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PROGRESS REPORT ON NUCLEAR DATA ACTIVITIES IN INDIA FOR THE PERIOD JANUARY 1989 TO JUNE 1990

Compiled by

R. P. Anand
Nuclear Physics Division

1990

B.A.R.C. - 1521

GOVERNMENT OF INDIA
ATOMIC ENERGY COMMISSION

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60 Abstract : The progress report covering the period from January 1989 to June 1990 contains brief descriptions of various activities such as measurements, evaluations, compilations and processing of nuclear data and other related works being carried out in India, mainly at Bhabha Atomic Research Centre, Bombay, and at Indira Gandhi Centre for Atomic Research, Kalpakkam.

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PREFACE

The present progress report on Nuclear Data activities in India is the Seventh in the current new series of progress reports, the first of which was brought out in the year 1981. This report covers the work carried on during the period from January, 1989 to June, 1990. It contains brief description on various activities such as measurements, evaluations, compilations and processing of Nuclear Data and other related works being carried out in India, mainly at Bhabha Atomic Research Centre at Bombay and Indira Gandhi Centre of Atomic Research at Kalpakkam.

The work related to basic nuclear physics including nuclear structure studies and heavy ion physics being carried out at Pelletron accelerator at Bombay and Variable energy cyclotron at Calcutta has not been included in this report and for this the reader should refer to the proceedings of the Symposium on Nuclear Physics, Vol 32B (1989), held at Aligarh from December, 26-30, 1989.

Most of the write-ups in this report describe the work in brief and in progress and these are not to be regarded as publications or quoted without permission from the authors.

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

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STUDIES OF FISSION FRAGMENT TEMPERATURES FROM THE MEASUREMENT OF NEUTRON EMISSION SPECTRA IN THERMAL NEUTRON INDUCED FISSION OF 235U

Mahesh Samant, R.P. Anand, Alok Saxena, B.R. Ballal, D.M. Nadkarni, R.K. Choudhury and S.S. Kapoor.

The energy spectra of prompt neutrons emitted in fission contain information on the statistical properties of the excited fission fragments. From the analysis of the neutron spectra one can deduce the level density parameters corresponding to the mass and excitation energy region of the fission fragments /1/. We have taken up studies to measure the neutron energy spectra at $0°$, with respect to the fragment direction in $^{235}U(n_{th},f)$ in order to systematically determine the level density parameter as a function of fragment mass.

The experiment was carried out using the thermal neutron beam at CIRUS reactor. Two surface barrier detectors were mounted on either side of a thin ^{235}U source (100 $\mu g/cm^2$) on 160 $\mu g/cm^2$ Ni backing) to measure the fragment energies. A (2"x2") NE213 liquid scintillator detector was placed at a distance of 66.8 cms from target along the direction of fission detectors to detect the fission neutrons. The n-γ separation was achieved by the pulse shape discrimination method and the neutron energies were obtained from the measurement of the time of flight with respect to the time of fragment detection in the semiconductor detector. The information on the energies of the fission fragments, the time of flight of the neutrons, height and shape of the pulses from thr neutron detector were recorded event by event on PDP 11/23 computer.

A total of 2×10^5 coincident events were recorded and analysed. The time resolution (FWHM) as obtained from the prompt γ-ray peak in the TOF spectrum was seen to be about 2.0 nsec. The energy of the neutrons was derived from the time of flight and the spectra were obtained after correcting for the efficieny of neutron detection as a function of neutron energy. The neutron detection efficiency of the NE213 detector was experimentally determined from an independent experiment using a ^{252}Cf fission sorce in the identical detector geometry. The analysis of the fragment mass and total energies was carried out by knowing the individual fragment energies /2/. An event by event analysis was then carried out to obtain the neutron energy E in the fragment centre of mass system as a function of fragment mass by knowing the fragment mass, energy and the laboratory energy of the neutron. These neutron spectra for various mass groups (of 4 mass units each) are shown in Fig. 1(a) and Fig. 1(b).

These spectra were then fitted to a cascade evaporation spectrum to obtain the average fragment temperatures. Fig.2 shows the fragment temperature distribution as a function of fragment mass. It is seen that the temperature increases with

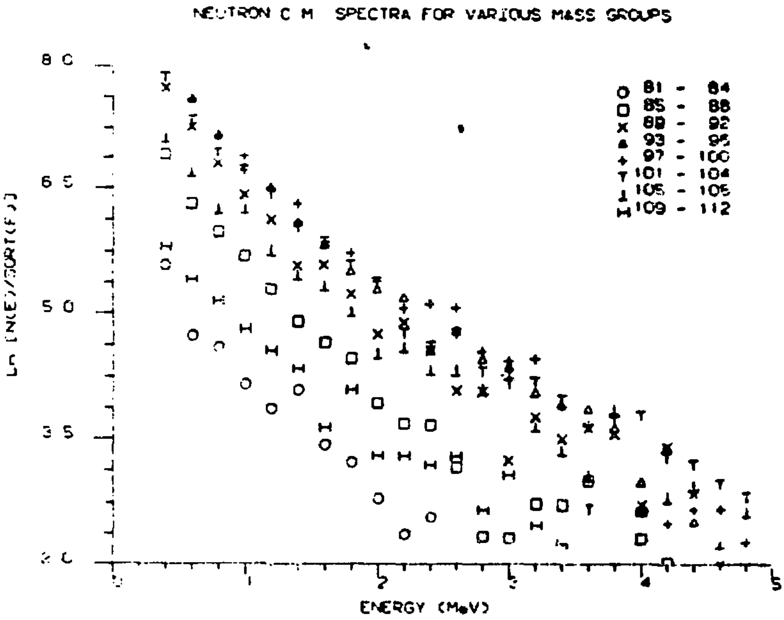
NEUTRON C M SPECTRA FOR VARIOUS MASS GROUPS
81 - 84
85 - 88
89 - 92
93 - 96
97 - 100
101 - 104
105 - 108
109 - 112
ENERGY (MeV)

Fig. 1(a)

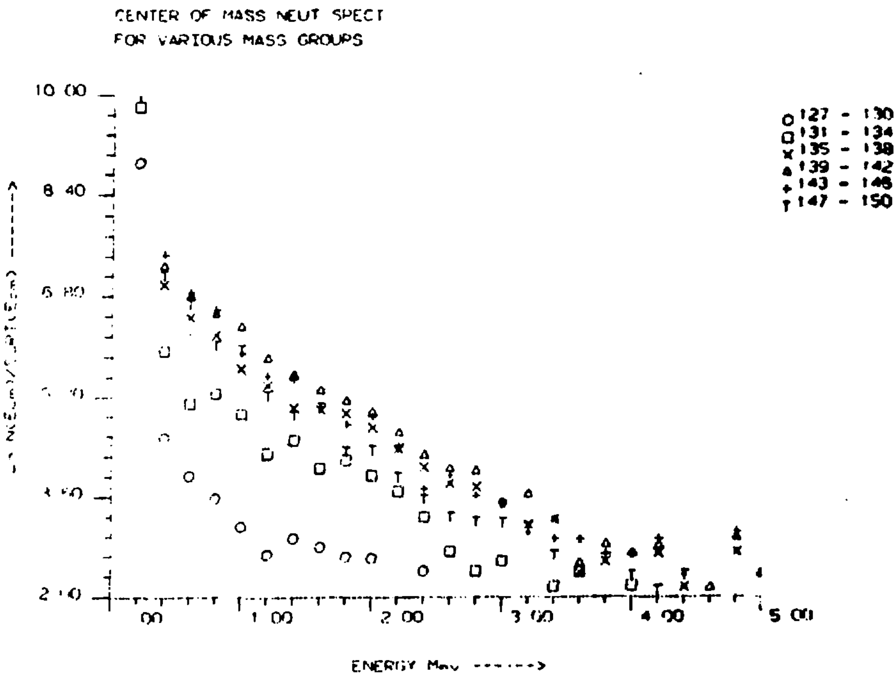
CENTER OF MASS NEUT SPECT
FOR VARIOUS MASS GROUPS
127 - 130
131 - 134
135 - 138
139 - 142
143 - 146
147 - 150
ENERGY Mev ------->

Fig. 1(b)

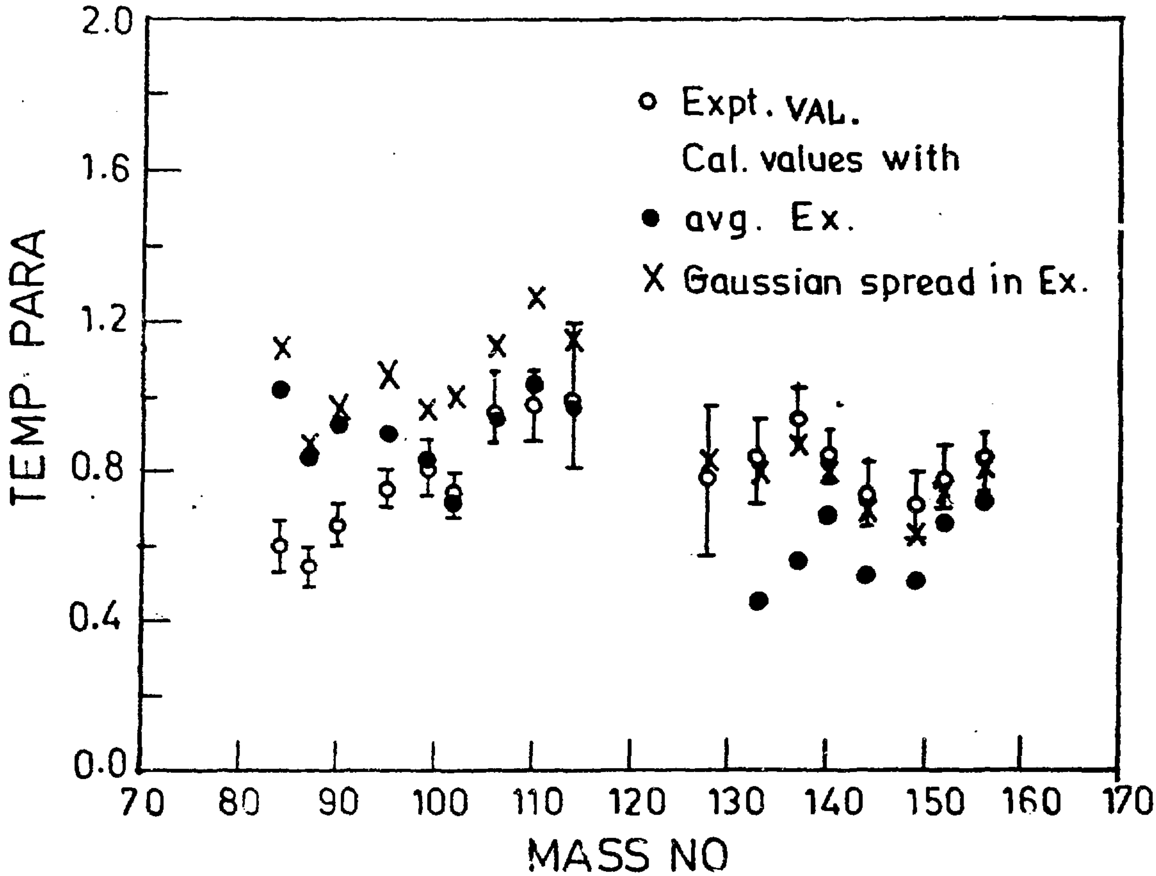

Fig. 2

fragment mass for the light fragment region, where as there is somewhat less variation in the heavy fragment region. These temperatures were then used to get the information on level density parameters of the corresponding fragments by comparing them with the calculated values of temperatures. The calculations were carried out using a statistical model code (ALICE) with a shell dependent level density parameter. The evaporation spectra given by this code at appropriate fragment excitation energies, was then compared with the measured spectra to obtain liquid drop level density parameters. The calculations were carried out (1) at average excitation energies of the fragments and also (2) by taking an appropriate Gaussian spread in the excitation energies and these calculated values of the temperature parameter for the case of a= A/7 are shown in Fig. 2 along with the measured values as a function of fragment mass. It can be seen that the value of a=A/7, taken along with guassian spread in the excitation energies, fits the data quite well for the heavy fragments only. Further analysis of the data is in progress.

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STUDIES OF PRE-SCISSION NEUTRON COMPONENT FROM THE MEASUREMENT OF PROMPT NEUTRON EMISSION SPECTRUM IN $^{235}U(n_{thermal},fission)$

R.P. Anand, M. Samant, B.R.Ballal, R.K. Choudhary, A.Saxena, D.M. Nadkarni and S.S. Kapoor

There is a lot of controversy regarding the component of pre-scission neutrons observed in the prompt neutron emission in the thermal neutron induced fission of ^{235}U. In order to study this effect and other correlations between the emitted neutrons and the fission fragments, we have setup an experiment at CIRUS reactor at B.A.R.C. The energy spectra of prompt neutrons emitted at different angles from various fission fragments contain information on the dynamics of the fission process itself.

The detector used to measure the energy and angle of the fission fragment-pair, is an ionisation chamber which consists of two parallel plate ionisation chambers with Frisch grids and one common cathode and filled with P-10 gas at pressure slightly more than one atmosphere pressure. ^{235}U target (100 ugm/cm^2) is mounted in the centre of the common cathode. The neutron detector consists of a cylindrical (2"X2") NE 213 liquid scintillator, which was kept at a distance of about 67 cms away from the U-target and in the direction of the electric field of the ionisation chamber.The energy of the neutrons was measured using Time of Flight technique. The start signal was taken from the common cathode and the stop signal from the neutron detector. The time resolution of the system, as obtained from the gamma peak of the time of flight spectrum, is found to be less than 2 nsec. Pulse shape discrimination technique is used to distinguish between neutrons and gammas detected by the liquid scintillator.

The following seven parameters are stored in coincidence in a PDP 11/23 computer. 1) Time of flight, 2) Zero Cross-over time for Pulse shape discrimination, 3) & 4) Collector pulse heights for both fragments, 5) & 6) Grid pulse heights for both fragments and 7) Pulse height of the neutron detector. the energies of the fragments are obtained from the collector pulse heights whereas the angle information can be determined from the grid pulses.

The data has being collected and more than one million coincident events have been stored on a magnetic tape. Simultainiously we have also collected singles-events (not in coincidence with neutrons) which are required for the calibration of the ion chamber.The total number of fission events is also being recorded.At present the analysis of the data is under progress and proper calibration constants of the ionisation chamber have been obtained from the singles data and it has been verified that the angular distributions as a function for various mass and energy windows gives an isotropic distribution for the singles data. Further analysis for the coincident data is in progress.

STUDIES OF COLD FRAGMENTATION IN FISSION OF 235U

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The sharing of the available energy in the form of the kinectic and excitation energies of the fragments is one of the long standing problems in the study of the fission process. In rather rare cases, fission events take place in which the G-value of the fission reaction is nearly exhausted by the total kinectic energy of the fragments. Determination of the mass distribution for such cold fission events can yield important information on the potential energy surface (PES) and the dynamics of the fission process. It has been reported /1/ that in fission of ^{234}U, cold fragmentation is obtained for the pair fragments of A_L=100-105 and A_H=129-134 masses, which are both stablized by shell effects. There is very little experimental data on cold fragmentation for other fissioning systems. We report here the measurements on the cold fragmentation mass distributions and the excitation energy distributions of fission fragments in thermal neutron fission of ^{235}U.

The experimental setup consists of a back-to-back gridded ionisation chamber for fragment mass, energy and angle measurements. By the method of electronic collimation of the fission fragments /2/, a mass resolution of better than 1.5 mass units was obtained. An event by event analysis of the excitation energy of the fragments was carried out using the measured total kinetic energies and the Q-values from the mass tables of Liran and Zeldes /3/ . Fig.1 shows the variation of the average excitation energy with fragment mass alongwith the average Q-value used for the calculations. A sharp local minimum around A_L=104 amu is seen indicating the influence of the strong heavy fragment shell (Z_H=50 and N_H=82) in bringing down the average excitation energy of the fragments. For A_L=113-115 amu the excitation energy distribution had two peaks as shown by two symbols in the figure. This result indicates that for these mass splits there are two scission configurations or fission modes.

In fig.2 we show the fragment mass distributions for various excitation energy windows. The mass distribution for the fragmentation events corresponding to the excitation energy window of 0-5 MeV shows that apart from peak corresponding to A_L=104 and A_H=132 amu, there is also higher yield for fragments of A_L=92 and A_H=144 amu. PES calculations with inclusion of shell corrections indicate that there may be a deformed shell for these masses and the present results suggest therefore that nuclei in this region may possess large ground state deformations. The structures in mass distributions disappear with increasing excitation energy but at very high excitation energies they are seen to reappear. These results can be understood if excitation energies correspond to cold but highly deformed nuclei at the scission point. Further detail calculations are required to explain the new features seen in the present results on the cold fragmentation processes in the fission of ^{235}U.

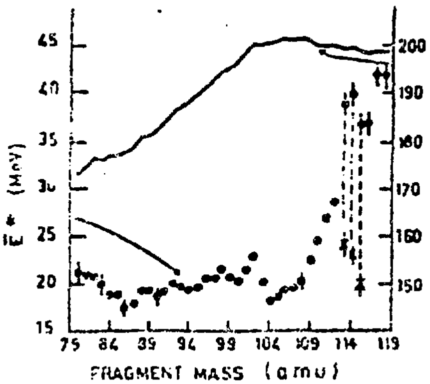

Fig. 1

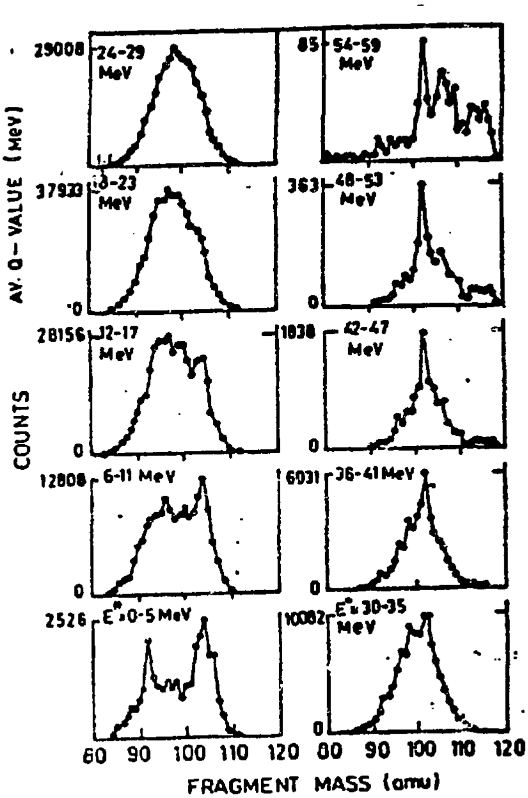

Fig. 2

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FAST NEUTRON RADIATIVE CAPTURE CROSS-SECTION OF FISSION PRODUCT Nd-150

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The knowledge of fast neutron capture cross-sections of the fission products is important in the design of fission reactors/1/. Neodymium isotope Nd-150 is one of the fission product and hence its neutron absorption cross-section should be known precisely. In the present work neutron radiative capture cross-sections for Nd-150 have been measured using enriched Nd-150 (94.0%) isotope at ten neutron energies, between 0.46 MeV and 3.04 MeV. The comparative activation technique has been employed and neutrons are produced by H-3(p,n)He-3 and Li-7(p,n)Be-7 reactions. The experiment was performed using 5.5 MV Van-de-Graaff accelerator at Bhabha Atomic Research Centre, Bombay. The sample under investigation was placed between the two standard samples of potassium iodide of similar size and was irradiated simultaneously by wrapping them in a cadmium foil to check any contribution from thermal and low energy neutrons. Irradiation was performed in the zero degree forward beam direction to get the least angular spread in proton beam energy. The product nucleus was identified by its characteristic γ-rays and its abundance using a 100 cc ORTEC Ge(Li) spectrometer. The two intense γ-rays of 117 KeV and 256 KeV were considered for the present measurement.

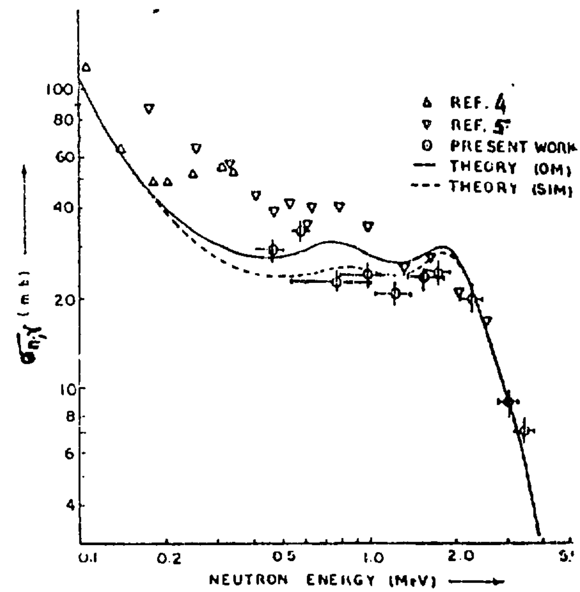

Fig. 1

The spread in the neutron energy was estimated by taking into account the angular spread of proton energy and the proton energy degradation in tritium and lithium targets /2/. The neutron flux at the place of irradiation was of the order of 10^7 neutrons cm^{-2} s^{-1} . The uncertainties involved in cross-section values are due to statistical fluctuations and systematic errors. Measured values are compared with the earlier reported values. The experimental values are also compared with the FISPRO II /3/ computer code in the framework of H-F statistical model qhich also takes care of cascade capture and direct capture.

* - Imphal College, Thoubal, Imphal.

$ - Janta College, Bakewar, Etawah.

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Helium ion induced fission excitation functions of Holmium, Erbium and Gold

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As part of a long range programme of work on the measurement of fission excitation functions of low Z ($Z \leqslant 80$) elements, the fission cross sections of Holmium ($Z = 67$), Erbium ($Z = 68$) and Gold ($Z = 79$) induced by He-ions in the energy range 30-70 MeV have been measured using the sensitive "fission track" technique. The fission events were recorded using lexan plastic track detector. Targets of high purity Rare Earths Oxide (purified by a series of anion exchange columns) and gold were deposited on high purity silver foils and were irradiated with the He-ions of different energies from the Variable Energy Cyclotron at Calcutta. The heavy element contents of both the rare earth oxides and the silver foils were estimated to be not more than 1-3 ppb. The experimental values for the fission cross sections are as follows:

A. Ho^{165} + He

E MeV	E^* MeV	(nano-barns)
40	37.9	$\leqslant$ 1
45	42.8	$\leqslant$ 3
45.3	43.1	$\leqslant$ 6
50	47.7	99±34
55	52.6	240±39
60	57.4	877±25
65	62.3	1550
70	67.2	2750

B. Er (Nat.) + He

E MeV	E^* MeV	(nano-barns)
34	32	$\leqslant$ 7
45	42.5	36.9
50	47.4	203
55	52.3	396
60	57.2	987
65	62.0	2090

C. Au^{197} + He

E MeV	E MeV	(Micro-barns)
30	27.9	10.0
40	37.7	186.8
50	47.5	2189
55	52.4	3195
70	67.1	16657

The excitation energies E^* are calculated by assuming full momentum transfer and using Q-values calculated from the mass tables of Myers and Swiatecki (1).

From the experimental cross sections, the ratios $\Gamma f/\Gamma n$ which measures the competition between fission and neutron emission was calculated. For low Z elements this ratio is very nearly equal to σ_f/σ_R where σ_R is the total reaction cross section which can be computed by standard optical model codes. These experimental ratios i.e. $\Gamma f/\Gamma n$ were analysed using the statistical model expression given by Vandenbosch and Huizenga(2).

$$\frac{\Gamma_f}{\Gamma_n} = \frac{Ko.\ a_n\ [\ 2\ a_f^{\frac{1}{2}}\ (E-E_f)^{\frac{1}{2}}\ -1]}{4\ a_f\ A^{\frac{2}{3}}\ (E-B_n)}\ \exp\ [\ 2a_f^{\frac{1}{2}}\ (E-E_f)^{\frac{1}{2}}\ -\ 2\ a_n^{\frac{1}{2}}\ (E-B_n)^{\frac{1}{2}}\ ]$$

to get the fission barrier E_f and the level density parameters a_f and a_n. A least square fitting procedure was used to fit the experimental Γ_f/Γ_n ratios by varying the values of a_f between A/8 and A/20 (reasonable upper and lower limits) and getting the "best fit" values for E_f , a_f and a_f/a_n. The values for Ho and Er are given in the following Table.

Table

Fission Barriers and level density parameters ('Best-fit' values)

Ho^{165} + He^4 ----------> Tm^{169}				
E_f (MeV)	a_f $(MeV)^{-1}$	a_n $(MeV)^{-1}$	a_f/a_n	ψ^2 (Least sum of squares of the deviations)
27.40	21.20	19.93	1.06	0.143
29.00	16.90 a	15.94	1.06	0.143
29.80	14.08 b	13.51	1.04	0.136
30.80	12.07 c	11.60	1.04	0.136
31.40	10.56	10.36	1.02	0.139
32.00	9.39	9.70	1.02	0.152
32.60	8.45	8.45	1.00	0.164

Er (Nat.) + He -----> $Yb^{171.3}$				
E_f (MeV)	a_f $(MeV)^{-1}$	a_n $(MeV)^{-1}$	a_f/a_n	ψ^2 (least sum of squares of the deviations)
27.0	21.40	19.40	1.10	0.398
27.0	17.10 a	16.40	1.00	0.096
27.0	14.20 b	14.20	1.00	0.021
28.0	12.20 c	12.20	1.00	0.021
29.0	10.70	10.70	1.00	0.028
30.0	9.50	9.50	1.00	0.037
31.0	8.50	8.50	1.00	0.047

where a) is A/10; b) is A/12; c) is A/14

This work represents the lightest compound nuclei for which fission barrier have been determined experimentally. It considerably extends the range of elements studied at these intermediate energies in the region of deformed rare earth nuclei. The results were compared with predictions of a semiemperical mass formulation based on the charged liquid drop and the agreement is good. The predicted liquid drop barriers E_f(L.D) are 32.6 MeV and 31.2 MeV (weighted average value) for Tm-169 and Yb-171.3 respectively. Further theoretical correlations of these experimental data and the evaluation of Au data and experimental work on ytterbium and terbium are in progress.

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Measurement of Alpha to Fission Branching ratios of Heavy element isotopes

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The importance of measurements of alpha emission to spontaneous fission branching ratio of heavy radionuclides is two fold. First, it is amongst the most fundamental property of heavy radionuclides and can be used for partial half life measurement. Secondly, the ratio can be used for identification of heavy element isotopes. The results of α/F branching ratio measurements of Cf-252 and Cm-244 are presented in this report.

In a systematic study of the various parameters associated with track registration and revelation of charged particle in CR-39, we have observed that it is possible to unambiguously reveal both alpha particles and fission fragments by a sequential etching procedure. This procedure has been used to determine the α/F ratio of Cf-252 and Cm-244 accurately.

Table - α/f Branching ratio of Cf-252

Expt. No.	Detector	Fission tracks in specified area	Alpha tracks in specified area	2π geometry	4π geometry
1.	CR-39 (DOP)	1.81×10^4	28.40×10^4	15.69	31.38
		1.83×10^4	29.05×10^4	18.57	31.74
		1.86×10^4	29.58×10^4	15.90	31.80
2.	CR-39 (DOP)	2.04×10^4	32.13×10^4	15.75	31.50
	CR-39	1.67×10^4	26.68×10^4	15.97	31.95
3.	CR-39	521	8073	15.50	31.00

Average = 31.56 ± 0.34

Table - α/F Branching ratio of Cm-244

Expt. No. and Detector used	Exposure time	Area counted mm^2	No. of tracks	Td $\#/Cm^2/min$	α/F ratio 2π geometry	α/F ratio 4π geometry
1. CR-39	10305 min.	48.92	4513[a]	0.895	-	-
(DOP)	30 sec.	6.61	10298[b]	3.116×10^5	3.482×10^5	6.963×10^5
	30 sec.	7.43	11620[b]	3.128×10^5	3.495×10^5	6.990×10^5
2. CR-39	19834 min.	34.48	5396[a]	0.789	-	-
	1 min.	7.14	19657[b]	2.753×10^5	3.489×10^5	6.979×10^5

Average = $(6.977 \pm 0.014) \times 10^5$

a - Fission tracks , b - Alpha tracks

Fission of Au-197, Lu-175 and Ho-165 with O^{16} ions - Cross sections, fragment angular distributions and mass distributions.

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With the availability of heavy ions from the Pelletron at TIFR, preliminary experiments have been carried out on the measurement of fission cross sections mass distribution and angular distribution of the fission fragments formed in the interaction of Au-197, Lu-175 and Ho-165 and O^{16} ions of energy 81, 85, 91, 96 and 108 MeV. Heavy ions bring in large amounts of angular momentum to the compound nucleus. These measurements will provide important information on the nature of angular momentum effects on the probability of heavy-ion induced fission. Angular distribution was measured over the range of Laboratory angles from 17° to 90°. The preliminary values for the fission fragment anisotropy in the Lu-175 + O^{16} system appears to be rather high; 10.25 and 11.6 at 81 and 85 MeV respectively. Further detailed analysis of the experimental data are in progress. A typical angular distribution curve is shown in figure 1.

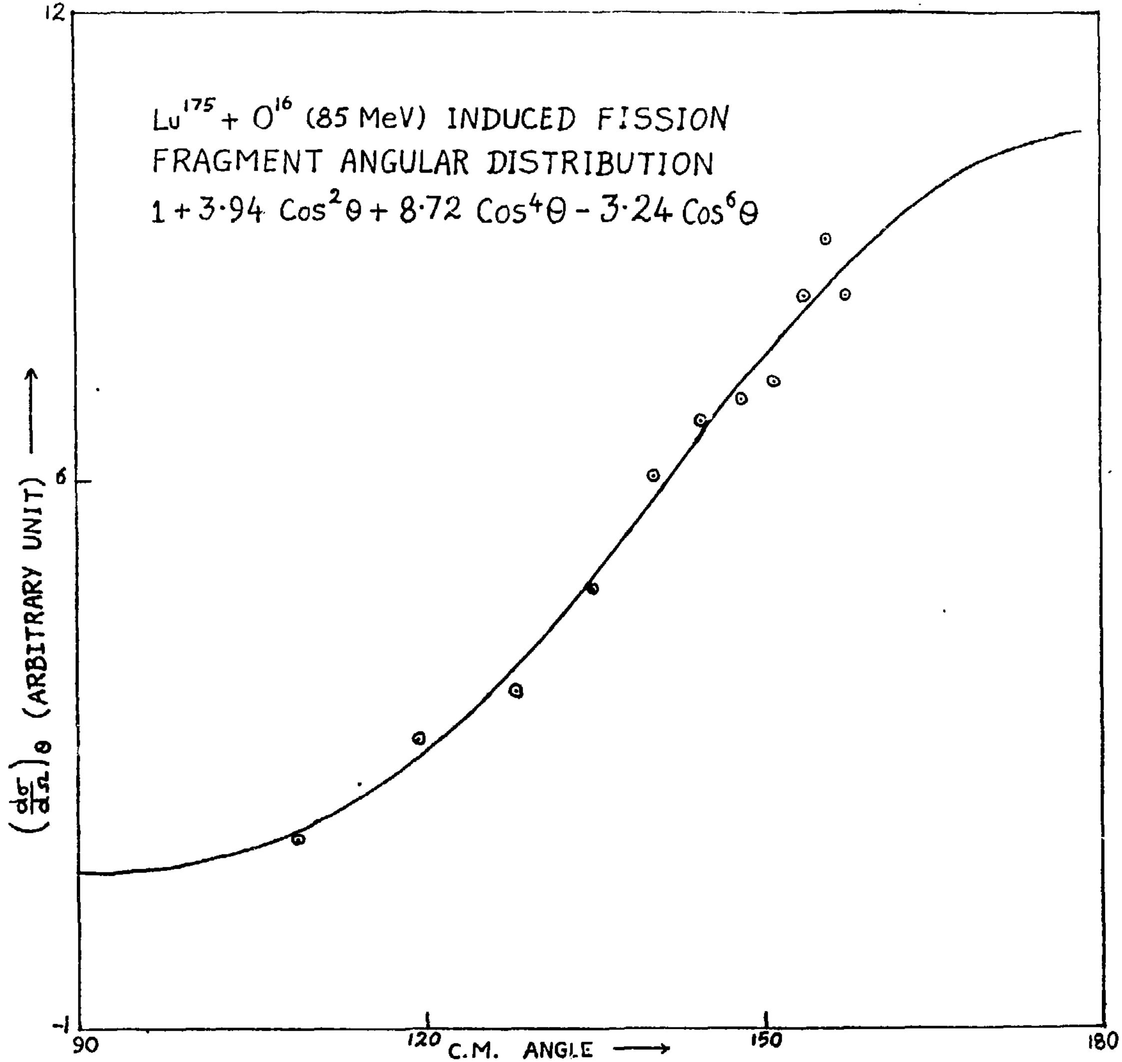
Lu175 + O16 (85 MeV) INDUCED FISSION
FRAGMENT ANGULAR DISTRIBUTION
1 + 3·94 Cos2θ + 8·72 Cos4θ - 3·24 Cos6θ
(dσ/dΩ)θ (ARBITRARY UNIT)
12
6
-1
90
120
C.M. ANGLE
150
180

FIGURE - 1

NUCLEAR DATA ACTIVITIES OF NUCLEAR CHEMISTRY SECTION FOR THE PERIOD FROM JANUARY, 1989 TO JUNE 1990

RADIOCHEMISTRY DIVISION
TROMBAY, BOMBAY 400 085.

Nuclear data activities have been carried out under the following categories.

1. <u>Nuclear Fission Studies</u>:

   a) Charge distribution
   b) Mass distribution
   c) Fragment angular momentum

2. <u>Nuclear Reactions and Nuclear Decay Data</u>

   a) Cross section measurements
   b) Isomeric transition
   c) γ-abundances.

1. <u>Nuclear Fission Studies</u>:

The charge distribution studies in SF and neutron induced fission of actinides have been carried out to arrive at the width of the distribution, most probable charge etc by determining fractional cumulative yields. The yield data are given in Table 1 to 3.

Isotopic yields of Iodine in fission of ^{238}U in ^{16}O induced fission have been carried out at Pelletron, TIFR and the data is presented in Table - 4.

The mass yield data in 38 MeV α-induced fission of ^{232}Th has been reported in Table - 5.

Alpha induced fission of ^{209}Bi at 52.8 and 58.7 MeV has been presented in terms of cross section for the production of fission products in Table - 6.

Investigation of fragment angular momentum of ^{95}Nb in ^{16}O induced fission of ^{197}Au at different excitation energies was carried out using medium heavy ion accelerator facility Pelletron at TIFR and data is presented in Table-7.

2. Nuclear Reaction and Decay Data:

Production cross-section data of a few radionuclides in alpha induced reactions and gamma ray intensity data in decay of ^{84m}Rb are given in Table 8 to 11.

In addition, to this a review on the methods of determination of fission yields has been communicated to Radiochimica Acta (1990).

TABLE - 1

Fractional cumulative yields in the thermal neutron induced fission of ^{241}Pu

Nuclide	zp	FCY
^{134}Te	52.07	0.794±0.067
^{135}I	52.52	0.942±0.024
^{138}Xe	53.75	0.778±0.017
^{140}Ba	54.55	0.972±0.004

A. Ramaswami, S.S. Rattan, R.J. Singh and Satya Prakash
DAE Symposium on Radiochemistry and Radiation Chemistry at Nagpur University, Nagpur, Feb. 1990.

TABLE - 2

Cumulative yields and fractional cumulative yields in SF at ^{252}Cf

Nuclide		Cumulative yield	FCY
^{108}Ru		5.89±0.29	0.897±0.043
^{109}Ru		4.08±0.30	0.725±0.046
^{137}Xe		4.11±0.13	0.833±0.026
^{138}Xe		5.24±0.16	0.894±0.027
^{139}Xe		3.86±0.29	0.678±0.046
^{139}Ba		6.08±0.21	0.964±0.034
^{141}Ba		5.66±0.10	0.927±0.016
^{142}Ba		5.41±0.10	0.898±0.017
$^{140}Ba(^{140}La)$	-	0.9978±0.0018	
$^{142}Ba-(^{142}La)$	-	0.915±0.034	
$^{146}Ce-(^{146}Pr)$		0.989±0.006	

S.B. Manohar, A. Ramaswami, B.K. Srivastava, A.V.R. Reddy, A.G.C. Nair, G.K. Gubbi, A. Srivastava and Satya Prakash
Nuclear Physics A 502, 3076 (1989).

TABLE - 3

Yields of Rare Earth Fission products in the SF of ^{252}Cf

Nuclide	% Fission Yields
^{144}La	4.78±0.65
^{145}Ce	4.20±0.21
^{148}Ce	4.14±0.35
^{147}Ce	3.76±0.40
^{147}Pr	4.38±0.43
^{148}Pr	4.81±0.15
^{147}Nd	2.43
^{151}Nd	1.21±0.06
^{152}Nd	2.47±0.06
^{158}Sm	0.70±0.04

M.S. Oak, A. Ramaswami and Satya Prakash
DAE Symposium on Radiochemistry and Radiation Chemistry held at Nagpur University, Nagpur (Feb. 1990)

TABLE - 4

Cumulative yield of Iodine Isotopes in ^{238}U + ^{16}O

Isotope	T 1/2	Cumulative yield
^{130}I	12.5 h	1.0
^{131}I	8.06 d	1.60
^{132}I	2.28 h	0.66
^{133}I	20.3 h	1.09
^{134}I	53.0 min.	1.04
^{135}I	6.68 hr.	0.55

A.V.R. Reddy, S.B. Manohar, A. Goswami, B.S. Tomar and Satya Prakash
DAE Symposium on Radiochemistry and Radiation Chemistry held at Nagpur University, Nagpur, Feb. 1990.

TABLE - 5
Fission yields in 38 MeV alpha induced fission of ^{232}Th

S.No.	Fission Product	Measured fission yield %
1.	^{78}As	0.22±0.02
2.	^{85m}Kr	1.37±0.29
3.	^{87}Kr	2.22±0.14
4.	^{88}Kr	2.43±0.26
5.	^{89}Rb	3.24±0.31
6.	^{91}Sr	3.18±0.22
7.	^{92}Sr	3.22±0.23
8.	^{93}Y	2.99±0.13
9.	^{97}Zr	3.61±0.64
10.	^{99}Mo	4.19±0.06
11.	^{103}Ru	3.36±0.08
12.	^{105}Rh	2.52±0.04
13.	^{107}Rh	1.78±0.18
14.	^{111}Ag	1.34±0.22
15.	^{112}Pd	1.36±0.18
16.	^{113}Ag	2.12±0.94
17.	^{115}Cd	1.69±0.15
18.	^{123}Sn	1.84±0.20
19.	^{127}Sb	2.37±0.01
20.	^{129}Sb	3.06±0.34
21.	^{131}I	4.40±0.21
22.	^{132}Te	5.63±0.25
23.	^{133}I	5.30±0.15
24.	^{138}Cs	5.27±0.45
25.	^{139}Ba	4.52±0.05
26.	^{140}Ba	4.72±0.01
27.	^{141}Ce	3.59±0.13
28.	^{142}La	2.12±0.23
29.	$^{143}C2$	2.20±0.14
30.	^{151}Pm	0.37±0.03

R. Guin, S.M. Sahakundu, S.B. Manohar and Satya Prakash
Radiochica Acta (In press) 1990.

TABLE - 6

Experimental cross section in microbarn for the formation of fission products in alpha induced fission of ^{209}Bi

Nuclide	E_α - Alpha Energy	
	52.8 MeV	58.7 MeV
^{91}Sr	1015±41	1779±131
^{92}Sr	1090±63	1620±40
^{95}Zr	1483±90	2850±40
^{97}Zr	1679±271	2726±151
^{99}Mo	2295±570	3726±372
^{103}Ru	2293±217	4090±251
^{105}Rh	2626±101	4909±100
^{111}Ag	1400±79	2429±439
^{113}Ag	1715±6	2879±225
^{115}Cd	1237±26	1886±267

S.S. Rattan, A. Ramaswami, R.J. Singh and Satya Prakash
DAE Symposium on Radiochemistry and Radiation Chemistry held at Nagpur University, Nagpur, Feb. 1990.

TABLE - 7

Fragment Angular Momentum of ^{95}Nb in ^{213}Fr ($^{197}Au+^{16}O$)

E_{Lab} MeV	E^*_{MeV}	< I >	Ym/Ym+Yg	J_{RMS} (h)
87.1	48.25	15.7	0.145	9.5
93.4	54.07	23.3	<0.105	12.0

T. Datta, H. Naik, S.P. Dange, P.K. Pujari and Satya Prakash
DAE Symposium on Radiochemistry and Radiation Chemistry held at Nagpur University, Nagpur, Feb. 1990.

TABLE - 8

The energies and Intensities at the gamma rays in ^{84n}Rb

E (MeV)	I_γ (relative)
215.78±0.04	45.8±2.2
247.90±0.03	100
463.55±0.07	63.2±3.9

R.Guin and S.K. Saha
DAE Symposium on Radiochemistry and Radiation Chemistry held at Nagpur University, Nagpur, Feb. 1990.

TABLE - 9

Experimental cross section in millibarns for the production of nuclides in Irradation of ^{27}Al with alpha particles

Nuclide	E_{α} in MeV (55.2)	(58.2)
^{7}Be	1.025	1.136±0.054
^{22}Na	49.77±0.82	43.98±1.63
^{27}Mg	2.34±0.07	3.01±0.12
^{28}Mg	0.435±0.039	0.41±0.032
^{28}Al	74.66±8.73	71.94±11.35
^{29}Al	5.81±0.54	7.37±0.31

S.S. Rattan, A. Ramaswami, R.J. Singh and Satya Prakash
Radiochimica Acta (In press)

TABLE - 10

Formation cross sections of nuclides in alpha induced fission of ^{209}Bi at different energies

Nuclides	Alpha Energy in MeV 52.3	58.3
^{22}Na	55.73±1.46	52.52.±0.65
^{24}Na	8.83±0.14	24.14±0.63
^{28}Mg	0.521±0.05	0.579±0.032

S.S. Rattan, A. Ramaswami, R.J. Singh and Satya Prakash
DAE Symposium on Radiochemistry and Radiation Chemistry held at Nagpur University, Nagpur, Feb. 1990.

TABLE - 11

Cross Sections in millibarns for the ^{4}He ion induced reactions on ^{197}Au targets

Residual Nuclide	E_α (MeV)	mb	Residual Nuclide	E_α (MeV)	mb
^{200}Tl	35.3	9.85±3.0	^{197g}Hg	43.9	1158.7±173.6
	42.5	6.39±0.78		48.6	1385.5±196.6
	43.9	7.76±1.94	^{197g}Hg	43.9	36.6
	48.6	4.36±1.09	r	48.6	38.6
^{199}Tl	43.9	99.6±24.9	^{198}Au	35.3	0.63±0.19
	48.6	79.7±19.9		42.5	1.67±0.43
^{198}Tl	45.1	274.7±52.8		43.9	4.23±0.85
	48.4	168.8±17.6		48.6	8.96±1.79
^{198}Tl	45.1	139.6±22.8	^{196}Au	35.3	21.85±1.06
	48.4	113.9±42.0		42.5	50.68±2.80
^{197}Tl	45.1	1104.9±266.9		43.9	99.51±11.8
	48.4	1311.0±318.8		48.6	130.9±12.8
^{197m}Hg	35.3	2.23±0.33	^{195}Au	35.3	0.34±0.17
	42.5	11.9±1.79		43.9	37.58±18.7
	43.9	18.51±2.77		48.6	55.16±27.5
	48.6	38.7±5.8	^{194}Au	48.6	6.27±1.56
^{197g}Hg	35.3	9.57±1.43			
exp	42.5	402.4±39.4			

N. Chakravarty, R.J. Singh, S.S. Rattan, A. Ramaswami, S.M. Sahakundu and Satya Prakash
Radiochimica Acta 48, 1 (1989).

SHELL DEPENDENT LEVEL DENSITIES IN NUCLEAR REACTION CODES

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&
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Statistical mode codes for compound nucleus decay are useful tools in interpreting and in predicting yields for many nuclear reactions. One of the important inputs to these calculations is the nuclear level densities as a function of the excitation energy. In this work we present the results of the incorporations into the computer code ALICE of a shell dependent level density formula due to Kataria, Ramamurthy and Kapoor (KRK) /1/. This model relates the shell effects in the nuclear level densities to the shell correction term of the nuclear mass surface. Because the ALICE code /2/ contains a library of experimental masses as well as a liquid drop mass formula, the necessary shell correction terms can be generated internally, without any free parameters.

We compare calculated and experimental excitation functions for the reactions $^{54}Fe(\alpha,n)$, $(\alpha,2n)$, (α,p) and (α,pn) as measured by Llouck and Muller/3/ and $^{56}Fe(\alpha,3n)$, $(\alpha,4n)$, $(\alpha,p2n)$ and $(\alpha,p3n)$ as measured by Ewart et al /4/ with calculated results. All experimental yields were measured using the stacked foil method with considerable energy degradation in the foil stacks. This means that the thresholds of the experimental excitation functions may be in error by perhaps +(-) 2 MeV; the peak yields, which are our main interest, are rather insensitive to this problem. All the calculated results will be from the code ALICE with default precompound decay preceding compound nucleus decay.

Several sets of the excitation function are shown in Figs.1 the yields of $^{56,57}Ni$ are significantly lower than their isobars $^{56,57}Co$. Is this because of Q value differences or because the Ni isotopes each have closed $f_{7/2}$ proton shell (and ^{56}Ni a closed neutron shell as well). A calculation using standard Fermi gas level density with Q values and binding energies calculated from the experimental masses should take into account differences in yields which are attributable to Q-value and Coulomb barrier effects. This option of the code has been used and the results are shown in the fig.1. It may be seen that the yields of the singly closed shell ^{57}Ni are overpredicted and the discrepancy is still greater for the doubly closed shell ^{56}Ni excitation function.

The results of this new level density routine with those using standard Fermi-gas level density for reactions leading to the products in the vicinity of the $f_{7/2}$ shell closure are also shown in fig.1. The shell dependent level density results shown as continuous curve results in a significantly improved results when compared with experimental excitation functions. More extensive comparisons with data are desirable for the final evaluation of the benefits of this approach to the problem of nuclear structure dpendent level densities.

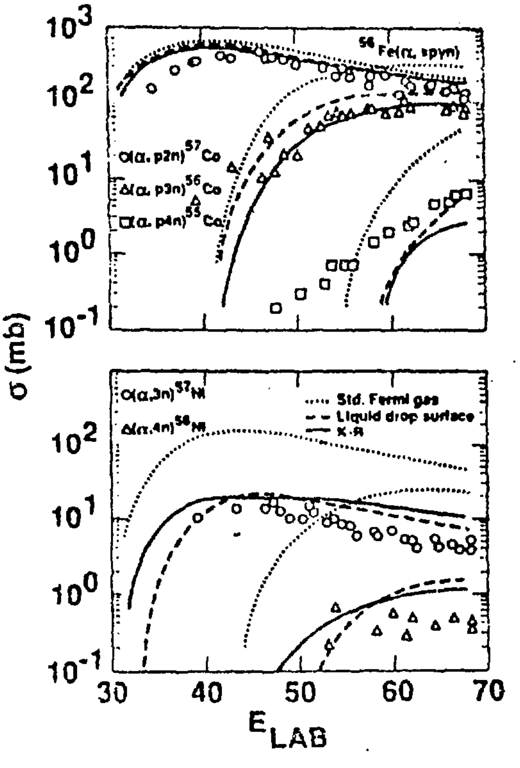

Fig.1

The calculated excitation functions are shown using a standard Fermi Gas level density (dotted). Fermi gas level densities with Q values and B.E. calculated from LDM (dashed lines) and with KRK level density approach (solid lines).

1. S.K.Kataria, V.S. Ramamurthy and S.S. Kapoor, Phys. Rev. C18. 949 (1978).

2. M.Blann and T.Plasil, University of Rochester report UR NSRI 181 (1973) unpublished and M.Blann. Lawrence Livermore National Laboratory Report UCID-20169 (1984).

3. T.S.Houck and J.M.Miller, Phys. Rev. 123. 231 (1961).

4. A.Ewart, C.Valentine and M.Blann, Nucl. Phs.69. 625 (1965).

FIXED PARTICLE HOLE LEVEL DENSITIES USING SADDLE POINT APPROXIMATION

S.K. Kataria and V.S.Ramamurthy
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Partial nuclear level densities characterised either by the number of quasi-particles or by the number of particle-hole excitations are of great interest in the study of nuclear reaction cross-sections in the form of intermediate structures or in the form of high energy part of the evaporation spectrum during the pre-equilibrium decay. For the case of non-interacting Fermion gas model of the nuclei, Williams introduced the concepts of effective excitation energy and Pauli correction energy in the fixed particle-hole level density expressions. Grand partition function approach has also been used in the literature by considering the nucleus as a system of non-interacting Fermi-particles and holes. For equidistant single particle level scheme, Bloch obtained simple analytical expressions for low p-h exciton numbers.For the special case of equal number of particles and holes i.e. $n_p = n_h = n$; this level density expression reduces to a normalised gaussian distribution in n with the variance σ^2 is given by gt/π^2. In other words, total nuclear level density is recovered in this method by integration over all values of n.

It is well known that shell structure effects play very important role in determining the total nuclear level densities as well as in fixed particle hole level densities. Several studies have been carried out to include the effect of the energy gap at the shell closures in fixed particle hole level densities. In the present work grand partition function has been used in the numerical evaluation of nuclear level densities starting from the given set of single particle levels for neutron and protons. We have used this approach to calculate the fixed particle hole level densities for a given excitation energy E_x and angular momentum projection M. We have solved this set of six simultaneous non-linear saddle point equations by gradient expansion around the estimated solutions and using iteration procedure. This method converges rapidly with good initial guess values. The linearised matrix becomes singular in case of unphysical conditions. We have carried these calculations using the neutron and proton levels for ^{44}Ca as earlier used in the ref./1/, wherein we had reported the results of our exact recursive methods for particle hole level densities and for nuclear level densities .

For the case of uniform level scheme, closed expressions for the entropy S and excitation energy E_x as a function of the particle hole numbers for a given kind of nucleons as derived by Bloch are:

$$S = 2\epsilon g t + 2 (N_p + N_h) \ln 2$$

$$E_x = \epsilon g t^2 + (N^2_p + N^2_h) g$$

where $\epsilon = (\pi^2/6 - 2\ln^2 2) = 0.684$

By expanding the entropy S around the maximum in powers of deviations δN_p and δN_h, the following expansion clearly brings out the role of p-h excitons in the second term.

$$S = \pi^2/3gt - 1/2gt [(\delta N_p - \delta N_h)^2 + \pi^2/6\epsilon(\delta N_p + \delta N_h)^2]$$

In other words the total level density can be recovered by summing over all values of δN_p, δN_h. Work is in progress to see if these expressions can be fitted to the calculated nuclear level densities with shell structure effects by defining an effective g value only in the second correction term, whereas the first term will be evaluted with full shell structure effects independently. In this way, we can obtain a fully consistent set of particle-hole level densities along with total nuclear level densities inclusive of shell structure effects. The comparison of the numerically calculated particle hole level densities against the values obtained using the above mentioned prescription are in progress and will be reported later.

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EVALUATION OF NEUTRON CROSS SECTIONS FOR TH-232 ABOVE 0.1 MeV

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As a part of nuclear data evaluations for the important nuclides of the ^{232}Th-^{233}U fuel cycle, the computer code HAUSER-V was used and modified to calculate the total neutron cross section. This code is based on Hauser-Feshbach statistical model and it uses Spherical Optical Model (SOM). This modified version of the code was first checked by performing the total neutron cross section calculations for the case of ^{238}U using the SOM parameters given in Ref./1/ and the calculated values were found to agreevery well with their measured values. To test the SOM parameters the measured total cross section data forms the benchmark. Similar calculation were repeated for the case of ^{232}th using the conventional SOM parameters for actinides given in Table 1 of Ref. /2/. In HAUSER-V program the default option uses the Wilmore-Hodgson, WH (4), parameters and it is found that using these parameters, the calculated total cross section values do not agree well with the measured data, while the Madland and Young (MY 2) parameters gives satisfactory values. Using these (MY2) SOM parameters, the calculations of neutron cross sections for the other actinides are being carried out.

The current status and accuracy of neutron cross sections data for each of the nine important isotopes (^{231,232,233}Th, ^{231,232,233}Pa and ^{232,33,234}U) relevant to ^{232}Th-^{233}U fuel cycle in fast energy range was reviewed. A paper on this review was submitted at 2nd INS Conference on "Nuclear Power-Advanced Fuel Cycles", T.I.F.R., Bombay, Jan. 1990.

1. A. Smith et.al., "Evaluation of the 238 U Neutron Total Cross Section", ANL/NDM-74 (1982).

2. P.D. Hodgson, "The Neutron Optical Model in the Actinide Region", Report 101/81.

NUCLEAR OPTICAL MODEL CALCULATIONS OF NEUTRON CROSS SECTIONS FOR THORIUM - 232

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(An Abstract of Ph.D. Thesis)

For our present illustrative spherical optical model calculations for Th-232 the various optical model parameters have been taken from SHELDON et al.(1986). A spherical optical model potential with a derivative Woods-Saxon form factor and Thomas spin orbit term is followed. The inelastic neutron cross sections are calculated for Th-232, using the computer code ABAREX, developed by MOLDAUER (1981). This computer code is an optical statistical model program for the calculation of energy averaged neutron induced nuclear reaction cross sections.

Using the "LEVELS" card of the ABAREX code, we specified 27 nuclear energy levels, the energy of the last level being at 1182.5 Kev. A continuum level density starting at the highest discrete level is derived from a Fermi - gas temperature of 0.385 Mev. a shift of -0.25 Mev value, and a spin cut-off parameter 5.93 . The energies, spins, and parities of the 27 discrete levels are assigned. The excitation functions for the selected energy states upto 2.5 Mev are analysed and the angular distributions are studied.

The investigations made by the author over the sensitivity analysis of the physical quantities deduced from the nuclear optical model on the various nuclear optical model parameters (real radius, imaginary radius, real diffuseness, imaginary diffuseness... etc.) in the particular case of Th-232 neutron induced nuclear reactions are briefly given. The sensitivity of the calculated energy averaged neutron induced nuclear reactions cross sections for Th-232 nucleus to various nuclear optical model parameters are investigated. The following cross sections are evaluated.

1. The total cross section
2. The absorption cross section
3. The total elastic cross section

The sensitivity of these cross sections to the follwoing parameters namely the optical model parameters, the continuum level density, gamma ray strength functions,...etc. were systematically investigated. The sensitivity of the calculated total, absorption, and the total elastic cross sections for Th-232 nucleus, to 1% change in the real radius keeping all the other parameters unchanged, as a function of energy of the incident neutron, are represented graphically.

It is interesting to note that the sensitivity curve shows a very complicated pattern and in our calculations for 1% change in real radius, the compound nucleus formation cross section sensitivity changes by more than 30% in the 0.05 to 0.2 Mev energy region. The calculated cross section as a function of "real radius" parameter at 0.1 Mev energy is represented graphically. The non-linear behavious is evident. The non-linear character of the sensitivity function is further demonstrated. Whenever the percentage change in cross section is plotted against the percentage change in "real radius". Additional sensitivity graphs performed using the ABAREX code for other optical model parameters of the nuclear optical model potential, are also shown. A successful attempt has been made by the author to elucidate the non-linear character of the sensitivity of the excitation functions.

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ROLE OF DIFFERENT LEVEL DENSITY FORMULATIONS IN THE PREDICTION OF NEUTRON INDUCED PARTICLE PRODUCTION CROSS-SECTIONS OF NI-58: *S.B. Garg, Neutron Physics Division, Bhabha Atomic Research Centre, Trombay, Bombay 400 085*

It is well known that level density is one of the most important parameters influencing the neutron nuclear reaction cross-sections of target nuclides. In this paper a detailed investigation of the four different level density prescriptions has been carried out for Ni-58 with special reference to total production cros-sections of neutron, hydrogen, helium and gamma-rays in the energy range 5-26 MeV. These data are useful in radiation damage studies, shield designs and the neutronics analysis of fusion and fast fission reactors. The study has been conducted in the frame work of multistep Hauser-Feshbach data evaluation scheme /1/ which includes Kalbach pre-equilibrium exciton model /2/ and Brink-Axel giant dipole resonance model /3/. The following four level density formulations have been considered in this work:

1. *Original Gilbert - Cameron Formula /4/ (OGCF) :* $= 0.0888 * \sqrt{aU}\, A^{2/3}$
2. *Improved Gilbert - Cameron Formula /5/ (IGCF) :* $= 0.[illegible]146 * \sqrt{aU}\, A^{2/3}$
3. *Back-Shifted Fermi Gas Formula of Dilg et al (BSFGF) /6/.*
4. *Formulation to include the effect of shell closures developed by Ignatynk et al (IF) /7/.*

Fig.1 *depicts the decay chain considered in the computations. Competition of neutron, proton and alpha-particle emission is included both in the precompound and compound stages of the reaction while gamma-ray competition is included only in the compound decay. Energy, spin, parity and the gamma-ray branching ratios of the discrete levels of all residual nuclides are included. Good optical model potential parameters for the emitted particles have been selected from the literature to obtain their transmission coefficients. Gamma-ray s-wave strength function is used to compute transmission coefficients for gamma-rays; level density and energy shift parameters for BSFGF are taken from Ivascu et al /8/. For the other three formulations pairing and shell energy corrections of Cook et al /9/ have ben used.*

OGCF has been taken as the base case for drawing inferences from this investigation. Fig.2 shows total helium and gamma ray production cross-sections in the above stated four cases. To save space, total neutron and hydrogen production cross-sections are not shown graphically. The conclusions are briefly stated below:

(a) Neutron emission in BSFGF case is enhanced (1-4%)upto neutron energies of 15 MeV and above this energy it is depleted marginally (~5%). In the IGCF case it is depleted marginally (~4%) and in IF case it is enhanced marginally (1-10%).
(b) Total hydrogen production cross-sections in the IGCF case are reduced over a wide energy range by 5-15%. IGCF and IF cases show marginal depletions but these two cases produce almost identical results above 12 MeV.
(c) As shown in Fig.2 total helium production cross-sections are most affected in the BSFGF case, the reduction being ~40% or more. IGCF case shows enhancement while IF case is identical to the base case above 12 MeV.
(d) Gamma-ray production cross-sections are enhanced in all the cases. IGCF and IF cases produce almost identical results as shown in Fig.2.

In summary it can be stated that hydrogen and helium production cross-sections are adversely affected in the BSFGF case while the gamma-ray production cross-sections are enhanced. IGCF and IF cases yield almost identical results but since IF accounts for the energy dependence of the 'a' parameter, it may be adopted in the computation of binary, tertiary and multiparticle reaction cross-sections required in reactor technology.

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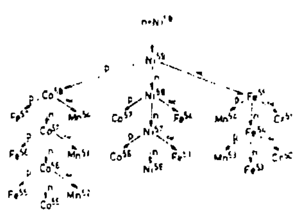
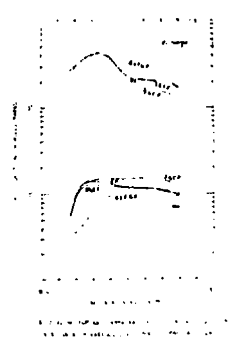


FIG. 1 NUCLIDES PARTICIPATING IN THE DECAY CHAIN OF $n+Ni^{58}$ REACTION



NEUTRON INELASTIC SCATTERING CROSS-SECTIONS OF NI-58 FOR TECHNOLOGICAL APPLICATIONS: *S.B. Garg, Neutron Physics Division, Bhabha Atomic Research Centre, Trombay, Bombay 400 085.*

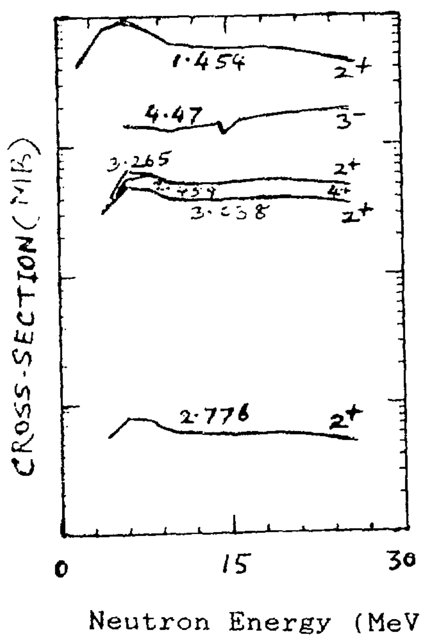
Ni-58 is one of the constituents of stainless steel which is frequently used as a structural material in nuclear reactors. Discrete level and total inelast'c scattering cross-sections of this element are impor .nt from the considerations of determining accurate neutron energy spectra in fusion and fast fission reactor systems. Sparse measurements of inelastic cross-sections at some selected energy points exist for this element. To fill the wide energy gaps, consistent computations of Ni-58 have been carried out in the frame work of multistep Hauser-Feshbach /1/ statistical theory covering the energy range 2-26 MeV. In this theory, competition for emission of neutron, proton, alpha-particle and gamma-rays has been considered at every stage of the reaction. Pre-equilibrium emission of neutron, proton and alpha-particles has been taken into account according to the Kalbach exciton model /2/. The Brink-Axel giant dipole resonance model has been employed to estimate the radiative capture cross-sections. Gamma-ray transmission coefficients have been determined from the s-wave gamma-ray strength function with the energy and width of the resonances taken into consideration. Transmission coefficients for neutron, proton and alpha-particles have been computed in the frame work of local spherical optical model with the parameters of neutrons and alpha-particles taken from Strohmaier et al /3/ and for protons taken from Mani /4/. The experimental information on level schemes and gamma-ray branching ratios for the various residual nuclides taking part in the reaction chain have been explicitly included in the calculations. Gilbert Cameron /5/ formulation for the level density with the pairing energy and shell correction parameters of Cook et al /6/ have been utilized for computations in the continuum energy region.

Direct level excitation cross-sections have been calculated with the Distorted Wave Born Approximation /7/. Deformation parameters for the discrete energy levels have been selected from the literature. The direct cross-sections have been explicitly used in the calculations to determine in a consistent manner the compound inelastic contributions.

The calculated direct level excitation cross-sections for the six discrete levels of Ni-58 are shown in Fig.1. It is noted that the total direct inelastic component for all the levels adds about 85 mb for neutron incident energies of 10 MeV or higher. Below 10 MeV the direct component is higher.

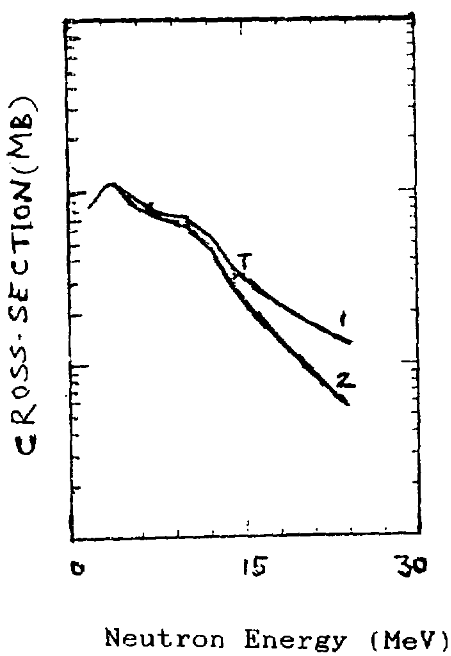
Total inelastic cross-sections with and without the inclusion of direct component are depicted in Fig.2. Curve 1 is generated with direct component included and curve 2 represents the results of compound statistical theory. It is noted that curve 1 gives higher values of the inelastic cross-sections at each energy point than curve 2 and represents the measured data rather well. It is concluded that the direct component of inelastic cross-section should be evaluated and included in the prediction of inelastic cross-sections for technological applications even though the computations become more cumbersome and time consuming.

References: 1. W. Hauser and H. Feshbach, Phys. Rev. 87, 366(1952) 2. C. Kalbach, Z. Phys. A283, 401(1977) 3. B. Strohmaier et al., Paper presented at the IAEA Advisory Group Meeting on Nuclear Data for Radiation Assessment and Related Safety Aspects, Vienna (1981) 4. G.S. Mani, Nucl. Phys. A165, 225(1977) 5. A. Gilbert and A.G.W. Cameron, Can. J. Phys. 20, 477(1967) 6. J.L. Cook et al., Aust. J. Phys. 20, 477 (1967) 7. G.R. Satchler, Nucl. Phys. 55, 1 (1964).



Direct level Excitation Cross sections of Ni-58

Fig. 1



Total neutron Inelastic Cross sections of Ni-58 with and without direct reaction component

Fig. 2

GENERATION OF MULTIGROUP CROSS-SECTIONS FOR FEW ELEMENTS FROM ENDF/B-IV LIBRARY USING THE COMPUTER CODE NJOY

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WIMS cross section library obtained from UK is being extensively used here in both light water as well as heavy water moderated reators. This library has been obtained from the point data taken from UKNDL files. There are number of elements of our interest for which cross sections are not there in this library. Recently, we developed the capability of generating multigroup cross sections from ENDF/B-IV and V libraries by adapting a code system NJOY /1/ on the ND computer /2/.

The computer code NJOY is a versatile code for nuclear data processing. It is a comprehensive computer code package for producing pointwise and multigroup cross sections from ENDF/B-IV and B-V evaluated nuclear data. The code NJOY is a modular system and consists of 16 modules. Each module is essentially a free standing code. All the modules except two have been adapted on the ND computer. The two modules which have not been adapted essentially give results in a particular format and these use special system subroutines.

In WIMS library, the cross sections are written in a particular format. In this library nuclear data are stored on two files. One file gives infinite dilution data for all the isotopes and other file gives resonance integral for different dilution and temperature for resonance materials only. We have added a new module called WIMSR to the NJOY code system which prepares the required number of files for an isotope in which cross sections are written in the format of WIMS library.

The code NJOY has now been used to generate 69-group cross sections for number of elements such as, V-51, Mn-55, Co-59, Nb-93, Ag-107, Ag-109 and Pa-233. The cross sections for all the above elements except Mn-55 were not there in the old WIMS library available with us. These cross sections were included in the WIMS library.

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GENERATION OF TRANSPORT CROSS-SECTION FOR HYDROGEN USING B-N APPROXIMATION.

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For the analysis of light-water reactors, it is important to use proper transport cross-section in place of total cross section for taking into account the effect of P-1 scattering in hydrogen. In the multi-group picture, one can obtain the transport cross-section by using the follwoing expression /1-2/.

$$\sigma(tr,g) = \sigma(t,g)-[\Sigma\, \sigma_1\ (g' \dashrightarrow g)\ J(g')]/J(g) \qquad (1)$$

where, $\sigma(tr,g)$, $\sigma(t,g)$, σ_1 ($g' \dashrightarrow g$) and $J(g)$ represent transport cross section in energy group-g, total cross-section in group-g, P-1 scattering cross-section from group-g' to group-g and current in group-g, respectively. In Eq.(1), the summation in the last term on rhs is on group-g'. In the thermal energy range by using the principle of detailed balance, a simple expression for transport cross-section is obtained /1-2/.

$$\sigma(tr,g) = \sigma(t,g)-[\Sigma\sigma_1\ (g \dashrightarrow g')] \qquad (2)$$

Here also the summation in the last term on rhs is on group-g'. This expression has been used in WIMS 69-group library/3/ in the whole energy range and these $\sigma(tr,g)$ values were being used by us in the analysis of light-water reactors. However, the principle of detailed balance on which Eq.(2) is based is not true above thermal energy cut-off i.e. 4 ev. Thus, Eq.(1) should be used to generate the transport cross-section in the fast and epithermal energy ranges. For this purpose we have to evaluate the current spectra J(g) in this energy range.

After generating P-N scattering /2/ matrics for hydrogen in water, B-N approximation /2/ was used to generate the current spectrum in the energy range above 4 ev. Fission spectrun of U-235 given in WIMS-69 was used as the source and a buckling value $B^2 \leq 0.01$ was given to find the current spectrum. This current spectrum was, then, used to obtain the transport cross-section of hydrogen using Eq.(1) in WIMS-69. The most important observation was that the current spectrum was almost independent of order of B-N approximation and buckling value for $N \geq 1$ and $B^2 \leq 0.01$. These transport cross-sections calculated by us using Eq.(1) are significantly lower than those of WIMS-69 obtained by using Eq.(2). Thus, the K-eff (multiplication factor) values of light-water reactors will reduce by using these new transport cross sections and this may have significance in the design and analysis of water-moderated high leakage reactors.

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2. G.I. Bell and S.Glasstone, "Nuclear Reactor Theory", Van Nostrand Reinhold Company, New York (1970).
3. Wims-69 cross-Section Library, Version available with Theoretical Physics Division, BARC, India.

MULTIPLICATION FACTOR UNCERTAINTIES FOR SMALL SOLUTION REACTORS

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The experimental data obtained from the critical assemblies serve the main source for validation of the calculation methods and the cross-section libraries which are employed to predict various reactors parameters. WIMS library, for example, has been extensively validated against numerous experiments. These experiments include light-water as well as heavy-water moderated thermal lattices with varying enrichment of fissible materials. Very satisfactory values of the multiplication factors for these lattices have been predicted using this library and employing the usual transport approximation to take into account anisotropy of the scattering cross-section.

However, the critical assemblies containing the solution of highly enriched fissile material have not been extensively analysed using this library. The critical assemblies haveing very high enrichment of fissile material such as U-233, U-235 and Pu-239 in form of solutions have been analysed using discrete ordinate (Sn) method of solving the neutron transport equation. These include the bare as well as fully water reflected assemblies in spherical as well as cylindrical geometries. The hydrogen to fissile atom ratio (H/X) varies from approximately 50 to 2000. The analysis shows that the WIMS library predicts the multiplication factor satisfactorily (within 1%) for the assemblies which are characterized by low leakage of neutrons (less than 20%) from the core. The critical assemblies which are characterized by high leakage of neutrons from the core (more than 20%) are overpredicted depending upon the present observations that the transport cross-sections of hydrogen in fast energy range which play dominant role in determining the leakage from the core are not adequate to take into account the leakage properly.

We have further studied the effect of modified transport approximation on the prediction of leakage from such systems. In this approximation the energy transfer to lower groups due to anisotropic scattering of neutrons which is neglected in usual transport approximation, is accounted for by properly modifying the self scattering cross-section in each group. By using this modified transport approximation for hydrogen, significant improvement in the values of the multiplication fators for highly leaking critical assemblies is observed.

NEUTRON INDUCED REACTION CROSS-SECTIONS OF NI-58 AND NI-60: S.B. Garg, Neutro Physics Division, Bhabha Atomic Research Centre, Trombay, Bombay 400 085.

Under an IAEA Research Contract dealing with the 'Methods for the Calculati of Neutron Nuclear Data for Structural Materials of Fast and Fusion Reactor investigations have been carried out for the neutron induced reaction cross-sectio of Ni-58 and Ni-60 utilizing the three data evaluation schemes, namely, multist Hauser-Feshbach scheme with pre-equilibrium corrections, geometry dependent hybr model scheme and the unified exciton model scheme. In particular, the followi types of neutron cross-section data have been evaluated in the energy range 1- MeV:

(a) (n,p), (n,α), (n,γ), (n,2n), (n,np), (n,pn), (n,nα), (n,αn) and (n,2 cross-sections.

(b) Energy spectra of the emitted neutron, proton, alpha-particle and γ-rays.

(c) Discrete energy level excitation and total inelastic scattering cross-sections

(d) Angle-energy correlated double differential cross-sections for the seconda emitted neutrons

(e) Total production cross-sections for neutron, hydrogen, helium and gamma-rays.

Measured cross-section data for several reactions listed above do not exist f Ni-58 and Ni-60 in the entire energy range considered in this investigation and th this study serves to extrapolate or generate such data for these reactions. In th regard, the above mentioned three evaluation schemes have been intercompared a appropriate conclusions are drawn. The results are summarized in a paper /1/ a sent to IAEA.

REFERENCE

1. S.B. Garg; Investigation of Neutron Induced Reaction Cross-Sections of Ni-58 a Ni-60 with Various Nuclear Model Evaluation Schemes, Paper Presented at the 3 IAEA-CRP Meeting held at Vienna during June 1990.

DEVELOPMENT OF CODES FOR THE NEUTRON-PHOTON COUPLED CROSS-SECTION LIBRARY: S.B. Garg, Neutron Physics Division, Bhabha Atomic Research Centre, Trombay, Bombay 400 085

A very tedious but vital task of adopting and developing the following codes was completed for the inhouse computer in order to generate neutron-photon multigroup coupled cross-sections to carry out the design oriented core physics, safety and shielding studies of nuclear reactors:

1. XLACS-IIA Code /1/, which processes neutron resonances and generates multigroup cross-sections for all the neutron induced reactions together with the anisotropic scattering matrices of any order.

2. LAPHNGAS Code /2/, which generates multigroup gamma-ray production cross-sections via neutron interactions

3. SMUG Code /3/, which produces multigroup photon interaction cross-sections and the anisotropic scattering matrices.

4. CHOX Code /4/, which combines multigroup neutron, gamma-ray production and photon interaction cross-section libraries to produce a coupled masterfile by suitably arranging and merging the various types of reaction cross-sections.

5. NITAWL Code /5/, which makes use of the masterfile to produce the desired neutron-photon coupled cross-section library for the simultaneous transport of neutrons and photons.

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1. G.S. Robinson; XLACS-IIA: A Modified Version of XLACS-II for Processing ENDF Data into Multigroup Neutron Cross-Sections in AMPX Master Library Format, Informal Memo (1981); PSR-182 (1982)

2. D.J. Dudgiak et al; LA-4750-MS (1972)

3. J.L. Lucius and W.E. Ford III; SMUG: An AMPX Module For Generating Multigroup Photon Cross-Sections, PSR-63

4. N.M. Greene and J.E. White; CHOX: The AMPX Module For Cross-Section Interface Manipulation, PSR-117 (1978)

5. L.M. Petric et al; NITAWL: AMPX Module For Resonance Self-Shielding and Working Library Production, PSR-117 (1978)

MULTIGROUP PHOTON INTERACTION CROSS-SECTIONS: S.B. Garg, Neutron Physics Division, Bhabha Atomic Research Centre, Trombay, Bombay 400 085.

28 group photon interaction cross-sections have been generated for H,B,C,O,Na, Ni, S, Ca, Cl, Mg. Zn, Fe, Cu and Pb in order to carry out the mineral exploration studies using the basic DLC-7F cross-section library and SMUG Code /1/. P3-anisotropic scattering matrices have also been computed for a selected energy group structure which is specially tailored to facilitate such investigations. The generated cross-sections have been converted to a suitable format for application in DTF-IV code /2/.

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1. J.L. Lucius and W.E. Ford III; SMUG: An AMPX Module For Generating Multigroup Photon Cross-Sections, PSR-63

2. K.D. Lathrop; DTF-IV, A Fortran-IV Program For Solving the Multigroup Transport Equation with Anisotropic Scattering, LA-3373 (1965)

CREATION OF A MULTIGROUP CROSS SECTION LIBRARY BASED ON JENDL-2 FOR USE IN NEUTRONIC CALCULATION OF FAST REACTOR PARAMETERS

M.M. Ramanadhan, V. Gopalakrishnan, K. Devan and S. Ganesan

The work on creating a multigroup cross section library based on the Japanese evaluated nuclear data library JENDL-2, and compatible with a composition dependent cross section preparation code EFFCROSS used here, has been continued. The purpose and the earlier part of this work was reported in the last Progress Report/1/. Subsequently, 25 group constants for the nuclides U-233, U-234, U-236, U-238, Np-237, Np-239, Al, Be-9, Ca and Nb were added to the set. In addition, since data for Oxygen and Nitrogen were not available in JENDL-2, multigroup data based on ENDF/B-IV were also included in this set. This entire set of non-adjusted, 25 group constants which include unshielded cross sections, self-shielding factors for various temperatures and dilution cross sections, elastic and inelastic transfer matrices etc., has been saved in the binary form compatible with the retrieval routines of the code EFFCROSS. This work is being continued to include data for more nuclides.

In addition to the above broad group library, the fine group (roughly 30 fine groups per broad group) elastic scattering data, constituting the PART-2 library, required to account for the elastic scattering resonances of light nuclei have also been calculated, from the basic JENDL-2 data. These data were also saved in the binary form as required by EFFCROSS.

Integral validation of the above multigroup set by analyses of selected benchmark fast critical assemblies have been taken up.

1. R.P. Anand (Ed.), Progress Report on Nuclear Data Activities in India for the Period from July 1987 to December 1988, BARC-1464, page 44.

COMPARISION OF NUCLEAR DATA OF CALIFORNIUM-252 FROM ENDF/B-V AND ENDL/84-V

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

Cf-252 is a candidate nuclide for an external source of neutrons used in a fast reactor. This material whose half life is 2.645 years decays through alpha with a branching ratio of 96.908 % and through spontaneous fission with a branching ratio of 3.092 %. It emits nearly 3.77 neutrons per spontaneous fission. Recently the use of Cf-252 has been studied as an alternative to Sb-Be source in the Fast Breeder Test Reactor (FBTR) at Kalpakkam/1/. The evaluated nuclear data used for such a study plays a crucial role with regard to the estimations and recommendations. Hence a detailed comparison of all the important nuclear data of this nuclide taken from ENDF/B-V and ENDL/84-V files was made in the multigroup form/2/. It is to be noted that the number of neutrons released per neutron induced fission show marked differences (around 30 %) between the two files. In addition, capture and fission cross sections also show significant differences in the low energy (keV) region. Tables 1, 2 and 3 give these comparisons.

1. C. P. Reddy and S.M. Lee, Study of Cf-252 Source for FBTR, RG/RPD/317 Rev No. 0 (1989).

2. M. M. Ramanadhan, V. Gopalakrishnan and S. Ganesan, Multigrouping and Processing of Neutron Cross Section Data for Californium-252 from ENDL/84-V and ENDF/B-V Files and Intercomparisons of Multigroup Constants, Internal Note RPD/NDS/31 (1989).

Table 1

COMPARISON OF CF-252 CAPTURE CROSS SECTIONS

GROUP	UPPER ENERGY	ENDF/B-V VALUES (barns)	RATIO ENDL / ENDF/B-V
14	9.13 KEV	0.7849E 00	1.9146
15	5.54 KEV	0.7375E 00	2.8270
16	3.36 KEV	0.8352E 00	2.8125
17	2.04 KEV	0.1072E 01	2.3390
18	1.24 KEV	0.1457E 01	1.8040
19	750 EV	0.2045E 01	1.3846

Table 2
COMPARISON OF CF-252 FISSION CROSS SECTIONS

GROUP	UPPER ENERGY	ENDF/B-V VALUES (barns)	RATIO ENDL / ENDF/B-V
13	15.1 KEV	0.4015E 00	0.9956
14	9.13 KEV	0.6324E 00	0.8608
15	5.54 KEV	0.9718E 00	0.7528
16	3.36 KEV	0.1440E 01	0.6605
17	2.04 KEV	0.2086E 01	0.6213
18	1.24 KEV	0.2964E 01	0.5594
19	750 EV	0.4202E 01	0.5189
20	455 EV	0.7125E 01	0.4064
21	276 EV	0.1305E 02	0.6826

Table 3
COMPARISON OF NUMBER OF NUETRONS RELEASED PER FISSION FOR CF-252

GROUP	UPPER ENERGY	ENDF/B-V VALUES	RATIO ENDL / ENDF/B-V
1	14.5 MEV	0.5424E 01	0.6803
2	3.68 MEV	0.4784E 01	0.7013
3	2.23 MEV	0.4487E 01	0.7130
4	1.36 MEV	0.4316E 01	0.7205
5	822 KEV	0.4209E 01	0.7256
6	499 KEV	0.4145E 01	0.7287
7	302 KEV	0.4100E 01	0.7309
8	183 KEV	0.4074E 01	0.7322
9	111 KEV	0.4059E 01	0.7329
10	67.5 KEV	0.4051E 01	0.7334
11	40.9 KEV	0.4046E 01	0.7337
12	24.8 KEV	0.4042E 01	0.7339
13	15.1 KEV	0.4040E 01	0.7340
14	9.13 KEV	0.4039E 01	0.7340
15	5.54 KEV	0.4038E 01	0.7340
16	3.36 KEV	0.4038E 01	0.7340
17	2.04 KEV	0.4038E 01	0.7340
18	1.24 KEV	0.4038E 01	0.7341
19	750 EV	0.4038E 01	0.7341
20	455 EV	0.4037E 01	0.7341
21	276 EV	0.4037E 01	0.7341
22	101 EV	0.4037E 01	0.7341
23	22.6 EV	0.4037E 01	0.7341
24	3.06 EV	0.4037E 01	0.7341
25	.414 EV	0.4037E 01	0.7341

SENSITIVITY STUDIES USING OPTICAL STATISTICAL MODEL FOR NEUTRON CROSS SECTIONS OF PU-240,U-238 AND TH-232

K. Devan, V. Gopalakrishnan, S. Subbarathnam*, S. Ganesan and M. M. Ramanadhan

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A sensitivity study of the calculated cross sections to optical model parameters was made using ABAREX/1,2/, an optical-statistical model code for the calculation of energy averaged neutron induced nuclear reaction cross sections. In addition to s and p wave strength functions, the cross sections for total, absorption, shape elastic, compound elastic, inelastic to individual or to desired discrete levels, gamma ray production, radiative capture, and fission (if transmission factors are supplied) can be calculated using this code.

For our present calculations, we have taken parameters used by Sheldon et al./3/, which are the 'global' parameters fitted to actinide region. A spherical optical potential with a Woods-Saxon form factor in the real part and its derivative form in the imaginary part and Thomas spin-orbit term are employed. The parameters used are:Real well depth =49.82- 17(N-Z)/A-0.3E MeV; Imaginary well depth= 5.52- 9(N-Z)/A + 0.4E MeV; Real radius = Imaginary radius=1.26fm; Real diffuseness=0.63fm; Imaginary diffuseness=0.52fm; Spin-Orbit depth=6.2 MeV; Spin-Orbit radius=1.12fm and Spin-Orbit diffuseness=0.47 fm, where E is the incident neutron enegy in MeV.

Figures 1-3 give plots of the sensitivity of calculated total and absorption cross sections for the three nuclides to 1% change in the real radius with all other parameters unchanged, as a function of the incident neutron energy, and we observe more than 35% change in the absorption cross section below 0.2 MeV. It may be worth investigating such a high sensitivity of the cross section to a small change in radius.

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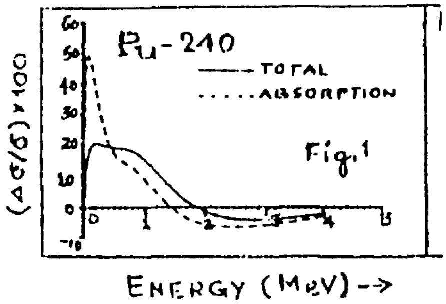
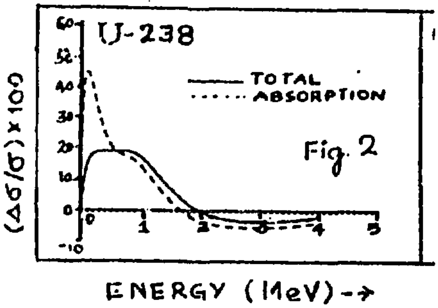
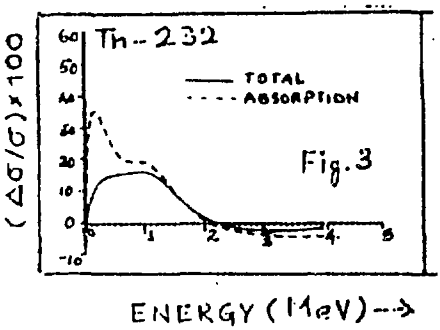
3. Sheldon et al.,(1986) J. Phys. G12, 237.







Figs. 1, 2 and 3: Sensitivity of total and absorption cross sections to 1% change in real radius parameter for Pu-240, U-238 and Th-232.

NEW RESULTS OF THE USE OF RECENT NUCLEAR DATA FILES FOR U-233 AND TH-232 ON DOUBLING TIME CHARACTERISTICS FOR A TYPICAL FBR*

S. Ganesan, M. M. Ramanadhan, V. Gopalakrishnan, P. V. K. Menon and S. M. Lee

The group constants of processed nuclear data of U-233 and Th-232 from recent evaluated nuclear data files are intercompared. U-233 data were processed from ENDL-84/V and JENDL-2 and ENDF/B-IV and Th-232 data were processed from ENDF/B-V(Rev.2) and JENDL-2. Comparison of these later data with the standard data in use in Kalpakkam for FBR design are presented. The pre-processing and processing of data were carried out using the IAEA codes LINEAR, RECENT, SIGMA1 and the Kalpakkam codes REX1, REX2 and REX3. The U-233 and Th-232 data thus obtained were used to study the characteristics of a 500 MWe FBR in terms of breeding ratios, fissile inventories and doubling times.

* Paper presented at the International Conference on the Physics of Reactors: Operation, Design and Computation (PHYSOR-90), held during April 23-27, 1990 in Marseille, France. See Proceedings of this Conference, Volume 1 (Oral Sessions I to VII), page III-1.

A PRELIMINARY ESTIMATION OF THE UNCERTAINTY IN K-EFF OF ZPR-6-7 CRITICAL ASSEMBLY USING CROSS SECTION COVARIANCE INFORMATIONS.

K. Devan, V. Gopalakrishnan, M.M. Ramanadhan and S. Ganesan

In reactor physics, the sensitivity methodology using the error informations on cross sections has the aim of quantifying the uncertainties, in the reactor performance parameters, that are arising due to the uncertainties in basic nuclear data. It helps us to ascertain the design margin in integral parameters of reactors with or without the critical experimental programme. We did some preliminary sensitivity studies/1/ using the covariance informations of the basic nuclear data and their impact on k-eff of a (PuO_2- UO_2) fueled fast critical benchmark assembly ZPR-6-7 having a core size of 3100 litres.

For the present study, we have taken, the reported/2/ four group covariance matrix data for capture cross sections of U-238 and Pu-239 and nu-fission ($\nu\sigma_f$) cross section of Pu-239. A four group effective cross section set was obtained by condensing the availbale 25 group cross sections using the core centre fluxes of the same assembly. With this cross section set, sensitivity coefficients are calculated groupwise and reactionwise by performing two exact k-eff calculations employing one-dimensional spherical model and diffusion theory as follows: In each case, a parallel set of four group cross section set was prepared by perturbing a particular group cross section of a particular nuclide by 10 % and correspondingly adjusting the total cross section for consistency. The group sensitivity coefficient is then calculated as a partial derivative using the values of k-eff obtained from the diffusion calculations with perturbed and unperturbed cross section sets. The sensitivity response function was calculated using two different multigroup data sets derived from different basic files and found to justify the linearity assumption in the sensitivity methodology. We calculated the uncertainties in k-eff due to the uncertainties in capture cross section of U-238, nu-fission cross section of Pu-239 and capture cross section of Pu-239 using the following relation. In mat ix notation, the variance is given as

$$VAR(R) = E_R^2 = (S E_x)^+ C_x (S E_x)$$

where S is the sensitivity matrix consisting of partial derivatives of the integral parameter and E_x is the error matrix and C_x is the correlation matrix. The superscript + indicates transpose of the matrix. The one-sigma error in U-238 capture cross section alone gives an error of 2.1 % in k-eff, capture cross section of Pu-239 gives an error of 0.9 % and nu-fission cross section of Pu-239 gives an error of 2.5 %. The estimated uncertainties in k-eff are significantly influenced by the values of off-diagonal elements in the correlation matrix.

When the one-sigma errors of the above three cases are used together in the calculation we obtain a net error of 3.[illegible] % in the k-eff which is also given by the quadratic addition of the individual errors when there is no correlation among these cases.

1. K. Devan, V. Gopalakrishnan and S. Ganesan, A Preliminary Estimation of the Uncertainty in k-eff of ZPR-6-7 Using Cross Section Covariance Informations, Internal Note RPD/NDS/26 (1989).

2. Takanobu Kamei and Tadashi Yoshida, "Error Due to Nuclear Data Uncertainties in the Prediction of Large Liquid-Metal Fast Breeder Reactor Core Performance Parameters", Nucl. Sci. & Engg. 84, 83-97 (1983).

ADAPTATION OF NJOY/ERRORR MODULE IN ND-570 COMPUTER

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The multigroup covariance processing code NJOY/ERRORR module /1/ has been adapted to the ND-570 computer. It calculates the multigroup covariance matrices of microscopic cross sections and $\bar{\nu}$ values from the basic covariance data given in ENDF/B-V or VI evaluated nuclear data file. A sample problem given in ref.2 was tried for U-235, using the available data from ENDF/B-V.2 special purpose actinide file. The output covariance data /3/ is found to be identical with the reported results.

We also repeated the covariance calculation in 25 group structure. Attempts are being made to process cross material and cross reaction covariances.

1. D. W. Muir and R. E. MacFarlane, "The NJOY Nuclear Data Processing System, Volume IV: The ERRORR and COVR Modules", Los Alamos National Laboratory, LA-9303-M, Vol. IV, (ENDF-324) (1985).

2. R. E. MacFarlane, D. W. Muir, and R. M. Boicourt, "The NJOY Nuclear Data Processing System, Vol. I: User's Manual," Los Alamos National Laboratory, LA-9303-M (ENDF-324) (1982).

3. K. Devan and V. Gopalakrishnan, "Adaptation of NJOY/ERRORR module in ND-570 computer", Internal note RPD/NDS/34.

MULTIGROUP CROSS SECTIONS CALCULATED FOR VARIOUS APPLICATIONS

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i. On a request from the Radiation Shielding and Statistical Physics Section of our Division, 100 group (DLC-2 Structure) cross sections of Li-6(n,α)T, Li-7(n,n'α)T, and B-10(n,2α)T were calculated/1/ from JENDL-2 basic data library. These were required for the study of Tritium Production in the proposed 500 MWe Prototype Fast Breeder Reactor (PFBR) under study at Kalpakkam. In addition, 25 group (Cadarache Type) and 100 group (DLC-2 Type) cross sections for over 15 structural nuclides were requested for application in activation analysis. We have preprocessed and multigrouped/3/ all the required data from JENDL-2.

ii. One group average cross sections of neutron nuclear reactions often become necessary to make approximate estimates of certain reactor parameters neutronically. To this end, one group cross sections were computed for all the nuclides available in the ENDL/84-V basic data library. This library being a preprocessed one, gives tabulated cross sections against energy with linear interpolation. Although reconstruction of the resonances into point data has been made with large tolerances (> 2%), this library is handy for rough estimates. The cross sections were first averaged to 25 energy groups, using REX1-87, and the spectrum averaged one group values were then obtained/2/ with the 25 group flux spectra obtained from a neutronic calculation corresponding to the core region of FBTR, and for the core-1, core-2, axial blanket and the radial blanket regions of PFBR.

1. K. Devan, V. Gopalakrishnan and S. Ganesan, 100 Group Cross Sections for Li-6, Li-7 and B-10 from JENDL-2 Library for Prediction of Tritium Production in PFBR, Internal Note RPD/NDS/30 (1989).

2. V. Gopalakrishnan and S. Ganesan, One Group Cross Sections of All Nuclides from ENDL/84-V Library, Internal Note RPD/NDS/25 (1989).

3. K. Devan, V. Gopalakrishnan and M. M. Ramanadhan, Activation Cross Sections of Sodium and Structural Nuclides in 25 and 100 Groups from JENDL-2, Internal Note RPD/NDS/32 (1990).

CREATION OF PC VERSIONS OF COMPUTER CODES

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The following is the status with respect to adapting our Computer codes to the PC/AT:

REX1-87: This code which is meant for calculating unshielded multigroup cross sections from a reconstructed evaluated data library has been adapted, and tested with a sample problem.

REX2-87: This code which is meant for calculating the self-shielding factors for various input dilutions from the Doppler broadened point data at a particular temperature, in the resolved resonance region, has been adapted and tested.

REX3: This program which uses the UNRESR module of NJOY to obtain energy dependant self shielded cross sections for speficied temperatures and dilutions and calculates multigroup self-shielding factors for those temperatures and dilutions, in the unresolved resonance region, has been adapted and tested.

COMREX2: This program which compares two sets of infinite dilution cross sections, self shielded cross sections and Doppler changes from the 25 group (Cadarache Type) library has been adapted.

LISTCOMP: This is a utility program that would list the identification numbers of the available materials in the multigroup set along with the length of space occupied by each of them. This has been adapted.

EFFCROSS: This code which prepares the composition dependent cross sections for further neutronic calculations has been adapted. The formatted (card image) Cadarache sets of Part-1 and Part-2 multigroup data were brought to the PC along with the program for recasting the data in the required binary form. The binary data was created and the code was tested with a sample problem of ZPR-6-7. The results obtained on the PC and on the Honeywell system for this case were found to agree.

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