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PROGRESS REPORT ON NUCLEAR DATA ACTIVITIES IN INDIA
FOR THE PERIOD FROM
JULY 1987 TO DECEMBER 1988

Compiled by

R. P. Anand
Nuclear Physics Division

1989

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GOVERNMENT OF INDIA
ATOMIC ENERGY COMMISSION

PROGRESS REPORT ON NUCLEAR DATA ACTIVITIES IN INDIA FOR THE PERIOD JULY 1987 TO DECEMBER 1988

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BHABHA ATOMIC RESEARCH CENTRE
BOMBAY, INDIA

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B.A.R.C. - 1464

INIS Subject Category : A34.00

Descriptors

NEUTRON REACTIONS

ALPHA REACTIONS

CROSS SECTIONS

NUCLEAR MODELS

EXPERIMENTAL DATA

COMPUTER CODES

RESEARCH PROGRAMS

INDIA

PROGRESS REPORT

PREFACE

The present progress report on Nuclear Data activities in India is the sixth 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 July, 1987 to December 1988. 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.

Most of the write-ups in this report describe the work 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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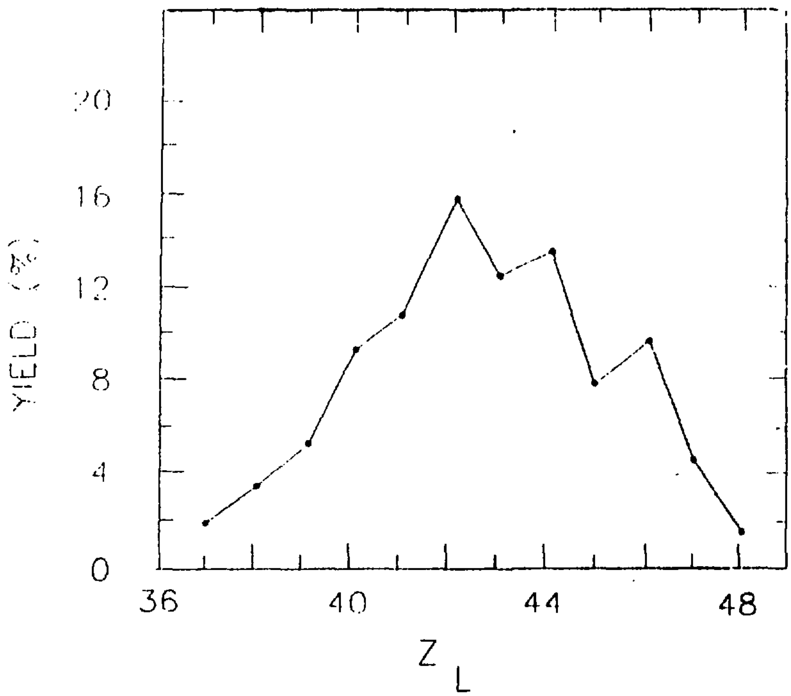
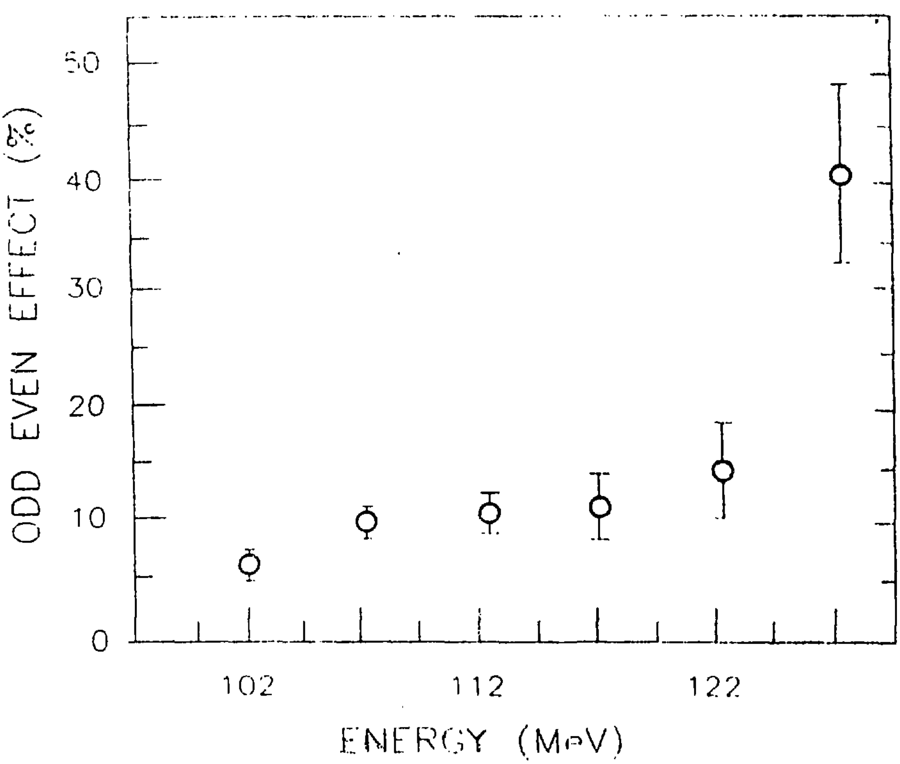
CONTENTS

Odd-Even Effects in Fragment Charge and Kinetic Energies in Fission of ^{252}Cf

M.N. Rao, R.K. Choudhury and D.C. Biswas
Nuclear Physics Division, B.A.R.C., Bombay 400 085.

The nuclear structure effects such as shell and pairing effects play a dominant role in determining the fission fragment characteristics in low energy nuclear fission. In order to study these, the fragment charge and kinetic energy measurements were carried out using a DE-E gas detector telescope, consisting of a gas ionisation chamber and a surface barrier detector. The ^{252}Cf source was mounted inside the detector and the chamber was filled with P-10 gas. A total of 8×10^6 events of DE-E_{Res} from the detector telescope were collected. The response function of the telescope for different fragment Z and E was determined by carrying out separate measurement of the E-E data in coincidence with a x-ray detector to record the K-rays from fission fragments. Using the data on the most probable energy loss as a function of Z and E of the fragments, an event by event analysis was carried out to obtain fragment charge from the full DE-E data. A charge resolution of 1.8 (FWHM) was achieved for all fragment charges and kinetic energies. Fig.1 shows the charge distribution of fragments for all the events. The odd-even effect, defined as the difference in the even and odd charge yields normalised to the total yield ($\Sigma_e - \Sigma_o / \Sigma_e + \Sigma_o$) is seen to be about 0.12 for all the events. The variation of with the kinetic energy of fragments is shown in Fig.2. It is seen that there is a slow variation of with energy, but at very large kinetic energy corresponding to cold fission events, increases sharply. The average kinetic energy did not have any noticeable fluctuations for odd and even charges.


YIELD (%)
20
16
12
8
4
0
36
40
44
48
Z_L


ODD EVEN EFFECT (%)
50
40
30
20
10
0
102
112
122
ENERGY (MeV)

MEASUREMENT OF NEUTRON EMISSION SPECTRUM IN THERMAL NEUTRON INDUCED FISSION OF ^{235}U

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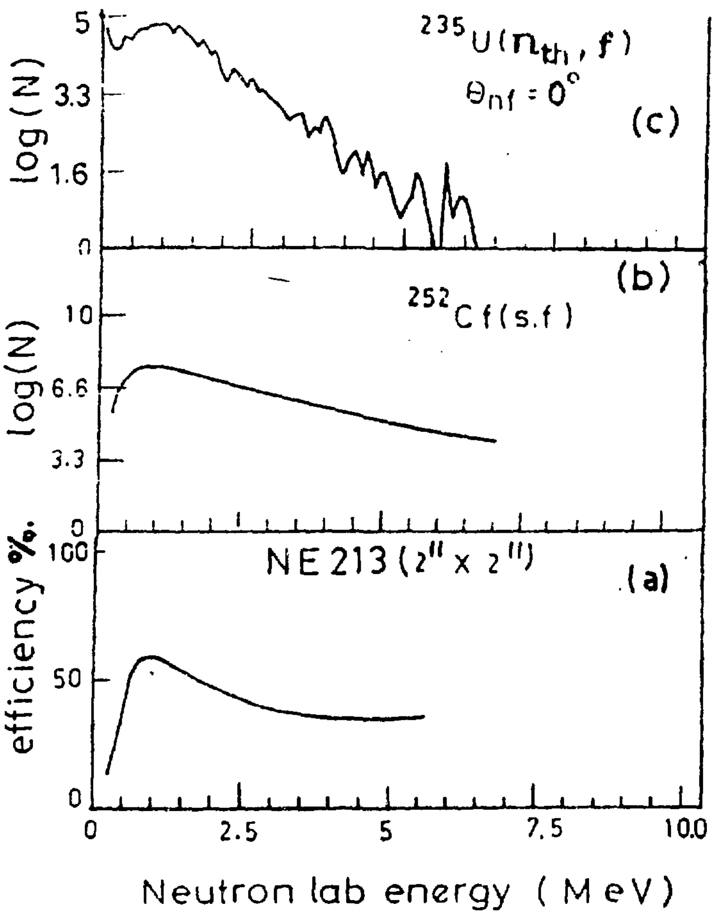
Nuclear Physics Division
Bhabha Atomic Research Centre, Bombay 400 085.

The energy spectrum of prompt neutrons emitted in fission contains information on the statistical properties of the excited fission fragments. It is possible therefore to determine the level density parameters from the analysis of neutron energy spectra. Although there exist many measurements of neutron emission characterstics in thermal neutron induced fission of ^{235}U, systematic determination of the level density parameters as a function of fragment mass has not been carried out. We have taken up studies to measure the neutron energy spectra at 0 deg with respect to the fragment direction in thermal fission of ^{235}U in order to experimentaly determine the level density parameter.

The experiment is carried out using the thermal neutron beam from the 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 detect the fission fragments. A (2" x 2") NE213 liquid scintillation detector was placed at a distance of 64 cms from the target, in the direction of the fission detectors to detect the fission neutrons. The n-γ separation was done by the pulse shape discrimination method using the zero crossover technique. The two fragment energies and the neutron time of flight with respect to the fragment detection were recorded in coincidence.

The efficiency of the neutron detector was independently determined by measuring the neutron energy spectrum of ^{252}Cf and using the known spectral shape avaliable from the earlier measurements as standard. Fig 1(b) shows the measured neutron spectrum in ^{252}Cf fission and Fig. 1(a) gives the neutron detection efficiency of the NE213 detector. The measured detector efficiency agreed quiet well with a Monte-Carlo stimulation corresponding to the electron equivalent energy threshold of 50 keV. The efficiency corrected spectrum in ^{235}U fission measured at 0 deg with respect to the fission fragment direction is shown in Fig. 1(c). This spectrum corresponds to all fission fragments and further analysis is in progress to obtain the neutron spectra as a function of mass and kinetic energy of the fragments. The analysis procedure to obtain the temperature and the level density parameter from the neutron energy spectra will be presented and discussed.

Reference 1. D.G.Madland and J.R.Nix, in Proc. Conf. on Nucl., data for Sci. & Tech., Antwerp, Belgium, Sept., 6-10(1982). p473.



$^{235}U(n_{th}, f)$
$\Theta_{nf} = 0°$
(c)
log (N)
5
3.3
1.6
0
(b)
$^{252}Cf(s.f)$
log(N)
10
6.6
3.3
0
NE 213 (2" x 2")
(a)
efficiency %
100
50
0
0
2.5
5
7.5
10.0
Neutron lab energy (MeV)

Fragment Mass and Energy Correlations in Fission of ^{235}U by Thermal Neutrons

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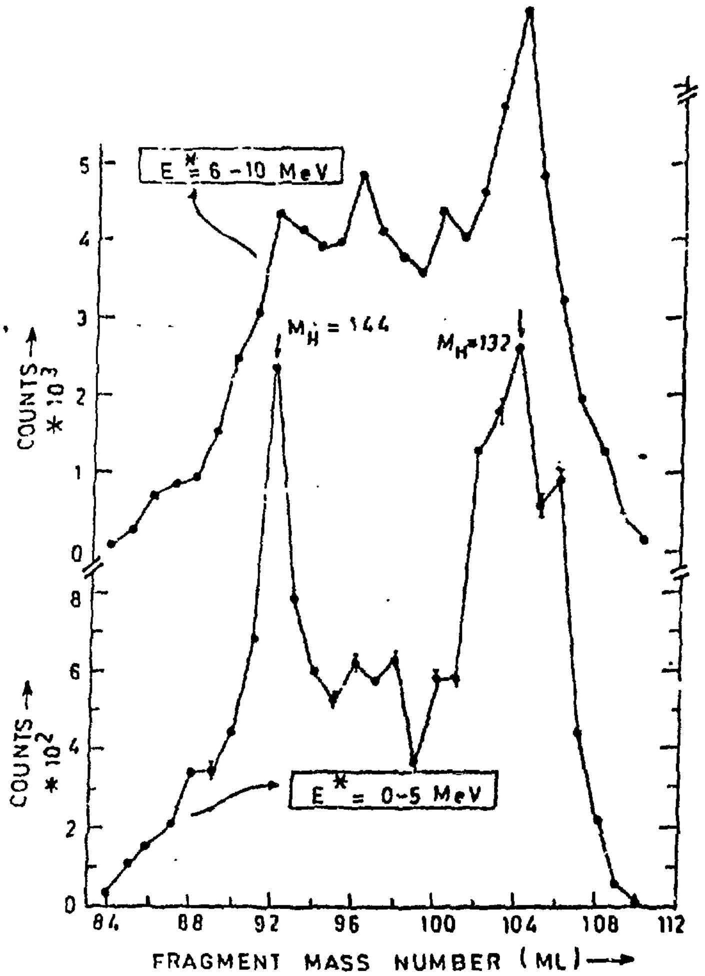
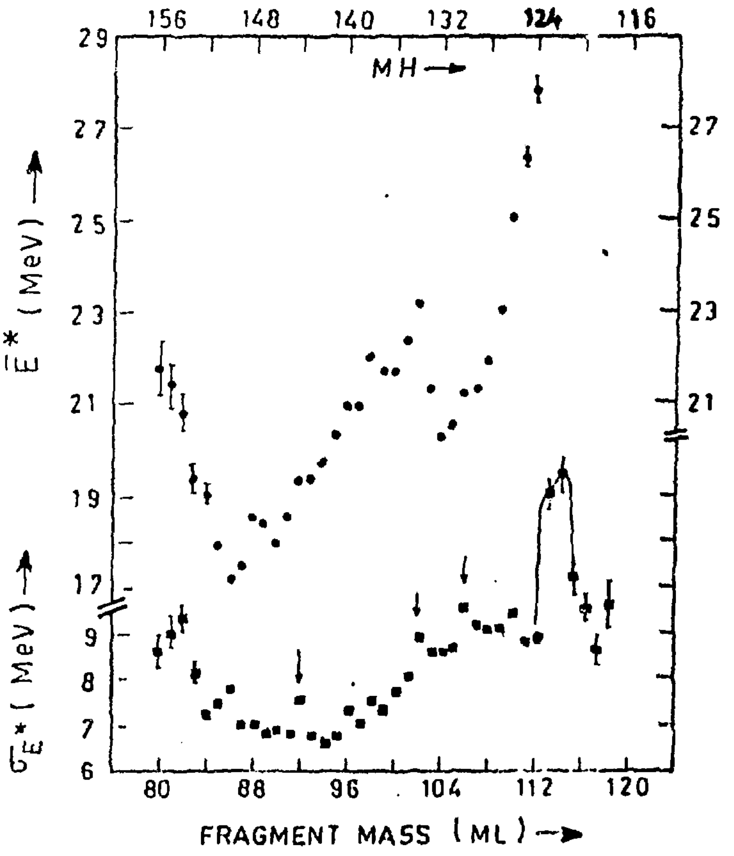
In the fission of heavy nuclei, a large amount of energy is released which appears in the form of kinetic and excitation energy of the fragments. The division of energy in these two forms contains significant information on the dynamics of the fission process. Studies on the variation of the excitation energy distribution with fragment mass can provide information on the effect of nuclear structure and dynamics governing the fission process. Measurements of mass and energy correlations were carried out using a back-to-back gridded ionisation chamber containing a thin ^{235}U target as the common cathode. The pulse heights from the two collectors (V_c), grids (V_g) and time difference between grid and collector pulses (T_{gc}) were recorded. The angle of fission fragments with respect to the electric field direction was obtained independently by analysing the V_g and T_{gc} pulses, and angle resolution was found to be 2°-3° in both these methods. The fragments were electronically collimated perpendicular to the target foil (60° cone) in order to eleminate the degraded events in the chamber, thereby improving significantly the mass resolution of the chamber. Analysis was carried out to obtain excitation energy event by event by using Q-value distributions for each mass. Fig.1 shows the variation of the average and width of the excitation energy distribution as function of fragment mass. It is seen that the average excitation energy shows minima around mass numbers M_H=132 and 148 amu. The width in the excitation energy distribution increases sharply around mass M_H=122 amu (M_L= 114 amu), which was found to be due to presence of two peak structure in the distribution. The mass distribution of the light fragment group for the extreme cold fission events is shown in fig.2. It is found that the yields of M_L=104 amu (M_H=132 amu) and M_L= 92 amu (M_H=144 amu) are enhanced for E^* 5 MeV indicating that these two modes can be formed at extremely low excitations. The structure at 132 amu corresponds to the doubly closed spherical shell and the structure at 144 amu is due to deformed shell for Z=56 and N = 88. This result is striking since the deformed shell has to be at energies close to the ground state energy, and theoretical calculations are required to see the effects of shells in this region.


E* = 6 - 10 MeV
E* = 0-5 MeV
M_H = 144
M_H = 132
COUNTS → * 10^3
COUNTS → * 10^2
FRAGMENT MASS NUMBER (ML) →


MH →
$\bar{E}^*$ (MeV) →
σ_{E^*} (MeV) →
FRAGMENT MASS (ML) →

RADIATIVE CAPTURE OF FAST NEUTRON IN Mn-55

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The precise knowledge of radiative capture cross section is useful in the study of reaction mechanism, design of fusion reactor and in extracting the useful information about nuclear structure. For this purpose a complete knowledge of capture cross-section is required. For the same purpose we have measured the cross-section for Mn-55 from 450-3450 KeV. We hope that our result would add a step in the betterment of earlier results.

The production of neutrons, their precise energy, energy-spread, photopeak detection efficiency and method of measurement are discussed in details elsewhere/1/. The specpure sample is used in the present work. Iodine is used as standard in the present measurement. The compound nucleus formed as a result of neutron bombardment has no isomeric state. The ground state has a half life of 2.578h and the characteristic gamma ray of 847 keV (98.9%). Iodine-128 emits 443 keV(16%) gamma ray. These rays were picked up with pre-calibrated 100 cc lead shielded Ge(Li) detector. The sample and standard were put in the same geometry at the time of irradiation and counting. The cross-sections have been calculated using a formula from lit./2/. The required photopeak counts were taken from the recorded spectrum for a specific time.

The comparision of measured values has been done with lit. values/3-9/ in fig. 1. It shows that the present cross-section is higher than that of Johnsrud et. al./3/ in lower energy region while a good agreement is seen at higher energies with values of Peto et. al./8/, Colditz et. al./7/ within experimental errors. The possible reason for the observed discrepancies in the results may be due to different monitoring reactions and technique used by different group of workers.

The measured cross-sections have also been compared with theoretically evaluated one using FISPRO II code. The best fit parameters are used. These are target spin=5/2+, B.E=7.2702 MeV, Do=2105 eV, Γ_γ=800mV, a(c)=6.45/MeV, a(t)=5.45/MeV. Both theoretical and measured cross-sections are compared in fig. 1 along with other lit. values/3-9/. It has been observed that the contribution from direct and collective capture processes is not appreciable upto 3.0 MeV. It exhibits good agreement between calculated and measured values and confirms the validity of statistical theory in the reproductions of the experimental results.

The authors are grateful to late Prof. M.L.Sehgal, ex-Chairman Deptt. of Physics, A.M.U Aligarh. His inspiration makes this work a success. We are also thankful to Dr. S.S. Kapoor for providing us the Van de Graaff facilities for this work.

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** Physics Deptt., Aligarh Muslim University, Aligarh.
*** Nuclear Physics Division, B.A.R.C., Trombay (Bombay).

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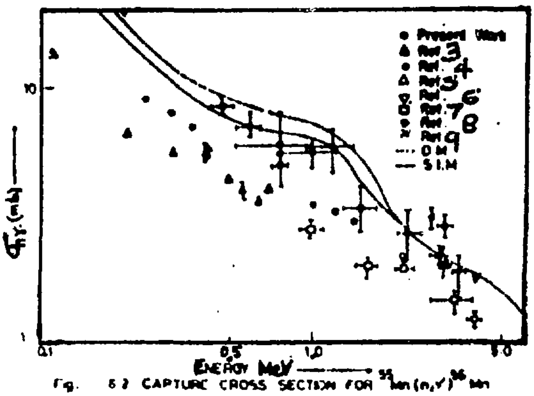

Fig. 6.2 CAPTURE CROSS SECTION FOR $^{55}Mn(n,\gamma)^{56}Mn$

MEASUREMENT OF NEUTRON EMISSION IN $^{238}U(\alpha, f)$ AT MODERATE EXCITATION ENERGIES

Alok Saxena, D.M.Nadkarni, R.P.Anand,
S.R.S.Murthy, R.K.Choudhury, V.S.Ramamurthy and
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There has been recent interest[1-3] in the study of neutron emission in fission since it can provide important information on the mechanism of fission process and on the saddle to scission dynamics. Neutron multiplicities corresponding to pre-scission and post-scission emission can be determined by the neutron spectra in coincidence with fission fragments at different angles. There have been anomalously large precission neutron multiplicities observed in heavy ion induced reactions which has been interpreted to be due to large saddle to scission transit times[1-2]. There exists very little information on the entrance channel effects and also neutron emission in light ion induced fission reactions. In the present work we have carried out measurements of neutron spectra in α-induced fission of ^{238}U at 40 and 50 MeV bombarding energies at two different angles with respect to fission direction. A liquid scintillator (NE213) of 2" dia and 2" thick was used to detect the neutrons. The neutron detector was located 88 cm away from target and at 30° with respect to the beam direction. Two fission fragment detectors were mounted at 0° and 90° with respect to the neutron detector for measuring fission fragment energies. The time of flight of neutrons was measured with respect to zero time corresponding to the fission fragment detection. Neutron time of flight spectra measured in 0° with respect to fission fragment for α-bombarding energy of 40 MeV is shown in fig. 1. Lab neutron energy spectra for 0° and 90° are shown in fig. 2. These energy spectra at 0° and 90° were fitted simultaneously using watt spectrum consisting of three different evaporation sources. The temperatures for post and pre-scission components were found to be 1.24 and .853 MeV respectively and the prescission multiplicity accounted or 40% of the neutrons emitted in coincidence with fission.

1. D.J.Hinde et. al. Phys. Rev. C37 (1988) 2923
2. A.Gavron et. al. Phys. Rev. C35 (1987) 579
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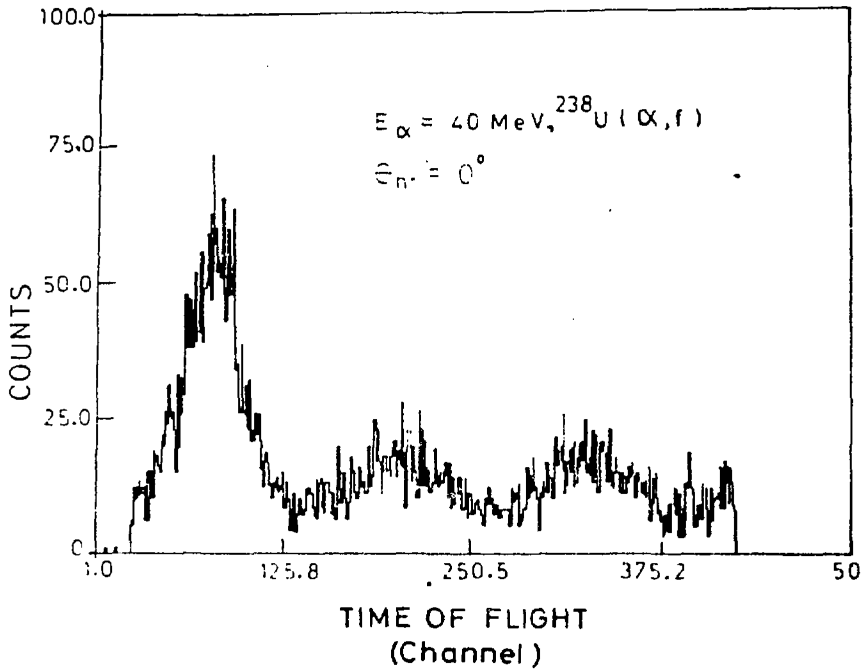
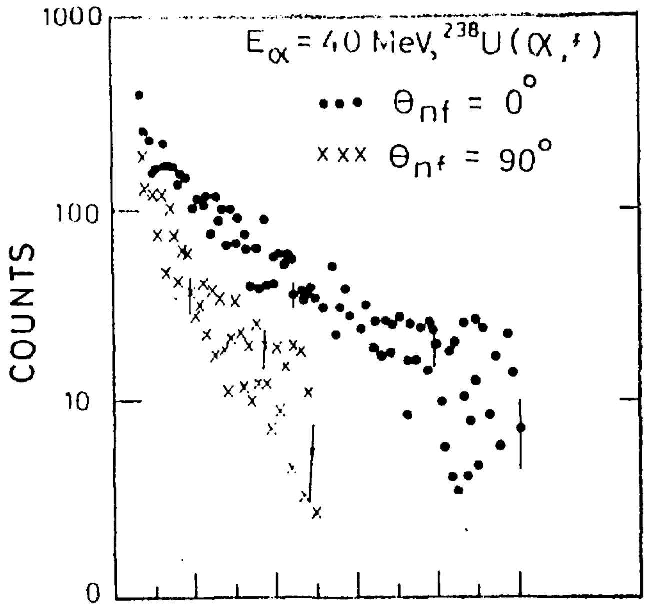


E_α = 40 MeV, ^{238}U (α, f)
θ_n = 0°
COUNTS
100.0
75.0
50.0
25.0
0
1.0
125.8
250.5
375.2
50
TIME OF FLIGHT
(Channel)

FIG. 1



E_α = 40 MeV, ^{238}U (α, f)
••• θ_{nf} = 0°
xxx θ_{nf} = 90°
COUNTS
1000
100
10
0
0
2
4
6
8
10
E_n

FIG. 2

ALPHA-INDUCED FISSION OF ^{235}U AT EXTREME SUB-BARRIER ENERGIES

N. N. Ajitanand, K. N. Iyengar, R. P. Anand, and D. M. Nadkarni
Nuclear Physics Division, Bhabha Atomic Research Centre, Trombay, Bombay 400085, India

The alpha-induced fission cross section for ^{235}U has been measured for E = 3,4 and 12 MeV for the first time. Taken together with the already published data for protons at extreme sub-barrier energies, these results suggest that the observed cross section below 5 MeV alpha energy may be due to fission of low-lying target states populated by Coulomb excitation. However, it is not clear how such states are able to have the required penetrability of the fission barrier, suggesting that an explanation of these results may lead to a better understanding of deep sub-barrier tunnelling, especially with regard to the fission process.

The measured cross sections have been shown in figure 1. The cross sections at 3 and 4 MeV were measured at Van de Graaff accelerator at B.A.R.C abd the 12 MeV data was taken at Cyclotron, Calcutta by reducing the energy of 25 MeV alphas by passing the beam through thin gold foils. The cross-sections were measured using Lexan track detectors where each fission event is recorded as a conical track. Fission events caused by backgound neutrons were also measured very close to the ^{235}U target and suitable corrections for the same, were taken into account for calculating the cross sections.

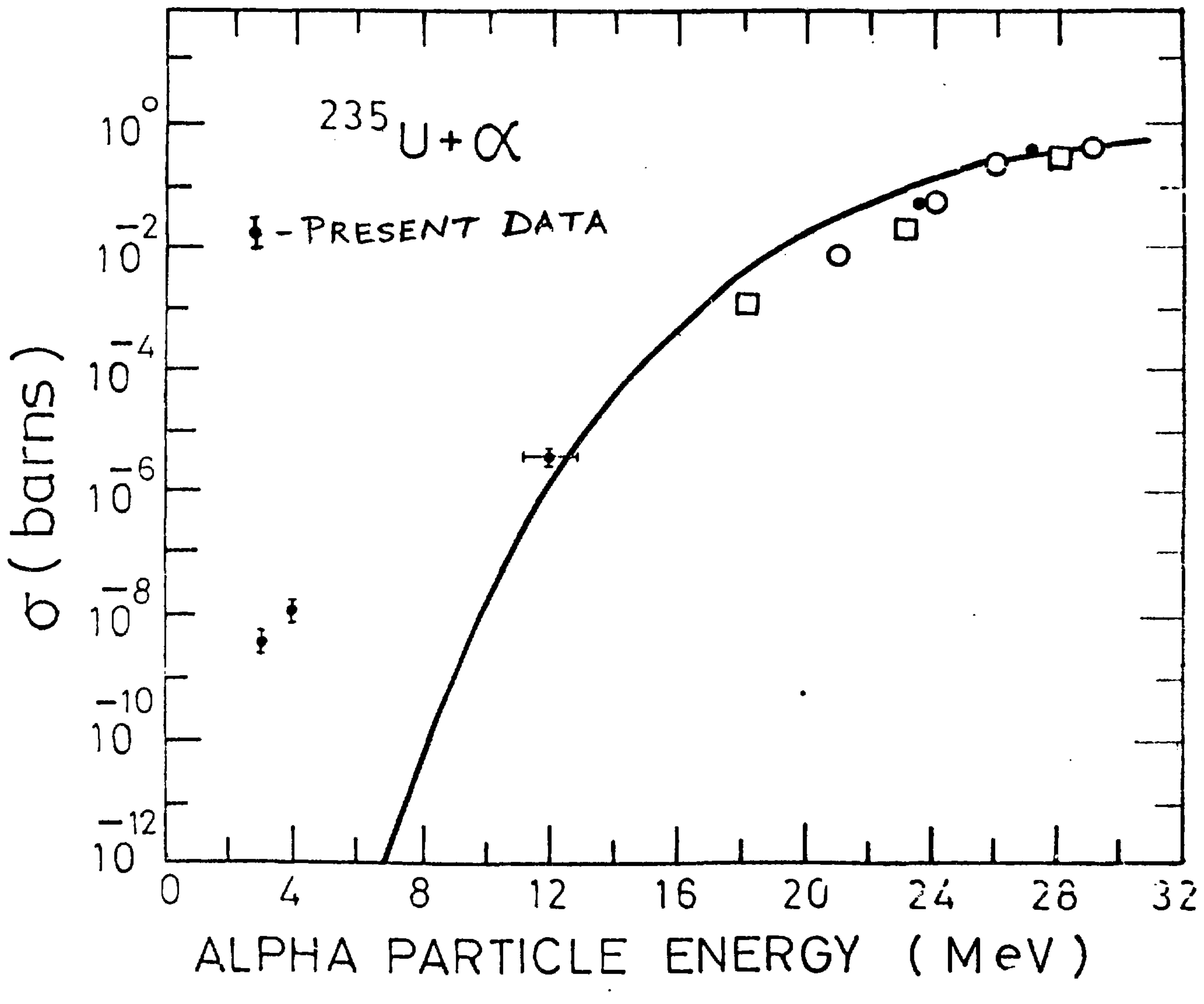

$^{235}U+\alpha$
- PRESENT DATA
10^{0}
10^{-2}
10^{-4}
10^{-6}
10^{-8}
10^{-10}
10^{-12}
σ (barns)
0
4
8
12
16
20
24
28
32
ALPHA PARTICLE ENERGY (MeV)

FIG. 1

MEASUREMENT AND ANALYSIS OF EXCITATION FUNCTIONS FOR Nb(α,xn) REACTIONS

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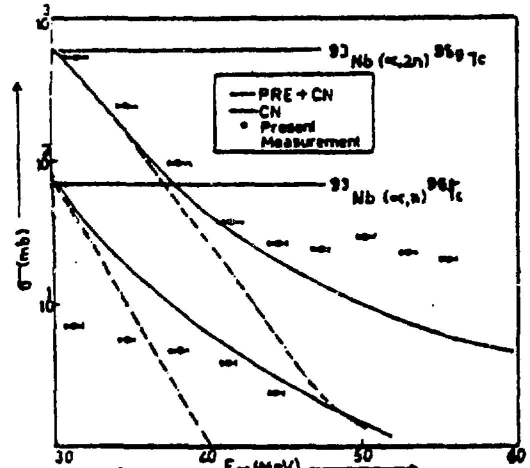
Department of Physics, A.M.U. Aligarh

Study of excitation functions via alpha-induced reaction provide valuable information on the pre-equilibrium emission of particles. Alpha-induced reactions in ^{93}Nb have been investigated in the energy range $\simeq$ 30-60 MeV using the stacked foil technique. Spectroscopically pure thin foils of 10.71 mg/cm^2 have been used for the irradiation. The stack was made of ten such foils using energy degradors in between foils. The stack has been irradiated with an alpha beam of energy $\simeq$ 58.32 MeV. The beam spot on the stack was kept to 10 mm in diameter using a tantalum collimator infront of the stack. The beam current on the target was $\simeq$ 200 nA. The alpha particle energy loss in the sample and the degrador thickness has been calculated using the stopping power table of Northcliffe et. al[1]. The average incident flux calculated from the total charge collected in the Faraday cup was found to be $\simeq 10^{12}$ α-particles/sec/cm^2. The post - irradiation analysis has been done using a high resolution Ge(Li) detector of 100 c.c. coupled with the multichannel analyser CANBERRA 88. The excitation functions for the production of radio-nuclides Tc^{96g}, Tc^{95g},Tc^{94g}, and Tc^{93g} have been measured in the above energy range.

The excitation functions for these reactions have been calculated theoretically with and without the inclusion of pre-equilibrium (PE) emission in the framework of hybrid model of Blann using the computer code ALICE/LIVERMORE 82[2]. In fig. 1 the experimentally measured and theoretically calculated excitation functions for the reactions (α,xn) (x=1-4) have been shown. A reasonable agreement between the theory and the experiment has been observed for the initial exciton number equal to 6 in the present calculations. The preequilibrium fraction FR, which is the measure of relative weight of PE componant has also been calculated and it is observed that FR increases very rapidly with energy.

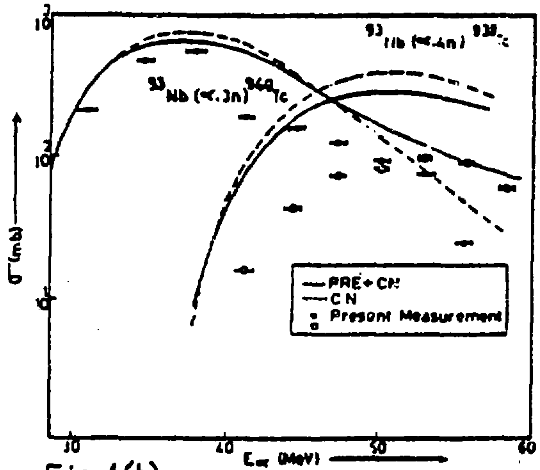
Ref. (1) L.C.Northcliffe and R.F.Schilling, Nuclear Data Tables A7 (1970)256

(2) M.Blann and J.Bisplinghoff, Lawrence Livermore Laboratory report UCID-19614(1982)



93Nb (α,2n) 95g Tc
PRE + CN
CN
Present Measurement
93Nb (α,n) 96g Tc
σ (mb)
Eα (MeV)

Fig.1 (a)



93Nb (α,3n) 94g Tc
PRE + CN
CN
Present Measurement
σ (mb)
Eα (MeV)

Fig.1(b)

CROSS SECTION MEASUREMENTS OF ALPHA INDUCED REACTIONS ON 69-Ga AND 71-Ga.

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Cross sections for the reactions 69-Ga(α,xnyp) x$\leq$ 4,y $\leq$ 2 and 71-Ga(α,xn) x$\leq$ 4 have been measured from threshold to 65.0 MeV using stacked foil technique to provide mostly new data. The gallium targets were obtained by vacuum evaporation of gallium nitrate on to 23.4 μm thick aluminium backings placed over the masking plates. The thickness of the targets was = 0.5 mg/cm^2. The stacks were formed by placing aluminium degraders (23.4 μm or 4.34 μm thick) over the targets. Fig 1 depicts the total residual production cross sections for the reactions (i)69-Ga(α,n)72-As symb (■), N=0.92 (ii) 69-Ga(α,2n) 71-As symb (▲) N=1.12 and (iii)69-Ga(α,p3n)69-Ge symb (▼) N=0.69 as a function of α -particle bombarding energy. Solid lines are the hybrid model calculation using the code Alice and normalized to the experimental data. N is the normalization factor. The dominant mechanism for reactions induced by alpha in the energy range of 10 MeV to 65.0 MeV of this work is fusion of the projectile and the target nuclei followed by decay of the resulting compound nucleus by statistical evaporation. We have attempted to compare the observed cross section with those predicted by such a model. The calculation were performed using the α oriori calculation method with the computer code ALICE. For all product nuclei we have taken exciton number n=4 with nn=2 and np=2. The intra-nuclear transition rates were calculated using imaginary optical potential. For a representative set of product nuclei the comparison is depicted in figure 1. Qualitatively, the model is able to reproduce the observed shapes. Quantitatively, the magnitude of the calculated cross sections are except for a few cases, within a factor of 2 as indicated by factor N and often closer than that, to the observed cross sections over the entire energy range. Given the approximation inherent in the code, especially the limited treatment of the angular momentum dependent effects and neglect of the contributions from non-fusion process the agreement is as good as could be expected. Overall these comparisons confirm rather quantitatively that fusion evaporation is the dominant mechanism.

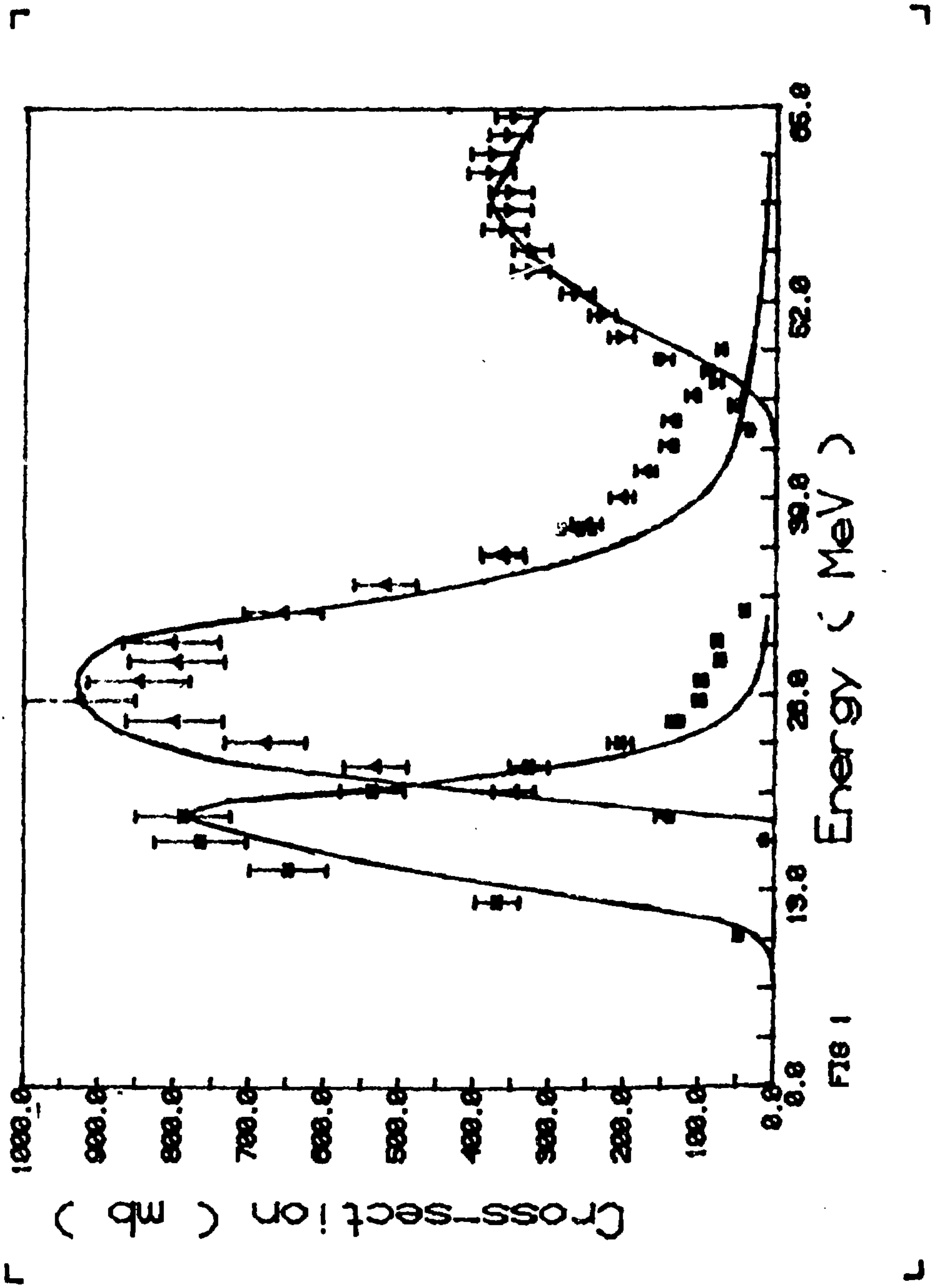
Cross-section (mb)
Energy (MeV)
0.0
100.0
200.0
300.0
400.0
500.0
600.0
700.0
800.0
900.0
1000.0
13.0
26.0
39.0
52.0
65.0

FIG 1

EXCITATION FUNCTIONS FOR THE HELIUM ION INDUCED FISSION OF Ho-165

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As part of a long range programme of work on the fission excitation functions of low Z-elements ($Z < 90$) with Helium ions, the fission cross-sections of Ho-165 ($Z = 67$) in the energy range 40 MeV - 60 MeV have been measured[1,2]. The fission events were recorded using lexan plastic track detector. Targets were prepared from highly purified holmium oxide in the form of thin coating of the oxide by Electrophoresis onto super pure Ag foils. Preliminary values for the fission cross-sections of Ho-165 are as follows:

Energy (MeV)	σ_f (nanobarns)
40	≤ 1
45	≤ 3
60	800 ± 25

Further work is in progress to get the complete excitation function in the energy range 30 - 80 MeV and also to extend these studies to other rare earths in order to study fission barrier systematics in the highly deformed region of rare earths nuclei.

1. P.C. Kalsi, R.C. Sharma, A.K. Pandey and R.H. Iyer, "Fission cross-section of Ho induced by 60 MeV He ions", DAE Symposium on Radiochemistry and Radiation Chemistry, IGCAR, Kalpakkam (1989).

2. A.K. Pandey, R.C. Sharma, P.C. Kalsi and R.H. Iyer, "Excitation functions for the Helium ion induced fission of Holmium", Paper presented at the Sixth Seminar on Solid State Nuclear Track Detectors: Application to Geochronology and Micro-analysis, Gauhati University, Guwahati, March 16-18 (1989).

A GEOMETRY DEPENDENT HYBRID MODEL BASED STUDY OF ANGLE-ENERGY CORRELATED DOUBLE DIFFERENTIAL CROSS-SECTIONS: S.B. Garg, Neutron Physics Division, Bhabha Atomic Research Centre, Trombay, Bombay 400 085

Anisotropy of scattered neutrons is to be accounted for in reactor physics and shielding calculations. Anisotropy becomes more pronounced in fast reactor and fusion based systems where nuclear interactions are caused by energetic neutrons in the MeV energy range. In such systems correct predictions of multiplication factor, flux distribution and power profiles can be made if angle-energy correlated double differential cross-sections for the secondary neutrons are known. The measured information on this type of data is rather scanty and thus nuclear models are employed to generate these data.

In the frame work of preequilibrium-statistical models three approaches are available to compute double differential cross-sections. The first approach is due to Costa et al/1/ and it makes use of the fast particle concept of the generalized exciton model of Mantzouranis et al/2/. The second approach is that of Kalbach and Mann/3/ who suggested a phenomenological expression based on the multistep direct and multistep compound concepts of Feshbach et al/4/. The third recipe is due to Blann et al/5/ and is based on the nucleon-nucleon scattering concept of Hayakawa et al/6/. In this approach the single nucleon-nucleon scattering kernel is modified by folding in multiple scattering fractions obtained with the geometry dependent hybrid model. Refraction effects in the entrance and exit channels, geometric information of local density variation and energy dependence of single particle level density have been included in the considerations.

In this paper the last approach based on the geometry dependent hybrid model has been employed to calculate the angle-energy double differential cross-sections of the secondary neutrons for Cr-52 and Fe-56. The results are shown in Figs. 1 & 2 at scattering angles of 60° and 120° for 14.1 MeV and 18 MeV of neutron incident energies. It is noted that the calculated double differential cross-sections are within the measured uncertainties. Some deviations are, however, noted for the high energy neutrons for the simple reason that such neutrons are emitted in direct reactions which are not adequately represented in these formulations.

These investigations indicate that the geometry dependent hybrid model may be employed to generate double differential cross-sections for technological applications.

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Acknowledgement.
I am thankful to Dr. S.S. Kapoor for many helpful discussions.

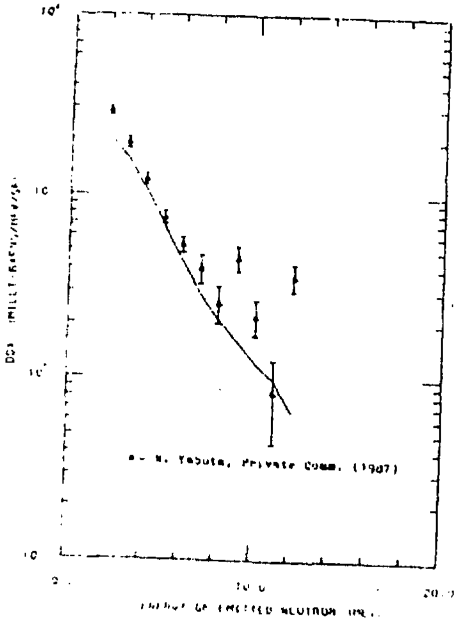

FIG. 1 : DOUBLE DIFF. CROSS-SECTION OF [illegible] AT 14.[illegible] MEV
ANGLE = 40 DEG.

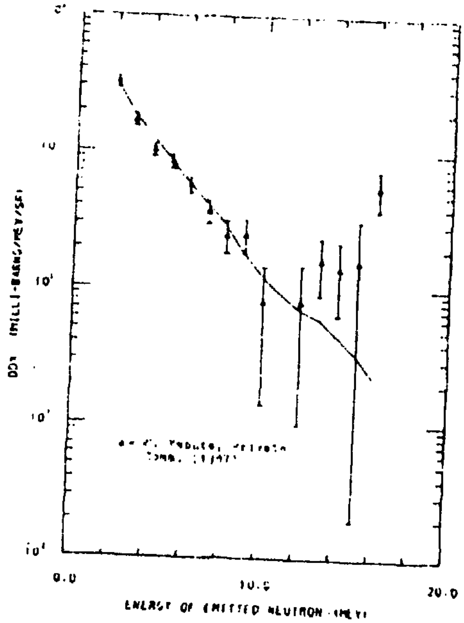

FIG. 2 : DOUBLE DIFF. CROSS-SECTION OF [illegible] AT 15.0 MEV
ANGLE = 120 DEG.

A COMPARATIVE STUDY OF NUCLEAR MODELS WITH SPECIAL REFERENCE TO NEUTRON CROSS-SECTIONS OF STRUCTURAL ELEMENTS: S.B. Garg, Neutron Physics Division, Bhabha Atomic Research Centre, Trombay, Bombay 400 085.

Binary, tertiary and multiparticle neutron induced reaction cross-sections are needed for several elements in reactor technology in order to estimate the neutronics, safety and shielding characteristics of fusion and fission based reactor systems. The measured cross-section data for several types of reactions are generally not available in the entire energy range extending upto 20 MeV and thus the desired information is generated with appropriate nuclear models. In this paper we have investigated the following three data evaluation model schemes with a special reference to neutron cross-sections of Cr-52 and Fe-56 which are constituents of stainless steel - a frequently used structural material in nuclear reactors:

(i) Multistep Hauser-Feshbach (MSHF) scheme comprising Hauser-Feshbach statistical theory, Brink-Axel Model, optical model, Kalbach's exciton and empirical direct reaction models.

(ii) Geometry Dependent Hybrid Model (GDHM) scheme consisting of geometry dependent hybrid model, Weisskopf-Ewing evaporation model and optical model

(iii) Unified Exciton Model (UEM) scheme which makes use of unified exciton model, Brink-Axel and optical models.

With these schemes we have computed (n,n'), (n,p), (n,α), (n,2n),(n,np),(n,pn),(n,nα), (n,γ) and total production cross-sections for neutrons, protons, α-particles and gamma-rays together with the energy spectra of the secondary emitted particles for Cr-52 and Fe-56 in the energy range 1 to 20 MeV. We have observed that the MSHF scheme reproduced the measured cross-section data rather well compared to the other two schemes. Some of the salient features of the intercomparison are given below:

(a) All the three schemes predict total neutron emission and (n,2n) cross-sections within 25% of one another as shown in Fig.1. GDHM scheme being the simplest one with fewer free parameters may be employed for quick estimates of the above noted reactions.

(b) The charged particle emission cross-sections predicted by the three schemes differ, sometimes by factors as shown in Fig.2.

In this analysis no attempt is made to tamper with the original level density and pairing energy systematics used in these schemes. However, in order to bring out the effect of these parameters on computed cross-section data; we used the same level density and pairing energy correction parameters in all the schemes and observed the following:

(i) (n,2n) and neutron emission cros-sections follow similar trends as indicated in the base case i.e. these cross-sections are predicted within 25% of one another

(ii) Proton emission crosssections are increased in the MSHF and UEM schemes but these cross-sections are still lower than those given by the GDHM scheme

(iii) Alpha emission cross-sections are also increased in the MSHF and UEM schemes but this increase is more prominent in the UEM scheme.

In summary it can be stated that the MSHF scheme may be used for interpolation and extrapolation of the measured data. GDHM and UEM schemes may be employed to obtain quick approximate estimates of neutron emission cross-sections. Level density systematics affects the calculated crosssections. Systematics derived from experimental information of level densities would yield realistic estimates of the required data particularly in the MSHF scheme.

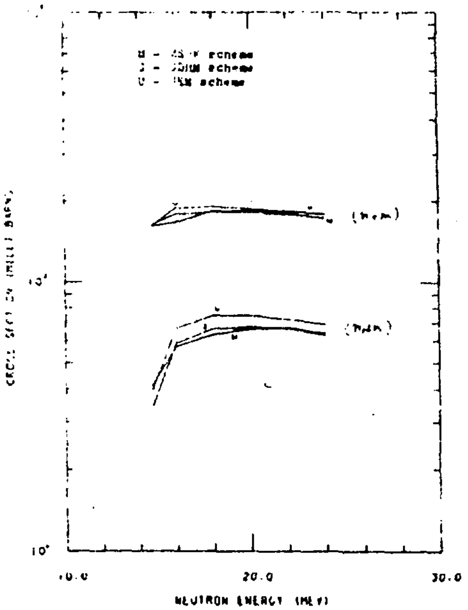

FIG. 1 : INTER-COMPARISON OF NEUTRON EMISSION AND (N,2N) CROSS-SECTIONS OF CR-52 WITH MSHF, GDHM, AND UEM SCHEMES

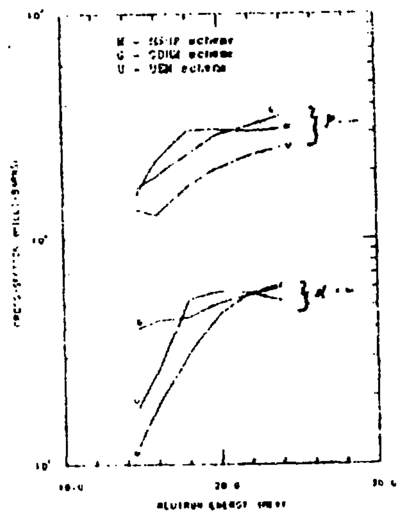

FIG. 2 : INTER-COMPARISON OF PROTON AND ALPHA EMISSION CROSS-SECTIONS OF CR-52 WITH MSHF, GDHM AND UEM SCHEMES

Determination Of Level Density Parameter By Analysing (N,2N) And (N,3N) Cross-Sections With Geometry Dependent Hybrid Model

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In this investigation the geometry dependent hybrid model has been used to calculate (n,2n) and (n,3n) cross-sections and angle-energy correlated double differential cross-sections for neutron emission. In this scheme the equilibrium component is computed with the multistep Weisskopf-Ewing evaporation model which employs the following Fermi gas formula to evaluate level densities of the nuclides taking part in a reaction chain:

$$\rho(U) \propto (U-\delta)^{-5/4} \exp\left[2\sqrt{a(U-\delta)}\right]$$

where a = level density parameter = A/X, A is the mass No. of the composite nucleus and X is a parameter.

U = excitation energy
δ = pairing energy correction

In this investigation the level density parameter 'a' of the evaporation model has been determined by simultaneously reproducing the measured (n,2n) and (n,3n) cross -sections of Sc-45, Co-59, Y-89, Nb-93, Rh-103, Tm-169, Lu-175, Ta-181, Au-197 and Bi-209. It has also been found that the value of 'a' determined in this investigation reproduces quite well the measured neutron emission spectrum at 14.1 MeV. The level density parameter 'a' is found to have the following systematics:

(a) For deformed nuclides belonging to rare earth region, a $\simeq$ A/13
(b) For nuclides approaching doubly closed shells a $\simeq$ A/20 and
(c) For other category of nuclides 'a' varies from A/8 to A/10, the most probable value being A/9
(d) The 'a' values determined in this study are found to be quantitatively correlated with the ground state shell correction energy.

It is known that in reactor technology accurate neutron induced cross-section data are desired for fuel, structural, coolant, control and shielding materials which span almost the entire chart of nuclides. The present work has been aimed at providing a computationally simple scheme of estimating the above listed cross-sections with acceptable accuracies for technologaical applications.

NEUTRON INDUCED MULTI-PARTICLE REACTION CROSS-SECTIONS OF CHROMIUM AND IRON : S.B. Garg, Neutron Physics Division Bhabha Atomic Research Centre, Trombay, Bombay 400 085.

This work has been carried out under the Co-ordinated Research Programme dealing with the methods for the calculation of fast neutron nuclear data for structural materials sponsored by the International Atomic Energy Agency /1/. The main objective of this programme is to develop and adopt nuclear models and methods for the computation of neutron induced reaction cross-sections of structural materials to be used in fission and fusion based reactor systems.

Chromium and iron are two main constituents of stainless steel which is commonly used as a structural material. Binary, tertiary and multiparticle reaction cross-sections; energy spectra of the secondary emitted neutron, proton, alpha-particle and gamma-rays; total production cross-sections for neutron, hydrogen, helium and gamma-rays; level inelastic cross-sections and radiative capture cross-sections have been generated for these elements in the energy range extending upto 30 MeV. Three nuclear model schemes, namely, the multistep Hauser-Feshbach (MSHF) scheme, geometry dependent hybrid model (GDHM) scheme and the unified exciton model (UEM) scheme have been employed in the investigations. It has been noted that the multistep Hauser-Feshbach scheme reproduces the measured cross-sections rather well with appropriate optical model potential parameters and it may be used for the extrapolation and interpolation of the measured data for the technological applications.

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MULTIGROUP CROSS-SECTIONS OF SPECIAL ACTINIDE ELEMENTS S.B. Garg Neutron Physics Division. Bhabha Atomic Research Centre. Trombay, Bombay 400 085.

35 group cross-sections with P_3-anisotropic scattering matrices and resonance self-shielding factors have been evaluated at several temperatures for special actinide elements such as Am 241, Am-242 m, Cm-242, Cm-243, Cm-245, Cm-247, Cf-249 and Cf-251 utilizing the ENDL/1/ basic cross-section library and the code MINX/2/.

The generated cross-sections are used to derive the criticality and safety data of the above noted elements.

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SYSTEMATICS OF CRITICALITY CHARACTERISTICS OF ACTINIDE NUCLIDES

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Considerable amounts of special actinide elements such as neptunium, americium, curium and californium are present in the accumulated spent fuel of nuclear power reactors operated with uranium or plutonium based fuels. Some of these nuclides have important applications in electric generators, neutron sources, space reactors and fusion reactor blankets. It is noted that all the isotopes of the above mentioned actinide elements irrespective of the fact whether these are fissible or fissile have favourable nuclear characteristics so as to render them suitable for fuelling the fast critical assemblies.

These characteristics are high fission cross-section, low capture cross-section and high $\bar{\nu}$ (average no. of neutrons released per fission) over a wide energy range of interest. High fission cross-section is also significant from the point of view of transmutation of the long lived radioactive isotopes to short lived ones by recycling and burning the so called 'nuclear waste' in power reactors.

Criticality characteristics of fast multiplying assemblies fuelled with the special actinide nuclides such as Np-237, Am-241, Am-242m, Am-243, Cm-243, Cm-245, Cm-247, Cf-249 and Cf-251 have been studied with a 35 group cross-section set which has been derived from ENDL/1/ library and are summarized in a recent paper /2/ together with the criticality data of uranium and plutonium based fuels. A number of interesting systematics relating K_{∞} and critical mass values with Z^2/A of the nuclide have been inferred.

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EVALUATION OF THE ^{232}Th TOTAL NEUTRON CROSS SECTION FROM 0.1 MeV TO 20 MeV

H.M. Jain, Experimental Reactor Physics Section

B. A. R. C.

As part of nuclear data evaluation for ^{232}Th - ^{233}U fuel cycle, the evaluation of ^{232}Th total neutron cross section is in progress. For this the computer program HAUSER-V was modified to calculate total cross section. The calculations were performed using the conventional spherical optical model parameters of Jary (6), MY (1), MY (2) and WH (4) from the TABLE I of Ref.1 . The calculations performed with Jary (6) and MY (2) SOM parameters are in good agreement with the recent experimental data. This exercise is done to select a suitable SOM parameters set for ^{232}Th to perform further calculations using HAUSER-V and to improve the evaluation based on model based calculations for interpolating the cross sections where measurements do not exit.

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Nuclear Data activities of Nuclear Chemistry Section, Radiochemistry Division, BARC for the period July 1987 to December 1988.

1. Nuclear Fission

For charge distribution studies in the spontaneous and helium ion induced fissioning systems, cumulative and fractional cumulative yields were determined for certain fission products. The values are reported in Table 1. For angular momentum of fission fragments, isomeric yield ratios of antimony isotopes were determined in the helium ion induced fission. The values are reported in Table 2. Fragment angular momentum of ^{132}I in different fissioning systems is shown in Table 3. Table 4 gives the results of our study on the effect of entrance channel parameters on fission fragment angular momentum in the medium energy fission. Table 5 gives the data on mass resolved angular distribution in helium ion induced fission. We have proposed a new approach to determine the elemental yield, charge polarisation and odd-even effects in fission. Table 6 gives the determined values of the elemental yield in the spontaneous fission of ^{252}Cf. The determined elemental yields are in good agreement with the literature data.

Table 1

Fractional Cumulative Yields in the spontaneous fission of helium ion induced fission

Sr.No.	Fissioning Nucleus	Fission Product	Cumulative Yield	Fractional Cumulative Yield
1.	^{252}Cf(S.F.)	^{139}Ba	6.08±0.21	0.9997
		^{141}Ba	5.66±0.1	0.9897
		^{142}Ba	5.41±0.1	0.9180
2.	^{252}Cf(S.F.)	^{137}Xe	4.11±0.13	-
		^{138}Xe	5.24±0.16	-
		^{139}Xe	3.86±0.29	-
3.	^{238}U(, f) 30 MeV	^{92}Sr	-	0.8812
		^{134}Te	-	0.8049
		^{135}I	-	0.8963
		^{142}Ba	-	0.9456

Table 2

Data on Isomeric Yield ratio and angular momenta

Fissioning system	Excitation Energy (MeV)	Initial $\langle L \rangle$ ($\hbar$)	Sb Isotopes	Isomeric yield ratio	Exp $\langle J \rangle$ $\hbar$	Calculated $\langle J \rangle$ $\hbar$
$^{241}Pu(n,f)$	6.2	4-5	^{128}Sb	0.53 ± 0.08	8.1 ± 1.2	
$^{238}U($ $,f)$	18.3	7.9	^{128}Sb	0.61 ± 0.02	9.9 ± 0.3	11.1
			^{130}Sb	0.52 ± 0.01	9.1 ± 0.2	10.6
	23.7	9.5	^{128}Sb	0.76 ± 0.08	12.3 ± 1.3	11.4
	24.8	11.0	^{126}Sb	0.88 ± 0.08	16.0 ± 1.0	13.7
			^{130}Sb	0.58 ± 0.01	9.9 ± 0.2	12.4
	30.2	12.9	^{126}Sb	0.80 ± 0.04	14.8 ± 0.8	14.8
			^{128}Sb	0.79 ± 0.02	12.9 ± 0.4	13.9
			^{130}Sb	0.67 ± 0.01	11.4 ± 0.2	13.5

Table 3

Fragment angular momentum of ^{132}I in different fissioning nuclei

Fissioning nucleus	Isomeric yield ratio	Fragment angular momentum
^{235}U	0.446±0.037	8.7±0.95
^{240}Pu	0.403±0.038	7.8±0.7
^{242}Pu	0.950±0.030	8.85±0.65
^{246}Cm	0.482±0.044	9.7±1.2
^{252}Cf	0.525±0.059	11.75±1.4

Table 4

Fragment angular momentum of ^{132}I in $^{238}U(\alpha, f)$

Projectile energy	I of the Fissioning nucleus	Isomeric yield ratio ($Y_m / Y_m + Y_g$)	RMS angular momentum ($\hbar$)
25.2	6.8	0.586 ± 0.076	11.1 ± 1.3
27.0	7.8	0.682 ± 0.055	13.2 ± 1.3
33.1	10.6	0.756 ± 0.051	15.5 ± 1.6
39.1	12.8	0.762 ± 0.045	15.6 ± 1.5
44.2	13.7	0.780 ± 0.035	16.5 ± 1.5

Table 5

Fission product angular anisotrppy in $^{238}U(\alpha_{40 MeV}, f)$

Fission product	Angular anisotropy (W(0)/W(90))	K_0^2 ($\hbar$)
^{99}Mo	1.59	72
^{117}In	1.27	87
^{126}Sb	1.29	95
^{128}Sb	1.37	65
^{132}Te	1.54	79
^{134}I	1.80	33
^{135}Xe	1.76	34
^{143}Ce	1.56	45

Table 6

Determined elemental yields in the spontaneous fission of ^{252}Cf

Fragment Charge	Elemental Yield	Fragment Mass	% Odd-even effect	Charge Polarisation
36	1.0±0.21	91.92	-	0.20±0.13
37	1.78±0.36	94.32	-	0.30±0.14
38	3.10±0.76	96.90	2.57	0.33±0.14
39	4.33±0.83	99.57	7.21	0.29±0.05
40	8.74±0.98	101.97	9.27	0.34±0.01
41	11.31±0.53	104.69	4.23	0.29±0.07
42	14.51±0.25	107.11	6.65	0.31±0.08
43	10.65±0.76	109.31	6.51	0.49±0.03
44	15.20±0.50	111.62	23.50	0.60±0.10
45	7.78±0.26	114.19	28.17	0.58±0.30
46	10.42±0.42	116.46	22.45	0.69±0.01
47	6.46±0.63	119.07	8.15	0.69±0.01
48	3.47±0.48	122.30	-	0.43±0.05
49	1.62±0.41	126.00	-	0.05±0.05

COMPRESSION MODULUS OF NUCLEAR MATTER IN THE INFINITE NUCLEAR MATTER MODEL

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The compression modulus K of infinite nuclear matter is a fundamental constant of nature. It has not yet been possible to determine with certainty the value of K directly from some data on nuclei. Using the energies of the giant isoscalar monopole resonances in Ca^{40}, Zr^{90} and Pb^{208}, Blaizot et al[1] had extracted the value of K to be 210 + 30 MeV through Hartree-Fock and RPA calculation using Skyrme interactions and extrapolating to nuclear matter. The soundness of this procedure has been questioned by Brown and Osnes[2] on several counts. Thus, the extraction of the value of K from the experimental data on breathing mode is plagued with inherent limitation and uncertainties. We present a method here to determine the value of K using the best known data of nuclei, namely, the ground-state energies of nuclei.

We will use here the mass formula based on infinite nuclear matter (INM) model[3] which has been proved very successful in predicting masses of nuclei far from stability. It is most suitable to extract the value of K from masses as it is exclusively built in terms of the properties of infinite nuclear matter. In this model the energy $E^S_{INM}(A,Z)$ *of a sphere containing A nucleons and Z protons is given by*

$$E^S_{INM}(A,Z) = -a^I_V A + a^I_S A^{2/3} + a^I_c \frac{Z^2}{A^{1/3}} + a_a \beta^2 A \tag{1}$$

where a^I_V, a^I_S, a^I_c *and* a^I_a *are the volume, surface, Coloumb and asymmetry parameters resfectively whose values are determined before[3]. The density* ρ_0 *of the nuclear matter in its ground-state obtained from the value of* a^I_c *is 0.219 nucleons/fm*^{3}.

We have chosen the mass formula of Moeller and Nix[4] which is based on Yukawa plus exponential model. In this model, the difuseness of the nuclear shape is taken into account by folding a equivalent spherical shape with Yukawa plus exponential function, and the experimental density of nucleus is used as input. This sphere should be considered as the XINM sphere in analogy of INM sphere but containing excited nuclear matter. The energy of XINM sphere is givan by

$$E^S_{XINM}(A,Z) = -a_V A + a_S A^{2/3} + a_c \frac{Z^2}{A^{1/3}} + a_a \beta^2 A \tag{2}$$

Here, a_v, a_s, a_c *and* a_a *are the volume, surface, Coulomb and asymmetry coefficients respectively whose values are determined by Moeller and Nix*[4].
The value of the density ρ *coming from* a_c *is 0.153 nucleons/fm*^{3}. *Now the compressibility K is given by*

$$K = 18\left[E^{S}_{XINM}(A,Z) - E^{S}_{INM}(A,Z)\right]\left(\frac{\rho_0}{\rho-\rho_0}\right)^2 A \qquad (3)$$

Using the abovs formula, K is calculated taking all the values of A and Z pertaining to all (181) the beta stable nuclei between mass number 60 to 240. The value is 158 ± 5 MeV. A plot of K vs A is shown in Fig.1. K remains remarkably constant with negligible A dependence. This is in confirmity with expectation. This is a unique value with variance of 5 MeV only. This is in accord with the data on supernovae explosion[5]. *Details of the present study can be found elsewhere*[6].

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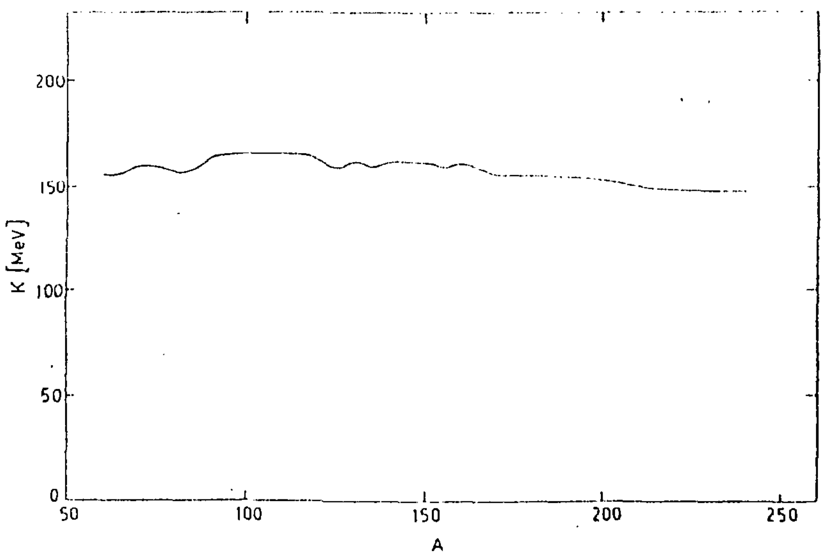

Fig 1

INFLUENCE OF ENERGY DEPENDENCE OF DILUTION CROSS SECTIONS ON CRITICALITY

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The neutron spectrum in energy is significantly influenced by the presence of resonances in cross sections of any material and of the surrounding materials in a mixture in a reactor assembly - the influence usually known as "self-shielding". In the Bondarenko method of obtaining spectrum weighted group average neutron cross sections in the resonance region, the cross sections of surrounding materials are represented by a parameter (dilution cross section) which is constant in energy whereas in reality it is energy dependent. The extent of agreement of the approximation was investigated earlier for the fast critical assembly ZPR-6-7. Very large deviations with respect to the recommended accuracies were found in the spectrum weighted group average neutron cross sections in the resonance region, in some cases. The investigation was extended to examine the influence of the approximations on criticality. K-eff calculations were made for ZPR-6-7 using the self shielding factors obtained with and without the assumption of energy dependence of the dilution with resolved resonance region. Table 1 presents a few case results bringing out the deviation in the value of K-eff and shows the inaccuracy of the conventional Bondarenko method.

Reference:

V. Gopalakrishnan and S. Ganesan,
"Numerical Investigation to Examine the Validity of Bondarenko Definition of Self-Shielded Cross Sections and Its Influence on Criticality." Paper presented at the Seventh National Symposium on Radiation Physics held at Mangalore, Karnataka, during November 16-20, 1987.

Table 1

K-eff Sensitivities

Case No.	Detail	K-eff	pcm difference from Case 1
1	Conventional cross section set used	.97855	-
2	Pu-239 fission cross sections modified using energy dependent dilutions	.98166	+ 312
3	U-238 capture modified as above	.96533	- 321
4	U-238 capture and Pu-239 fission and capture modified as above	.96000	- 230

USE OF PERSONAL COMPUTERS IN NUCLEAR DATA PROCESSING FOR REACTOR APPLICATIONS

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The nuclear data processing for reactor application and calculations relating to nuclear data evaluation, processing and testing demand very large CPU times and core memory and it is well known that the use of mainframe computers in a dedicated way will be ideally needed for 'one shot' generation of complete set of group constants starting from basic evaluated data file. It is thus very clear at the outset that the use of personal computers, despite their dedication, may provide results for only limited and part of the nuclear data processing activities. Nevertheless such personal computer based nuclar data activities can be tailored to compliment our use of mainframe systems in some cases of processing of data of isotopes for which the data is relatively simpler, for example, the neutron data of isotopes such as N-15, B-10 etc., which do not exhibit large number of resonances and such other complexities. In particular the use of personal computers has the following very attractive features.

1. For a few isotopes and neutron induced reactions which do not involve resonance reconstruction and Doppler broadening the CPU times in mainframe are small but the actual turnover time to get the outputs are large in view of multiprogramming environment and in such cases use of PC for quickly getting the results at the table may be useful. This has amply been proved by our experience for processing data of N-15 isotope in this paper.

2. Use of desk-top computers enable the obtaining of quick plots of data from the complex data base such as ENDF/B and comparison with experimental data base such as EXFOR. The use of personal computer has been used for such applications and found to be incredibly advantageous for quickly seeing the plots of vast amount of data on the video screen, and scan them for the first time and also later as and when needed when data is received from nuclear data centres such as International Atomic Energy Agency in Vienna, Brookhaven National Laboratory in U. S. A. etc.

3. In the field of neutron-nuclear data research it is well known that for the same given isotope the neutron cross section data retrieved from basic data files supplied by different countries differ from each other for any given nuclear reaction. Intercomparison of neutron reaction cross section data bases in the form of tables and plots is unavoidably the starting point to get information on discrepancies in evaluated cross section files. Use of a dedicated PC in such cases of applications can produce quicker results relative to the use of a mainframe system available in a multi-programming environment.

4. Education and training of personnel in complex areas such as nuclear data evaluation, processing and testing can be done in a very effective way using PC's without keeping the main frame computer occupied for the purpose of demonstration runs of illustrative examples.

The calculations presented in this paper were performed using a PC/AT model with the following features : (a) 640 Kilo Bytes (KB) of memory on the system board; (b) 20 Mega Bytes (MB) Fixed Disk drive; (c) 1.2 MB Diskette Drive; (d) 360 KB Diskette Drive (e) DOS 3.1; (f) 80287 math co-processor.

The design calculations of fast reactors using nitride fuel (PuN, UN) enriched in N-15 obviously requires accurate processed multigroup data of N-15 as input. The data base of N-15 of the latest ENDF/B-V (Rev.2) was obtained from BNL, USA. The entire data were retrieved and viewed as plots on video screen of the desk-top computer.

The following pre-processing codes developed by Nuclear data section of IAEA were used in the present study . (1) LINEAR (Version 87-1); (2) RECENT (Version 87-1); (3) SIGMA1 (Version 86-1) (4) GROUPIE (Version 86-2). Running times with sample problems run on the PC and the local mainframe are compared in Table 1. The N-15 data of ENDF/B-V (Rev.2) was processed on both the PC and the mainframe system. Actual times to get the outputs are indicated for the PC-AT. In the case of the mainframe, the times are those taken for batch jobs. The time taken for these jobs when they are run on an interactive time sharing basis on the mainframe exceeds one hour for each of the codes. As a dedicated system therefore, the PC-AT works out to be effectively faster than the mainframe for this particular example.

Reference:

S.Ganesan et al., Paper presented in the National Symposium on Personal Computers in Science and Engineering, BARC, Feb.3-5, 1988.

Table 1

Data on N-15 of ENDF/B-V (Rev.2)

Pre-processing Codes	Total clock time in (minutes)	
	IBM/PC-AT compatible	Honeywell Bull DPS-8
LINEAR	23.67	1.62
RECENT	23.25	6.60
SIGMA1	21.37	6.60
GROUPIE	15.60	4.38

IMPORTANCE OF CORRELATED UNCERTAINTIES IN THE CALCULATION OF TEMPERATURE DERIVATIVES OF CROSS SECTIONS

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When studying temperature dependent cross section effects such as a change in the self-shielded cross section as a function of temperature, it must be realised that even if the self-shielded cross sections are calculated to an accuracy of 0.5% at each temperature, a difference between the self-shielded cross sections at two temperatures may have much larger uncertainty unless special attention is paid to correlating the uncertainties in the self-shielded cross sections at the two temperatures. For example, if the uncertainty in the self-shielded cross sections at two temperatures are each independently only 0.5%, but the cross sections at the two temperatures differ by 0.5%, the uncertainty is the difference between the cross sections is over 100% and even the sign is uncertain.

To avoid this problem the SIGMA1 numerical Doppler broadening method is designed to strongly correlate the uncertainties in the energy dependent cross section at the initial (cold) and final (hot) temperatures. The Doppler broadening equation solved by the SIGMA1 method is merely the diffusion equation in spherical geometry. One of the important properties of the diffusion equation is that the integral of the reaction rate integrated over phase space is conserved independent of temperature. This implies that if we start from energy dependent cross sections which are 0.5% too high at the initial (cold) temperature we will obtain energy dependent cross sections which are also 0.5% too high at the final (hot) temperature, i.e., the uncertainties in the cross sections at the two temperatures are strongly correlated. This strong correlation is also illustrated in the results presented here. For example, we compare the results obtained by linearising, reconstructing and Doppler broadening using 0.1% uncertainty at each step to the results obtained by linearising and Doppler broadening to 0.1% but reconstructing to only 1%. If we consider the results obtained using the uncertainty of only 1% at each step to be 'exact' we can use the results obtained due to reconstructing the cross sections to 1% accuracy to predict (assuming independent errors at each temperature) and determine the true uncertainties introduced due to cross section processing.

From Table 1 we see that the actual uncertainty introduced due to cross section processing is consistently at least an order of magnitude less than the error predicted assuming that the uncertainties in the cross sections at the two temperatures are uncorrelated; i.e., the uncertainties are strongly correlated. From table it is noted that over the ranges of dilution normally encountered in application ($\sigma_0 < 10^3$) we can predict the change in the self shielded cross section due to Doppler broadening to within the target accuracy of 2%. In the unlikely situation where $\sigma_0 > 10^3$ barns, although the uncertainty in the change in the cross section with temperature rapidly increases with σ_0 , the change in cross section also decreases rapidly and plays little, if any, role in changing the reactivity of any reactor.

Reference

S. Ganesan, V. Gopalakrishnan, M.M. Ramanadhan and D.E. Cullen, "Verification of the Accuracy of Doppler Broadened Self-Shielded Multigroup Cross Sections for Fast Power Reactor Applications", Annals of Nuclear Energy, Vol. 15, No. 3, pp 113-140 (1988).

Table 1

IMPORTANCE OF CORRELATED UNCERTAINTIES IN THE CALCULATION OF DOPPLER CHANGES IN SELF-SHIELDED CROSS SECTIONS

Self-shielded Cross Sections 0.1% reconstruction			Doppler change 300 K to 2100 K	$\Delta\sigma$	Predicted uncertainty due to 1% reconstruction	Actual uncertainty due to 1% reconstruction
Dilution (barns) σ_0	Temperature 300 K	Temperature 2100 K	$\Delta\sigma$ *	Expressed as percent Q **	(percent) ***	(percent) ****
0	0.6345	0.8572	0.2227	35	2.85	0.27
1	0.6526	0.8746	0.2220	34	2.94	0.27
10	0.7586	0.9738	0.2152	28	3.57	0.28
100	1.0554	1.1963	0.1409	13	7.69	0.48
1000	1.2825	1.3152	0.0327	2.5	40	1.8
10000	1.3334	1.3357	0.0021	0.15	666	38

* $\Delta\sigma$ = Doppler change from 300 K to 2100 K = Self-shielded cross section at 2100 K self-shielded cross sections at 300 K

** Q = expressed as percent = 100 * $\Delta\sigma$/self-shielded cross section at 300 K

*** Predicted uncertainty due to 1% reconstruction = $\frac{100 * 91\% \text{ of self-shielded cross section at 300 K)}}{\Delta\sigma}$

**** Actual uncertainty due to 1% reconstruction =

$$\frac{100 * \text{(Doppler change calculated with 1.0\% reconstruction - Doppler change calculated with 0.1\% reconstruction)}}{\text{(Doppler change calculated with 0.1\% reconstruction)}}$$

RESONANCE CROSS SECTIONS OF STRUCTURAL MATERIALS WITH REICH-MOORE PARAMETRIZATION

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A new method of calculating Doppler broadened cross sections based on a fast converging series obtained by us/1/ for Doppler broadening function $\Psi(x,\theta)$ for structural materials was developed. A new method is necessitated due to the fact that almost all the recent measurements on structural material cross sections make use of Reich-Moore formalism for the asymmetric s-wave part and single level Breit-Wigner formalism for $\ell > 1$ waves. Such hybrid description of resonances is not, at present, allowed in standard evaluated files such as ENDF/B and JENDL and most of the processing codes follow the formats and procedures of only such standard evaluated files.

Finally, a sample calculation with the recent Fe-56 resonance parameters was presented which brought out large differences between the ENDF/B-IV data and recent experimental data for both broadened and unbroadened cross sections/2/.

References:

1. R.S. Keshavamurthy, J. Phys. A.: Math. Gen. 20 (1987) L 273.

2. R.S.Keshavamurthy, Resonance Cross Sections of Structural Materials with Reich-Moore Parametrization, RG/RPD-310, 1987.

ESTIMATION OF HELIUM PRODUCTION IN STAINLESS STEEL DUE TO NEUTRON IRRADIATION

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It is well known that helium produced in the components of stainless steel on irradiation in a neutron spectrum, through the (n,α) reactions, affect the mechanical properties of the stainless steel, like embrittlement, fracture and so on. In order to estimate the amount of helium produced in stainless steel in reactor environment, specimens of stainless steel DIN 1.4970 and DIN 1.4970 LB (low boron) were irradiated in the FRJ 2 reactor at KFA, Julich, FRG, for a known fluence of both thermal and fast neutrons (mixed spectrum) and the amount of helium atoms produced was measured accurately by mass spectrometry. In the present work, calculations were performed using the neutron interaction cross sections of the constiuents of the stainless steel specimen, taking into account important reactions like B-10 (n,α), the two step reaction process in Nickel i.e. Ni-58 (n,γ) Ni-59(n,α) Fe-56, and the details of the neutron specturm in a multigroup way. The Evaluated Nuclear Data Library ENDL/84-V was used for all the constituents, except Ni-59 for which the data was taken from KEDAK-4.

The calculated amount of helium was compared with the mass spectrometric measurements and the results are in good agreement.

DEVELOPMENT OF INDIAN CROSS SECTION DATA FILES FOR TH-232 AND U-233 AND INTEGRAL VALIDATION STUDIES

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An invited talk was given on the above topic at the International Conference on Nuclear Data for Science and Technology, held during May 30 - June 3, 1988, at Mito, Japan.

The following is the abstract/1/:

This paper presents an overview of the tasks performed towards the development of Indian cross section data files for Th-232 and U-233. Discrepancies in various neutron induced reaction cross sections in various available evaluated data files have been obtained by processing the basic data into multigroup form and intercomparison of the latter. Interesting results of integral validation studies for capture, fission and (n,2n) cross sections for Th-232 by analyses of selected integral measurements are presented. In the resonance range, energy regions where significant differences in the calculated self-shielding factors for Th-232 occur have been identified by a comparison of self-shielded multigroup cross sections derived from two recent evaluated data files, viz., ENDF/B-V (Rev. 2) and JENDL-2, for several dilutions and temperatures. For U-233, the three different basic data files ENDF/B-IV, JENDL-2 and ENDL-84/V, were intercompared. Interesting observations on the predictional capability of these files for the criticality of the spherical metal U-233 system are given. The current status of Indian data file is presented.

The complete details are documented in the reference.

Reference

S. Ganesan, "Development of Indian Cross Section Data Files for Th-232 and U-233 and Integral Validations Studies", pp. 929-936 in S. Igarasi (Ed), Proceedings of the International Conference on Nuclear Data for Science and Technology, May 30 - June 3, 1988, Mito, Japan, Saikon Publishing Co., Ltd.

MULTIGROUP CONSTANTS BASED ON JENDL-2 NUCLEAR DATA LIBRARY

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In IGCAR, the adjusted, 25 group French (Cadarache) cross section set containing data for 39 materials is frequently used for the fast reactor physics calculations. To augment this, to update this and to provide the data for the missing materials inthis set, we have developed an infrastructure to create multigroup constants based on an evaluated data library in the ENDF/B format. Mostly we use/1/ ENDF/B-IV (1974,USA) data for this purpose, and ENDL/84-V or parts of ENDF/B-V in some cases, and activity of updating or improving our multigroup data from these data bases continues. On taking stock of the present situation, we felt the need of parallel multigroup data set, based on JENDL-2 (Japanese Evaluated Nuclear Data Library - Version 2, 1982; distributed by IAEA) which enjoys a general international feeling to be among the recent good evaluations.

The IGCAR system of preprocessing and processing codes viz. LINEAR, RECENT, SIGMA1, FIXUP, REX1-87, REX2-87, REX3-87 and LCAT are being used to calculate the required unshielded and shielded cross section averages, group to group transfer matrices etc. and to store them in a format required by the code RECRHOM (the earlier version known as EFFCROSS) which prepares the mixture cross sections for a given assembly. We presently consider 4 temperatures viz. 300, 900, 1500 and 2100 kelvin and 6 dilutions viz. 0.0 (or 1.0), 10, 100, 1000, 10000, and 100000 barns. The UNRESR of NJOY, coupled to REX3-87 is being used for processing unresolved resonance region.

As indicated in Table 1, we have completed multigrouping the data for many materials from JENDL-2:

Reference:

1. P.V.K. Menon (Ed.), Activity Report of Reactor Physics Division - 1987, IGC-98 (1988), page 1.

Table 1

Materials processed from JENDL-2 library.

Sl.No.	MATERIAL	MAT NUMBER	Sl.No.	MATERIAL	MAT NUMBER
1	Li-6	2031	13	Mo-94	2422
2	Li-7	2032	14	Mo-95	2423
3	B-10	2051	15	Mo-96	2424
4	C-12	2061	16	Mo-97	2425
5	Na	2111	17	Mo-98	2426
6	Si	2140	18	Mo-100	2427
7	Cr	2240	19	Th-232	2903
8	Mn-55	2251	20	U-235	2923
9	Fe	2260	21	Pu-239	2943
10	Ni	2280	22	Pu-240	2944
11	Mo	2420	23	Pu-241	2945
12	Mo-92	2421	24	Pu-242	2946

ABAREX-PC, THE PERSONAL COMPUTER VERSION OF THE OPTICAL-STATISTICAL MODEL CODE ABAREX

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The code ABAREX, of late Prof. P. A. Moldauer, which calculates the energy averaged cross section at a given energy point of a neutron-nuclear interaction process based on Optical-Statistical nuclear reaction model, taking into account the width fluctuation corrections, has already been adapted to our main fame computers, Honeywell and Norsk Data.

The main frame version was received from Nuclear Energy Agency - Data Bank, Saclay, France.

Necessary modifications were made to run the code on an IBM/PC-AT and equivalent desk-top computers. Comment cards were added to document the changes. Input Specifications along with some details on the theory, and capabilities of the code has been brought out as a document/1/. The code was tested on our Microway PC/AT. The code was also demonstrated on the Olivetti PC/AT, to the participants of the recent Workshop on Applied Nuclear Theory and Nuclear Model Calculations for Nuclear Technology Applications, at the International Centre for Theoretical Physics, Trieste, Italy,(15 Feb - 18 Mar, 1988)/2/.

A sample calculation for Th-232, using the model parameters reported earlier/3/, was done on the PC/AT. Table 1 gives a sample result from this version of the code, showing the effect of the width fluctuation correction factor on the inelastic scattering cross sections for various residual nuclear levels. The results agree with those obtained on the main frame computers.

This version, called ABAREX-PC, along with its document and sample input and output, has been submitted to the NEA-DATA BANK, Saclay, on a request from there and is available for general distribution.

References:

1. Ganesan S. 'Input Specifications and Sample Problems for ABAREX, the Optical Statistical Model Code', Internal Note RPD/NDS-19,(1988).

2. S. Ganesan, 'Statistical Theory of Fluctuations in Neutron-Nucleus Cross Sections and Use of "ABAREX", the Optical-Statistical Model Code', lecture notes to be published by World Scientific Publishing Co. Pvt. Ltd., Singapore, in the Proceedings of the ICTP Workshop.

3. P.V.K. Menon (Ed.), Activity Report of Reactor Physics Division - 1987, IGC-98 (1988), page 9.

Table 1

EFFECT OF WIDTH FLUCTUATION CORRECTION (WFC) ON INELASTIC LEVEL EXITATION CROSS SECTIONS (THORIUM-232 + NEUTRON)

	CROSS SECTIONS IN BARNS			
	ENERGY = 1.0 MeV		ENERGY = 0.1 MeV	
LEVEL (MeV)	WITH WFC	WITHOUT WFC	WITH WFC	WITHOUT WFC
0.0000	0.68452	0.37815	2.50319	2.05959
0.0494	0.79414	0.91909	0.56575	0.95894
0.1621	0.38031	0.41637		
0.3331	0.02343	0.02324		
0.5569	0.00005	0.00004		
0.7143	0.20810	0.24758		
0.7304	0.13391	0.15589		
0.7741	0.27042	0.29923		
0.7744	0.10884	0.11774		
0.7852	0.25730	0.28378		
0.8274	0.08708	0.09851		
0.8296	0.15119	0.15926		
0.8730	0.05713	0.05769		
0.8833	0.00221	0.00218		
0.8901	0.04727	0.04743		
0.9602	0.00516	0.00506		

DEVELOPMENTS IN THE CODES REX1 AND REX2

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The code REX1, which calculates the unshielded average multigroup cross sections, scattering matrices etc., and the code REX2, which calculates the self shielding factors in the resolved resonance region, have undergone drastic restructuring to accommodate many of the present day requirements. The present versions of these codes are called respetively REX1-88 /1/ and REX2-88 /2/.

The differences between the old and the new versions are shown below:

Feature	Old version	New version
Precision	Single	Double
Integrations	All numerical	Analytical whereever feasible
Number of groups	Only 25	Upto 620
Builtin group limits	Only Cadarache (25 groups)	Cadarache(25), WIMS(69), DLC-2(100) and SAND-2(620)
Number of energy points handled	Maximum 20,000	No limit
Memory required	Nearly 180 kWords	Less than 80 kWords
Speed	Relatively slow	Faster
Transportability	Less (uses CHARACTER, NAMELIST, ENCODE DECODE etc.)	High (uses almost primitive FORTRAN-IV)
PC Version	Not possible	Available

On a request from the Nuclear Energy Agency - Data Bank, Saclay, France, these two codes have been submitted for general distribution. The following will bring out the major differences between the older and the present versions of the two codes:

References:

1. Gopalakrishnan V. and Ganesan S. 'REX1-87, A Code for Multigrouping Neutron Cross Sections from Pre-processed ENDF/B Basic Data File', Internal Note RPD/NDS-13.

2. Gopalakrishnan V. and Ganesan S, 'REX2-87, A Code for Calculating Self-shielded Multigroup Neutron Cross Sections and Self-shielding Factors from Pre-processed ENDF/B Basic Data File', Internal Note RPD/NDS-18.

EXFORET, AN EXFOR DATA RETRIEVAL PROGRAM

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In the EXFOR (EXchange FOrmat) data library, which contains experimental data and related information of nuclear data measurements, usually cross sections, angular distributions, associated errors etc., the information content for each material (element or isotope) is so huge that a computer program for selective retrieval or for editing is unavoidable. Since we attempted a limited evaluation of the Th-232 cross sections under an IGCAR-IAEA contract, we scanned the informations in the EXFOR library and the fission cross sections of Th-232 given by Meadows were preferred to those given by others based on our judgement. During this work, we developed a program EXFORET, for selective retrievals from this huge data library. EXFORET retrieves a cross section information along with the associated errors and necessary bibliographic details based on some simple keywords. The output of EXFORET is simple enough for comparisons, plotting and analyses.

Fig 1. gives a plot of the Meadows' fission cross section for Th-232 versus energy, retrieved from EXFOR. The recent EXFOR data was obtained from IAEA in March, 1988/2/

A copy of the program EXFORET has been given to BARC (Dr. H.M. Jain) on request. The program details and the input specifications are given in an internal document /1/.

1. Ramanadhan M.M., Ganesan S. and Gopalakrishnan V., 'EXFORET, A Program for Selective Retrieval of Cross Sections and other Relevant Data from EXFOR Library', Internal Note RPD/NDS-21.

2. Hans D. Lemmel, IAEA Nuclear Data Section, Private Communication (March 1988)

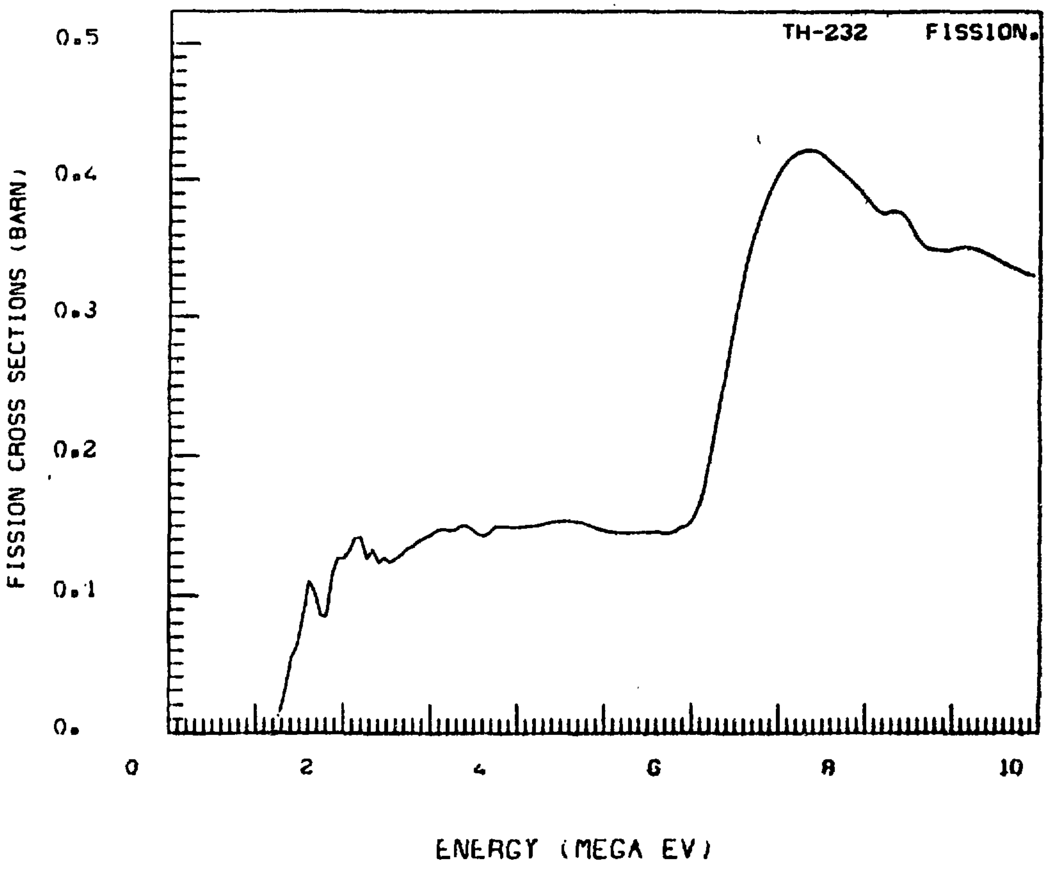

Fig 1. A plot of the Meadows' fission cross section for Th-232 versus energy, retrieved from EXFOR. This plot was made by Kalpakkam program EXFORET.

PROCESSING OF BASIC DATA FILES FOR SHIELDING AND ACTIVATION STUDIES IN FAST BREEDER TEST REACTOR

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Nuclear data services were rendered to Radiation Shielding and Statistical Physics Section of Reactor Physics Division for activation and shielding studies. Two of the significant nuclear data processing activities performed successfully are given below:

i. For the study of the regeneration strength of the neutron Source, i.e. the Sb-Be subassembly, in the FBTR spectra, the cross sections, shielded and unshielded, and the scattering matrices were requested by the above section. The request was met by processing antimony data from ENDF/B-IV by our codes system. Since the self-shielding was very significant, the dilution cross section was computed taking into account all the nuclides around antimony in the source region and the shielded cross sections were calculated at 625 kelvin temperature. Nuclear data of both the stable isotopes antimony viz Sb-121 and Sb-123 were processed.

ii. Activation cross sections in the DLC-2 type 100 group structure were given for all the nuclides of FBTR, in addition to the average fission spectra for Pu-239 from ENDF/B-IV.

iii. In order to sufficiently respect the sharp dips in the total and elastic cross sections ("windows"), a new 117 group structure was arrived at, and average unshielded cross sections and transfer matrices were obtained for fuel, structural and coolant materials of FBTR.

EXACT ANALYTICAL EVALUATION OF DOPPLER EFFECT IN LOW ENERGY NEUTRON RESONANCE REACTIONS

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One of the main problems in neutron resonance spéctroscopy as well as in reactor physics is to take into account Doppler effect, arising due to thermal motion of target nuclei, properly as it can drastically distort the shapes of resonances. Analytical evaluation of this effect would prove very useful in extraction of resonance parameters. Recently Hwang has given an exact analytical evaluation starting from the solbrig Kernel/1/. We have evaluated it analytically through a different approach and expressed it in terms of complex probability functions. Our result agrees with that of Hwang but for some errors in his paper in identifying complex probability function through its integral representation. We have also shown the approximation under which the exact result reduces to the approximate Ψ function approach. We have further suggested the use of Pade approximation for complex probability function. Our calculations show that Ψ function approximation works very well for most of the resonances but for the very low energy sharp ones. Typically, for these low energy sharp resonances, the deviations are upto 0.9% for U-238, 1.1% for Th-230, 0.9% for Pu-240 at room temperature. If the temperature is increased four-fold the respective deviations go up to 2.6%, 3.7% and 3% thus making the exact treatment important.

1. R.N. Hwang, Nucl. Sci. Eng. 96, 192 (1987)

COMPARISON OF ABSORPTION CROSS SECTIONS IN WIMS 69 ENERGY GROUP STRUCTURE

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On a request from Mrs. Kamala Balakrishnan, Reactor Engineering Division, BARC, two sets of unshielded cross section averages for Th-232 in WIMS 69 group structure have already been obtained by processing two evaluated data files viz. ENDF/B-V (Rev. 2) and JENDL-2. The processing of these files were carried out using Kalpakkam processing code system. In continuation, the self-shielding factors both in the resolved and in the unresolved resonance regions have been calculated for two temperatures and 12 dilutions. The ENDF/B-V (Revision 2) data was processed by LINEAR, RECENT, FIXUP, SIGMA1, REX1-87, REX2-87 and REX3-87 for this purpose. In addition, Maxwellian averaged cross sections were calculated for several values of temperature, for WIMS' groups lying below 0.625 eV.

Table 1 gives the ratios of multigroup absorption cross sections JENDL-2/WIMS, ENDF/B-V (rev.2)/WIMS and also the ratio JENDL-2/ENDF/B-V (rev.2). The WIMS library shows large deviations with respect to the two recent files JENDL-2 and ENDF/B-V (Rev.2). It is interesting to observe that even the two recent files have not converged and they exhibit large differences in 0.8 eV to 48 eV energy region. Between 27 eV and 48 eV the absorption cross section in WIMS library is too low by two orders of magnitude. The two recent files differ by a factor of two in this energy region.

TABLE 1

COMPARISON OF ABSORPTION CROSS SECTIONS $[\sigma(n,\gamma)+\sigma(n,f)-\sigma(n,2n)]$
OF TH-232 IN THE "WIMS" 69 ENERGY GROUP STRUCTURE

GROUP	ENERGY LIMITS	ABSORPTION CROSS SECTIONS (BARN)			CROSS SECTION RATIOS		
NO.	(UPPER) (EV)	WIMS	JENDL-2	ENDF/B-V	JENDL /WIMS	ENDF /WIMS	JENDL /ENDF
1	1.00000E+7	-1.57424E-1	-2.07729E-1	-2.71285E-1	1.320	1.723	0.766
2	6.06550E+6	1.71179E-1	1.59252E-1	1.57891E-1	0.930	0.922	1.009
3	3.67700E+6	1.67519E-1	1.65646E-1	1.64231E-1	0.939	0.980	1.009
4	2.23100E+6	1.86655E-1	1.80139E-1	1.78472E-1	0.965	0.956	1.009
5	1.35300E+6	2.08102E-1	1.33484E-1	1.35091E-1	0.641	0.649	0.988
6	8.21000E+5	2.14778E-1	1.44777E-1	1.61547E-1	0.674	0.752	0.896
7	5.00000E+5	1.84542E-1	1.29810E-1	1.44810E-1	0.703	0.785	0.896
8	3.25000E+5	2.06643E-1	1.37860E-1	1.52710E-1	0.657	0.739	0.903
9	1.83000E+5	2.62443E-1	1.73020E-1	1.90350E-1	0.659	0.725	0.909
10	1.11000E+5	3.58648E-1	2.37360E-1	2.50410E-1	0.652	0.698	0.948
11	6.73400E+4	4.96241E-1	3.45100E-1	3.70880E-1	0.695	0.747	0.930
12	4.08500E+4	6.93103E-1	4.68310E-1	4.48770E-1	0.676	0.647	1.044
13	2.47800E+4	7.19826E-1	5.81550E-1	5.55240E-1	0.532	0.604	1.047
14	1.50300E+4	1.19851E+0	7.15850E-1	6.87250E-1	0.597	0.573	1.042
15	9.11800E+3	1.54721E+0	8.32950E-1	8.49150E-1	0.571	0.549	1.040
16	5.53000E+3	1.96327E+0	1.09180E+0	1.08780E+0	0.556	0.555	1.002
17	3.51910E+3	2.45502E+0	1.06530E+0	1.50790E+0	0.434	0.614	0.706
18	2.23950E+3	3.07916E+0	1.74030E+0	2.05900E+0	0.565	0.669	0.845
19	1.42510E+3	3.83530E+0	1.89870E+0	2.30170E+0	0.495	0.600	0.825
20	9.06900E+2	5.36570E+0	3.05690E+0	3.65920E+0	0.570	0.682	0.835
21	3.67260E+2	7.37931E+0	1.07560E+1	1.10750E+1	1.458	1.501	0.971
22	1.48730E+2	1.04318E+1	9.74030E+0	1.00670E+1	0.934	0.965	0.968
23	7.55010E+1	2.96486E+1	3.91730E+1	3.71160E+1	1.321	1.252	1.055
24	4.80520E+1	1.01601E+1	7.36030E-2	1.48420E-1	0.007	0.015	0.496
25	2.77000E+1	1.12995E+2	6.67030E+1	7.57750E+1	0.590	0.671	0.880
26	1.59680E+1	3.36137E-1	1.08290E-1	2.24670E-1	0.322	0.668	0.482
27	9.87700E+0	4.70778E-1	1.63230E-1	2.91150E-1	0.347	0.618	0.561
28	4.00000E+0	5.83670E-1	2.53960E-1	3.86780E-1	0.435	0.663	0.657
29	3.30000E+0	6.31647E-1	3.19070E-1	4.53380E-1	0.505	0.718	0.704
30	2.60000E+0	6.86108E-1	4.00860E-1	5.35290E-1	0.584	0.780	0.749
31	2.10000E+0	7.59103E-1	5.21600E-1	6.54580E-1	0.687	0.862	0.797
32	1.50000E+0	8.28351E-1	6.44490E-1	7.74360E-1	0.778	0.935	0.832
33	1.30000E+0	8.69625E-1	7.20220E-1	8.48670E-1	0.828	0.976	0.849
34	1.15000E+0	8.93175E-1	7.63950E-1	8.91510E-1	0.855	0.998	0.857
35	1.12300E+0	9.00925E-1	7.78480E-1	9.05680E-1	0.864	1.005	0.860
36	1.09700E+0	9.08781E-1	7.94130E-1	9.20530E-1	0.874	1.013	0.863
37	1.07100E+0	9.16899E-1	8.09810E-1	9.35390E-1	0.883	1.020	0.866
38	1.04500E+0	9.25128E-1	8.25180E-1	9.50770E-1	0.892	1.028	0.868
39	1.02000E+0	9.33415E-1	8.40390E-1	9.66170E-1	0.900	1.035	0.870
40	9.96000E-1	9.47175E-1	8.57210E-1	9.81680E-1	0.905	1.036	0.873
41	9.72000E-1	9.63701E-1	8.73520E-1	9.97130E-1	0.906	1.035	0.876
42	9.50000E-1	9.87250E-1	8.95600E-1	1.01900E+0	0.907	1.032	0.879
43	9.10000E-1	1.02831E+0	9.34350E-1	1.05660E+0	0.909	1.028	0.884
44	8.50000E-1	1.08778E+0	9.89900E-1	1.11070E+0	0.910	1.021	0.891
45	7.80000E-1	1.20820E+0	1.10850E+0	1.22520E+0	0.917	1.014	0.905
46	6.25000E-1	1.39800E+0	1.29210E+0	1.40550E+0	0.924	1.005	0.919
47	5.00000E-1	1.60410E+0	1.49670E+0	1.60630E+0	0.933	1.001	0.932
48	4.00000E-1	1.78207E+0	1.67420E+0	1.80050E+0	0.951	1.010	0.941
49	3.50000E-1	1.90086E+0	1.80290E+0	1.90760E+0	0.948	1.004	0.945
50	3.20000E-1	1.98759E+0	1.88460E+0	1.98850E+0	0.948	1.000	0.948
51	3.00000E-1	2.06468E+0	1.96310E+0	2.06510E+0	0.951	1.000	0.951
52	2.80000E-1	2.17321E+0	2.07920E+0	2.17940E+0	0.957	1.003	0.954
53	2.50000E-1	2.32349E+0	2.23170E+0	2.33140E+0	0.960	1.003	0.957
54	2.20000E-1	2.54249E+0	2.46600E+0	2.56390E+0	0.970	1.008	0.962
55	1.80000E-1	2.87210E+0	2.80220E+0	2.90080E+0	0.976	1.010	0.966
56	1.40000E-1	3.35749E+0	3.30070E+0	3.39760E+0	0.983	1.012	0.971
57	1.00000E-1	3.89752E+0	3.79520E+0	3.89200E+0	0.974	0.999	0.975
58	8.00000E-2	4.31295E+0	4.20540E+0	4.30830E+0	0.975	0.999	0.976
59	6.70000E-2	4.68097E+0	4.57450E+0	4.67660E+0	0.977	0.999	0.978
60	5.80000E-2	5.04392E+0	4.93800E+0	5.04040E+0	0.979	0.999	0.980
61	5.00000E-2	5.47736E+0	5.36180E+0	5.47290E+0	0.979	0.999	0.980
62	4.20000E-2	5.99744E+0	5.87180E+0	5.99180E+0	0.979	0.999	0.980
63	3.50000E-2	6.52742E+0	6.40360E+0	6.52620E+0	0.981	1.000	0.981
64	3.00000E-2	7.10287E+0	6.97950E+0	7.10420E+0	0.983	1.000	0.982
65	2.50000E-2	7.83649E+0	7.72490E+0	7.87140E+0	0.986	1.004	0.981
66	2.00000E-2	8.89366E+0	8.78080E+0	8.94550E+0	0.987	1.006	0.982
67	1.50000E-2	1.05724E+1	1.04140E+1	1.06120E+1	0.985	1.004	0.981
68	1.00000E-2	1.36578E+1	1.34540E+1	1.37150E+1	0.985	1.004	0.981
69	5.00000E-3	2.25397E+1	2.22860E+1	2.27350E+1	0.989	1.009	0.980

EXPERIENCES IN THE THEORETICAL PREDICTIONS OF NEUTRON TOTAL CROSS SECTIONS FOR U-238 USING ABAREX CODE IN 100 keV to 5 MeV ENERGY REGION.

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The neutron interaction cross sections of U-238 were computed/1/ using the optical-statistical code ABAREX/2/ for several sets of recommended model parameters proposed for actinides(given in table 1). The comparison of total cross sections calculated using those model parameters /3/ with JENDL-2 values has been made. It is found that Madland-Young-2 parameter set /3/ predict well the total cross sections of U-238 in the energy region 0.1 to 5 Mev. The comparison of total cross sections with JENDL-2 values are shown in Fig.1

Using Madland-Young-2 parameter set, the influence of width fluctuation correction factor to radiative capture cross sections are studied. It is interesting to see that its effect become negligible at higher energies. Figure 2 shows the results we obtained. We also performed the sensitivity of total cross section to 1% changes in real potential depth and real radius parameter . It is found that total cross section is very much sensitive to real radius parameter as compared to real potential depth.(see Fig.3)

References:

1. K. Devan.(1989). M.Phil thesis submitted to University of Calicut on "Calculations of Neutron induced cross sections on U-238 using ABAREX code". Unpublished.

2.S. Ganesan. (1988). Lecture notes of "Statistical theory of fluctuations in Neutron- Nucleus cross sections and use of "ABAREX", The OPTICAL- STATISTICAL MODEL CODE", presented at the workshop on "Applied Nuclear Theory and Nuclear Model Calculations for nuclear Technology Applications" held at ICTP, Italy.

3. P.E.Hodgson. (1981). The neutron Optical Model in the Actinide Region, an invited paper presented to the meeting on "Fast Neutron scattering on Actinide Nuclei" held at Paris in Nov. 1981.

TABLE 1

OPTICAL MODEL PARAMETERS PROPOSED FOR ACTINIDES BY VARIOUS AUTHORS.

	ANTSIPOV	JARY	MY1	MY2	KONSHIN	WN	MAINO	SHELDON
U_0	45.87	45.9628	47.43	46.2354	47.5	47.01	47.01	49.82
U_1	-0.4	-0.30	-0.36	-0.354	-0.3	-0.267	-0.267	-0.3
U_2	----	----	----	----	---	-0.0012	-0.0012	---
r_1	1.258	1.26	1.203	1.264	1.233	1.2781	1.259	1.260
a_1	0.626	0.63	0.629	0.612	0.62	0.66	0.660	0.63
W_0	2.95	5.52	8.983	9.265	2.70	9.52	7.2	5.52
W_1	0.4	0.40	-0.22	-0.232	0.400	-0.053	---	0.40
W_2	----	-----	0.0325	0.0332	-----	-----	----	-----
r_2	1.26	1.26	1.297	1.256	1.25	1.178	1.237	1.26
a_2	0.53	0.52	0.555	0.553	0.58	0.48	0.48	0.52
U_{so}	7.5	6.2	6.2	6.2	7.5	0	7.2	6.2
r_1'	1.258	1.12	1.1	1.1	1.1	0	1.26	1.12
a_1'	0.626	0.47	0.75	0.75	0.75	0	0.66	6.2

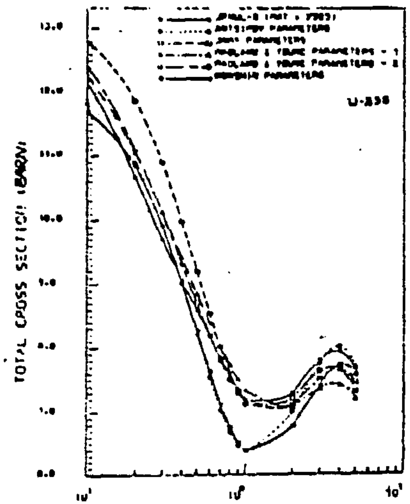


Fig. 4 TOTAL CROSS SECTIONS CALCULATED USING ABAREX FOR VARIOUS MODEL PARAMETERS.

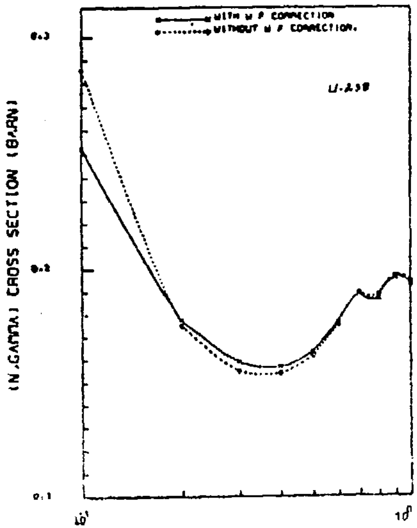
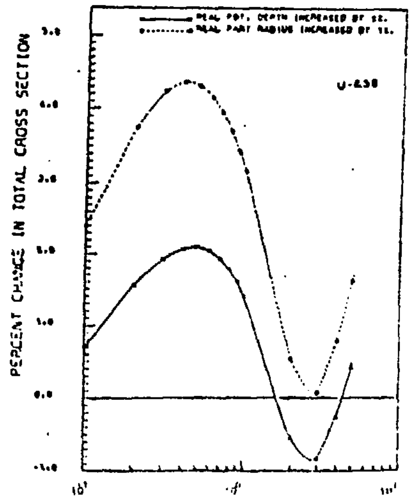


Fig. EFFECT OF WIDTH FLUCTUATION CORRECTION. (MOLDAUER-YOUNG PARAMETER SET 2 USED.)



ANALYSIS OF CENTRAL REACTION RATES FOR TH-232 IN CFRMF ASSEMBLY

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In the analysis of CFRMF it is stated /1/ that the integral measurement of Th-232 (n, γ) at CFRMF is inconsistent with ENDF/B-IV daa and that the capture cross section for Th-232 should be adjusted upwards by 10% in 0.1 keV - 1 MeV energy region. At 23 keV measurements of Baldwin and Knoll /2/ is consistent with this observation. This is in contradiction in 800 keV to 1 MeV energy region with our results /3,4/ based integral validation of $\langle \alpha \rangle$ for THOR assembly that ENDF/B-IV overestimates $\langle \sigma_c \rangle$ in 800 keV - 4 MeV energy region. The discrepancy can be resolved if we note that CFRMF covers /1/ such a wide energy region that THOR /5,6/ which covers just the fission source energy range.

Results of our calculations of central reaction rates for Th-232 in CFRMF using the BNL /5/ Benchmark specification for 620 group fluxes are presented in Table 1.

The inference is that the $\langle \sigma_c \rangle$ values in ENDF/B-IV are underpredicted by not just about 10% as indicated /1/ by CFRMF analysis in 100 ev to 800 keV energy region but some percent more than 10% as we have to provide for cancellation of higher $\langle \sigma_c \rangle$ values in ENDF/B-IV in 800 keV to 1 Mev energy region.

References:

1. R.A. Anderl et al. : Proc. of Fourth ASTM-EURATOM Symposium on Reactor Dosimetry, Maryland, March 22-26, 1982, CONF-820321/Vol.2, p.623; See also p.D1-16 of ref./5/.

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4. S. Ganesan et al., : Radiation Effects 96/1-4, 313-316 (1986).

5. Updated version BNL-19302; ENDF-202 (1983).

6. G.E. Hansen and H.C. Paxton : Nucl. Sci. Eng. 71 287 (1979).

Table 1

Integral cross section (mb) at core of CFRMF assembly

REACTION	MEASURED/5,1/	CALCULATED AT IGCAR	
		ENDF/B-V (Rev.2)	JENDL-2
Fission	19.6 $\pm$ 5.2%	19.15	20.10
Capture	290.0 $\pm$ 3.8%	257.98	239.42

International Intercomparison of Californium-252 Neutron Emission Rate.

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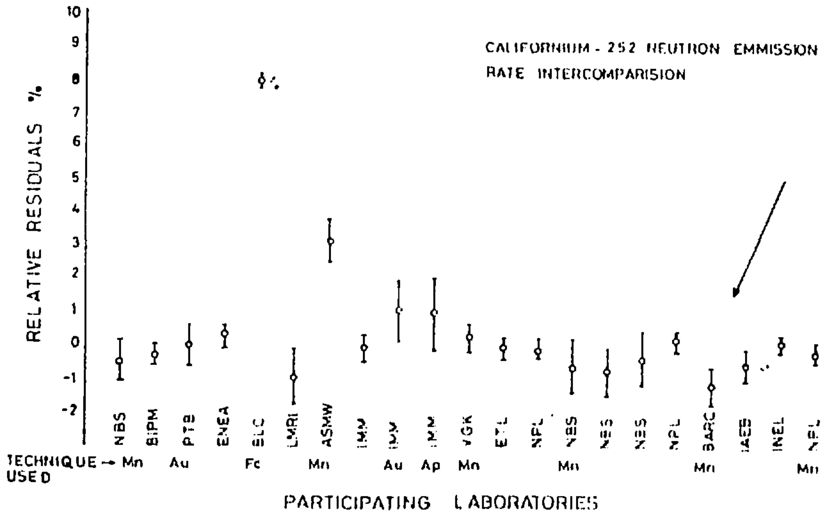
Fourteen countries including India took part in an international intercomparison of Californium-252 neutron source organised by the International Bureau of Weights and Measures, France. Three Californium sources provided by the National Bureau of Standards,U.S.A., National Physical Laboratory, U.K. and Idaho National Engineering Laboratry, U.S.A.were circulated to the participant laboratories. Participants were free to choose their method of standardization and four different methods were used namely (1) Manganese bath technique, (2) Gold foil integration of neutron density in water bath, (3) Associated alpha particle counting in T(d,n) reaction and (4) Fission counting of Califorium source.

Detailed measurements were carried out at BARC on the NPL source using an improved manganese bath technique. In order to check the consistency of the coincidence measurements on Manganese-56 a standard re-entrant high pressure ionization chamber along with a reference Radium source was circulated. The results of the intercomparison are shown in figure. The BARC value agrees within 1 % with the international mean when the input values are normalised to a common set of standard cross-section and to a common basis for calculation of corrections.



CR-39 RESPONSE MEASUREMENTS TO NEUTRONS IN THE ENERGY RANGE 0.45 MeV to 14.7 MeV.

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CR-39 is the most sensitive of nuclear track detectors and is recommended as an effective neuton dosimeter with energy response superior to other pesonnel dosimeters. As fast neutron detector and dosimeter, CR-39 is conceived as CR-39 sheet covered with a known thickness of hydrogen rich material such as polythene, to serve as a proton radiator. Be^-^on et al (1) have investigated the response of such a sy :em both experimentally and theoretically and found that the dose response is uniform to within 30 % between 0.2 MeV and 2.0 MeV for radiator thicknesses of 1 mm, 2 mm and 3 mm under certain etching conditions.

We have made an experimental determination of the CR-39 response to neutrons in the energy range 0.45 to 14.7 MeV and compared the results with those of Benton et.al. The energy points at 0.45 MeV, 1.6 MeV, 2.6 MeV and 4.3 MeV were obtained from Am/Li, Am/F , Am/B and Am/Be sealed neutron sources and represent the average energy calculated from the neutron spectra of these sources. For comparison with the sensitivity and dose response curve of Benton et. al. the CR-39 sensitivity and the dose weighted average energies of these sources were also calculated. The sensitivity was calculated using Benton's sensitivity curve for monoenergetic neutrons. The dose weighted average energies were calculated using the latest values of dose conversion factors published by IAEA (2) . The 14 MeV neutron flux was produced in the low energy accelerator at Purnima laboratories using the D - T reaction.

CR-39 samples used had been obtained from Pershore Mouldings, England. They were irradiated bare as well as covered with 1 mm. thick polythene in neutron fluences equivalent to 1 rem, usig perpendicular incidence for the neutrn flux. The samples were then etched in 6N, NaOH at (70 + 1) degrees celsius for periods extending to 16 hours, corresponding to an etched thickness of 21 microns for the purpose of comparison with other authors. Differential etching was employed to bring out the optimum etching times correspondig to individual energy groups.

Table shows the calculated and measured CR-39 response data.The sensitivity values are given in terms of tracks/n as well as tracks/cm /mrem and are for 16 hour etching time corresponding to an etched thickness of 21 microns which were the values used in Benton's work. The sensitivity values are in reasonable agreement with the calculated sensitivity values within the uncertainties of measurement. As can be seen from column 6, the sensitivity values in terms of tracks/cm##/mrem for the dose weighted energy points of 0.65 MeV, 1.63 MeV and 2.8 MeV are within 25% Of the mean value. The sensitivity to Am/Be neutrons has been measured to be 25.6 tracks/cm2/mrem.These values agree well with those obtained by Benton et. al. who obtained a sensitivity value of 27.2 tracks/cm2/sec. and whose dose response was uniform to within 30% from 0.2 to 2.0 MeV.

The measured values have an uncertainty of about 10% which was obtained from the spread in the data. The 14.7 MeV value has an additional systematic error of 5% due to the rather large correction of 30% which had to be applied for the contribution of secondary neutrons from the rather massive target mount.

REFERENCES:

(1) Benton E. V. et.al. Health Physics Vol 40(1981)
(2) Radiation Protection Dosimetry Vol 12 No. 2 (1985)

NEUTRON RESPONSE OF CR-39

Neutron Source	Average Energy MeV	Dose Weighted Average Energy MeV	Calculated Sensitivity 10^{-3} Tracks/n	Measured Sensitivity 10^{-3} Tracks/n
^{241}Am/Li	0.45	0.65	0.17	0.23
^{241}Am/F	1.6	1.63	0.36	0.40
^{241}Am/B	2.6	2.8	0.58	0.55
^{241}Am/Be	4.3	4.4	0.92	1.04
D - T	14.7	14.7	0.45	0.41

A Multisphere Spectrometer for Neutron Dosimetry in Reactor Environment.

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A multisphere spectrometer system for neutron dosimetry has been developed and tested using a standard Am-Be neutron spectrum. The multisphere spectrometer (1) is a high sensitivity four pi spectrometer especially suited for low level, nearly isotropic neutron fluences such as those obtaining in and around reactor installations, fuel reprocessing plants and accelerators. Our system consists of a set of seven high density polypropylene spheres of diameters from 5 cms. to 30 cms., containing a thermal neutron detector in the form of a gold foil or a small 4 mm * 8 mm. LiI(Eu) crystal coupled to a photomultiplier.

The fast neutrons entering the sphere are slowed down to thermal energies and detected by the thermal neutron detector at the centre of the sphere. The response of each sphere detector peaks at a different energy for each sphere, peaks occuring at higher energies for larger diameter spheres. The fluence spectrum is determined from the known response function of each sphere detector and the measured response of each sphere detector in the unknown field. The spectrum unfolding involves solving a set of linear equations corresponding to the number of multisphere detectors, using a scheme that seeks the best fit in a least squares sense, between the trial neutron spectra and the measured count rates in the case of the LiI detector or the induced activity in the case of gold foils. The solution is achieved by a process of iteration to get the spectra that best fits the experimental data.

The LiI neutron detector assembly was calibrated at thermal energy with reference to the thermal neutron flux density syandard (2). The measured response agrees well with the theoretically calculated response which is 0.3 counts per second per unit flux.

The fast neutron response of the system as a whole, that is the moderator sphere with the thermal neutron detector at the centre and its capability for spectrometry at low flux levels has been confirmed by taking measurements with the neutron spectrum of a standard Am-a-Be neutron source. The measured response for each sphere detector for the Am-Be neutron spectrum in the case of gold foils is given in Table (1). Table (2) shows the response of sphere detectors with 96% enriched LiI(Eu) assembly and other relevant data.

REFERENCES:

[1] BRAMBLETT, R.L.,EWING, R.I.,BONNER, T.W.,Nucl. Instrum. Meth. 9(1960)1-12.

[2] PHADNIS, U.V., KAMBOJ, B.K., SHAHANI, M.G., and SHARMA, D.,6th National Symposium on Radiation Physics, IGCAR,

Table 1. Experimentally determined response of Bonner spheres with thin gold foil detector at the centre to Am-Be neutron source (1.2 * 10^7) n/s at 40 cms.

Sphere Diameter (inches)	Foil Weight mgs.	Saturation Activity c/s	Response dps/gm	Thermal neutron flux n/cm2/s
3	24.5	0.85	53.7	127.9
4	26.36	1.00	60.2	150.2
5	24.5	2.98	190.3	452.4
6	27.34	3.62	221.5	529.6
8	24.1	3.64	249.5	608.4

Table 2 Multisphere Spectrometer Parameters and Data.

1.	Number of Spheres	Seven
2.	Material	Polypropylene
3.	Diameter (inches)	3, 4, 5, 6, 8, 10, 12
4.	Filler Rod	Polypropylene
5.	Detector	1. 1/2 mil and 1 mil thick Gold foil 2. 4 mm * 8 mm LiI(Eu) crystal
6.	Thermal neutron Response	0.3 counts /unit flux for LiI(Eu) 0.4 dps/gm/unit flux for Gold foils
7.	Fast neutron Response (8 inch sphere, Am-Be spectrum)	0.28 counts/unit flux for LiI(Eu) 0.04 dps/gm/unit flux for Gold foil

14 MeV NEUTRON FLUX MEASUREMENT USING IRON FOILS AND ITS INTERNATIONAL INTERCOMPARISON.

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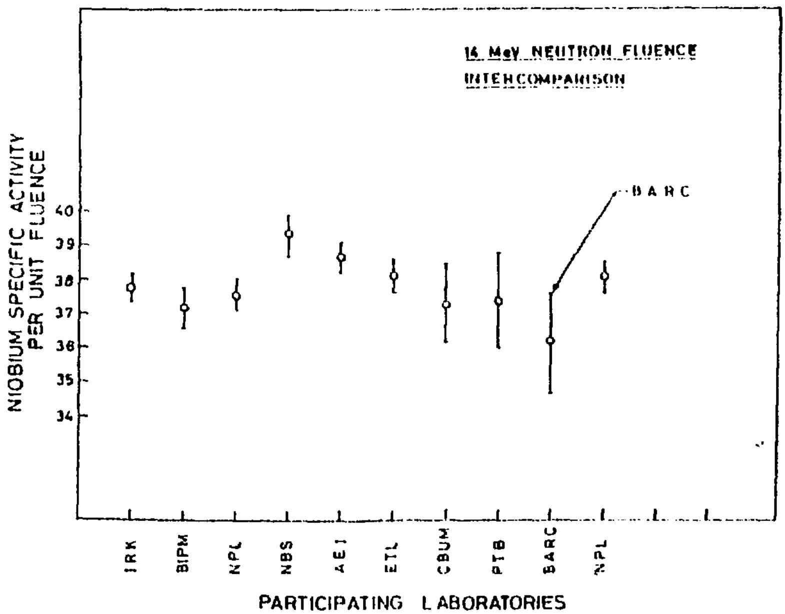
14 Mev fast neutron flux was produced and measured absolutely using the neutron telescope. The telescope measures the flux in terms of n-p scattering cross-section which is considerd a standard cross-section for fast neutron measurement. The telescope was also used to measure the Fe56(n,p) Mn56 reaction crosssection at 14.7 MeV to devlope Fe 56 as secondary fast neutron detector at BARC. This involved a detailed estimation of the corrections due to the elastic and inelastic scattering of neutrons. The value of this crossection was found to be 104 ± 4 %.

An iternational intercoparison with nine standards laboratories around the world was carried out through the auspices of NPL, U.K. The Fe56 was used as detector for the flux measurement at BARC whereas the NPL, U.K. circulated Niobium and Zirconium as the transfer standards for simultaneous irradiation during the flux measurements. This combination of the detectors not only measures the fast neutron flux but also determines the average energy from the knowledge of the ratio of Niobium and Zirconium induced activities. The results of this intercomparison are shown in Figure.



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