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Institut für Angewandte Reaktorphysik
Institut für Neutronenphysik und Reaktortechnik
Projekt Schneller Brüter

Theoretical and Experimental Analysis of Fast Zero Power Assemblies with Special Consideration to Nuclear Data Check

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GESELLSCHAFT FÜR KERNFORSCHUNG M.B.H.

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Zusammenfassung

In diesem Bericht wird ein Überblick über die in Karlsruhe verwendeten theoretischen und experimentellen Methoden zur Analyse kritischer Anordnungen gegeben. Insbesondere wird die Möglichkeit, mikroskopische Wirkungsquerschnitte durch integrale Experimente zu überprüfen, in Betracht gezogen.

Der Stand der Unsicherheiten mikroskopischer Daten wird in Abschnitt 1 kurz dargestellt. Die Prinzipien und Probleme der Auswertung dieser Daten werden diskutiert. Im Abschnitt 2 werden die wichtigsten integralen Kerndaten für den Entwurf eines schnellen Reaktors beschrieben und die Möglichkeit, diese Vorhersagen in Null-Leistungsanlagen zu überprüfen, dargelegt. Die angewendeten theoretischen Methoden zur Berechnung schneller Leistungsreaktoren und die Analyse schneller Null-Leistungsanordnungen werden im Abschnitt 3 beschrieben. Die experimentellen Methoden und die Genauigkeit der gemessenen integralen Daten werden in Abschnitt 4 diskutiert. Im Abschnitt 5 wird untersucht, ob diese Experimente zur Überprüfung von Kerndaten benutzt werden können. Im abschliessenden Abschnitt wird gezeigt, wie die Information kritischer Experimente zur Unterstützung der Auswertung mikroskopischer Wirkungsquerschnitte und Verbesserung der physikalischen Reaktorentwürfe benutzt werden kann.

Abstract

In this paper a review is given about the theoretical and experimental methods used at Karlsruhe to analyse critical assemblies. Special consideration is given to the possibility of checking microscopic data by integral experiments.

The situation of microscopic data uncertainties is briefly illustrated in chapter 1. Principles and problems in evaluating these data for the use in reactor calculations are discussed. In chapter 2 the main integral nuclear data to be predicted in the design of a fast reactor are stated and the principle possibility to check these predictions in zero power facilities is outlined. In chapter 3 the theoretical methods used to calculate fast power reactors and to analyse fast zero power assemblies are described. The experimental methods and the accuracy of the measured integral data are discussed in chapter 4. In chapter 5 it is analysed whether these experiments can be used to check nuclear data. The concluding chapter shows how the information from critical can be used to support the evaluation of microscopic data and to improve the physics design of fast power reactors.

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Theoretical and Experimental Analysis of Fast Zero Power Assemblies with Special Consideration to Nuclear Data Check

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INTRODUCTION

In the days of fast breeders of the first generation (EBRII, FERMI-REACTOR, DOUNRAY FAST REACTOR) critical assemblies were mostly designed as mockups. In the past few years they have taken on a different role. The large size of breeder reactors currently under design, and the lack of inventory of materials to simulate the power reactor in a critical assembly have led to the approach that critical assemblies are built mainly with the purpose to check nuclear data and calculational methods. Extrapolation to the power

reactor is then done by analysis.

D. Okrent presented in the 1965 Conference on Safety, Fuels and Core Design in Large Fast Power Reactors at Argonne [1] the results of an international inter-comparison on theoretical predictions of integral neutronic properties of some specified fast systems. It became evident that fast reactors could not be calculated reliably. In 1966 another inter-comparison on ZPRIII-48 [2] showed that the analysis of fast critical assemblies was far from being in a satisfactory state. It was assumed that the discrepancies were both due to the uncertainties in the microscopic data, and to inadequate theoretical methods. In the last years it was possible to improve the theoretical methods considerably and at the 1969 BNES Conference on the Physics of Fast Reactors [3] in London it was generally agreed that the remaining differences between theoretical and experimental results are mainly caused by incorrect nuclear data.

In this paper a review is given about the theoretical and experimental methods used at Karlsruhe to analyse critical assemblies. Special consideration is given to the possibility of checking microscopic data by integral experiments.

The situation of microscopic data uncertainties is briefly illustrated in chapter 1. Principles and problems in evaluating these data for the use in reactor calculations are discussed. In chapter 2 the main integral nuclear data to be predicted in the design of a fast reactor are stated and the principle possibility to check these predictions in zero power facilities is outlined. In chapter 3 the theoretical methods used to calculate fast power reactors and to analyse fast zero power assemblies are described. The experimental methods and the accuracy of the measured integral data are discussed in chapter 4. In chapter 5 it is analysed whether these experiments can be used to check nuclear data. The concluding chapter shows how the information from critical experiments can be used to support the evaluation of microscopic data and to improve the physics design of fast power reactors.

1. PRINCIPLES AND PROBLEMS IN MICROSCOPIC DATA EVALUATION

The steady improvement and refinement of reactor theory programs in the last years has led to a status where the reliability of theoretical predictions of the physical properties of fast reactors depends to the largest extent upon the detailed and reliable knowledge of the microscopic nuclear data involved. The dominating influence of the heavy nuclei on the physical behaviour of fast reactors has provoked high accuracy requests for the nuclear data of these materials so that the experimental methods for neutron cross section measurements had to be more and more refined. Because of the large amount of data produced the task of the evaluator to derive complete sets of so-called "best" data by taking into account all available experimental information and by judging its reliability has become more laborious and complicated.

1.1. Consistent experimental results

In the ideal case different measurements come into agreement after reduction to the same experimental conditions, at least within the range of their mutual uncertainties. Then these data sets can simply be averaged by least squares and other adjustment procedures in order to elaborate recommended values.

Problems can arise here because of gaps in the available experimental information either due to the limitedness of the experimental facilities or due to the fact that not all data have been determined experimentally because until now they might have not been urgently requested. For a cross section smoothly dependent on energy the gaps can rather reliably be closed by numerical or graphical inter- or extrapolation using theoretical models and empirical or semi-empirical systematics.

In most cases nuclear theory can only then be applied successfully for interpolation if at least some of the theoretical parameters are determined from experiment. The reliability of the nuclear models is restricted to a certain energy range, reaction and nucleus. For example outside the resolved resonance region the optical model allows the prediction of total cross sections without differentiation for the various exit channels like fission, radiative capture, etc. The various reaction cross sections for

elastic and inelastic scattering, fission, radiative capture, etc. can be predicted here by using a statistical theory for the decay of the compound nucleus. The evaporation model can serve for the completion of the data available for the energy distribution of inelastically scattered neutrons in the continuum range whereas the discrete-level inelastic model has to be used in the range of resolved excitation levels of the target nucleus. For the prediction and interpolation of cross sections in the resonance region single-level and multi-level formulae are available. The necessary statistical parameters for the unresolved region can be derived from available resolved resonance data or, concerning the strength functions, from optical model calculations. The Fermi gas model e.g. predicts here the spin and energy dependence of the level densities, the Hill-Wheeler formula the energy dependence of the fission widths.

Gaps in the resolved resonance range can principally not be closed in a similar way because no theory exists at the moment which is able to predict the position and properties of resonances. This shows clearly the limited scope of the various nuclear models and illustrates how problematic the closing of gaps can become even in the case of consistent experimental results because no unified nuclear theory exists. A more extensive survey about the possibilities for interpolation by nuclear theory is given in reference [4].

1.2. Discrepant experimental results

Instead of agreement between different data sets, however, one encounters more often deviations between the results by an amount larger than the uncertainties of the individual measurements, in spite of the corrections already applied. These discrepancies and inconsistencies represent the main problem in almost every evaluation regardless of neutron energy, reaction type and nucleus concerned. In the simplest case the differences are due to normalization to different standard values. In other cases, however, their sources cannot so easily be detected, as they are very closely connected with the experimental facility and methods used.

An example for an encountered discrepancy due to different measuring techniques represent the experiments for the mean number of neutrons released in the spontaneous fission of ^{252}Cf. This value is the most important standard for $\bar{\nu}$-measurements for all fissile and fertile materials. Neutron detection

with liquid scintillators yield higher values for $\bar{\nu}_{sp}(^{252}Cf)$ than the boron pile and the manganese sulphate bath measurements. The differences are larger than the uncertainties quoted for the individual measurements. This discrepancy may be partly attributed to the loss of fast neutrons due to other nuclear reactions in the manganese sulphate bath, but the situation is till now still rather unsatisfactory.

Another well-known example for strong not yet resolved systematic discrepancies between different measurement series represent the capture cross section measurements for ^{238}U. For illustration different measurements at 30 keV are quoted in Table I, together with the deviations relative to the experiment of de Saussure (arbitrarily chosen).

Since some of these measurements have also been repeated with great care yielding the same results it seems that the persistent deviations are, at least partly, due to different experimental methods used. Each laboratory relies with greatest confidence on ist own special detector although systematic errors might obviously be caused by it.

Such unresolved discrepancies have the consequence that the evaluated data will have uncertainties larger than the accuracies quoted for the individual experimental data in regard. The evaluator is forced to come to a decision concerning the reliability of the different discrepant experimental data sets in order to derive "best" values. Empirical and semi-empirical nuclear systematics are restricted in their reliability. Nuclear theory does not give necessarily an univocal decision because it may be that the individual series can all be reproduced by nuclear theory using different sets of nuclear parameters or different models of which some of them are quoted in the preceding section.

Thus in the end the evaluators subjective judgement based on the experience and the physical understanding plays an essential role in the evaluation procedures. Important aspects in this context are the reputation of the experimentalist and of the laboratory at which the measurements have been performed.

From here it is obvious that the evaluations carried out by evaluators in different countries and scientific institutions may yield different results. For demonstration a comparison of three evaluated data files is given in Fig. 1 for the capture cross section of ^{238}U and in Fig. 2 for the fission

cross section of ^{239}Pu. We compared the American ENDF/B-file, the English UKAEA Nuclear Data Library and the German KEDAK-file in the versions available in April 1970 at CCDN in Saclay.

1.3. Possible ways to reduce data discrepancies

The present situation with so many unresolved discrepancies in the important experimental nuclear data information is rather unsatisfactory. Naturally the steady improvement of the experimental methods leads gradually to a reduction of the discrepancies. But one has to pay attention to the fact that new precision measurements do not clarify the situation in any case. That is e.g. the case for the spontaneous fission $\bar{\nu}$-measurements for ^{252}Cf, where recent measurements have not succeeded in resolving the discrepancies but only in increasing the number of discrepant measurements. In some cases certainly more detailed considerations of the experimental conditions followed by a thorough re-evaluation could help. This procedure will be much facilitated in the near future since the compilation centers have started to compile together with the experimental data sets also physical and technical comments characterizing the measurement. These comments will be structured according to the internationally agreed exchange format. The final discussion on it has taken place on the last Four-Centre-Meeting in November 1969. In a standardized form information is given about the experimental facility, the experimental method, the properties of the sample, the detector, the standard used, the data analysis, the corrections applied, the error analysis and others. These compiled schematic descriptions of the experimental conditions of a measurement help the experimentalist to survey quickly what kind of information is required by the evaluator for a comparison of his measurements with others and for the judgement of it.

In the case of large discrepancies the experimentalists themselves should meet to compare their experimental results and to re-consider the experimental conditions of their measurements in all details because they know best the difficulties in their measurements and the possible source of errors in them. This procedure is considered to be highly efficient in detecting the deficiencies of the various experiments and their analysis. Thus improved techniques for measurement and analysis can be applied, a search for new independent

experimental methods may be stimulated, the experimental equipment, or at least parts of it, may be exchanged and so all of these efforts should lead to more precise and consistent results. A close cooperation with the evaluators guarantees that the reactor physicists as users of these data can rely on successfully elaborated "best" data sets. An international organization such as the IAEA could have the important function to coordinate these activities.

The procedure of data evaluation and the ways to reduce the data discrepancies are given in Fig. 5 and 6.

2. THE INTEGRAL NUCLEAR DATA OF INTEREST IN FAST REACTORS

The neutronic properties to be predicted for the design of a fast reactor are:

a) the fuel inventory,
b) the power distribution,
c) the conversion of fertile to fissile material during operation together with the build-up of fission products,
d) the behaviour of the system as a consequence of perturbations of the normal operating reactor.

The theoretical prediction is supported by information from critical experiments, but one must realize that the latter cannot deal directly with the transient behaviour, or with long-term effects. The check of methods and data by critical experiments is mainly devoted to the determination of critical mass, reaction rate traverses and reaction rate ratios, reactivity coefficients, reactivity worths of the materials or isotopes in question and also the reactivity effect of higher Pu-isotopes and fission products.

If the experimental investigation is aimed to clarify nuclear cross section uncertainties, the theoretical methods used to analyse integral measurements as well as the experimental accuracy have to have a high precision, so that discrepancies between theoretical and experimental results can be attributed to cross section uncertainties in a unique way.

3. THEORETICAL METHODS IN FAST REACTOR CALCULATIONS

In fast power reactors now under design the neutron mean free path over a wide energy range is larger than the pitch of the core lattice. Therfore each core zone can be treated as homogeneous for the calculation. In most cases the diffusion approximation of Boltzmann's transport equation is sufficient to calculate the neutronic properties of the system. Higher order transport approximations are normally used only to determine corresponding corrections. Even control rod calculations can be performed by diffusion theory with tolerable accuracy. The main effort in the calculation of fast reactors is therfore spent not so much to obtain the spatial distribution of neutrons, but rather the distribution in energy. With respect to the nuclear quantities of interest, mentioned in chapter 2, the energy range from some MeV down to some hundred eV is equally important.

The analysis of experiments in fast critical or subcritical assemblies requires definitely more sophisticated theoretical methods also for the space and angular distributions of neutrons, especially if it is desired to check the accuracy of nuclear data.

3.1. The multigroup procedure

Because no analytical solutions to the transport or diffusion equation exist for cases of interest, the neutron distribution has to be obtained numerically by discretisation in all variables. With respect to energy this means integration of the balance equation over a certain energy interval, the energy group. In order to preserve the reaction rates in one energy group, the cross sections are averaged over the energy group with the true neutron flux density as a weighting function. This procedure implies two main problems:

a) By definition the group cross section is not a constant because the weighting function depends on position and angle.

b) A unique set of group cross sections for reactor calculations does not exist because diffusion theory, S_N, P_N or collision probability methods each requires a different system of weighting functions.

To overcome the first difficulty, one usually assumes a separation of the neutron distribution in energy and the other variables, so that the group cross sections become independent of space and angle. To account for the space variation of the neutron spectrum in homogeneous zones, these may be subdivided into several regions with different group constants. If the groups are broad, it is important to use the proper weighting spectrum, which is not easy to determine. In principle it can be obtained from an iterative procedure.

In the resonance region of the cross sections the neutron spectrum varies strongly, across one resonance proportional to the inverse of the total macroscopic cross section (narrow resonance(NR)-approximation). Therefore group constants of one isotope depend on the material composition of a region. The resonance character of the cross section also introduces the temperature as a variable in the group constants.

At Karlsruhe we have adopted the scheme, developed at Obninsk [5], of splitting the effective group constant into

$$\sigma_{eff}^{(i)} = \sigma_{\infty}^{i} \cdot f^{i} \text{ (composition, temperature)},$$

where σ_{∞}^{i} is the group constant for infinite dilution of the isotope in question. This splitting is especially advantageously in the unresolved resonance range, because f^{i} is not too sensitive to uncertainties in the statistical resonance data and σ_{∞}^{i} can be calculated from measured values across the energy group. The determination of the resonance self shielding factors f^{i} is based on the single level Breit-Wigner formula. Interference of potential and resonance scattering and resonance overlap is taken into account. The interaction between resonances of different isotopes is neglected by definition. This is not a severe limitation in the energy range of interest in fast reactors.

Two different procedures are currently in use at Karlsruhe to account for the dependence of the resonance self shielding on the composition of a reactor region.

a) The total cross sections of all isotopes except the one whose resonance self shielding is being calculated, are approximated by an appropriate value, called σ_{o}, within the group. Usually

one takes the total cross sections for infinite dilution. We call this the σ_o-concept.

b) Cross sections for elastic moderation of neutrons by nuclei of structural and coolant materials are not approximated by the σ_o-concept, but are calculated exactly within the NR-approximation. The numerical procedure uses about 1000 energy points. In the case of anisotropic scattering in the CMS experimental angular distributions are directly used. This procedure is called REMO (from elastic removal) [6]. It should be noted that reactor calculations are performed only in the broad group scheme.

This limitation is just being eliminated by 200 group fundamental mode and space dependent consistent P_1-calculations. Here again the macroscopic elastic removal constants are determined exactly, using as a basis about 1000 groups. A full documentation of the procedure used and the calculated results will be published soon.

3.2. Number of energy groups

All design calculations of fast reactors are based at Karlsruhe on standarized 26-group sets [6], according to the scheme first introduced by Abagjan et al. [5]. The energetic fine structure due to resonances within a group is treated in NR approximation so that the collision density is only weakly dependent on composition and energy. However, the group width is too broad that a standard collision density weighting function could be used for all types of fast reactors, mainly because of a miscalculation of the neutron slowing down. In consequence we use different weighting spectra for most of the reactors calculated, especially if the REMO procedure is applied.

The desired target is to perform reactor calculations with data sets which do not include any prescribed weighting function. Then both the main problems, as stated in section 3.1, in establishing group constants are avoided. This is possible for example if the group width is so small that simple energetic averages of the cross sections in the fine groups can be used. This condition leads to more than 10^5 groups, caused by the resonance

structure of the heavy nuclei (e.g. ^{238}U). It is obvious that the application of reactor codes with such high energy resolution is restricted to very special investigations (GENEX [7], RABBLE [8]).

A condition, which can much more easily be fulfilled, is to request a constant collision density within a group. This leads to a group number of some hundred, which is mainly related to the resonance structure of medium weight nuclei as coolant and structural material and, in case of a reactor with plutonium oxide, also the energy dependence of the oxygen cross sections. This procedure requires the calculation of resonance self shielding within the groups, especially for the heavy nuclei, for instance according to the scheme described above, but not using the σ_o-concept for the description of neutron down scattering. To avoid immense tabulations on the composition dependent resonance self shielding factors for the heavy nuclei, an appropriate interpolation formula must be used. This group scheme can also satisfactorily deal with threshold type cross sections as $\sigma_f(^{238}U)$.

These some hundred group calculations originated by Hummel and Rago [9] in fundamental mode calculations can now easily be performed for one-dimensional problems on modern computers. The resulting spectra can then be used to condense the group constants regionwise in order to perform coarse group calculations with multi-dimensional diffusion or transport codes.

Though the effect of a scattering resonance on the neutron spectrum in a homogeneous fast reactor medium is restricted to energies around the resonance itself, the necessity of a proper treatment of these resonances stems from the fact that they are relatively broad and sometimes overlapping for structural materials (the main resonances cover a complete group in the 26-group scheme) and thus influence the absorption rates of neutrons by other nuclei in this energy range.

Following this calculational scheme, it is obvious that the success in predicting integral nuclear parameters of fast reactors depends strongly on the accuracy of the various cross sections, resonance data and also on the methods of generating group constants. From the physical nature of the effects to be studied in fast reactors, following cross sections are important:

a) For neutron energies above some hundred keV:
fast fission of even-even nuclei (^{238}U, ^{240}Pu, ^{242}Pu), fission spectra, elastic and inelastic scattering data which influence the neutron spectrum and leakage (critical mass, shape of power distribution)

b) For neutron energies above some 10 keV to some hundred keV:
fission and absorption of heavy nuclei, elastic scattering resonance data (critical mass, breeding, coolant density and void effects)

c) For neutron energies from some hundred eV to some 10 keV:
resonance parameters for fission and absorption of heavy nuclei, absorption in fission products (Doppler effect, critical mass, breeding, fission product poisoning)

Besides the already mentioned unsatisfactory state of heavy nuclei fission and absorption cross sections, we want to emphasize that the gaps in the data for anisotropic elastic scattering distribution should be closed.

3.3. Special theoretical methods to analyse experiments in zero power facilities

In connection with the development of accurate experimental techniques at Karlsruhe it was recognized that adequate calculational methods had to be developed to interpret the experiments. Special attention was given to heterogeneity effects which are either due to the plate structure of the core or else arise in a sample experiment, such as material worth or Doppler experiment where one has to look at the heterogeneous configuration of the sample in its environment. In all cases the collision probability method was used, and resonance self shielding was included [10, 11, 12]. Table II shows the influence of heterogeneity in

a) a Na-void experiment [13], where the homogeneous results are lowered by about 20%,

b) a Doppler experiment [14], where the resonance interaction between a hot sample and the cold environment is considerable at high energies,

c) material worth measurements [12], which are sensitive to the sample size and the environment. These experiments were done in SNEAK-5C, an assembly with k_∞=1 and a softspectrum. The heterogeneity was very strong, which is reflected by the fact that the 5 g sample of ^{238}U was worth three times as much in the graphite, than in the fuel. It is obvious that in such an assembly large corrections for heterogeneity are necessary.

3.4. Theoretical methods to be developed

From the theoretical point of view the following aspects are not yet included in our procedure outlined above. Across core-blanket or core-reflector interfaces the space dependence of heavy isotope resonance self shielding has to be investigated more properly. Work is underway at Karlsruhe [14] to treat the case of plane geometry. The general case of space dependent resonance shielding in multizoned reactor systems or cells involving transient neutron spectra is rather complicated to deal with and has been done yet only approximatively [10, 14, 15]. A re-investigation of the resonance parameters of fissile nuclei in terms of a multi-level formula and the corresponding determination of resonance self shielding is important for soft spectrum systems and high content of fissile material, both for criticality and for Doppler effect calculations. For fast reactors now under design the multi-level effect is not very important [11].

It was stated in section 3.1. that no unique group constant set can be established for reactor calculations, with generally acceptable number of groups. In any case one has to make sure whether these group constants, which are originally prepared for instance for diffusion theory calculations, can be used also in transport calculations or in the calculation of the adjoint flux. A thorough investigation of the effects on integral parameters, caused by this inconsistency (mainly the neutron leakage is influenced), has not yet been made. Kiefhaber for example [16] showed that the use of flux weighted group constants in adjoint calculations may have an non-negligible effect in the calculations of neutron life-time and material worths.

4. METHODS USED IN INTEGRAL EXPERIMENTS

At Karlsruhe two fast assemblies and a fast-thermal coupled one are available for checking fast reactor calculations and the empirical determination of the behaviour of mocked-up fast reactors.

a) The subcritical fast assembly SUAK [17] is designed for pulsed neutron experiments on uranium and plutonium fuelled assemblies. The assembly is laid out for core volumes up to 600 liters and multiplication constants $k_{eff}<0.95$. It is pulsed either with a 200 keV Cockroft-Walton accelerator or a neutron flash tube. For time of flight (TOF) measurements flight channels up to 100 m length are provided.

b) The fast-thermal reactor STARK [18] consists of a 50 liter cylindrical zone with the composition of a uranium fuelled fast reactor surrounded by a thermal driver zone. The fast zone of this reactor is mainly used for checking experimental methods to be applied in SUAK and SNEAK and for supporting studies of fast reactor lattices.

c) In the fast critical facility SNEAK [19] experiments on clean physics cores and measurements on technical mock-ups of fast power reactors are performed. The maximum core size is limited by the fuel inventory, which is now 800 kg of ^{235}U-metal and 200 kg ^{239}Pu oxide fuel.

The fast cores of all three assemblies are built of fuel and diluent platelets of the dimensions 50.7 x 50.7 mm and thicknesses ranging from 1.6 to 6.3 mm. These platelets are contained in a stainless steel or aluminum matrix. Unit cells of less than 100 cm^3 volume can be constructed.

In the following sections the various experimental methods and their accuracies are discussed.

4.1. Determination of the critical mass

The most accurate but also most complex quantity determined in a critical experiment is the mass of fissile isotopes. The errors inherent in the

critical mass m_{cr} are due to

- the content and the isotopic composition of fuel (usually in the order of 0.1 to 0.3% of m_{cr}). Especially the Pu-content in oxide fuel is mostly not known exactly.
- corrections for partially inserted control rods and for detectors located in the core.

The critical mass can be given with a total uncertainty of less than 0.5% in m_{cr} or 0.3% in k_{eff}. For an evaluation detailed information on the geometry of the reactor has to be given in addition to m_{cr}.

Especially the irregular core boundaries and the internal heterogeneity of core zones may influence the critical mass considerably. For critical experiments carried out in the past this information is not always easily to obtain.

In SNEAK assemblies the shape correction is in the order of a few tenth of a percent.

Differences in critical mass of different compositions may be obtained by progressive substitution experiments [20, 21]. These experiments yield information on the nuclear properties of reactor media which can be built in small zones in fast critical reactors of not too different a composition, especially of not too different a diffusion constant. By this means only a small stock of about 200 kg Pu-fuel allows the investigation of core compositions as foreseen for large fast breeder reactors. In SNEAK-3 experiments, the critical mass of a 500 l plutonium-fuelled reactor was inferred with a 2% accuracy ($\approx$ 0.3% $\Delta k/k$) from the substitution of a 230 l Pu-zone in an uranium fuelled reactor.

4.2. The measurement of reactivity changes

The uncertainties in reactivity measurements depend strongly on the magnitude of the reactivity change to be measured.

a) Reactivity changes $\rho \gtrsim 2\ \$$, as measured for shut down rods and in substitution experiments, cannot be determined by reactivity compensation in SNEAK due to excess reactivity limitations. Therefore, such large reactivity changes are compensated by a change in critical mass.

b) Reactivity changes in the range $2\ \$ > \rho \geq 1\ ¢$, as measured for sodium void effects in zones of the reactor, poisoning of a few elements etc., are determined with calibrated control rods. The control rods are calibrated by period measurements and/or solution of the inverse kinetics equation after incremental movements of the rod. Calibration by inverse kinetics results in a systematic error up to about ± 3% in control rod worth [22], which is mostly due to the shift in statistical weight distribution. An absolute limit in accuracy is given by the reproducibility of the geometrical arrangement and the temperature distribution after a change in loading. A single change in loading results in an uncertainty of about ± 0.2 ¢. Careful elimination of error sources and repetitive measurements yielded small reactivity changes due to sodium loss with a precision of ± 0.05 ¢.

c) For measurements of reactivity changes $\rho < 1\ ¢$ special techniques are developed, which eliminate influences of temperature shift or loading changes. For material worth measurements a pile oscillator is used in connection with an automatic sample changer. The reactivity measurements are performed by recording either the flux signal while oscillating or the position of a auto-rod. This is a control rod, driven by a servo-mechanism to keep the flux level constant. The accuracy of these measurements is limited by statistics and the reproducibility in sample positioning. Both effects amount to an error of about $10^{-7}\ \Delta k/k$.

Any uncertainty in β_{eff} does not effect the accuracy of the techniques described above, but has to be taken into account, if reactivity measurements are compared with calculations or calculated corrections are applied to the critical mass.

Such a correction for the irregular core boundary is usually calculated by two-dimensional codes. More difficult to obtain is a calculated correction for the heterogeneity of the core. Bunching experiments, i.e. increasing the heterogeneity of a medium, give an experimental check of these calculations and allow an extrapolation to homogeneous medium properties [22].

4.3. Reaction rate measurements

Measured reaction rates give valuable information on the neutron balance in critical reactors. They allow to identify the sources of discrepancies in criticality calculations even if - due to compensation - no effect on k_{eff} is noticable. Of main importance are naturally the fission and capture rates in the fuel. Since measurements in zero power facilities do not yield the capture rates of ^{235}U or ^{239}Pu they are deduced from the neutron or reactivity balance equation [23, 24].

At the London Conference of BNES, 1969, detailed information on the techniques to measure reaction rates were presented. It is now widely accepted that accurate measurements have to consider the perturbation introduced by any detector very accurately. For capture and fission rate measurements minimum perturbation is achieved, if the detectors themselves are part of the fuel. In practice, this is accomplished by activating foils made of fuel material inside the lattice [25, 26]. Fission chambers are mainly used in measurements of reaction rate distributions, either to deduce material bucklings of large uniform zones or power distributions in complicated geometrical arrangements.

The precision of measurements of the fission rates in ^{235}U and ^{239}Pu and the capture rate in ^{238}U is quite high. But error margins quoted in literature do not always distinguish clearly statistical and estimated systematic errors. Since there is still some doubt about systematic errors, most laboratories are developing several independent techniques to measure these rates. An extensive study to intercalibrate the equipment for reaction rate measurements used at Karlsruhe and Cadarache was started recently.

The techniques currently in use are summarized below.

4.3.1. Fission rate measurements with chambers

The evaluation has to take into account perturbations introduced by the chamber walls, the connections to the electronic equipment and any guide channel for the chamber itself. Then, the precision of the measurements is limited by the counting time available and the determination of the effective fissile mass of the chambers. Reaction rate measurements thus are restricted

to an accuracy of ± 1%. Bucklings can be obtained from traverse measurements with less than 0.5% standard deviation [27].

4.3.2. ^{235}U and ^{239}Pu fission rate measurements with foils

These measurements introduce only very small perturbations. They may be performed in different manners:

a) Radiochemical analysis [28] of the samples.
Absolute β-counting of ^{99}Mo yields the fission rates of ^{235}U and ^{239}Pu with an error of about 1.5% and fission ratios with about 3% accuracy [23, 29]

b) Comparison of the γ-activity due to fission products with the activity of foils irradiated in a thermal neutron spectrum [26, 29] or inside a calibrated fission chamber. The accuracy quoted in these measurements is about 2% in fission ratios.

4.3.3. Fission rate measurements with solid state track recorders [26, 30]

This method is still in the stage of development. The tracks produced by fission fragments emerging from thin layers of fissile isotopes are recorded in a suitable catcher foil. After etching, these tracks can be counted visually. Accuracies of 3% in the fission ratios were obtained. The main difficulty seems to arise in the construction of a reliable and fast automatic counting device.

4.3.4. ^{238}U capture rate measurements

Radiochemical analysis [28] and absolute β-counting of ^{239}Np yields capture rates with an accuracy of about 1.5% [23]. Widely used is the comparison of γ-rays and or X-rays emitted in the ^{239}U or the ^{239}Np decay after simultaneous irradiation of foils in a thermal and the fast neutron spectrum. The accuracy achieved is about 2% in the ratio of capture in ^{238}U to fission in either ^{235}U or ^{239}Pu [26, 29]. At Karlsruhe, coincidences of 106 keV γ-rays and 104 kev X-rays of ^{239}Np are counted in a fast electronic circuit. The sensitivity of the equipment is determined by means of a calibrated ^{234}Am source, which also decays to ^{239}Pu via ^{239}Np [31].

An improved version of this technique [32] yields the capture rate of ^{238}U about 1% and the ratio of ^{238}U capture to ^{239}Pu fission with less than 2% accuracy.

In Table III the estimated accuracies of reactivity measurements and measured reaction rate ratios are listed.

4.4. Spectrum Measurements

The different techniques used to measure neutron spectra are discussed. The accuracy of these methods is stated, to define the role of neutron spectra in checking nuclear data.

4.4.1. Proton-Recoil-Counters

The lower limit of the measurable energy range lies around 1 keV and is determined by the fact that 1 keV protons create only about 30 electron-ion pairs. In fact, it seems, that already below about 5 keV results with reasonable accuracy cannot be obtained because of the lacking knowledge on the detailed energy dependence of the energy loss per ion pair for hydrogen.

On the high energy side, there is no principle limit and it has been shown [33] that with large counters and moderate pressure (below 5 atm) measurements up to 10 MeV are possible. For incore measurements, on the other hand, the dimensions of the counters are limited to some ten centimeters and therefore high pressures (above 10 atm) would be necessary to stop the high energy protons. Because of the difficulties caused by these high pressures incore measurements are restricted until now to the energy range below about 2 MeV.

The experimental errors in the proportional counter measurements are currently assessed as follows [33, 34]

Energy	5-30 keV	30-100 keV	100keV-1MeV	1-2 MeV	1-4 MeV	4-10 MeV
Statistics	2%	2%	2%	2%	2%	2%
Total systematic errors	20%	13%	10%	20%	10%	20%
Comments				(Small counters Incore)	(large counters)	

A large part of the overall experimental error is due to uncertainties in the correction of the distortions in the measured proton-recoil distribution, which arise from the truncation of proton-recoil tracks by the counter walls or by the extension of tracks into the end region, where there is little or no multiplication

Two difficulties encountered with incore proton-recoil measurements should be mentioned: Firstly, due to the relatively high efficiency measurements in Pu cores are restricted until now to the subcritical region. The second problem, common also to some other reactor physics experiments, which complicates the comparison of measured and calculated data, is the necessary void of some hundred cm^3 to install the detectors.

4.4.2. Li^6- and He^3-Semiconductor Sandwich Spectrometer

Both spectrometers are used above some hundred keV. Due to the better signal-to-noise ratio the He^3-spectrometer seems to be more favourable. A serious handicap for both spectrometers are the uncertainties of about 8% in the cross sections, which contribute a large part to the total experimental error. For a typical measurement with a Li^6-spectrometer figures for the total experimental error are given below [33].

Energy	0.4-2.5 MeV	2.5-4 MeV	4 - 10 MeV
Statistics	3%	10%	15%
Total systematic error	7%	10%	12%

4.4.3. The Time-of-Flight-Method

Measurements with the time-of-flight method are restricted to subcritical assemblies and comparison with the results from other methods are difficult, if the beam spectrum is different from the angle averaged spectrum.

The neutron detector used at Karlsruhe is a Li^6-glas scintillator Ne 905. It was calibrated in the energy range 100 eV to 100 keV against a B^{10}-NaJ-slab detector, the efficiency of which has been calculated by a Monte Carlo Code /35/. In the energy range between 60 keV and 320 keV the Li6-detector was calibrated with monoenergetic neutrons from a van de Graaf relative to a long counter. In the remaining energy range the efficiency relies on calculations with the recommended $Li^6(n,\alpha)$ cross section.

The experimental error is mainly due to uncertainties in the detector efficiency. In the energy range where it relies on the Monte Carlo calculation the error in the detector efficiency is assumed to be 10% and from 100 keV to 350 keV it may amount to 15%. Typical figures (12 hour run with flash tube at 100 pps and an energy resolution of 10%) for the errors in the measured spectra, are /33/:

Energy	200 keV	40 keV	10 keV	1 keV	0.1 keV
Statistics	< 1%	1%	3%	10%	50%
Total systematic error (Efficiency, zero time)	25%	15%	13%	10%	10%

4.4.4. Resonance Foil Activation in Sandwich Geometry

Up to twenty isotopes are used for the determination of the low energy end of the neutron spectrum by the resonance foil activation technique /26/. The energies of the main resonances range from 1 eV to 10 keV. The calibration factors, which depend on the γ-counter efficiency, are difficult to calculate and were determined experimentally in a 1/E-neutron spectrum and by comparison

with time-of-flight results at various SUAK assemblies. For the evaluation effective resonance integrals are calculated using the TRIX-1 program and most recent microscopic data.

The total experimental error depends strongly on the neutron spectrum. It amounts to 10-20% in the soft spectra of steam-cooled assemblies and to 20-40% in the harder spectra of sodium-cooled reactors and is composed mainly of the following contributions:

a) Correction for activities not related to the main resonance:
Soft spectrum 10%, hard spectrum 20%.

b) Resonance parameters and calculation of effective resonance integrals:
Soft spectrum 2%, hard spectrum 3%.

c) Counting statistics:
soft spectrum 4%, hard spectrum 8%.

d) Others: 2%.

4.4.5. Accuracy of spectrum measurements

Although the attempt has been made to use spectrum measurements for adjustment of neutron cross section data [37], in our opinion, this will be useful only after a considerable improvement of the measurements and after a more reliable estimation of errors is possible, compared to the figures quoted in this paper.

If one starts, according to [34], with the required standard deviation for fast reactor prediction of 0.01 in k, and 0.03 in breeding gain, examination of a range of systems leads (as a rough guide) to the target accuracy for the amplitude of a broad group spectrum as indicated in the upper part of Fig. 3. Comparing this target accuracy with the quoted experimental errors - where the statistical errors, except for the resonance foils, are ommitted because the uncertainties quoted in Fig. 3 are related to a broad group spectrum - one concludes, that at least in the energy range 10 keV to 4 MeV the experimental accuracy has to be improved considerably before spectrum measurements will be useful to improve the prediction for fast reactors.

5. CHECK OF NUCLEAR DATA BY INTEGRAL EXPERIMENTS IN CRITICAL ASSEMBLIES

A check of nuclear data in critical assemblies poses strict requirements on both experimental and calculational techniques, because a valid check of cross sections by integral experiments is possible only under the following conditions:

a) the experiments must be carried out with sufficient precision
b) a meaningful interpretation must be possible; this means that the measured effect can be calculated by well-established techniques, for which errors are small.
c) the result must be sensitive to the cross section data of interest.

From these requirements, the following definition of a "clean" critical experiment suitable for data check can be given: It is a set of good precision measurements on a critical assembly in simple geometry, such that errors due to calculational methods are small.

The characteristics of some important clean critical experiments are given in Table IV. Of course, none of them is ideally clean, but they all are either reflected single-zone cores, or have a large enough uniform test zone. An improvement would be desirable to reduce the heterogeneity effects: These are quite large, expecially in the benchmark series, and it would be useful to have at least a few experiments in a homogeneous zone.

The important techniques used in the experiments are:

1) criticality measurements (critical mass, substitution)
2) reaction rates (ratios of reaction rates at the center, fission rate traverses to obtain B^2)
3) reactivity worth measurements of absorbing materials
4) spectrum measurements.

These techniques were discussed in detail in section B. The first three give good experimental precision, typically 2% for reaction rates, 1% in the critical mass, and 10^{-7} $\Delta k/k$ for reactivity measurement. The

accuracy of spectra measured is too low to be directly used for data check.

In the following the usefulness of these experiments for data check will be discussed under the aspects b) and c) above.

5.1. Criticality and ratios of reaction rates

k_{eff} of a critical configuration, though easy enough to measure, is rather difficult to interpret, mainly because of the many corrections. First, there are the corrections for irregular boundary, and for transport effects, which are fairly small in large cores, but may be sizeable in small cores. Obviously that they may be eliminated by experiment, if either a $k_\infty=1$ experiment is performed, or B^2 is obtained from measured fission rate traverses.

The main correction, however, is for heterogeneity. It is quite large for assemblies with metal fuel, and it is uncertain, especially for large and complicated plate cells, like in ZPR-III/48. In fact, the Δk due to heterogeneity was calculated to be 0.0182 by Broomfield and Palmer [43], 0.0126 by Fillmore et al. [38], whereas the Karlsruhe ZERA Code [10] gives 0.0151. The values scatter by more than 0.5% in Δk, and therefore, the calculation of the heterogeneity effect is the dominating uncertainty in k_{eff} calculations for a large cell. The bunching experiments at Argonne [44] gave only agreement in order of magnitude. Though it was found, that the effect is much smaller, and therefore probably less uncertain, with oxide fuel, it is certainly worthwhile to build, for some typical cases, quasi-homogeneous assemblies, where these effects are not present. Such experiments were carried out in England in an epithermal spectrum [49], and are being planned at Ispra in fast spectra.

Ratio of reaction rates can be measured with chambers, or with foils. However, foil measurements can generally be well interpreted, and, therefore, qualify for good precision data check, whereas chamber measurement should

be used with caution. Foil measurements were carried out in the $k_\infty = 1$ assemblies on ZPR-III, and on ZEBRA, in SNEAK-3A-2, and in MASCURA.

The multiplication factor k_{eff} is sensitive to $\nu\sigma_f - \sigma_a$, and errors in the different cross section may compensate in part. Typical sensitivities for a 5000 liter oxide breeder, as taken from Zaritsky and Troyanov [62] are given in Fig. 4. On the other hand, measurements of the ratios σ_f^5 / σ_f^9 and σ_c^8 / σ_f^5 are sensitive mainly to the cross sections in the ratio, and such measurements help to resolve compensating errors in k_{eff}. This is clear from Table V, which shows some typical sensitivities of spectrum-averaged cross sections in the SNEAK-3A-series, normalized to a change of -10% in the cross sections, However, the table also shows that the sensitivities are not strongly spectrum dependent, and a number of measurements in strongly different spectra would be required to allow conclusions in the 4-groups frame used in the table.

Broeders [61] has determined the influence of uncertainties in $\sigma_f(^{235}U)$ and $\sigma_\gamma(^{238}U)$ for integral parameters in SNEAK-3A-2, a uranium fuelled assembly with soft neutron spectrum to simulate steam cooled systems. The changes in the cross sections are listed in Table VI, the basic group constant set is the KFK-SNEAK set [6]. The results of fundamental mode calculations, corrected for heterogeneity, are quoted in Table VII. It is: $\Delta k_L = k(N_H{=}0) - k(N_H)$, where N_H is the normal hydrogen concentration in SNEAK-3A-2; $\Delta k_{\rho/2} = k(N_{H/2}) - k(N_H)$; $\Delta k_{2\rho} = k(2N_H) - k(N_H)$; RSDC $= \frac{dk}{k} / \frac{d\rho}{\rho}$.

It is realized from Table VII that the assumed uncertainties of about 10% in $\sigma_f(^{235}U)$ and up to 20% in $\sigma_\gamma(^{238}U)$ have rather large effects on most of the integral data quoted. From this it follows that the differential nuclear data have to have a much higher precision. It would indeed be a drastic and highly appreciated improvement, if the data of independent differential measurements are consistent within a 5% margin, though even a higher accuracy has been requested in literature.

5.2. Central material worth measurements

The interpretation of material worth measurements is problematic. It should be mentioned first that the worth of materials with a strong slowing

down cross section depends very much on the details of the adjoint spectrum, so that they cannot be used presently for data check. Therefore, the following discussion will be restricted to absorbing or fissile materials.

The interpretation is complicated because of two problems

a) Uncertainties in the kinetics parameters, especially in β_{eff} for Pu-^{238}U-fuelled assemblies.

b) Dependence of the reactivity worth on the sample size and on the heterogeneity of the environment.

The importance of the first problem in Pu-fuelled assemblies will be illustrated by the following results obtained in the U-fuelled core 3A-2 and in the partially Pu-fuelled core 3B-2 in SNEAK. The Pu-zone was substituted into 3A-2, and the difference in buckling was determined, and found to be small. The reactivity worth of 1 cm^3 core material in the center is given by

$$\rho_c = \frac{(\phi^+ DB^2 \phi)}{F \cdot \beta_{eff}}$$

where F is the usual normalization integral. The ratio of calculated over experimental reactivity ρ_c was found to be

0.94 for SNEAK-3A-2
1.07 for SNEAK-3B-2

Although these results are only preliminary, the difference of 13% in going from the U-fuelled to the partially Pu-fuelled assembly seems to indicate an inconsistency between the kinetics parameters used for Pu assemblies and for U assemblies.

The results are in line with the observation, published, for example by Little and Hardie [46], that materials worths in Pu-fuelled assemblies are consistently overpredicted by 20-25%. The conclusion is that, so far, one can only use ratios of reactivity worths.

The second effect, namely the dependence of reactivity worth on the sample size and the environment, has been studied recently at Karlsruhe [12], and these effects are understood now about as well as heterogeneity effects. Thus although material worth measurements require a careful analysis they will probably give useful information on data of fuel and structural materials, which complements the information from reaction rate measurements.

In several cases, material worth measurements were used as integral checks in cases where large uncertainties in the cross sections existed. For example, Oosterkamp has carried out experiments in SNEAK to check the data of ^{240}Pu. Reactivity measurements were made with two PuO_2UO_2 compositions which contained the same amount of ^{239}Pu, but different amounts of ^{240}Pu. The measured difference in reactivity in going from 8% to 22% in ^{240}Pu is shown in the table, and compared with calculations using the ABN-data [5], for σ_f and σ_c, and also data evaluated by Pitterle [47], which are based on recent differential measurements.

Difference in reactivity for 2 fuel compositions, cents

	SNEAK-3B-2 (soft spectrum)	SNEAK-4B (hard spectrum)
Experiment	16.6	22.9
ABN-Set	11.3	13.3
Pitterle	20.0	21.8

The results in the hard spectrum clearly favour the Pitterle data, whereas the results in the soft spectrum show only a slight bias. It should be noted, that σ_c values by Yiftah which are slightly higher than those by Pitterle would give better agreement in the soft spectrum.

An other example is taken from a paper by Barré et al. [40]. Worth measurements of nickel in the French reactor ERMINE, in addition to reflector worth measurements, give a check on the capture cross section. The results are in good agreement with recent capture data by Spitz.

Differential cross section data on fission products were compiled by different authors, but are still highly uncertain [48]. On the other hand,

data on the reactivity effect of fission products can be obtained by material worth measurements. The critical facility STEK in the Dutch research center Petten [49], has been designed to measure highly radioactive samples by the oscillator technique. Directly applicable results will be available soon.

5.3. Special experiments to check nuclear data

Measurements of the Doppler effect are sensitive to the low energy spectrum, and adjoint spectrum, which in turn depend on many cross section data. For ^{238}U, in spectra of Na reactors, discrepancies between Karlsruhe calculation and experiment are about 10% though somewhat better agreement is obtained in assemblies containing hydrogen.

Though the calculational methods are well developed now, such discrepancies are difficult to trace, because too many cross section data are involved. However, an interesting check on data was possible by means of the Doppler effect in ^{239}Pu [11]. The Doppler effect occurs mainly between 0.1 and 5 keV, and this is the energy range where the large discrepancy between the KAPL "lowα" data and the "highα" data by Schomberg and Gwin existed. Experiments were carried out in SNEAK-3B-2 in the normal core, and in a boron environment, which was designed to suppress the absorption effect, but to retain the fission effect. The analysis of both experiments shows clearly that the calculation with "highα" values is compatible with the experimental results within about 25%, which may be expected in a Pu Doppler experiment, whereas the calculation with "lowα" values is not. Table VIII shows the breakdown of the calculated values.

One could, in principle, try to use reaction rate traverses for data check. There are disagreements in the SNEAK measurements in the vicinity of interfaces, and in the blankets, and there is evidence that a large portion of the disagreement is due to errors in cross section data. However, the measurements depend on the cross sections in a very complicated way, which certainly also involves the spectrum, and, therefore, they are not well suited to trace errors.

Also, for the same reason, the fine structure of reaction rates in a cell is not a good quantity to trace cross section errors. Furthermore, the reactor physicist tries to design his experiments with small heterogeneity effects, so that they may be considered as a correction. Calculational methods, at the present stage, are just about adequate to give an estimate of this correction, and there is still dis-agreement between heterogeneity calculations in different laboratories. In order to use heterogeneity effects for data check, one would have to design an experiment with large heterogeneity effects, and be sure that the calculations describe these effects adequately. This has probably not been carried out so far.

6. CONCLUSION

The foregoing discussion shows that in most cases the theoretical methods are adequate to analyse the experimental results in critical assemblies, which can be performed also with a relatively high accuracy. For a meaningful analysis one should be sure to use sufficiently accurate calculational methods and precise and reliable experimental results. Spectrum measurements definitely need more precision. On the other hand it is generally agreed that the differential data presently available are not good enough to allow a reliable calculation of the reactor parameters with the desired accuracy. This is the main reason why critical experiments are still being carried out. The information obtained in critical experiments can be used for power reactor design in different ways. The first not very sophisticated way is to use scaling factors obtained from calculated and measured integral data. This procedure is nevertheless useful and allows a normally sufficient prediction of the main characteristics of a power reactor, if the scaling is based on engineering mock-up experiments [51].

The second way is to analyse the discrepancies between measured and predicted integral data in order first to locate the main responsible microscopic data uncertainties and secondly to give preference to specific cross section measurements according to the direction which is indicated by the

analysis of discrepancies. It can be stated generally that only a systematic study of a variety of fast cores, which differ in geometry, material composition, and neutron energy distribution combined with a thorough comparison and re-evaluation of the main microscopic data can provide more definite conclusions about the reliability of nuclear data to be used in fast reactor calculations. The results of such investigations performed at Karlsruhe were reported by Küsters [52] and Kiefhaber [53], and led for instance to the conclusion to prefer lower ^{238}U capture data than formerly used in our calculations (now Moxon's data [54] are included). The low ^{235}U fission cross section measured by Pönitz [55] must be excluded. Our interpretation of the 1967 Schomberg's data [56] for the α-value of plutonium should better be replaced by Gwin's data [57]. $\sigma_f(^{239}Pu)$ should be increased above White's results [58] to the Pfletschinger and Käppeler data [59]. Oosterkamp [12] could rule out the formerly used capture data of ^{240}Pu. The calculation of criticality for various assemblies, as given in Table IX shows that a relatively good prediction is possible because of the fact, that for most Pu-fuelled assemblies investigated k_{eff} is underpredicted by about 1 to 2%, while for most U-fuelled assemblies k_{eff} is overpredicted by nearly the same amount.

This procedure in the end will lead to and support those measured microscopic data, which are consistent with the integral data obtained in critical facilities.

Finally there is the third way of adjusting cross sections to fit available integral data, using a least-squares fit technique. This way, in our opinion, is just a more systematic way compared to the first one to use integral data in reactor design. One has to be careful about certain pitfalls: The number of adjustments must be lower than the number of integral data, otherwise one gets meaningless oscillations in the adjusted cross sections [50]. It must be realized, that adjusted and non-adjusted cross sections generally have about the same standard deviation so that the procedure does not yield additional information on a single cross section except in cases where there are large uncertainties or errors in the original values. Thus, it is not surprising that the fitting procedure, when applied to different original values, does not necessarily lead to the same adjusted cross sections. An example is given for adjusted values

obtained by Barré and by Rowlands, as reported in [49].

Cross section adjustment

	Energy Range keV	Barré		Rowlands		Difference % in adjusted values
		Adjusted	Adjusted value	Adjusted	Adjusted value	
^{238}U capture	302-498	-14.3%	.111	-7.8%	.120±6%	8%
	24.8-40.9	-11 %	.392	+3.8%	.472±6%	20%
	3.36-5.53	+ 6 %	.927	+7.4%	.949±6%	2%
^{239}Pu alpha						
	0.768-9.12	+40 %	.78	+11%	.94 ±15%	20%

It is apparent that the difference of 20% in the adjusted data of ^{238}U capture in energy group 25 - 40 keV is rather large, and so is the difference in ^{239}Pu alpha which is also 20%. However, these differences are of the same order as the uncertainties in differential data (compare the difference between Moxon and Pönitz for ^{238}U capture and the uncertainty of 15% quoted by Gwin for Pu alpha), and if it is accepted that the cross section fit does not improve differential data, there is no contradiction.

As a final remark, it should be emphasized that work with adjusted cross sections does not ensure a meaningful extrapolation to integral data of assemblies which differ widely in composition or spectrum from those assemblies used in the fit.

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Table I. Various ^{238}U capture measurements at 30 keV

Neutron energy (keV)	$\sigma_{n\gamma}$ (mbarn)	Authors References given in [60]	Comments	Deviation relative to de Saussure (%)
30.0±8	470±38	de Saussure, Weston et al.	relative to $(\sigma_{nf}+\sigma_{n\gamma})$ (^{235}U)	taken as reference
30.0±7	473±74	Gibbons, Macklin, Miller, Neiler	normalization to the absorption cross section of Indium	+0.6
30.0±1.5	479±14	Menlove, Pönitz	"grey" detector, absolute measurement of $\sigma_\gamma(^{238}U)$ at 30 keV	+2
30.0	467±18	Pönitz	grey detector; relative to $\sigma_f(^{235}U)$	-0.6
30.1	549±55	Macklin, Gibbons, Pasma	Moxon-Rae detector; relative to σ_γ(Ta)	+17
30.5	480	Bilpuch, Weston, Newson	normalized to those of other experimenters	+2
30	350* (±50)	Bergquist	relative to capture in Ag, normalized to σ_γ(Ag) at 24 keV	-26
30	373 * (±77)	Hanna, Rose	relative to the hydrogen elastic scattering cross section	-21
30	458 * (±70)	Linenberger, Miskel	relative to $\sigma_f(^{235}U)$	-3
30	420 * (±30)	Moxon	relative to the absorption cross section of ^{10}B	-11
30	526 *	Tolstikov et al.	normalized to the ^{10}B cross section at 24 keV	+12

* linearly interpolated between neighbouring experimental points

Table II. Effect of heterogeneity in different experiments

a) Na-Void Experiment (205 liters, SNEAK-6A), reactivity in cents

Experiment	2.32
Calcul. Homogeneous	2.72
Calcul. Heterogeneous	2.30

b) Doppler Experiment (25% enriched U in SNEAK-3A-2, depleted U in ZPR-6/4Z), relative units of reactivity change

Energy Range	Effect without interaction		Correction for hot sample-cold environment interaction	
	SNEAK	ZPR6	SNEAK	ZPR 6
Above 10 KeV	-16.5	-25.5	-3.5	-11.8
1 - 10 KeV	-42.4	-46.1	-2.6	- 3.2
Below 1 keV	-28.1		+1.2	
Total	-87.0	-71.6	-4.9	-15.0

c) Material worth experiment SNEAK-5C

	Weight, g	Position	Exper., μ ϕ/g	Homog. Calc. / Exp.	Heter.Calc. / Exp.
^{238}U	60	1	-37.7	0.97	1.16
	60	2	-24.4	1.49	1.26
	5	1	-86	0.43	0.80
	5	2	-25	1.42	1.22
^{239}Pu	5	1	443	0.98	1.09
	5	2	390	1.13	1.19

Position 1 in the graphite region of the unit cell

Position 2 in the fuel region of the unit cell

Table III. Estimated accuracies of integral experiments

A) Reactivity measurements at SNEAK

	$1\ ¢ \lesssim \rho \lesssim 2\ ¢$	$\rho \lesssim 1\ ¢$
Relative	± 3%	-
Absolute	> ± 0.2 ¢	$\pm 2 \cdot 10^{-3}$ ¢

B) Recent reaction rate ratio measurements by foils

	radiochemical analysis	calibration in thermal flux	calibration by fission chambers	capture with Seufert/Stegemann method [31]
σ_f^5/σ_f^9		2.2% ANL [29]	2.2% ANL [29] 2.1% UKAEA [26]	
σ_f^8/σ_f^9	3.1% ANL [29]		2.9% ANL [29]	
σ_f^8/σ_f^5	3.3% ANL [29]		2.2% UKAEA [26] 3.1% ANL [29]	
σ_c^8/σ_f^9	3.6% ANL [29]	1.7% UKAEA [26] 1.4% ANL [29]		1.3% UKAEA [26] 1.6% UKAEA [32]
σ_c^8/σ_f^5	3.0% ANL [23]	1.5% ANL [29]		2.9% Khe [25]

Table IV. Some important clean critical experiments

	Benchmark series ZPR-III/48,49, 50,53	k_∞-series ZEBRA,8A-8C, ZPR-III/55	SNEAK-3 series	MASURCA-2A,2B
Core-Geometry	cylindrical 1-zone core, reflector	k_∞-zone, with degraded spectrum, driver and buffer	cylindrical 1-zone core, reflector (3A-1, 3A-2) substitution of inner Pu-zone (3B-2)	cylindrical 1-zone core reflector (2B) substitution of inner Pu-zone (2A)
Fuel	$Pu/^{238}U$	$Pu/^{238}U$	enriched U(3A) $Pu/^{238}U$(3B-2)	enriched U(2B) $Pu/^{238}U$(2A)
Cell-Geometry	complicated plate cell (3 drawers)	relatively simple, but large cell	simple in the 3A-series (U) complicated in 3B-2 (inner Pu-zone)	simple rod cell (4 rods)
Important Measurements	k_{eff}, fission ratio with chambers(not with foils),reactivity worth with samples of various sizes	k_∞, ratios of reaction rates with foils across the fuel plates, spectrum (TOF, proton recoil)	k_{eff}, ratios of reaction rates (only in 3A-2 with foils), reactivity worths, spectrum (^6Li, (proton recoil, sandwich foils), Pu-α by Doppler experiment	B^2, ratios of reaction rates (with foils across fuel rods)
Special Feature	composition similar to a mixed carbide fuelled Na cooled fast breeder; various modifications	degraded spectrum to emphasize the region where α_{Pu} was uncertain	composition similar to a steam-cooled UO_2 (or mixed oxide) fuelled fast reactor	
References	29	26, 29	22, 38	42

Table V. Sensitivities of spectrum-averaged cross sections to change of -10% in cross sections (SNEAK 3A-series)

Energy Groups (ABN)	$\overline{\sigma^8_c}$			$\overline{\sigma^9_f}$		
	3A-0	3A-1	3A2	3A-0	3A-1	3A-2
1 - 5	-0.006	-0.006	-0.006	-0.019	-0.019	-0.018
6 - 8	-0.030	-0.024	-0.020	-0.046	-0.039	-0.031
9 -11	-0.043	-0.037	-0.031	-0.030	-0.024	-0.019
12 -16	-0.009	-0.020	-0.027	-0.027	-0.005	-0.021
Total	-0.088	-0.087	-0.084	-0.100	-0.097	-0.089

Table VIII. Break-down of the calculated reactivity effect in a Doppler experiment (sample: 450 g PuO_2, temperature change 400°C) 10^{-6} $\Delta k/k$ for different Alpha-values for ^{239}Pu

	Normal Core		Boron Environment	
	Low Alpha	High Alpha	Low Alpha	High Alpha
Fission	+21.2	+19.9	+13.4	+11.0
Absorption	-18.7	-21.4	- 7.1	- 7.4
^{240}Pu	+ 2.5	- 1.5	+ 6.3	+ 3.6
	- 1.3	- 1.2	- 0.4	- 0.4
	+ 1.2	- 2.7	+ 5.9	+ 3.2
Expansion	- 0.6	- 0.6	- 0.1	- 0.1
Total calculated	+ 0.6	- 3.3	+ 5.8	+ 3.1
Experiment	-3.7 ± 0.1		+2.5 ± 0.1	

Table VI. Cross section variation used for the calculations

Group	Energy Range	Variation of σ_f (U235) (%)	Variation of σ_γ (U238) (%)
1	6.5 -10 0.5 MeV	-10 x)	-10
2	4.0 - 6.5 "	-10	-10
3	2.5 - 4.0 "	-10	-10
4	1.4 - 2.5 "	-10	-10
5	0.8 - 1.4 "	-10	-10
6	0.4 - 0.8 "	+ 7	-10
7	0.2 - 0.4 "	+ 7	-20
8	100 - 200 keV	+ 7	-20
9	46.5 - 100 "	+ 7	-20
10	21.5 - 46.5 "	+ 7	-20
11	10.0 - 21.5 "	+ 7	-20
12	4.65 - 10.0 "	+ 7	-20
13	2.15 - 4.65 "	+ 7	-20
14	1.0 - 2.15 "	+ 7	-20
15	0.456 - 1.0 "	+ 7	-15
16	215 - 465 eV	+ 7	-15
17	100 - 215 "	+ 7	-15
18	46.5 - 100 "	+ 7	-15

x) Because in group 1-5 σ_f (U235) has as upper limit the KFK-SNEAK data, the calculations are performed with the lower limit.

Table VII. Influence of data uncertainties on integral nuclear parameters in SNEAK-3A-2

Reactor parameter	KFK 793/KFK 776		Fundamental Mode Calculations *)											
			$B^2=24.010^{-4}$	$B^2 = 25.49\ 10^{-4}\ cm^{-2}$										
	Experiment	KFK SNEAK SET 2 - Dim.	KFK SNEAK SET	KFK SNEAK SET	σ_γ(U238) Low Gr. 1-5	σ_γ(U238) Low Gr. 6-9	σ_γ(U238) Low Gr.10-14	σ_γ(U238) Low Gr.15-18	σ_f(U235) Low Gr. 1-5	σ_f(U235) High Gr. 6-9	σ_f(U235) High Gr.10-12	σ_f(U235) High Gr.13-15	σ_f(U235) High Gr.16-18	σ_γ(FE) Low Gr. 9-14
k_{eff}	1.000	0.9886	1.0047	0.9886	0.9897	0.9995	1.0101	0.9926	0.9811	1.0017	0.9961	0.9944	0.9916	0.9934
$\Delta k_L \times 10^2$	- 7.0	- 5.91	- 6.32	- 6.55	- 6.54	- 6.13	- 7.04	- 6.91	- 6.63	- 5.85	- 6.42	- 7.00	- 6.83	- 6.17
$\Delta k_{\rho/2} \times 10^2$	- 3.8	- 3.17	- 3.30	- 3.43	- 3.43	- 3.22	- 3.54	- 3.64	- 3.48	- 3.10	- 3.35	- 3.63	- 3.62	- 3.42
$\Delta k_{2\rho} \times 10^2$	+ 4.8	+ 3.74	+ 3.89	+ 4.09	+ 4.08	+ 3.86	+ 3.97	+ 4.34	+ 4.05	+ 3.76	+ 3.96	+ 4.21	+ 4.34	+ 4.04
R.S.D.C. $\times 10^2$	+ 5.8	+ 4.70	+ 4.54	+ 4.76	+ 4.75	+ 4.46	+ 4.76	+ 5.01	+ 4.83	+ 4.31	+ 4.61	+ 4.98	+ 5.06	+ 4.72
$-\Delta k(\Delta T) \times 10^2$	-	-	1.04	1.01	1.01	1.02	0.96	0.961	1.01	1.00	1.00	1.01	1.02	1.02
$^{25}\sigma_\gamma/^{25}\sigma_f$	-	-	0.321	0.320	0.320	0.321	0.324	0.321	0.324	0.312	0.315	0.315	0.317	0.321
$^{28}\sigma_\gamma/^{25}\sigma_f$	0.130	0.137	0.139	0.138	0.137	0.131	0.124	0.136	0.140	0.135	0.136	0.137	0.138	0.136
$^{49}\sigma_\gamma/^{25}\sigma_f$	-	-	0.303	0.301	0.302	0.303	0.307	0.304	0.305	0.294	0.296	0.296	0.297	0.303
$^{28}\sigma_f/^{25}\sigma_f$	0.0338	0.0297	0.0288	0.0291	0.0291	0.0288	0.0284	0.0290	0.0295	0.0287	0.0289	0.0289	0.0290	0.0290
$^{49}\sigma_f/^{25}\sigma_f$	-	-	0.965	0.966	0.965	0.966	0.963	0.967	0.978	0.944	0.953	0.954	0.959	0.966

*) With heterogenity corrections from KFK 776

$\delta k_{eff} = + 0.46\ 10^{-2}$; $\delta\Delta k_L = - 0.3\ 10^{-2}$; $\delta\Delta k_{\rho/2} = - 0.14\ 10^{-2}$; $\Delta k_{2\rho} = + 0.32\ 10^{-2}$

Table IX. Best available criticality values calculated for various Fast Zero Power Assemblies

Assembly	Best available value for k_{eff} (MØXTØT-set)	Experimental result	Δk Theory-Experiment
SUAK U1B	0.856	0.86 ± 0.01	-0.004
SUAK UH1B	0.930	0.945± 0.01	-0.015
ZPRIII-10	1.011	1.000	+0.011
ZPRIII-25	0.997	1.000	-0.003
SNEAK-series 3A0	0.937	0.930	+0.007
3A1	0.968	0.962	+0.006
3A2	1.000[1)]	1.000	±0.000
3A3	1.036	1.048	-0.019
SNEAK-3A1	1.020	1.000	+0.020
SNEAK-3A2	1.013	1.000	+0.013
SNEAK-3B2	1.000	1.000	±0.000
ZPRIII-48	0.989	1.000	-0.011
ZPRIII-48B	0.987	1.000	-0.013
ZEBRA 6A	0.985	1.000	-0.015
SNEAK 5C	1.040	1.03 ±0.01[2)]	+0.010
ZPRIII-55	0.983	1.000	-0.017

1) Normalization point for the SNEAK-3A-series

2) Preliminary experimental results

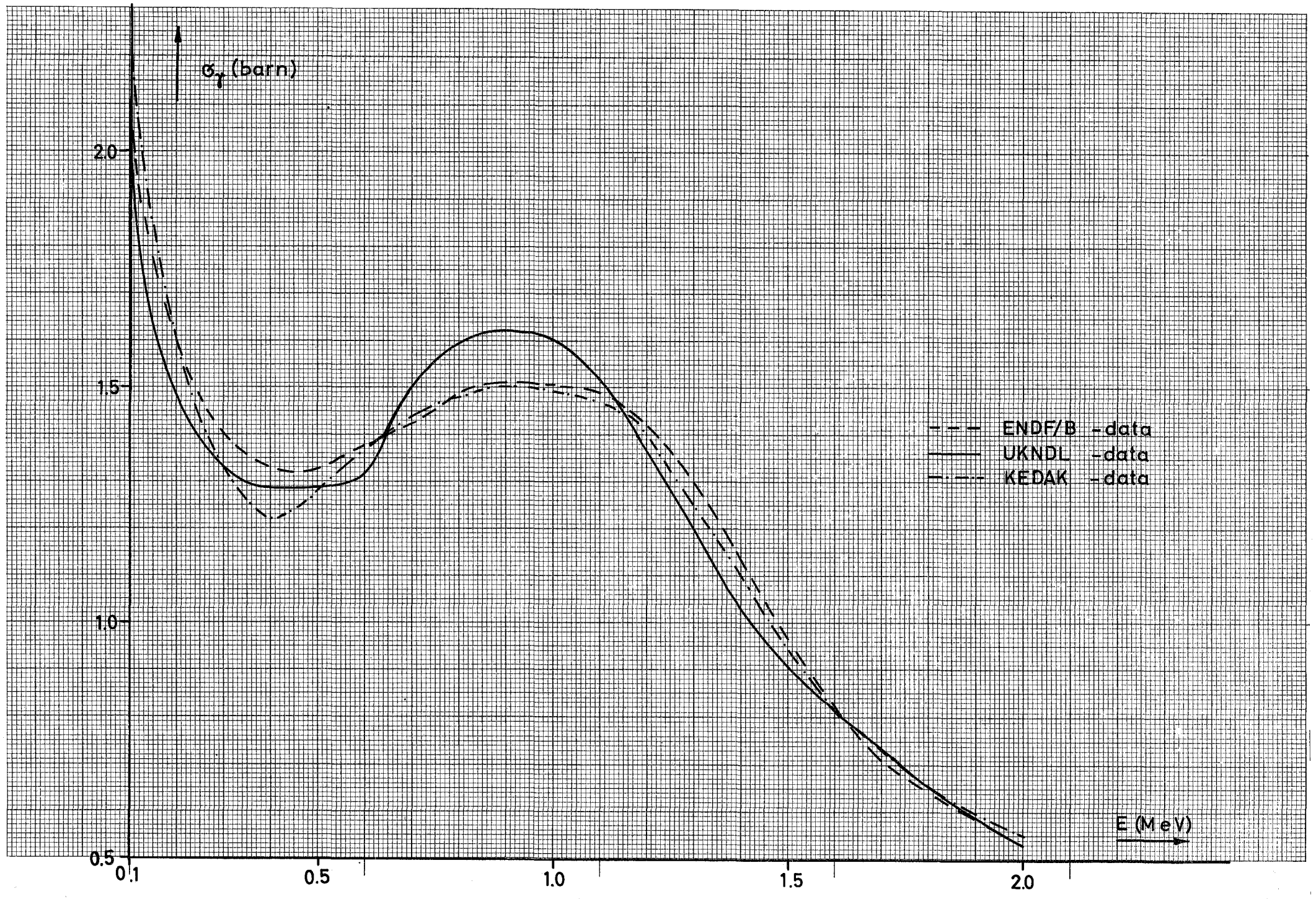

Fig.1. ^{238}U Capture Cross Section

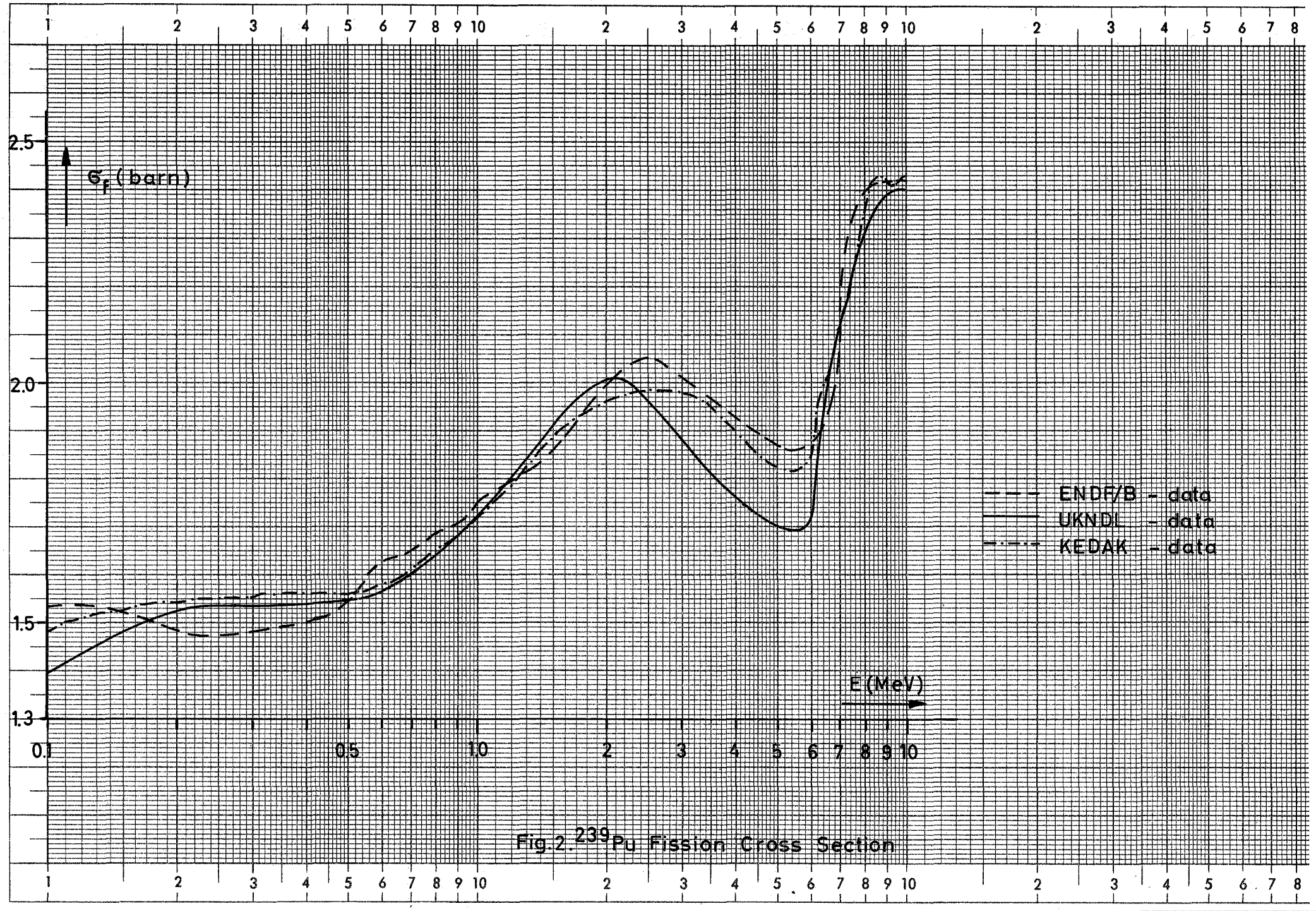

Fig.2. ^{239}Pu Fission Cross Section

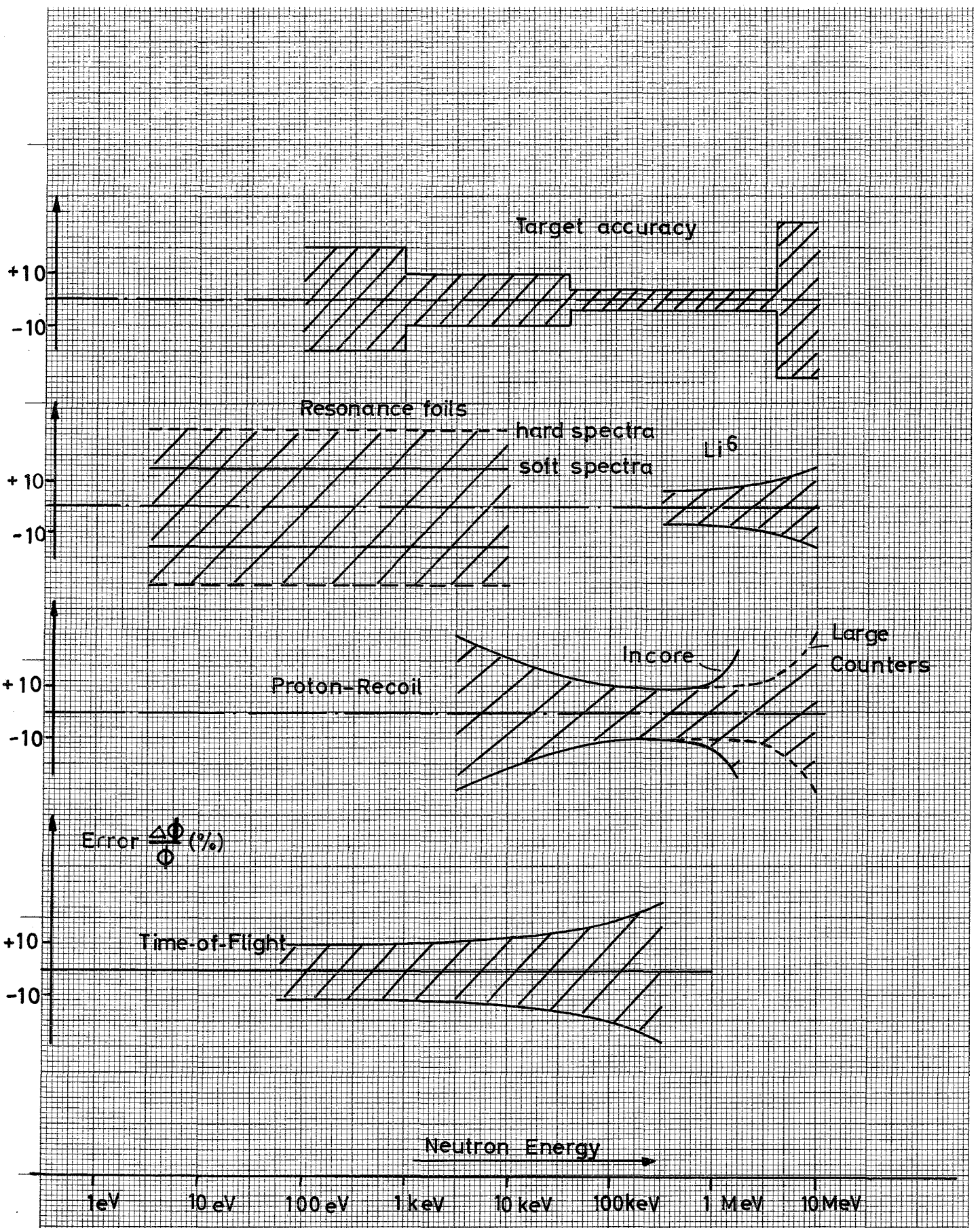

Fig. 3. Tentalive target accuracy and quoted systematic errors for broad group neutron spectra.

Fig.4

Sensitivities $\frac{\sigma}{k}\frac{\partial k}{\partial \sigma}$ for a 5000 Liter Oxide Breeder (after Zaritskiy and Troyanov)

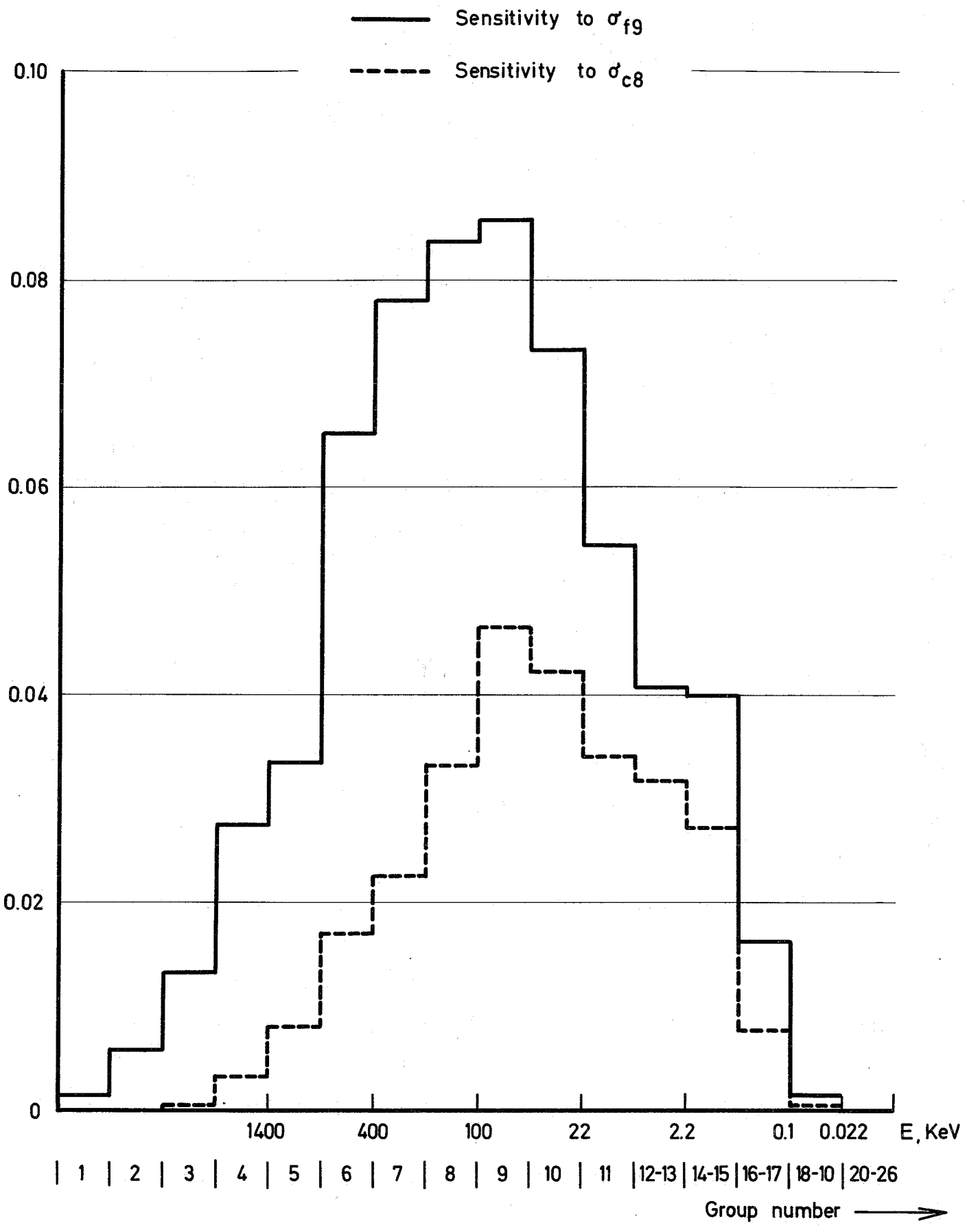

FIG. 5 PROCEDURE OF DATA EVALUATION

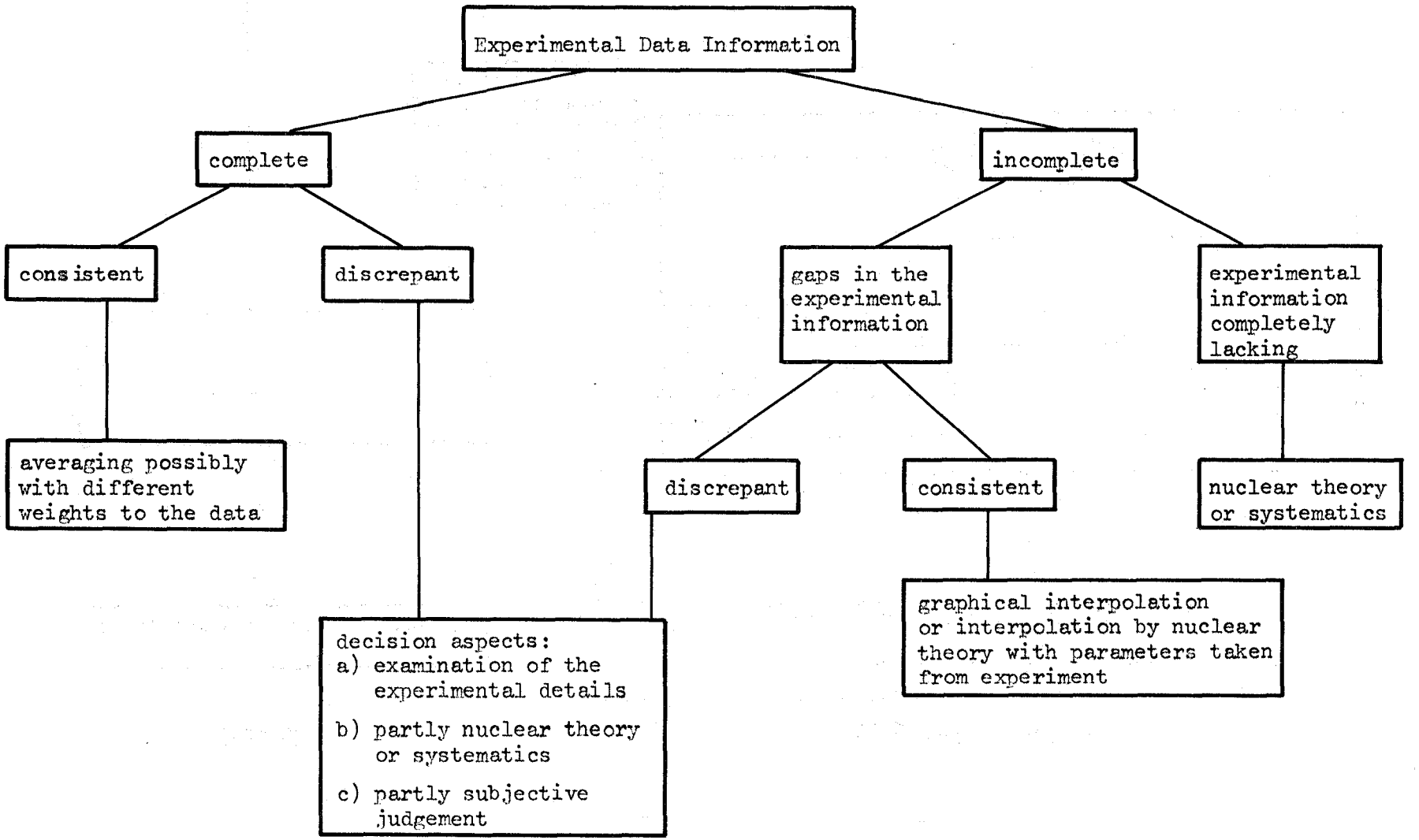

FIG. 6 WAYS TO REDUCE DATA DISCREPANCIES