START

RECORD COPY

HEDL-SA-1572

STATUS REPORT ON DOSIMETRY BENCHMARK NEUTRON FIELD DEVELOPMENT CHARACTERIZATION APPLICATION

W. N. McElroy

E. P. Lippincott

ASTM EURATOM SYMPOSIUM

October 3-7, 78

Palto Alto, CA

HANFORD ENGINEERING DEVELOPMENT LABORATORY
Operated by Westinghouse Hanford Company, a subsidiary of Westinghouse Electric Corporation, under the Department of Energy Contract No. EY-76-C-14-2170

COPYRIGHT LICENSE N[illegible]

By acceptance of this article, the publisher and/or recipient acknowledges the Government's right to retain a nonexclusive, royalty-free license in and to any copyright covering this paper.

DISTRIBUTION OF THIS DOCUMENT IS UNLIMITED

DISCLAIMER

This report was prepared as an account of work sponsored by an agency of the United States Government. Neither the United States Government nor any agency thereof, nor any of their employees, makes any warranty, express or implied, or assumes any legal liability or responsibility for the accuracy, completeness, or usefulness of any information, apparatus, product, or process disclosed, or represents that its use would not infringe privately owned rights. Reference herein to any specific commercial product, process, or service by trade name, trademark, manufacturer, or otherwise does not necessarily constitute or imply its endorsement, recommendation, or favoring by the United States Government or any agency thereof. The views and opinions of authors expressed herein do not necessarily state or reflect those of the United States Government or any agency thereof.

DISCLAIMER

Portions of this document may be illegible in electronic image products. Images are produced from the best available original document.

STATUS REPORT ON DOSIMETRY BENCHMARK NEUTRON FIELD DEVELOPMENT, CHARACTERIZATION AND APPLICATION

A. Fabry
Centre d'Etude de l'Energie Nucléaire
S.C.K./C.E.N., Mol, Belgium

J.A. Grundl
National Bureau of Standards
Washington, D.C., USA

W.N. McElroy, E.P. Lippincott
Hanford Engineering Development Laboratory
Richland, Washington, USA

Harry Farrar IV
Rockwell International
Atomics International Division
Canoga Park, California, USA

INVITED REVIEW PAPER, Second ASTM-EURATOM Symposium
on REACTOR DOSIMETRY, Palo Alto, California, October 3-7, 1977

STATUS REPORT ON DOSIMETRY BENCHMARK NEUTRON FIELD DEVELOPMENT, CHARACTERIZATION AND APPLICATION

ABSTRACT

This report attempts to present a brief, but comprehensive review of the status and future directions of benchmark neutron field development, characterization and application in perspective with two major objectives of reactor dosimetry:

1) fuel fission rate and burn-up passive monitoring,

2) correlation of materials irradiation damage effects and projection to commercial power plants.

The report focuses on the Light Water Reactor and Fast Breeder Reactor program needs.

Current interfaces between dosimetry, reactor physics, cross section metrology, solid state research and metallurgy are highlighted through a tentative classification of the benchmark fields and its justification.

A summary is given of indications drawn regarding the accuracy of fundamental fission cross sections, of dosimetry sensor nuclear data, of helium production assessment, of dosimetry fission yields, and of some other nuclear cross sections, on the basis of integral measurements in selected benchmark neutron fields. The ENDF/B IV data files are used as reference for the discussions. The success of this benchmark field approach to reactor dosimetry must not hide its present limitations and it is tried to adopt throughout a critical attitude to this respect, so as to delineate the areas where improvements are most needed.

1. DIRECTIONS OF DOSIMETRY BENCHMARK FIELD DEVELOPMENT AND APPLICATION

Benchmark neutron fields for reactor dosimetry have been documented and discussed at recent international conferences; see the proceedings of:

- First ASTM-EURATOM Symposium on Reactor Dosimetry, Petten, September 1975
- International Conference on the Interactions of Neutrons with Nuclei, Lowell, Mass., July 1976
- IAEA Consultants' Meeting on "Integral Cross Section Measurements in Standard Neutron Fields for Reactor Dosimetry", Vienna, November 1976
- International Symposium on Neutron Standards and Applications, NBS, Washington D.C., March 1977

The benchmark approach to the validation and calibration of dosimetry techniques is shown schematically in Figures 1 and 2.

As shown in Figure 1 input information is the best estimate $X_0(E)$ of an energy-dependent function $X(E)$ and differential-energy response functions $y_i(E)$ which are:

- reaction cross sections $\sigma_i(E)$ when $X(E)$ is a differential-energy flux spectrum $\phi(E)$,
- flux spectra $\phi_i(E)$ when $X(E)$ is a differential-energy reaction cross section $\sigma(E)$ or a damage function $\sigma_d(E)$

Adjustment or unfolding codes are used to modify $X_0(E)$ within its assigned uncertainties until a solution $X(E)$ is obtained which produces measured integral responses - reaction rates R_i or materials property changes ΔP_i - within their assigned uncertainties, according to the constraint

$$Y_i = \int y_i(E)\, X(E)\, dE.$$

Given the above, the purpose is then to derive consistent projected integral responses Y_i', generally materials property changes, for in-service conditions in commercial reactors.

The functionals $y_i(E)$ and $X(E)$ can and possibly should be simultaneously adjusted. These functions are by no means of direct technological interest, but are essentially transfer functions between the measured integral quantities Y_i and the projected ones Y_i'. These energy-dependent transfer functions, are amenable however to observation and/or theoretical assessment.

A strategy, which has become widely accepted in the last few years, to take maximum advantage of all available information, is detailed schematically in Figure 2.

Two main objectives of reactor neutron dosimetry are well indicated on this second figure:

1) calibration and validation of radiometric and stable fission product passive sensors for fuel fission rate and burn-up measurements

2) correlation of materials radiation induced damage.

This includes the validation and/or adjustment of nuclear data - cross sections, fission yields - necessary for interpretation of the dosimetry sensor responses.

The uncertainty ranges shown in Figure 2 are typical of the current state-of-the-art capabilities developed in the frame of the Fast Breeder Reactor (FBR) program.[1]

In the last two years, renewed emphasis has been placed also on covering the needs of the Light Water Reactor (LWR) program, in particular pressure vessel surveillance (PVS).[2,3,4,5]

A number of new, relevant neutron fields have been identified, some of which do not match the current classification,[6] Table I, into standard, reference or controlled environment benchmark fields; it has been proposed to label them as "test regions". (See the chairman's report of the workshop on benchmarks is conference). The essential feature in which test regions differ from, say, any arbitrary position in a nuclear reactor, is the extent and quality of their environmental characterization by dosimetry methods, making them most suitable for the improvement of radiation damage analysis (Figure 2).

Note that only neutron benchmark fields are considered in this report; e.g., gamma ray benchmark fields developed in relation to heating and to shielding applications are not reviewed.

Also, only dosimetry fields are discussed; e.g., fields in which passive dosimetry sensors are or have been or will be used; this thus excludes from the present report most reactor physics benchmarks, whether clean assemblies or mock up criticals, and most one-material nuclear data testing experiments.

Table II is an attempt to delineate somewhat further the directions of benchmark neutron field development and use for LWR and FBR programs. The table is a matrix in which major fields are grouped within the four field categories and four interrelated areas of applications:

1) Validation and/or Calibration of Sensors

This application encompasses:

- Performance testing of passive dosimetry sensor sets; in particular, careful interlaboratory comparisons of individual sensors

- Dosimeter sensor quality assurance; most crucially, mass assay

- Calibration of passive and active sensors with respect to relevant standards; some examples are:

  a) calibration of commercial fission chambers by exposure to the MOL-II field[24]

  b) calibration of radiometric fission foils as fission rate indicators by irradiation in CFRMF[12]

  c) distribution of threshold activation foils exposed to a certified ^{252}Cf fission spectrum fluence (NBS)

- Flux transfer[25] from ^{252}Cf standard fields to actual reactor environments by means of ^{239}Pu and/or ^{238}U fission sensors (usually fission chambers).

2) Validation and Adjustment of Dosimetry Sensor Nuclear Data[1][26][27][28]

This area is reviewed in subsequent sections.

3) Validation of Damage Exposure and Correlation Parameters

This area, which is a most crucial one, especially if effective advantage is to be taken from advanced dosimetry methods, calls for good interdisciplinary cooperation between the dosimetrist, the metallurgist and the solid state physicist.

International adoption of the dpa, number of displacements per atom,[29] as a common exposure unit for damage experiments, whether in reactors or in particle accelerators, constitutes a first significant achievement. Property changes in materials are, however, related to specific types of residual defects rather than to dpa. Physically grounded multiparametric data fits are

desirable and indeed necessary to quantify temperature,[3] gas production,[30] and other effects that influence the correlation of radiation damage in different environments;[31] the data base would be beneficially expanded if specimen microstructure, chemistry and even mechanical stresses could be properly accounted for using advanced correlation methods: clearly, a long way to go. In this respect, it must be pointed out that the existence of non-linear interferences between various defect production mechanisms may in the long term reveal that the damage function approximation,[32] e.g. the definition of energy-dependent damage cross sections, is not satisfactory, and minor alterations of the logistics schematized in Figures 1 and 2 may thus become warranted.

The important trend however is the separate identification of test region neutron fields for data development and testing: an appropriate focus for the needed continuation and amelioration of interdisciplinary efforts.

In Table II are indicated those neutron fields in which important joint metallurgy and high quality dosimetry programs are underway or firmly planned.

4) Validation of Reactor Physics Data and Codes for Dosimetry Applications

Reactor physics and dosimetry are inseparable. Physics calculations, even when inaccurate as in complex geometrical arrangements or in locations subject to time-varying perturbations, are needed to define a best estimate of the neutron spectrum required as input to unfolding codes (Figure 1); conversely, the adjustment of neutron spectra on the basis of integral reaction rate measurements provides insight into biases of physics modeling and of nuclear data (see next sections). Often, reactor physics computations are used also to extrapolate dosimetry and metallurgical data from test or surveillance to in-service conditions (Figure 2). The validation of such calculations is essential when the range of extrapolation is sizeable as in LWR pressure vessel surveillance.

Significant recent progress has been made on the application of transport theory methods in this area of benchmark fields by the development of a capability for credible flux spectral and reaction rate uncertainty analysis employing multigroup cross section covariance files.[33] This capability has been applied so far to GODIVA and JEZEBEL.[34]

2. FLUX SPECTRAL CHARACTERIZATION OF DOSIMETRY BENCHMARK NEUTRON FIELDS

An adequately accurate characterization of multigroup neutron flux spectra in dosimetry benchmark fields generally requires[21,35] a combination of reactor physics calculations, neutron spectrometry, and integral reaction rate measurements. The weighting to be made between these complementary types of information depends on the benchmark and the energy range considered; this is often a matter of delicate judgment for the evaluator. For example, a family of fields, the Intermediate-energy Standard Neutron Field (ISNF)[10,36] (and the near-1/E version labeled ISNF/CV) has specifically been designed so as to be governed by simple and well understood physical processes[37] and thus to be computable with group flux uncertainties less than $\pm$ 5% below 1 MeV. Fission neutron spectra on the other hand are obviously not amenable to a transport theory assessment (except small corrections for neutron scattering and absorption within the sources and experimental structures). Finally it is noteworthy that in-pile neutron spectrometry techniques[38,39] are usually not applicable to test regions; an exception is, perhaps, the cavity behind the pressure vessel of a LWR plant just before the initial start of full power operation.[4]

Neutron spectrometry in benchmark fields is often essential in view of the current deficiencies of calculations and also because of the lack of response of present dosimetry sensors in the energy range ~10 - 500 keV.

A great achievement of modern neutron spectrometry is to have established the ^{252}Cf spontaneous fission neutron spectrum as a primary standard field in the energy range ~0.25 - 8 MeV, where it is known with an accuracy of

± 2 - 5%;[40] outside of this range, more work is warranted and is in progress. A most valuable test of neutron spectrometers today is to expose them to a Californium source; if these are devices used for in-pile measurements however, precaution and corrections are needed to relate this beam-type geometry with the one in environments where the angular flux spectrum may range from isotropic to one highly peaked along some streaming direction.

The ISNF and ISNF/CV assemblies are the other primary standard fields relevant to dosimetry applications. Californium-252 driven versions will provide an ideal test bed for confrontation of in-pile spectrometry techniques with reliable physics calculations and with integral measurements.

All other dosimetry benchmark fields are more poorly characterized than ^{252}Cf and ISNF, even the thermal-neutron induced uranium-235 fission neutron spectrum,[21,26,28,40,41] which is nevertheless considered a standard (Table II). A brief discussion of the present status for some reference fields of the FBR program is relevant here; further comments are given in the next section.

Neutron spectroscopy techniques applied so far in FBR reference dosimetry benchmarks are generally limited to proton recoil proportional counters, and to the $^{6}Li(n,\alpha)$ sandwich spectrometer.[42] In the MOL-II facility,[11] a sophisticated $^{3}He(n,p)$ spectrometer[43] has also been used successfully.[44] Time-of-flight has been done in the RPI iron blocks[14] in relation to physics applications, but not yet in an arrangement suited for dosimetry purposes. The II interlaboratory comparison (of five independent spectrometers) has shown[44] that group flux spectra can be measured to within ± 5% in the energy range ~20 keV - 2 MeV. The ^{6}Li (n,α) and gas proton recoil technique have been applied also in BIGTEN[45] and CFRMF;[46] in this last assembly, yet unsatisfactorily resolved discrepancies between ~100 and ~500 keV are a challenge to the experimenters and the evaluators.

So, neutron spectrometry in reference dosimetry neutron fields may presently provide a good characterization between 20 keV and 2 MeV, where it is

considered a most valuable validation constraint (Figure 2) on both calculations and integral unfolding schemes. Outside of this range, less weight is given to spectroscopy results.

Below 20 keV, the experimental data are less quantitative, but are still useful because calculations are often so unreliable[47] that even predicted fission rates may be affected in soft spectra. Time-of-flight measurements, whenever feasible, would be helpful. It is true however that the low energy spectral detail below 20 keV, so important to safety aspects of the FBR, is generally of marginal significance in terms of major dosimetry and metallurgical objectives (Figure 2). The most reasonable strategy to overcome low-energy spectral uncertainties in FBR benchmarks consists in validating selected sensors by exposure in the ISNF and using them for subsequent spectral adjustment of reference fields. Such adjustments have already been done[21,28] on the basis of the gold capture reaction and ISNF validation is in progress.

The energy range above 1 MeV is extremely important for radiation damage correlation. In this respect, the application, whenever feasible, of alternate proton recoil techniques such as nuclear emulsions, liquid scintillators and simple solid state sandwich spectrometers, or of other methods as yet exploratory,[39] should be most actively encouraged.

At this moment, from an evaluation viewpoint, integral spectral adjustment based on the best known threshold reactions (category I[21,28,43] cross sections on Figure 2) remains the most accurate approach to assess the group flux spectrum above 1 MeV in reference benchmark neutron fields.[21,28] This is true because of the lack of systematic spectrometry efforts in this range[*] and of the comparatively large uncertainties of transport theory computations - which is often traceable to uncertainties in neutron inelastic

(*) It seems that the focus of neutron spectrometry has been on FBR physics where, in contrast to FBR dosimetry applications, the energy range above 2 MeV is not very important.

scattering data. An appropriate illustration is given in Table III for two reference dosimetry fields, CFRMF and BIGTEN. Uranium-238 inelastic scattering cross section changes between the ENDF/B III and ENDF/B IV files are responsible for up to ~10% variations of the predicted integral reaction rates of threshold reactions sensitive in the range 0.5 - 5 MeV. Note that the hardening of the neutron spectra suggested by ENDF/B IV is relatively consistent with the implications of spectral adjustment (by means of the SAND-II code[21,49]) based on category I - type integral reaction rates. This is, however, in disagreement with $^{6}Li(n,\alpha)$ spectrometry results[45,46] which support ENDF/B III in this energy range. The integral adjustment results have been published[21,28] well before calculation results based on ENDF/B IV became available; they were and still are considered reference information; on this basis, version IV may be considered an improvement over version III, despite evident residual deficiencies.

In summary, despite considerable progress, further improvement of the accuracy of benchmark neutron field spectral characterization is desirable.

3. VALIDATION OF HIGH-ENERGY FISSION CROSS SECTIONS BY INTEGRAL MEASUREMENTS IN DOSIMETRY BENCHMARK NEUTRON FIELDS

One of the traditional ways to check the consistency of fission cross sections of a new evaluated nuclear data file is to examine how well critical masses and measured integral fission rate ratios are predicted in clean multiplying assemblies. For example, in the U.S., the successive versions of the ENDF/B files are submitted to such integral data testing using a series of fast criticals selected and documented by the Cross Section Evaluation Working Group (CSEWG).[50]

Dosimetry standard and reference neutron fields, Table II, generally do not provide adequate criticality information for data testing; some of them however, have been subject to careful, well documented and accurate fission rate ratio measurements. Their spectral characterization, despite its weaknesses (Section 2), is comparable in quality to the physics benchmarks and

sometimes it is even superior. Consequently, some of the dosimetry benchmark fields are most suited to integral data testing of fundamental fission cross sections. A brief examination of the implications of such testing for the ENDF/B IV file is warranted and is presented in Table IV.

The table is constructed using the latest absolute integral measurement results (see for example references[37,51,55]), most of which have been obtained by means of NBS absolute fission chambers;[56] exceptions are GODIVA and JEZEBEL (for these assemblies, the data relate to monoenergetic calibrations at 2.5 MeV where absolute ratio measurements have been performed for normalization and are backed by 1966 interlaboratory comparisons in Flattop).[53]

The differential-energy flux spectral shapes accepted for the calculated terms in this confrontation are as follows:

χ_{82}, χ_{25}: NBS evaluation[40]

JEZEBEL, GODIVA, BIGTEN, CFRMF: transport theory characterization based on ENDF/B IV

ISNF: transport theory characterization based on ENDF/B III (minor variation expected if ENDF/B IV had been used instead)

ST: CEN/SCK evaluation,[11] known however to be in error above 1 MeV.

In the fourth column of the table, the absolute $^{235}U(n,f)/^{238}U(n,f)$ measurement results have been renormalized to uranium-235 fission neutron spectrum calibrations : for this highly spectral-sensitive index, the latter calibration provides measured-to-computed ratios more strictly indicative of transport theory calculation deficiencies for all fields which are not pure sources: this normalization eliminates potential biases in absolute measurements and partially compensates for uncertainties in the knowledge of the source spectrum. Comparison of columns 3 and 4 highlights how different the

testing results can be for $^{235}U(n,f)/^{238}U(n,f)$ if the experimental values are based on absolute fission rate measurements rather than on fission neutron spectrum calibrations. This is the case also for the $^{237}Np(n,f)/^{235}U(n,f)$ and $^{237}Np(n,f)/^{238}U(n,f)$ ratios, but not for the $^{239}Pu(n,f)/^{235}U(n,f)$ ratio.

The $^{239}Pu(n,f)/^{235}U(n,f)$ ratio is largely spectrum insensitive in each benchmark and as absolutely measured, it departs from ENDF/B IV predictions by a constant 4 (± 2) %, a trend also observed for the ZPR-6 physics benchmarks.[54] Furthermore, the measured absolute $^{235}U(n,f)$ average cross section in the Californium -252 fission spectrum is about 3 (± 2) % lower than the computed one using ENDF/B IV.[51]

It is tempting to conclude that in the energy range 0.1 - 5 MeV the $^{235}U(n,f)$ cross section in ENDF/B IV may on the average be overestimated by ~3 - 4% while fortuitously, $^{239}Pu(n,f)$ would be consistent with integral data. Such paradoxical speculation seems to be supported by JEZEBEL criticality. Indeed, column 4 of Table IV suggests that the transport theory flux spectrum characterization of JEZEBEL is adequate. This is not too surprising: the JEZEBEL spectrum is comparatively insensitive to uncertainties in inelastic scattering data because the importance of neutron downscattering relative to the fission spectrum component is modest (to give an idea : the $^{235}U(n,f)/^{238}U(n,f)$ ratio is 3.94 in χ_{25}, 4.74 in JEZEBEL, 6.21 in GODIVA and 26.9 in BIGTEN). If the JEZEBEL spectrum is not grossly in error (and according to the table, it is not) the critical mass is a close test of the $^{239}Pu(n,f)$ cross section scale, as its uncertainties are smaller. The adequate prediction of JEZEBEL criticality by ENDF/B IV supports the suggestion of Table IV that the $^{239}Pu(n,f)$ cross section may be correct in integral terms while the discrepancy on GODIVA critical mass is consistent with two other implications of integral cross sections in dosimetry fields, namely:

a) that the $^{235}U(n,f)$ cross section in ENDF/B IV may be too high, and

b) that the ^{235}U inelastic scattering cross section in ENDF/B IV is too low, or the downscattering elements seriously biased (column 4 of Table IV).

These speculations seem to be at variance with a recent sensitivity analysis of GODIVA and JEZEBEL,[34] and indeed, if attention is restricted to the absolute $^{235}U(n,f)/^{238}U(n,f)$ ratio for these two fields, column 3 of Table IV, it is JEZEBEL which seems poorly predicted by transport theory and not GODIVA. But the contradiction is only apparent and would presumably disappear if a large enough array of data, i.e. that in Table IV, was considered in the sensitivity studies; column 4 of the table supports this statement rather unequivocally.

The $^{237}Np(n,f)/^{238}U(n,f)$ ratio is much less spectral sensitive than the $^{235}U(n,f)/^{238}U(n,f)$ ratio; indeed, at the present level of accuracy of spectral characterization for all benchmarks in the table, except FF, it is possible that the trend indicated by the last column is traceable to a small bias, -3 ($\pm$ 2.5) %, in the $^{237}Np(n,f)$ cross section in ENDF/B IV.

The BIGTEN and CFRMF discrepancies in column 4 of Table IV point towards a bias in the ^{238}U downscattering data in ENDF/B IV; e.g. not enough neutron removal from above to below ~2 MeV is calculated. It such bias was to be expressed only in terms of the total inelastic scattering cross section for ^{238}U, this cross section would be too low in ENDF/B IV and much too high in ENDF/B III. Note that $^{6}Li(n,\alpha)$ spectrometry however tends to support ENDF/B III while the RPI time-of-flight measurements on depleted uranium[55] suggest to lower $\sigma_{n,n'}$ for ^{238}U even more than has been done in ENDF/B IV. Such conflicting implications of integral microscopic measurements and differential macroscopic observations relative to ^{238}U downscattering data call for more work in this area of relevance to both the FBR and LWR programs.

Summarizing, integral fission cross section measurements in dosimetry benchmark neutron fields indicate that:

- the ^{239}Pu and ^{238}U fission cross sections in ENDF/B IV may be adequate within uncertainties of $\pm$ 3 %, on the average(*)

- the ^{235}U and ^{237}Np fission cross sections in ENDF/B IV may be biased by respectively + 3 - 4 (± ?) % and - 3 (± ?) %, on the average(*)

- the selection of ^{235}U and ^{238}U inelastic scattering data in ENDF/B IV is still unsatisfactory, but constitutes a significant improvement over ENDF/B III.

4. STATUS OF DOSIMETRY CROSS SECTIONS

A number of new compilations[27,56] and evaluations[57] to [60] have been published recently for the differential-energy cross sections of nuclear reactions relevant to reactor dosimetry; see reference[61] for a survey and intercomparison of part of these data. The ENDF/B dosimetry cross section file version IV[57] has gained growing international recognition as a reference file and is the focus of the present brief outline.

The ENDF/B IV dosimetry file has been subjected to extensive integral data testing[28] in benchmark neutron fields using an approach recommended by an IAEA Consultants' meeting in 1973,[48] see illustration on the upper part of Figure 2. A set of reactions, labeled category I, has been selected for which differential-energy cross sections are a priori among the best known. They have been used to adjust the low- and high-energy tails of benchmark field neutron spectra (see discussion in Section 2): the selection of these category I reactions was indeed heavily weighted also by the criterion that such spectral adjustments should be negligibly small in energy ranges where the spectrum is well established by neutron spectrometry

(*) No suggestion is possible for what concerns the detailed shapes of fission cross sections; compensation effects due to erroneous shapes may lead to apparent consistency or inconsistency between differential and integral data.

measurements (for instance, 0.25 to 8 MeV in the ^{252}Cf fission spectrum, 20 keV to 2 MeV in ZZ). Transport theory calculations based on ENDF/B III were generally used to provide input neutron spectra (Figure 1) for the unfolding code, SAND II[49] in this case. The trends indicated by these spectral adjustments are illustrated in reference[21] and discussed in Sections 2 and 3 of this paper. On the basis of these adjusted benchmark neutron spectra, the consistency of differential-energy cross sections for category II reactions has then been examined.[28]

The status of the testing is summarized in Table V. A few reactions not included in the ENDF/B IV file, but requested for version V, have been assessed directly in terms of a quick appraisal of published differential measurements.

In this table, consistency at a level of $\pm$ 5 % (1 σ) or better has been singled out to attempt a simplified classification of the dosimetry reactions: all reactions falling in the upper part of the table belong to category I or should be considered as serious category I candidates. The $\pm$ 5% accuracy level is not entirely arbitrary: although accuracies in the range of $\pm$ 1 - 2% are requested for some cross sections - most noticeably for fission in ^{235}U and ^{239}Pu, the accuracy presently attainable by differential measurements for many of the reactions in the table, and especially for the threshold reactions, is at best $\pm$ 5%. Therefore, generally speaking, state-of-the-art consistency is achieved for the 17 reactions in the upper part of Table V.

For many reactions, integral cross section measurements have an accuracy better than $\pm$ 3%. This is the case for essentially all the reactions in the upper part of Table V and on another hand, gross errors in the energy-dependent shapes of these cross sections are believed unlikely. It has been suggested that under these conditions, differential-energy cross sections for category II threshold reactions might valuably be rescaled (by less than 5%) so as to provide improved matching between their measured and computed values - computed on basis of neutron spectra adjusted by means of category

I reactions.[48] More generally, these field-independent measured-to-computed integral cross section ratios, called bias factors,[28] may help the evaluator to reduce current uncertainties of differential-energy cross section data.

It is worth stressing here that the benchmark fields used for testing of the ENDF/B IV dosimetry file have so far been limited to fission spectra - ^{252}Cf and ^{235}U -, [illegible], CFRMF and BIGTEN, all of which have been extensively studied by the Inter-Laboratory Reaction Rate (ILRR) programme.[1] For threshold reactions with major energy response above ~2 MeV, these fields provide complementary and somewhat redundant information: redundant because the spectral shapes above ~2 MeV are very similar for the last four fields; complementary because the most accurately known spectrum, ^{252}Cf, provides a less complete and accurate integral data set, while CFRMF (less well characterized in spectral shape), has the most complete and accurate microscopic integral cross section data base. For non-threshold reactions, ΣΣ and CFRMF are largely redundant to each other, as are the two fission spectra with each other, but the major drawback in terms of data testing is the lack of accuracy in the characterization of the low-energy spectral details. High accuracy measurements in the ISNF[36] are needed to bring firm ground to the integral data testing for non-threshold reactions (see also Section 2).

In summary, it must be stated that dosimetry for LWR and FBR reactors benefits from a relatively satisfactory sensor cross section data base validated by careful benchmark neutron field testing, and that areas for further work and improvements are rather well delineated. By contrast, the knowledge of sensor cross section data in the energy range of relevance to fusion reactors is poor at this state. This gap must be filled if the best cost to benefit ratio for all three types of reactor projects, LWR, FBR, and CTR, is to be derived from current and planned materials testing research and development efforts.[2]

5. HELIUM PRODUCTION MEASUREMENTS IN DOSIMETRY BENCHMARK NEUTRON FIELDS

An important addition to the multiple foil approach to neutron dosimetry has been development and use of helium accumulation fluence monitors (HAFM's).[62,63,64] These monitors generally consist of miniature capsules containing various elements and isotopes with different (n,α) cross section energy responses. HAFM's operate on essentially the same principle as the radiometric multiple foils except that the nuclear product of interest in each case is helium, which can be measured with ~2% absolute accuracy over a very broad concentration range using isotope dilution mass spectrometry.[64] The fact that the product is stable makes this method very attractive for long term fluence measurements.

The first HAFM's contained natural boron and enriched ^{6}LiF, and now these are included routinely as an integral part of the U.S. Breeder Reactor Program's multiple foil flux-fluence spectral sets.[63] For typical power reactor fluences, the amounts of boron and ^{6}LiF are minimized (~0.1 mg) to keep the helium generation down to levels of $<10^{17}$ atoms for convenience of mass spectrometric measurement. The sensitivity of the mass spectrometer system,[64] however, is such that helium levels as low as 10^{11} atoms can also be measured with ~2% absolute accuracy. This has made it possible to measure the helium generation rates of boron and ^{6}LiF in CFRMF, BIGTEN, ΣΣ, and in the fission cavity neutron spectrum of the BR1 reactor at MOL. A planned increase in the power level of CFRMF, moreover, will make it feasible to integrally test HAFM's containing materials such as ^{9}Be, ^{14}N, and ^{32}S, which have relatively high integrated helium production cross sections for breeder reactor neutron spectra.

When the results of the ^{10}B and ^{6}Li helium production cross sections for CFRMF and BIGTEN are compared with certain radiometric reaction measurements made during the same irradiations, a discrepancy appears between the measured and calculated helium production rates. Even though the ^{10}B and ^{6}Li (n,α) cross sections are considered to be well known, particularly at energies below 100 keV, it appears that adjustments to each, and also other

standards such as the ^{235}U (n,f) cross section, may be required to get consistency.[63] It is expected that further analysis of the results, and the additional data from [illegible] and the fission cavity spectrum, will provide an improved knowledge of the ^{10}B and ^{6}Li cross sections, and will continue to increase the accuracy of the HAFM's using these reactions for fluence-spectrum determinations.

6. DOSIMETRY FISSION YIELD DATA

Dosimetry fission rate measurements are carried out in most environments by measurement of one or a few fission products, which may have short half-lifes, long half-lifes, or be stable. Thus the yield of these fission products, together with the efficiency of detection of these products, must be known in order to determine the absolute number of fissions produced. This need for highly accurate fission yield data on a few isotopes is in contrast to needs of other users of fission yield data, e.g. decay heat evaluators, who need information on all fission products normalized to give exactly one light and one heavy mass product per fission.

It has been shown in addition, that some fission yields vary significantly with the energy of the neutron causing fission.[65] Thus for these isotopes, the fission yield in the spectrum of interest must be known. Accordingly, measurements of fission yields in benchmark neutron environments provide a means for increasing accuracies of fission rate measurements in similar neutron fields.

In addition, measurement of fissions in benchmark fields has been used to directly reference fission product measurement techniques to absolute fission rate measurements using fission chambers or solid state track recorders. This "K factor" technique thus combines the fission yield data together with all efficiency factors into one absolute normalization constant. This technique has been shown to be very accurate in the ranges in

which it has been applied. However, benchmark field normalizations to absolute fission rates have only been available in low power facilities. Application to dosimetry measurements in high power facilities, which use long half-life products such as ^{137}Cs or stable products like ^{148}Nd has not yet been done. Thus extrapolation from the other measurements is required.

Present uncertainty estimate for absolute fission rate measurements in high power facilities are in the range of 5%. It is expected that referencing to the benchmark field measurements will allow this uncertainty to be reduced to 2% under carefully controlled conditions. Further reduction to goal accuracies of 1% will require substantial efforts.[2]

7. PRESENT LIMITATIONS OF THE BENCHMARK FIELD APPROACH TO REACTOR NEUTRON DOSIMETRY

The purpose of this last section is to tentatively indicate further areas of dosimetry benchmark neutron field research which presently are major weak links in the perspective of meaningful reactor fuels and materials irradiation data development, testing, exchange, and application. Various critical comments have been made in the previous sections, in particular Section 2, about spectral characterization, and are not repeated here.

Perhaps, the most striking weakness of the benchmark dosimetry approach to-day is the lack of a high flux standard or reference neutron field, adequately tied to the existing low flux level benchmarks.

In Section 1, it has been stressed for instance, that benchmark neutron fields allow the calibration and validation of dosimetry sensors. The sensors referred to are different from the ones applied in actual reactors, and so are the geometries for measurement in the case of radiometric detection. These two gaps, a) sensor identity and b) detector sensitivity for sensor response assessment, are difficult to bridge in practice. Mass assay and neutron field perturbation problems largely depend on the sensor specifications. An interlaboratory radiometric measurement consistency established

for specimens exposed to traditional standard, reference, and controlled environment benchmark fields[1] (Table II) may not exist or be easily attainable for high fluence samples.

It is important to note that essentially all radiation damage effect studies are carried out in reactor test and surveillance facilities which are only indirectly characterized; even if this is done by application of a validated dosimetry capability, Figure 2, this is a weak link in the application of standardized dosimetry methods and data. In this regard, current LWR dosimetry programs may be considered to represent a more rigorous approach in which low power mock-ups of high flux materials-test environments are realized and thoroughly investigated from a dosimetry standpoint.[2,4,23]

The FBR program is also somewhat open to such considerations. Examples are the current plans to mock up in SNEAK the start up configuration for SNR 300, and maybe most impressively, the FFTF In-Reactor Irradiable (IRT),[19] which will allow, if fully implemented, the systematic application in an actual FBR environment of all characterization and dosimetry techniques developed by the ILRR program participants[1] and by other specialized groups.

Along the same line, a high flux FBR reference field, NIGHTMARE (the NBSR Intermediate-energy Glory Hole Tailored Mass Assay and Reference Environment) has been proposed recently: it is a boron-10 tailored cylindrical arrangement to be installed in a glory hole at the center of the NBS reactor, in the heavy water gap decoupling the upper and lower core halfs. The flux spectrum is expected to be similar to the one in the IRT, but the flux level will be in the range of 10^{14} cm^{-2} sec^{-1}.

Another serious weakness of the current benchmark approach is the restricted availability of high-quality fissionable dosimetry sensors of ^{235}U, ^{238}U, ^{239}Pu and ^{237}Np, and the lack of interlaboratory assay and quality assurance comparisons for sensors of widely different masses. International cooperation in this area is essential and is urgently needed.

The helium production measurement capability developed at Atomics International[62] for application in benchmark neutron fields is so far unique in terms of its remarkably low sensitivity. International standardization of dosimetry requires such techniques to become more widely available. Also, the serious discrepancies relative to helium production in ^{10}B and ^{6}Li in CFRMF and BIGTEN (Table III) call for an independent method to measure these reaction rates.

Another, long-standing weakness, of dosimetry methods in terms of radiation damage correlation is the lack of benchmark field validated sensors with threshold response below ~0.5 MeV.

Damage response monitors such as the graphite GAMIN detector[66] may in principle help to remedy this situation, but in practice, they are delicate to apply and so restricted in fluence range that they cannot be exposed in most benchmarks nor in power environments. An exception may be quartz damage sensors[67] insofar as pressure vessel surveillance applications are concerned: these sensors can be made sensitive in the fluence range $\sim 10^{16}$ to 10^{19} cm^{-2} with expected negligible temperature response below 500°C.[68] Additional work in this area is recommended.

Finally, it is believed that improved error propagation methods that allow one to account[69] for the off-diagonal elements of covariance matrices in dosimetry adjustment schemes (Figure 1) may lead to a most beneficial progress in the application of benchmark neutron field data.

REFERENCES

1. W. N. McELROY et al. - Special Series, Nucl. Techn. 25, 177-422 (1975).

2. W. N. McELROY, D. G. [illegible], R. GOLD, L. P. LIPPINCOTT, J. G. [illegible], W. C. MASON, G. A. GRUNDL, E. D. McGARRY, F. B. K. KAM, and C. F. [illegible] - "Standardization and maintenance of dosimetry and damage analysis for U.S. LWR, LMBR and HTGR development programs." [illegible] SYMPOSIUM ON REACTOR [illegible]: [illegible] AND SPECTRAL MATERIALS, October 3-7, Palo Alto, [illegible].

3. G. P. GOLLIE, N. DUDLEY, W. N. McELROY, R. WILLARD and A. FABRY - "Application of Advanced Irradiation Analysis Methods to [illegible] Reactor Pressure Vessel Test and Surveillance Programs." (See Reference 2)

4. F. J. RAHN, C. Z. SERPAN, A. FABRY, W. N. McELROY, J. A. GRUNDL and J. DEBRUE - "Trends of Light Water Reactor Dosimetry Programs." (See Reference 2)

5. C. Z. SERPAN - "Influence of Neutron Dosimetry on Radiation Damage Considerations in LWR Pressure Vessels" - Proceedings of the First ASTM-EURATOM Symposium on Reactor Dosimetry, EUR 5667 e/f Part I, 41 (1977).

6. J. A. GRUNDL, A. FABRY - "Workshop on Reactor Dosimetry Benchmarks" Proceedings, Report EUR 5667 e/f/ Supplement, p. 7 (1977).

7. J. A. GRUNDL, V. SPIEGEL, C. M. EISENHAUER, M. T. HEATON II, D. M. GILLIAM and J. BIGELOW - Nucl. Techn. 32, 315 (1977).

8. W. G. ALBERTS, J. BORTFELDT, E. GUNTHER, K. KNAUF, M. MATZKE, G. PASSL, V. SIEGEL AND K. F. WALZ - Proceedings, Nuclear Cross Sections and Technology Conference, NBS Special Publication 425, vol. I, 271 (1975).

9. A. FABRY, J. A. GRUNDL, C. EISENHAUER - Ibid. vol. I, 254 (1975).

10. C. M. EISENHAUER, J. A GRUNDL, A. FABRY - Proceedings, Neutron Standards and Applications, NBS Special Publication 493, 379 (1977).

11. A. FABRY, G. AND S. DE LEEUW - Nucl. Techn. 25, 349 (1975).

12. J. W. ROGERS, D. A MILLSAP, Y. D. HARKER - Nucl. Techn. 25, 330 (1975).

13. G. AND S. DE LEEUW - "One-Material Experiments in the Frame of Power Reactor Pressure Vessel Benchmarks." Consultants' Meeting on Integral Cross Section Measurements in Standard Neutron Fields, IAEA, Vienna, November 15-19 (1976).

14. R. C. BLOCK, M. BECKER, B. K. MALAVIYA - "Benchmark Integral Spectrum Measurements in Single Materials for the Resolution of Uncertainties in Data and Methods." (See Reference 2)

15. G. E. [illegible], S. C. PAYNE - "Revaluated Critical Specifications of Some GCFR/LASL Fast Neutron System." Report LA-4208 (1969).

16. E. [illegible], F. J. [illegible], E. A. PLASSMAN - Nucl. Technol. 25, 301 (1975).

17. A. [illegible] "International Intercomparison of Neutron Spectra Evaluation Methods Using Activation Foils." Report [illegible], Kernforschungsanlage Julich, Germany (1977).

18. A. [illegible], M. MARTINI, M. SALVATORES - Energia Nucleare (Milan) 20, 11, 612 (1973).

19. J. W. DAUGHERTY, R. A. BENNETT, W. L. BUNCH, W. N. McELROY, T. J. KING - "FFTF Reactor Characterization Program." (See Reference 2)

20. K. KOBAYASHI, I. KIMURA, M. HAYASHI, M. AKIYAMA - J. Nucl. Sci. and Technn. 13, 10, 581 (1976).

21. W. N. McELROY, R. GOLD, E. P. LIPPINCOTT, A. FABRY, J. H. ROBERTS - "Spectral Characterization by Combining Neutron Spectrometry, Analytical Calculations and Integral Measurements." (See Reference 13)

22. J. A. GRUNDL, C. EISENHAUER - "Benchmark Neutron Fields for Reactor Dosimetry." (See Reference 13)

23. F. CASALI, J. M. CERLES, J. DEBRUE - "The Role of Irradiation Reactor Mock-ups." (See Reference 2)

24. A. FABRY, I. GIRLEA - "Quality Control and Calibration of Miniature Fission Chambers by Exposure to Standard Neutron Fields." (See Reference13)

25. J. A. GRUNDL, C. EISENHAUER - Proceedings, NBS Special Publication 493, 156 (1977).

26. A. FABRY, H. CEULEMANS, P. VANDEPLAS, W. N. McELROY, E. P. LIPPINCOTT - Proceedings, EUR 5667 e/f Supplement, p. 23 (1977).

27. M. F. VLASOV, A. FABRY, W. N. McELROY - "Status of Neutron Cross Sections for Reactor Dosimetry." Proceedings, International Conference on the Interactions of Neutrons with Nuclei, Lowell, Mass., July 6-9 (1976).

28. A. FABRY, W. N. McELROY, L. S. KELLOGG, E. P. LIPPINCOTT, J. A. GRUNDL, D. M. GILLIAM, G. E. HANSEN - "Review of Microscopic Integral Cross Section Data in Fundamental Reactor Dosimetry Benchmark Neutron Fields." (See Reference 13)

29. Proceedings, IAEA Specialists Meeting on Radiation Damage Units, Harwell, UK, 2-4 November 1976.

30. R. L. SIMONS - Proceedings, Irradiation Effects on the Microstructure and Properties of Metals, ASTM STP 611 p. 181-192 (1976).

31. D. G. DORAN, G. R. ODETTE, R. L. SIMONS, L. K. MANSUR - "Damage Correlation in Theory and Practice." Proceedings, International Conference on Radiation Effects in Breeder Reactor Structural Materials, Scottsdale, Arizona, June 19-23 (1977).

32. W. N. McELROY, R. L. SIMONS, D. G. DORAN, G. R. ODETTE - J. Testing and Evaluation 3, 229-239 (1975).

33. C. R. WEISBIN, J. H. MARABLE, J. L. LUCIUS, E. M. OBLOW, F. R. MYNATT, R. W. PEELLE and F. G. PEREY - "Application of FORSS Sensitivity and Uncertainty Methodology to Fast Reactor Benchmark Analysis." Report ORNL/TM-5563 (1976).

34. C. R. WEISBIN and R. W. PEELLE - Proceedings, NBS Special Publication 493, 269, (1977).

35. W. N. McELROY, R. A. BENNETT, D. L. JOHNSON and N. D. DUDEY - "Neutron Environmental Characterization Requirements for Reactor Fuels and Materials Development and Surveillance Programs". Proceedings, EUR 5667 e/f Part I, 1 (1977).

36. C. M. EISENHAUER, D. M. GILLIAM, J. A. GRUNDL and V. SPIEGEL - "Utilization of NBS Standard Neutron Fields for Reactor Dosimetry." (See Reference 2)

37. A. FABRY - Proceedings, NBS Special Publication 493, 292 (1977).

38. G. AND S. DE LEEUW - "Neutron Spectrometry for Reactor Dosimetry." (See Reference 2)

39. R. GOLD - "Neutron Spectrometry for Reactor Applications: Status, Limitations and Future Directions" Proceedings, EUR 5667 e/f Part I, 119 (1977).

40. J. A. GRUNDL, C. M. EISENHAUER - Proceedings, NBS Special Publication 425, 250 (1975).

41. L. STEWART, C. M. EISENHAUER - Proceedings, NBS Special Publication 493, 198 (1977).

42. G. AND S. DE LEEUW - "Lithium-6 Neutron Spectrometry In-core" Proceedings, EUR 5667 e/f Part I, 176 (1977).

43. H. BLUHM, G. FIEG, H. WERLE - Nucl. Sci. Eng. 54, 300 (1974).

44. H. BLUM, G. FIEG, H. WERLE, H. AMES, J. P. BRAAK, R. J. S. HARRY, J. MONTIZAAN, G. DE LEEUW-GIERTS, S. DE LEEUW - "Intercomparison of Differential Neutron Spectrometry Techniques in the MOL-II Fast Assembly" Report EUR 1655-RCN 172-BLG 471 (1972).

45. G. DE LEEUW-GIERTS, S. DE LEEUW, G. E. BACKER, H. H. HELMICK - "Neutron Spectrum of BIGTEN by Lithium-6 Spectrometry" To be published.

46. G. DE LEEUW-GIERTS, S. DE LEEUW, H. H. HELMICK and J. W. ROGERS - "Neutron Spectrometry Data in LMFBR Benchmark and Standard Neutron Fields" Proceedings, EUR 5667 e/f Part I, 99 (1977).

47. Y. A. KAZANSKIJ, A. A. VAN'KOV, E. I. INYUTIN - Atom. En. Rev. 13, 807 (1975).

48. C. L. DUNFORD, M. F. VLASOV - Proceedings of a Consultant's Meeting on Nuclear Data for Reactor Neutron Dosimetry, Vienna, 10-12 September 1973, Report INDC(NDS) - 56/U, IAEA (1973).

49. C. A. OSTER, W. N. McELROY, R. L. SIMONS, E. P. LIPPINCOTT and G. R. ODETTE - "A Modified Monte-Carlo Program for SAND-II With Solution Weighting and Error Analysis" Report HEDL-TME 76-60 (1976).

50. H. ALTER, R. B. KIDMAN, R. LABAUVE, R. PROTSIK, B. A. ZOLOTAR - "Cross Section Evaluation Working Group Benchmark Specifications" Report ENDF-202 (1974).

51. D. M. GILLIAM - Proceedings, NBS Special Publication 493, 299 (1977).

52. J. A. GRUNDL, D. M. GILLIAM, N. D. DUDEY, R. J. POPEK - Nucl. Techn. 25, 237 (1975).

53. P. I. AMUNDSON et al. - Proceedings, Report ANL 7320, 679 (1966).

54. E. M. BOHN, R. MAERKER, B. A. MAGURNO, F. J. McCROSSON, R. E. SCHENTER - "Benchmark Testing of ENDF/B IV" Report ENDF-230 (1976).

55. A. PARVEZ, M. BECKER - Nucl. Sci. Eng. 62, 571 (1977).

56. M. F. VLASOV, K. OKAMOTO, H. DERRIEN, L. EDWARDSON - "Review of Differential Neutron Data for Important Reactions Not Yet Included in ENDF/B V Dosimetry File." (See Reference 2)

57. B. A. MAGURNO, Editor - "ENDF/B IV Dosimetry File" Report BNL-NCS-50446/ NEACRP-L-145/ENDF-216 (1976).

58. L. G. A. KRAMER, H. J. NOLTHENIUS, W. L. ZIJP - "Cross Section Library LAPENAS" Report RCN-76-020 (1976).

59. W. G. CROSS, H. ING - Nucl. Sci. Eng. 58, 378 (1975).

60. D. L. SMITH, J. W. MEADOWS - "Response of Several Threshold Reactions in Reference Fission Neutron Fields" Report ANL/NDM-13 (1975) and other reports in the same series.

61. W. L. ZIJP, H. J. NOLTHENIUS, N. J. C. M. VAN DER BORG - "Cross Section Library IOSCROS77 (in the Sand II Format)" Report ECN-[illegible] (1977).

62. H. FARRAR IV, E. P. LIPPINCOTT, W. N. McELROY - "Dosimetry for Fission Applications" Proceedings, EUR 5667 e/f Part I, 675 (1977).

63. H. FARRAR IV and E. P. LIPPINCOTT, "Helium Production Measurements for Fast Neutron Dosimetry and Damage Correlations." (See Reference 2)

64. H. FARRAR IV, W. N. McELROY AND E. P. LIPPINCOTT, "Helium Production Cross Section of Boron for Fast Reactor Neutron Spectra," Nucl. Techn. 25 305 (1975).

65. W. J. MALCK, "Fission Yields: Measurement Techniques and Data Status," Proceedings of the International Specialists Symposium on Neutron Standards and Applications, March 28-31, 1977, NBS Special Publication 493.

66. J. P. GENTHON, J. C. NIMAL, F. VERGNAUD - "Determination des flux de dommage, detecteur à effet physique et calcul Monte Carlo." Proceedings, EUR 5667 e/f Part I, 778 (1977).

67. H. MORET, J. GRULMANS, Y. DROULERS, P. MAS, A. FABRY, J. P. DEWORM - "Fast Neutron Dosimetry with Quartz Detectors" Annales du Colloque sur les procédés de mesures de flux associés aux dispositifs d'irradiation dans les réacteurs de recherche et d'essais de matériaux, Société Européenne de l'Energie Atomique, GRENOBLE, 15-17 Novembre 1965.

68. A. VAN DEN BOSCH, C.E.N.-S.C.K. (Solid State Physics), private communication (1977).

69. F. G. PEREY - "Least-Squares Dosimetry Unfolding: The Program STAY'SL" Report ORNL-TM-6062/ENDF-254 (1977).

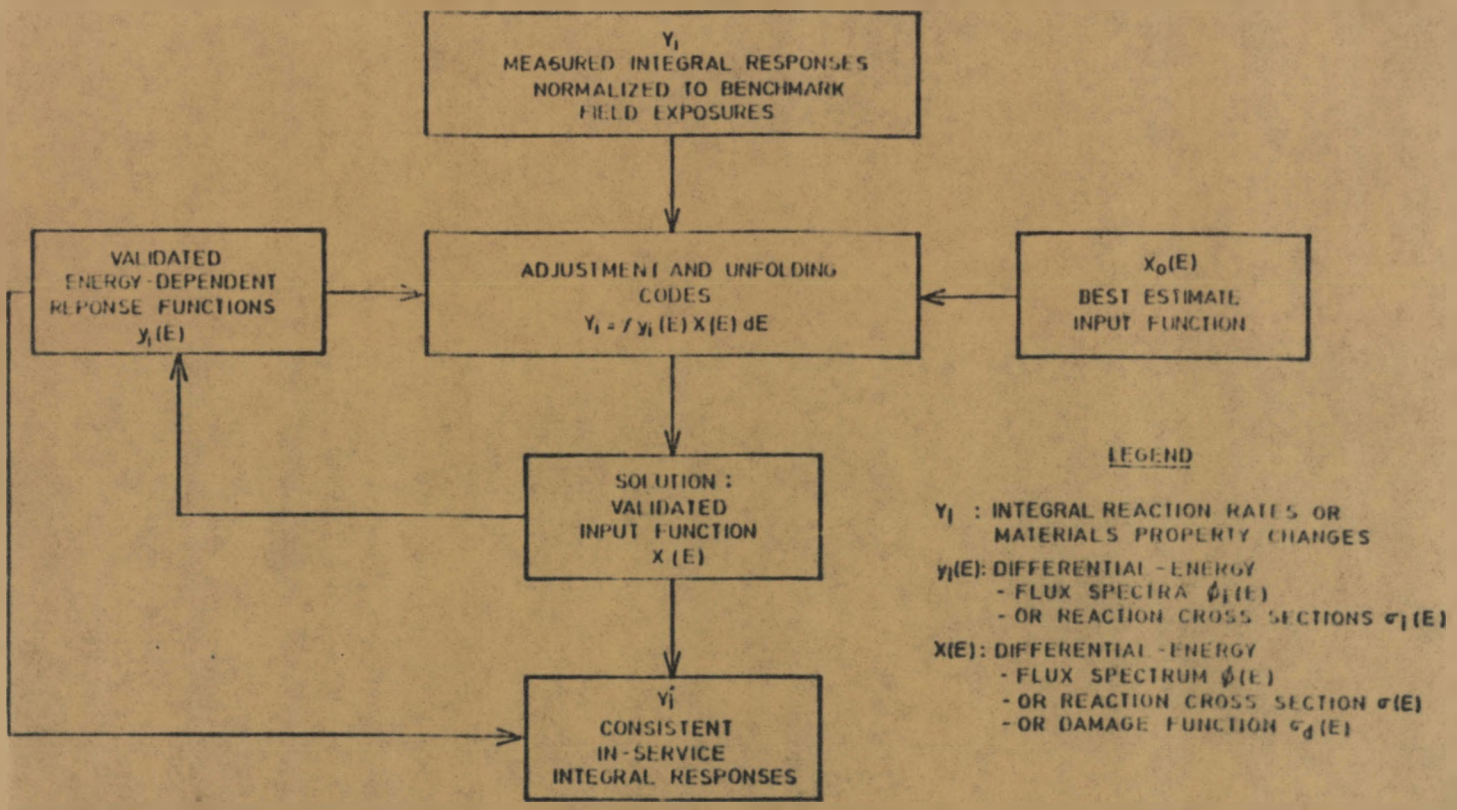
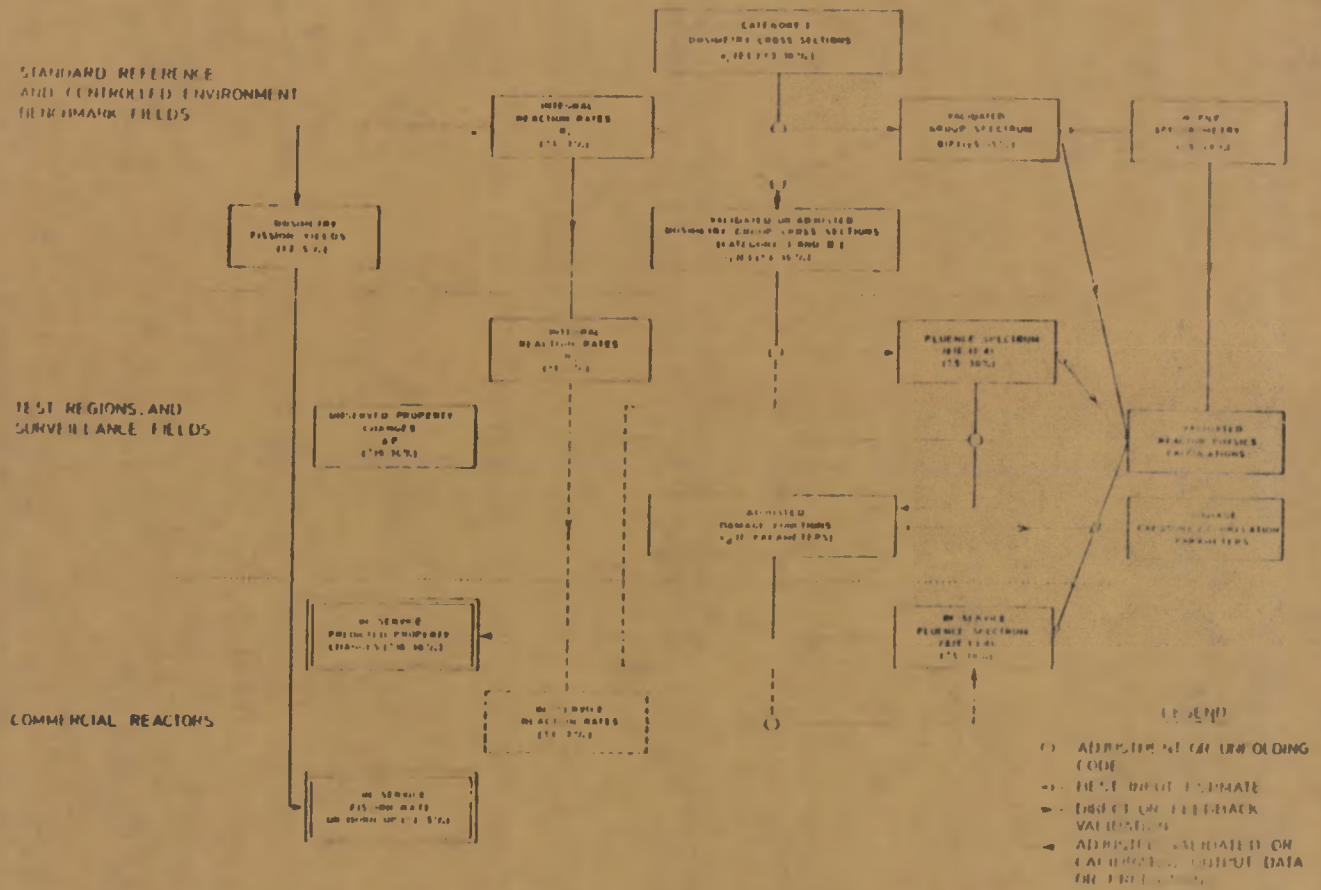

Y_i
MEASURED INTEGRAL RESPONSES
NORMALIZED TO BENCHMARK
FIELD EXPOSURES
VALIDATED
ENERGY-DEPENDENT
REPONSE FUNCTIONS
$y_i(E)$
ADJUSTMENT AND UNFOLDING
CODES
$Y_i = \int y_i(E)\, X(E)\, dE$
$X_0(E)$
BEST ESTIMATE
INPUT FUNCTION
SOLUTION :
VALIDATED
INPUT FUNCTION
X (E)
Y_i'
CONSISTENT
IN-SERVICE
INTEGRAL RESPONSES
LEGEND
Y_i : INTEGRAL REACTION RATES OR
MATERIALS PROPERTY CHANGES
$y_i(E)$: DIFFERENTIAL - ENERGY
- FLUX SPECTRA $\phi_i(E)$
- OR REACTION CROSS SECTIONS $\sigma_i(E)$
X(E): DIFFERENTIAL - ENERGY
- FLUX SPECTRUM $\phi(E)$
- OR REACTION CROSS SECTION $\sigma(E)$
- OR DAMAGE FUNCTION $\sigma_d(E)$

FIGURE 1. Flow Chart - Neutron Environment and Damage Exposre Correlation Parameter Determination.


STANDARD REFERENCE
AND CONTROLLED ENVIRONMENT
BENCHMARK FIELDS
TEST REGIONS AND
SURVEILLANCE FIELDS
COMMERCIAL REACTORS
LEGEND

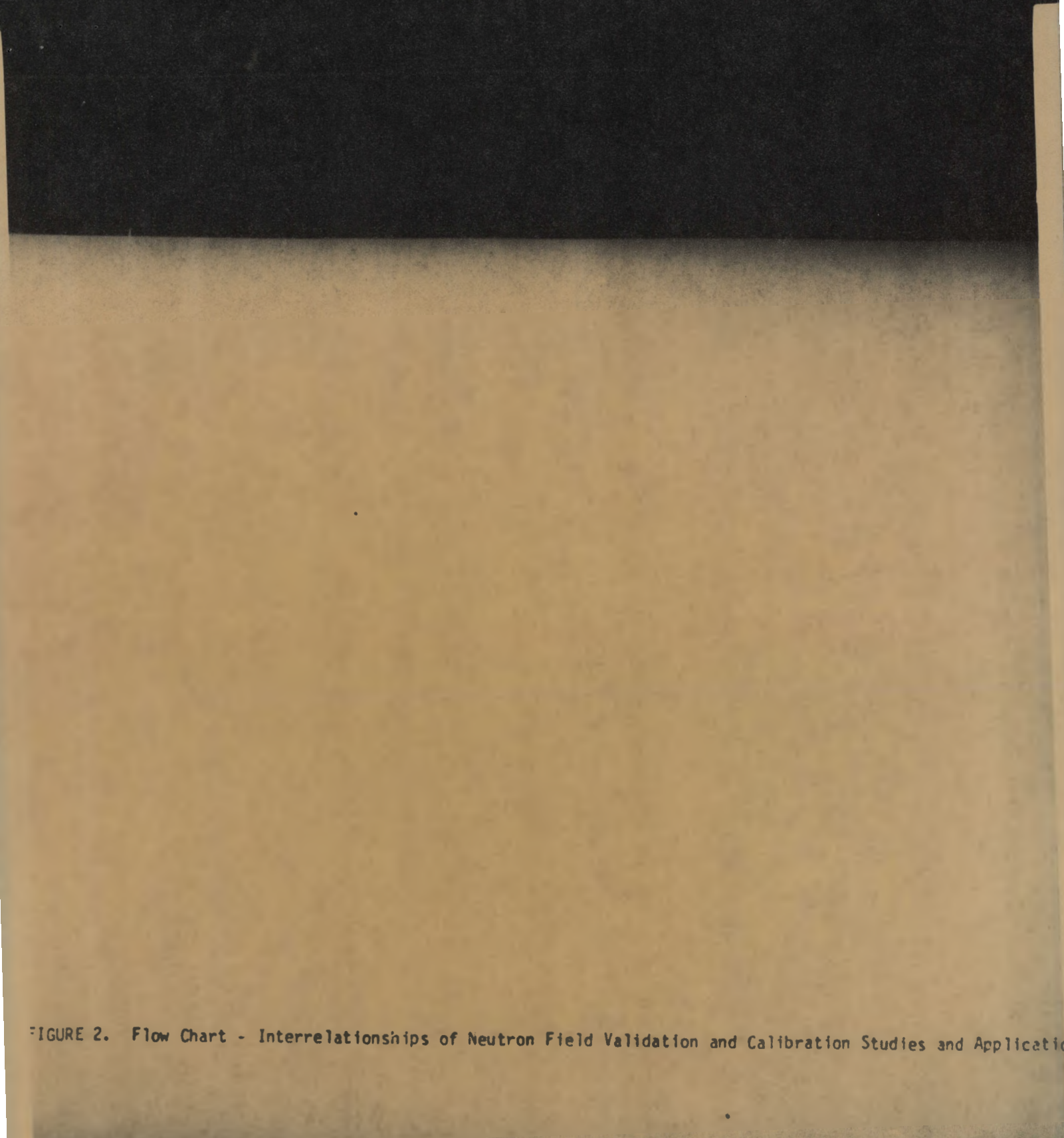


FIGURE 2. Flow Chart - Interrelationships of Neutron Field Validation and Calibration Studies and Applicati

TABLE I. CLASSIFICATION OF REACTOR DOSIMETRY BENCHMARK RADIATION FIELDS

STANDARD:	• PERMANENT, STABLE, REPRODUCIBLE
	• WELL-SPECIFIED PHYSICAL SURROUNDINGS
	• STATE-OF-THE ART ACCURACY CHARACTERIZATION OF FLUX AS FUNCTION OF ENERGY, SPACE AND ANGLE
	• INTERLABORATORY MEASUREMENT OF IMPORTANT FIELD QUANTITIES
REFERENCE:	• PERMANENT, REPRODUCIBLE
	• KNOWN PHYSICAL SURROUNDINGS
	• REASONABLY GOOD CHARACTERIZATION
	• ACCEPTED AS REFERENCE BY A COMMUNITY OF USERS
CONTROLLED ENVIRONMENT:	• KNOWN PHYSICAL SURROUNDINGS
	• EMPLOYED FOR RESTRICTED SETS OF WELL-DEFINED EXPERIMENTS
TEST REGION:	• EMPLOYED TO INVESTIGATE DOSIMETRY MEASUREMENT PROBLEMS AND RELATED CORRELATIONS OF MATERIALS RADIATION DAMAGE

TABLE II. LWR AND FBR DOSIMETRY BENCHMARK NEUTRON FIELDS[a]

AREA OF APPLICATION	TYPE OF BENCHMARK NEUTRON FIELD: STANDARD	REFERENCE	CONTROLLED ENVIRONMENT	TEST REGION
VALIDATE/CALIBRATE SENSORS	FISSION SPECTRA[7,8,9] 252Cf(10) 1/E(235U/CV(10)) THERMAL	[illegible](11) CFRM(12)	LWR: ORR PSF[2](b)	LWR: BR1 (PWR)[4] UVa[d] CARIGLIANO (BWR)
			FBR: FFTF/IRT(19)(c)	FBR
VALIDATE/ADAPT SENSOR NUCLEAR DATA AND SPECTRUM UNFOLDING	ALL STANDARD FIELDS	ALL REFERENCE FIELDS	ALL (TRENDS ONLY)	
VALIDATE DAMAGE EXPOSURE/CORRELATION PARAMETERS			LWR: ORR PSF[2](b)	LWR: PCA[c] UVa[d] BR1[4], CARIGLIANO
			FBR:	FBR: Selected locations in BARSEBÄCK, FBR II, FFTF, EBR II, BR2[f], KNK II
VALIDATE REACTOR PHYSICS DATA AND CODES FOR DOSIMETRY APPLICATIONS		LWR: MOL IRON SHIELDS(g)(13) RPI IRON BLOCK (h)(14)	LWR: PCA[2](b) ORR PSF[2](b)	LWR: MOL(PWR)[4] [illegible] FERRY(BWR)[4] [illegible] CARIGLIANO YAYOI(FBR)
		FBR: GODIVA(15)(i) JEZEBEL(15)(j) BIGTEN(16)(k) [illegible](11)	FBR: ECIL (ORE 16(17)(m) ERR(n) TAPIRO(18)(o) FFTF/IRT(19)(c) YAYOI(20)(p)	FBR: Selected locations in FFTF, EBR II, BARSEBÄCK, FBR

(a) Fields applicable to both the LWR and FBR programs, unless otherwise indicated by labels in the table.
More complete, though less differentiated list of benchmark fields (including fusion reactor type ones) may be found in reference (21). The NBS compendium, reference (22), provides summary descriptions and all currently available data for many of the standard, reference-type controlled-environment fields, in particular fission spectra, [illegible], CFRMF.

(b) Oak Ridge Reactor (ORR) Pool Side Facility USNRC high flux pressure vessel mock up, to be operational in 1979.

(c) Fast Flux Test Facility In-Reactor Thimble: a reentrant tube through FFTF core center in which extensive passive and active neutron spectrometry and dosimetry characterization measurements will be performed in Fall 1979.

(d) University of Virginia swimming pool reactor used for the EPRI (Electric Power Research Institute) dosimetry-metallurgy program.

(e) Bulk Shielding Reactor (swimming pool) at Oak Ridge used for the EPRI program and for the USNRC Heavy Section Steel Technology (HSST) program.

(f) Cadmium-screened sodium loops in the high flux materials testing reactor BR2 at Mol, Belgium, used in particular for LMFBR fuel and clad experiments. A zero power engineering mock up, BR02, is available for spectral characterization and dosimetry calibration (23).

(g) Spherical assemblies of iron shells inside of natural uranium driver shells in the graphite thermal column of the BR1 reactor at Mol, Belgium.

(h) Spectral characterization by time-of-flight.

(i) Los Alamos bare critical sphere of uranium-235 metal.

(j) Los Alamos bare critical sphere of plutonium-239 metal.

(k) Los Alamos natural uranium reflected critical cylinder with homogenous core central zone of 10% enriched uranium.

(l) Oak Ridge, Pool Critical Assembly USNRC low flux pressure vessel mock up, to be operational by mid 1978.

(m) Atomics International thermal-fast coupled low power facility, with fast zone central flux spectrum similar to FFTF. Now dismantled. Integral data in ECIL core 16 are used for international comparisons of spectral unfolding methods.

(n) Engineering Mock-up Critical, zero power mock up of FFTF at Argonne National Laboratory, Idaho.

(o) Fast source critical reactor, Italy; uranium-235 cylindrical core surrounded by a copper reflector.

(p) Fast source critical reactor, Japan, similar to Tapiro, but with a lead reflector.

TABLE III. RATIO OF MEASURED TO COMPUTED INTEGRAL CROSS SECTIONS IN DOSIMETRY BENCHMARK NEUTRON FIELDS (FERMI AND BIG-

REACTION	FERMI			BIG-II		
$^{59}Co(n,\gamma)^{60}Co$	1.074	1.123	1.098	1.021	1.047	1.033
$^{54}Fe(n,p)^{54}Fe$	1.001	0.989	1.000	1.357	1.362	1.364
$^{63}Cu(n,\gamma)^{64}Cu$	0.946	0.946	0.937	0.932	0.914	0.915
$^{197}Au(n,\gamma)^{198}Au$	1.019	1.014	1.000	1.041	1.036	1.019
$^{238}U(n,\gamma)^{239}U$	0.961	0.946	0.956	1.002	1.000	0.980
$^{10}B(n,\alpha)^{7}Li$	1.071	1.068	1.053	1.141	1.131	1.108
$^{45}Sc(n,\gamma)^{46}Sc$	1.166	1.175	1.158	1.125	1.114	1.102
$^{115}In(n,\gamma)^{116m}In$	0.939	0.932	0.939	-	-	-
$^{6}Li(n,\alpha)^{3}H$	0.959	0.977	0.949	1.001	1.015	0.980
$^{235}U(n,f)$	0.979	0.976	0.974	0.995	0.990	0.987
$^{239}Pu(n,f)$	1.016	1.001	1.003	1.017	1.003	1.017
$^{237}Np(n,f)$	1.007	0.951	0.999	0.985	0.944	1.015
$^{115}In(n,n')^{115m}In$	1.074	0.977	1.007	1.092	0.995	1.034
$^{238}U(n,f)$	1.089	0.983	1.004	1.107	0.979	1.018
$^{47}Ti(n,p)^{47}Sc$	0.889	0.815	0.811	0.949	0.891	0.848
$^{58}Ni(n,p)^{58}Co$	1.043	1.017	0.999	1.089	1.058	1.027
$^{54}Fe(n,p)^{54}Mn$	1.061	0.963	0.967	1.120	1.028	0.975
$Ti(n,x)^{46}Sc$	1.214	1.145	1.125	1.290	1.207	1.129
$^{27}Al(n,p)^{27}Mg$	0.965	0.929	0.918	-	-	-
$^{27}Al(n,\alpha)^{24}Na$	0.994	0.910	0.999	1.078	1.029	1.014
$^{48}Ti(n,p)^{48}Sc$	1.787	1.578	1.686	1.993	1.980	1.859

(a) Normalized by $^{239}Pu(n,f)$ transfer from californium.

(b) $\int_0^\infty \sigma_r(E)\phi(E)dE \Big/ \int_0^\infty \sigma_r(E)\phi_{ENDF/B\text{-}IV}(E)dE$, $\phi_{ENDF/B\text{-}IV}(E)$ as indicated in text, section 3.

(c) Using as input $\phi(E)$ from ENDF B-III computation and the six category I reactions underlined in Table I, see references 21 and 28.

TABLE IV. MEASURED TO COMPUTED (ENDF/B IV) FISSION CROSS SECTION RATIOS IN GEOMETRY BENCHMARK NEUTRON FIELDS[a]

FIELD	MATERIAL NOMINAL NEUTRON SPECTRUM	$^{238}U(n,f)/^{235}U(n,f)$ ABSOLUTE	$^{238}U(n,f)/^{235}U(n,f)$ NORMALIZED TO $^{252}Cf/^{235}U$	$^{239}Pu(n,f)/^{235}U(n,f)$ ABSOLUTE	$^{239}Pu(n,f)/^{235}U(n,f)$ ABSOLUTE	$^{237}Np(n,f)/^{235}U(n,f)$ ABSOLUTE
^{252}Cf[b]	PURE SOURCE	0.99[d]	1.01	0.99	1.04	0.93
^{235}U[c]	PURE SOURCE	0.94[d]	1	0.98	1.00	0.96
JEZEBEL	^{239}Pu	0.96	0.98	0.98	1.02	0.95
GODIVA	^{235}U	1.03	1.10	1.06	1.01	0.96
FLATTOP	^{238}U	1.01	1.07	1.01	1.03	0.96
	-	(0.95)[e]	(1.01)[e]	(0.97)[e]	1.0[x]	(0.93)[e]
ORSPHERE	-	0.99	1.05	1.0	1.0[x]	0.97
LANL	$^{10}B(n,\alpha)^{7}Li$	0.91	0.97	0.95	1.01	

(a) Some of these fields are also considered as reactor physics benchmarks. The differential energy cross sections used for the computation are ENDF/B IV and the assumed spectral shapes are the most recently evaluated ones, see text.

(b) ^{252}Cf spontaneous fission neutron spectrum

(c) Thermal neutron induced ^{235}U fission neutron spectrum

(d) $\pm$ 2.5%

(e) Evaluated spectral shape known to need updating.

TABLE V SUMMARY STATUS OF THE KNOWLEDGE OF DOSIMETRY CROSS SECTIONS FOR RADIOMETRIC PASSIVE SENSORS

ACCURACY LEVEL	CATEGORY I REACTION(s)	CATEGORY II REACTION(s)	REACTIONS NOT IN ENDF/B IV FILE(b)
CONSISTENT TO ± 5 % OR BETTER	$^{197}Au(n,\gamma)^{198m}Au$ $^{239}Pu(n,f)$ $^{237}Np(n,f)$ $^{238}U(n,f)$ $^{58}Ni(n,p)^{58}Co$ $^{56}Fe(n,p)^{56}Mn$ $^{27}Al(n,\alpha)^{24}Na$	$^{59}Co(n,\gamma)^{60}Co$ $^{235}U(n,f)^{237}U$ $^{10}B(n,\alpha)$ $^{115}In(n,n')^{115m}In$ $^{32}S(n,p)^{32}P$ $^{63}Cu(n,\alpha)^{60}Co$ $^{55}Mn(n,\gamma)^{56}Mn$	$^{103}Rh(n,n')^{103m}Rh$ $^{31}P(n,p)^{31}Si$ $^{24}Mg(n,p)^{24}Na$
INCONSISTENT BUT AREAS OF DEFICIENCY IDENTIFIED(c)		$^{47}Ti(n,p)^{47}Sc$ $Ti(n,x)^{46}Sc$ $^{48}Ti(n,p)^{48}Sc$ $^{61}Ni(n,\alpha)^{58}Fe$	
INCONSISTENT (MORE WORK NEEDED)	$^{58}Ni(n,2n)^{57}Ni$(d)	$^{54}Fe(n,\alpha)^{51}Cr$ $^{61}Ni(n,\gamma)^{62}Ni$ $^{10}B(n,\alpha)^{7}Li$ $^{45}Sc(n,\gamma)^{46m}Sc$ $^{115}In(n,\gamma)^{116m}In$ $^{6}Li(n,\alpha)^{3}H$ $^{238}U(n,f)$ $^{60}Ni(n,p)^{60}Co$ All (n,2n) REACTIONS	$^{109}Ag(n,\gamma)^{110m}Ag$ $^{58}Fe(n,\gamma)^{59}Fe$ $^{93}Nb(n,n')^{93m}Nb$ $^{181}Ta(n,\gamma)^{182}Ta$ $^{93}Nb(n,n')^{93m}Nb$ $^{64}Zn(n,p)^{64}Cu$ $^{55}Mn(n,\gamma)^{56}Mn$ $^{61}Ni(n,2n)^{60m}Co$(d)(e) Other (n,2n) reactions

(a) Currently on the ENDF/B IV file

(b) Requested for ENDF/B V

(c) Expected to be consistent on ENDF/B V

(d) Uncertainties believed to be less than ± 10 %

(e) Category I reaction