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भारत सरकार
GOVERNMENT OF INDIA
परमाणु ऊर्जा आयोग
ATOMIC ENERGY COMMISSION

PROGRESS REPORT ON NUCLEAR DATA ACTIVITIES IN INDIA

for the period from January 1983 to June 1984

Compiled by
R. P. Anand
Nuclear Physics Division

भाभा परमाणु अनुसंधान केन्द्र
BHABHA ATOMIC RESEARCH CENTRE
बंबई, भारत
BOMBAY, INDIA
1984

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INIS Subject Cagtegory: A3400

Descriptors

NEUTRON REACTIONS

FISSION

FISSION YIELD

CROSS SECTIONS

FAST NEUTRONS

MULTIGROUP THEORY

EXPERIMENTAL DATA

COMPUTER CALCULATIONS

RESEARCH PROGRAMS

INDIA

PREFACE

The present progress report on Nuclear Data Activities in India is the third in the new series of progress reports, the first of which was brought out in 1981. It covers the period from January, 83 to June, 84.

It contains information about nuclear data measurements, compilation and evaluation works being carried out at B.A.R.C., Bombay; R.R.C., Kalpakkam and at other institutions in the country.

The document contains information of a private nature and should not be quoted without prior permission from the authors.

S.S. Kapoor
Member, International Nuclear
Data Committee.

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I. NUCLEAR FISSION

MASS, ENERGY CORRELATIONS IN ^{4}He INDUCED FISSION OF ^{238}U AT MEDIUM ENERGIES

R.K. CHOUDHURY, ALOK SAXENA, S.S. KAPOOR, D.M. NADKARNI, V.S. RAMAMURTHY and P.P.SINGH

Bhabha Atomic Research Centre
Trombay, Bombay 400085.

The energy and mass distribution of fission fragments in the fission of ^{238}U induced by alpha particles of bombarding energies of 32,40,48,56 MeV, were studied. The measurements were done by using two surface barrier detectors in the back-to-back geometry and the coincident pulse height information for each fission event was recorded on a magnetic tape. The pulse height data were analyzed off line to obtain mass distributions and total kinetic energy distributions. The variation in the peak to valley ratio of the mass distribution was studied with increasing bombarding energy. The detailed correlations between the fragment mass (M_H) and the total kinetic energy (E_K) were obtained. The width of the mass distribution, (σM_H) is found to decrease with increasing kinetic energy whereas, $\overline{M}_H$ remains almost constant upto most probable kinetic energy of 174 MeV and then it decreases with E_K. The implications of these results on the excitation energy dependence of the fission process will be discussed. Also from the present data the possibility of deducing the first chance mass distribution at these energies will be discussed.

ANALYSIS OF SIMULTANEOUS FRAGMENT MASS AND CHARGE DATA FROM THERMAL NEUTRON INDUCED FISSION OF ^{235}U

REKHA GOVIL, D.M. NADKARNI and S.S. KAPOOR
Bhabha Atomic Research Centre
Bombay 400 085.

The mean values and the variances of fragment mass, charge and neutron distributions are determined as a function of the total kinetic energy E_k, in the thermal neutron induced fission of ^{235}U using a back-to-back ΔE-E detector system. These parameters, which are a function of the kinetic energy, summarize the properties of the nucleon exchange process at the time of scission. The results suggest that in the fission process also the distribution of mass and charge is brought about by nucleon exchange process and the degree of neutron-proton correlation is found to be strongly dependent on E_k.

ANGULAR DISTRIBUTION OF LONG-RANGE ALPHA PARTICLES IN FAST NEUTRON-INDUCED FISSION OF ^{235}U.

M.M. Sharma, S.C.L. Sharma, A.K. Sinha and G.K. Mehta
Indian Institute of Technology, Kanpur-208016
INDIA

Angular distribution of long-range alpha particles emitted in the fission of ^{235}U induced by fast neutrons has been measured about the neutron-beam direction. The technique /1/ employs a ΔE-E semiconductor detector telescope to derive the angle information about the detector axis. The technique is also capable of identifying the particles by the virtue of using a particle telescope.

The experiment was carried out at several neutron energies between thermal and 1 MeV and the anisotropies in the angular distribution of alpha particles about the neutron-beam direction were determined.

The following table gives the results of such measurements/2/.

Table : Anisotropies $(Y(0^{\circ})/Y(90^{\circ})$ of the ternary alpha particle angular distribution.

Neutron Energy	Anisotropy
(140 ± 30) keV	(-85 ± 28) %
(170 ± 25) keV	(-87 ± 32) %
(200 ± 25) keV	(-94 ± 31) %
(400 ± 200) keV	(-10 ± 28) %
(600 ± 180) keV	(-25 ± 19) %
(1000 ± 170) keV	(-50 ± 27) %

It turns out that the alpha particle angular distribution in ternary fission of ^{235}U is peaked perpendicular to the neutron-beam direction at neutron energies in the above region. This would imply that the ternary fragment angular distribution peaks fore and aft about the beam direction. This behaviour is similar to that of binary fragments in this energy region.

/1/ M.M. Sharma, S.C.L. Sharma, A.K. Sinha and G.K. Mehta, Nucl. Inst. and Meth. in Phys. Res. (In press)

/2/ M.M. Sharma, G.K. Mehta, Communicated to Prämana (1984).

ANGULAR DISTRIBUTION OF POLAR LIGHT CHARGED PARTICLES IN THERMAL NEUTRON-INDUCED FISSION OF ^{235}U

M.M. Sharma, A.K. Sinha and G.K. Mehta
Indian Institute of Technology, Kanpur-208016, INDIA

D.M. Nadkarni
Bhabha Atomic Research Centre, Bombay-400085, INDIA.

The angular distribution of polar light charged particles (LCPs) emitted in thermal neutron-induced fission of ^{235}U has been measured. The method of detection of polar LCPs consists of using a pair of ionization chamber with a multi-hole collimator as a common cathode. The ionization chambers detect the fission fragments related to polar and equatorial LCPs, the LCPs being detected by a semiconductor ΔE-E telescope. The collimator arrangement thus separates the polar and equatorial events. Using different collimation for polar LCP region of angular distribution, the yields of polar protons, tritons and alpha particles were measured. Using Mote Carlo simulation for the detection system, $\sigma(\theta)$ for the angular distribution of polar protons, tritons and alpha particles were determined (refer to table).

Table

The width of the angular distribution of polar LCPs.

LCPs	$\sigma(\theta)$
Alpha particle	$(28.0\pm7.0)^{\circ}$
Proton	$(13.0\pm6.0)^{\circ}$
Triton	$(25.0\pm15.0)^{\circ}$

The errors indicated in the table are statistical only. The angular distribution of polar protons was found to be narrow compared to a wide angular distribution of polar tritons and alpha particles as indicated in the table.

Cumulative yields of rare-earth fission products in the spontaneous fission of ^{252}Cf

B.S. Tomar, H. Naik, A. Ramaswamy, Satya Prakash and M.V. Ramaniah

In view of the paucity of data on the mass yields of short lived rare-earth fission products, the present study was aimed at determining the cumulative yields of mainly short lived rare earth isotopes in spontaneous fission of ^{252}Cf.

Fission products from a $\sim 3/\mu g$ ^{252}Cf source were collected on 1 mil aluminium catcher foils. The collection time was varied depending on the half life of the isotope under study. The catcher foils were dissolved in dil HCl in presence of lanthanum carrier. Rare earths were separated by fast one step fluoride and hydroxide precipitation and this precipitate was dissolved in 5 ml dil HCl and counted. The counting was done on a precalibrated 8% HPGe detector, (having a resolution of 1.8 Kev at 1332 Kev) coupled to a 4096 channel analyser.

The gamma ray peak areas were corrected for absolute disintegration rate by using gamma ray abundance[1] and detector efficiency. The yields of rare-earths isotopes were determined relative to the yield of ^{146}Ce[2] and are shown in table 1 alongwith the literature values[3-5]. The present yield values are in good agreement with the literature data except in case of ^{151}Nd. The fission yield of ^{145}Ce is determined for the first time. The large difference between our value and the literature value for ^{151}Nd could be plausibly due to the large variation in gamma ray abundance values taken for yield calculation.

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A.J. Deruytter.
Nucl. Inst. Meth, 134, 299 (1976).

5. L. Toppare, H.N. Erten and N.K. Aras.
Technical Journal, Turkish, Atomic Energy Commission I, 8 (1980).

Table 1

Fission yields of Rare Earth Isotopes in the spontaneous fission of ^{252}Cf

	Nuclide	Half life (a)	Gamma Energy Kev	Present work	Yield value Ref 3	Ref 4	Ref 5
1.	^{141}Ce	32.5 D	145.4	6.00±0.67	6.00	6.13	-
2.	^{143}Ce	33.0 H	293.3	6.20±0.71	6.22	6.13	-
3.	^{145}Ce	2.98 M	724.0	4.18±0.71	-	-	-
4.	^{146}Ce	14.2 M	218.0	4.08±0.41	-	5.18	3.82
5.	^{147}Nd	10.98 D	91.0	3.30±0.37	4.26	4.10	-
6.	^{149}Nd	1.73 H	211.3	2.42±0.27	2.71	2.74	2.42
7.	^{151}Nd	12.44 M	116.3	0.94±0.12	-	-	1.72
8.	^{151}Pm	28.4 H	340.0	1.40±0.14	1.81	1.99	-
9.	^{153}Sm	46.7 H	103.2	1.35±0.16	1.33	1.31	-
10.	^{155}Sm	22.1 M	104.0	0.69±0.07	-	0.84	-

(a) D = days H = Hours, M = Minutes

Isotopic yield distribution of technetium isotopes in the thermal neutron induced fission of ^{233}U, ^{235}U and ^{239}Pu.

Alok Srivastava, A. Goswami, A.G.C. Nair, B.K. Srivastava, S.B. Manohar, Satya Prakash and M.V. Ramaniah.

In continuation of our work on the systematic study of nuclear charge distribution the fractional independent yield of technetium isotopes mainly ^{101}Tc, ^{103}Tc, ^{104}Tc and ^{105}Tc have been determined by separating Tc at various cooling times as Tetraphenyl arsonium technate from other fission product which were obtained by irradiating fissile materials in solution form in CIRUS reactor for about 20 sec, using pneumatic carrier facility. The activity of the samples were assayed using a precalibrated 8% HPGe detector coupled to a 4 K channel analyser. From the activity and cooling time, the fractional independent yields of technetium isotopes were obtained using standard decay growth relations.

The yields are considerably lower for ^{233}U and ^{235}U compared to ^{239}Pu fission. The low yields are explained in terms of the neutron to proton ratio of products being away from the neutron to proton ratio of the fissioning system. Also the 82 closed N shell of the complementary product is ^{239}Pu has influence on the yields.

1. Alok Srivastava, A.G.C. Nair, B.K. Srivastava, S.B. Manohar, Satya Prakash and M.V. Ramaniah,
Radiochim Acta (In Press).

FRACTIONAL INDEPENDENT YIELD (FIY) OF TECHNETIUM ISOTOPES

NUCLIDE	$T_{\frac{1}{2}}$	GAMMA RAY ENERGY USED	$^{233}U(n_{th},f)$	$^{235}U(n_{th},f)$	$^{239}Pu(n_{th},f)$	$^{*252}Cf(SF)$
^{101}Tc	14.0 mins	307.0 Kev	0.007	-	0.0023±0.0015	0.007±0.002
^{103}Tc	50 secs	346.0 Kev	0.01±0.02	-	0.0346±0.0141	0.045±0.010
^{104}Tc	18.0 mins	358.0 Kev	0.02±0.02	0.02	0.0900±0.0139	0.078±0.022
^{105}Tc	7.6 mins	108.0, 159.0 Kev	0.04±0.02	0.03	0.1947±0.0452	0.339±0.064

* Ref. 1.

Effect of excitation energy on the isotopic yield distribution of Iodine in the fission of ^{236}U.

A.V.R. Reddy, T. Datta, S.M. Deshmukh, S.M. Sahakundu, S.B. Manohar, Satya Prakash and M.V. Ramaniah

With an aim to examine the role of excitation energy on the charge/mass polarization, studies on isotopic yield distribution of iodine in the alpha induced fission of ^{232}Th ($^{236}U^*$, $E^* \approx 23.4$ MeV) were carried out and the obtained systematics of the distribution were compared with the same in the thermal neutron induced fission of ^{235}U ($^{236}U^*$, $E^* = 6.5$ MeV). Irradiations were carried out using external α-particle beam of energy 30 ± 0.3 MeV at the 88" Variable Energy Cyclotron, Calcutta using thorium metal foil (29.7 mg/cm^2) target at a beam current of 2.5 μA. Independent yields of iodine (130-134) and cumulative yields of its precursors (Tellurium 131-134) were determined with and without radiochemical separations for iodine using high resolution gamma spectrometry, relative to the cumulative yield of ^{135}I, and are given in Tables 1 and 2 respectively.

Using the obtained independent yields of iodine in the present work, the parameters of isotopic yield distribution via most probable mass (A_p) and width of distribution (σ_A) were evaluated as 135.09 and 1.6 ± 0.16 mass units respectively and the corresponding values in low energy system are 135.90 and 1.21 ± 0.12 mass units respectively. Increase in σ_A and less deviation of most probable mass with increase in E^* from the one expected from unchanged charge distribution indicates that the charge division approaches towards statistical nature[1] resulting in unchanged charge distribution with increase in excitation energy.

1. A.V.R. Reddy, T. Datta, S.M. Deshmukh, S.M. Sahakundu, S.B. Manohar, Satya Prakash and M.V. Ramaniah. Communicated to Phys. Rev. C.

TABLE 1

Independent Yields of Iodine with relevant Nuclear Data

S.No.	Nuclide	Half-life	Gamma energy (keV)	% Abundance	Independent Yield	Isomeric Yield Ratio
1.	130	12.38 h	536.0	99	0.0178 + 0.0023	-
2.	131	8.04 d	364.5	81.2	0.0241 + 0.0038	-
3.	132 m	1.39 h	667.7	13.2	0.151 + 0.025	0.73 + 0.06
	132 g	2.30 h	667.7	99		
4.	133	20.80 h	529.9	87	0.448 + 0.068	
5.	134	52.6 m	847.0	95.4	0.755 + 0.098	1●
			884.0	65.4		
6.	135	6.61 h	1260	28.6	1*	

h = hours, d = days, m = minutes

● = assumed, * = standard value

TABLE 2

Cumulative Yields of Tellurium Isotopes

S.No.	Nuclide	Half-life	Gamma energy (keV)	% Abundance	Cumulative yields	Isomeric Yield Ratio
1.	131 m	30.0 h	149.7	24.2		
	131 g	25.0 m	149.8	67.7	0.605 + 0.060	0.516 + 0.050
2.	132	78.0 h	228	85.0	1.15 + 0.10	-
3.	133 m	55.4 m	912	80.0	0.489 + 0.050	0.593 + 0.085
4.	134	41.8 m	210	23.0	0.440 + 0.055	

Studies on fragment angular momentum in low energy fission

T. Datta, S.P. Dange, R. Guin, H. Naik and Satya Prakash

In order to study the influence of fragment deformation and charge split on fragment angular momentum, angular momenta have been evaluated from radiochemically determined independent isomeric yield ratio of $^{138}Cs^{m/g}$ in $^{252}Cf(SF)$ and of $^{131,133}Te^{m/g}$ in $^{241}Pu(n,f)$ systems using statistical model analysis for spin redistribution during fragment deexcitation. The independent isomeric yield ratio for $^{138}Cs^{m/g}$ was obtained from follow-up of the 463 KeV composite gamma line after dissolution of the aluminium foil used to collect the fission fragments from $^{252}Cf(SF)$ source. The independent isomeric yield ratio for $^{131,133}Te$ were obtained from irradiation of ^{241}Pu at the APSARA reactor using the same methodology as described elsewhere[1]. The independent isomeric yield ratios and the fragment angular momenta are shown in Table 1.

The high angular momentum for ^{138}Cs has been explained in terms of high scission point deformation of fragments in the vicinity of shell closure (N 82) due to core polarization caused by odd proton. The angular momenta of $^{131,133}Te$ remain low and are comparable to the same in other low energy actinide fissioning system possibly due to low deformation expected in view of shell closure proximity and even Z nature showing that the influence of post-scission coulomb excitation is not very prominent.

1. T. Datta, S.M. Sahakundu, S.P. Dange, N. Chakravarty, R. Guin and Satya Prakash, Phys. Rev. C. 28, 1206 (1983).

TABLE 1

Fissioning System	Fission Product	Independent Isomeric Yield Ratio	Fragment Angular Momenta
$^{241}Pu(n, f)$	^{131}Te	0.526 ± 0.063	5.8 ± 1.1
	^{133}Te	0.536 ± 0.064	5.95 ± 1.2
$^{252}Cf(SF)$	^{138}Cs	0.582 ± 0.068	9.8 ± 1.2

Studies on fragment angular momentum in medium energy fission

T. Datta, S.M. Sahakundu, S.P. Dange, R. Guin, Satya Prakash and M.V.Ramaniah

With the objective of investigating the influence of entrance channel angular momentum and excitation energy of the compound nucleus one fission fragment angular momentum, the angular momentum of fragments corresponding to fission products $^{131,133}Te$ in $^{232}Th(\alpha_{40\ MeV}, f)$ have been determined.

The irradiations were carried out using external α-particle beam of energy 40 ± 0.4 MeV at the 88" Variable Energy Cyclotron, Calcutta using thorium metal foil (29.7 mg/cm^2) target at beam current ~ 1-2 μA. Independent isomeric yield ratios of the isomeric fission product pairs were determined radiochemically from follow up of the specific composite gamma lines 149, 312 KeV for $^{131}Te^{m/g}$, $^{133}Te^{m/g}$ respectively on a high resolution semiconductor gamma spectrometer, after correcting for precursor contribution evaluated from charge distribution systematics. Fragment angular momenta were deduced in the framework of statistical model to correct for spin-change (redistributed) during fragment deexcitation(1).

It was seen that for $^{131,133}Te$ the angular momenta do not change in particular for ^{131}Te, significantly compared to the same in the same fissioning system ^{236}U formed through $^{235}U + n_{th}$ process, although excitation energy and angular momenta are much different in $^{232}Th + \alpha_{40\ MeV}$ system as given in Table 1.

The near constancy of fragment angular momenta in the magic shell region nuclei (mass 132) compared to the expected enhancement due to (i) increase in temperature (excitation energy effect), (ii) retainment of part of initial angular momentum of $^{236}U^*$ by the fragments, (iii) increased deformation at higher energy is ascribed to (a) higher rigidity towards bending mode oscillation of $^{236}U^*$ at higher angular momentum, (b) low deformability of even Z, shell region fragments, (c) faster time of descent at higher energy fission.

1. T. Datta, S.M. Sahakundu, S.P. Dange, N. Chakravarty, R. Guin and Satya Prakash, Phys. Rev. C. 28, 7206 (1983).

TABLE 1

Fissioning System	Excitation Energy (MeV)	Angular Momentum	Fragment Angular Moments	
			^{131}Te	^{133}Te
$^{235}U(n_{th}, f)$	6.5	3-4	6.0 ± 1.5	5.9 ± 1.0
$^{232}Th(\alpha, f)$	33.2	14.2	5.7 ± 1.0	7.3 ± 1.2

Cumulative yields of short-lived ruthenium isotopes in the thermal neutron induced fission of ^{233}U, ^{235}U and ^{239}Pu.

A.G.C. Nair, Alok Srivastava, A. Goswami and B.K. Srivastava

Cumulative yields of ^{107}Ru, ^{108}Ru and ^{109}Ru in the fission of ^{234}U, ^{236}U and ^{240}Pu have been determined using fast radiochemical separation techniques followed by high resolution γ-ray spectrometry. The fissile materials (10 μg) were irradiated in pneumatic carrier facility of CIRUS reactor for 20 sec. The ruthenium isotopes were separated from the irradiated target by distilling ruthenium as RuO_4 from fission products in presence of $NaBiO_3$ and was collected in ice cooled 1M HCl. Using this method a chemical yield of 50% and high decontamination ($\gtrsim$1000) from technetium and halogens were obtained. Cumulative yields of ^{107}Ru, ^{108}Ru and ^{109}Ru were calculated from the experimentally obtained count rates which are corrected for detection efficiency and -ray abundances, using standard equations. Further, these yields were converted into chain yields assuming normal charge distribution systematics for comparison with the literature data on chain yields. The following table gives the values of cumulative yields and chain yields.

TABLE 1

Cumulative Yields of Ruthenium Isotopes in the thermal neutron induced fission of ^{233}U, ^{235}U and ^{239}Pu

Mass	^{233}U(%)	^{235}U(%)	^{239}Pu(%)	Reference
107	0.111 ± 0.012	0.145 ± 0.032 0.13 ± 0.01*	3.068 ± 0.284	This work
108	0.077 ± 0.005	0.067 ± 0.017 0.042 ± 0.002*	2.005 ± 0.165 1.89 ± 0.05@	This work
109	-	0.034 ± 0.004 0.027 ± 0.003*	1.088 ± 0.98	This work

* P. Fettweis, P. Del Marmol, Z. Physik A. 275, 359 (1975).

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Mass Distribution of $^{232}Th(\alpha,f)$ with 40 MeV Alpha Particles

R. Guin, N. Chakravarty, S.M. Saha Kundu, S.B. Manohar, Satya Prakash and M.V. Ramaniah

This work was performed as a continuation of the programme to study the energy and angular momentum dependence of the fission product mass distribution in the fission of ^{232}Th induced by alpha particles. The earlier work with 30 MeV alphas are reported in (1).

Thorium metal foils (29.7 mg/cm^2) covered with 0.0025 cm thick pure Al foils and were irradiated using external α-particle beam of energy 40±0.4 MeV at the VEC in Calcutta. The beam current was around 1.5 μA and the duration of the irradiation was varied depending upon the half life of the nuclide studied. The fission products were assayed using high precalibrated HPGe detector coupled to a 4K channel analyser. The yields for twenty six fission products are given in Table 1 which were evaluated using charge distribution systematics[2]. The mass distribution shows the existence of a symmetric peak giving a triple humped distribution. To find out the contribution of symmetric component in mass distribution, and its dependence on excitation energy further work at different excitation energies using radiochemical separations is being pursued.

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Nucl. Phys. A 165, 65 (1971).

TABLE 1

Fission Products Yields in the Reaction $^{232}Th(\alpha, f)$ with 40 MeV Alpha Particles

Nuclide	Gamma Energy (KeV)	Abundance (%)	Half-Life	Chain Yields (cm^2)
^{78}As	613.6	54.0	1.51 h	4.42×10^{-28}
^{85}Kr	304.5	14.0	4.48 h	$(2.94 \pm 0.15) \times 10^{-27}$
^{87}Kr	402.7	49.7	76.4 m	$(4.34 \pm 0.31) \times 10^{-27}$
^{88}Kr	196.3	26.3	2.84 h	$(5.37 \pm 0.24) \times 10^{-27}$
^{89}Rb	1248.1	46.7	15.4 m	6.23×10^{-27}
^{91}Sr	1024.3	33.5	9.48 h	$(5.9 \pm 0.24) \times 10^{-27}$
^{92}Sr	1384.0	90.0 ± 11.0	2.71 h	$6.21 \pm 0.7 \times 10^{-27}$
^{94}Y	918.8	49.9	18.6 m	6.5×10^{-27}
^{99}Mo	181.1	6.06	66 h	6.5×10^{-27}
^{103}Ru	497.1	89.3	39.4 d	5.6×10^{-27}
^{104}Te	358	89.0	18.2 m	5.25×10^{-27}
^{105}Rh	319	19.2	35.4 h	4.32×10^{-27}
^{107}Rh	392.5	8.9	21.7 m	3.06×10^{-27}
^{112}Pd	617.4	49.9	21.05 h	2.84×10^{-27}
^{113}Ag	298.4	9.46	5.37 h	$(2.88 \pm 0.8) \times 10^{-27}$
^{115}Cd	336.2	46.1	53.5 h	1.75×10^{-27}
^{117}Cd	1303.3	18.3	2.40 h	$(1.41 \pm 0.02) \times 10^{-27}$
^{127}Sb	685.7	35.7	3.85 d	4.25×10^{-27}
^{129}Sb	544.7	18.1	4.32 h	6.9×10^{-27}
^{131}I	364.5	81.2	8.04 d	6.72×10^{-27}

II. NEUTRON CROSS-SECTIONS

RADIATIVE CAPTURE OF FAST NEUTRONS IN Gd-160

R.K.Y. SINGH, R.P. GAUTAM, M.A. ANSARI and M.L. SEHGAL. Department of Physics, Aligarh Muslim University and S.KAILAS, Nuclear Physics Divn., Bhabha Atomic Research Centre, Trombay, Bombay.

Neutron Capture cross sections have been measured for the reaction $^{160}Gd(n,\gamma)^{161}Gd$ in the neutron energy range of 0.48 MeV to 3.04 MeV. Monoenergetic neutrons of suitable intensity covering this energy range were produced by using the $^{7}Li\ (p,n)^{7}Be$ and $^{3}T(p,n)^{3}He$ reactions and the protons from the Van de Graaff accelerator at Trombay. The capture cross sections were obtained by measuring the activity of ^{161}Gd produced in the reaction with a 100 cc Ge(Li) detector. We employed a ^{127}I sample along with the ^{160}Gd target and carried out the irradiation. We determined the Gd(n,γ) cross sections normalising it to $^{127}I\ (n,\gamma)^{128}I$ cross section as standard, at each neutron energy. The results of the present measurement are given in the Table.

$^{160}Gd(n,\gamma)^{161}Gd$

E_n (MeV)	$\sigma_{n,\gamma}$ (mb)	E_n (MeV)	$\sigma_{n,\gamma}$ (mb)
0.480±0.040	29.2±6.0	1.200±0.290	30.2±2.6
0.600±0.060	37.7±7.2	1.540±0.260	35.2±2.2
0.780±0.070	35.4±2.3	2.190±0.210	15.7±2.1
0.980±0.220	41.9±8.6	3.040±0.180	7.7±1.5

NEUTRON CAPTURE CROSS -SECTION OF ^{232}Th

R.P.Anand, H.M. Jain*, S. Kailas
S.K.Gupta, V.S. Ramamurthy and
S.S.Kapoor

Bhabha Atomic Research Centre,
Bombay-400085, India.

The neutron capture cross-sections of ^{232}Th have been measured relative to gold in the neutron energy range of 450 -1100 KeV. Monoenergetic neutrons were produced from the ^{7}Li(p,n) reaction using the mono-energetic protons from the Trombay Van de Graaff Accelerator and a liquid nitrogen cooled Li - metal target. A high resolution Ge(Li) detector was employed for measuring the gamma activity of the product nucleus ^{233}Th. These measurements were also carried out at thermal neutron energy for the overall normalization. Small variations in the neutron flux with time were taken into account while calculating the cross-sections. The effect of self absorption and multiple scattering of neutrons in the Thorium sample were also considered. The measured cross section values are compared with the other recent measurements and evaluations.

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* Experimental Reactor Physics Section, Bhabha Atomic Research Centre, Bombay 400085, India.

PREEQUILIBRIUM PARTICLE EMISSION SPECTRA AND MULTIPARTICLE REACTION CROSS-SECTIONS OF NIOBIUM*, S.B. Garg and Amar Sinha, Neutron Physics Division, Bhabha Atomic Research Centre, Trombay, Bombay 400 085

Preequilibrium model and multistep Hauser-Feshbach theory are two important tools which are increasingly utilized in the analysis and prediction of nuclear induced multiparticle reaction cross-sections, particle emission spectra and their angular distributions. We have selected niobium for the application of these models particularly because its neutron induced particle emission spectra and their angular distributions have been well measured at 14.6 MeV and it adequately serves as a test case for these models. Niobium is also considered to be a neutron multiplier in fusion-fission reactor systems and thus it is a fit case for cross-section evaluation in the MeV energy range.

We have adopted the generalized exciton model of Mantzouranis et al /1/ for the analysis of preequilibrium angular distributions. According to this model the exciton state of a composite nucleus is represented by the exciton number n and a direction Ω which corresponds to the direction of the incoming particle. It is assumed that on emission the direction of the emitted particle coincides with Ω.

A combination of preequilibrium exciton model and multistep Hauser-Feshbach theory /2/ has been used to evaluate (n,n'), (n,2n), (n,p), (n,α), (n,np), (n,pn) and (n,αn) reaction cross-sections at 10,14.6, 20, and 25.7 MeV. Gamma emission has been accounted for according to the Brink-Axel giant resonance theory /3/.

Transmission coefficients for the neutron, proton and alpha channels have been computed with the spherical optical model. The potential parameters for neutrons and protons have been taken from Perey /4/ and those for alpha particles from Huizenga and Igo /5/.

The discrete levels of the residual nuclides and their level density parameters have been taken from ref. /6/ and the continuum of levels has been estimated according to Gilbert-Cameron /7/ formalism of level densities.

The preequilibrium contribution has been obtained by extracting the value of K- the interaction matrix constant as given by Kalbach /8/ to fit the angle integrated total neutron emission spectrum $d\sigma/d\epsilon$ at 14.6 MeV and ϵ varying from 6 to 9 MeV. We have also used the following expression for the average transition matrix elements

$$|M|^2 = K A^{-3} E^{-1}$$

and extracted the value of K for evaluating the angular distributions of the first emitted neutron, proton and alpha particles. The master equation and the closed form approaches have been adopted to obtain angle integrated particle emission spectra. It has been observed that the preequilibrium emission becomes more predominant with the increasing energy of the incident particle. Some of the evaluated data are given in the following tables. The experimental data are taken from Hermsdorf et al /9/.

Table 1

Angle Integrated Neutron Spectrum $\frac{d\sigma}{d\epsilon}$ at 14.6 MeV

ϵ (MeV)	Exptl (mb/MeV)	$\|M\|^2$ (Kalbach) calculated (mb/MeV)	$\|M\|^2$ (other) calculated (mb/MeV)
6-7	56.0 ± 6	53.27	56.9
7-8	46.9 ± 5	45.89	43.7
8-9	36.5 ± 4	38.21	35.0

Table 2

Angular Distributions at 14.6 MeV (mb/st. MeV)

Angle (degrees)	ϵ = 7 MeV		ϵ = 8 MeV		ϵ = 9 MeV	
	Calc	Expt.	Calc	Expl.	Calc	Expl.
40	7.07	7.30	5.96	6.94	4.99	4.95
60	5.68	5.23	4.70	5.93	3.89	5.62
90	3.51	3.11	2.74	2.48	2.19	0.76
120	1.86	1.92	1.27	1.38	0.92	0.75

* Communicated to NEA Data Bank for inclusion in the International Nuclear Model Code Comparison Studies. Also included in the Nuclear Physics and Solid State Physics Symp. (1983)

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MULTIGROUP PHOTON INTERACTION CROSS-SECTIONS WITH P5-ANISOTROPIC SCATTERING MATRICES, S.B. Garg, Neutron Physics Division, Bhabha Atomic Research Centre, Trombay, Bombay 400 085

Secondary gamma ray production and photon interaction cross-sections are needed for shielding, safety and radiation transport studies. 25 group photon interaction cross-sections in the energy range 20 MeV to 100 eV have been generated for about 44 elements. P5-scattering moments have also been evaluated by making use of the Klein-Nishina type of recipes.

The multigroup cross-sections include total, coherent scattering, incoherent scattering, pair production and photoelectric cross-sections and are available on request.

ISOMERIC CROSS_SECTIONS OF INDIUM

M. AFZAL ANSARI, R.K.Y. SINGH, M.L. SEHGAL, V.K. MITTAL
D.K. AVASTHI and I.M. GOVIL

Physics Department
Aligarh Muslim University
Aligarh 202 001

Neutron capture cross-sections are measured in the energy region of 1 to 3 MeV for the isomeric states of In-116. The compatative gamma activation technique has been used. The present results are compared with the previous data whereever available.

PROTON AND ALPHA-PARTICLE INDUCED REACTION CROSS-SECTIONS OF CARBON, COBALT AND LEAD*, S.B. Garg and Amar Sinha, Neutron Physics Division, Bhabha Atomic Research Centre, Trombay, Bombay 400 085

This optical model based study was suggested by the NEA Data Bank, France under the International Nuclear Model Code Comparison Project to determine the accuracy of transmission coefficients obtained with the spherical optical model potential parameters of charged particles.

Reaction cross-sections, Differential cross-sections, polarizations, and spin rotation data have been estimated for the proton and alpha particle induced reaction cross-sections of carbon, cobalt and lead at the incident energies of 5,10,15 and 20 MeV in the laboratory system.

Becchetti and Greenlees /1/ optical model parameters have been used for the protons and McFadden and Satcheler /2/ parameters have been used for the alpha-particles.

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BARC35-A35 GROUP CROSS-SECTION LIBRARY WITH P3-ANISOTROPIC SCATTERING MATRICES AND RESONANCE SELF-SHIELDING FACTORS*, S.B. Garg and Amar Sinha, Neutron Physics Division, Bhabha Atomic Research Centre, Trombay, Bombay 400 085

Multigroup cross-sections are the basic input constants to study the neutronic behaviour of reactor assemblies. These basic-constants should account for the effects of different temperatures and material compositions in the various assembly regions and should represent well the anisotropic behaviour of the scattering phenomena. To meet these requirements a 35 group cross-section set, called BARC35, has been derived from the basic ENDF/B-IV cross-section library for 57 elements. P3-anisotropic scattering matrices and resonance self-shielding factors have also been evaluated and included in this library. Resonance self-shielding factors are given at 300°K, 900°K and 2100°K for a variety of dilution constants. At any intermediate temperature and dilution the self-shielding factors are evaluated by appropriate interpolation schemes.

This cross-section library has produced close agreements in the measured and calcuated multiplication constants of a few selected fast critical assemblies representing large oxide and carbide fuelled uranium and plutonium based systems. The reason for selecting large assemblies is that the resonance self-shielding factors play a dominant role in determining their physics characteristics.

The selected assemblies include ZPPR2, ZPR-3-48, ZPR-3-53, ZPR-6-6A, ZPR-6-7, ZPR-9-31 and ZEBRA2 and are amongst those recommended by the US Nuclear Data Evaluation Working group for testing the accuracy of cross-sections.

BARC35 library is well suited for the safety and neutronics studies of thermal, fast or fusion-fission hybrid systems.

The library is available on request.

* Issued as a BARC Report BARC-1222 (1984)

Generation of Doppler broadened Cross Sections for ^{239}Pu, ^{238}U, ^{235}U, ^{240}Pu, ^{241}Pu, ^{242}Pu, Fe, Cr, Ni, Mn, Mo, Na for Higher Temperature for use in Accident Analysis

(M.M. Ramanadhan, V. Gopalakrishnan and S. Ganesan)

Doppler broadened cross sections to temperatures varying from 300 K to 4500 K were generated for various fissile, fertile and structural materials using the pre-processing code LINEAR-RECENT and SIGMA1[1]. Self shielded cross sections were obtained by using the program REX2[2].

Summarised below are the steps adapted for obtaining the cross sections for various elements.

1. Using code 'LINEAR', the evaluated cross sections of the particular elements from the ENDF/B-IV file, were linearised to high precision (0.1%).
2. The above linearised data was reconstructed to a reasonable precision by using code 'RECENT'.
3. For materials like ^{238}U and ^{239}Pu which had large data points, the output obtained from the code RECENT was further thinned using LINEAR code to an acceptable precision before going to step 4. For the other materials step 4 was directly followed.
4. Using the code SIGMA1 the above cross sections were Doppler broadened to 300 K with 0.0 thinning criteria.

For broadening to temperatures higher than 300 K, one thinned output for 300 K was taken as input and cross sections were obtained at a number of temperatures. Since the running time of code SIGMA1 depends on the number of points input to it this method of going to higher temperatures was adapted. The preprocessing experience in the case of U-238 is detailed by a flow chart in Fig.1.

The doppler broadened outputs were input to the program REX2 for obtaining the self shielded cross sections.

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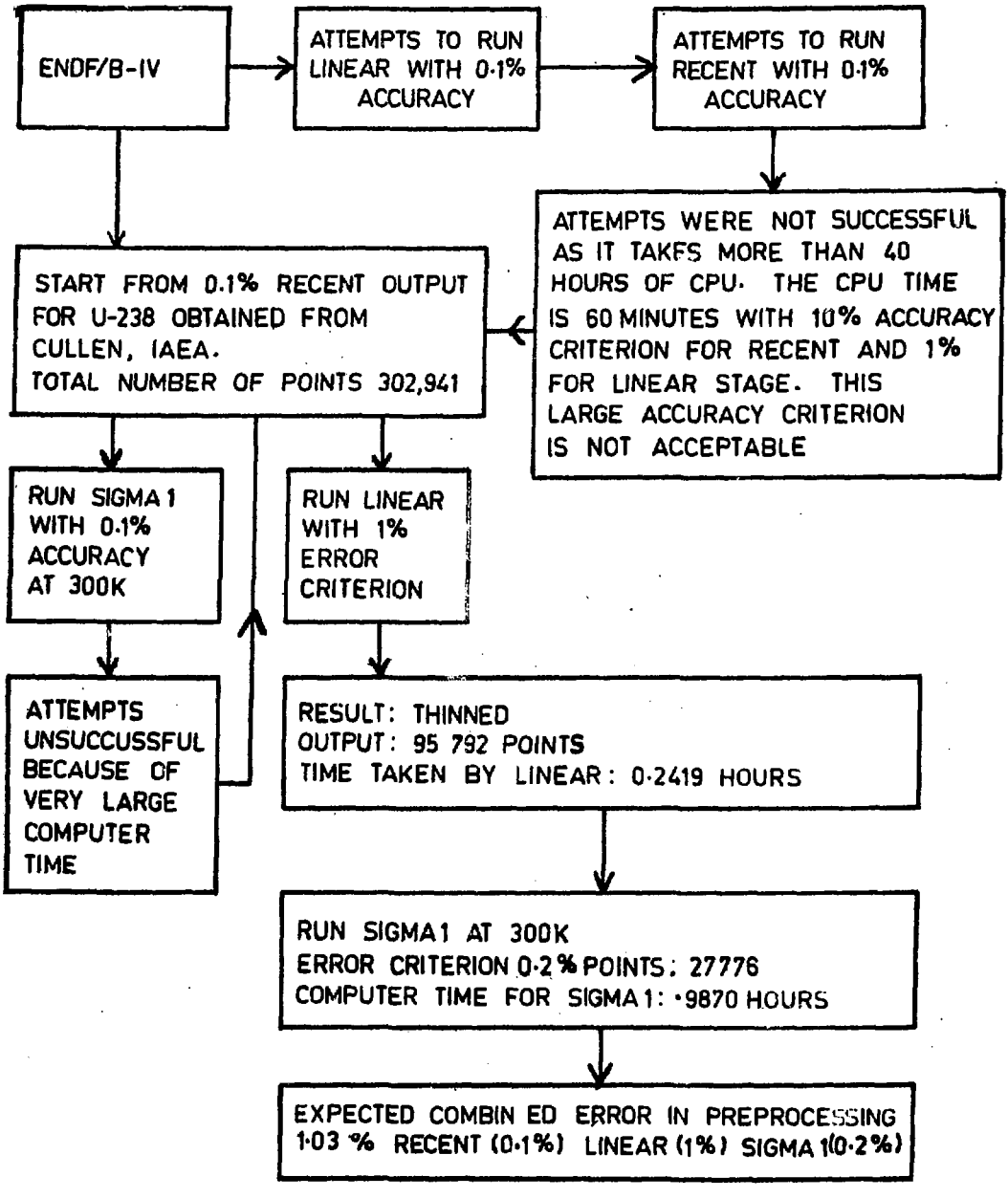

Fig.1 **Flow Chart describing the Preprocessing Experience for the Nuclide U-238**

Generation of Activation Cross Sections in the DLC 100 Group Structure for Structural Isotopes

(V.Gopalakrishnan, M.M.Ramanadhan, S.Ganesan)

Activation cross sections were required in the DLC 100 energy group structure for the computation of activation rates in stainless steel of Intermediate heat exchangers in fast reactor system. The program 'REX1'[1] was used to obtain these cross sections using the data available with us in the ENDF/B-IV format. The isotopes and the basic data library from which they were retrieved are as follows:

Mn^{55} and Co^{59} - ENDL 78 of LLL[2]

Fe^{54}, Fe^{56}, Fe^{58}, Cr^{50}, Ni^{58}, Ni^{60}, Ni^{64} } JENDL1[3] of Japan

In addition to the activation cross sections viz. $\sigma(n,p)$ and $\sigma(n,\alpha)$, 100 group cross sections have also been generated for other neutron nuclear reactions for which basic data are available with us. These group cross sections have been saved on magnetic tape and can be made available to interested users on request.

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Role of Pre-equilibrium Emission on (n,xn) Cross-Sections

R.P. Anand, M.L. Jhingan, S.K. Gupta and
M.K. Mehta

Nuclear Physics Division

Bhabha Atomic Research Centre, Bombay 400 085.

Introduction

The knowledge of (n,2n) and (n,3n) cross sections is quite essential in the reactor technology. Recently hybrid fusion-fission reactors have gained considerable importance. The energy of the neutrons from the fusion reaction D-T is about 14 MeV, above the threshold of (n,2n) and (n,3n) reactions in most of the reactor materials. These cross sections are needed in shielding and breeding calculat ions also. Many of the nuclides produced in the reactor have short half lives and it is not possible to measure their cross sections directly. Also it is interesting to see the role played by the preequilibrium emission in the case of (n,xn) reactions. Here a simple method to calculate (n,xn) cross sections has been developed. In our earlier calculations[1,2] the preequilibrium mode of decay was not taken explicitly but all nonequilibrium effects were taken care by an empirical factor obtained by Kondaiah [3] by the analysis of a large number of (n,2n) cross sections measured at 14 MeV. This empirical factor is valid only around 14 MeV and does not hold good at higher energy. As a result in our earlier calculations upto about 16 MeV there was agreement with the measured cross sections but at higher energy systematically calculated (n,2n) cross sections were lower than measured ones and the reverse was true in case of (n,3n). In the present work the preequilibrium mode of decay along with the equilibrium mode is considered for the first particle emission. Subsequent emissions are considered to be due to the equilibrium mode only. In the preequilibrium decay both proton and neutron channels are considered while in the equilibrium decay proton channel is ignored as in this case proton is likely to have low energy insufficient to cross the Coulomb barrier. Gamma deexcitation competes with neutron emission near the threshold as the neutron having low energy is angular momentum forbidden and further the level density is low at that energy. This effect is indirectly compensated by using the level density parameters of Pearlstein[4] which are lower by about a factor of 2.7 as compared to those of Gilbert and Cameron[5].

Method of Calculation

Cross sections for (n,xn) reactions are calculated on the following assumptions:

(1) Neutrons are emitted in a statistical manner from the composite nucleus formed after the capture of incident neutron.

(2) The emission of the first particle is considered to be due to both preequilibrium and equilibrium processes while subsequent emissions are due to the equilibrium process only.

(3) In the case of the preequilibrium emission both proton and neutron channels are considered while in the case of the equilibrium emission the proton channel is neglected.

(4) The competition due to the gamma deexcitation is taken care indirectly by using level density parameters given by Pearlstein[4] which are less by a factor of 2.7 as compared to those given by Gilbert and Cameron[5].

(5) Equilibrium emission is calculated according to the Weisskopf model neglecting angular momentum.

When a target nucleus of mass A captures a neutron of energy E_n a composite nucleus of mass A+1 is formed at an excitation energy E_n above the ground state of nucleus A. The composite nucleus deexcites by an emission of neutron of energy ϵ_1 and the residual nucleus is left with an excitation energy $(E_n-\epsilon_1)$. If this energy is more than the binding energy of one neutron in the nucleus A, a second neutron is emitted with energy otherwise further neutron emission cannot take place and the event results in (n,n'). Similarly after the emission of a second neutron if the residual nucleus possess sufficient energy to emit a third neutron the same is emitted. Otherwise the event results in (n,2n). In this work maximum four neutron emission is considered. The first neutron emission consists of two components, preequilibrium and equilibrium and may be written as follows as used by Chatterjee and Gupta[6]

$$\frac{df(\epsilon_1)}{d\epsilon_1} = \left\{ \frac{df_{PE}(\epsilon_1)}{d\epsilon_1} + (1-\delta)\frac{df_{EQ}}{d\epsilon_1} \right\} \qquad (1)$$

where $f_{PE}(\epsilon_1)$ is the fraction of neutrons emitted with energy ϵ_1 due to the preequilibrium process and $f_{EQ}(\epsilon_1)$ is a similar term due to the equilibrium process. δ is the sum of neutron and proton fractions emitted due to the preequilibrium process.

Preequilibrium Component

The emission of particles of type ν, of kinetic energy ϵ_ν from an n-exciton state at an excitation energy E is proportional to the emission rate $w_\nu(n,\epsilon_\nu,E)$ multiplied by the time $\tau(n,E)$ the system spends in a particular state and accordingly $df_{PE}/d\epsilon_\nu$ is given by

$$\frac{df_{PE}}{d\epsilon_\nu} = \sum_{\substack{n=3 \\ \Delta n=2}}^{\bar{n}} w_\nu(n,E,\epsilon_\nu)\cdot\tau(n,E)\cdot D_n \qquad (2)$$

The emission rate $w(n,E,\epsilon_\nu)$ of particle ν from an n-exciton state composed of p-particles and h-holes $(n=p+h)$ is given by

$$w_\nu(n,E,\epsilon_\nu) = \frac{2s_\nu+1}{\pi^2\hbar^3}\mu_\nu\epsilon_\nu\sigma(\text{inv})\cdot R_\nu\cdot\frac{1}{gE}\left(\frac{U}{E}\right)^{n-2}\cdot\frac{(n+1)(n-1)}{2} \qquad (3)$$

R_ν is a factor which arises due to proton neutron distinguishability. g is the single particle level density and is taken to be $A/13$. μ_ν is the reduced mass of ν-particle and s_ν is its spin. τ_ν is the life time of n-exciton state and is given by

$$\tau(n,E) = \frac{1}{\lambda_n^+ + \gamma_n} \qquad (4)$$

where λ_n^+ is the transition rate to the next higher exciton state and is given by

$$\lambda_n^+ = \frac{g^3 E^3 \overline{|M|^2}}{2(n+1)} \cdot \frac{2\pi}{\hbar} \qquad (5)$$

$\overline{|M|^2}$ is the average squared matrix element and is taken to be dependent on average energy per exciton as given by Kalbach[7]. γ_n in equation (4) is given as follows,

$$\gamma_n = \sum_{\nu=n,p} \int_0^{\epsilon_{max}} w_\nu(n,E,\epsilon_\nu)\, d\epsilon_\nu$$

Dn in equation (2) is the term responsible for depletion and is given by

$$D_{n+2} = D_n (1 - \gamma_n \tau_n) \quad \text{and} \quad D_3 = 1 \qquad (7)$$

δ in equation (1) is related with Dn as follows

$$1 - \delta = D_{n+2} \quad \text{or} \quad \delta = 1 - D_{n+2} \qquad (8)$$

Inverse reaction cross sections have been obtained by the parametrisation developed by Chatterjee et al[8] to get Wilmore-Hodgaon[9] optical model reaction cross sections. At energy lower than 2 MeV their parametrisation is not accurate. Hence at these energies an expression given by Blatt and Weisskopf[10] is usedby matching the cross section at 2 MeV.

Equilibrium Component

The equilibrium component is calculated according to the Weisskopf model neglecting angular momentum. The probability of neutron emission with energy ϵ is given as follows.

$$P \sim \epsilon . \sigma . \rho(U)$$

σ is the inverse reaction cross section and is calculated as explained in section III. $\rho(U)$ is the level density at the excitation energy U and is given by

$$\rho(U) = C e^{2\sqrt{aU}}$$

C is taken to be constant, any slow variation in C is neglected. a

is level density parameter given by Pearlstein. Accordingly, the expression for (n,2n) cross-section is given as follows,

$$\sigma(n,2n) = \sigma_M \int_0^{E_n - EB1} \left\{ \frac{d f_{PE}}{d\epsilon_1} + (1-\delta) \frac{\epsilon_1 \sigma_1 P_1}{\int_0^{E_n} \epsilon_1 \sigma_1 P_1} \right\} \cdot P_2 \cdot d\epsilon_1 \qquad (11)$$

σ_M is taken to be σ_R for non-fissionable nuclei and for fissionable nuclei it is taken as $\sigma_R - \sigma_f$. σ_f is the fission cross section and is taken from the measured data if available otherwise it may be estimated from the empirical formula given by Jhingan et al[11]. P_2 is the probability that after the emission of the first neutron the second neutron is emitted with such an energy that the third neutron cannot be emitted and is given as follows

$$P_2 = \int_{E_n - \epsilon_1 - EB2}^{E_n - \epsilon_1 - EB1} \epsilon_2 \cdot \sigma_2 \cdot \rho_2 \, d\epsilon_2 \Big/ \int_0^{E_n - \epsilon_1 - EB1} \epsilon_2 \cdot \sigma_2 \cdot \rho_2 \, d\epsilon_2 \qquad (12)$$

EB1 and EB2 are the binding energy of one and two neutrons in the nucleus A respectively. If $E_n - \epsilon_1 - EB_2$ happens to be negative it is taken to be zero. In the numerator limits of integration ensure that second neutron is emitted with an energy so that third neutron cannot be emitted. In a similar way expressions for (n,3n) and (n,4n) are written.

In this work a maximum of four neutron emissions have been considered. Accordingly a computer code has been developed. All the integrals have been evaluated using Simpson's rule. Cross sections have been calculated and compared for a number of nuclei in the mass region 89 to 238 at incident neutron energies upto 28 MeV. However cross sections for ^{238}U have been calculated upto 20 MeV only as the fission cross sections have been taken from ENDF-B-IV where data are given upto 20 MeV only. In some typical cases the comparisons between calculated and measured cross sections are shown in figure 1 to 4.

Conclusion

The agreement in case of (n,2n) and (n,3n) cross section is within 10%. Data on (n,4n) are scanty and wherever measured data are available they have been compared with the calculated ones and the agreement is within a factor of two. The preequilibrium component is found to be considerable at energies higher than about 16 MeV and its inclusion is essential. Average squared matrix element given by Kalbach[7] can be used without any adjustment. It is not very sensitive to (n,xn) cross sections. In the present work there is no fitting parameter. All the parameters have been teken from the literature. The level density parameters given by Pearlstein give good agreement in the case of (n,2n) and (n,3n) cross sections. It appears to compensate the effect of neglecting gamma deexcitation and makes the calculation easier. The (n,xn) cross section for unstable nuclei can be predicted by this method where direct measurement is not possible.

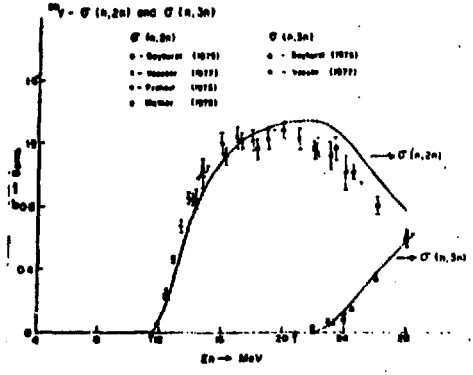

Fig.1

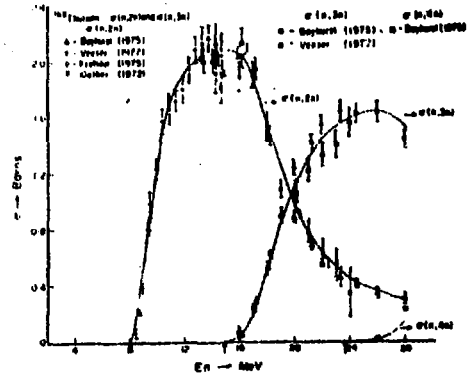

Fig.2

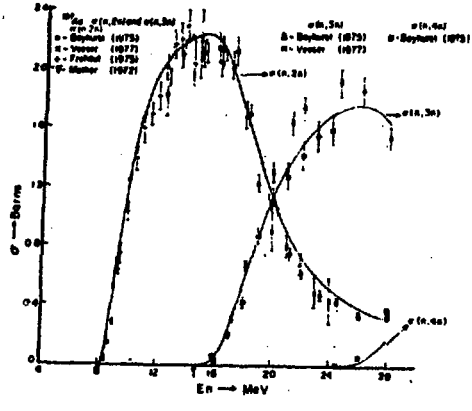

Fig.3

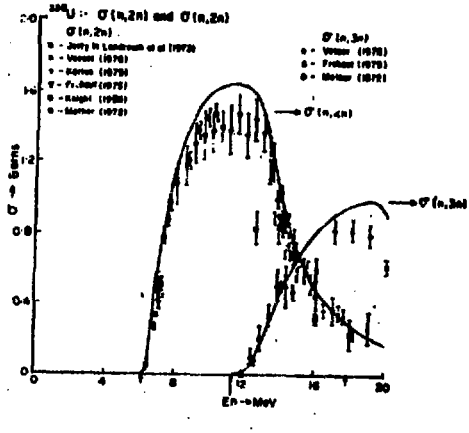

Fig.4

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III. OTHER NUCLEAR DATA ACTIVITY

Generation of New Multigroup Cross Section Set for Various Materials for Fast Reactor Applications

(M.M.Ramanadhan, V.Gopalakrishnan, S.Ganesan)

Using the Nuclear Data Processing Code system RAMBHA developed at RRC, Kalpakkam, multigroup cross sections were generated from the basic data library ENDF/B-IV for various elements. The isotopes covered thus far include the following:

^{235}U, ^{238}U, ^{239}Pu, ^{240}Pu, ^{241}Pu, ^{242}Pu, Fe
Cr, Ni, Na, O, C, Al, Si, Mo, Mn and Ga

The multigroup cross section data of these elements are considered 'improved' compared to our earlier set[1] due to the following improvements in the processing code system that is used for multigrouping:

1. Abandoning the resonance integral method, the infinite dilution cross section computed using the code system LINEAR-RECENT-REX1[2] which is the most suited for ENDF/B type of basic data representation.

2. Abandoning the J* method of Hwang, the shielded cross sections are obtained using the code system LINEAR-RECENT-REX2[3] in the resolved resonance region.

3. The LINEAR-RECENT-REX1[2] code system also improves the elastic removal cross section in the resonance region.

These new infinite dilution group cross sections were compared in detail (groupwise, reactionwise and materialwise) with the 1969 adjusted French multigroup cross section set using the program COMPLOT[4] and this comparison not only brought out the differences due to improvement in our own processing code system but also the significant changes in the basic cross section data itself, thanks to improved differential data base.

The groupwise and reactionwise comparisons of these

two multigroup infinite dilution cross section sets are available with the authors for each of the isotopes mentioned above. Interestingly this comparison brings out the changes in the cross sections of some isotopes reflecting the refinement in the ENDF/B-IV data after nearly a decade of improvement in the techniques of evaluation and measurement in constrast to the old differential data base used in the generation of 1969 French set. For example, for transactinium isotopes such as ^{238}U, ^{235}U, ^{239}Pu, ^{240}Pu, ^{241}Pu etc. in the new set, the total inelastic scattering cross sections are increased considerably and in some cases even by a factor of 3 or more around the inelastic threshold. Also for fertile isotopes such as ^{238}U and ^{240}Pu, the effect of the newly discovered sub threshold fission and its intermediate structure is seen in the RRC set in the form of non zero fission cross sections in lower groups. The absorption cross sections for ^{12}C and other medium mass elements such as Fe, Cr and Ni are found to be very different by several factors in the new set because of improved helium and hydrogen production cross sections in the 1974 ENDF/B-IV data base.

Part of these comparison study was presented in a recent conference[5].

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Multigroup Constants for Gallium for use in Analysing Fast Critical Assemblies

(S.Ganesan and V.Gopalakrishnan)

In many multigroup data sets like SETR for instance the multigroup data are not available for Gallium. The basic data library ENDF/B-IV available with us does not include Gallium data. In such a situation, the usual practice had been to replace Gallium with Copper as an approximation for neutronic calculations for fast critical assemblies. This approach might be roughly valid in cases where the concentration of Gallium is not so significant. For instance, in SNEAK-9C2 assemblies POZ and C, the nuclides concentration was only 0.00014104 nuclides/b-cm which was added[1] to that of Cu in analysis. In critical assemblies like THOR[1], the element Ga is found to be present in significant quantity. (Pure plutonium metal has the unique characteristic of existing in six allotropic forms between room temperature and its melting point at 913K. The delta phase of Pu which occurs in the temperature range of 588 and 730K can be stabilised by the addition of a nominal amount of Ga to the Pu metal. The alloying increases the delta phase stability range from 730K well bebw room temperature). In THOR, the nuclide concentration is 0.00133 nuclides/b-cm in the core, and a replacement of the same with a different element in neutronic calculations will lead to uncertain intrepretation of calculated results. Therefore, the necessity of obtaining the multigroup data Ga was felt. The basic library ENDL-78[3] which was identified to be the only library available with us containing basic data for Gallium was processed by the code RAMBHA[4]. The multigroup constants thus generated in 25 group structure are being used in the analysis of THOR[5] assembly.

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The Program 'REX1' for Accurate Generation of Multigroup Infinite Dilution Cross Sections, Elastic and Inelastic Transfer Matrices and the Program 'REX2' for Accurate Calculation of Self Shielding Factors in the Resolved Resonance Region for Fast Reactor Applications

(V.Gopalakrishnan and S.Ganesan)

The computer programs REX1[1] and REX2[2] are two parts of RAMBHA[3], a nuclear data processing code system for fast reactor applications, developed at RRC to obtain multigroup cross section data using the basic library in the ENDF/B-IV[4] format.

REX1: This program is intended for multigrouping point data available in the ENDF/B-IV format. The program expects the cross sections to be directly specified as point data without parameterization through resonance formalisms. If the actual cross sections are represented by a combination of resonance parameters and floor corrections in some energy region and as point data in other energy regions, then, one has to use the preprocessing codes LINEAR[5]-RECENT[6] before using REX1 for multigrouping. In addition to group cross sections, the program calculates the multigroup transfer matrices for elastic scattering by Bucholz' method[7] and the multigroup transfer matrices for inelastic scattering by the method followed in MIGROS-3[8].

The program follows any interpolation scheme specified according to ENDF/B-IV conventions and uses Gauss' quadrature whenever numerical integrations are needed. Standard fast reactor flux spectra expressed in a functional form in various energy regions are utilised for weighting.

REX2: The resolved resonance has been extended considerably for many isotopes of interest in reactor design by means of high resolution cross section measurements. The improved quality of data in the resonance region has called for faithful multi-

grouping such that the multigroup constants exactly represent the evaluated data. Thus improvements in processing has resulted in the routines for direct numerical Doppler Broadening of cross sections in resonance region without using the J* formalism.

The program REX2 calculates the self shielded cross sections and the self shielding factors for a set of input dilutions from the exactly Doppler broadened point cross sections made available at a specified temperature by running the preprocessing code SIGMA1. If the cross section data comes from the original ENDF library at OK, then, SIGMA1[9] requires the data to be preprocessed by the LINEAR[5]-RECENT[6] code system.

The program REX2 replaces the older routines of RAMBHA based on approximate J* method of Hwang in the resolved resonance region.

The temperature (T) and dilution(σ_p) dependent cross section known as the shielded cross section of any material for a group g is computed using Bondarenko definition as

$$\sigma_{x,g}(T,\sigma_p) = \frac{\displaystyle\int_{E_{g+1}}^{E_g} \frac{\sigma_x(E,T)\,S(E)}{(\sigma_t(E,T)+\sigma_p)^N}\,dE}{\displaystyle\int_{E_{g+1}}^{E_g} \frac{S(E)}{(\sigma_t(E,T)+\sigma_p)^N}\,dE}$$

where

$\sigma_x(E,T)$ is the cross section at energy point E and temperature T for a process x; x is t, C, f or s.

S(E) is the slowly varying weighting function

and N= { 1 for flux weighting used for x other than t
2 for current weighting used when x is t.

The self shielding factor is defined as

$$f_{x,g}(T, \sigma_p) = \frac{\sigma_{x,g}(T, \sigma_p)}{\sigma_{x,g}(T, \infty)}$$

where $\sigma_{x,g}(T, \alpha)$ is known as the infinite dilution cross section. In REX2 $\sigma_{x,g}(T, \infty)$ is taken as $\sigma_{x,g}(T, 10^{20})$

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Cross section Sensitivity Studies for Fast Neutron Transmission of Sodium

(P.T.Krishnakumar)

All the present designs of Liquid Metal Fast Breeder Reactors (LMFBR) involve thick layers of sodium of about five metres above the core. The manner in which fast neutrons penetrate particularly from a fission source is of considerable practical interest from the point of view of shielding. To assist in the design of experiments intended to check the accuracy of sodium cross sections required for LMFBR shielding calculations, cross section sensitivity profiles have been generated for geometry and detector combination approximating experimental set up.

A relative cross section sensitivity function is defined as the relative change in reactor performance parameter R due to relative change in cross section Σ i.e.

$$\text{Relative sensitivity function} = \frac{\Delta R/R}{\Delta\Sigma/\Sigma}$$

A plot of relative sensitivity function versus neutron energy is known as sensitivity profile. The linear perturbation code SWANLAKE[1] has been used for the generation of sensitivity profiles for the total flux for a plane isotropic fission source incident on one side of a slab of sodium, five metres thick and a neutron detector at the other.

The striking features in the sensitivity profile reflect the corresponding behaviour of total cross section of Na. Little transport can occur in the high cross section region at 3 KeV, 52 KeV and between 700 KeV and 1 MeV, so that sensitivity plot has a deep minima in these energy regions. Conversely the prominent spikes in the sensitivity plot corresponds to four regions of minimum cross sections (0.297 MeV, 0.522 MeV, 1.885 MeV and 3.075 MeV). The results[2] of the above sensitivity study clearly indicates the importance of high

energy region particularly 300 KeV and 500 KeV for the neutron penetration.

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A Critical Assessment of Multigroup Nuclear Data for Structural Elements

(S.Ganesan, M.M.Ramanadhan, V.Gopalakrishnan and R.S. Keshavamurthy)

Our calculated values of central worths of structural materials using the non-adjusted, ENDF/B-IV based, 1983 RRC set and those using the 1969 adjusted French multigroup set showed discrepancies ranging from + 20 to + 60% in comparison with experimental values in our[1,2] analyses of selected fast critical assemblies[3] viz. ZPR-6-7, ZPR-9-31, ZPR-3-48 and ZPR-3-56B. The K_{eff} of the last assembly which has nickel reflector showed an overprediction of 1.63% when calculated using the French set. One clear conclusion of the present investigation is that the capture cross sections for Fe, Cr and Ni in French set at RRC are definitely outdated and incorrect. The modified French set in which the multigroup sets for Fe, Cr and Ni alone were replaced in full by RRC set gives a Keff close to unity.

Further, the currently reported[4] discrepancies in resonance parameters and measured cross sections in structural elements are also being surveyed to identify problem areas which need further attention.

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COMPLOT : A Program for Comparison of the Multigroup Constants

(M.M.Ramanadhan)

Program COMPLOT[1] compares the values of the multigroup constants obtained from the nuclear data processing code RAMBHA[2] with those obtained from the 1969 adjusted French cross section set[3]. The program has also been utilized for comparing multigroup data sets derived from the same basic file using different processing approximations. The output can be in a tabular form or as a graphic plot depending on the option exercised in the input, for various isotopes and for different reactions. Option also exists for listing the various multigroup constants for any particular isotope in SETR format. With suitable modifications the code can easily be adapted for comparison of the multigroup constants from various evaluated data libraries.

This program plays a major role in checking, plotting correcting and processing of SETR formatted multigroup libraries. Using this program many of the physics aspects of cross sections of any particular isotope can be verified by graphical plots and comparison tables. This ensures automatic checking of the enormous amount of numerical data that go into the various multigroup sets for various energy ranges, reactions and nuclides. The importance of these comparisons-study during the preparation of a new multigroup cross section set for a given isotope is quite enormous for it quickly identifies any large deviations between the new and old cross sections and also to the identifications of problem areas forsensitivity studies in subsequent analysis of critical experiments to validate the new multigroup cross section data. The specifications of this COMPLOT program have been documented in Ref.1.

An interesting feature of this program is that the number of inputs to this program is confined to specifying of the number of isotopes to be compared, their identification numbers, the old and new cross section sets and options to obtain print outs or graphical plots.

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Analysis of ZPR-9 Assembly 31, the Advanced Fuels Program Carbide Benchmark Critical Assembly

(S.Ganesan, M.M.Ramanadhan and V.Gopalakrishnan)

The analysis of the 1000litre (core) sized carbide benchmark fast reactor assembly[1] using the RRC non adjusted multigroup set and the adjusted French 1969 set[2] shows nc surprising results as compared to our predictional capability for oxide cores. The results for central worths, compare very well with those reported by U.S. team using ENDF/B-IV Set and also show that non adjusted set maynot be satisfactory in the prediction of central worths of materials. On the other hand K_{eff} is best predicted by the non adjusted ENDF/B-IV based RRC set and the ratio of fission reaction rates are predicted to the same extent by the two sets available at RRC.

Based on our present analysis the following conclusions and recommendations are drawn:-

1. Our attempts of writing our own nuclear data processing code system RAMBHA and the generation of non adjusted ENDF/B-IV based RRC multigroup datausing RAMBHA and its validation have been quite successful.

   The K_{eff} for the carbide benchmark critical assembly is well predicted by the non adjusted ENDF/B-IV based RRC multigroup set.

2. The over predictions of central worths by non adjusted RRC and US sets are more, compared to the over prediction by the adjusted French set, for most of the elements.

   Special attention should be given to reducing the over prediction of central worths of various elements in future with stress on improving the status for structural elements.

3. The non adjusted ENDF/B-IV based RRC set is superior, as

compared to the 1969 non adjusted French set, for isotopes such as ^{241}Pu for which it is recognised that recent (1969-78) evaluations/measurements of cross sections have brought in many improvements over the earlier evaluations.

4. The predictional performance of RRC non adjusted set is inbetween that of adjusted French set and the US non adjusted set for the reaction rate ratios in ZPR-9-31 assembly. The details are given in Ref.3.

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A Programme of Evaluation, Processing and Testing of Nuclear Data for Th-232

(S.Ganesan, M.M.Ramanadhan, V.Gopalakrishnan and R.S.Keshavamurthy)

Within the framework of a RRC-IAEA research contract[1] on validation and benchmark testing of actinide nuclear data, work has been initiated at Kalpakkam on a programme of evaluation and validation of nuclear data for Th-232 and U-233 for fast reactor applications.

Generations of four different new multigroup cross section sets in 25 energy group structure using the following basic data files have been completed for Th-232.

a. ENDF/B-IV
b. JENDL-1 (Ref.2)
c. Rumanian file (INDL/A-83) (Ref.3)
d. JENDL-INDIAN file presently created at Kalpakkam

The first three files a, b, c were obtained from IAEA Nuclear Data Section. The processings of all these data files are being done by processing code system RAMBHA[4] developed at Kalpakkam in the lastfew years.

The analysis of THOR assembly discussed in Ref.5 clearly favours σ_c values lower than French and ENDF/B-IV based set. This trend favours the Indian evaluation for capture for Th-232 by Mehta and Jain[6].

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Participation in IAEA Code Verification Project

(S.Ganesan and M.M.Ramanadhan)

On a suggestion from IAEA Nuclear Data Section, we started participating in IAEA Code Verification Project[1]. This project attempts to assure that for a given input or evaluated data and physical assumptions the output of cross section processor is accurate. This project does not deal with errors in the evaluated data or the modelling accuracies of the transport or diffusion approximations of the physical problem. The objectives are therefore limited to the following:

1. To examine the behaviour and test the accuracy of processing programs.
2. To eliminate, through identification of weak points, the error producing options and sources (of discrepancies) in the processor.

Once these two objectives are satisfi , it is possible to have the generation of multigroup constants ithout introducing significant errors in processing, within reasonable computing time using correct models for processing.

Using the latest so called[2] 'mod. 1 Library' (referred to at Vienna as EN5-D2-p ENDF/B-V dosimetry library) we generated flat weighted zero kelvin unshielded cross sections using SAND II 620 group structure. The codes LINEAR-RECENT-GROUPIE[3] were utilised.

A comparison of our results were made at IAEA, with the benchmark standard results. Differences were noticed for very small values close to threshold and the reason was traced to a procedure built into LINEAR[3] to handle the case of positive and negative background cross sections in the resonance region which should not have been applied near thresholds. The versions of LINEAR used does not further subdivide the interval if the

cross section at either end of an energy interval is less than a user specified value. This design problem in LINEAR was overcome in a current version of LINEAR.

The participation increased our confidence in the sucessful adaptation and use of preprocessing codes supplied by IAEA, in our DPS-8 system.

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Generation of Broad Group Errors and their Correlation Matrix for Total Reaction Cross Section of U-238

(S.Ganesan and V.Gopalakrishnan)

A recent Argonne National Laboratory evaluation has provided the fine group values of errors and their correlation matrices for U-238 from 44 keV to 20.0 MeV energy region. This fine group error matrix has a dimension of 55 x 55.

A practical problem in sensitivity analysis is concerned with proper collapsing of the large correlation matrix into a small sized matrix for determination of error associated with calculations of neutronic parameters arising from cross section uncertainties and their correlations.

A derivation made by us, invoking certain assumptions, as detailed in Ref.2 gives the following expression.

$$\delta\hat{\Sigma}_I \, \delta\hat{\Sigma}_J \, \hat{C}_{IJ} = \frac{\sum_{i=m}^{m+Q} \sum_{j=n}^{n+P} C_{ij} \, \delta\Sigma_i \, \delta\Sigma_j \, \phi_i \, \phi_j}{\phi_I \, \phi_J}$$

where $\delta\hat{\Sigma}_I$ and $\hat{C}_{IJ}$ are the broad group values of the errors and their correlation matrix. Q is the number of fine groups in the broad group I and p the number of fine groups in broad group J. It is assumed that the broad group boundary and a fine group boundary coincide. ϕ_I, ϕ_J are broad group fluxes and ϕ_i, ϕ_j represent fine group values. An assumption involved in deriving Eq.(1) is that ϕ_i does not depend on Σ_i. Our derivation appears to correspond to option LB = 0 given by Perey[3] in ENDF/B uncertainty file.

Calculations are in progress to collapse the above 55 x 55 matrix into 10 x 10 matrix. The aim is to obtain familiarity and experience in processing of uncertainty files and get a practical idea about the whole problem.

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