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U.K. Nuclear Data Progress Report January – December 1986

Editors: M R Sené and J A Cookson

Nuclear Physics Division
Harwell Laboratory, Oxfordshire OX11 0RA

June 1987

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PREFACE

This report is prepared at the request of the United Kingdom Nuclear Data Committee (UKNDC) and summarises nuclear data research in the UK between January and December 1986.

Nuclear data are presented by laboratory with contributions this year from the UKAEA establishments at Harwell and Winfrith, from the National Physical Laboratory, and from the Universities of Birmingham, Edinburgh and Oxford. Included in these contributions are reports of work carried out by collaborations involving the following institutions: Harwell Laboratory, AEE Winfrith, Culham Laboratory, JET, the Universities of Birmingham and Manchester, Imperial College London, Hammersmith Hospital London, the Netherlands Energy Research Foundation (ECN), ENEA at Frascati and Bologna, the Instituto di Fisica Applicata Generale Milan, the Scuola Normal Superiore Pisa, IPK Jülich, the Physicalisch Technische Bundesanstalt West Germany, Los Alamos National Laboratory, and the NEA Data Bank.

Contributions on Chemical Nuclear Data are grouped under that heading, and have been gathered by the Chemical Nuclear Data Committee of the UKNDC. Much of this work is again the product of collaborative effort involving Harwell, Winfrith, Dounreay, Aldermaston, the University of Birmingham, the Central Electricity Generating Board and British Nuclear Fuels Plc.

A report of the 19th UK Nuclear Data Forum held at Harwell Laboratory, including summaries of the three invited lectures, appears as Chapter 8.

Where work is clearly relevant to requests in WRENDA 83/84 (INDC(SEC)-88/URSF), the appropriate numbers are given after the title of the contribution. In addition, a CINDA-type index is included at the end of the report.

Contributions to the report on nuclear data topics are welcome from all sources, and an invitation is extended to researchers in other laboratories of industry, government, universities and polytechnics to use this medium.

CONTENTS

1. NUCLEAR PHYSICS DIVISION, HARWELL LABORATORY

(Division head: Dr. A. T. G. Ferguson)

1.1 Evaluation of cross-sections for the Joint Evaluated File (M. C. Moxon, B. H. Patrick and M. G. Sowerby)

The UK in cooperation with France and Germany is responsible for producing a new evaluation of the neutron induced cross-sections of ^{238}U which will be submitted to JEF (the Joint Evaluated File Project which operates within the framework of the NEA Data Bank as a collaboration between laboratories in Member Countries) for inclusion in the JEF-2 library. Up to the present time much of our work has involved the resolved resonance parameters.

The resolved resonance region is being reanalaysed (in cooperation with the NEA Data Bank) taking into account the lessons learned in the work of the NEANDC Task Force on ^{238}U[1]. In particular the analyses are:

(a) using the improved resolution functions that have been shown to be necessary;

(b) applying renormalisation factors to the capture data of de Saussure et al.[2] above the 6.6 eV resonance

and (c) using shape analysis methods on both transmission and capture data

The new analyses use the code REFIT[3] and the work can be divided up into a number of stages:

(1) An evaluated list of neutron widths obtained from previous shape analyses of transmission data is produced; this adopts the energy scale of Olsen[4].

(2) These parameters are used, assuming that Γ_γ = 23.5 meV, to compute the shape of the capture yield using the version of REFIT which does a simplified multiple scattering correction. The shapes of measured and computed capture yields are then compared and the resonances not seen

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(1) H. Derrien, M. Moxon, Y. Nakajima, D. Olsen, F. Poortmans, G. de Saussure, M. Sowerby and B. Syme, Nuclear Data for Basic and Applied Science, Gordon and Breach, New York, Vol. 2 (1986) p.1511.

(2) G. de Saussure, E. G. Silver, R. B. Perez, R. Ingle and H. Weaver, Nucl. Sci. and Eng. 5 (1973) 385.

(3) M. C. Moxon, "Neutron Data of Structural Materials for Fast Reactors", EUR 6108 (1979) p.644 and Private Communication (1986).

(4) D. K. Olsen, Nucl. Sci. and Eng. 94 (1986) 102.

in transmission but visible in capture are identified; neutron widths are estimated and added to the evaluated list. Care is taken in all this to ensure that the estimated parameters are consistent with transmission data.

(3) Stage 2 is repeated until all possible small levels have been identified in the capture data.

(4) A simultaneous fit of the transmission and capture data is performed using the version of REFIT which does an exact multiple scattering correction and values of Γ_n and Γ_γ are obtained as appropriate.

The analysis is carried out in this way because of the need to minimise the cost of computing. The full shape analysis of capture yield data is, in particular, expensive in computer time. Such an analysis has not been performed before and is only possible with the code REFIT.

Table 1.1 lists the data being used in the analysis. At the present time Stage 3 is essentially complete below 4 keV. Above 4 keV where the work is being done in cooperation with Nakajima (NEA Data Bank) the analysis is being carried out in a number of energy ranges, starting with the lowest range. Stage 3 has now been reached for the lowest of these (4 to 6.3 keV). The analysis is only possible above 4 keV, and is significantly improved at lower energies, because Perez et al.[1] have sent us their high resolution capture cross-section data obtained using 3 different ^{238}U samples on a 150 m flight path at Oak Ridge. These data have not been used previously to obtain resonance parameters.

Above 1 keV some difficulties have been experienced in obtaining a set of parameters that is consistent with both transmission and capture cross-sections, which could imply that occasionally either Γ_γ deviates considerably from a constant value or more than one resonance of the same spin are very close together. Below 4 keV satisfactory compromises have been achieved but at higher energies, where p-wave resonances are stronger, some problems remain. In the energy range below 4 keV the present set of parameters includes data for 570 resonances. In comparison the ENDF/B-V and JENDL-2 evaluations have 442 and 416 resonances over the same energy range. Many of the

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(1) R. B. Perez, R. L. Macklin, G. de Saussure and R. W. Ingle, Private Communication (1985).

Table 1.1

Data being used in U-238 Resolved Resonance Analysis

Energy Range eV	0 - 900	900 - 4,000	4,000 - 10,000
Analyses used to obtain starter set of neutron widths	Olsen et al[3] Haste et al[4]	Olsen[2] Moxon[7] analysis of Olsen et al data[8]	Olsen[2] Moxon[7] analysis of Olsen et al data[8]
Energy scale	Olsen et al[3]	Olsen[2]	Olsen[2]
Capture data used to expand starter set of neutron widths	de Saussure et al[1] Moxon[9]	de Saussure et al[1] Perez et al[10]	Perez et al[10]
Data that will probably be used in final simultaneous fit to capture and transmission data	Olsen et al[3] (trans) Haste et al[4] (trans) de Saussure et al[1] ($\sigma_{n\gamma}$) Moxon[9] ($\sigma_{n\gamma}$)	Olsen et al[8] (trans) de Saussure et al[1] ($\Gamma_{n\gamma}$) Perez et al[10] ($\sigma_{n\gamma}$)	Olsen et al[8] (trans)

additional resonances are close to previously known ones and it has only been possible to identify them because of the ability of REFIT to do shape analyses of capture data. As an example consider the energy range 3680 to 3883 eV where the present analysis gives 28 resonances, 9 assumed s-wave and 19 assumed p-wave (41 resonances would be expected for a level spacing of 20 eV). Over the same energy range Olsen[2] fits the transmission data with 14 resonances. The resonances that are missing contribute 14% to the

(1) G. de Saussure, E. G. Silver, R. B. Perez, R. W. Ingle and H. Weaver, Nucl. Sci. and Eng. 5 (1973) 385.
(2) D. K. Olsen, Nucl. Sci. and Eng. 94 (1986) 102.
(3) D. K. Olsen, G. de Saussure, R. B. Perez, E. G. Silver, F. C. Difilippo, R. W. Ingle and H. Weaver, Nucl. Sci. and Eng. 62 (1977) 479.
(4) T. J. Haste, M. C. Moxon and J. E. Jolly, AERE-R 8666 (1979).
(5) F. Poortmans, L. Mewissen, G. Rohr, R. Shelley, T. Van de Veen, H. Weigmann, E. Cornelis and G. Vanpraet, INDC(NDS)-129/GJ (1981) 112.
(6) Y. Nakajima, Ann. Nuc. En. 7 (1980) 25.
(7) M. C. Moxon, Private Communication (1986).
(8) D. K. Olsen, G. de Saussure, R. B. Perez, F. C. Difillipo, R. W. Ingle and H. Weaver, Nucl. Sci. and Eng. 69 (1979) 202.
(9) M. C. Moxon, AERE-R 6074 (1969).
(10) R. B. Perez, R. L. Macklin, G. de Saussure and R. W. Ingle, Private Communication (1985).

infinitely dilute capture cross-section. In the ENDF/B-V and JENDL-2 evaluations each has 8 s-wave resonances and there are 8 and 10 p-wave resonances respectively. The resonances that are missed contribute 15 and 12% respectively to the infinitely dilute capture cross-section. In these evaluations this is made up by adding a smooth underlying cross-section which, if it is a large contribution, leads to incorrect self-screening factors being obtained in reactor physics calculations.

In the ^{238}U evaluation it is hoped to make use of the ^{238}U fission and capture cross-sections that are being evaluated as part of the ENDF/B-VI standard cross-section evaluation. We are at present involved in the phase 1 review of the standards evaluation. This has involved one of us attending a meeting of the CSEWG Standards Sub-committee in Oak Ridge and we are currently investigating the data used in the evaluation and the preliminary results and errors obtained. The errors are smaller than most people expect and a JEF Working Group has been set up to consider the problem.

1.2 Iron capture cross-section measurements on HELIOS (D. B. Gayther, J. E. Jolly, R. B. Thom* and C. A. Burke**) [WRENDA 356-360,396]

Work has continued with measurements on natural iron samples using the Total Energy Detector of the Maier-Leibnitz type. The principle of the detector and the problems encountered in its application were discussed in last year's report (UKNDC(86)P113, p.6).

A series of 14 experimental runs was made over a six week period in late 1986. The samples differed from previous ones in being a lamination of iron and gold sheet in the form, 0.3 mm Fe - 25 μm Au - 0.3 mm Fe. This arrangement is similar to that of Corvi et al.[1] and is used to provide absolute normalisation of the iron capture yield (capture events per incident neutron) by the 'saturated resonance' method, based on the 4.9 eV resonance in gold. This method relies on the fact that in a resonance with a large peak cross-section where the predominant interaction is neutron capture, the capture yield will approach unity near the peak, even when the sample is relatively thin.

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(1) F. Corvi, C. Bastian and K. Wisshak, Nucl. Sci. Eng. 93, (1986) 348.
* AEE Winfrith
** Imperial College

Calculation of the exact yield for the purpose of normalisation is thus insensitive to the precise values of the resonance parameters. Figure 1.1 shows a calculation of the capture yield for the present sample obtained with the code REFIT (M. C. Moxon, private communication) compared with the normalised experimental data. The time-of-flight measurements were made on the Fast Neutron Target at a flight path length of 42 m. The neutron energy range covered was 2.4 eV to 110 keV. The shape of the incident neutron spectrum was determined by replacing the iron/gold sample with a ^{10}B sample.

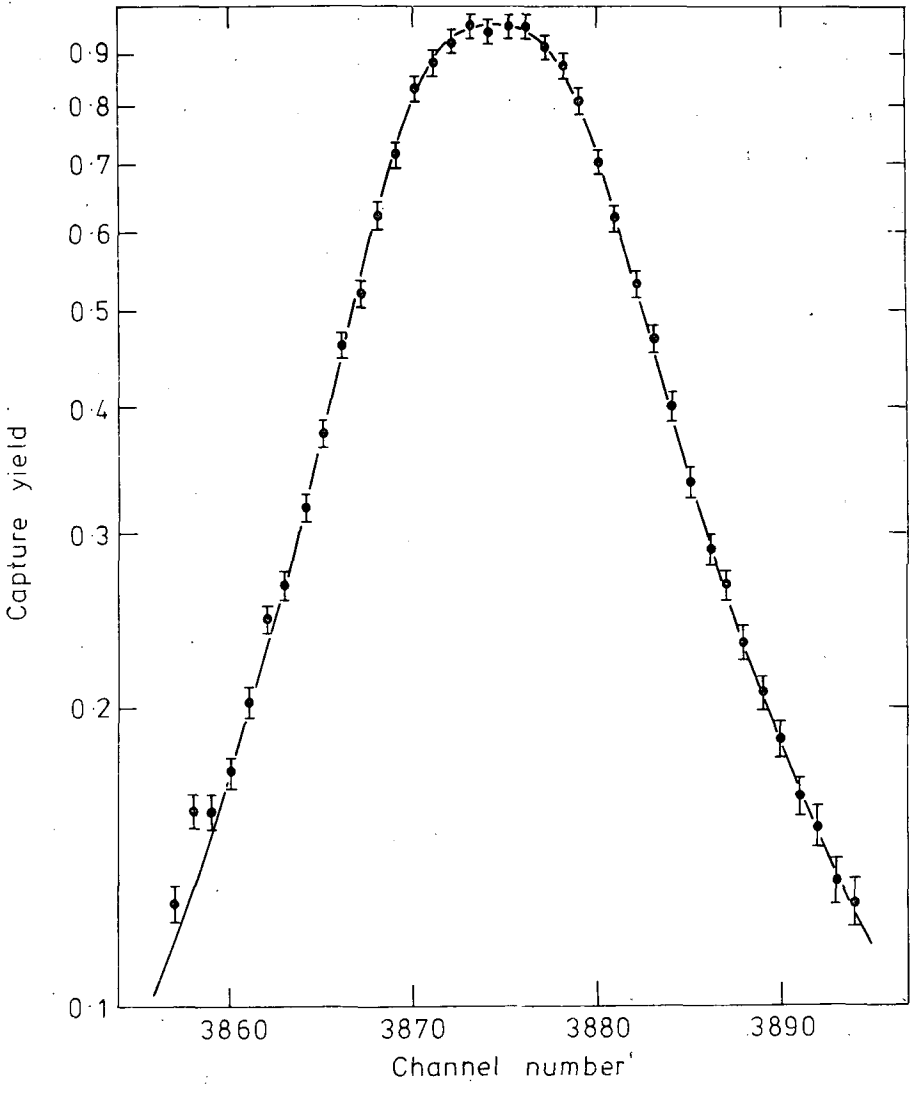

Figure 1.1 Comparison of calculated (continuous line) and experimental capture yields versus time-of-flight in the region of the 4.9 eV resonance in gold.

The main aim of these measurements is to determine the resonance parameters of the 1.15 keV resonance in ^{56}Fe. As reported previously, there are discrepancies between measurements of these parameters made with various capture detectors and values obtained from transmission data. Since the transmission values are considered to be reliable, doubt has thus been cast on the techniques used in the capture measurements. In particular, the technique used in the present measurement, where a calculated pulse amplitude weighting function (UKNDC(86)P113, p.6) is used to obtain the required detector response to gamma-rays, needs further validation.

A problem with the weighting technique has been that even the most comprehensive computer codes, which have been used to generate the weighting function, are unable to reproduce faithfully the observed response of the detector to the higher energy gamma-rays that are emitted in the capture process. At gamma-ray energies below a few MeV the calculated and observed spectra are in reasonable agreement. At higher energies, however, the calculated detector pulse amplitude spectrum shows fewer small amplitude pulses than are observed, the difference increasing with increasing gamma-ray energy. For a 6 MeV gamma-ray incident on the present detector, the number of small amplitude pulses is underpredicted by a a factor of ~2. In the absence of a physical explanation for this discrepancy it has been decided to derive an empirical weighting function based on experimental pulse amplitude spectra. Agreement between the calculated and observed spectrum for the 6.13 MeV gamma-rays emitted in the $^{19}F(p,\alpha\gamma)$ reaction, for example, can be obtained by applying an asymmetric resolution function to the calculated response. It is considered that by applying such a function with an asymmetry which depends on gamma-ray energy the calculated and observed spectra can be made to agree, and an empirical weighting function can thus be generated.

Development of an empirical weighting function requires considerable computational effort. In the meantime a preliminary analysis of the capture yield data for the 1.15 keV ^{56}Fe resonance has been made using the original calculated weighting function. At this stage it was considered that a comprehensive resonance analysis was not justified and a simple area analysis treatment has been made with the code CAREA (M. C. Moxon, private communication). The value obtained for the capture area, $g\Gamma_n\Gamma_\gamma/\Gamma$, is 59.4±6.0 meV. The quoted uncertainty is dominated by the estimated uncertainty

in the weighting function. This result is to be compared with the capture area for this resonance of 57.8±0.7 meV obtained by neutron transmission measurements on HELIOS (G. D. James and M. C. Moxon, private communication).

1.3 Determination of resonance parameters for natural iron from neutron transmission data (G. D. James and M. C. Moxon) [WRENDA 356-358]

The neutron transmission measurements for natural iron carried out on a 150 m flight path of the time-of-flight spectrometer at HELIOS have been analysed using REFIT, running on a CRAY 1 XMP accessed through an IBM 3084Q, to determine the parameters of the resonance at 1.15 keV and of the s-wave resonances below 317 keV. Two thicknesses of iron, 2 mm and 15 mm, were used in the experiments and two thicknesses of copper, 3 mm and 6 mm, were used to determine the background at 0.577 keV and 2.04 keV. Transmissions were calculated with copper in and out of the neutron beam and resulted in twelve transmission data sets. Analysis of the resonance at 1.15 keV was carried out in a sequence of steps designed to reveal the difference between transmission data sets with and without copper in the beam, and between analyses over a narrow energy range near 1.15 keV and analyses over wider energy ranges extending up to 50 keV. One of these, for the simultaneous analysis of twelve data sets and involving 29 free parameters, reached array limitations in the programme. These limits have been extended and the final analysis is in progress.

1.4 FISPIN calculations of fuel reprocessing (R. A. Forrest and D. A. J. Endacott)

A series of calculations has been made using the inventory code FISPIN in support of several Authority programmes. Recycling of Pu and higher actinides in both fast and thermal reactors, the effect of storage of Pu prior to fabrication and changes in data libraries to investigate specific fission products are typical of the work undertaken.

1.5 The $^{93}Nb(n,n')^{93m}Nb$ reaction (D. B. Gayther, J. E. Jolly, C. A. Uttley, M. F. Murphy* and K. Randle**) [WRENDA 563-569,571-572,582]

In the period since the last progress report, one further measurement of the cross-section for the production of the 30 keV isomer in ^{93}Nb has been made.

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** Department of Chemistry, University of Birmingham, UK

This was at 2.7 MeV neutron energy using the 6 MV Van de Graaff accelerator at Harwell as the source. It was decided not to make a further attempt to measure the cross-section at 600 keV because of difficulty in manufacturing adequate lithium targets (UKNDC(86)P113, p.16) and the present experimental programme has thus been completed. This work is described in Harwell Report AERE-R 12612 (to be published) which has the following title and abstract:

The $^{93}Nb(n,n')^{93m}Nb$ cross-section

D. B. Gayther, M. F. Murphy, K. Randle, W. H. Taylor, C. A. Uttley

Abstract Measurements of the 30 keV isomer production cross-section in ^{93}Nb excited by neutron inelastic scattering have been made in the neutron energy range 1 to 6 MeV. Small foils of niobium were exposed to high fluences of 'monoenergetic' neutrons and the cross-section was determined from the K X-ray activities so induced. The neutron fluence was measured with a low efficiency fission chamber in which the fissile deposit is located immediately behind the niobium foil. Comparisons are made with nuclear model calculations of the cross-section and recommended values based on this and the experimental data are presented.

Figure 1.2, taken from this report, shows the final cross-section values in comparison with cross-sections deduced from $(n,n'\gamma)$ measurements[1,2,3]. Also shown is the theoretical cross-section of Strohmaier et al[4]. It can be seen that the present data do not confirm the broad peak in the theoretical curve in the region of 2.8 MeV.

1.6 The measurement of eta for ^{235}U in the neutron energy region below 1 eV (M. C. Moxon) [WRENDA 964-967]

1.6.1 History of the measurement

The measurement was made by observing the fast neutrons produced in a thick sample of ^{235}U as a function of neutron energy using a 10 metre flight path on the Condensed Matter Target of HELIOS. (In practice measurements were carried out with several samples of different thickness to check the thickness dependence of various corrections which have to be applied to the raw data.)

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(1) I. J. van Heerden and W. R. McMurray, Z. Physik, 260, (1973) 9.
(2) G. H. Williams, Thesis, University of Texas (1975).
(3) H. Göbel, E. J. Feicht and H. Vonach, Z. Physik, 240, (1970) 430.
(4) B. Strohmaier et al., Physics Data Series 13-2 (1980) 62.

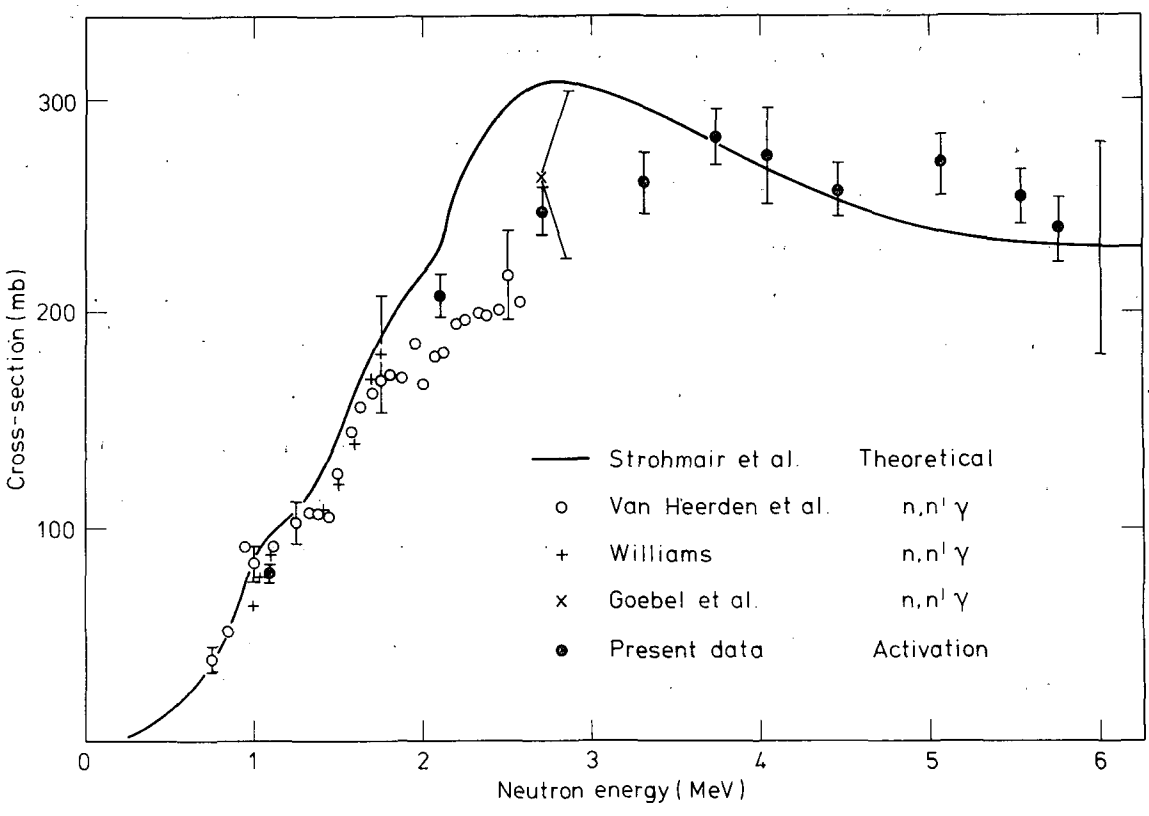

Figure 1.2 Comparison of evaluated $^{93}Nb(n,n')^{93m}Nb$ cross-section with experimental data.

The detector used was an NE 213 liquid scintillator with a pulse shape discrimination unit. Neutron, gamma-ray and pile-up signals were recorded as a function of time-of-flight. The background was determined using the 'black resonance' technique i.e. samples of materials with large resonances are placed in the neutron beam. The background is determined from the minima in the transmitted neutron beam around the resonance energy. The attenuation of the background due to the presence of the filters in the beam is determined by measurements on two different thicknesses of filter.

The initial measurements carried out in December 1984 (UKNDC(85)P112, p.35) to determine the energy dependence of eta for ^{235}U in the energy region below

1 eV indicated signal to background ratios of several hundred to 1 in the region down to ~20 meV and better than 5 to 1 at 3 meV. In early 1985 the tantalum target of the Condensed Matter Cell of HELIOS was replaced by one made from natural uranium. The measurements undertaken in July and August 1985 (UKNDC(86)P113, p.3) gave a much poorer signal to background ratio. Tests and measurements carried out on the equipment indicated that the additional background was due to neutrons coming from the neutron source. In late 1985 and early 1986 measurements on samples of lead showed that fast neutrons were coming from the target with delays of 100 μsec and greater. These neutrons were not present in a similar measurement carried out on the tantalum target. This additional background could be fitted with the sum of three exponentials (see Fig. 1.3), the main component having a half life of 30 μsec. This is in good agreement with the half life for thermal neutrons in a 40 mm thick slab of hydrogenous moderator. Thus the additional background was thought to be due to the return of thermal neutrons to the natural uranium target causing fission in the ^{235}U. The fast neutrons from these fissions give rise to an exponential tail ($t_{1/2}$ ~30 μsec) to the resolution function. The source of the background has been confirmed in measurements carried out in January 1987 with the moderators of the target decoupled from the uranium target by sheets of cadmium, when it was reduced by a factor of three.

The shape and magnitude of the background is now being redetermined and further analysis will be carried out on the data.

1.6.2 Measurement of eta

The shape of the neutron spectrum has been determined from measurements on a ^{10}B slab and ^{10}B parallel plate ion chamber placed in the neutron beam.

Figure 1.4 shows the time dependence of the incident neutron spectrum as a function of neutron time-of-flight, together with the observed counts and background from the ^{10}B slab measurement. In the energy region below ~20 meV the shape was determined mainly from the ^{10}B ion chamber data which had a signal to background ratio of 10 to 1 at 3 meV and extended to a neutron energy of 1.4 meV. In the low energy region the background is the main source of uncertainty and has been determined to accuracies of 1% and 5% for the ^{10}B slab and ion chamber measurements respectively.

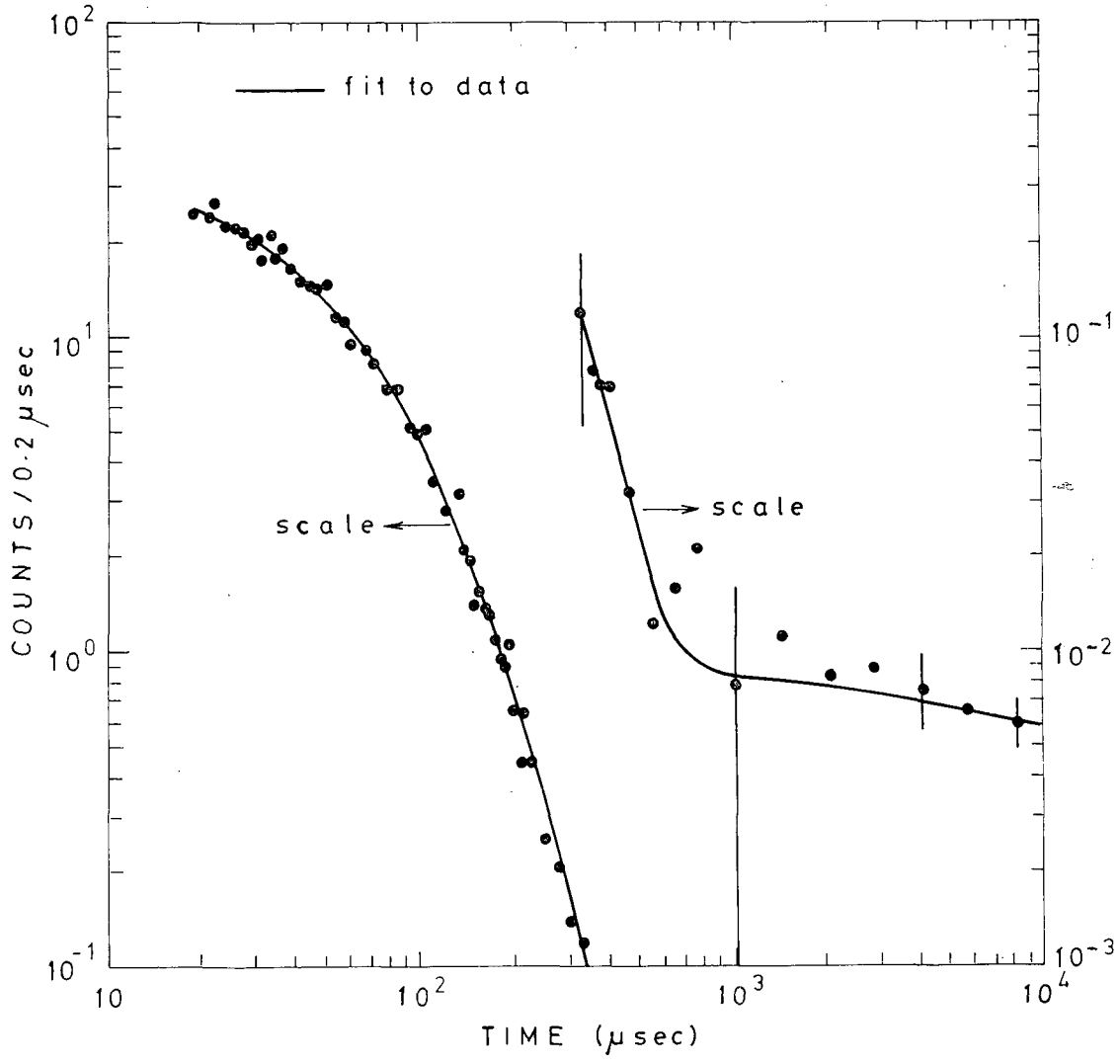

Figure 1.3 Least squares fit to the fast neutrons scattered by a lead sample into the detector. The fitted curve is

$$B(t) = 0.332\times10^{-2} + 36.36\,(1-\exp(-t))\exp(-0.02385\,t) + 3.686\,(1-\exp(-t))\exp(-0.01097\,t) + 0.00329\,(1-\exp(-t))\exp(0.000204\,t),$$ where t is in μs.

The time-of-flight spectra from the ^{235}U samples are corrected for count loss and signal pile-up effects. The background was determined using the 'black resonance' technique and subtracted from the observed counts. The resultant

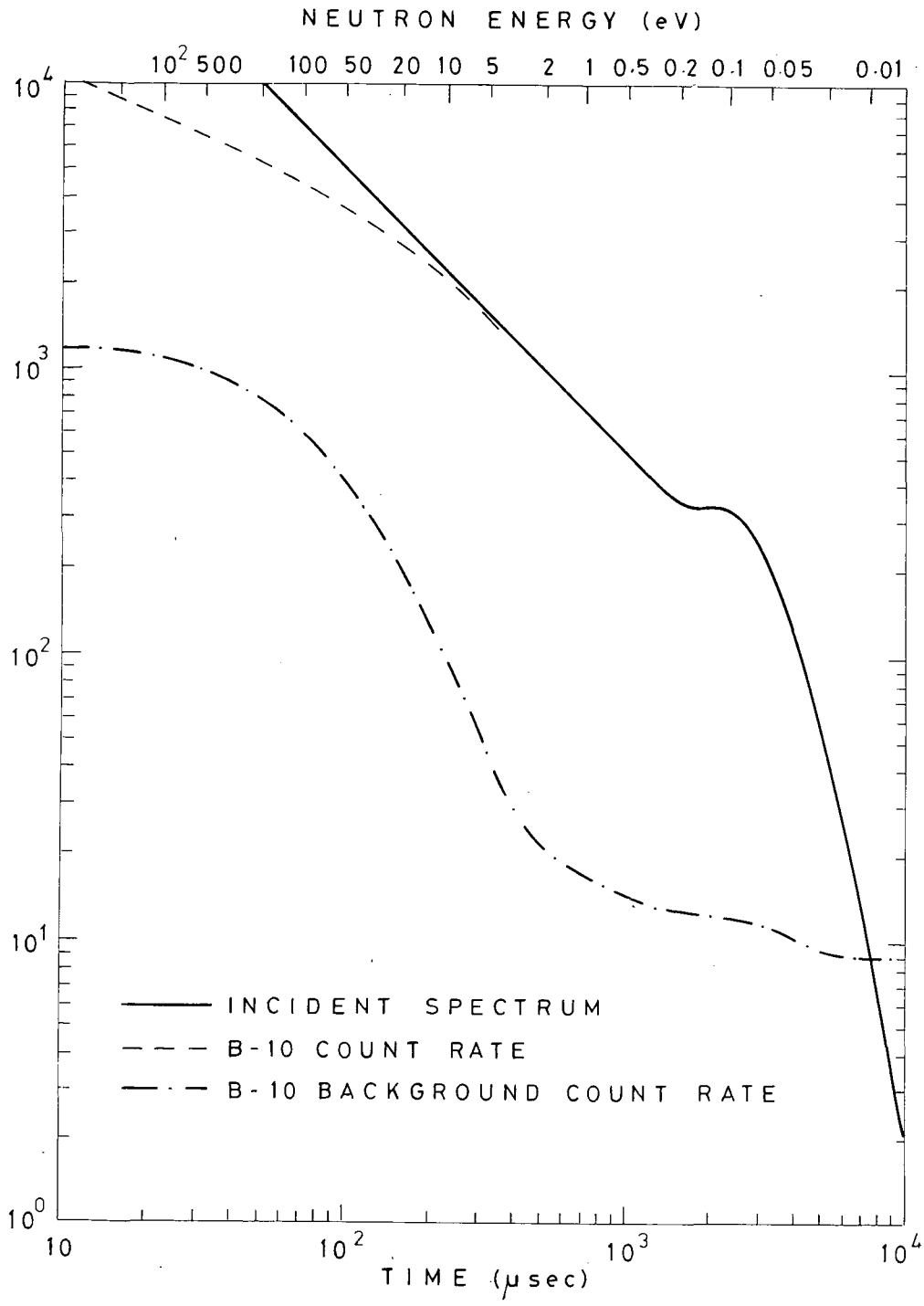

Figure 1.4 The incident neutron spectrum calculated from the ^{10}B slab and ion chamber measurements. Also shown are the observed counts from the ^{10}B slab and its background.

counts are then divided by the incident spectrum to give a relative fission neutron yield per incident neutron. The yield for the thickest ^{235}U sample is shown in Figure 1.5. The statistical errors on these data are generally less than ½% (one standard deviation). Uncertainties due to the errors in the background both for the ^{235}U and the incident neutron spectrum measurement are 2% at 50 eV decreasing down to a constant value of 0.17% between 0.5 and 0.02 eV. Below this energy the error due to the background increases with decreasing energy to a value of ~0.65% at 10 meV.

The energy dependence of eta can be determined directly from the thick sample yield data. Small corrections for the transmitted and scattered neutrons have been calculated using the cross-sections in the JEF-1 data library and assuming the angular dependence of the scattering cross-section to be isotropic. The measured energy dependence of eta is shown in Figure 1.6 compared with presently accepted evaluations. An anisotropic scattering angular distribution due to solid state effects may account for the larger than expected observed spread of the data points especially in the region of 14 meV. These effects are under investigation at present.

During the measurement of eta for ^{235}U many subsidiary measurements, listed in Table 1.2, were carried out to check the validity of the eta results, e.g. background measurement, spectrum checks and the measurement of total cross-sections for corrections. These data, which need further analysis, should yield as good if not better nuclear data than exist at present in addition to their main purpose.

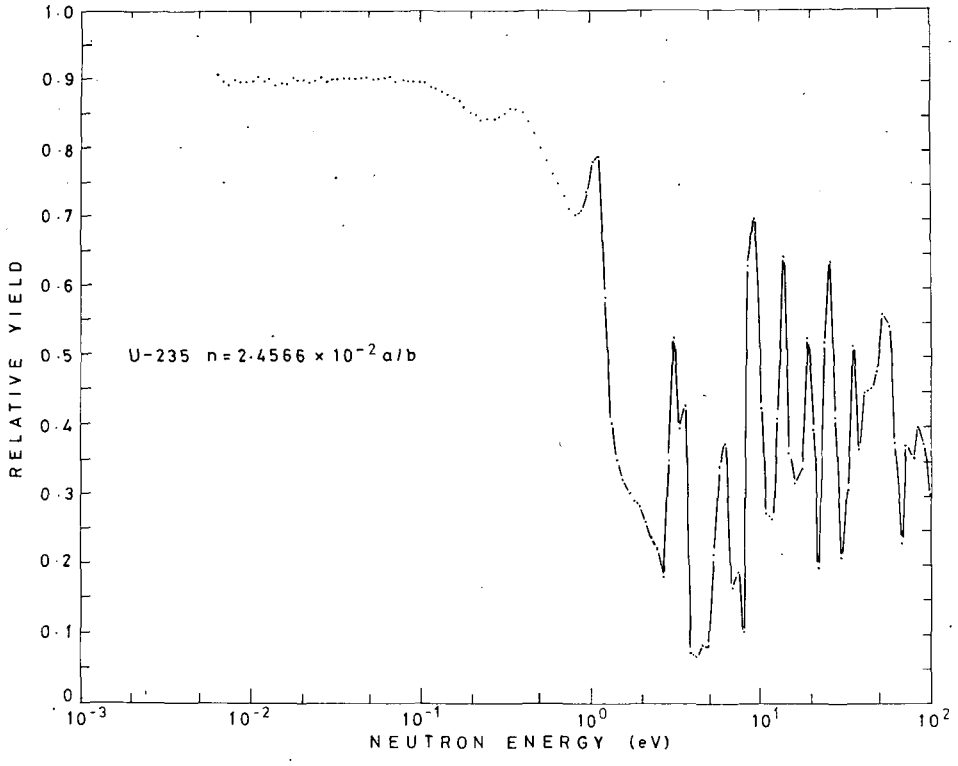

Figure 1.5 The measured ratio of the neutron output from the PSD unit to the incident neutron spectrum for the thickest ^{235}U sample with n = 2.4566 x 10^{-2} atoms per barn.

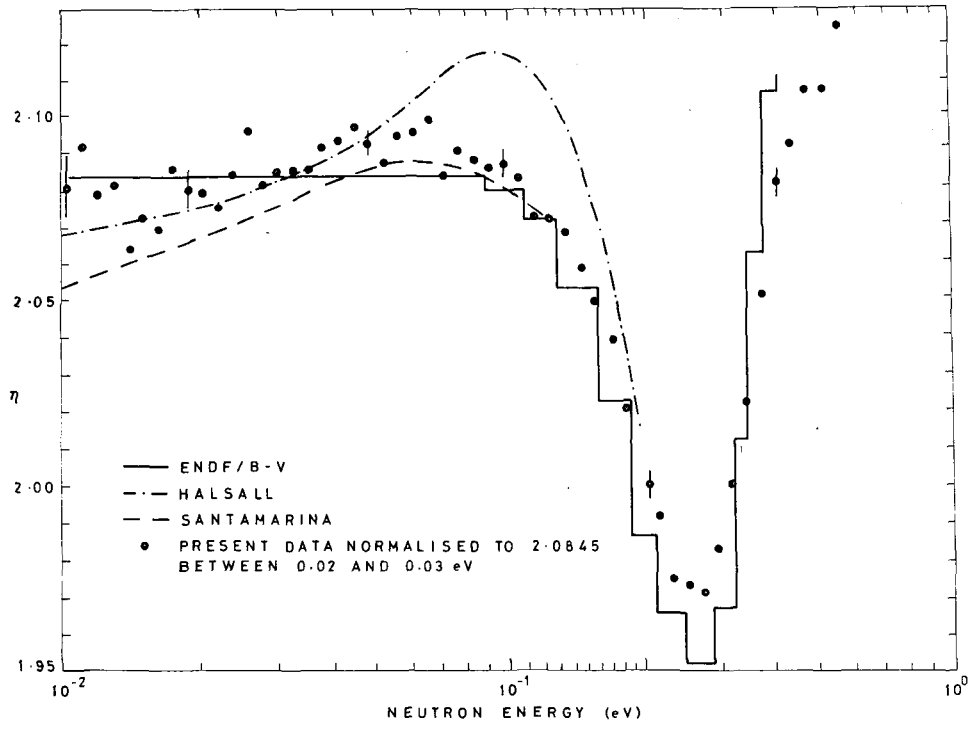

Figure 1.6 The energy dependence of eta calculated from the data of figure 1.5.

Table 1.2

1.	^{235}U	the shape of the fission cross-section and alpha from ~0.01 to ~10 eV
2.	Au	capture cross-section from 0.01 eV to ~20 eV
3.	Cd	capture cross-section from 0.2 to 20 eV
4.	Ta	resonance parameters for first 3 resonances
5.	Er	resonance parameters for first 2 resonances
6.	In	resonance parameters for first 3 resonances
7.	Ag	resonance parameters for first resonance
8.	^{235}U	total cross-section from 0.006 to ~10 eV to an accuracy of ~1/3 %
9.	Pb	possible measure of the energy dependence of the capture cross-section

Reason for measurement:

1	The gamma-ray output from the PSD unit was recorded at the same time as the neutron output. Both can be used to derive values of the relevant cross-section.
2 and 3	Checks on the shape of the incident neutron spectrum.
4,5,6,7	Samples of these elements were used to measure the background.
8	The total cross-section was measured to check the values given in the JEF data files as well as to check the isotopic composition of the samples.
9	Two lead samples were used to measure the background and it may be possible to determine the energy dependence of its capture cross-section from the gamma-ray output data.

1.7 Activation of fusion reactors (R. A. Forrest, M. G. Sowerby, B. H. Patrick and D. A. J. Endacott)

Work has continued over the past year to improve the nuclear data required for calculations of induced activity in fusion reactor materials exposed to a flux of fusion neutrons. This has involved collaboration with Culham Laboratory (Dr. L. Giancarli), the University of Oxford (Dr. P. E. Hodgson), ECN Petten (Dr. H. Gruppelaar) and Imperial College (Prof. A. Goddard).

A library of cross-section data has been produced by Petten based on the work of Mann at Hanford, and including some UK input. This still requires some additions to ensure that all reactions for all nuclides with half lives greater than 1 day are included. The missing data will be generated in the first instance by THRESH[1]. In addition to cross-sections, data on the half lives, decay modes and average energies of all relevant nuclides must also be assembled. Of the approximately 1200 nuclides for which information is required about 850 are covered by the JEF-1 library. The remainder will have ENDF/B-V format files created specially for this work and this is in progress.

(1) L. Giancarli and H. Gruppelaar, UKAEA Culham Report CLM-R 261.

A change from the earlier work involving UKCTRIIIA[1] is the use of the inventory code FISPIN[2] rather than ORIGEN. One advantage of the former is that sensitivities of the amounts of activation products to uncertainties in cross-sections can be calculated. This will mean that the most important data can then be improved by use of the code CADE[3] or better evaluations. Development of an existing sensitivity version of FISPIN has been carried out at Imperial College.

Improvements to the nuclear systematics used in the code THRESH have been made by analysis of all the experimental data on various threshold reactions up to 1985. This work is described in a recent Harwell report[4], which also gives details of likely errors for the various types of reaction; these are needed for the sensitivity calculations.

Once the cross-section and decay data have been correctly assembled and made accessible to FISPIN, preliminary calculation of particular activation products will be undertaken. However, it is already known that some experimental measurements will be necessary to back up the theoretical calculations. An irradiation in the high flux source RTNSII at Livermore is planned to give cross-sections for reactions in the materials Hf, Ta and W. These involve production of isomeric states for which theory and systematics are either very difficult or lacking.

A summary[5] of the status of this work at the end of 1986, including more details of both the theory and the current problems, was given at the IAEA Advisory Group Meeting on Nuclear Data for Fusion Reactor Technology.

1.8 Nuclear materials assay

Good measurement of nuclear material quantity is essential throughout the nuclear fuel cycle for reasons of accountancy, criticality control, plant performance monitoring, fuel performance, safeguards and non proliferation and nuclear waste management. In Neutron Systems Group there are current programmes of research on non-destructive assay of fissile materials by different methods

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(1) O. N. Jarvis, UKAEA Report AERE-R 9601 (Rev) (1980).
(2) R. F. Burstall, UKAEA Report ND-R-328(R) (1979).
(3) D. Wilmore, UKAEA Report AERE-R 11515 (1984).
(4) R. A. Forrest, UKAEA Report AERE-R 12419 (1986).
(5) R. A. Forrest, IAEA Report on Meeting at Gaussig GDR, Dec. 1986, DIDSG(87)P357.

of pulsed neutron interrogation and also by passive observation of spontaneously emitted neutrons and gamma radiation. Progress on these is described below.

1.8.1 Matrix effects in the neutron die-away assay of fissile material in 200ℓ drums (B. H. Armitage and A. C. Sherwood)

A sensitive means of measuring the fissile content of waste is provided by the neutron die-away technique. This technique is based on a pulsed 14 MeV neutron source inside a graphite and polyethylene chamber, which produces an interrogating thermal flux within the enclosed space. As the thermal neutron flux dies away fission is induced in any fissile material present, and the ensuing fast fission neutrons are counted by fast neutron detectors embedded in the walls of the chamber. This method may be used to assay contaminated material present in waste drums. To be able to assay fissile material in matrix-filled drums by the neutron die-away technique it is necessary to define a procedure which does not require a prior knowledge of the matrix.

A study based on the use of eighteen different matrices with a 200ℓ capacity neutron die-away chamber at Harwell has been described in previous reports. The limitation of this work, however was that the measurements of fissile material response were confined to fissile material at the centre of the drum. As a considerable range of fission neutron response was observed, an empirical approach to matrix compensation was made by dividing the matrices into four different categories. This method led to the compensated fission neutron response for seventeen of the eighteen different matrices varying over the range 0.35 to 1.01 with a standard deviation of 15%.

The eighteen matrix study has now been extended to include variations in the fission neutron response due to the presence of fissile material at any point in the drum. In addition to monitoring the interrogating thermal neutron flux by a bare BF_3 detector attached to a wall of the chamber, a bare BF_3 monitor was also located in direct contact with the underside of the drum. This detector is referred to as the external matrix monitor (EMM), as a Cd shield placed below the detector ensured that it was sensitive only to the thermal neutrons emerging from the matrix. The spatial variability in response was measured at six locations in each of the drums. The results were expressed in the form $\bar{R}(0.4\text{-}5)$: the volume weighted fissile material response from each drum integrated over the time interval 0.4 to 5 ms after each pulse from the

14 MeV pulsed neutron source. The values obtained (Table 1.3) for seventeen of these matrices cover a numerical range from 0.68 to 3.05 with a standard deviation of 34%. Matrix 18, which consists of 226 kg iron and 53 kg water, has been omitted as the spatial variability in response is much greater than for any other of the matrices.

In order to obtain an approximate value $\bar{R}(0-5)$, the mean fissile material response integrated over the interval 0 to 5 ms after the pulse, use was made of the EMM. This approach was justified by the observation that the lifetime of thermal neutrons observed in the EMM was approximately equal to that observed with the fast neutron detector. On this basis $\bar{R}(0-5)$ was obtained from

$$\bar{R}(0-5) = \bar{R}(0.4-5).EMM(0-5)/EMM(0.4-5).$$

The resulting values for $\bar{R}(0-5)$ were found to have a reduced standard deviation (23%) significantly lower than the $\bar{R}(0.4-5)$ standard deviation.

Finally, a modest improvement in standard deviation was obtained by taking into account inferred values for the detection efficiency E for fission neutrons emitted from fissile material at the centre of the drum. Such a value for E can be obtained for an unknown drum by measuring the yield at the detectors of a Cf spontaneous fission source placed immediately beneath the drum, and from appropriate calibration data. The inclusion of a term depending on E was approached empirically, with the result that the response is given as $\bar{R}(0-5)E^{-0.2}$ in column 4 of Table 1.3.

The matrix categories shown in the Table are based on high/low absorption and high/low moderation. Drums can be placed in these categories on the basis of measured values of thermal neutron lifetimes as observed with the wall mounted thermal monitor and the EMM. The justification for this procedure is that if an unknown drum can be placed in one of these categories then a more directly relevant value for the standard deviation can be given. A more detailed account of this work is in course of preparation.

Table 1.3

Matrix Number	$\bar{R}(0.4\text{-}5)$	$\bar{R}(0\text{-}5)$	$\bar{R}(0\text{-}5)E^{-0.2}$	Category	$\bar{x}$	Sigma
1	1.00	1.00	1.00			
2	0.97	1.05	1.05			
3	1.00	1.13	1.13	1	1.08	10%
4	1.01	1.23	1.19			
5	0.96	1.02	1.02			
6	0.84	1.02	1.01			
7	0.76	1.05	1.04	2	1.06	10%
8	0.75	1.14	1.13			
9	0.73	1.06	1.06			
10	0.91	1.21	1.25			
11	0.61	0.98	1.15	3	1.09	26%
12	0.64	0.99	1.03			
13	0.46	0.72	0.92			
14	0.35	0.64	0.98			
15	0.44	0.91	1.05			
16	0.41	1.01	1.14	4	1.16	28%
17	0.51	1.23	1.29			
$\bar{x}$	0.73	1.02	1.09			
Sigma	34%	23%	19%			

1.8.2 A pulsed neutron interrogation chamber for 500ℓ drums (B. H. Armitage)

In order to extend studies of pulsed neutron interrogation, an assay chamber able to accommodate a 500ℓ drum has been constructed at Harwell. Its primary purpose is to study the assay of hard waste by the delayed neutron technique using a 14 MeV pulsed neutron source. As the same source and chamber can be used in die-away mode it will no doubt prove as adaptable in operation as the existing 200ℓ assay chamber.

The interior dimensions of the cell are 1.60 m height, 1.03 m width and 1.05 m depth. The inside graphite wall has a thickness of 155 mm and the outer polythene wall is 54 mm in thickness. A horizontally sliding door enables the drum to be introduced into the chamber.

Fast neutron detector packages are embedded into each of the four vertical walls of the chamber. Each package consists of a single 51 mm dia. 0.63 m

effective length ^{3}He tube filled to 2 atm. with ^{3}He. The tube is encased in a 13 mm thick cylinder of polyethylene covered with 2.5 mm Cd. Although provision has been made in each wall for three detector packages, only two detector packages of less than optimum size are currently available. These are borrowed from the 200ℓ assay chamber for the duration of work on the 500ℓ one.

By also sharing the pulsed neutron source, electronics and data processing systems with the existing 200ℓ chamber, the cost of the new 500ℓ system has been kept to that of the chamber. This was constructed in less than four months and at minimum cost using existing stocks of machined graphite blocks.

1.8.3 Differentiation between 500ℓ drums of radioactive waste at the 10mCi level of alpha activity by active neutron interrogation (B. H. Armitage and M. J. Cogbill)

The 500ℓ interrogation chamber has been used to establish whether active neutron techniques can determine which 500ℓ waste drums can be classified as low level waste. More specifically the requirement was to provide an assurance that 500ℓ drums of encapsulated and uncapsulated waste contain less than 10mCi alpha activity. Thus a fail safe approach is needed, but without the necessity of a quantitative measurement.

Differential die-away measurements have been made with three 500ℓ drums containing fissile samples (^{235}U). The first of these drums was matrix free, the second contained steel shot at a bulk density of 1.5 gcm^{-3}, while the third was filled with plain grout.

The fissile material response was measured at a minimum of six positions in each drum. As expected the grout-filled drum demonstrated the greatest spatial variability with the response at the centre being a factor of about a hundred less than at the surface. Significant spatial variability was also observed with the simulated hard waste matrix where the response at the centre was about six times less than at the surface (and also six times less than the volume average). The behaviour of the hard waste drum may, however, be dependent on the heterogeneity of the matrix.

Table 1.4

Matrix	None	Steel shot	Grout#
Lower limit of detection (mg ^{235}U)	3	30	10,000

water to solid ratio of 0.32

Lower limits of detection were obtained for fissile samples at the centre of each drum, using eight detectors instead of the designed array of twelve larger volume detectors. These lower limits, shown in Table 1.4, are for standardized assay conditions of 5.10^6 interrogating neutrons per pulse, 10 pulses per second, 1000 second assay time and twelve fast neutron detector packages.

From a knowledge of specific alpha activity (Ci g^{-1}) in the relevant isotopes, the mass of any given fuel composition at the 10 mCi level can be determined. If waste is known to contain only LEU, the limit is provided by the ^{234}U content which is $7.64.10^{-3}$ times that of ^{235}U. This means that a 500ℓ drum can contain 209 g of ^{235}U at the 10 mCi level. Here we also take 209 g of ^{235}U as the limit for HEU.

In Table 1.5 lower limits of detection in mCi per 500ℓ are given for two Pu-bearing fuels, CAGR fuel, LEU and HEU, on the assumption that the fissile material is in dispersed form and not in the form of lumps. From this table it may be concluded that differentiation at the 10 mCi level is achievable for unencapsulated dense hard wastes.

The situation for grout encapsulated wastes appears to be less satisfactory, in that differentiation is possible for HEU and LEU but not for the other wastes.

Table 1.5

Lower limit of detection (mCi per 500ℓ)

Waste	Steel shot	Grout
LEU,HEU	1.4×10^{-3}	0.5
Pu-bearing fuel 1	2.8	1,000
Pu-bearing fuel 2	8.7	3,000
Low burn-up CAGR	2.4	800
High burn-up	8.7	3,000

A fail safe aspect of this approach for encapsulated drums is that the presence of fissile material close to the surface may result in severe over estimation of the alpha-activity actually present in the drum.

Self-shielding of thermal neutrons in lumps of fissile material is an inherent problem in neutron interrogation. In the 'worst case', all the fissile material is present at the centre of drum in the form of a single spherical lump. Here differentiation in the dense hard wastes should be possible for LEU and HEU and for low burn-up CAGR fuel. As far as grout encapsulated wastes are concerned, differentiation should be possible for LEU and HEU in dispersed form, and for LEU in the 'worst case' situation, but not for plutonium wastes.

1.8.4 Pulsed neutron source development (J. W. Leake* and B. H. Armitage)

The development programme with GEC Avionics described in last year's report (UKNDC(86)P113, p.28) has been subject to some delay but is now nearing a conclusion. The 14 MeV source equipment comprises a sealed tube and canister within which deuterons are accelerated onto a ^{3}H target, along with a separate high voltage transformer and a control unit. An improved neutron source will soon be available for delivery to Harwell, while a new version of the neutron tube and its canister (outside the scope of the development contract) has already been delivered to Sellafield. More recently it has become apparent that the high voltage transformer is subject to premature failure, although at the commencement of the development contract there was no evidence that the

*I & AP Division

transformer was a source of weakness. As a result a larger heavy-duty transformer is to be incorporated into the new neutron source equipment.

The main thrust of the programme has been in improving the reliability and longevity of the sealed neutron tube. In the early tube tests, a build-up of gas pressure was the principal mode of failure. This appears to have been overcome by decreasing the ^{3}H content. Lifetime tests on sealed tubes incorporating a new target material, low sputter materials, and a suppressor have been undertaken. As a number of tubes have failed due to overheating of the suppressor electrode, the final phase of the work will be concerned with overcoming this particular problem.

1.8.5 Development of californium shuffler methods for delayed neutron interrogation (N. P. Hawkes, G. B. Huxtable, T. W. Packer and M. T. Swinhoe)

Thermal neutron interrogation with a Cf shuffler gives an enhanced sensitivity to fissile material compared with fast neutron interrogation, but is subject to problems of self-shielding and matrix effects associated with it. Therefore we have been investigating the use of epi-cadmium interrogation, in which the sample is shielded from externally generated thermal flux by a cadmium sheet. We have used the existing pneumatic shuffler to make measurements on uranium metal and oxide samples. A modest amount of source tailoring, in the form of a block of iron 6.5 cm thick, was added between the Cf source irradiation position and the drum to reduce the contribution of ^{238}U fission to the overall response. The results indicate that for an empty drum the ratio of the response per gram of ^{235}U to ^{238}U is 36±7. The result implies that for natural uranium, 80% of the response would be due to ^{238}U. There is a factor of 4 difference in response between the centre and ends of the empty drum. MORSE calculations for the same system gave a result of 38±19, in good agreement. The calculation gives the self-shielding for a 100 g lump of ^{235}U as 0.35 (i.e. it would give the same signal as 35 g of dispersed material).

These results indicate that more tailoring of the spectrum is required to reduce ^{238}U contribution and that a more uniform assay of the drum is desirable.

Our plan is to install a Teleflex cable source movement mechanism similar to that used at Los Alamos. This should allow source movement during the

irradiation to give a more uniform assay. This system, as presently envisaged, should provide a reliable experimental facility which will allow experiments on source tailoring and matrix effects and allow comparisons between differential-die-away and shuffler systems under similar conditions.

Calculational work is beginning on the new shuffler design using the LANL program MCNP rather than MORSE.

1.8.6 Optimisation of timing parameters for a californium shuffler (N. P. Hawkes and R. Wyatt*)

A shuffler system for fissile assay operates by bringing up a neutron source to the sample under test, leaving it there for a short period (the 'irradiation') to induce fissions, then withdrawing the source and counting delayed fission neutrons for another period (the 'counting time'). The whole cycle can be repeated many times, and delayed neutron activity can build up from one cycle to the next.

The aim of the work reported here was to find the optimum values of the irradiation and counting times, and to evaluate the effect of reducing the source transfer speed (as would happen if the pneumatic source transfer system in the existing shuffler were replaced with a mechanical system - see Section 1.8.5 of this Report).

The formula of Binney and Scherpelz (1978)[1], which gives the number of delayed neutrons counted as a function of the timing parameters, was modified to allow for the fact that the source cannot be moved instantaneously. A calibration factor was also included so that the formula agreed with actual measurements from the shuffler for the most frequently used set of timing parameters. The formula could then be used to calculate how the quality $m/\Delta m$ of the measurement varies with irradiation and counting times, where m is the mass of fissile material deduced from the measurement and Δm is the statistical uncertainty in m. The detection limit for the system corresponds to a $m/\Delta m$ value of about 3.

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(1) S. E. Binney and R. I. Scherpelz, Nucl. Instrum. Meth. 154 (1978) 413.

* Present address: Physics Department, University of Bath

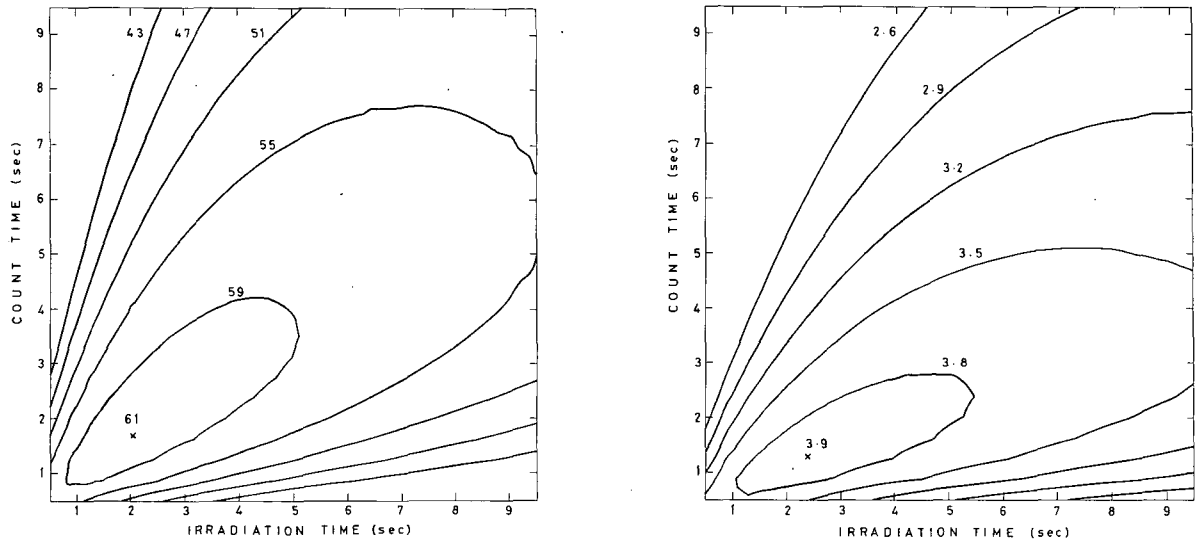
A convenient way of displaying the results is as a contour plot of $m/\Delta m$. Figure 1.7a shows such a plot for a fissile mass much greater than the detection limit of the device. The $m/\Delta m$ surface is quite gently curved, showing that the timing parameters are not highly critical. Figure 1.7b shows the corresponding plot for a mass close to the detection limit; again, the curvature is not severe. The optimisation point has moved slightly to a longer irradiation time and a shorter counting time. Figure 1.7c shows the effect, for the small mass of Figure 1.7b, of degrading the source transfer time from 0.2 s to 0.8 s. Even with the slower speed there is still a large region for which $m/\Delta m$ exceeds 3, ie the slower system would still be able to detect the material.

The conclusion from this work was that changing to a mechanical transfer system - which has significant advantages of reliability and ease of control over the pneumatic system - would not seriously degrade the quality of measurements.

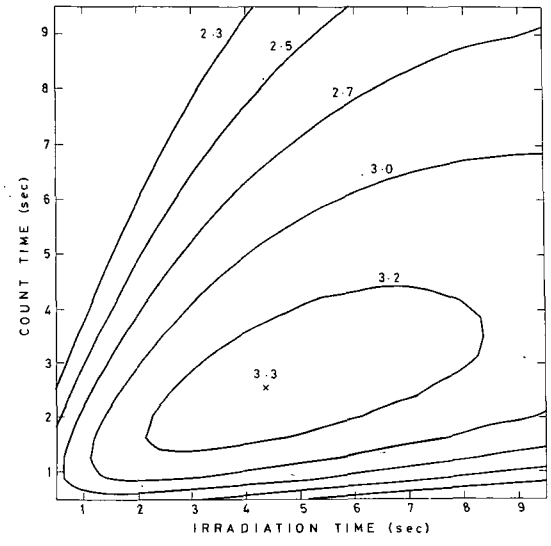
1.8.7 The effects of (α,n) production on neutron coincidence counter measurements (M. T. Swinhoe)

Work has continued on understanding the behaviour of neutron coincidence measurements. Extension of the method to a wider range of materials has required the calculation of the relevant (α,n) yields e.g. for nitrate solutions. Also Monte Carlo calculations of various sample types have been carried out to determine the relative importance of the effects of water: increased multiplication, change in detection efficiency and increase in (α,n) production. The preliminary conclusion is that the change in efficiency can be ignored compared with the other two effects, and this allows measurements with plutonium nitrate solutions and moist plutonium oxide samples to be made with the same calibration curve as plutonium metal or dry plutonium oxide samples, provided the moisture content of the plutonium oxide is known. A report on this work is in preparation.

Matthes, of JRC Ispra, has proposed a variation in the use of neutron coincidence counters which involves a change in reflectivity of the cavity to give information on the multiplication occurring in the sample. Some Monte Carlo estimates of the expected performance of this technique have shown that the change in total counting rate in the different geometries is small but significant and that the method may lead to useful results. This needs to be



(a) (b)



(c)

Figure 1.7 Contour plot of the calculated quality of measurement $m/\Delta m$, as a function of irradiation and counting time, (a) for a large fissile mass, (b) for a small mass close to the detection limit, (c) showing the effect of degrading the source transfer time, with the same mass as for (b).

verified experimentally, and it is planned to conduct the necessary experiments soon.

1.8.8 Electronics for neutron-interrogation systems (G. Huxtable, J. Argyle, M. Calkin, R. Oliver-Hall)

Neutron differential-die-away chambers and neutron source-shuffler systems have many common features, though they operate on very different timescales. Both have groups of neutron detectors which provide a time-varying pulse rate, and it is this variation that has to be recorded.

We are developing a multiscaler front-end module for such systems, to use with standard Harwell 6000 series electronics. It will handle up to 8 parallel data streams and will feed data to Instrumentation & Applied Physics Division's new histogramming memory module type 6191. The whole unit will be controlled by an IBM-PC through an IEEE interface, and the intention is that the PC will also control a Teleflex-cable positioning system for a Californium neutron source, to make up a self-contained and simple neutron interrogation system. The multiscaler unit is complete, but not finally tested.

1.8.9 Methods of monitoring the ^{235}U enrichment of UF_6 gas in centrifuge enrichment plants operating at low pressure (T. W. Packer)

Work has continued on the development of an NDA instrument that will rapidly confirm that UF_6 gas in the pipework of a uranium centrifuge enrichment plant is in the LEU range (i.e. $\lesssim$20% enrichment). As described in a previous report (UKNDC(85)P112, p.54), the instrument is based on two types of measurement:-

(i) X-ray fluorescence analysis to indirectly determine the pressure of the UF_6 gas in the pipe.

(ii) A gamma-ray spectrometry procedure which determines the total mass of ^{235}U in the pipe by measuring the number of emitted 185.7keV gamma-rays. It incorporates a choice of two methods of distinguishing those emitted by the UF_6 gas from those emitted from any uranium deposited on the pipework.

The shorter measurement is based on the "deposit correction" technique which only requires an X-ray fluorescence measurement followed by a gamma-ray spectrometry determination. However it is subject to systematic errors. The "two geometry" technique, which is independent of major systematic errors, requires the same two initial measurements, but also requires a second gamma-ray spectrometry determination made under collimated geometry. The extra time that this additional measurement requires has been reduced by approximately a factor of three by using a variation of a more efficient third geometry developed for making similar measurements on the smaller diameter pipes at Almelo. It is now possible to make determinations at Capenhurst, using both techniques, in 30 to 45 minutes.

Our continued collaboration with the Los Alamos Laboratory in this work has resulted in the production of an operational system based on U.S. firmware (multi-channel analyser incorporating a dedicated UF_6 gas enrichment programme) and U.K. hardware (detector, collimator and shielding assembly). The complete system was demonstrated to, and operated by IAEA and Euratom Inspectorates at Capenhurst in October 1986. The equipment is now on loan to the IAEA at their Seibesdorf laboratory in Vienna.

1.8.10 Plutonium dating (D. West)

In UKNDC(86)P113, p.32) it was shown that a technique using gamma-ray analysis for dating the chemical separation of a Pu sample was capable of measurements over the range 3 months to 15 years with errors of circa 1%. A report of this work has now been issued as AERE-R 12181 (July 1986) and has been accepted for publication in Nuclear Instruments and Methods.

1.9 Work for JET - Nuclear techniques applied to fusion plasma diagnostics

The Joint European Torus (JET) situated at Culham is the world's leading plasma fusion experiment and an important step towards the eventual realisation of fusion reactors. The Neutron Systems Group at the Harwell Laboratory have been responsible for the design, construction, installation and commissioning of the major neutron detection instruments in JET's portfolio of diagnostics for plasma conditions. We are now involved, through a Task Agreement with JET, in the regular exploitation of the instruments for best information on plasma conditions, while further developing their capability and exploring the development of new diagnostics methods.

1.9.1 Development and commissioning of the neutron-activation yield monitor at JET (KN2 diagnostic) (G. Huxtable, J. Argyle, P. Dixon, M. Calkin, M. Philbin, and R. Oliver-Hall)

Construction of the capsule-transfer ("rabbbit") system for neutron-activation measurements at JET (UKNDC(85)P112, p.57), and first-stage installation in the JET torus hall/diagnostic hall areas is finished. At present four irradiation positions are in use, all in the lower half of the torus. Final-stage installation of an additional four upper irradiation stations will be done at completion of the current upgrading of JET.

For counting the induced activities, the capsules are delivered to stations coupled to appropriate detectors. A germanium detector, calibrated for quantitative gamma-activity measurements, and a NaI detector have been provided and coupled to the automatic control system. For the next phase of JET operations there are plans to provide additional detectors. The rabbit system is also used to transport capsules containing fission foils to delayed-neutron detectors supplied by a group from Mol, Belgium; these detectors are also coupled to the automatic control system.

The rabbit control system has been designed with an interface to the JET computer system, and this has been tested. Operation under JET computer control awaits suitable JET software: until now, transfer operations have been controlled directly by an operator.

At present, gamma-spectra are accumulated and stored in a pulse height analyser, before transfer, via diskette, for detailed analysis on a mainframe computer. A version of GAMANAL[1] has been used for data reduction. Direct transfer of this data to JET's NORD system awaits new JET software.

Even in its present non-automated form the KN2 system has provided a useful tool for measuring the neutron fluxes immediately adjacent to the vacuum wall of the JET torus.

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(1) J. A. B. Goodall, UKAEA Report, AERE-M 3185 (1982).

1.9.2 Use of KN2 activation system for measuring JET neutron fluxes

(G. Huxtable, J. Argyle, D. B. Syme, with P. Battistoni[1], M. Martone[1], M. Pillon[1], S. Podda[1], M. Rapisarda[1], E. Bittone[2], K. Verschuur[3], G. Gorini[4], O. N. Jarvis[5], J. Kallne[5], and G. Sadler[5])

In order to understand the effects of changing parameters as the JET experiment develops, it is fundamental to measure well the neutron yield from the plasma. The existing KN1 neutron monitors respond to plasma neutrons rather indirectly, as the shielding shell around the torus intervenes, so it is very useful to get a more direct check on the neutron output from a neutron-activation measurement within the shell. Even at this position it is necessary to make some neutron transport calculations to relate activity to yield, but these calculations can be of a simpler nature: the code "Furnace" was used (ECN 86-097). The reaction $^{115}In(n,n')^{115m}In$ was used, in which neutrons over 1 MeV activate indium to a metastable state with a half-life of 54 min. Neutrons arriving directly from a D-D plasma will predominantly be 2.5 MeV, but there is a tail down to thermal energies from scattered neutrons. A complication is the intense activity from $^{115}In(n,\gamma)^{116m}In$, due to its high cross-section at low energies, which masks the ^{115m}In activity initially. Consequently the ^{116m}In activity has to be allowed to decay somewhat before making the ^{115m}In measurement.

Some corrections to the data remain to be made but first results imply reasonably good agreement of absolute calibration between KN2 and the present standard diagnostic KN1, and excellent reproducibility of the ratio from shot to shot.

The reaction $^{63}Cu(n,2n)^{62}Cu$, which gives an appreciable response for neutrons above about 10 MeV, was used to detect 14 MeV neutrons. The activation gives a 511 keV gamma-ray from positron annihilation with a half-life of 10 mins. The high-energy neutrons are less than 1% of the 2.5 MeV neutrons: this

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1 ENEA, Frascati, Italy
2 ENEA, Bologna, Italy
3 ECN, Petten, Holland
4 Scuola Normal Superiore, Pisa, Italy
5 JET

is the "burn-up" ratio. Unfortunately, low energy neutrons can also give rise to 511 keV gammas from a (n,γ) reaction in copper, and this is a significant contribution which has to be allowed for by counting at later times to quantify the longer half-life involved.

A series of measurements of this "burn-up" ratio has been made covering a wide range of JET shots. The data are presently being analysed, mainly at Frascati. This information is sufficiently interesting that it is planned to make these burn-up ratio measurements routinely for most JET shots in the future.

1.9.3 Installation of neutron profile monitor diagnostics (KN3) at JET (J. M. Adams, P. Dixon*, and N. Watkins)

The neutron profile monitor diagnostic (KN3), designed, developed and constructed by Harwell, was successfully installed at JET in the Autumn for initial testing and commissioning. The diagnostic comprises 2 large heavy concrete shield/collimator assemblies which enable the viewing of neutrons emitted from the JET plasma both horizontally and vertically. Each shield/collimator assembly contains an array of NE213 fast neutron detectors**; the horizontal array contains 10 detectors, and the vertical array contains 7 detectors. Each detector was enclosed in a triple magnetic shield comprising mild steel(5 mm)/radiometal(2 mm)$^{+}$/ μ-metal(1.2 mm)$^{+}$ to reduce the effect of the large magnetic fields that arise during a JET shot; up to 1 kgauss has been observed in the region of the vertical array. Each detector was connected to a pulse shape discriminator (PSD)$^{++}$ in order to discriminate between neutron and gamma events. Separate logic pulse trains from neutron, gamma and livetime outputs from each PSD were then fed via latching scalers into the JET computerised data acquisition system (CODAS) to record line-of-sight time-resolved profiles of neutron and of gamma emission during the JET shot, together with analogue data corresponding to a specific detector channel.

Before the inclusion of the radiometal/μ-metal shields there was clear evidence of magnetic field effects in the horizontal array. No such evidence

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* Engineering Projects Division
** Supplied by Nuclear Enterprises Limited, Sighthill, Edinburgh
\+ Supplied by Telcon Metals Limited, Crawley, Sussex
++ Link Systems Model 5020, supplied by Link Analytical Limited, High Wycombe, Bucks.

was observed once the shields were included in both arrays. However, the effect of totally enclosing each detector in a triple magnetic shield was to increase the detector environmental temperature to ~36°C, even after the inclusion of a cooling water supply.

Overall, this initial KN3 installation and partial commissioning was considered successful. After the JET shutdown for a major upgrade at the end of November, the KN3 diagnostic was completely dismantled, and a number of modifications set in train. These include modifications of the NE213 detector dynode chains, installation of a suitable cooling water system, modification of the vertical array detector box to accommodate 9 NE213 detectors, filling of both detector array boxes with a $LiCO_3$/paraffin wax mixture to reduce 'cross-talk' between adjacent detector channels, and a variety of improvements to the electronics.

1.9.4 First measurements using the neutron profile monitor diagnostic (KN3) on JET (J. M. Adams and N. Watkins)

The initial installation and partial commissioning of the Neutron Profile Monitor Diagnostic (KN3) on JET enabled the recording of line-of-sight time-resolved neutron and gamma emission profiles from the plasma for some 600-800 JET shots under a variety of JET operating conditions. In the time-resolved neutron emission profiles there is clear evidence of 'sawteeth' (a plasma relaxation phenomenon) in detector channels viewing the central region of the plasma, accompanied by 'sawteeth reversals' in some of the detector channels viewing the outer regions of the plasma. In this case the oscillations in neutron emission occur at the same time but in opposite phase to those observed in the 'plasma centre'. As would be expected the gamma emission profiles do not exhibit this phenomenon since the gamma emission emanates from the torus wall and not the plasma. The neutron and gamma emission profiles give a good indication of where the plasma strikes the torus wall in the event of plasma disruptions, and, for some JET shots, clearly indicate that the plasma has struck the torus wall early in the shot.

The vast amount of preliminary data is in the process of being analysed, and, to this end, a comprehensive data library of all the recorded KN3 data has been produced to aid in the selection of similar JET shots.

1.9.5 2.5 MeV neutron spectrometer (KM1) for JET deuterium plasmas

(N. P. Hawkes, P. Dixon, M. Hone*, O. N. Jarvis*, M. Loughlin+, G. Sadler*, A. C. Sherwood, M. T. Swinhoe, D. B. Syme and P. van Belle*)

The aim of the 2.5 MeV neutron spectrometry system is to deduce the energy distribution of ions in the plasma by measuring the energy spectrum of the 2.5 MeV D-D fusion neutrons. The system consists of a set of detectors (see UKNDC(83)P109, p.69) which can be placed inside a well-shielded enclosure in the JET Torus Hall.

The shielding for the enclosure consists of a mixture of 90% paraffin wax and 10% lithium carbonate, 100 cm thick at the front and 60 cm thick elsewhere, cast inside a steel shell. The interior of the enclosure measures 1.8 m long by 1 m high by 0.6 m wide, and access is provided by a motorised door at the rear. The total weight of the shield is approximately 15 tonnes. It is supported on a steel pillar over 4 m high to raise it to the mid-plane of the torus, and a collimator which partly protrudes from the front of the shield defines the (horizontal) line of sight to the plasma (see Figure 1.8).

The shield was installed at JET in October 1986, and data have been taken with the ^{3}He gridded ion chamber for about 1000 shots.

In November 1986, the Si diode from the prototype proton recoil spectrometer was installed in the shielded enclosure, without a proton radiator, to assess the severity of background. The diode was placed in a graphite pot (not evacuated), with an ^{241}Am calibration source. It was positioned on the floor of the enclosure, out of the direct beam but exposed to the substantial scattered flux from the ^{3}He spectrometer. The results were very encouraging, with no background signal exceeding a pulse height equivalent to 800 keV. The diode has now been returned to the proton recoil spectrometer vacuum chamber, and the re-assembled prototype system will now be tested on the Harwell 500 keV Van de Graaff.

At the same time that the Si diode was installed, an NE213 detector was also placed in the shield, again out of the direct beam but exposed to scattered

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*JET
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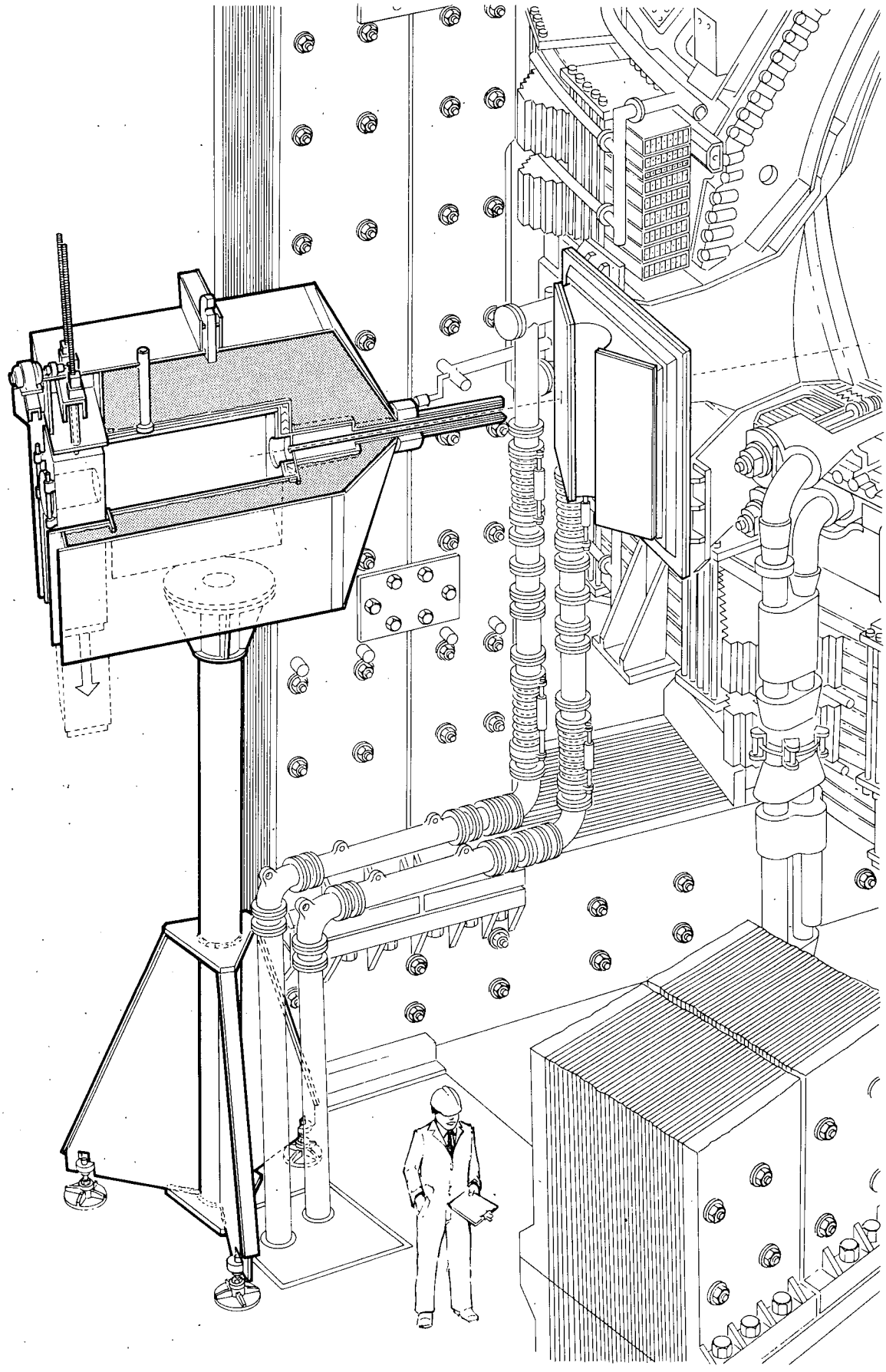

Figure 1.8 The shielded enclosure for the 2.5 MeV neutron spectrometry system is shown at the upper left hand side of this drawing.

flux. The photomultiplier was protected by three layers of magnetic shielding, which successfully prevented the large fields generated by JET from distorting the signal. However, difficulties were experienced with the pulse shape discrimination system, intended to distinguish pulses due to neutrons from those due to gamma rays. The reason for this is being investigated.

During the Winter 1986/Spring 1987 JET shutdown, an adjustable collimator will be installed in front of the spectrometer enclosure in the Torus Hall. This will allow the spectrometer aperture to be changed at short notice to suit JET neutron yields, and will greatly improve the useful range of the instrument.

1.9.6 First measurements using the 2.5 MeV neutron spectrometer (KM1) on JET (N. P. Hawkes, D. B. Syme and M. Loughlin[+])

Following installation of the KM1 shield in the JET torus hall in October 1986, JET neutron spectra have been measured for about 1000 shots using the ^{3}He detector in KM2 and a second ^{3}He detector in the roof laboratory above JET. Data analysis methods have been developed to produce neutron spectra from the raw data. Information on plasma temperatures and ion velocity distributions has thus been acquired for various plasma heating regimes.

For plasmas heated ohmically or with additional RF power, the neutron energy spectrum is usually assumed to be Gaussian in shape, with a width determined by the temperature of the plasma. This assumption greatly simplifies the task of de-convoluting (unfolding) the detector response from the measured data. With plasmas heated by neutral beam injection (NBI), however, no simple parameterisation of the spectrum shape is known, so we have developed analysis programs which carry out the unfolding with no prior assumptions about the form of the spectrum. A typical result appears in Figure 1.9a. This shows the unfolded neutron spectrum during the middle two seconds of neutral beam injection (three shots have been summed together to improve statistics). Comparison with Figure 1.9b, which shows RF-only shots unfolded using the same program, demonstrates clearly how the neutral beams broaden the spectrum and change its shape.

[+]Birmingham University

A second ^{3}He chamber is also in use at JET. This detector is installed in the Roof Laboratory, and views the plasma along a vertical line of sight (ie perpendicular to the neutral beams instead of parallel). For ohmic and RF plasmas, it gives the same results as the first detector, but for neutral beam shots there are interesting differences - the horizontal spectrum shows a small shift in mean energy, whereas the vertical spectrum does not (this is related to the kinematic energy shift and other effects like plasma rotation introduced by the neutral beams); and there are indications that the detailed shapes of the two spectra also differ. Figure 1.9c shows the unfolded vertical spectrum for the same plasma as for Figure 1.9a.

1.9.7 NE213 fast neutron detector response function measurements (J. M. Adams, M. Loughlin* and N. Watkins)

The NE213 fast neutron detectors used in the JET Neutron Profile Monitor (KN3) (see Section 1.9.3) comprise 50 mm diam. by 10 mm thick NE213 BAI cells** coupled to 50 mm EMI 9815B photomultipliers+. These detectors are used in conjunction with Link Systems Model 5020 Pulse Shape Discriminator (PSD)++ to discriminate between neutron and gamma events. Each detector has a built-in ^{22}Na source, (~200 nanocuries), which enables a calibration of the detector pulse height spectrum in terms of the equivalent electron energy. On JET these detectors are used to detect the 2.45 MeV neutrons emitted from a deuterium plasma and these data are recorded as time-resolved neutron profiles. In order to reduce the detection of neutrons scattered by the torus wall etc., these detectors are set to count events above a threshold of about 2 MeV in neutron energy. To reduce the number of gamma events processed by the PSD unit, events depositing more energy than that of 3 MeV neutrons are rejected by the pulse height discrimination circuitry.

Experience on JET indicated a discrepancy between neutron light output relations available in the literature and the results from 2.45 MeV neutrons emitted from fusion reactions in a deuterium plasma. This caused errors in setting the pulse height levels corresponding to the required threshold energies.

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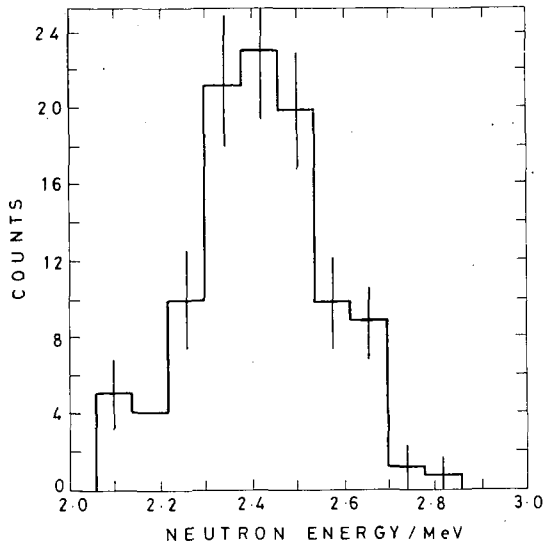
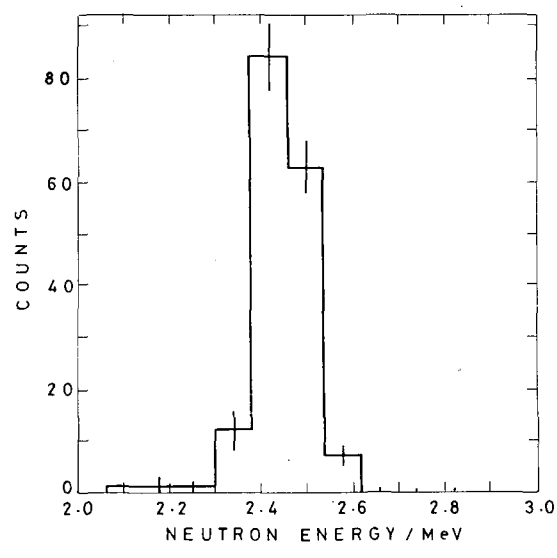
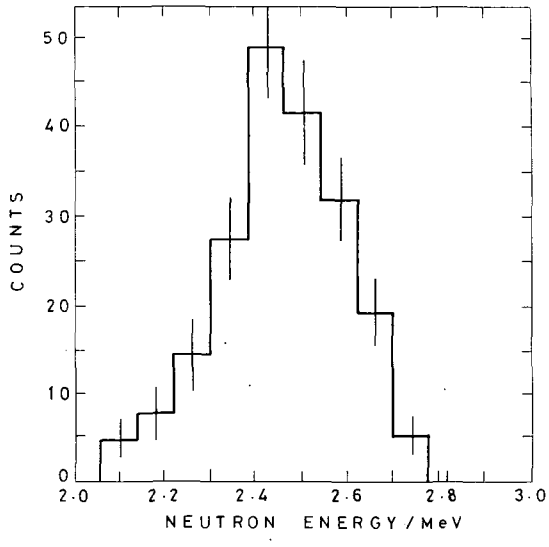
* University of Birmingham
** Supplied by Nuclear Enterprises Limited, Sighthill, Edinburgh
\+ Supplied by Thorn EMI Electron Tubes Limited, Ruislip, Middlesex
++ Supplied by Link Analytical Limited, High Wycombe, Bucks.



(a)



(b)



(c)

Figure 1.9 Unfolded neutron spectra, (a) for a period of 2 seconds during neutral beam injection, along a horizontal line of sight (summed data from pulses 10954-10956), (b) for R.F. only shots (summed data from shots 10350-10354), (c) as (a) but viewed along a vertical line of sight.

To resolve this question a series of neutron response function measurements have been performed on the 3MV Van de Graaff utilising monoenergetic neutrons from the T(p,n) reaction, using a thin TiT_2 target of ~50μg/cm^{2++}, over the energy range 1.5 - 3.2 MeV. Preliminary results are consistent with the JET data, and confirm the discrepancy with published equivalent electron energy data for neutrons for NE213.

1.9.8 ^{3}He spectrometer response function measurements (J. M. Adams, J. Källne*, M. Loughlin**, V. Merlo*, G. Sadler* and N. Watkins)

The ^{3}He spectrometer used as one of the detectors comprising the 2.5 MeV Spectrometer Diagnostic (KM1) (see Section 1.9.5), on JET is a commercially available gridded ^{3}He ionisation chamber$^+$. It is used to measure fusion neutron spectra from various types of deuterium plasma on JET, under heating systems such as ohmic heating and neutral beam injection, from which the plasma temperature or ion velocity distributions can then be calculated. To do this properly a good knowledge of the ^{3}He spectrometer response to neutrons in the energy range 2-3 MeV is required.

A series of neutron response function measurements have been performed on the Harwell 5MV Van de Graaff accelerator, using a thin TiT_2 target of ~50μg/cm^{++} to produce monoenergetic neutrons from the T(p,n) reaction over the energy range 2-3 MeV. These measurements included an investigation of various combinations of ionisation chamber and amplifier, and different orientations of the ionisation chamber with respect to the monoenergetic neutron source. Several amplifiers of different manufacture were employed with shaping times ranging from 4-10 μs.

An analysis of the response function data is currently being made. The data are fitted with a function consisting of 2 Gaussians plus a wall effect term:

$$Y(E)=a\ \exp\{-0.5[(E-b)/c]^2\} + d\ \exp\{-0.5[(E-e)/f]^2\} + g\ \ WALL$$

where E is the neutron energy; a,d are the heights of the Gaussians; b,e are the

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\+ Supplied by Jordan Valley, Israel
++ Supplied by Multivolt Limited, Crawley, Sussex.

mean positions of the Gaussians; c,f are the standard deviations of the Gaussians; and g is a normalising constant for WALL, the shape of the wall effect term as given by Batchelor[1]. The fitting procedure involves freely adjusting the parameters to minimise χ^2. The resultant parameterisation as a function of neutron energy is required to unfold the neutron spectra obtained on JET under the various operating regimes.

1.9.9 CH_4 spherical ionization chamber response function measurements (J. M. Adams, F. Hoenen*, M. Loughlin**, G Sadler+ and P. van Belle+)

The CH_4 spherical ionization chambers used as one of the proposed detectors comprising the 2.4 MeV Spectrometer Diagnostic (KM1) (see Section 1.9.5) on JET were developed by F. Hoenen[2] at Jülich. The neutron response of a number of these detectors was measured on the Harwell 5MV Van de Graaff utilising monoenergetic neutrons from the T(p,n) reaction, using a thin TiT_2 target of ~50μg/cm^{2}++, over the energy range 2-3 MeV. The measurements involved the examination of the effect of different amplifiers, and various shaping times in the range 6-10μs.

Preliminary results indicate a proton recoil edge resolution of 3-4%. A full analysis of the data is currently in progress.

1.9.10 Time resolved measurement of 14 MeV neutron production (JET diagnostic KN7) (G. Huxtable, O. N. Jarvis+, S. Conroy+)

This is a preliminary investigation into the operation of a silicon detector close to the torus. It uses fast electronics for crude discrimination between the few large current pulses resulting from 14 MeV neutron reactions in the silicon, and the much larger number of smaller pulses from lower energy neutrons and gammas (which present pile-up problems).

So far, we have shown that the detector's current pulses can be transmitted directly via superscreened cable to processing electronics in the diagnostic hall, preserving modest but adequate resolution despite the electrically noisy JET environment. When operating in this undemanding current-mode, the

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(1) R. Batchelor et al., Rev. Sci. Instr. 26 (1955) 1037.
(2) F. Hoenen, Nucl. Inst. Meth. 223 (1984) 150.
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** University of Birmingham
\+ JET
++ Supplied by Multivolt Limited, Crawley, Sussex.

detectors have an acceptable life as far as neutron damage is concerned, even remaining usable when leakage currents have increased far above the level that will be tolerable for high-resolution spectrometry.

To date, space limitations have prevented the installation of our preferred detectors, and it has been difficult to interpret observations from substitute diodes. We hope to improve this position soon and to install a simplified signal-processing system in the next phase of JET operations.

1.9.11 Fusion gamma-rays from JET (G. Sadler*, N. P. Hawkes, O. N. Jarvis*, D. B. Syme and P. van Belle*)

Detectors with substantial efficiency and energy resolution for high energy gamma-rays have been mounted in the roof laboratory above JET with lines of sight down through collimated apertures in the thick floor to the region of maximum plasma density in the torus. The objective is to observe gamma-rays produced in fusion reactions and evaluate the information on plasma conditions to be gained from such observations.

Gamma-rays with an energy of 16.6 MeV from the reaction $d(^3He,\gamma)^5Li$ have been observed during the last period of JET operation in 1986, using a 125 mm diameter by 125 mm long NaI(Tl) crystal and a 75 mm diameter by 75 mm long BGO crystal. The intensity of this gamma line indicates a fusion power of about 10 kW, leading to a Q-value of about 0.002, the highest ever achieved.

The 4.4 MeV gamma line from the decay of the first excited state of ^{12}C was also observed, showing carbon to be present as an impurity in the plasma.

1.9.12 14 MeV neutron spectrometer for D-T plasmas (N. P. Hawkes, D. B. Syme and D. West)

The current design for this device is shown schematically in Figure 1.14. Neutrons strike an annular proton radiator, and recoiling protons impinge on a semiconductor detector mounted on axis. The semiconductor detector is shielded from the direct neutron beam by a shadow bar. To increase the efficiency of the spectrometer, several such radiator-detector pairs are stacked one behind the other.

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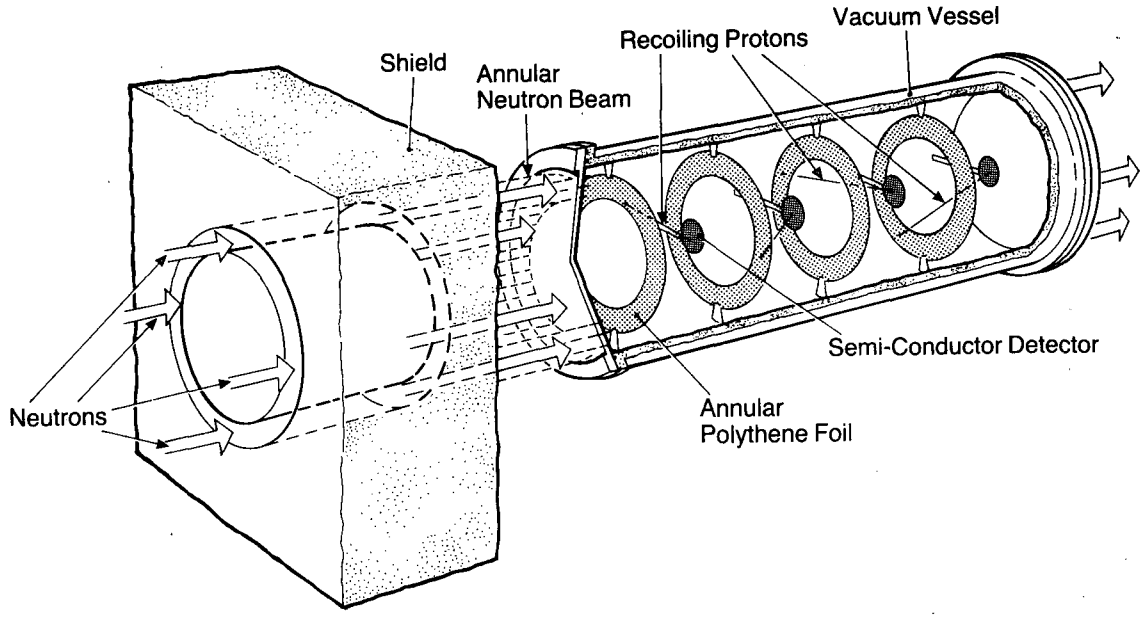

Figure 1.10 A schematic view of the 14 MeV neutron spectrometer design.

A computer program has been written to calculate the resolution and efficiency of a radiator-detector pair as a function of the radii of the annular radiator and of the proton detector, and the distance between the two. The results indicate that optimisation of these parameters could produce an improvement of about 90% in efficiency or more than 40% in resolution over the original design.

A silicon surface barrier detector is being obtained so that a detailed assessment of the relative performance of silicon and germanium proton detectors can be carried out. This will be done using the new well-collimated beam line on the Harwell 500 keV Van de Graaff accelerator. The major part of this work will continue, as part of a recently granted contract for development and construction of a 14 MeV neutron spectrometer for the (d,T) phase of JET operations. Some effort will continue on the underlying programme towards the design of the most appropriate variant of the device for the absolute determination of neutron fluxes in nuclear data measurements in the range of interest to fusion reactors.

2. CHEMICAL NUCLEAR DATA

2.1 Introduction

During 1986 the UK Chemical Nuclear Data Committee (UKCNDC) held two meetings (Chairman: A. L. Nichols (AEEW) and Secretary: F. G. Eltham (AEEW)). The Data Library Sub-committee (Chairman: A. Tobias (CEGB) and Secretary: M. F. James (AEEW)) also met twice during the year. Efforts at the committee level have focussed on assessing and re-defining the chemical nuclear data needs, and an updated UKCNDC Request List has been prepared and approved (AEEW-M 2330). There are increasing demands to improve the decay data for specific fission-product radionuclides that contribute significantly to the decay heat of fuel at cooling times of approximately 1000 s. Anomalies have also been identified in the fission yield data for tritium and ^{129}I in the fast reactor and reprocessing programmes, and laboratory studies are planned in the UK.

Evaluation effort is linked to the development and needs of the European Joint Evaluated File (JEF). Data evaluations in the UK include fission yields and decay data, and this work has been geared towards specific requirements for JEF-2.

Support for chemical nuclear data still stands at between 2 and 3 scientist-years in the UK, shared over many people within AWRE, CEGB, UKAEA, NPL and the universities. Specific items of work involve Extra Mural Research (EMR) contracts between the nuclear industry and specific universities, and this type of study needs to be encouraged to measure radionuclide half-lives, decay data, neutron emissions and fission yields.

2.2 Measurements

2.2.1 The measurement of tritium yield in the fast neutron fission of actinides: a proposal (J. W. McMillan*)

A detailed programme has been proposed for the measurement of the tritium yield in the fast neutron fission of specific actinides. The actinide nuclides ^{235}U, ^{238}U, ^{239}Pu, ^{240}Pu, ^{241}Pu and ^{242}Pu would be irradiated in a fast-reactor

*Harwell Laboratory

neutron spectrum (probably in MASURCA), and the tritium separated chemically from the specimens prior to measurement by liquid scintillation counting. Technical aspects of the proposal, timescale and probable cost have been determined and assessed.

During the 1970s Crouch (Harwell) surveyed the production of tritium by ternary fission and commenced a measurement programme which included irradiation of ^{235}U and ^{239}Pu in the fast-reactor neutron spectrum of ZEBRA. This programme was not completed, although the ZEBRA irradiated specimens could still be profitably examined. The continuing need to obtain tritium yields for the fast neutron fission of several actinide nuclides has been identified by Nichols in the 'UKCNDC Request List 1986' (AEEW-M 2330). The actinides identified were ^{235}U, ^{238}U, ^{239}Pu, ^{240}Pu and ^{241}Pu. While the 'Request List' also indicated that thermal neutron yields of tritium were required for the fission of ^{235}U, ^{239}Pu and ^{241}Pu, subsequent information provided by James and Banai (AEEW) suggests that they are known with an adequate accuracy of ~5%. Consequently a proposal has been prepared for an experimental programme which is confined to the measurement of tritium yields for the fast neutron fission of ^{235}U, ^{238}U, ^{239}Pu, ^{240}Pu, ^{241}Pu and ^{242}Pu; the last is added for completeness as a highly enriched supply of this actinide is available.

Critical aspects of the experimental programme have been identified, including minimum irradiation times to obtain suitable amounts of tritium for counting purposes, identification of low-temperature irradiation facilities, provision of a suitable supply of enriched actinides, and appropriate procedures for chemical separation. The zero-energy assembly MASURCA is the preferred irradiation facility. A supply of highly-enriched actinides is available at Harwell, but checks on their B and Li contents will be necessary to ensure that tritium production from these critical impurities is insignificant. Similar checks will be necessary on the aluminium used as irradiation containers. A high-temperature separation route has been chosen to minimise tritium dilution by water. Separation equipment contained within a glove box will be tested using specimens of ^{235}U and ^{239}Pu irradiated a number of years ago in ZEBRA. Approval for these measurements is currently being sought, and the programme could be completed by the end of 1988.

2.2.2 Absolute fission yields of selected fission product nuclides (T. W. Kyffin*)

The attainable accuracies of fission yield data have been reassessed for samples being currently irradiated in PFR. Because of changes in the programme, the expected burn-up of the samples has increased significantly to give the following estimated accuracies of measurement:

Material	Burn-up	Accuracy (1σ)
^{235}U	35%	2%
^{238}U	1.5%	5%
^{239}Pu	35%	2%
^{240}Pu	10%	2%
^{241}Pu	50%	2%

Proposed fission yield measurements are for Nd nuclides, $^{95}Nb/^{95}Zr$, ^{106}Ru, ^{137}Cs and ^{144}Ce.

2.2.3 Mass spectrometric measurements of alpha in PFR (R. C. Horscroft** and A. Tyrrell**)

One ^{239}Pu and two ^{240}Pu samples irradiated in PFR Dounreay have been analysed at AWRE. The quantities of specific nuclides have been measured and the total atoms calculated for zero cooling time (Table 2.1).

These data have been used to derive the desired Pu nuclide ratios (Table 2.2). This work is part of a collaborative effort between AWRE, DNPDE and Harwell to measure alpha at various positions in the PFR core.

2.2.4 Measurements of radionuclide decay data (M. F. Banham***, R. McCrohon*** and A. J. Fudge***) [WRENDA 1072]

Measurements of alpha and gamma-ray emission probabilities ($P_{\alpha,\gamma}$) and half-life ($t_{\frac{1}{2}}$) have been undertaken for a number of nuclides in collaboration with other laboratories. A range of germanium detectors is used coupled to a Canberra system comprising series 80 and 90 MCAs interfaced to a PDP 11/23 computer. Efforts also continue to improve the current measurement facilities. Details of the work are listed below.

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* Dounreay Nuclear Power Development Establishment (DNPDE)
** AWRE, Aldermaston
*** Harwell Laboratory

Table 2.1

Total Atoms in Irradiated Samples at Zero Cooling Time

Nuclide	DI/RST(39)		A2/JWW(25)		A2/JWW(47)	
	Atoms	RSD	Atoms	RSD	Atoms	RSD
^{137}Cs	3.463×10^{12}	2.368	1.220×10^{12}	0.1812	3.805×10^{12}	0.0072
^{144}Ce	8.738×10^{11}	0.5687	4.979×10^{11}	1.557	1.439×10^{12}	0.2385
^{155}Eu	9.596×10^{10}	0.2207	3.578×10^{10}	0.3812	8.941×10^{10}	0.0287
^{241}Am	1.437×10^{13}	0.0891	7.689×10^{12}	0.8800	1.390×10^{11}	0.0157
^{239}Pu	2.661×10^{12}	2.265	1.707×10^{12}	0.1881	6.552×10^{14}	0.3344
^{240}Pu	1.366×10^{15}	1.769	8.855×10^{14}	0.1399	1.505×10^{13}	0.8060
^{241}Pu	4.444×10^{13}	109.2	1.846×10^{13}	3.248	1.934×10^{11}	5.300
^{242}Pu	2.989×10^{12}	2.265	1.616×10^{12}	2.891	6.161×10^{9}	9.434

Table 2.2

Plutonium Ratios

Pin No.	Can No.	Sample No.	$^{239}Pu/^{240}Pu$+	$^{241}Pu/^{240}Pu$+	$^{242}Pu/^{240}Pu$+	Date of Analysis
DI/RST* (^{240}Pu)	39	40	0.001941(3)	0.02462(3)	0.002097(6)	25/7/84
A2/JWW (^{240}Pu)	25	10	0.001941(8) 0.001924(7)	0.01549(3) 0.01447(2)	0.001785(10) 0.001830(7)	24/7/84 26/11/85
			$^{240}Pu/^{239}Pu$+	$^{241}Pu/^{239}Pu$+	$^{242}Pu/^{239}Pu$+	
A2/JWW (^{239}Pu)	47	29	0.02292(2) 0.02298(1)	0.000187(1) 0.000174(1)	0.000010(1) 0.000010(1)	22/11/85 25/2/86

*only one result obtained because effort was requested on the A2/JWW samples
+numbers in parenthesis are 1σ standard uncertainties

(a) Half-life measurements of ^{233}Pa and ^{237}U: half-lives of ^{233}Pa and ^{237}U are currently being measured by means of an ion chamber and freshly purified sources of the nuclides.

(b) Liquid scintillation equipment for α-spectrometry as well as for low energy β-spectrometry is being developed and assembled.

(c) Development of a liquid scintillation counting method for ^{93m}Nb; the greater sensitivity of liquid scintillation relative to X-ray counting with a Si(Li) detector is being exploited to provide a rapid method for counting the reactor neutron dosimeter ^{93m}Nb.

(d) P_γ measurements of transplutonium nuclides: separations of americium and curium, after PFR irradiation of pellets of these elements, are being carried out to enable high resolution gamma-ray spectrometry to be carried out for P_γ measurements.

(e) Fission product studies: gamma scanning is being used to provide fission product distribution data in irradiated nuclear fuels; these data are used to provide information on neutron flux profiles, fission product migration and burn-up. Fission product ratios are being studied as a means of providing fuel history information. A start has also been made to develop gamma-emission tomography using the gamma-scanners to provide, for example, cross-pin flux gradients of intact fuel pins.

Publication: M. F. Banham and R. McCrohon, The measurement of gamma-ray emission probabilities for the nuclides ^{231}Pa, ^{233}Pa, ^{232}U, ^{235}U, ^{237}U and ^{237}Np, AERE-R 11353, Sept. 1986.

2.2.5 On-line chemical separation and studies of short-lived nuclides in fission and other nuclear reactions (H. E. Sims*, G. F. Blower**, G. W. A. Newton** and V. J. Robinson**)

The decay properties of short-lived nuclides ($t_{\frac{1}{2}}$<5 min) have been studied using the Harwell Variable Energy Cyclotron (VEC), the helium jet recoil transport facility and Ge(Li) spectroscopy. This work has been completed, and

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* Harwell Laboratory
** University of Manchester

five papers have recently been published in Radiochimica Acta on the gamma-ray abundances of Re and W radionuclides.

2.2.6 Direct transfer reactions in the heavy ion systems ^{12}C, ^{14}N and ^{16}O with ^{197}Au (H. E. Sims*, G. W. A. Newton**, V. J. Robinson** and H. L. Wilkinson**)

Single neutron transfer reactions in heavy ion systems are not fully understood. Absolute yields of the single nucleon transfer products in the systems ^{197}Au + ^{12}C, ^{14}N and ^{16}O have been studied as a function of heavy ion energy. This work has been completed and is in the process of being written up for publication (see also UKNDC(85)P112, p.73). The reaction mechanisms are not simple elastic processes and do not fit theory. Significant quantities of ^{197m}Hg, ^{196}Au and ^{198}Au were detected, with evidence of some neutron evaporation for ^{196}Au.

2.2.7 Production of actinides (G. W. A. Newton**)

Agreement has been reached to produce pure actinides for Harwell by chemical separation and judicious use of the excitation functions. The relevant excitation functions will be measured more precisely, and the products will be used as actinide standards.

2.3 CNDC Data Library Sub-committee

Current membership: A. Tobias (Chairman, CEGB/BNL), M. F. James (Secretary, AEEW), J. Banai (AEEW/University of Birmingham), A. J. Fudge (Harwell), A. L. Nichols (AEEW) and A. Whittaker (BNF PLC).

2.3.1 Data library developments

The current status of the UKCNDC Data Libraries is summarised in Table 2.3. Evaluation efforts have been restricted during the past 12 months with relatively little progress being made towards revisions of the decay data files. Spectral data from these files may be accessed by the retrieval system described by Tobias (RD/B/5170N81 (1981)). Much of the decay data and the fission yields from the files have been incorporated in the JEF-1 data library.

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Table 2.3

UK Chemical Nuclear Data Libraries: Status Table, December 1986

Data	Present Status	File Development
1. Fission Product Data	Exists as UKFPDD-2 (ENDF/B-IV format) - replaces UKFPDD-1 total no. of nuclides = 855 radioactive nuclides = 736 ground state = 175 1st excited state = 133 2nd excited state = 5 nuclides with spectra = 390 total no. of γ lines = 11,978 total no. of β^- lines = 3,592 total no. of β^+ lines = 91	Data acquisition for future revision. Some data have been converted to ENDF/B-V format. Adoption of delayed neutron emission probabilities, and consideration to include delayed neutron spectra
2. Activation Product Decay Data	Available in ENDF/B-IV and V format for 91 nuclides as UKPADD-1. Now includes detailed K X-ray spectra.	60 nuclides have been evaluated for UKPADD-2; 56 are in ENDF/B-V format. These evaluations have been temporarily suspended whilst ^{51}Cr, ^{65}Zn, ^{75}Se and ^{198}Au have been evaluated as part of an IAEA-CRP exercise.
3. Heavy Element and Actinide Decay Data	Completion of UKHEDD-1, including spontaneous fission data in June 1982. Data in ENDF/B-V format for: total no. of nuclides = 125 ground state = 111 1st metastable state = 13 2nd metastable state = 1 total no. of α lines = 767 total no. of β^- lines = 527 total no. of β^+ lines = 39 total no. of γ lines = 3,475 total no. of discrete electrons = 6,755 total no. of X-rays = 381	Evaluation of 4n series proceeding: ^{236m}Np, ^{236g}Np, ^{236}Pu, ^{232}U, ^{228}Th, ^{224}Ra, ^{220}Rn and ^{216}Po completed. ^{231}Pa, ^{234}U and ^{239}U decay data and ^{242m}Am half-life have been evaluated.
4. Fission Yields	Available in ENDF/B-V format based on Banai/James revision (UKFY1): UKIFYU1 - unadjusted independent yields UKIFYA1 - adjusted independent yields UKCFYA1 - adjusted cumulative yields	Report on evaluation being prepared. Initially work for next evaluation will concentrate on production of new database and revised fractional independent yields.

(a) Heavy elements (A. L. Nichols*)

An extensive report has been published by the IAEA, describing all the measurements and evaluations made under the auspices of the Coordinated Research Programme on the Measurement and Evaluation of Transactinium Isotope Decay Data (IAEA Technical Reports Series No. 261). Efforts are now in hand to include these improved data in a new version of the UKCNDC heavy element decay data library (UKHEDD-2). Decay data have also been evaluated for specific radionuclides in the 4n decay chain of ^{232}U. This work is half completed and evaluations have been undertaken for ^{236m}Np, ^{236g}Np, ^{236}Pu, ^{232}U, ^{228}Th, ^{224}Ra, ^{220}Rn and ^{216}Po. Further work is planned to complete this decay chain series.

(b) Fission products (A. Tobias**)

There have been no evaluations made specifically for the UKCNDC data files during this period. All efforts have been directed towards the JEF collaboration.

(c) Activation products (A. L. Nichols*)

Gamma-ray emission probability data have been evaluated for ^{75}Se. This concludes a preliminary assessment of data for four radionuclides to be considered as calibration standards for gamma-ray detectors (^{51}Cr, ^{65}Zn, ^{75}Se and ^{198}Au). The evaluations have been undertaken for the IAEA Consultants Meeting on Gamma-ray Standards for Detector Calibration held at Grenoble, May 1985.

(d) Fission product yields (J. Banai+, D. R. Weaver+ and M. F. James*)

Under a three year cooperative agreement funded by CEGB, UKAEA and BNF PLC, a research fellowship (J. Banai) has been set up at AEE Winfrith. This agreement covers the evaluation of the fission product yields following neutron induced fission of ^{232}Th, ^{233}U, ^{235}U, ^{238}U, ^{239}Pu, ^{240}Pu and ^{241}Pu at various neutron energies, based upon experimental results and theoretical modelling. These results are of importance in nuclear reactor technology for kinetics and control, decay heat, burn-up, materials damage surveillance, safety and accident studies.

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* AEEW
** CEGB/BNL
\+ University of Birmingham

Earlier work in the United Kingdom has been undertaken by Crouch at Harwell[1]. The initial phase of the present programme consisted of reorganising Crouch's analysis codes and introducing some improvements. Thus, ternary fission yields are now included in a systematic way, and the treatment of chains involving isomeric yields has been modified. The codes now identify those data which contribute significantly to the χ^2 value and thereby permit the rapid identification of mass chains in which the basic data are discrepant. Currently, the philosophy has been adopted that these discrepant data are not being removed from the evaluation (as previously done by Crouch), but are left in the data set for future reconsideration in the light of any new experimental data. The code produces an error estimate based upon the larger of the internal and external estimates using weighted averaging.

Chain yields and independent yields have been evaluated; the chain yields are a completely new evaluation while the independent yields make use of the earlier fractional independent yields of Crouch but combined with the new chain yields. Future work will include the revision of these fractional independent yield data, and work is in hand on the extension and revision of the data-base of experimental results required to implement these changes. Currently, two libraries (UKIFYU1 and UKIFYA1) have been produced and these replace the interim C4U and C4A files produced by Crouch et al.[2]. The UKIFYA1 file is the result of the application of physical constraints to the evaluation process so that a number of conditions are met by the resulting values. These include the conservation of nucleons, conservations of charge, balance between the light and heavy mass peaks, etc. On the other hand, the UKIFYU1 file has been produced on the basis of the best fit to the experimental data without application of the constraints. Both files have been submitted to the Joint Evaluated File (JEF) and include error estimates based on the fitting procedures; full covariance matrices can be provided but are large and rather sparse.

Traditionally independent yields are quoted before delayed neutron emission while cumulative yields allow for the prior emission of delayed neutrons. Work is in progress on a cumulative yield file (UKCFYA1) and a preliminary version has been submitted to JEF. Branching ratios have been derived from

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(1) E. A. C. Crouch, Atomic and Nuclear Data Tables, 19 (1977) 417.
(2) E. A. C. Crouch, M. F. James, P. S. Whitworth and C. Dunn, New Fission Product Libraries C4A and C4U, CNDC(85)P3 (1985).

UKFPDD-2[1] or, if the data are not available from this source, from the Table of Isotopes[2] and Nuclear Data Sheets[3].

In order to gain first hand experience of techniques associated with measurement of fission yields, Banai has participated in experiments at ILL Grenoble and at the Studsvik Laboratory in Sweden, by courtesy of the administrations of both facilities.

2.3.2 Joint Evaluated File (JEF)

The preliminary version of the JEF-1 decay data file included a number of short-lived fission product nuclides whose decay data were inadequate for the calculation of average decay energies. Suitable values for these parameters have been extracted from published measurements and predictions and were incorporated into the file by mid-1986.

Decay heat predictions using the JEF-1 decay and fission yield data have been compared with corresponding values obtained with the UK data files. Differences of up to a few percent were observed in the predictions for a fission pulse. These occurred mainly at short cooling times and can be attributed to revised fission yield data. For practical applications the differences in the integral predictions using JEF-1 and UK data were shown to be much smaller. As a consequence of improved data for short-lived fission products in JEF-1, predicted gamma spectra at short cooling times were more complete than those obtained with the UK data. It was concluded that for total decay heat predictions there was little to choose between UK data and that of JEF-1. However, for applications which require spectral predictions at short cooling times the use of JEF-1 is preferred.

UK and JEF-1 total decay heat predictions have also been compared with results of a least squares fit to measured data for both ^{235}U and ^{239}Pu and directly with results of measurements for ^{238}U and ^{241}Pu. Acknowledged

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(1) A. Tobias and B. S. J. Davies, UKFPDD-2: A Revised Fission Product Decay Data File in ENDF/B-IV Format, CEGB Report RD/B/N4942 (1980), Berkeley Nuclear Laboratories.

(2) C. M. Lederer and V. S. Shirley, Table of Isotopes, Seventh Edition, John Wiley & Sons, New York (1978).

(3) Nuclear Data Sheets, edited by the National Nuclear Data Centre, Brookhaven National Laboratory, Upton, New York, USA.

deficiencies in decay data for short-lived fission products (half lives $<$100 s) were confirmed by the observed differences between measurement and predictions at short cooling times. However, it was found that both UK and JEF-1 predictions displayed a discrepancy of 5-10% with respect to measurements for cooling times around 1000 s when the decay data were considered reliable. Since this discrepancy may lead to uncertainties which exceed the accuracy requirements for some applications (typically 5%), it was necessary to identify the cause and seek suitable improvements to the basic data.

The principal decay heat nuclides at a cooling time of 1000 s were identified, and it was found that, for a number of them, the decay characteristics had been determined from relatively few measurements. It was also shown that the decay data for these nuclides, present in both UK and JEF-1 data libraries, offered scope for revision which would result in better agreement between measurement and prediction. As plans for JEF-2 are prepared, consideration should be given to improve decay heat predictions. It has been shown that while there is a general requirement for further data on short-lived nuclides, there is also a specific need to re-examine experimentally the decay schemes of a number of fission products with effective half-lives of around 1000 s.

3. REACTOR PHYSICS DIVISION, AEE WINFRITH

(Division head: Mr. A. J. Briggs)

3.1 Few group cross-sections for use in FISPIN (R. W. Smith)

An evaluation of the production cross-section of ^{236}Pu (from the (n,2n) reaction on ^{237}Np) has been completed. The evaluation is very similar to Fort's[1] recently recommended data. In a comparison with the recent integral measurement by Gromova[2] of the ^{252}Cf spontaneous fission spectrum the agreement is within 25%. Further adjustments to the present evaluation may be undertaken when further integral evidence becomes available. The present evaluation has been used to produce new 3 group data for use in FISPIN with MAGNOX, AGR, BWR and PWR fuels.

3.2 The automated production of group cross-sections for use in FISPIN (R. W. Smith)

A code for the automated production of irradiation dependent actinide libraries (containing 129 actinides) in the FISPIN (3 group) library format using appropriate WIMS weighting spectra (69 group fluxes) has been developed.

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(1) E. Fort, JEF-DOC-86.
(2) E. A. Gromova et al., Atomnaya Energiya 60 (1986) 68.

4. DIVISION OF RADIATION SCIENCE AND ACOUSTICS

NATIONAL PHYSICAL LABORATORY

Superintendent: Dr. K. C. Shotton

(Radioactivity and Neutron Measurements Branch)

(Branch head: Dr. P. Christmas)

4.1 International intercomparisons (under the auspices of the International Bureau of Weights and Measures, BIPM)

4.1.1 Radioactivity measurements (D. Smith and M. J. Woods)

Following earlier participation in a pilot intercomparison, NPL took part in a full-scale intercomparison of ^{109}Cd involving 18 laboratories around the world. At NPL a large, pressurised proportional counter was used to determine the total conversion electron emission, and the gamma-ray emission was measured with a calibrated ionisation chamber. A preliminary report shows the NPL result to be close to the mean value of the other participants.

4.1.2 Neutron measurements (J. B. Hunt, V. E. Lewis and T. B. Ryves)

The results of the measurements at NPL for the recent neutron dosimetry intercomparison have been analysed, and the work presented in an internal report: a full report will be produced when all participants have completed their measurements.

Participation in intercomparisons of neutron fluence rate measurements using (a) Harwell fission chambers and (b) an NPL dual moderating sphere system, has been postponed due to delays elsewhere. The fission chambers have been brought to Teddington on two occasions, however, for periodic checking against a radioactive neutron source.

4.2 Neutron cross-sections (V. E. Lewis and T. B. Ryves)

Measurements of the thermal neutron capture cross-section ratios of B, S and Mn to H, carried out in collaboration with the University of New Mexico, Albuquerque and using a modified manganese sulphate bath technique have been completed, and a full co-variance analysis published[1].

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(1) T. B. Ryves, A. Arbildo and J. C. Robertson, Ann. Nucl. En. 13 (1986) 237 and 679

Using natural Ag targets, cross-sections have been measured at 14.3 MeV for the reactions $^{107}Ag(n,2n)^{106m+g}Ag$, $^{109}Ag(n,2n)^{108g}Ag$, $^{109}Ag(n,p)^{109m+g}Pd$ and $^{109}Ag(n,\alpha)^{106m}Rh$. In each case the induced beta activity was measured, yielding results of high precision. A paper describing this work is in course of preparation.

4.3 Decay data (P. Christmas, D. Smith, M. J. Woods, W. Gelletly* and S. L. Waters**)

^{152}Eu, ^{154}Eu: Measurements of the half-lives of these nuclides have been published[1]. The final values were (4943±4) days and (3138±2) days respectively.

^{65}Ni: The half-life has been determined, and measurements of the gamma-ray emission probabilities are underway.

^{90}Sr: Measurements of the half-life continue, using an ionisation chamber to detect bremsstrahlung from a solid source.

^{24}Na: It has been suggested that the chemical state of a radionuclide may affect its half-life; this is being investigated using two chemically distinct sources in a differential ionisation chamber system.

^{82}Sr: The half-life has been determined by gamma-ray spectrometry to be (25.342±0.053) days; use of the conventional ionisation chamber technique was precluded by the presence of gamma-emitting contaminants. This work has been accepted for publication[2].

^{237}Np, ^{233}Pa: An account of measurements of a number of internal conversion coefficients of these nuclides was presented at the 1986 Nuclear Data Forum. The work involved Bergkvist-type multi-strip sources, made by Chemistry Division, Harwell, and a 16-element proportional counter.

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(1) M. J. Woods and S. E. M. Lucas, Int. J. Appl. Radiat. Isot., 37 (1986) 1157.

(2) S. M. Judge, A. M. Privitera and M. J. Woods, Int. J. Appl. Radiat. Isot., to be published.

* Schuster Laboratory, University of Manchester.

** MRC Cyclotron Unit, Hammersmith Hospital, London.

^{241}Am: Under a continuing SERC-supported collaboration with Manchester University, measurements similar to those described immediately above have been made on ^{241}Am, again using a multi-strip source prepared at Harwell.

^{47}Ca: The endpoint energies of the two beta branches have been determined using the NPL iron-free spectrometer together with an electron-gamma coincidence system.

^{180m}Hf: The existence of the rarest naturally occurring 'stable' nuclide, ^{180}Ta (0.012% of all Ta), has been explained in terms of the hitherto unobserved beta decay of ^{180m}Hf. An initial search for this decay suggests the presence of two beta branches, in accordance with the accepted level schemes. This investigation continues.

4.4 Evaluations (P. Christmas and M. J. Woods)

Within the framework of an IAEA Co-ordinated Research Programme a preliminary evaluation has been carried out of half-life data for nuclides commonly used to calibrate gamma-ray spectrometers. The work has been shared with the Physicalisch Technische Bundesanstalt (PTB), W. Germany.

5. DEPARTMENT OF PHYSICS RADIATION CENTRE, UNIVERSITY OF BIRMINGHAM

(Director: Professor J. Walker)

5.1 Delayed neutron spectroscopy (S. J. Chilton, D. R. Weaver, J. G. Owen and J. Walker) [WRENDA 979]

Measurements of spectra of delayed neutrons from the fission of ^{235}U induced by neutrons from the Be(d,n) reaction have continued during the year. As described in last year's report (UKNDC(86)P113, p.54) attempts are being made to determine delayed neutron spectra for the Keepin six group representation(1) by performing experiments with modulated beam cycles of different beam-on and beam-off periods, giving emphasis to different delayed neutron groups through the alteration of the irradiation to counting time ratios. However, it became apparent that the process of extracting group spectra from a small number of experimental runs with different cycle times would lead to unacceptably high statistical uncertainties on the derived group spectra. The same problem was also being experienced by other groups(2).

However, the development of a response function representation for our ^{3}He spectrometers (UKNDC(86)P113, p.54) which removes the need for collection of both pulse-height and pulse rise-time data, has made it possible to utilise the second parameter of our dual-parameter data acquisition system to record the time of arrival of a delayed neutron pulse. Thus a single experiment can provide information on the wide variety of delayed neutron emission periods simply by taking into account the temporal information available from the second parameter of the counting system. The first parameter remains as before the energy spectrum of the ^{3}He counts. A number of experiments of this type have been carried out. The bombarding neutrons have been derived from the $^{9}Be(d,n)$ reaction using a thick target because this provides an excellent neutron yield per incident deuteron. Earlier work had indicated that there is little change in the delayed neutron spectrum with the energy of the bombarding neutron (UKNDC(86)P113, p.54) so little adverse effect was expected from using a source spectrum which was not monoenergetic.

An iterative technique has been employed to analyse the data(3). This first estimates the group spectra, calculates the corresponding pulse height

(1) G. R. Keepin, T. F. Wimett and R. K. Zeigler, Phys. Rev., 107 (1957) 1044.
(2) G. P. Couchell and K-L. Kratz, Private Communications.
(3) S. J. Chilton, Birmingham University Ph.D. thesis (1987) to be published.

spectra for the various time slices in the time of arrival two-parameter data sets, and then minimises the differences. This has the effect of determining the absolute ratios of the groups to each other. Using this information a second iteration refines the estimate of the group spectra. It was not possible to separate all the six groups and initial analysis was undertaken with five groups only, the fifth combining the effects of the fifth and sixth Keepin groups. Even so, there was still a tendency to unstable solutions and the problem was debated at length at the Specialists Meeting on Delayed Neutron Properties, held in September 1986 at Birmingham[1]. The result of this debate was a concensus that some form of constrained fitting procedure was appropriate and should be investigated. The results of the current unconstrained iterative process for one of the ^{3}He spectrometers are shown in Figure 5.1 where the contributions from groups 4, 5 and 6 have been combined in order to produce a stable solution.

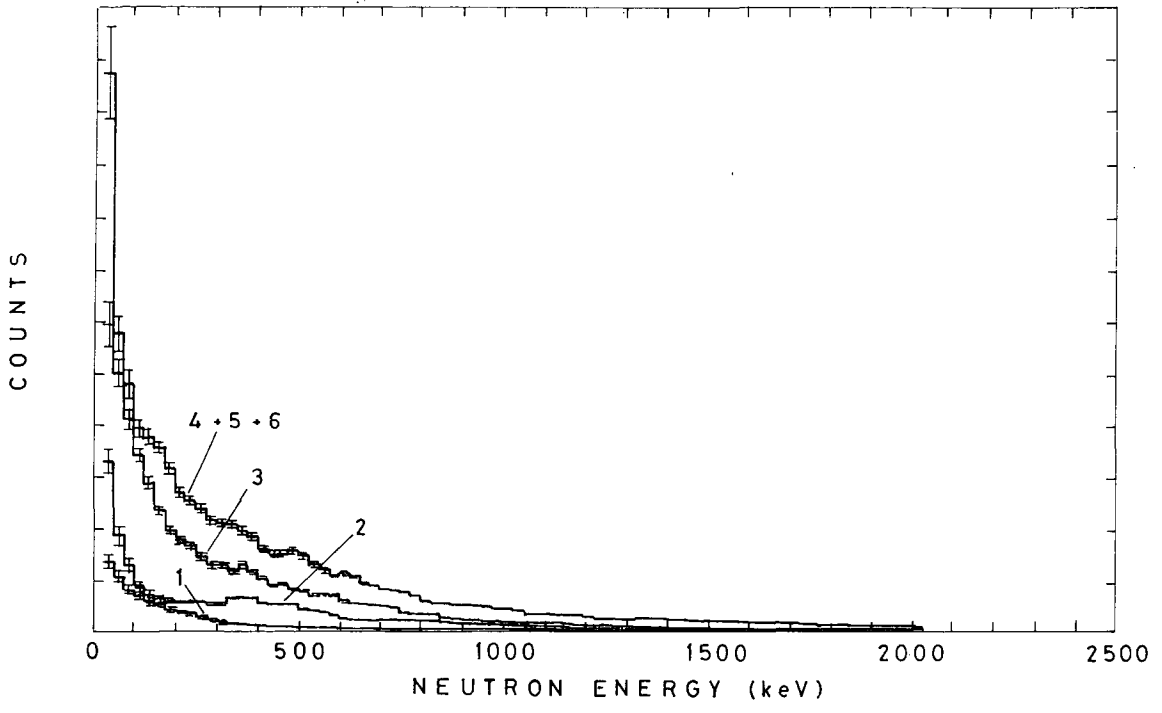

Figure 5.1 The best fit pulse-height spectra for groups 1-3 and 4+5+6 for the Seforad detector total response. The scale is arbitrary but shows the relative amplitudes of the contributions.

(1) S. J. Chilton, D. R. Weaver, J. Walker and J. G. Owen, Proceedings of the Specialists Meeting on Delayed Neutron Properties, Birmingham, September 1986. To be published as a University of Birmingham Report.

6. DEPARTMENT OF PHYSICS, UNIVERSITY OF EDINBURGH

6.1 The analysing power and differential cross-section for elastic scattering by 3 MeV neutrons (R. B. Galloway)

The results of recent investigations are due to be published shortly[1]. The analysing powers and the differential cross-sections of Cd, Sn, Sb, Te and I for the elastic scattering of 3 MeV neutrons have been measured at 7 degree intervals over the angular range 20 to 167 degrees. The scattering samples were of natural isotopic abundance. Comparison was made with the results of optical model and Hauser-Feshbach calculations based on parameters recently proposed[2] for low-energy neutron scattering in this mass range. The parameters which give the best fit to the data were also found. It was concluded that the spin-orbit potential in this region is less deep and less diffuse than that usually employed.

Measurements are continuing on In and lighter samples.

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(1) R. B. Galloway and H. Savaloni, J. Phys. G, in press.
(2) A. B. Smith, P. T. Guenther and J. F. Whalen, Nucl. Phys. A415 (1984) 1.

7. NUCLEAR PHYSICS LABORATORY, UNIVERSITY OF OXFORD

7.1 Neutron inelastic scattering

7.1.1 The inelastic scattering of neutrons by ^{232}Th (A. M. Street and P. E. Hodgson)

This work, described last year (UKNDC(86)P113, p.67), has now been published in Nuclear Science and Engineering 92 (1986) 459.

7.2 Neutron reaction cross-section calculations

7.2.1 Neutron scattering and reactions on ^{93}Nb from 1-20 MeV (D. Wilmore and P. E. Hodgson)

An analysis of neutron reactions on ^{93}Nb is in progress using the methods previously applied to ^{59}Co and described last year (UKNDC(86)P113, p.68).

7.2.2 Weisskopf-Ewing calculations of neutron-induced reactions (S. Ait-Tahar and P. E. Hodgson)

The cross-sections of several neutron-induced reactions on ^{55}Mn, ^{54,56}Fe, ^{59}Co, ^{58,60}Ni and ^{63,65}Cu have been calculated for energies below 20 MeV using the Weisskopf-Ewing theory and compared with experimental data. The total (n,p) and (n,α) cross-sections are generally well fitted, especially when they are dominant channels. At the higher energies the (n,p) cross-sections have important contributions from pre-equilibrium processes, and these have been fitted using the theory of Feshbach, Kerman and Koonin. A paper describing this work has been accepted for publication in Journal of Physics G.

7.2.3 Weisskopf-Ewing and pre-equilibrium calculations of nuclear reaction cross-sections (S. Ait-Tahar)

This work formed the subject of an M.Sc. thesis submitted in 1986 with the following abstract:

This thesis is devoted to the study of nuclear reactions using the Weisskopf-Ewing theory and the exciton model and to finding a convenient way to calculate nuclear reaction cross-sections. The validity of the Weisskopf-Ewing theory is tested by comparing its predictions with experimental observations

over a wide range of nuclei and energies. We consider reactions induced by neutrons and protons in the 1-20 MeV energy range and those induced by alpha particles in the 1-30 MeV energy range. We find that this theory presents some shortcomings essentially at high incident energies where processes other than the evaporation process described by this theory become increasingly important. Consequently, the precompound processes have to be included in order to account for the experimental observations. In the present work these contributions are evaluated within the framework of the exciton model. The combined model thus developed (Weisskopf-Ewing + exciton model) is tested for several reactions and it is found that its predictions are in good agreement not only with the excitation functions of various reactions but with the neutron and proton spectra as well. However, in this model the description of the nuclear reaction is purely phenomenological and the compound and the precompound processes are combined in an ad-hoc way. It is thus important to know to what extent this model compares to a more fundamental theory and in particular to the quantum mechanical theory of Feshbach, Kerman and Koonin (FKK). The statistical multistep compound (SMC) model of this theory is used to analyse some spectra and excitation functions of reactions induced by neutrons on ^{59}Co and ^{58}Ni and the results for the neutron emission spectrum from the n + ^{59}Co reaction are compared with those obtained from the present model. The two predictions are found to be in reasonable agreement with each other and also the details of the two spectra do not differ appreciably. It is emphasised that an important advantage of the present model is its convenience for practical purposes. The calculations are very fast and a large number of spectra and cross-sections in a wide incident energy range can be obtained using a very small amount of computing time.

A systematic analysis of the cross-sections of the (n,p) reactions around 14 MeV calculated using the evaporation model is carried out. From this study an approximate formula for the (n,p) excitation function is derived where this latter is expressed in terms of a parameter related to the neutron excesses of both the target and the residual nucleus. This simple expression is compared with other formulae and is found to give a better fit to the experimental data.

7.3 Pre-equilibrium processes

7.3.1 Pre-equilibrium processes in nuclear reactions (P. E. Hodgson, G. M. Field, H. Gruppelaar* and P. Nagel**)

Work on this topic described in UKNDC(85)P112, p.185 was presented at the International Conference on Nuclear Data for Basic and Applied Science at Santa Fé (May 1985) and has subsequently been published[1,2].

A review article entitled "Pre-Equilibrium Processes in Nuclear Reaction Theory: The State of the Art and Beyond" has also been published[3] with the following abstract:

The semi-classical and quantum-mechanical theories of pre-equilibrium processes in nuclear reactions are reviewed, with special attention to the exciton, hybrid and unified models, and the Feshbach-Kerman-Koonin quantum mechanical theory. The results of the 'International Nuclear Model and Code Comparison on Pre-equilibrium Effects' carried out under the auspices of the NEA Data Bank are presented and the physical bases and approximations of the models are discussed. The quantum-mechanical theory is described and compared with the semi-classical models. Selected data for pre-equilibrium processes in ^{59}Co and ^{93}Nb are analysed using both the semi-classical models and the quantum mechanical theory and their reliabilities for fitting and prediction are compared. The future development of analyses of pre-equilibrium processes is discussed with particular reference to the scopes of the various models and their optimum areas of applicability.

A paper on some aspects of this work was also presented at the 4th International Symposium on Neutron-Induced Reactions at Smolenice in Czechoslovakia (1985) and appears in the proceedings of that meeting.

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(1) P. E. Hodgson, G. M. Field, H. Gruppelaar and P. Nagel, Radiat. Eff. 95 (1986) 27

(2) P. E. Hodgson, G. M. Field, H. Gruppelaar and P. Nagel, Proc. Int. Conf. Nuclear Data for Basic and Applied Science, Santa Fé (Gordon & Breach) Vol. II (1986) p.1033

(3) H. Gruppelaar, P. Nagel and P. E. Hodgson, La Rivista del Nuovo Cimento 9 (1986) 1

* Netherlands Energy Research Foundation, ECN, Petten, Netherlands

** NEA Data Bank, Gif-sur-Yvette, France

7.3.2 Pre-equilibrium processes in the reactions of neutrons on ^{59}Co and ^{93}Nb (G. M. Field, R. Bonetti* and P. E. Hodgson)

The work on this topic described in UKNDC(85)P112, p.104 has been published[1].

The calculated cross-sections of all the more important reactions of neutrons on ^{59}Co and ^{93}Nb have been compared with the experimental data. The optical model has been used to obtain the differential elastic scattering cross-section and the total and reaction cross-sections, and the Weisskopf-Ewing, Hauser-Feshbach and Feshbach-Kerman-Koonin theories for the compound and the statistical multistep compound contributions to the neutron emission cross-section. Several other reactions, in particular the (n,2n), (n,p) and (n,α) have also been studied using one or more of these theories. Aspects of this work have been presented in invited papers to the 4th International Conference on Nuclear Reaction Mechanisms in Varenna (1985) and the Specialists Meeting on the Use of the Optical Model for the Calculation of Neutron Cross-sections below 20 MeV in Paris (1985) both of which have subsequently been published[2,3]. In addition, a contributed paper to the Paris meeting summarised the multi-step compound pre-equilibrium theory and presented comparisons with experimental data for neutron inelastic scattering and reactions on ^{59}Co and ^{93}Nb at 14 MeV. This paper has also been published[4].

7.4 Calculation of Optical Potentials (A. M. Street)

Optical potentials and the corresponding elastic scattering cross-sections are being calculated from a microscopic first-order Brueckner theory using self-consistent fields in a Hartree-Fock method.

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(1) G. M. Field, R. Bonetti and P. E. Hodgson, J. Phys. G 12 (1986) 93
(2) P. E. Hodgson, Ricerca Scientifica ed Educazione Permanente Supplement 46 (1985) 1
(3) P. E. Hodgson, OECD Report NEANDC-222U (1986) 117
(4) R. Bonetti and P. E. Hodgson, OECD Report NEANDC-222U (1986) 267
* Istituto di Fisica Applicato Generale, Milano, Italy

To test the computer program the elastic scattering of 30.3 MeV protons by ^{40}Ca is being calculated. This will enable the effect of long-range Coulomb terms on the scattering cross-section and on the imaginary potential to be investigated.

It is planned to study the isospin dependence of the potential, and the problems of complex binding energies, the non-zero temperature approximation and over-counting in the Born approximation. These calculations should provide a reliable method of determining optical potentials for reaction calculations.

It is intended to extend this work to neutron reaction calculations.

8. UK NUCLEAR DATA FORUM 1986

The nineteenth UK Nuclear Data Forum took place at the Harwell Laboratory on Tuesday 14th October 1986. Summaries of the three invited papers, The Origin and the Present Status of the JEF Project (B. H. Patrick and M. G. Sowerby, Harwell Laboratory), The Use of JEF-1 Data for Decay Heat Evaluation (A. Tobias, CEGB Berkeley) and The Validation and Benchmark Testing of JEF-1 (J. L. Rowlands, AEE Winfrith) are reproduced in this report (8.2.1 - 8.2.3). The nine contributed papers are listed below (8.1). The Forum closed with an entertaining and informative talk by Professor J. M. Reid (Glasgow University) entitled "Four Decades - a rapid review of the forty years in which 'nuclear' has slipped from being an adjective of high approval to being a near term of abuse".

8.1 Contributed papers

A Survey of the Specialists Meeting on Delayed Neutron Properties, September 15-19, 1986, University of Birmingham.
D. R. Weaver (Birmingham University)

A Group Analysis of the Delayed-neutron Spectrum from the Fast-fission of ^{235}U.
S. J. Chilton, D. R. Weaver, J. Walker and J. G. Owen (Birmingham University)

Gamma-ray Spectra Following Resonance Neutron Capture in ^{58}Ni and ^{60}Ni.
J. P. Mason (Harwell Laboratory)

The $^{19}F(p,\alpha\gamma)$ Reaction as a Source of Photons in the Binding Energy Region for Reactor Dosemeter Instrumentation Calibration and Benchmark Experiments.
S. Croft (Birmingham University)

A New Library of Fission Yields UKIFYA1.
M. F. James and J. Banai (AEE, Winfrith)

Proposed New Data Base for Neutron Induced Fission Product Yields (part of a project funded by the CEGB, BNF and UKAEA)
J. Banai (AEE, Winfrith)

Decay Studies of ^{237}Np.
S. A. Woods and W. Gelletly (Manchester University), P. Christmas, P. Cross and S. M. Judge (NPL)

Modelling the Double-differential Cross-section for the $^{9}Be(n,2n)2\alpha$ Reaction.
T. D. Beynon and B. S. Sim (Birmingham University)

Systematics of Some (n,x) Reactions at about 14.5 MeV.
R. A. Forrest (Harwell Laboratory)

8.2 Invited papers

8.2.1 The origin and present status of the Joint Evaluated File (JEF) project (B. H. Patrick and M. G. Sowerby, Nuclear Physics Division, Harwell Laboratory)

Origin of JEF:

Up to the mid-1970s, the evaluated nuclear data files, produced by the fission reactor communities in a number of countries, were freely available but the situation changed about that time. Following a major initiative by the USA to produce ENDF/B-V, it was decided that only certain parts (standards, fission products, dosimetry, etc.) would be released; the evaluations of the main materials being withheld. This situation continued and by 1980 there was still no timetable for the general release of ENDF/B-V. Partly as a result of heavy reliance on USA data files, the European nations found themselves in a very fragmented situation so far as evaluation activities were concerned at this time.

In 1980, the Nuclear Energy Agency Committee on Reactor Physics (NEACRP) asked the NEA to convene a meeting of European and Japanese members of NEACRP, together with nuclear data experts, to discuss possible co-ordination of evaluation activities in these countries. It is to be noted that this initiative had the full support of the USA members of NEACRP. The meeting was held in November 1980, the objectives being

- to agree on a common format for evaluated files
- to discuss common evaluation procedures
- to establish a mechanism for selecting evaluations, and co-ordination and review of new ones
- to establish a common data file for fission reactors

The meeting concluded that a European-Japanese evaluated data library should be established and, from this, JEF was born. It was agreed that the ENDF/B-V format was the most appropriate and, in order to get off the ground as quickly as possible, a starter file would be assembled for the most important materials by selecting the best existing evaluated files, based on agreement with experimental data.

Organisation of JEF:

The co-ordination of the work was to be carried out by a management group, consisting of representatives from actively participating member countries and from the nuclear data users. The focus of the work was established at the NEA Data Bank where the starter file was assembled during 1981, containing chosen evaluations on the principal structural materials and major actinides. As a result of the successful completion of this initial phase, a formal application was made to the NEA Steering Committee in 1981 to establish a Joint Programme on Neutron Data Evaluation and this was approved. All NEA Data Bank member countries were invited to participate and the management group was formalised by the formation of the Scientific Co-ordination Group (SCG).

At the present time, countries represented on the SCG are UK, France, West Germany, The Netherlands, Italy, Sweden, Switzerland and Japan; in addition, the EURATOM laboratory at Geel participates. The JEF file is made available to all NEA Data Bank member countries, but outside that area, there has been no distribution so far.

Details of JEF programme of work:

The programme of work for JEF can be divided into essentially 2 stages. In the first, which has been completed, the existing and available evaluations were reviewed and the best one for each nuclide or material was selected for inclusion into the first version of the JEF library (JEF-1). In the second stage, which is currently going on, benchmark tests of JEF-1 are being undertaken to identify weaknesses in the data and lead to a programme of new and revised evaluations for the JEF-2 library.

The timescales for the work are as follows:

JEF-1:

The general purpose file (i.e. neutron cross-sections) was distributed in April 1985.

The special purpose file (decay data, fission yields, scattering law data, photon interaction data) was distributed September 1985. Some revisions to the decay and fission yield data were frozen in July 1986.

A revised version of JEF-1, which includes minor changes to cross-sections, and is called JEF-1.1, was issued in Autumn 1986.

JEF-2:

The JEF-2 library is due for completion at the end of 1988 and this implies that the evaluations should be finished at the end of 1987.

Contents of JEF-1:

JEF-1 contains the following information:

(a) neutron cross-section data for 301 isotopes or elements
(b) decay data for 1130 isotopes
(c) scattering law data for H in H_2O, D in D_2O, H in CH_2 and C as graphite
(d) fission yield data for ^{232}Th, ^{233}U, ^{235}U, ^{238}U, ^{239}Pu, ^{240}Pu and ^{241}Pu
(e) photon interaction data (from NBS) for Z = 1 to 100.

JEF-1 is comparable in quality to other modern data libraries such as ENDF/B-V and JENDL-2. In some areas, particularly where new evaluations have been used, the quality is particularly high.

Work for JEF-2:

The JEF-2 library will probably be produced in ENDF/B-VI format. It is hoped to use the ENDF/B-VI standard cross-sections in the library so as to maintain compatibility with ENDF/B-VI. A number of new and revised evaluations are being produced for inclusion in the library. Both JEF-1 and JEF-2 are primarily designed for use in fission reactor programmes. The European Fast

Reactor collaboration will use JEF as the source for their nuclear data. Requirements for further experimental work will come through the JEF project.

Data for fusion:

In parallel with the JEF project is the European Fusion File (EFF). A library of cross-section evaluations is being produced for fusion blanket design for the Next European Torus (NET) project. The work is part of the fusion technology programme of the CEC; the B2 task - development of tools for neutronic calculations. The main contractors for EFF are Saclay, Petten, Bologna and Karlsruhe.

The EFF takes files from JEF and other sources and improves them for fusion applications (e.g. improving the data at ~14 MeV by taking into account pre-equilibrium effects and using a double differential representation). New evaluations are also being performed. The new and improved file will be fed back into JEF. The first version of EFF (EFF-1) was distributed early in 1986 and it contains the data for 26 materials. The EFF programme is a continuing one.

The following references give more information about JEF and EFF:

(1) "The Joint Evaluated File: A New Nuclear Data Library for Reactor Calculations", J. L. Rowlands and N. Tubbs. Proc. Santa Fé Conference on Nuclear Data for Basic and Applied Science, Gordon and Breach, New York, p.1493 (1986).

(2) "Status of European Fusion File", H. Gruppelaar. IAEA Advisory Group Meeting on Nuclear Data for Fusion Reactor Technology, Dresden, December 1986.

(3) The up to date status is always given in the Neutron Nuclear Data Evaluation Newsletters published by the NEA Data Bank (the latest is NNDEN/38 published July 1986).

(4) Further information on the projects can be obtained from the NEA Data Bank (JEF)and H. Gruppelaar (ECN Petten, The Netherlands) (EFF).

8.2.2 The use of JEF-1 data for decay heat evaluation (A. Tobias, Technology Planning and Research Division, CEGB, Berkeley Nuclear Laboratories)

The data requirements, in terms of decay data and fission yields, for decay heat calculations have been examined and the evaluation procedures used are described. The data available in the recent calculations are summarised and their use in decay heat calculations has been studied. From comparisons of JEF-1 decay heat predictions with both UK data predictions and measured data it was found that:-

1. The exclusion from JEF-1 of short lived 'theoretical' nuclides, previously in the UK decay data library, had only a relatively small effect on decay heat predictions at short cooling times following fission pulses. The corresponding effect on integral decay heat predictions is negligible.

2. A change in fission yield data, from UK to JEF-1, resulted in differences of up to a few percent in decay heat predictions for a wide range of cooling times following fission pulses. The differences were attributed not only to significant changes in independent fission yields (at short cooling times) but also to significant changes in mass chain yields (at longer cooling times).

3. The use of JEF-1 decay and fission yield data resulted in additional small differences in fission pulse predictions at short cooling times. On the whole, the differences in prediction between JEF-1 and UK data could be attributed largely to the change in fission yield data. Changes in decay data had a less marked effect on decay heat predictions. It was also noted that JEF-1 decay heat predictions are more complete than those with current UK data files in terms of the gamma spectra calculated at short cooling times.

4. Discrepancies at cooling times of less than 100 s were noted between JEF-1 fission pulse predictions and corresponding values derived from measurements. These can be attributed to the acknowledged deficiencies in decay data for short lived fission products. Underpredictions of 5-14% in total decay heat were also observed at a cooling time of 1000 s. The

magnitude of these discrepancies is such that they are more likely to be due to errors in decay data for a small number of nuclides than to errors in fission yields. The principal nuclides contributing to the decay heat at this time were identified as ^{89}Rb, ^{93}Sr, ^{94}Y, ^{95}Y, ^{101}Mo, ^{102}Tc, ^{104}Tc, ^{105}Tc, ^{130}Sb, ^{131}Sb, ^{133}Te, ^{134}I, ^{137}Xe, ^{138}Xe, ^{138}Cs, ^{139}Cs, ^{141}Ba, ^{142}Ba and ^{143}La.

It was noted that for a number of these nuclides the decay data had been taken from single or relatively few measurements and also that a number of them had average gamma energies that were relatively small compared to the Q values. Recognised problems in decay scheme measurements suggest that this group of nuclides offers the greatest scope for revision should their decay schemes be in error. Further experimental studies of these nuclides are desirable in order to resolve the observed discrepancy. Another, but less likely, cause of this discrepancy would be a very large error in the half life of some other short lived fission product.

A full account of this work is available as a CEGB report (TPRD/B/0863/R86) with the same title as this Data Forum paper.

8.2.3 The validation and benchmark testing of JEF-1 (J. L. Rowlands, AEE Winfrith)

Introduction

The validation and benchmark testing of JEF-1 are being carried out in stages. An extensive series of tests was made prior to the release of JEF-1 and these are summarised in Table 8.1. Following on from these tests more extensive studies are being made with the aim of guiding the re-evaluation work necessary for JEF-2 and assessing more carefully the accuracy of predictions made using JEF-1 and, eventually, JEF-2.

The evaluations of the capture cross-sections for the most important fission product isotopes have taken account of measurements made in fast reactor spectra. This is also the case for fission and capture in ^{241}Am and ^{243}Am.

Table 8.1

Validation and benchmark testing of JEF-1 prior to release in 1985

(a) Format consistency checking.

(b) Graphical display.

(c) Intercomparison with data from other sources
  (i) 2200 m/sec
  (ii) thermal Maxwellian
  (iii) resonance integral
  (iv) ^{235}U fission spectra
  (v) 14 MeV values

(d) Fundamental mode calculations in which just $\sigma_{infinity}$ was replaced in the KFK-INR fast reactor set. (62 fast criticals calculated) ($\delta k_{eff} \lesssim 1\%$ excepting for ^{238}U σ_c).

(e) Analysis of reactivity worth measurements and reaction rate ratios for fission products (e.g. Gruppelaar et al).

(f) k_∞, k_{eff} and central reaction rate ratio measurements for simple geometry systems:
  (i) infinite media (homogeneous and simple lattice cells)
  (ii) homogeneous spheres (e.g. Los Alamos criticals and ORNL uranyl nitrate solutions)
  (iii) approximate equivalent homogeneous spheres (with heterogeneity and shape factor corrections) representing fast reactor critical assemblies

(g) Single material shielding benchmarks (e.g. iron).

Several different methods have been used to process the nuclear data for the analysis of benchmark experiments. Different energy group structures and different methods to treat resonance shielding have been used. These differences have resulted in discrepancies which indicate inadequacies in some of the methods. In some analyses not all of the data have been taken from the JEF-1 library. This is the case for the MONK Monte Carlo calculations in which only the data for the primary actinides have been taken from JEF-1 (and not the thermal scattering data).

In the first series of calculations the NEA Data Bank took an existing library, the Karlsruhe KFK-INR set, and changed the infinite dilution cross-sections, (but not the resonance shielding factors, nor scattering

matrices) to JEF-1 values. Calculations were made for fast reactor benchmarks changing the cross-sections for each substance in turn to the JEF-1 values and repeating the calculations. In this way it was found that JEF-1 performed no worse than the KFK-INR set and, on balance, somewhat better. Calculations were also made for benchmarks having neutron spectra in which resonance shielding could be neglected, or treated very simply. These also gave satisfactory results.

Fast reactor group cross-section sets for the first stage benchmark tests

The NEA Data Bank then produced group cross-section sets suitable for fast reactor benchmark calculations. Sets were produced in three different group structures:

(i) 71 group 1/4 lethargy width groups; (JEF1-71F). This is also available in 72 group form, with an extra group above 10 MeV.
(ii) the Cadarache 25 group 1/2 lethargy width group structure, (JEF1-25F).
(iii) the Karlsruhe 26 group ABBN structure (3/4 lethargy width groups), (JEF1-26).
(iv) a 50 group set intermediate between the 71 group and 25 group sets, (JEF1-50F).

A simple (flat in lethargy or 1/E dE) weighting spectrum was used. The resonance shielding factors were calculated assuming the narrow resonance approximation and tabulated for a range of background cross-sections, $\sigma_b = 10^n$. Anisotropic scattering was approximated by an equivalent isotropic scattering cross-section (the transport approximation).

It was found that there were significant differences between the results of calculations made using the three different sets, up to about 1.3% in k_{eff} (figure 8.1). These differences are attributed to the use of a flat weighting spectrum, rather than a typical fast reactor weighting spectrum, which results in significant errors, particularly for the sets with a broader group structure.

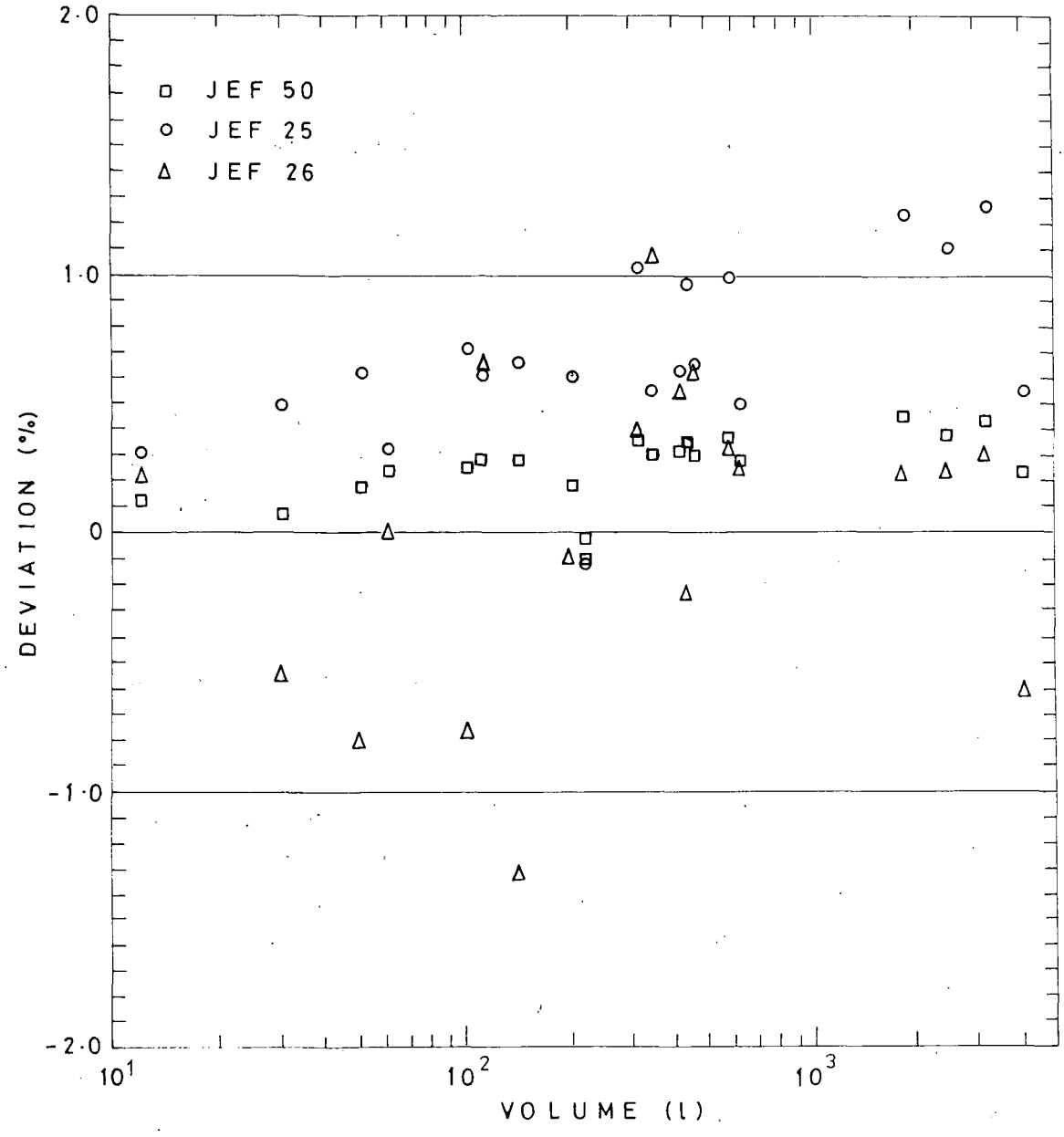

Figure 8.1 Deviation of k_{eff} for JEF1-50F, JEF1-25F and JEF1-26 from the ones for JEF1-72F.

Differences were also found between calculations made at the NEA Data Bank and at Winfrith using the 72 group set. These differences ranged up to about 1% in k_{eff}. It was concluded that these differences were attributable to differences in the method of interpolation in the tables of resonance shielding factors. The importance of this effect depends on the proportion of reactions in the ^{238}U resonance region.

It was concluded that the accuracy of calculations made using the 72 group set, as an approximation to the JEF-1 library, was about ±1%. Results obtained for a number of fast criticals are presented in figure 8.2. Taking into account the approximations made in the representation of the critical assemblies (homogeneous spherical models) and the approximations in the nuclear data processing, the results are satisfactory. More accurate representations and calculational methods are necessary before conclusions can be drawn about the need to improve particular items of nuclear data.

It was also concluded that for more accurate calculations either a finer group structure or a better flux weighting spectrum should be used and that the representation of resonance shielding should be improved.

Thermal reactor benchmark calculations

The JEF-1 thermal scattering data for H_2O, D_2O, graphite and polyethylene were produced by IKE Stuttgart. Their evaluation work included comparisons of calculation with neutron spectra measured at different temperatures for both the pure moderators and moderators admixed with neutron absorbers, (figures 8.3 - 8.6). Comparisons were also made with measurements of the thermal spectrum averaged diffusion coefficient and neutron diffusion length made at a range of temperatures (figures 8.7 and 8.8). The agreement is good.

IKE Stuttgart have also made calculations for a wide range of thermal reactor benchmarks. Their group cross-section library comprises three partly overlapping ranges:

151 thermal groups from 10^{-5} to 3.059 eV
8500 resonance region groups from 0.876 eV to 4.3 keV (lethargy width 10^{-3})
100 groups between 0.414 eV and 14.98 MeV (GAM-II structure)

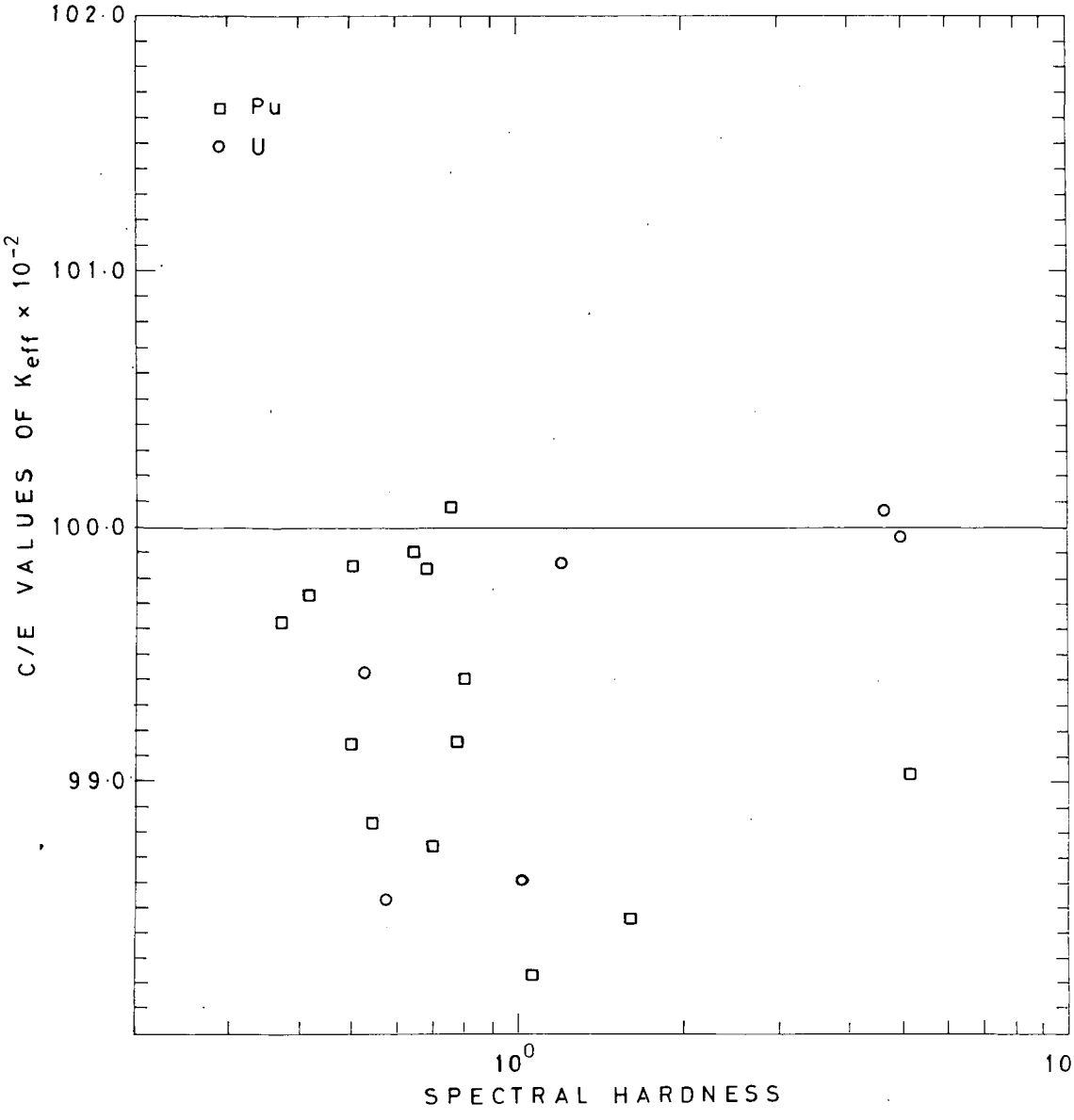

Figure 8.2 C/E-values of k_{eff} for JEF1-71F as a function of the spectral hardness.

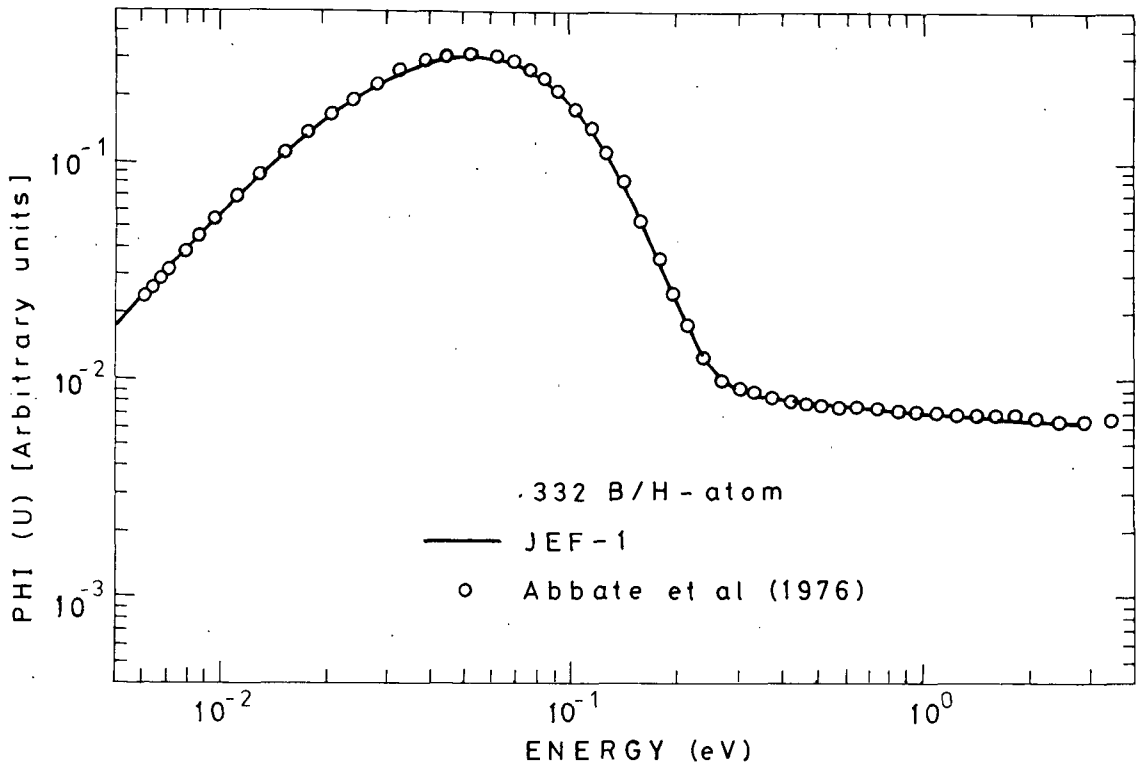

Figure 8.3 Infinite medium neutron spectrum in pure water (H_2O) at T = 23°C.

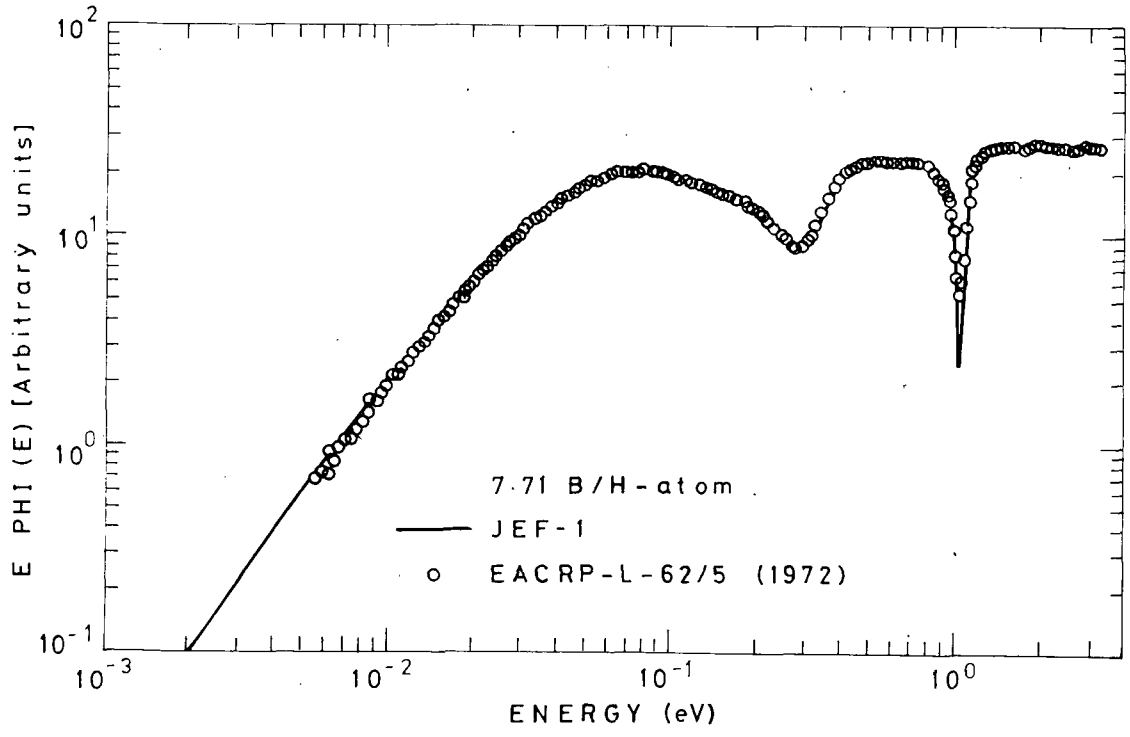

Figure 8.4 Infinite medium neutron spectrum in a plutonium nitrate solution (197.9 g Pu/ℓ - at 23 wt % ^{240}Pu).

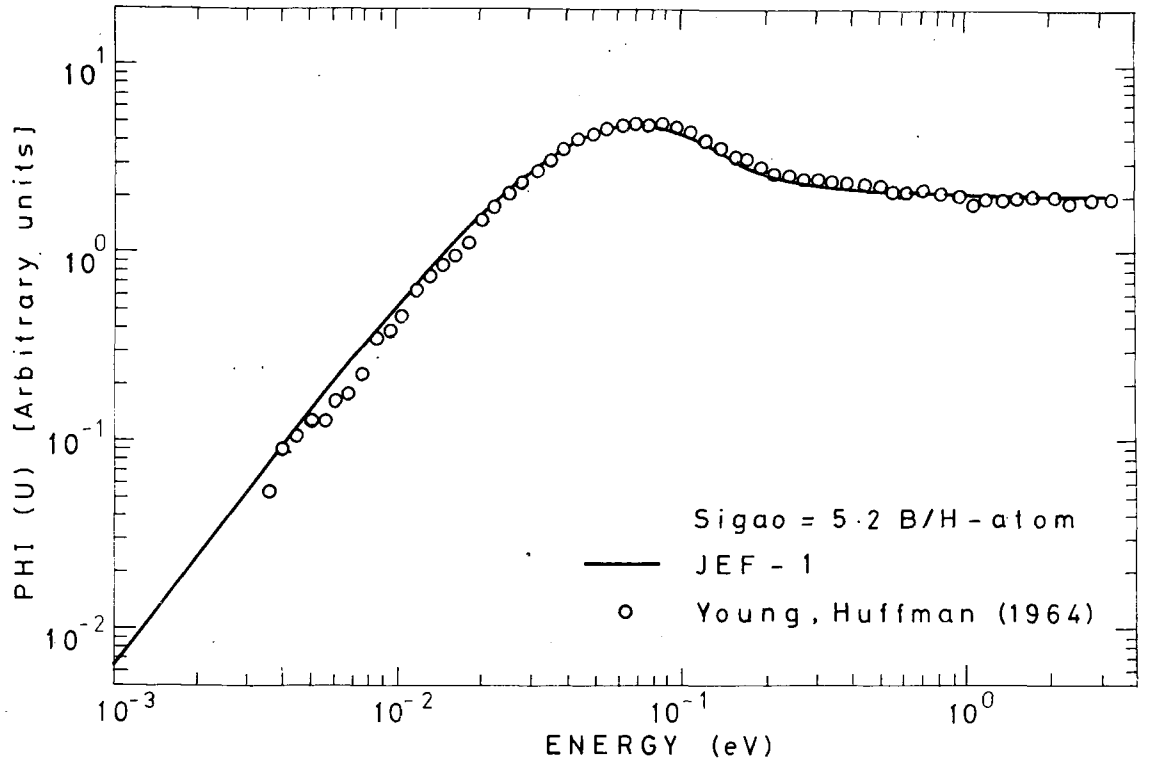

Figure 8.5 Infinite medium neutron spectrum in borated water (H_2O) at room temperature.

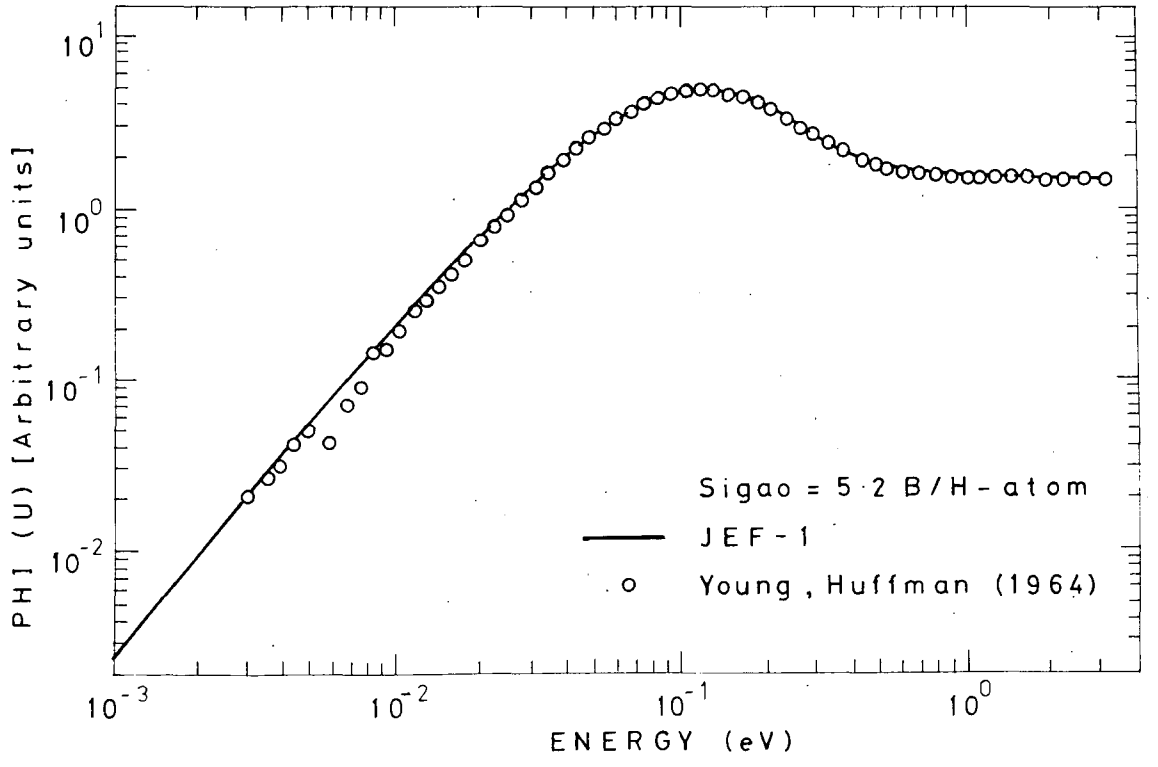

Figure 8.6 Infinite medium neutron spectrum in borated water (H_2O) at T = 316°C.

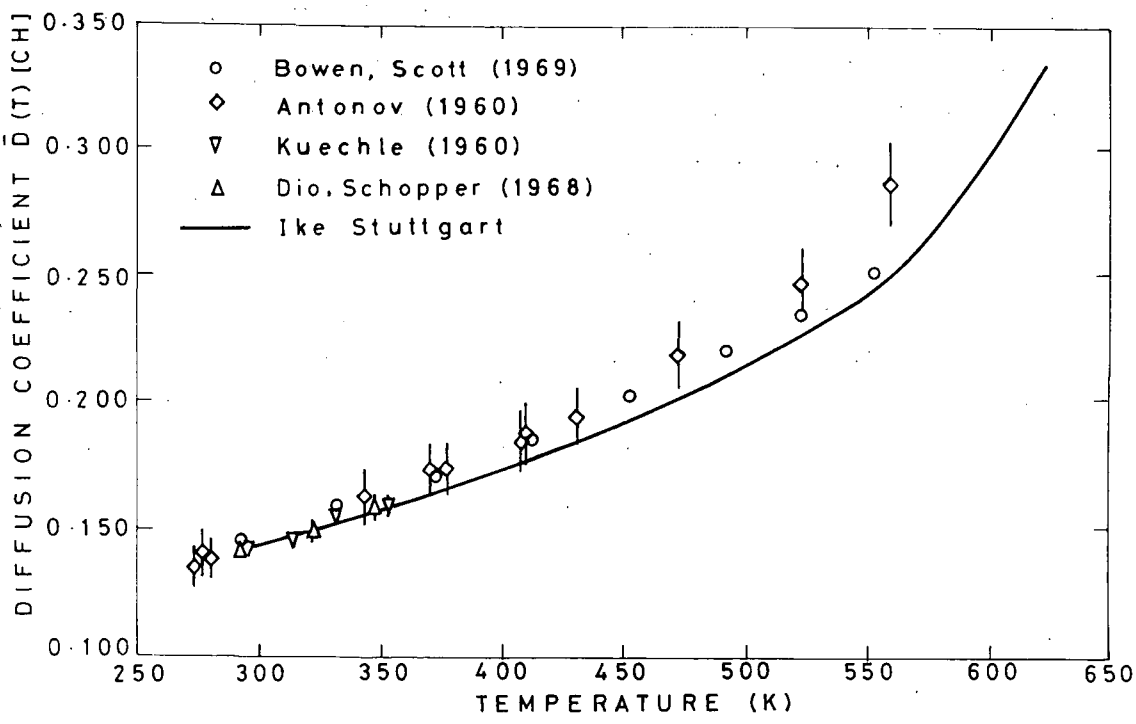

Figure 8.7 Neutron diffusion coefficient $\bar{D}(T)$ in water (H_2O).

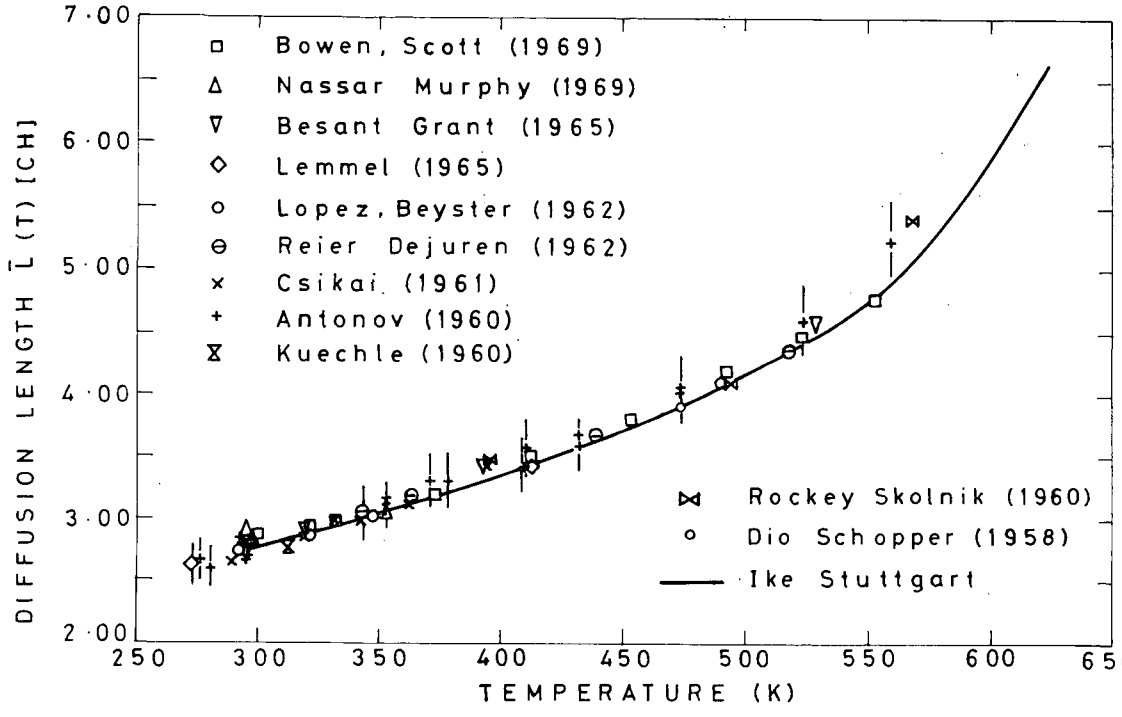

Figure 8.8 Neutron diffusion length $\bar{L}(T)$ in water (H_2O).

This GAM-II structure comprises 49 groups above 111 keV (0.1 lethargy width) and 51 groups between 0.414 eV and 111 keV (0.25 lethargy width). Three of these groups are below 0.876 eV, eight are below 3.059 eV and thirty eight are below 4.3 keV.

Anisotropic scattering is treated in the P3 approximation. Resonance shielding factors are also provided in the 100 group set and, for ^{238}U, in the 8500 group set. This level of detail in the representation of cross-sections is considered to be adequate for thermal reactor benchmark calculations. The narrow resonance approximation is assumed in the interpolation in the shielding factor tables (with heterogeneity taken into account via a background cross-section).

The cross-sections are first condensed to fewer groups (30 to 60) using a simplified representation of the system in the spectrum calculations (homogeneous or one-dimensional cell). This is then followed by a broad group transport theory calculation for the system, made using the one-dimensional Sn code ANISN.

The thermal benchmarks comprise five series of ^{235}U fuelled systems and two series of ^{239}Pu fuelled systems. These are:

(a) Uranyl nitrate solutions. ORNL-1,2,3,4,10. These are unreflected spheres of highly enriched uranyl nitrate in H_2O. Three of the solutions (ORNL-2, 3 and 4) contain boron, the decreasing N_H/N_{235_U} being compensated by an increase in boron fraction. The N_H/N_{235_U} ratios are in the range 972 to 1835. ORNL-10 is a larger sphere than ORNL-1 to 4 and has a larger N_H/N_{235_U} ratio.

(b) Lattices of uranium metal rods in water. TRX-1,2,3,4. The uranium enrichment is 1.3%. The moderator/fuel volume ratios are in the range 1.0 to 8.11.

(c) Lattices of uranium oxide rods in water. BAPL-UO_2-1,2,3. The uranium enrichment is 1.3% and the moderator/fuel ratios are in the range 1.43 to 2.40.

(d) Tight pitched lattices of uranium oxide in water. HI-C-UO_2-3,10,11,13. The uraniunm enrichment is 3%. The moderator/fuel ratios are in the range 0.42 to 1.4.

(e) D_2O moderated natural uranium lattices. MIT-1,2,3.

(f) Plutonium nitrate solutions. PNL-1,2,3,4,5,6,7,8 and 12. Seven of these are unreflected spheres and two (PNL-7 and 12) are water reflected. PNL-6 is a respecification of PNL-2.

(g) Mixed uranium-plutonium oxide lattices in water. PNL-30,31,32,33,34,36. The fuel contains 2% PuO_2. PNL-31, 33 and 35 contain boron absorber in the water.

In addition JEF-1 calculations are compared with spectrum averaged cross-section measurements (relative to ^{235}U fission).

(h) Actinide capture cross-sections measured by irradiation. The Sherwood assembly. Measurements are made for ^{235}U, ^{238}U, ^{239}Pu, ^{240}Pu, ^{241}Pu, ^{242}Pu, ^{241}Am and ^{243}Am.

The k_{eff} values calculated for the ORNL uranyl nitrate spheres are compared with experiment in figure 8.9. Percentage deviations between measured and calculated values for the uranium lattices moderated by water are given in Table 8.2 and k_{eff} values for the D_2O moderated lattices in figure 8.10. Values of k_{eff} for the homogeneous plutonium nitrate solutions are given in figure 8.11 and for the U-Pu oxide lattices in figure 8.12.

k_{eff} values for uranium fuelled thermal systems:

The JEF-1 calculated values are summarised in Table 8.2 and figures 8.9 and 8.11. They agree with experiment:

(a) to within ±0.15% for the 5 ORNL highly enriched uranium nitrate solutions, (typical s.d. on measurement ±0.2%)

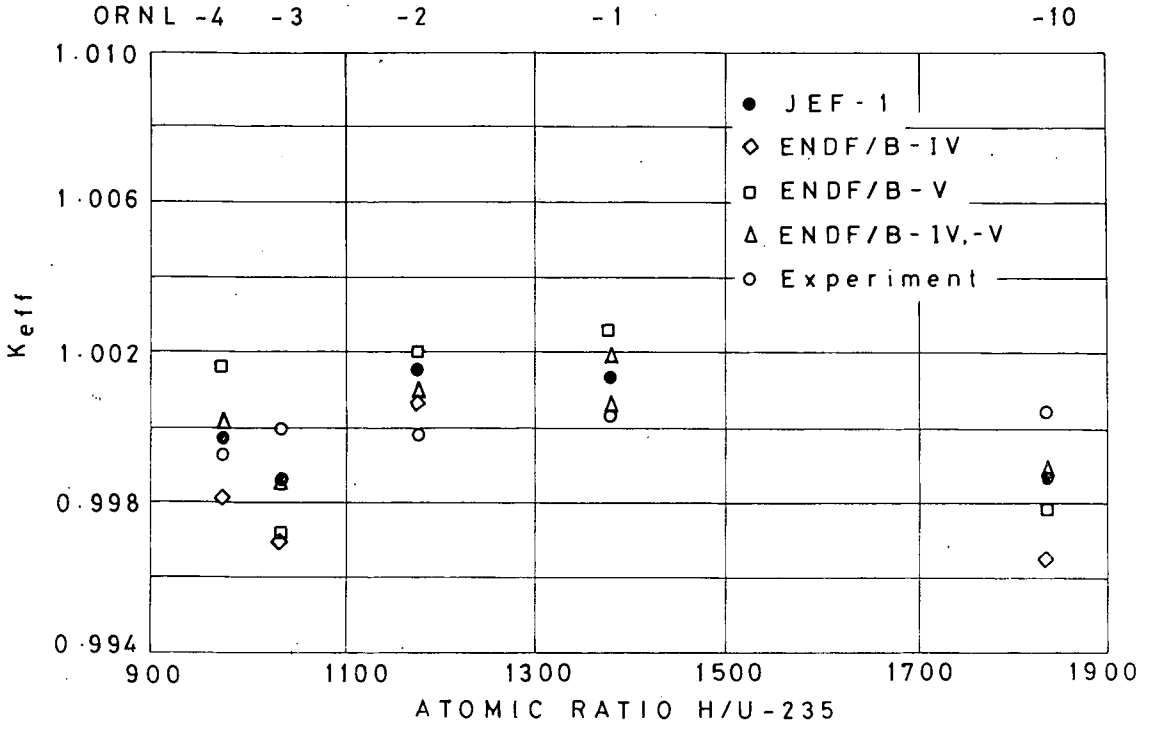

Figure 8.9 k_{eff} against the $H/^{235}U$ atomic ratio for homogeneous ^{235}U-H_2O benchmarks (ORNL).

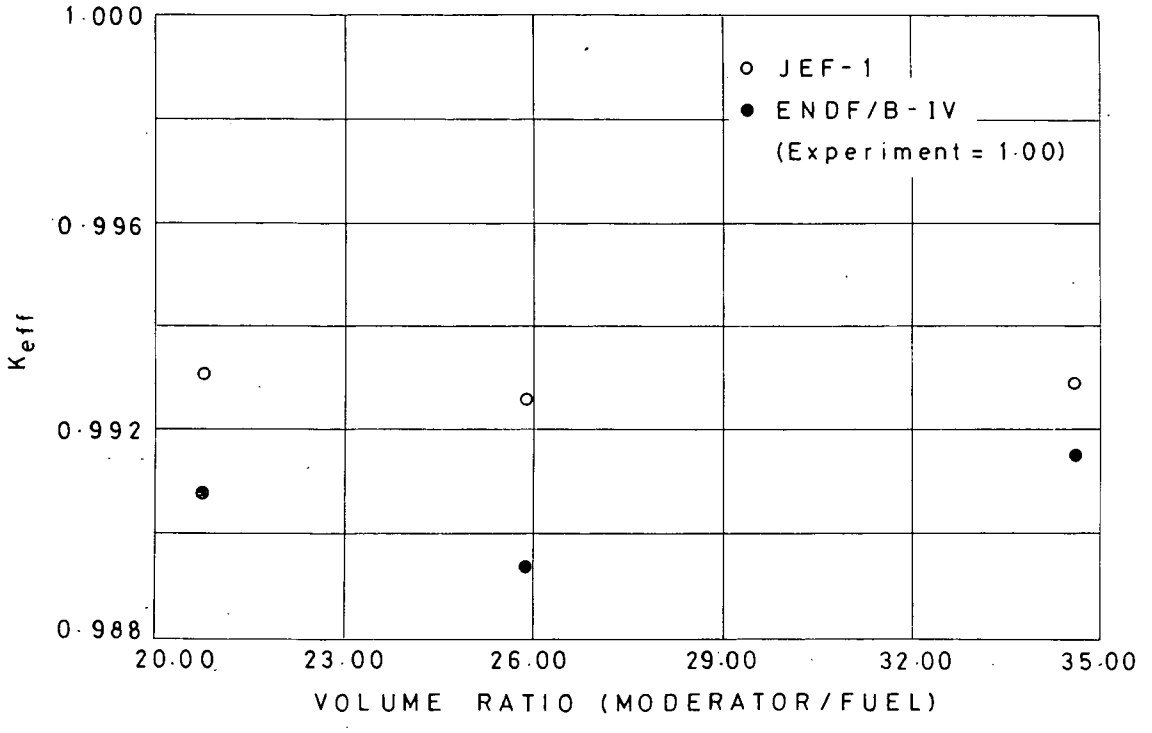

Figure 8.10 k_{eff} for the CSEWG heavy water benchmarks MIT-1,2,3.

Table 8.2

Percentage deviation of JEF-1 results from experiments for uranium lattices moderated by H_2O

Assembly	Parameter				
	Epithermal/Thermal C8	Epithermal/Thermal F5	F8/F5	C8/F5	k_{eff}
TRX-1	2.80 (1.59)	0.51 (1.01)	5.81 (4.33)	0.18 (1.00)	-0.40
TRX-2	0.05 (1.91)	-0.98 (1.30)	3.03 (5.05)	-1.19 (0.93)	-0.27
TRX-3	-0.56 (1.65)	2.16 (1.30)	3.29 (4.79)	-1.18 (0.88)	0.10
TRX-4	-1.04 (2.29)	-3.91 (1.40)	-1.87 (4.15)	-1.24 (0.75)	-0.21
BAPL-UO_2-1	2.13 (0.72)	0.12 (2.38)	-0.64 (5.13)	-	0.01 (0.07)
BAPL-UO_2-2	5.10 (0.89)	0.74 (1.47)	-4.57 (5.71)	-	-0.01 (0.06)
BAPL-UO_2-3	1.56 (1.10)	1.15 (1.92)	-3.68 (5.26)	-	-0.12 (0.05)
HI-C-3					0.15
HI-C-10					0.12
HI-C-11					0.16
HI-C-13					-0.01

Bracketed figures are experimental uncertainties

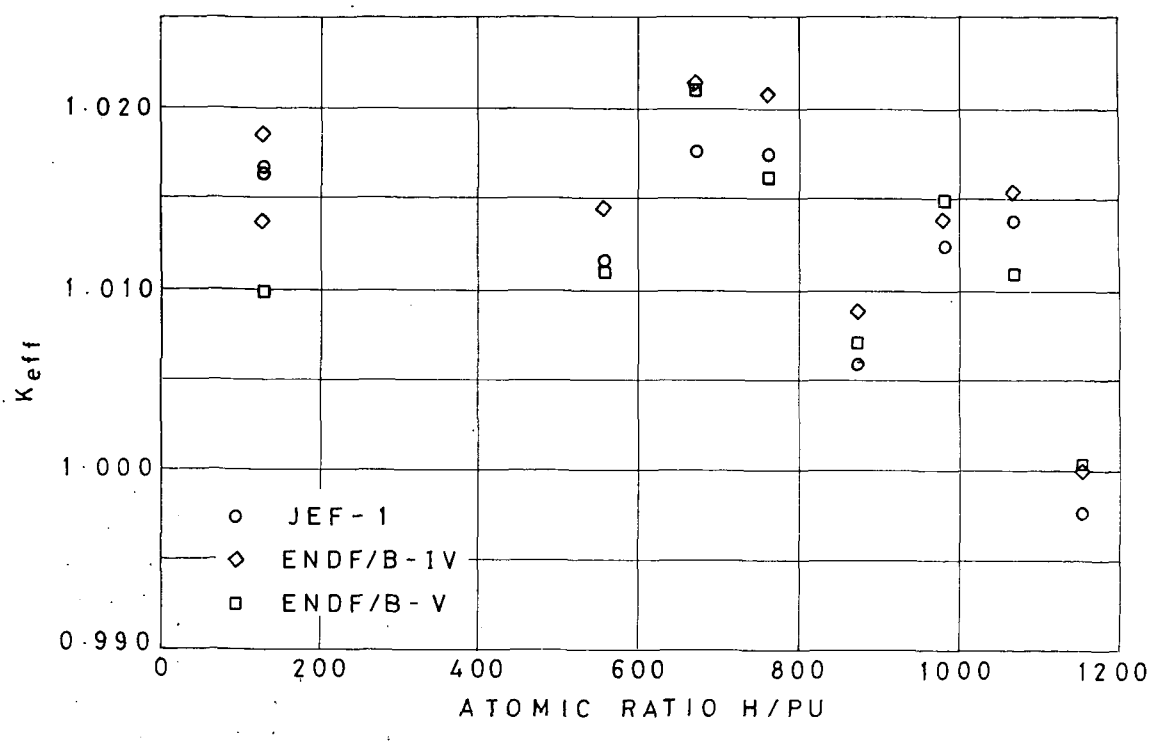

Figure 8.11 k_{eff} against the H/Pu atomic ratio for homogeneous Pu-H_2O benchmarks.

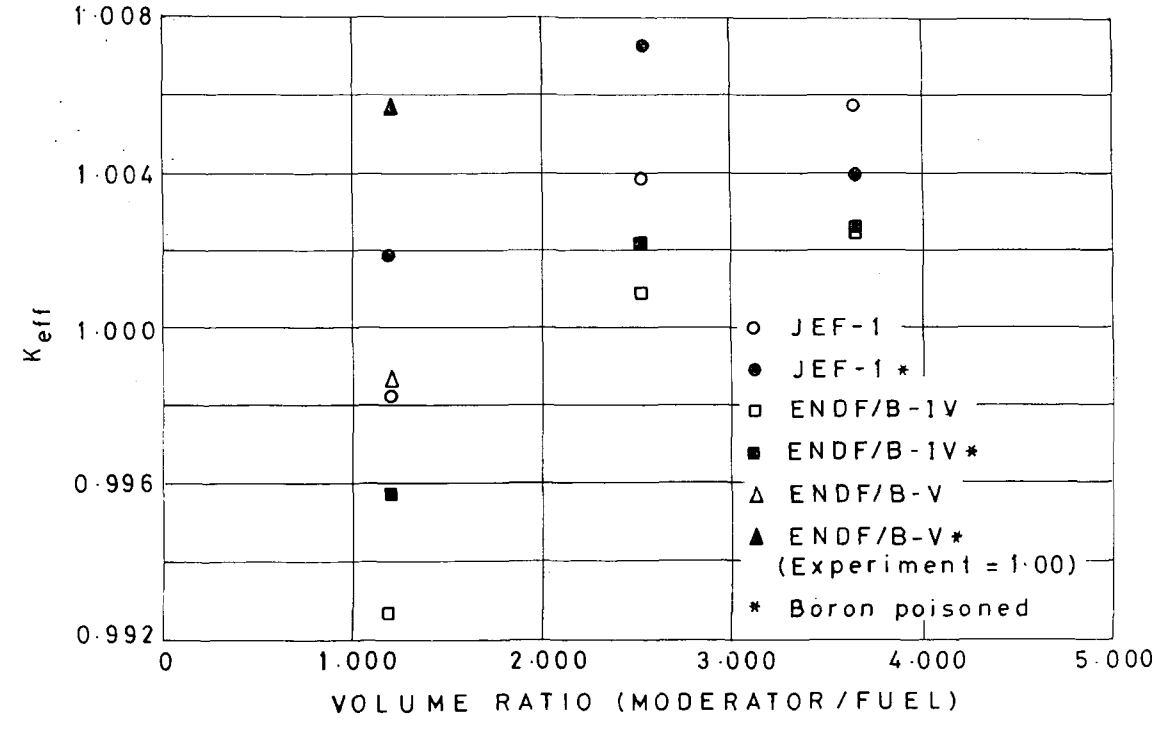

Figure 8.12 k_{eff} for the CSEWG and U-Pu-oxide benchmarks (UO_2 + 2 wt % PuO_2; 8% ^{240}Pu).

(b) to within ±0.4% for the 4 TRX H_2O uranium metal lattices, (typical s.d. on measurement ±0.3%), the average value of C/E being 0.998

(c) to within ±0.1% for the 3 BAPL lattices

(d) to within ±0.16% for the 4 HI-C H_2O-UO_2 tight pitched lattices

(e) for the D_2O moderated natural uranium metal fuelled lattices, MIT-1,2,3, k_{eff} is calculated 0.7% low

Ratio of ^{238}U capture to ^{235}U fission:

Measured values are only available for the TRX and MIT lattices. The largest difference between measurement and calculation is 2.9%. No difference exceeds 2 s.d. on the measurements.

Ratio of ^{238}U fission to ^{235}U fission:

Values are available for the TRX, BAPL and MIT lattices. The s.d. on the measurements is about ±5% and only one calculated value exceeds the s.d. on the measurements.

Epithermal/thermal ratios:

The ratio of the fraction of reactions occurring above 0.5 eV to the fraction occurring below has been measured for ^{238}U capture and ^{235}U fission in the TRX, BAPL and MIT lattice experiments. There are some large discrepancies (for BAPL-UO_2-2 ^{238}U capture, TRX-4 ^{235}U fission, and for MIT-2 and 3 both ^{235}U fission and ^{238}U capture). However, there is no obvious consistent pattern in that the discrepancies occur for both ^{238}U capture and ^{235}U fission for only two cores, MIT-2 and MIT-3, and not for all three of these similar lattices.

k_{eff} for homogeneous plutonium nitrate solutions:

The calculated values of k_{eff} for these nine PNL critical systems (figure 8.11) show a wide spread of values, from 0.996 to 1.017. However, it should be noted that PNL-6 is a respecification of PNL-2, which

gives a k_{eff} value about 0.5% different when calculated using ENDF/B-IV data. PNL-7 and 12 are spheres reflected by water and have similar H/Pu ratios. They differ in the ^{240}Pu fractions. The k_{eff} values differ by only 0.13%. However, the k_{eff} value for the unreflected sphere, PNL-3 which has a similar H/Pu ratio to PNL-12, has a markedly different calculated k_{eff} value, 0.9978 compared with 1.0139. Similar results are obtained for calculations made using ENDF/B-IV and B-V data.

Mixed UO_2-PuO_2 lattices:

The lattices PNL-30 to 35 form a series of three H_2O/fuel ratios each of which has two different levels of boron absorption in the water. The calculated k_{eff} values range from 0.998 to 1.006 for the versions with very little boron absorption and from 1.002 to 1.007 for the versions with boron absorber. The average calculated value of k_{eff} is 1.0035. The spread and the mean deviation from unity are thus much smaller than for the plutonium nitrate solutions.

Conclusions from the thermal benchmark studies:

For the chosen series of thermal reactor benchmarks it is only the plutonium nitrate solutions for which unsatisfactory results are obtained. There are some doubts about the reliability of these measurements and a more extensive series of plutonium fuelled benchmarks must be analysed before conclusions can be drawn.

Monte Carlo calculations for aqueous ^{235}U fuelled critical assemblies have been reported by P. F. Rose (Brookhaven National Laboratory). He made calculations using ENDF/B-V data. The evaluation for ^{235}U adopted in JEF-1 is the ENDF/B-V data and so these calculations will differ from the JEF-1 calculations only because of the different thermal scattering data. The range of $^{235}U/H_2O$ systems for which calculations have been made is wider than in the IKE Stuttgart study. In particular, the $H/^{235}U$ ratios are such that in several of the systems the fraction of ^{235}U fissions occurring above thermal energies is 20% or more whereas this fraction is only a few per cent for the $^{235}U/H_2O$ systems calculated in the IKE Stuttgart study.

The $^{235}U/H_2O$ systems calculated in the study made by Rose are 5 ORNL systems and 5 Rocky Flats systems. One of the Oak Ridge systems was included in both studies, ORNL-10. The values of k_{eff} obtained are 0.9986 in the IKE Stuttgart study and 0.9988±0.0176 in the Brookhaven study, a good consistency. For the systems with 20% to 45% of fissions above the thermal energy range the values of k_{eff} are about 1% higher than for the well thermalised systems. When the resonance region data are scaled to be consistent with the fission and capture resonance integrals in the Brookhaven compilation "Neutron Cross sections", by S. Mughabghab, the k_{eff} values become more consistent (within ±0.4% of unity). However, it is noted by Rose that if the Brookhaven evaluation of 2200 m/sec data made by Divadeenam and Stehn were adopted the k_{eff} values for the well thermalised systems would be reduced by 0.7% and the k_{eff} values would then be about 1% low.

Thermal capture cross-sections for actinide isotopes:

Capture cross-sections for actinides have been derived from the changes in compositions of actinide samples, following irradiation in a thermal reactor spectrum in the MELUSINE reactor at Grenoble. The measurements are relative to the ^{235}U fission cross-section. The JEF-1 calculated values for the uranium, plutonium and americium isotopes are compared with the measured values in Table 8.3. The largest discrepancy is 13% for ^{241}Am capture.

Table 8.3

Capture reaction rates for actinides measured in SHERWOOD assembly (irradiated in MELUSINE reactor, Grenoble/France) normalised to the fission rate of ^{235}U

Reaction Rate	Experiment	Calculation JEF-1	C/E
^{235}U absorption	1.233 ±0.017	1.224	0.993
^{235}U capture	0.233 ±0.003	0.224	0.961
^{238}U capture	0.01917±0.00014	0.0199	1.037
^{239}Pu capture	1.508 ±0.010	1.484	0.984
^{240}Pu capture	5.1 ±0.075	5.228	1.025
^{241}Pu capture	1.022 ±0.029	0.919	0.900
^{242}Pu capture	0.604 ±0.016	0.661	1.094
^{241}Am capture	3.185 ±0.065	2.762	0.867
^{243}Am capture	1.248 ±0.025	1.148	0.920

Calculations for high energy spectrum systems

Calculations have been made by EIR Wurenlingen for a series of thirteen systems with spectra having a high mean energy. These consist of nine Los Alamos criticals, two VERA criticals (VERA 1B and 11A), ZEBRA-3 and ZPR-III Assembly 12. Resonance shielding effects in uranium and plutonium isotopes are small in these systems and have not been treated in these calculations. The 174 group cross-section set (VITAMIN-E* structure) has been chosen to provide greatest detail at high energies. There are two groups below 0.414 eV and one-quarter lethargy width groups up to 2.035 keV. There is a progressively finer sub-division at higher energies, reducing from 1/4 to 1/10, with 1/20 and 1/40 lethargy width sub-divisions to represent important resonances (for example in oxygen and iron). The numbers of groups in decade intervals are:

1 keV - 10 keV	15 groups
10 keV - 100 keV	22 groups
100 keV - 1 MeV	43 groups
1 MeV - 10 MeV	46 groups
10 - 20 MeV	15 groups

Anisotropic scattering is treated in the P3 approximation and the one dimensional transport theory code, ONEDANT, is used (in S16 approximation).

The results are summarised in Table 8.4. For the ^{235}U and ^{239}Pu fuelled systems the calculated k_{eff} values agree with measurement to within 1.5%, the largest discrepancies being for the two VERA cores for which k_{eff} is underestimated by about 1.5%.

For the ^{233}U fuelled cores k_{eff} is underestimated by up to about 4%. For JEF-1 the ^{233}U cross-section data were selected from ENDF/B-IV. The ^{233}U evaluation in ENDF/B-V is a significant improvement over ENDF/B-IV but this evaluation is not generally available. The ^{232}Th data in JEF-1 are also from ENDF/B-IV and these have also been significantly improved in ENDF/B-V.

Comparisons have also been made for reaction rate ratio measurements made in these cores. For the ratio of ^{239}Pu fission/^{235}U fission the agreement is within ±3% (within ±1.5% excluding VERA-1B). For ^{238}U fission/^{235}U fission the spread in C/E values is larger, ±9%. The accuracy of these measurements is,

Table 8.4

Ratios of calculation to experiment for fast spectrum systems
(calculations made at Wurenlingen in P3, S16 approximation)

Name	Core	Blanket	k_{eff}	F9/F5	F8/F5
^{235}U fuelled cores					
GODIVA	^{235}U metal	bare	0.9995	1.008	1.060
FLATTOP25	^{235}U metal	U metal	0.9984	1.013	1.062
VERA-1B	^{235}U graphite	U metal	0.9847	0.973	0.971
BIG-TEN	10% ^{235}U	U metal	1.0016	1.003	1.032
ZPR-III-12	20% ^{235}U/graphite	U metal	0.9968	1.010	1.087
^{239}Pu fuelled cores					
JEZEBEL	Pu metal	bare	1.0095	0.988	0.976
JEZEBEL-Pu	Pu metal, 20% ^{240}Pu	bare	1.0024	-	0.981
FLATTOP-Pu	Pu metal	U metal	1.0054	-	0.984
VERA-11A	Pu/graphite	U metal	0.9860	1.014	0.922
ZEBRA-3	Pu/U	U metal	0.9953	1.006	0.989
THOR	Pu metal	Th metal	0.9923		0.999
^{233}U fuelled cores					
JEZEBEL-23	^{233}U metal	bare	0.9620	-	0.919
FLATTOP-23	^{233}U metal	U metal	0.9766	-	0.918

typically, ±2% to ±3%. There is more consistency in the results if the VERA values are excluded. For F8/F5 the mean value of C/E for ^{235}U fuelled systems is 1.060 and the range is ±3% and for the Pu fuelled systems the mean C/E is 0.986 and the range is ±1.3%. For the Pu fuelled systems the C/E values are consistent with unity excepting for the VERA-11A value.

Calculations for infinite media with fast and intermediate neutron spectra:

Calculations have been made by M. Grimstone (Winfrith) using the 72 group set generated by the NEA Data Bank for some fast and intermediate neutron spectrum zero leakage test zones built in ZEBRA and PROTEUS. These calculations are approximate in the treatment of resonance shielding, particularly for sodium and the constituents of steel. The results are summarised in Table 8.5.

Table 8.5

Ratios of calculation to experiment for zero leakage systems
(calculations made at Winfrith)

Name	k_∞	F9/F5	F8/F5	C8/F5
Uranium fuelled				
SCHERZO-556 (U metal)	0.996	1.024	1.077	1.020
UO_2-740 (uranium oxide)	0.997	0.999	1.022	0.995
			F8/F9	C8/F9
Plutonium fuelled				
ZEBRA 8A/2 (Pu, U, C)	0.994	0.951	0.998	0.997
8B (Pu, U, C)	0.993	1.008	0.999	1.00
8C/2 (Pu, s.s., U, C)	0.985	0.989	1.004	0.992
PROTEUS 6 (PuO_2, UO_2)	0.997	-	0.983	0.988
7 (PuO_2, UO_2, s.s.)	0.992	-	0.937	0.974
8 (PuO_2, UO_2, Fe)	0.997	-	0.936	0.989
10 (PuO_2, UO_2, Fe)	0.987	-	0.948	0.974

The calculated k_∞ values are close to unity excepting for two test zones which contain a high proportion of steel for which k_∞ is calculated about 1.5% low. For the F9/F5 ratio the only significant discrepancy is for ZEBRA 8A. A similar discrepancy is obtained using the adjusted cross-section set FGL5 and this suggests that the measurement or calculation method is discrepant, rather than the nuclear data. For the F8/F5 and F8/F9 ratios there are some larger discrepancies, with the values being overestimated by up to 8% for the uranium fuelled systems and underestimated by up to 6.5% for the PuO_2 fuelled systems. The discrepancy is largest for the PROTEUS test zones; an average underestimation of 5%. The discrepancy for the uranium fuelled systems is consistent with that found for the hard spectrum systems. The C8/F5 and C8/F9 ratios are all well predicted.

Calculations for French and German critical experiments

A JEF-1 cross-section set has been generated at Cadarache for the cell code HETAIRE. This has been used to calculate some MASURCA and SNEAK critical assembly measurements and MASURCA test zones (R1, R3, Z1, Z2, Z3 and ZOCO1). The results are summarised in Table 8.6.

Table 8.6

Ratios of calculations to experiment for fast reactor assemblies
(calculations made at Cadarache)

Name	k_{eff}	F9/F5	F8/F5	C8/F5
^{235}U fuelled cores				
SNEAK 9A	-	1.023	1.028	0.970
R3	1.0239	1.000	0.999	0.981
MASURCA 1B	1.0075	0.992	1.088	0.922
SNEAK 9B	1.0052	1.022	0.970	0.950
R1	1.0132	1.000	1.036	1.008
SNEAK $9C_1$	-	-	1.011	0.953
Pu fuelled cores				
Z3	1.0212	0.967	0.941	0.992
MASURCA 2A	-	0.979	1.029	0.927
PHENIX	-	1.011	0.977	0.949
Z1	1.0045	1.008	0.993	0.972
SNEAK 7B	1.0011	1.022	1.028	0.970
SNEAK 7A	1.0053	0.980	1.021	0.952
Z2	1.0105	1.048	1.026	0.995
SNEAK $9C_2$	-	1.008	0.937	0.983
ZOC01	1.0075	1.002	1.024	1.013

For two of the cases, R3 and Z3, the difference between calculated and measured k_{eff} values exceeds 2% and, on average, the values are 1% high. The calculated F9/F5 fission ratio values are within 3% of the measured values excepting for the two cores Z3 and Z2.

For F8/F5 fission ratio calculations there are four cases in which the agreement is outside ±3% (MASURCA 1B, R1, Z3 and SNEAK $9C_2$). In one case the discrepancy is about 9% and in another two about 6%. However, there is no obvious systematic pattern in the differences, although, for the ^{235}U fuelled cores, the tendency is for the calculated values to be about 2% high (which compares with the 6% average overestimate found for the hard spectrum and zero leakage cores) and for the ^{239}Pu fuelled cores there is, on average, a small underestimation by 0.3% (which compares with the average underestimation by 1.4% found for the hard spectrum cores and an average underestimation by 2.8% for F8/F9 in the zero leakage cores).

Capture cross-section measurements for actinide isotopes in a fast reactor spectrum

Spectrum averaged values of the capture cross-sections and (n,2n) cross-sections of actinide isotopes have been derived from sample irradiations in the PHENIX fast reactor (d'Angelo et al.). The JEF-1 calculated values are compared with the measured values in Table 8.7. There are large discrepancies for the (n,2n) cross-section of ^{239}Pu, which is overestimated by 38%. The capture cross-sections of ^{241}Pu and ^{242}Pu are also significantly overestimated, by 11% and 16% respectively. For the other isotopes the agreement is within about ±5%.

Table 8.7

C/E values for the PROFIL experiments using CARNAVAL-IV data and JEF-1 data

Type of data	C/E values with CARNAVAL-IV			C/E values with JEF-1
	PROFIL-1	PROFIL-2	Average value	Average value (PROFIL-1 + PROFIL-2)
σ_c (^{235}U)	0.99±2%	1.00±2%	0.99±1.4%	0.97±1.4%
σ_c (^{238}U)	0.97±2.3%	0.98±2.3%	0.98±1.6%	0.96±1.6%
σ_c (^{238}Pu)	-	-	-	0.95±3%
σ_c (^{239}Pu)	1.00±2.6%	1.01±2.6%	1.01±1.8%	0.97±1.8%
$\sigma_{n,2n}$ (^{239}Pu)	2.12±15%	1.89±15%	2.00±11%	1.38±11%
σ_c (^{240}Pu)	1.00±2.2%	1.02±2.2%	1.01±1.6%	1.06±1.6%
$\sigma_{n,2n}$ (^{240}Pu)	0.83±20%	0.85±20%	0.84±14%	0.83±14%
σ_c (^{241}Pu)	1.03±4%	1.01±5.3%	1.02±3.7%	1.11±3.7%
σ_c (^{242}Pu)	-	-	-	1.16±3.5%
σ_c (^{241}Am)	-	-	-	1.03±1.4%
σ_c (^{243}Am)	-	-	-	0.94±5%

Point-energy Monte Carlo calculations

Cross-sections are represented in a different way in the point-energy Monte Carlo code MONK6 from the way used in group cross-section sets. For example, secondary energy and angular distributions are represented by equiprobable bin data. The resonance structure of cross-sections is represented in great detail below 73 eV, by about 7000 energy points. The treatment of resonance structure at higher energies is more approximate but is considered to be sufficiently accurate for calculations on thermal spectrum systems. Because the cross-section processing route is different from that used to generate the group cross-section libraries (although some stages of the processing of primary cross-sections are the same) MONK6 calculations provide a check on processing methods as well as calculational methods.

Detailed intercomparisons are continuing.

The reasons for some differences have been identified but there are a number of differences between the MONK6 and deterministic calculations which have still to be resolved.

Conclusions

The performance of the JEF-1 neutron interaction cross-section library is broadly satisfactory, performing as well as could be expected for an unadjusted cross-section library. The JEF-1 benchmark calculations have not yet indicated specific items of nuclear data which require improvement although the parallel ENDF/B-V calculations have shown the need to improve ^{235}U fission and capture cross-sections in the resonance region.

Benchmark calculations made using different processing methods and different approximations have shown deficiencies in some of the methods.

9. CINDA-type Listing

Element Name	A	Quantity	Energy Min	Energy Max	Lab	Type	Reference NEANDC(E) 282 Vol 8 Jun.87, Page:	Author	Comments
Na	24	Half-life	-		NPL	Expt.prog.	55	Christmas +	Investigation of chemical effects on half life
Mn	55	N,Reaction	0	2+07	OXF	Theo.prog.	60	Ait-Tahar +	Calculation of reaction cross-section using Weisskopf-Ewing theory
Fe	nat	N,Gamma	2.4+00	1.1+05	HAR	Expt.prog.	4	Gayther +	
Fe	nat	Total SIG	3.0+02	3.0+05	HAR	Expt.prog.	7	James +	
Ni	58	N,Reaction	0	2+07	OXF	Theo.prog.	60	Ait-Tahar +	Calculation of reaction cross-section using Weisskopf-Ewing theory
Ni	60	N,Reaction	0	2+07	OXF	Theo.prog.	60	Ait-Tahar +	Calculation of reaction cross-section using Weisskopf-Ewing theory
Ni	65	Half-life	-		NPL	Expt.prog.	55	Christmas +	Measurement to be completed
Co	59	N,Reaction	0	2.0+07	OXF	Theo.prog.	63	Field +	Calculation of important neutron reaction cross-section
Cu	63	N,Reaction	0	2+07	OXF	Theo.prog.	60	Ait-Tahar +	Calculation of reaction cross-section using Weisskopf-Ewing theory
Cu	65	N,Reaction	0	2+07	OXF	Theo.prog.	60	Ait-Tahar +	Calculation of reaction cross-section using Weisskopf-Ewing theory
Sr	82	Half-life	-		NPL	Expt.prog.	55	Christmas +	25.342±0.053 days
Sr	90	Half-life	-		NPL	Expt.prog.	55	Christmas +	Measurement to be completed
Nb	93	N,Inelastic	2.7+06		HAR	Expt.prog.	7	Gayther +	Reported in AERE-R 12612
Nb	93	N,Scat	1+06	2+07	OXF	Theo.prog.	60	Wilmore +	Analysis of data in progress
Nb	93	N,Reaction	1+06	2+07	OXF	Theo.prog.	60	Wilmore +	Analysis of data in progress
Nb	93	N,Reaction	0	2.0+07	OXF	Theo.prog.	63	Field +	Calculation of important neutron reaction cross-section
Cd	nat	Diff.Elastic	3.0+06		EDB	Expt.prog.	59	Galloway	Differential elastic from 20 to 107 degrees
In	nat	Diff.Elastic	3.0+06		EDB	Expt.prog.	59	Galloway	Differential elastic from 20 to 107 degrees
Sn	nat	Diff.Elastic	3.0+06		EDB	Expt.prog.	59	Galloway	Differential elastic from 20 to 107 degrees
Sb	nat	Diff.Elastic	3.0+06		EDB	Expt.prog.	59	Galloway	Differential elastic from 20 to 107 degrees
Te	nat	Diff.Elastic	3.0+06		EDB	Expt.prog.	59	Galloway	Differential elastic from 20 to 107 degrees
I	nat	Diff.Elastic	3.0+06		EDB	Expt.prog.	59	Galloway	Differential elastic from 20 to 107 degrees
Eu	152	Half-life	-		NPL	Expt.prog.	55	Christmas +	4943±4 days
Eu	154	Half-life	-		NPL	Expt.prog.	55	Christmas +	3138±2 days
Pa	233	Half-life	-		HAR	Expt.prog.	44	Banham +	
U	235	Eta	1.0-02	1.0+00	HAR	Expt.prog.	8	Moxon	
U	235	^{3}H yield	PFR		HAR	Expt.prog.	42	McMillan	Proposed measurement of tritium yield in fast fission
U	237	Half-life	-		HAR	Expt.prog.	44	Banham +	
U	238	^{3}H yield	Fast		HAR	Expt.prog.	42	McMillan	Proposed measurement of tritium yield in fast fission
Pu	239	^{3}H yield	Fast		HAR	Expt.prog.	42	McMillan	Proposed measurement of tritium yield in fast fission
Pu	239	Alpha	PFR		ALD	Expt.prog.	44	Horscroft +	
Pu	240	Alpha	PFR		ALD	Expt.prog.	44	Horscroft +	
Pu	240	^{3}H yield	Fast		HAR	Expt.prog.	42	McMillan	Proposed measurement of tritium yield in fast fission
Pu	241	^{3}H yield	Fast		HAR	Expt.prog.	42	McMillan	Proposed measurement of tritium yield in fast fission
Pu	242	^{3}H yield	Fast		HAR	Expt.prog.	42	McMillan	Proposed measurement of tritium yield in fast fission