using ENDF/B-V data [after Magurno, Ref. 63].

Reaction		Sig Avg$_{CALC}$* (mb)	Sig Avg$_{CALC}$** (mb)	Difference***
Li-6(n, total He)		453.10	---	---
B-10(n, total He)		483.36	---	---
Na-23(n,G)Na-24		0.268	---	---
Al-27(n,p)Mg-27		4.26	3.86±0.25	+9.4%
Al-27(n,A)Na-24	I	0.720	0.705±0.040	+2.1%
Sc-45(n,G)Sc-46		5.27	---	---
Ti-46(n,p)Sc-46		11.18	11.80±0.75	-5.5%
Ti-47(n,n-p)Sc-46		0.0084	---	---
Ti-47(n,p)Sc-47		22.48	19.0±1.6	+15.5%
Ti-48(n,n-p)Sc-47		0.0014	---	---
Ti-48(n,p)Sc-48		0.282	0.300±0.018	-6.4%
Mn-55(n,2n)Mn-54		0.202	0.244±0.015	-21%
Fe-54(n,p)Mn-54	IC	81.08	79.7±4.9	+1.7%
Fe-56(n,p)Mn-56	I	1.04	1.035±0.075	+0.5%
Fe-58(n,G)Fe-59		1.64	---	---
Co-59(n,G)Co-60	IC	6.02	---	---
Co-59(n,2n)Co-58		0.183	---	---
Co-59(n,A)Mn-56	IC	0.150	0.143±0.010	+4.7%
Ni-58(n,2n)Ni-57	I	0.0028	0.0058±0.0003	-107%
Ni-58(n,p)Co-58	IC	105.1	108.5±5.4	-3.2%
Ni-60(n,p)Co-60		2.61	---	---
Cu-63(n,G)Cu-64		9.31	9.30±1.40	+0.1%
Cu-63(n,A)Co-60		0.558	0.500±0.056	+10.4%
Cu-65(n,2n)Cu-64		0.309	---	---
In-115(n,n')In-115m	IC	179.5	189±8	-5.3%
In-115(n,G)In-116		123.68	134.5±6.0	-8.7%
I-127(n,2n)I-126		1.22	1.05±0.065	+13.9%
Au-197(n,G)Au-198	I	76.60	83.5±5.0	-9.0%
Th-232(n,f)F.P.		75.10	81.0±5.4	-7.9%
Th-232(n,G)[Pa-233]		91.10	---	---
U-235(n,f)F.P.	IC	1233.71	1203±30	+2.5%
U-238(n,f)F.P.	I	304.7	305±10	-0.1%
U-238(n,G)[Np-239]	IC	69.37	---	---
Np-237(n,f)F.P.	I	1350.00	1312±50	+2.8%
Pu-239(n,f)F.P.	I	1790.66	1811±60	-1.1%

* Differential cross sections and Watts U-235 thermal fission neutron spectrum from ENDF/B-V.

** Supplied to B. Magurno (BNL) by W. McElroy (HEDL). Values come from an evaluation by A. Fabry of integral data for a pure U-235 thermal fission neutron spectrum.

*** Difference = (Sig Avg$_{CALC}$ - Sig Avg$_{MEAS}$)/Sig Avg$_{CALC}$ expressed in percent. Values underlined indicate discrepancies exceeding 5% (or 10% for high threshold).

Conclusions

Damage dosimetry accuracy requirements have become stringent. Uncertainties in basic nuclear data will probably prevent realization of stated goals for the near term. Greater emphasis on quality rather than quantity of data is needed. It will be necessary to reduce the lists of requested cross section data rather than to expand them as has been the past trend. This should be possible if users assess their needs carefully and refrain from asking for every conceivable type of information which might be related to their specific application and focus on important requirements.

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REFERENCES

1. W. N. McElroy et al., Proc. 2nd ASTM-EURATOM Symp. Reactor Dosimetry, Palo Alto, Calif., Oct. 3-7, 1977, p. 17, NUREG/CP-0004 (Vol. 1).
2. S. M. Qaim, IAEA-Advisory Group Meeting on Nucl. Data for Fusion Reactor Technology, Vienna, Austria, 11-15 Dec., 1978.
3. J. Grundl and C. Eisenhauer, Neutron Cross Sections for Reactor Dosimetry (Vol. I), p. 53, IAEA-208, Proc. IAEA Consultants' Mtg. on Integral Cross Sections and Std. Neutron Fields for Reactor Dosimetry, Vienna, Austria, 15-19 Nov., 1976. Also, ibid. Ref. 1 (Vol. 2), p. 1177.
4. L. R. Greenwood, Argonne National Laboratory, Private Communication (1979).
5. H. Farrar, IV and E. P. Lippincott, ibid. Ref. 1 (Vol. 2), p. 725. Also, Proc. 1st ASTM-EURATOM Symp. on Reactor Dosimetry, Petten, Holland, 22-26 Sept. 1975, EUR 5667ef (Part I), p. 675.
6. E. P. Lippincott and J. A. Ulseth, ibid. Ref. 1, p. 271.
7. G. L. Simmons et al., EPRI NP-1056, Electric Power Research Institute, Palo Alto, Calif. (April 1979).
8. C. D. Bowman et al., ibid. Ref. 1 (Vol. 2), p. 1365.
9. C. A. Oster, ibid. Ref. 1 (Vol. 3), p. 1365.
10. W. L. Zijp et al., ibid. Ref. 1 (Vol. 3), p. 1333.

11. C. Ertek et al., ibid. Ref. 1 (Vol. 3), p. 1385.
12. F. W. Stallman and F. B. K. Kam, ibid. Ref. 1 (Vol. 3), p. 1411.
13. F. G. Perey, ORNL/TM-6062, Oak Ridge National Laboratory, Oak Ridge, Tennessee (October 1977). Also, ibid. Ref. 1 (Vol. III), p. 1449.
14. A. Alberman et al., ibid. Ref. 1, p. 145, Also Nucl. Technology 36, 336 (1977).
15. G. P. Genthon et al., EUR-5274 d,e,f,n, CBNM, Geel, Belgium (1975).
16. W. L. Zijp et al., ECN-36, Netherlands Energy Research Foundation, Petten, Holland (Feb. 1978).
17. S. Ganesan, Reactor Research Centre, Kalpakkan 603102, Tamil Nadu, India, private communication (notes from lectures given in 1976).
18. B. Goel, Nucl. Sci. Eng. 69, 99 (1979).
19. D. G. Doran et al., invited paper, Proc. Radiation Effects in Breeder Reactor Structural Materials, Scottsdale, Arizona, June 1977, Amer. Met. Soc. of AIME, New York (1977).
20. H. Ullmaier and W. Schilling, Proc. Int. Course on Physics of Modern Materials, ICTP Trieste, June 1979 (to be published in 1979 by IAEA).
21. A. A. Sagues et al., ibid. Ref. 19, p. 367.
22. W. G. Johnston et al., J. Nucl. Materials 48, 330 (1973).
23. J. E. Sanders, U.K.A.E.A., Reactor Physics Division, A.E.E.-Winfrith, private communication. Document based on a lecture given at the Nuclear Data Forum, London, (Dec. 1979).
24. K. Takeuchi et al., ibid. Ref. 1, p. 257.
25. G. L. Simmons, ibid. Ref. 1 (Vol. 2), p. 627.
26. F. J. Rahn et al., EPRI NP-380-SR, Electric Power Research Institute, Palo Alto, California (April 1977). Also, ibid. Ref. 1 (Vol. 3, p. 1069.
27. G. C. Martin, Jr., ibid. Ref. 1, p. 229.
28. G. Bartholome et al., ibid. Ref. 1, p. 285.
29. S. L. Anderson, ibid. Ref. 1 (Vol. 3), p. 1069.
30. E. B. Norris and J. S. Perrin, ibid. Ref. 1 (Vol. 3), p. 1109.
31. R. Odette et al., ibid. Ref. 1 (Vol. 3), p. 1123.
32. Symposium on Neutron Cross Sections from 10 to 40 MeV, Brookhaven National Laboratory, U.S.A., May 3-5, 1977, BNL-NCS-50681. Many papers.
33. M. R. Bhat, BNL-NCS-25295, Brookhaven National Laboratory (1979).
34. A. B. Smith et al., invited paper at the NEANDC Meeting held in Oak Ridge, Tennessee, U.S.A., April 3-7, 1978. Text available from authors at Argonne National Laboratory, Argonne, Illinois, U.S.A.
35. R. A. Konynenberg, UCRL-51393, Rev. 1, Lawrence Livermore Laboratory, Livermore, California, U.S.A.
36. The Fusion Reactor Materials Program Plan, Section II - Damage Analysis and Fundamental Studies, DOE/ET-0032/2, U.S. Dept. of Energy, Washington, D.C., U.S.A. (July 1978).
37. H. L. Adair, J. Van Audenhove et al., ibid. Ref. 1 (Vol. 2), p. 975; (Vol. 3) p. 987 and p. 1003.
38. Interlaboratory Reaction Rate Program, 11th Progress Report, HEDL-TME 77-34, UC-79b,d, compiled by W. McElroy, Hanford Eng. Dev. Laboratory, Richland, Washington, U.S.A. (March 1978). Many papers.
39. J. A. Grundl et al., Nucl. Tech. 25, 237 (1975).
40. G. DeLeeuw-Gierts and S. DeLeeuw, ibid. Ref. 1 (Vol. 2) p. 591. Also, ibid. Ref. 38.
41. A. Fabry et al., ibid. Ref. 39, p. 349.
42. E. Bennett, Nucl. Sci. Eng. 27, 16 (1967).
43. E. J. Mowdy, ibid. Ref. 39, p. 381.
44. J. W. Rogers et al., ibid. Ref. 39, p. 330.
45. H. D. Warren et al., ibid. Ref. 1 (Vol. 2), p. 775.
46. J. H. Roberts and R. Gold, ibid. Ref. 1 (Vol. 2), p. 739.
47. H. Farrar, IV et al., ibid. Ref. 1 (Vol. 2), p. 725. Also, see Proc. 1st Topical Mtg. on Fusion Reactor Materials, Miami Beach, Florida, U.S.A. (January 1979).
48. J. P. Genthon et al., Nucl. Inst. and Meth. 131, 1 (1975).
49. A. Alberman et al., Contrib. paper, 3rd ASTM-EURATOM Symp. on Neutron Dosimetry for Reactors, Ispra, Italy, October 1979.
50. J. P. Genthon et al., EUR 5795 e,n, ECN, Petten, Holland (1977).
51. F. Mathieu et al., ibid. Ref. 1 (Vol. 2), p. 789.
52. D. L. Smith, ibid. Ref. 3, p. 321.
53. D. L. Smith, ANL/NDM-30, Argonne National Laboratory, Argonne, Illinois, U.S.A. (March 1977).
54. W. Mannhart, RSIC Seminar-Workshop on Theory and Applic. of Sensitivity and Uncertainty Analysis, Oak Ridge, Tenn., Aug. 22-24, 1978. Text of paper available from author at PTB-Braunschweig, F. R. Germany.
55. A. Fabry et al., Proc. 1st ASTM-EURATOM Symp. on Reactor Dosimetry, Petten, Holland, 22-26 Sept. 1975, EUR 5667 e,f.
56. Summary Report, ibid. Ref. 3, ed. M. F. Vlasov.
57. J. Grundl and C. Eisenhauer, Conf. on Nucl. Cross Sections and Technology, Washington, D. C., March 3-7, 1975, NBS Special Publ. 425, p. 250.
58. W. N. McElroy et al., ibid. Ref. 3, p. 147.
59. Evaluated Nuclear Data File, Brookhaven National Laboratory, Brookhaven, N.Y., U.S.A.
60. L. R. Greenwood et al., Nucl. Tech. 41, 109 (1978).
61. Table of the Isotopes, 7th Edition, ed. by C. M. Lederer and V. S. Shirley, Lawrence Berkeley Laboratory, published by John Wiley and Sons, Inc., New York (1978).
62. Articles on the status of fission mass yields in Ref. 1 (Vol. 3): W. J. Maeck et al., p. 983, p. 1267 and p. 1279; D. M. Gilliam et al., p. 1289; L. S. Kellogg et al., p. 1307.
63. S. A. Magurno, Brookhaven National Laboratory, private communication, (1979).