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भारत सरकार

GOVERNMENT OF INDIA

परमाणु ऊर्जा आयोग

ATOMIC ENERGY COMMISSION

भाभा परमाणु अनुसंधान केन्द्र

PROGRESS REPORT ON NUCLEAR DATA ACTIVITIES IN INDIA

for the period from July 1981 to December 1982

Compiled by

R. P. Anand

Nuclear Physics Division

BHABHA ATOMIC RESEARCH CENTRE

बंबई, भारत

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PREFACE

The present progress report on Indian Nuclear Data Activities is the second of the new series of progress reports, the first of which was brought out in 1981. It covers the period from July, 81 to December, 82.

As the proceedings of the annual nuclear physics and solid state physics symposia give full coverage of the nuclear physics activities in the country this report is limited to the compilation of nuclear data related activities only.

Bulk of the nuclear data related work in the country is being carried out at B.A.R.C., Bombay and R.R.C., Kalpakkam. The concerned divisions and sections of these research centres were requested to submit contributions for this report of their nuclear data related work, both published and unpublished, carried out during the above period. The nuclear data related work carried out in the other institutions and universities, which were reported at the nuclear physics and solid state physics symposium held at Varanasi in Dec.82, have also been included here.

(S.S. Kapoor)
Convener
Indian Nuclear Data Group.

FISSION FRAGMENT ANGULAR DISTRIBUTION DATA FOR NEUTRON INDUCED FISSION OF ^{235}U+

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Introduction

The fission fragment angular distributions provide important information on the quantum states available at the saddle point to the fissioning nucleus which in turn provides a basis for a theoretical understanding of the fission process. A knowledge of the fragment angular distributions is also important in evaluating those experimental fission cross section measurements in which fission fragments were detected at specified angles and not in a 2π geometry. In the case of neutron induced fission of an even-odd target nucleus like ^{235}U, at neutron energies exceeding a few MeV the number of K-states at the saddle point become sufficiently large for a statistical description[1] to be valid. In this region the fragment anisotropies depend on the value of a parameter K_0^2 (the variance of the assumed Gaussian distribution in K), which is related to the effective moment of inertia (and hence the nuclear shape) at the fission barrier. It has been pointed out earlier[2] that for nuclei exhibiting a double-humped barrier shape, the transition state is expected to gradually shift from the position of the outer barrier (barrier II) to the liquid drop model shape, with an increase in the excitation energy, as a result of the washing out of the nuclear shell effects with excitation energy. Some evidence of this feature in the experimental fragment anisotropy data for ^{242}Pu fissioning nucleus was also pointed out earlier[2]. In this work, we have looked into this feature of the expected excitation energy dependence of the fission transition state in the fragment anisotropy data of the fast neutron induced fission of ^{235}U. The present analysis provides evidence for the presence of the above feature expected in the case of all nuclei with double-humped barriers. This suggests that evaluation of the fission data for nuclei with double-humped fission barrier shape should incorporate the above important feature resulting from the gradual washing out of shell effects.

Description of the Data

The measured fragment anisotropies $W(0°)/W(90°)$ in the neutron induced fission of ^{235}U in the neutron energy range of 0 to 23 MeV taken from Refs.3-17 are given in Figs.1 and 2. Results of recent measurements of J.W. Meadows and C. Budtz-Jorgensen obtained through private communication[17] have also been included in the figures. Figs.1 and 2 provide a summary of the present status of the measured fragment anisotropies for different neutron energies for the case of the target nucleus ^{235}U, resulting from thorough literature survey. Although the average trend

+Contribution to International Conference on Nuclear Data for Science and Technology, Antwerp (6-10 Sept.1982).

of the variation of the anisotropy with the neutron energy is clearly brought out in the figures the different measurements of the anisotropy are generally in agreement with each other to only within a value of 0.05. Comparing the two most recent measurements[16,17] the anisotropies of Ref.17 appear to be systematically lower by a few per cent as compared to those of Ref.16. For example, at 300 keV and 1040 keV where both have reported measurements, the values reported in Ref.17 are 1.098 and 1.017 as compared to 1.170 and 1.040 of Ref.16. Further careful measurements at these two energies will be useful in resolving the above small descrepancy between the above two recent measurements.

Analysis of the Data and Discussion

We have carried out statistical model analysis of the compiled fragment anisotropy data (Figs.1 and 2) on the basis of the statistical theory[4]. Below the second chance fission threshold, the theory predicts an angular distribution of the form

$$W(\theta) = (1+\alpha \cos^2\theta)/(1+\alpha/3)$$

where α is related to the quantity $p = l^2_{max}/4K_0^2$, l_{max} being the maximum angular momentum brought in by the neutron in a sharp cut-off model and K_0^2 is a parameter which is the product of the effective moment of inertia J_{eff} and temperature T of the saddle point nucleus. With the opening up of the multichance fission, one has to also consider the angular distributions of multiple chance fission fragments. In addition, it is known that for incident neutron energies of about 14 MeV, significant pre-equilibrium effects set in leading to uncertainties in the angular momentum distribution and excitation energy of the fissioning nucleus. We have therefore restricted the present analysis to the range $0 < E_n < 11$ MeV where third chance fission is also not open.

In the present analysis, the first chance fission anisotropy of the ^{236}U fission fragments for neutron energies upto 11 MeV have been deduced, from the experimental data on Γ_f/Γ_n and anisotropies for neutron induced fission of ^{235}U and ^{234}U. In this way, the observed anisotropies of ^{236}U have been corrected for the second chance fission in a nearly model independent way. The maximum angular momentum l_{max} is obtained from the relationship given by Blann[18].

The continuous line in Figs.1 and 2 represents the values of the anisotropy resulting from the variation of K_0^2 versus E_x shown in the inset in Fig.1 as solid line. Also shown in the inset, are the expected values of K_0^2 versus E_x for the second barrier shape (dashed line) and the shape predicted by the liquid drop model (LDM) (dash-dot line). For the LDM shape the shell and pairing corrections have been included as in Ref.2. Thus, the above analysis demonstrates that the fission of ^{236}U exhibits the same features as was earlier pointed out for the ^{242}Pu fission, namely, that due to washing out of shell effects the shape of the transition state shifts from the second barrier shape to the LDM shape. This feature is expected to be common to all actinide nuclei with double-humped fission barriers and should be consider-

ed in any fission data evaluation.

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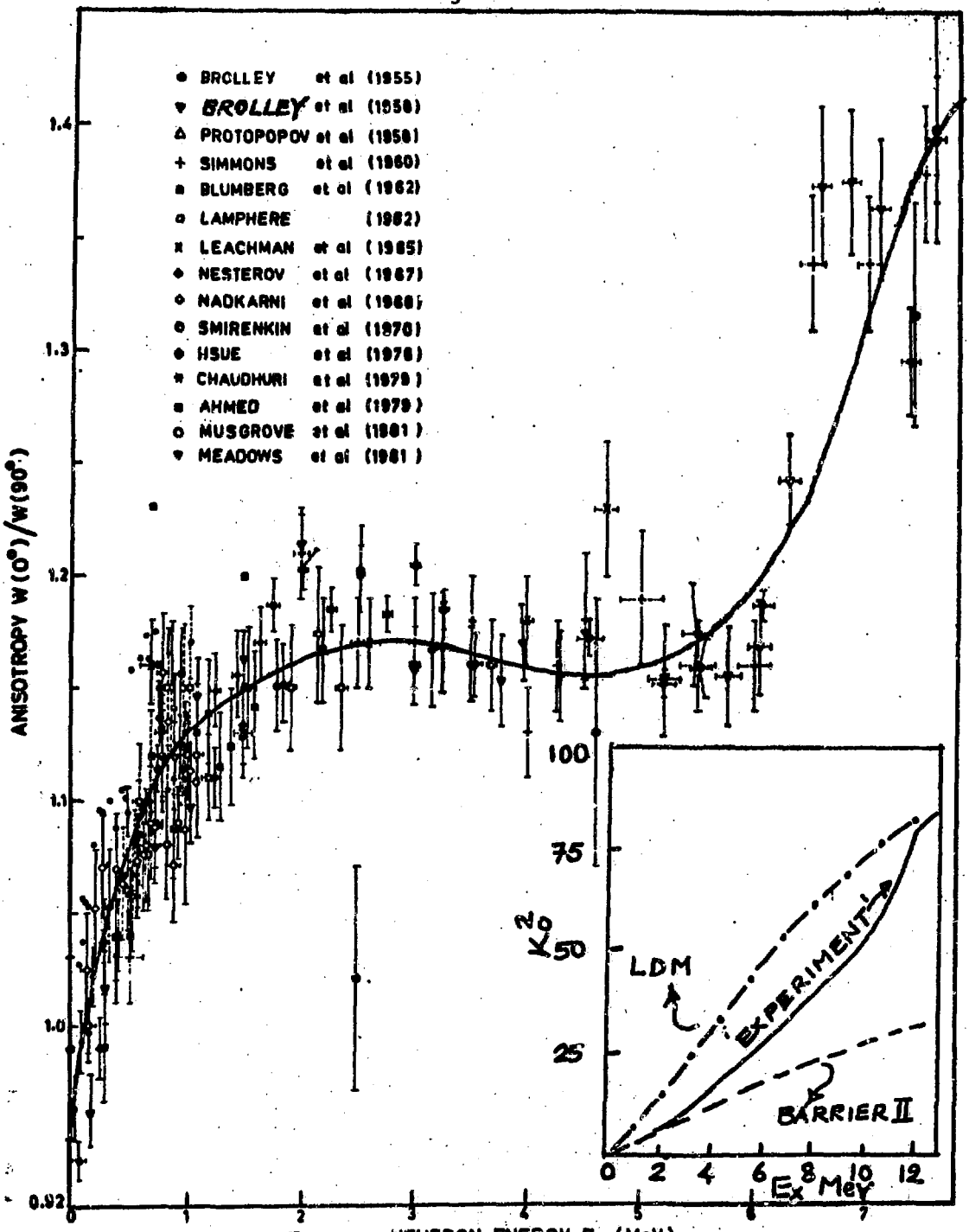
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LEACHMAN et al (1965)
NESTEROV et al (1967)
NADKARNI et al (1968)
SMIRENKIN et al (1970)
HSUE et al (1976)
CHAUDHURI et al (1979)
AHMED et al (1979)
MUSGROVE et al (1981)
MEADOWS et al (1981)
ANISOTROPY W(0°)/W(90°)
NEUTRON ENERGY En (MeV)
LDM
EXPERIMENT
BARRIER II
Ex MeV

FIG. 1

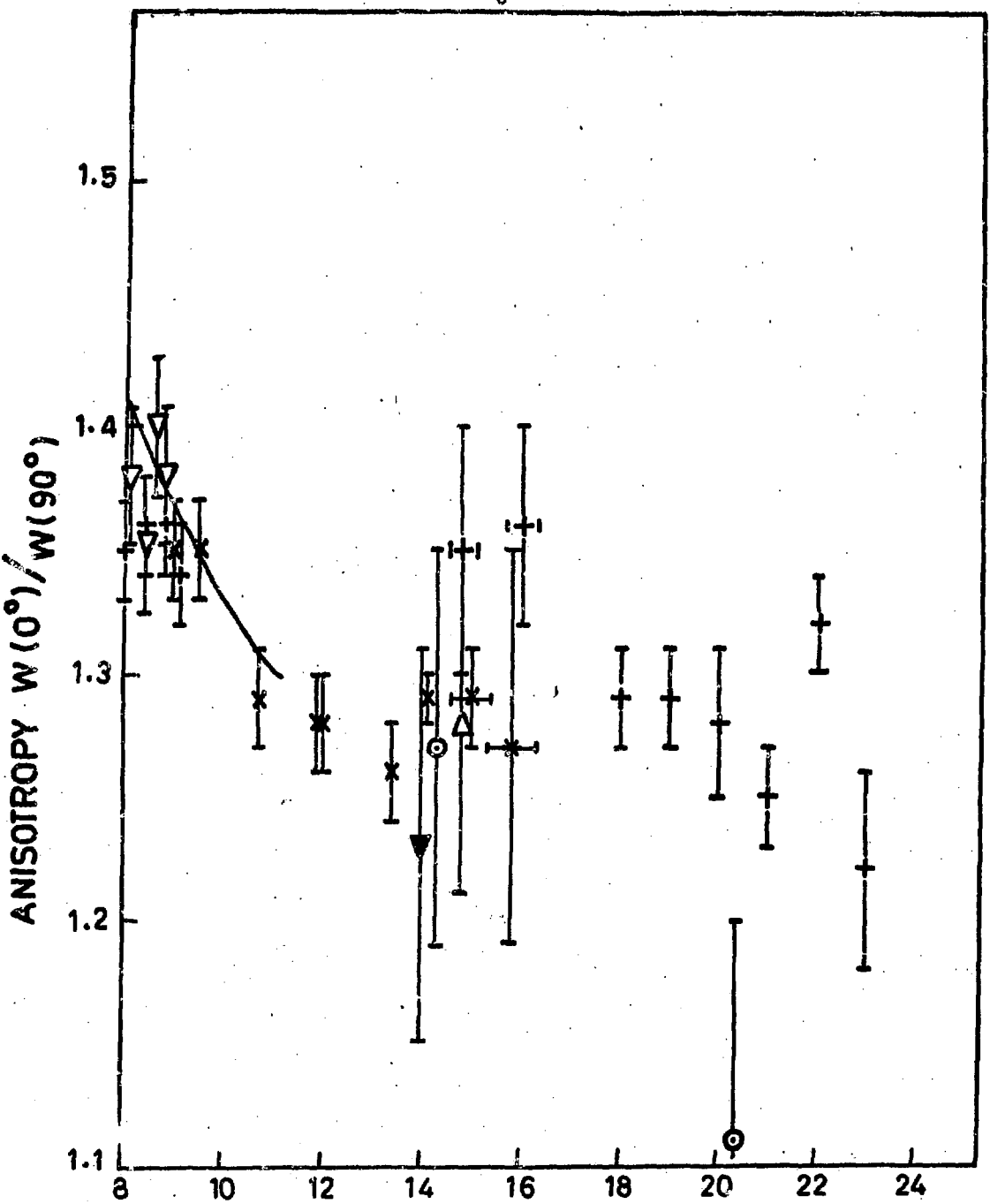

FIG.2 - NEUTRON ENERGY (MeV)

PROMPT NEUTRON KINETIC ENERGY DISTRIBUTION IN FISSION

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The average kinetic energy distribution $\bar{\eta}_F$ for prompt neutrons in fission is found by modifying Le Couteur's formalism /1/, including the (i) nuclear structure effects explicitly and (ii) excitation dependence of the level density parameter a (U). Contrary to earlier attempts /2,3/ we have consistently used all input parameters from our Structure-sensitive Statistical Saddle-scission Point model /4,5/ without introducing any new parameters.

Assuming that only neutrons are emitted and that the fragment excitation $U_F > B_F$, where B_F is the average neutron binding energy, the number of average prompt neutrons $\bar{\nu}_F(A_F)$ emitted from the fragment A_F is given by

$$\bar{\nu}_F(A_F) = \int_0^{U_F - E_{\gamma F}} dU/(B_F + 2T) \qquad (1)$$

where the denominator is the average energy carried away by a neutron at any stage U of compound nuclear excitation. The maximum energy available for neutron emission is $U_F - E_{\gamma F}$, where $E_{\gamma F}$ is the γ de-excitation energy, T is the nuclear temperature. A functional form of U is necessary to evaluate eq.(1). The level density $\rho(U)$ is assumed to have the form

$$\rho(U) = K \exp\left[2(a(U).U)^{1/2}\right]/U^{5/4}, \qquad (2)$$

where K is a constant and a(U) is our excitation dependent level density parameter of the form /6/

$$a(U) = \bar{a}(1 + x_1 U + x_2 U^2) \qquad (3)$$

Here $\bar{a}$ is the ground state a-parameter and x_1, x_2 are the structure-sensitive coefficients. Using the definition $1/T = (d/dU)\ln \rho(U)$, U is deduced as a function of T; this helps solving eq.(1). The actual integration gives

$$\nu_F(A_F) = (\bar{a}B_F/2)\left[(1+c)\gamma_m - \tfrac{1}{2}c\gamma_m^2 + \tfrac{1}{3}c\gamma^3 - \{1 + c + 5/(4\bar{a}B_F)\}\ln(1+\gamma_m)\right] \qquad (4)$$

where $c = (3/2)x_1\bar{a}B_F$ and $\gamma_m = 2T/B_F$ and T_m is the temperature at the highest limit $U_F - E_{\gamma F}$. The average energy $\bar{\epsilon}_{nF}$ and average kinetic energy $\bar{\eta}_{LF}(A_F)$ of each neutron emitted are

$$\bar{\epsilon}_{nF} = (U_F - E_{\gamma F})/\bar{\nu}_F(A_F)\ ;\ \bar{\eta}_{LF}(A_F) = \bar{\epsilon}_{nF} - B_F \qquad (5)$$

A Maxwellian shape for the neutron kinetic energy spectrum is used to find the average fission neutron energy $\bar{\eta}$ in the c.m. frame, the corresponding value $\bar{E}$ in the laboratory frame and the average number of neutrons ($\bar{\nu}$) per fission event.

Input parameters for the microscopic calculations are estimated from SSSP model /4,5/. Results for the $\bar{\nu}_F$ distribution, the $\bar{E}$ and $\bar{\eta}_L$ values are better than earlier simple studies /2,4/. Our predictions for $\bar{\eta}_F$ for ^{235}U (n_{th},f), ^{239}Pu (n_{th},f) and ^{252}Cf (s.f) are compared in fig.1 with the available standard results /2,3,7,8/. A fair agreement is achieved; for ^{252}Cf the discrepancy with expt. /8/ seems to be real in the range $A_F \simeq 125-145$. We predict a minimum in $\bar{\eta}_F$ in the neighbourhood of $A_F = 132$ (corresponding to Z = 50 and N = 82) in all these cases.

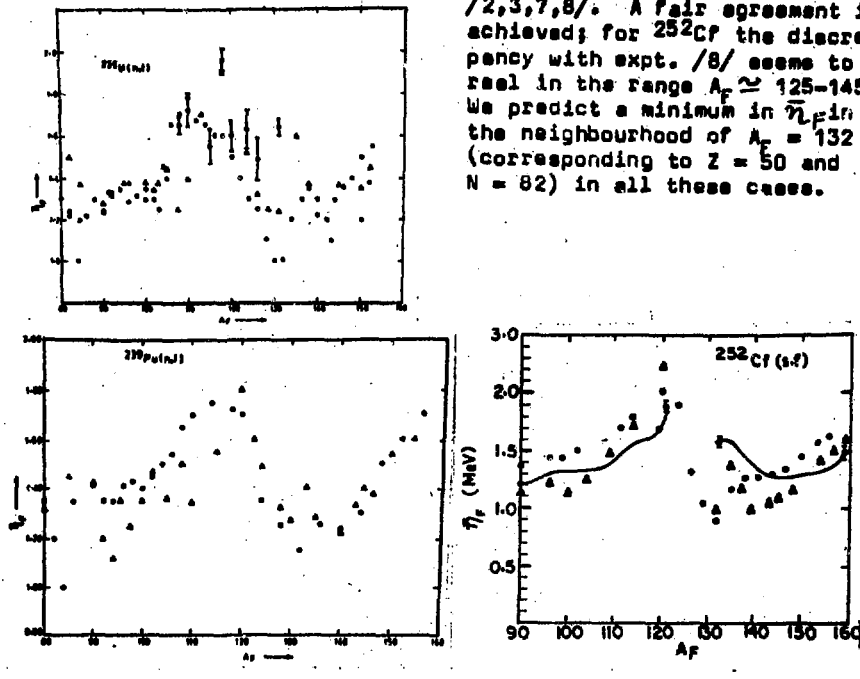

Fig.1. Distribution of $\bar{\eta}_F$ as a function of A_F
^{235}U : • = presen △ =ref /2/, o=ref./7/;
^{240}Pu: • = present, ▲ =ref/2/; ^{252}Cf:
o = present, ▲ =ref /3/,— = ref./8/

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STUDY OF FISSION FRAGMENT MASS DISTRIBUTIONS FOR SPECIFIED TOTAL KINETIC ENERGIES

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The correlation between the fission fragment mass (M_f) and total kinetic energy (E_k) has been earlier /1,2/ depicted in terms of the dependence of $\bar{E}_k$ and σ_{Ek} on specified mass ratio or M_f. However, for a direct comparison of the mass distribution with model predictions it is desirable to have experimental information on the fragment mass distribution for specified total kinetic energy (E_k). In the present work we report results of a double kinetic energy measurements which are different from earlier published work in the following two respects. Firstly, the fission fragment detectors used in the back-to-back geometry were high resolution heavy ion detectors with central collimator holes (2mm) to limit the detection area to the central region of the detectors. Secondly, the analysis of the data was carried out to obtain prensutron emission mass distribution for narrow 4 MeV intervals of E_k.

The experimental assembly consisted of an evacuated chamber containing two semiconductor detectors placed on either side of a 30 $\mu g/cm^2$ ^{235}U source on VYNS backing and this chamber was irradiated in the neutron beam of CIRUS reactor. About 2×10^5 two parameter events were recorded on mag tape and analysed off-line on a computer. The analysis involved mass dependent energy calibration of the detectors, the correction for the effects of neutron emission using $\bar{\nu}(M,E_k)$ data and the usual momentum and mass conservation relations event by event. Fig.1 shows the observed mass distribution of fission fragments for a few representative E_k windows. Fig.2 shows the variation of the most probable values of the light fragments ($\bar{M}_L$) and the heavy fragments ($\bar{M}_H$) with E_k. It is seen that as E_k increases $\bar{M}_L$ and $\bar{M}_H$ approach each other and, in particular for very large E_k values $\bar{M}_H$ approaches the value of 132 which corresponds to the doubly magic configuration (Z=50, N=82). It may be noted that the highest kinetic energy window ($E_k \sim 205$ MeV) shown in the figure nearly corresponds to the Q-value of the reaction for the fragment masses observed, based on the calculations carried out with the existing mass formulas /3/. This implies that for these cases the final excitation energies are very small and at scission they may be virtually cold as some energy may be locked up in the light fragment deformation energy. It is an interesting feature that even for these cold fission cases the character of the mass distribution with its finite mass width is not much altered. This appears to be contrary to what would be expected on the basis of the statistical models where the excitation energy at scission crucially determines the mass yield distribution. The present data

therefore point to the importance of the dynamical effects in determining the mass distributions.

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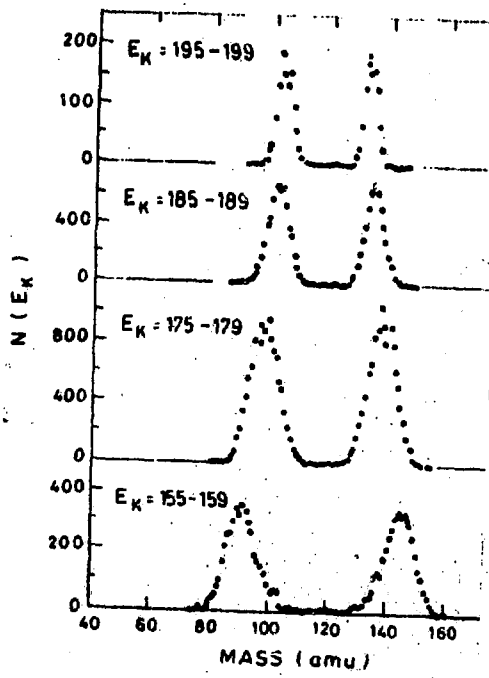

Fig.1. $N(M_F)$ distribution for some E_k window.

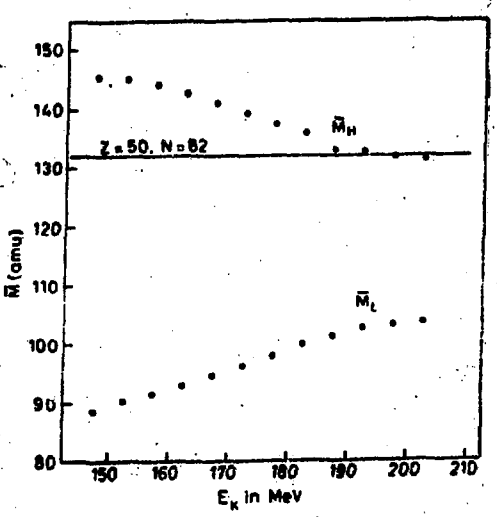

Fig.2. $\bar{M}_L$ and $\bar{M}_H$ variation with E_k

CHARGE DIVISION MECHANISM IN BINARY FISSION IN TERMS OF DCM

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The charge and mass values obtained from our DCM /1/ agree fairly well with the experimental findings of Unik et al /2/. This has led us to investigate, how far the DCM agrees with the existing charge division mechanisms of binary fission. There are at present two well known charge division hypothesis, namely, UCD and ECD /3/.

The experiments of Unik et al /2/ (from x-ray measurements) suggests a modified form, which gives much better agreement with the expts.

$$A_{HF} = \frac{A_F}{Z_F}(Z_H + .5) \; ; \; A_{LF} = \frac{A_F}{Z_F}(Z_L - .5)$$

where the symbols have their usual meanings.

In order to arrive at the modified UCD formula, from the DCM, we start with

$$A_{LF} = A_{LC} + Ne/2 \text{ and } A_{HF} = A_{HC} + Ne/2$$

where Ne is the excess neutrons, which is equi-partitioned between the heavy (A_{HC}) and light (A_{LC}) core.

After some trivial algebra, we finally arrive at

$$A_{HF} = \frac{A_F}{Z_F}(Z_H + C) \; ; \; A_{LF} = \frac{A_F}{Z_F}(Z_L - C)$$

where $C = \frac{F(Z_F, Z_L)}{2} Z_L - \frac{F(Z_F, Z_H)}{2} Z_H$, is positive.

We worked up this relation for fissioning nuclei ^{230}Th, ^{240}Pu and ^{252}Cf, and found that for all possible complementary fragment pairs, C ranges between .2 and 1. Particularly in the high yield regions ($Z_H = 52 - 56$) C has a value between .4 and .8.

The ECD hypothesis attempts to connect the prompt fragments in fission with the radiochemical products obtained after neutron and beta emissions.

From DCM, the increment of mass of each fragment w.r.t. the isotopically abundant core, on the line of stability, and before neutron emission, is given by

$$\Delta A_H = A_{HF} - A_{HC} \; ; \; \Delta A_L = A_{LF} - A_{LC}$$

Radiochemical masses are obtained after neutron emission, viz.

$$A_{HR} = A_{HF} - \nu_H = A_{HC} + Ne/2 - \nu_H$$

$$A_{LR} = A_{LF} - \nu_L = A_{LC} + Ne/2 - \nu_L$$

where ν_H and ν_L, are the no. of neutrons emitted from the heavy and light core respectively.

Finally to obtain the charge displacements as

$$\Delta Z_H = \frac{Ne/2 - \nu_H}{(\Delta A/\Delta Z)_{A_{HC}}} \quad ; \quad \Delta Z_L = \frac{Ne/2 - \nu_L}{(\Delta A/\Delta Z)_{A_{LC}}}$$

The above result was tried for ^{252}Cf and ^{254}Fm where ν_H and ν_L are experimentally known /4/ for different complementary fragment pairs. The agreement between ΔZ_H and ΔZ_L is satisfactory as evidenced from the following table:

Table - I

Charge Displacements from DCM for ^{252}Cf

H.F.	L.F.	Ne	ΔZ_H	ΔZ_L
^{129}Sn	^{123}Cd	20	2.59	2.07
^{137}Te	^{115}Pd	18	2.45	2.15
^{141}Xe	^{111}Ru	20	3.36	2.89
^{146}Ba	^{106}Mo	18	3.23	3.16
^{150}Ce	^{102}Zr	20	4.62	4.31

Thus DCM can also fairly reproduce ECD.

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/2/ J.P. Unik et al, Proc. 3rd IAEA Symp. on the Physics and Chemistry of Fission, Rochester, New York, August 1973. Paper IAEA - SM - 174/209.
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J.E. Gindler et al, Phys. Rev. C16 (1977) 1483.

Cumulative yields of short-lived Ru isotopes in the spontaneous fission of ^{252}Cf

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Cumulative yields of the short-lived ^{105}Ru, ^{107}Ru, ^{108}Ru and ^{109}Ru in the spontaneous fission of ^{252}Cf have been determined by a fast radiochemical separation followed by gamma spectrometry. The cumulative yields for ^{105}Ru, ^{107}Ru, ^{108}Ru and ^{109}Ru are 7.20 ± 0.30, 6.35 ± 0.33, 5.89 ± 0.29 and 4.08 ± 0.30 respectively. The ^{105}Ru yield values reported in the literature have poor agreement. This has been explained in terms of the chemical behaviour of radio ruthenium. The fast radiochemical separation of ruthenium from other fission products was achieved within 40 sec. after the end of irradiation. The overall uncertainty in the measured yield values is around 5 to 7%.

Charged particle induced fission: Dependence of fragment angular momentum on entrance channel in ^{232}U fission

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In this work fission fragment angular momenta for fragments corresponding to fission product isomers. $^{131m,g}Te$ and $^{133m,g}Te$ have been deduced from the radiochemically determined independent isomeric yield ratios in 30 MeV alpha-particle induced fission of ^{232}Th fission. The objective of this work was to investigate the influence of initial excitation energy and angular momentum of the compound nucleus on the fission fragment angular momentum, an aspect, not so well understood so far although scanty data that are available indicate somewhat uncertain increase of fragment angular momentum with initial excitation energy.

Experimental:

Table : Comparison of Angular Momenta in ^{232}Th ($\alpha_{30\ MeV}$,f) and ^{235}U (n_{th},f) system

Fissioning system	Excitation Energy (MeV)	$\langle I^2 \rangle_{av}^{\frac{1}{2}}$ ($\hbar$)	RMS Angular Momenta ($\hbar$)	
			^{131}Te	^{133}Te
$^{235}U(n_{th},f)$	6.55	3 to 4	6.0 ± 1.5 7.5 ± 0.7	5.9 ± 1.5 6.5 ± 0.4
$^{232}Th(\alpha,f)$	23.38	10.2	5.7 ± 0.6	6.9 ± 0.9

Charge distribution in the spontaneous fission of ^{252}Cf, Determination of fractional cumulative yields of ^{140}Ba and ^{146}Ce

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The fractional cumulative yields (FCY) of ^{140}Ba and ^{146}Ce in the spontaneous fission of ^{252}Cf have been determined using radiochemical separation and direct gamma-ray counting of the irradiated catcher foil on a 60 c.c. Ge(Li) coupled to a 4 K analyser. The fractional cumulative yields are calculated from the growth and decay of the daughter products ^{140}La and ^{146}Pr respectively. The values of fractional cumulative yields are 0.9977 ± 0.0018 and 0.989 ± 0.002 for ^{140}Ba and ^{146}Ce respectively. These values correspond to a width (σ) of 0.59 ± 0.07.

Isotopic yield distribution of Iodine in the spontaneous fission of ^{252}Cf: Influence of proton pairing effect

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Independent yields of $^{131,132,133}I$ in the spontaneous fission of ^{252}Cf were determined gamma spectrometrically following separation of iodine from other fission products. The experimental data were used to parametrise the isotopic yield distribution (Ap model) for iodine (Z=53) in ^{252}Cf (SF). The values for most probable mass (Ap) and the width of the distribution (σA) as obtained in this work are 135.75 and 1.66 $\pm$ 0.14 mass units respectively. Total elemental yield (Y_z) was estimated as 8.2 $\pm$ 0.7%. Using this value of Y_z in this work and the literature data for other elements, the odd-even effect due to proton pairing was estimated as 13.9 $\pm$ 4.6% for ^{252}Cf (SF).

Experimental Results:

Nuclide	I.Y.
^{131}I	$(2.74 \pm 0.31) \times 10^{-2}$
^{132}I	$(4.65 \pm 2.32) \times 10^{-3}$
^{133}I	$(5.74 \pm 0.30) \times 10^{-1}$
^{134}I *	(1.097 ± 0.137)

* From ref: S.B. Manohar et al. Phys. Rev. C 17 (1) (1978) 188.

Isotopic yield distribution of Technetium isotopes in the spontaneous fission of ^{252}Cf

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The fractional independent yields of ^{101}Tc, ^{103}Tc, ^{104}Tc and ^{105}Tc in the spontaneous fission of Cf 252 have been determined by following thedecay of technetium after rapid chemical separation from molybdenum. From the FIYs the most probable mass Ap and the isotopic dispersion parameter (σ_A) were determined and are 107.30 ± 0.60 and 1.80 ± 0.30 respectively. The yields of ^{103}Tc, ^{104}Tc and ^{105}Tc are consistent with the yields of cesium isotopes which are complementary to Tc.

Table 1. Fractional Independent Yields of Tc

^{101}Tc	0.007 ± 0.002
^{103}Tc	0.045 ± 0.010
^{104}Tc	0.078 ± 0.022
^{105}Tc	0.339 ± 0.064

Fission fragment angular momentum: Independent isomeric yield ratios of ^{131}Te, ^{133}Te and ^{111}Pd in ^{252}Cf (SF) system

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In low energy fissioning system with low initial angular momentum, generation of fragments with high angular momenta is ascribed to strong coulombic interaction between the fragments for non-linear scission configuration. Therefore, one expects a strong correlation between fragment angular momentum, mode of charge split and fragment deformation. Investigation on angular momenta of charge complementary fragments is therefore useful to study the effect of coulombic torque which in turn depends on the charge split and deformation of individual fragments, on the fragment angular momentum.

In this work angular momenta of fragments corresponding to $^{131,133}Te$ and ^{111}Pd, which are charge-complementary have been deduced from the radiochemically determined independent isomeric yield ratios of these fission product isomers in the spontaneous fission of ^{252}Cf.

Table : Isomeric yield ratio and angular momenta

Fission product	Yield Ratio	RMS Angular Momentum ($\hbar$)
^{111}Pd	0.601 ± 0.018	8.5 ± 0.15
^{131}Te	0.503 ± 0.026	5.8 ± 1
^{133}Te	0.544 ± 0.062	6.1 ± 1

A SIMPLE MODEL TO CALCULATE FAST FISSION CROSS SECTIONS INCLUDING THE EFFECT OF PRE-EQUILIBRIUM

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Fission cross sections for fast neutrons play an important role in the design of fast reactors. Although the measurements of fast fission cross sections have been carried out extensively for the important fissile nuclei but these cross sections for the less stable nuclei, which are produced in the core of the reactors, can not be measured very easily. Here a semi-empirical approach, based on actual physical processes involved at higher neutron e energies, has been adopted to calculate fission cross-sections from 1 to 18 MeV (upto third chance fission) using the first fission plateau (2-4 MeV) cross sections as imputs. The basic assumption employed here is that, the ratio of neutron width to the fission width is considered to be independent of excitation energy (at least upto 25 MeV) for the decay of the nucleus through the statistical process. The effect of pre-equilibrium emission of first neutron is considered alongwith its equilibrium emission. The subsequent neutron emissions are assumed to be purely due to equilibrium process. When the first neutron is emitted due to pre-equilibrium, the first chance fission of the nucleus is not possible but the second- and third--chance fission probabilities are not effected. Near the fission-barrier. The barrier penetration factor as given by Hill and Wheeler was used. The gamma emission is neglected as it is very small in comparison to neutron emission. The reaction cross sections required are taken from the Wilmore and Hodgsonjs optical cross-sections.

The first-chance fission cross-sections were taken from the experimentally measured data wherever available, otherwise they were taken from the systematics of Behrens and Howerton (1978).

Based on this prescription a computer code was developed and fission cross sections were calculated for the nuclei ^{232}Th, ^{235}U, ^{238}U, ^{237}Np and ^{239}Pu in the energy range 1-18 MeV and these calculated cross sections were found to agree within 10% with the recently measured fission cross sections for these nuclei. Thus after establishing this prescription it was used to predict the fission cross sections for the less stable nuclei like ^{233}Th and ^{233}Pa where measurements are not available.

Ref.
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THE RADIATIVE NEUTRON CAPTURE CROSS-SECTIONS FOR ^{51}V IN MEV ENERGY REGION

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INTRODUCTION: Vanadium has been considered as one of the important elements for the future fast reactor designing, as its alloys are being used as one of the fuel-cladding material /1/. The energy range of 10 KeV to 3 MeV is of principal interest to such a reactors. Besides, measurements of fast neutron capture cross-sections furnish valuable informations about nuclear structure. Hence the precise knowledge of the reaction cross-section is essential in this energy region. In the present work, radiative neutron capture cross-section for the reaction $^{51}V(n,\gamma)^{52}V$ has been measured at three neutron energies, i.e. 1.07±0.20MeV, 1.89±0.17 MeV and 2.85±0.15 MeV, using activation technique. The experimental results have been compared with the statistical theory calculations by Holmes et al /2/, to see whether statistical theory is good enough to predict the radiative neutron capture cross-section up to 3 MeV energy.

Experimental Details: In the present measurements, activation technique has been applied. Monoenergetic neutrons of the desired energies were produced by means of the $T(p,n)^{3}He$ reaction using the Variable Energy Cyclotron at Chandigarh. Neutron energies were calculated from reaction kinematics. The spread in the neutron energy was estimated from the known target thickness and the geometry involved. The tritium target was cooled by lateral heat conduction process.

Spectrographically pure (99.99%) vanadium oxide was used as sample. The target sample was made by uniformally spreading the powder within the perspex ring of specified radius, which was sandwiched between two thin cellulose tapes. This sample was then placed between two standard samples of potassium iodide of similar size to get the average neutron flux passing through the sample under investigation. The target sample was irradiated in the zero degree forward direction with respect to the proton beam.

A high resolution 50 c.c. Ge(Li) detector (FWHM of 2KeV at 1.33 MeV) was used to detect the characteristic gamma lines. The photopeak detection efficiency of the Ge(Li) detector for gamma ray at various energies were determined experimentally by using standard gamma ray sources. The gamma rays spectra in the decay of samples were recorded with a pre-calibrated 4096 channel ND-100 analyser. The half life was also followed upto several halflives to be sure about the formation of ^{52}V with $T_{1/2}$ = 3.76 min.

The formula used for computing the cross-section in the present work is,

$$= \frac{A \cdot \lambda \exp(\lambda \cdot t2)}{N \phi \epsilon \theta K [1 - \exp(-\lambda t_1)] [1 - \exp(-\lambda t_3)]}$$

where A is the total counts under the photopeak of the characteristic gamma ray, λ is the decay constant of the product nucleus, N is the number of nuclei of the isotope under investigation, ϵ is the effective photopeak detection efficiency, ϕ is the average neutron flux, θ is the absolute intensity for characteristic gamma ray, K is the self absorption correction factor for the gamma ray in the sample, t_1 is the irradiation time, t_2 is the elapsed time between stopping of irradiation and starting of counting and t_3 is the counting time. For better accuracy of the results the measurements were done for two or more irradiations at each incident neutron energy. The errors and uncertainties associated with the reported results comprises of various factors /3/, whose corrections has been taken into account in the measured values.

Results and Discussion: The measured radiative neutron capture cross-section for ^{51}V at three neutron energies are given in table I. A comparison of our measured cross-sections with the statistical theory calculations of Holmes et al /2/ has also been given in the same table. Since the statistical theory calculations are parameter dependent, /2,4/ the exact agreement between experiment and theory is hardly possible. However, the agreement between our experimental result and theoretical predictions, is within a factor of two.

Table I

The (n, γ) cross-section for the reaction $^{51}V(n, \gamma)^{52}V$

S.No.	Neutron Energy En/MeV/	$\sigma_{exp}(n,\gamma)$ /mb/	$\sigma_{theo}(n,\gamma)$ /mb/	$\left\{\frac{\sigma_{exp}(n,\gamma)}{\sigma_{theo}(n,\gamma)}\right\}$
1.	1.07±0.20	2.84±0.35	1.9	1.495±0.185
2.	1.89±0.17	1.78±0.20	1.15	1.548±0.174
3.	2.85±0.15	0.92±0.13	1.06	0.868±0.126

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THERMAL CROSS-SECTIONS OF ^{232}Th and ^{233}U

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1. Nature of the Study

The recent finding of R.C.Little et.al[1] that the 0.0253 eV capture cross-section of ^{232}Th to be 6.45 barns as against the ENDF/B-V normalization of 7.40 barns arouses a lot of interest. These are the first ever direct measurements of ^{232}Th capture cross-section below 0.0346 eV. To see whether integral measurements warrant the use of reduced capture cross-section of ^{232}Th below 0.03 eV, we have analysed the Brook-haven National Laboratory (BNL) exponential lattices[2] using ENDF/B-V cross-sections, as well as the reduced thermal cross-sections normalised to 6.45 barns at 0.0253 eV.

The following microscopic point cross-sections were obtained on request from IAEA, Vienna[3].

(i) neutron capture cross-sections of ^{232}Th from ENDF/B-V in the range 0 to 4.0 eV.

(ii) total cross-sections of ^{232}Th from ENDF/B-IV in the energy range 0 to 4.0 eV.

(iii) total, capture and fission cross-sections and ν values of ^{233}U from ENDF/B-IV in the energy range 0 to 0.625 eV.

The above mentioned point cross-sections were converted to multigroup cross-sections with the energy limits matching with those of WIMS 69 group library using a hardened Maxwellian shape in neutron energy (E) in the range 0.0 to 0.14 eV and 1/E shape in the range 0.14 to 4.0 eV as the weighting spectra. For using Little's results, we prepared another set of cross-sections by reducing the capture cross-section of ^{232}Th below 0.03 eV from ENDF/B-V values by a factor 7.4/6.45.

The BNL lattices chosen for analysis were seven H_2O moderated single fuel rod lattices (moderator to fuel volume ratio - 0.997 to 9.2747) and four D_2O moderated single fuel rod lattices (moderator to fuel volume ratio - 11.7159 to 37.8276). The fuel consisted of 3% enriched $^{233}UO_2$ in ThO_2. Multigroup integral transport theory was used for spectrum calculations inside the unit cell of the lattice followed by multigroup diffusion theory calculations using the measured buckling[2] to get the multiplication factors

and also the leakage corrected spectra for finding the reaction rates.

2. Conclusions

With ENDF/B-V cross-sections for ^{232}Th, good predictions in the criticality of H_2O moderated lattices are obtained. However, there is an underprediction of multiplication factors by 1.9% in D_2O moderated lattices. Another parameter which shows deviation between calculated and measured values is the ratio of epithermal to thermal captures in ^{232}Th(ρ_c^{02}) which is underpredicted in both H_2O and D_2O moderated lattices by about 11%.

With the use of "Little et.al" cross-sections below 0.03 eV for ^{232}Th, the multiplication factor prediction of D_2O moderated lattices improve to 0.999 ± 0.009. At the same time, multiplication factors prediction of H_2O moderated lattices remain good at 1.006 ± 0.001. There is an improvement in ρ_c^{02} predictions in both H_2O and D_2O moderated lattices - deviations reducing to 8%.

It is concluded from our analysis that it is preferable to use a value of around 6.45 barns for ^{232}Th capture cross-section at 0.0253 eV (found in R.C.Little et.al's measurements) instead of ENDF/B-V normalisation of 7.40 barns.

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EVALUATION OF THE ^{232}Th NEUTRON CAPTURE AND FISSION CROSS-SECTION ABOVE 50 keV

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As a part of the IAEA-NDS sponsored Coordinated Research Programme on the Intercomparison of Evaluation of Actinide Neutron Nuclear, the evaluations of the measured differential data for ^{232}Th (n, γ) and (n, f) cross-section above 50 keV were completed. A paper (E-11) was presented at International Conference on Nuclear Data for Science and Technology, Antwerp, Belgium (Sept.1982).

Most of the existing differential measurements were reviewed and scrutinised for reliability. Selected data sets were renormalized to ENDF/B-V values of the standard cross-sections and fitted with computer program SPLINE. The end results of both these evaluations are sets of "recommended" values at energy intervals close enough to enable linear interpolation (Ref.NDE (BARC)-9 and 10). The present evaluations are compared with some recent ones. Regions of larger uncertainties are pointed out where new measurements would be of value.

* ERPS

** NPD

EVALUATION OF NEUTRON CROSS-SECTIONS FOR ^{232}Th*

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Thorium-232 is a fertile element which leads to the production of fissile Uranium-233 when irradiated with neutrons in nuclear reactors. In order to assess its physics potential neutron induced reaction cross-sections are needed in the energy range extending upto 20 MeV. To meet this end evaluations of total, elastic, inelastic, level excitation, capture, fission, (n, 2n) and (n, 3n) cross-sections have been carried out under a project of Intercomparison of Evaluation of Actinide Neutron Nuclear Data sponsored by Nuclear Data Section of International Atomic Energy Agency.

The measured total and elastic scattering cross-sections have been evaluated /1/, /2/. Deformed optical model with adiabatic approximation has been used with the parameters of Guenther et al /3/ to generate neutron total and elastic cross-sections upto 20 MeV. The spherical optical model parameters of Sinha and Garg /4/ have yielded the compound nuclear contributions and the level excitation cross-sections. (n, 2n) and (n, 3n) cross-sections have been derived using preequilibrium-statistical model /5/. Upto 18 MeV of neutron energy an evaluation of measured (n,2n) cross-sections carried out by Ananet al /6/ has been accounted.

For fission and capture cross-sections, evaluations of Jain and Mehta /7/, /8/ have been taken into account in order to obtain the total inelastic cross-sections.

/1/ V.K. Shukla and S.B. Garg ; Evaluation of Measured Neutron Total Cross-Sections of ^{232}Th, NDE (BARC) - 4.

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/8/ H.M. Jain and M.K. Mehta ; Evaluation of ^{232}Th Neutron Capture Cross-Sections Above 50 keV ; NDE (BARC) - 10.

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MULTIPARTICLE REACTION CROSS-SECTIONS*

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Neutron induced multiparticle reaction cross-sections are needed for nuclear safety and radiation damage studies in reactor systems. V, Cr, Mn, Fe and Ni are the structural elements and their charged particle emission cross-sections have not been measured in the energy range extending upto 20 MeV. To fill this gap (n,2n), (n,p), (n,α), (n,np), (n,pn) and (n,nα) reaction cross-sections have been evaluated by making use of Hauser-Feshbach multistep statistical theory /1/ and the preequilibrium exciton model /2/.

Optical model parameters for neutron and proton particle emission are those of Becchetti and Greenlees /3/ and for alpha particle emission are those of McFadden and Satchler /4/. Neutron total cross-sections calculated with these parameters agreed within 6% with those given in ENDF/B-IV cross-section library indicating their adequacy. Transmission coefficients and branching ratios for various reactions have been obtained by considering the energy level structure of target and residual nuclides, their continuum being represented by back shifted Fermi gas model. Pairing energy corrections and level width fluctuations have also been accounted for.

Precompound emission of particles has been taken at 8 MeV or above of neutron incident energies and the evaluations have been made by making use of the exciton model with the reaction matrix constant given by Kalbach /5/. It has been noted that (n,2n), (n,p) + (n,np) + (n,pn) = (n,xp) and (n,α) reaction cross-sections are calculated within 20% of the corresponding measured entities in the magic nuclides ^{51}V and ^{58}Ni. In the case of others the deviations are more. It may be stressed that the measured information on the reactions mentioned in this paper is rather scanty and hence a systematic study of these model based reactions can yield reasonable estimates of charged particle emission cross-sections for direct application in radiation damage calculations.

A few reaction cross-sections are listed in the following Table at 15 MeV to indicate agreements between the calculated and the measured values:

Table

Element	Cross-Section Type	Calculated (mb)	Measured (mb)
Cr-52	(n, 2n)	360.5	412 ± 29
V-51	(n, xp)	89.1	91 ± 14
Ni-58	(n, xp)	1042.3	1000 ± 120
	(n,α)	77.6	98 ± 6
	(n, 2n)	49	39 ± 4
Fe-56	(n, 2n)	478	570
	(n, xp)	309.5	290 ± 22
Cr-52	(n, xp)	384.3	260 ± 25

Spectra of emitted particles, elastic, inelastic and level excitation cross-sections have also been evaluated in the energy range 1 MeV to 20 MeV.

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ANALYSIS OF THERMAL REACTOR BENCHMARKS

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Thermal reactor benchmarks recommended by the Cross-sections Evaluation Working Group of National Nuclear Data Centre of BNL[1] were analysed using the multigroup cross-sections derived from UKNDL. The benchmarks included (i) unreflected critical spheres of uranyl nitrate solution in H_2O, (ii) homogeneous ^{239}Pu critical bare spheres, (iii) TRX-1 and 2 lattices, (iv) BAPL - 1,2,3 lattices and (v) MIT - 4,5,6 lattices. The first two sets of benchmarks are useful for (a) testing H_2O fast scattering data, (b) thermal absorption of hydrogen, (c) ^{235}U thermal data, (d) ^{239}Pu thermal data. The remaining sets of experiments were analysed with the object of validating our calculational procedures vis a vis more sophisticated ones.

The calculated eigenvalues for the ^{235}U spheres were found to be consistently lower than the measured ones with maximum deviation being around 1.5%. The cross-sections for H_2O were in agreement with those recommended in ENDF/B-IV. Use of the ENDF/B-V suggested fission spectrum (with E = 2.03 MeV) and an increase of 0.74% in the thermal ν of ^{235}U brings the calculated eigenvalues very close to the measured values. The effect of these modifications on the calculated parameters for lattice benchmarks is being investigated.

With regard to the examination of ^{239}Pu criticals, the computed eigenvalues were overestimated with a large scatter ($\simeq$ 2.5%) with average calculated K_{eff} = 1.0134. These results are in good agreement with the observations made by other evaluators of the same benchmarks. It has been found that these benchmarks have poor consistency and there exists the need for some well determined experiments for evaluating data for plutonium isotopes.

Analysis of lattice benchmarks brought out very clearly rather poor consistency of D_2O moderated exponential experiments. The cross-sections used by us which are based on WIMS library (derived from UKNDL) are in general adequate in predicting the physics characteristics of thermal reactors.

The results are discussed in detail in Reference 2.

References

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NEUTRON ANGULAR DISTRIBUTIONS IN REACTIONS INDUCED BY 14 MeV NEUTRONS

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In the fast particle model /1/ presquilibrium angular distributions described in terms of a generalized master equation, a crucial assumption is that at each stage there is an identifiable leading particle. The predictions of this model generally underestimate the emission at backward angles /2/. In the present work the exciton states are further classified /3/ as multistep compound (MSC) - including only bound excited states and multistep direct (MSD) - with at least one unbound particle. The angle dependent emission is assumed to arise only from the MSD states. The branching is assumed to occur at the first step with the unbound branch probability,

$$R_u = P_p r_p + P_n r_n \qquad (1)$$

where P_p and P_n are the probabilities of the initial interaction with a target proton or target neutron and r_p and r_n are the ratio of the unbound to the total state densities /4/ for the two resulting configurations taking account of neutron-proton distinction. Assuming identical transition rates for the two chains of states,

$$\frac{d\sigma}{d\epsilon\, d\Omega} = (1-R_u)\sum_n W_\beta(n,\epsilon)\,\mathcal{G}(n) + R_u \sum_l P_l(\cos\theta) \sum_n W_\beta(n,\epsilon)\,\mathcal{G}(n) \qquad (2)$$

where $\mathcal{G}(n)$ are the Legendre coefficients of exciton state lifetimes as in the usual fast particle model.

The results using the present model are shown in Figs 1 and 2 for S(n,n') and Ca(n,n') for $E_n \simeq$ 14 MeV. The dashed curve is the prediction of the usual fast particle model. The present model results in an improved agreement with the data.

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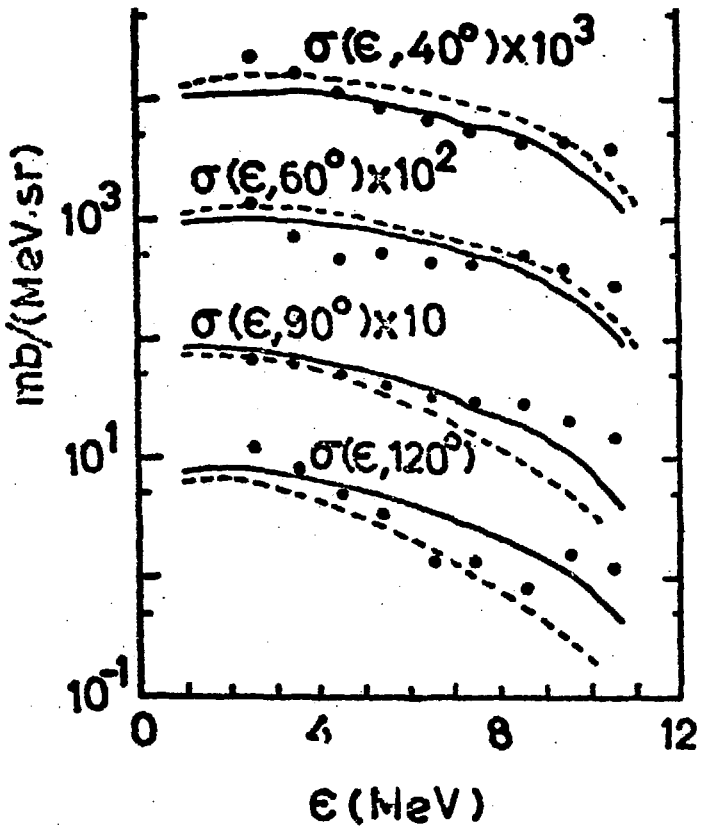

Fig 1 S(n,n')

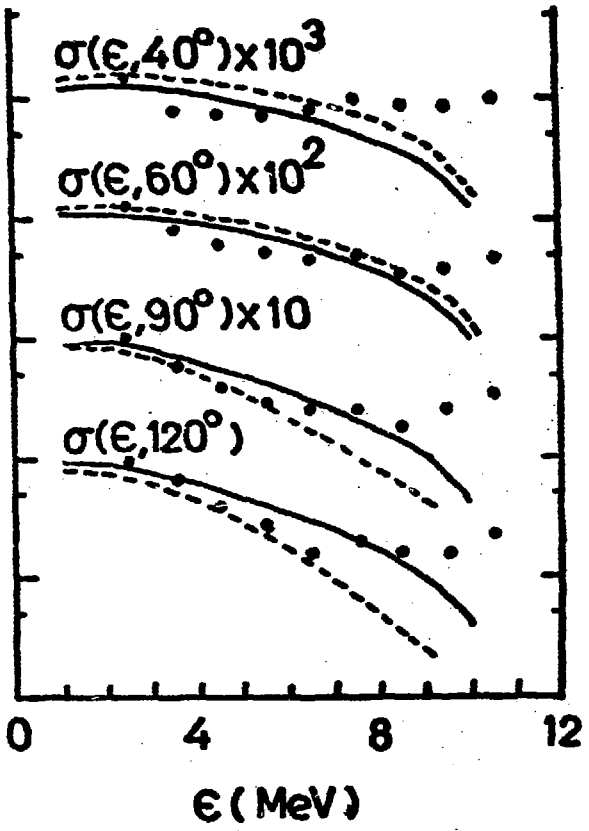

Fig 2 Ca(n,n')

A STUDY OF PRE-EQUILIBRIUM EMISSION IN (n,p) REACTIONS BETWEEN 10-20 MeV

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A computer code ACT has been used to calculate the excitation functions for $^{48,49,50}Ti$, ^{51}V, ^{52}Cr, ^{65}Cu, and $^{64,66}Zn(n,p)$ reactions in the energy range 10 to 20 MeV/1/. Hybrid model /2/ frame work has been used for treating the pre-equilibrium emission. Pre-equilibrium emission is considered only in the first step of de-excitation where the excitation energy is sufficiently large. In these calculations conservation of the parity and angular momentum have been taken into account at each step of de-excitation. Back-shifted Fermi level density parameters have been used, Pauli exclusion principle and width fluctuation correction can also be incorporated optionally. For all calculations level density parameter, the effective moment of inertia and fictive ground state position have been taken from ref./3/. The following values for the other parameters have been used in these calculations : p=2, h=1 and FM=430, where p and h stands respectively for the particles number and the hole degrees of freedom of the initial state. FM is the constant (MeV^3) which defines /4/ the matrix element for internal transitions competing with pre-equilibrium decay. As can be seen from Fig.1 the function of the pre-equilibrium emission increases for each case with increasing excitation energy /1/. At 15.0 MeV excitation energy the pre-equilibrium fraction for $^{48,49,50}Ti$, ^{51}V, ^{52}Cr, ^{65}Cu and $^{64,66}Zn$ target nuclei have been plotted in Fig.2 as a function of atomic mass number A, proton number Z, neutron number N and asymmetry parameter (N-Z)/A. It can be seen from Fig.2 that the pre-equilibrium fraction at this energy varies linearly with mass number, proton number and neutron number in general, except for ^{49}Ti, ^{65}Cu and ^{64}Zn target nuclei. However, no regular variation of the pre-equilibrium fraction has been observed as a function of asymmetry parameter (N-Z)/A. Further investigations are in progress to see

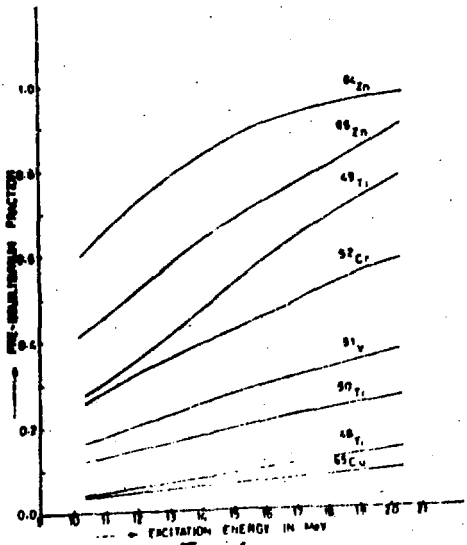

Fig. 1

this trend in other nuclei also.

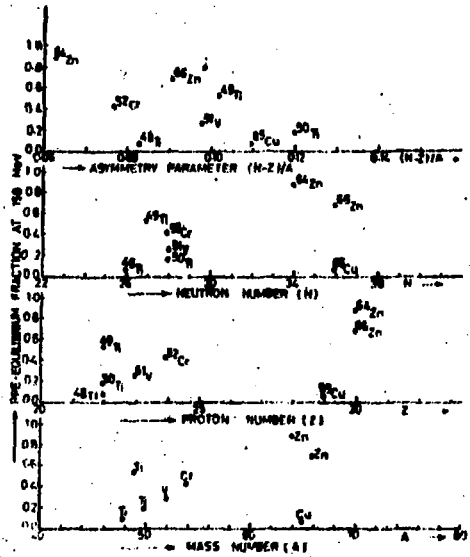

FIG. 2

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EVALUATION OF THRESHOLD REACTION INTERFERENCES IN REACTOR NEUTRON ACTIVATION ANALYSIS USING (n,γ) REACTIONS.

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The estimation of impurities by (n,γ) reactions in reactor neutron activation analysis can suffer from the presence of the high energy component in the reactor neutron spectrum, which can produce the same product nuclide through (threshold) nuclear reactions with the other elements constituting the matrix. Theoretical evaluation of such interferences requires knowledge of the excitation functions and the neutron spectrum in such details that it still remains not "quantitative" for analytical applications. The ultimate interest in analytical estimations is the "yield" which is a convoluted product of $\sigma(E)$ and $\Phi(E)$, integrated over the entire neutron energy spectrum at the irradiation position. Characterising the irradiation positions of the reactor enables the activation analyst to have an awareness of the extent of hard component present and the experimental measurement of the fast neutron reaction yield improves the accuracy in the analytical determination by the (n,γ) reaction. However, this fast neutron component itself can be used with great advantage in analytical applications, as in the case of titanium through scandium by the (n,p) reaction[1] and its application to the estimation of titanium in steels[2].

Measurements of the thermal flux were made with high purity foils of Co/Al alloy containing 0.1% and 1% Cobalt. Irradiations were done at two in-core and three out-of-core positions of the APSARA reactor and the induced ^{60}Co activity was measured using calibrated γ- spectrometers.

Preliminary investigations regarding the hard components of the neutron spectrum were provided by the cadmium ratios, defined as

$$\text{Cd ratio} = \frac{\text{Bare foil activity at zero-time}}{\text{Cd covered foil activity at zero-time}}$$

All cadmium ratios were calculated using 1mm thick interlocking Cd covers.

Five different nuclear reactions were used to determine the integrated fast flux . The threshold monitors were high purity foils irradiated

simultaneously at the five positions (as in the thermal flux measurements) for 24 hours. Table I summarises the observations for all the flux measurements and the cadmium ratios.

The threshold reaction interferences have been evaluated for six nuclear reactions and are expressed as the quantity, in mg, of the element, which through the fast neutron reaction 'would' produce the activity of the product nuclide equivalent to that from 1 ug of the impurity element by (n,γ) reaction. The observations are summarised in Table 2 for two positions A7 and C8. All measurements were done by gamma spectrometry except for ^{32}P for which a chemical separation procedure (phosphorous precipitated as ammonium phospho molybdate) followed by beta counting using a GM counter was adopted. All the irradiations were for a duration of 24 hours except iron, used to evaluate the $^{56}Fe(n,p)^{56}Mn$ reaction, in which case it was for 1 hour. The observations are in accordance with the expectations that the in-core position(A7) with a higher fast component requires much less amount of the interfering element compared to C8 to yield the same activity as would be produced by 1 ug of the (n,γ) element, except in case of $^{63}Cu(n,\alpha)^{60}Co$ reaction which has a high positive 'Q' value.

This type of characterisation has proved to be much useful in our routine analysis of varieties of samples. This exercise of characterisation has to be revised to reflect significant core (fuel) changes.

TABLE I

Flux Measurements made at Apsara

Reaction	$E_{eff.}$ [3,4] MeV	$\sigma_{eff.}$ [3,4] / $\sigma_{th.}$ mb.	Flux ($n.cm^{-2}.s^{-1}$) (in units of 10^{10})				
			A1	A7	C8	E8	D9
$^{59}Co(n,\gamma)^{60}Co$	thermal	37500	260	270	210	130	20
$^{59}Co(n,\gamma)^{60}Co$	Cadmium ratio		17	ND	ND	18	30
$^{48}Ti(n,p)^{48}Sc$	7.2	50	0.074	0.084	0.071	0.046	0.014
$^{46}Ti(n,p)^{46}Sc$	5.5	230	0.480	0.417	0.363	0.230	0.056
$^{54}Fe(n,p)^{54}Mn$	4.2	610	1.96	1.95	1.22	1.19	0.130
$^{58}Ni(n,p)^{58}Co$	2.9	420	3.05	2.83	2.45	1.57	0.357

ND : Not Determined

TABLE 2

Evaluated Threshold Reaction Interferences

Reaction	(n,γ) equivalents), mg	
	A7	C8
$^{35}Cl(n,p)^{32}P$	0.06	0.13
$^{46}Ti(n,p)^{46}Sc$	99	212
$^{56}Fe(n,p)^{56}Mn$	6.5	ND
$^{60}Ni(n,p)^{60}Co$	120	180
$^{63}Cu(n,\alpha)^{60}Co$	500	450
$^{86}Sr(n,p)^{86}Rb$	26	97

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GENERATION AND TESTING OF MULTIGROUP CROSS SECTION SET FOR FAST REACTOR APPLICATIONS

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Using ENDF/B-IV, the generation of multigroup cross section set has been completed with the present version of RAMBHA code system[1] for ^{239}Pu, ^{240}Pu, ^{241}Pu, ^{242}Pu, ^{235}U, ^{238}U, Mo, Na, O, C, Si, Fe, Cr, Ni and Mn in Cadarache 25 group energy structure. A program COMP[2] has been written to compare groupwise and reactionwise partial, total reaction cross sections, inelastic and elastic scattering matrices of the present set with the available French set for each isotope. The comparisons clearly show that as compared to 1969 adjusted French set the present multigroup constants widely differ by more than 50% for structural elements and by 10-15% for heavy elements in some energy regions for both partial and total reaction cross sections.

Sensitivity studies and validation of the new cross section set by analysis of selected fast critical assemblies are nearing completion. The results obtained thus far are encouraging.

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SENSITIVITY METHODLOGY AND APPLICATION PROGRAM AT RRC FOR FAST REACTOR

Analysis

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The measurement, evaluation and processing of radiation interaction data together with experimental and analytic data testing efforts have been the major activities of Reactor Physics Community for several decades. Implicit in such a data acquisition is the goal of generating good quality data to assess the reactor design parameter uncertainties arising from nuclear data related uncertainties. Developments in the field of sensitivity analysis and in particular to those related to generalised perturbation theory have been central to the successful definition of such procedures.

The following program of sensitivity studies at RRC haveben taken up:

1. To determine nuclear data related and method, uncertainties for specified performance parameters in radiation shielding and reactor design problems of Prototype Fast Breeder Reactor from basic cross section data.

2. To provide quantitative guidance to the cross section data measurement, evaluation and processing code development programs at RRC in the form of estimated accuracy requirements for basic nuclear data to meet design constraints and error margins.

3. To provide qualitative and quantitative understanding of the relative importance of basic physical data and interaction process involved in analysing any design or experimental problem so that conclusion drawn from one study can be extrapolated to another especially in the shielding experimental analyses to commercial reactor shield design.

COMPARISONS OF CALCULATED SELF SHIELDING FACTORS WITH MEASURED VALUES FOR ^{239}Pu, ^{235}U, Fe and Na

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The central question about the data representations and their processing in the resonance region is whether or not the present approach leads to a correct estimate of resonance self-shielding[1-4]. Obviously, a direct comparison of calculated self shielding factors (SSFs) with measured values as a function of temperature, energy and background dilution will answer this question. The present work compares the SSFs calculated by J* method of Hwang[5] with those calculated by Cullen's codes[6-9] for ^{235}U and ^{239}Pu using ENDF/B-IV data. The reported experimental values[10-11] of SSFs have been used for comparison with calculated values. For ^{23}Na and Fe, the experimental values[10] of SSFs are compared with values calculated using Cullen's codes[6-9].

The principle behind the semi-empirical determination of self shielding factors is well documented in the literature[10-12]. For J* method the code RAMBHA[13] was used.

The results show that at lower background dilutions there is a wide dispersion among the values of self shielding factors calculated by two different methods and those obtained experimentally. At higher dilutions the values of self shielding factors calculated by both the methods converge towards experimental values satisfactorily. The results were presented in Ref.14.

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TOWARDS EXACT PROCESSING OF CROSS SECTIONS IN RESOLVED RESONANCE REGION

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It has been found desirable to use the preprocessing codes[1-4] LINEAR-RECENT-SIGMA1-GROUPIE for generation of self shielding factors in resolved resonance region so that the self shielded cross sections accurately represent the evaluated data. These codes have been commissioned, tested and extensively used for generation of accurate group constants in the resolved resonance region[5] for many nuclides at RRC. The functions are explained briefly.

1. The code LINEAR converts evaluated cross sections in the ENDF/B format[6] into a tabular form that is subject to linear-linear interpolation in energy and cross section, and removing points that are not needed for linear interpolability. A major advantage of this code is that it allows subsequent codes to consider only linear-linear data.
2. The code RECENT reconstructs the resonance contribution to the cross section in linearly interpolable form, adds in any linearly interpolable background cross section and outputs the result in ENDF/B format. The output includes the original resonance parameters in a form that can be used in Doppler broadening and self-shielding calculations. Since the code RECENT requires the background cross sections to be linearly interpolable, it is necessary to process the ENDF/B file by the code LINEAR before RECENT is used. On the IBM 370/155 the program requires 7.5 mins to reconstruct ^{235}U cross sections to 2% accuracy and on Honeywell DPS 8/46 it is 5.6 mins. The program requires 42 K words of core storage.

In the resolved region the ENDF/B format allows parameters from four resonance formalisms[6] a. Breit-Wigner single-level parameters, b) Breit-Wigner multilevel parameters, c) Reich-Moore parameters and d) Adler-Adler parameters. The program reads parameters in any of these formats and computes the resonance contribution for all except Reich-Moore parameters, which are ignored.

3. The code SIGMA1 Doppler broadens evaluated cross sections in the ENDF/B format. The code requires data as linearly interpolable functions of energy and outputs broadened cross sections in the ENDF/B format.

The SIGMA1 method can be used to Doppler broaden cross sections for any particle viz. neutron, proton etc. incident at non-relativistic energies on a target in which the free atom approximation is valid.

4. The code GROUPIE reads evaluated data in the ENDF/B format and calculates any combination of the following quantities a. unshielded group averaged cross sections, b. Bondarenko self-shielded group averaged cross sections, c. Multiband parameters.

The code allows for specifying arbitrary energy groups and an arbitrary energy dependent neutron spectrum. All integrals are performed analytically to the required accuracy.

The code system has been utilised to generate self shielding factors for many elements of interest: Fe, Na, Pu isotopes, U isotopes Mn, Mo etc. in resolved resonance region.

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ON THE NEED FOR CHANGING THE ENDF/B CONVENTION FOR THE REPRESENTATION OF CROSS-SECTIONS IN THE UNRESOLVED RESONANCE REGION OF FERTILE AND FISSILE NUCLEI

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A more fundamental scientific basis for the prediction of the Doppler effect in fast reactor systems demands that the usual ENDF/B and KEDAK conventions for the representation of cross-sections in the unresolved resonance region should be changed. This conclusion[1] is based upon the experience acquired in evaluations of statistical mean resonance parameter sets made at RRC Kalpakkam, for use in neutronic calculations of fast reactors in the recent past for ^{235}U, ^{239}Pu, ^{232}Th, ^{233}U and ^{238}U. A brief account of several sensitivity studies and neutronic calculations made with these mean resonance data sets was presented[1]. The existence of non- uniqueness of mean resonance data sets has been clearly established and an uncertainty due to the choice of the mean resonance data set is found to be associated with the theoretical calculations of neutronic parameters of reactor systems. This uncertainty, identified at Kalpakkam, has not received attention in the past and is found to be significant in the case of the calculation of the Doppler coefficient in fast reactors. It should have been taken into account in the interpretation of the Doppler effect experiments performed in fast critical facilities for both fissile and fertile samples.

The present method of statistical representation of cross-sections leads to a higher statistical uncertainty in the prediction of the Doppler reactivity effect under coolant voided conditions in fast power reactors than in the normal case. Also the effect of inclusion of intermediate structure

in the fission cross section for ^{239}Pu on the temperature derivative of self shielding factors is significant and cannot be modelled satisfactorily at present.

If the high resolution cross-section measurements, as suggested by de Saussure and Perez, are made available for the main fissile and fertile isotopes, thinning and direct Doppler broadening of the cross-section data by the preprocessing methods developed by Cullen and his coworkers appear to be a promising approach. Until such time the evaluated nuclear data files should attempt to contain self-shielding factors directly evaluated with the support of self-indication and transmission measurements instead of the present convention of having mean resonance data sets in the unresolved region.

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Half-life of ^{242}Cm

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The half-life of ^{242}Cm has been determined by two independent methods (i) by direct decay in α- proportional counter and (ii) by preparing synthetic mixtures of ^{242}Cm and ^{244}Cm and studying periodically the decrease in $^{242}Cm/^{244}Cm$ alpha activity ratio by α- spectrometry. The half-life values obtained are 163.17 ± 0.06 day and 162.82 ± 0.21 day respectively. Average of these two independent values gives a half-life value of 163.00 ± 0.11 days. The uncertainty specified on the values is one standard deviation obtained in the present work is significantly different from the value 161.35 ± 0.30 d reported from JAERI in 1981 and compares well with the values reported in 1975 and 1979.