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PROGRESS REPORT ON NUCLEAR DATA ACTIVITIES IN INDIA

for the period from July 1984 to December 1985

Compiled by

R. P. Anand
Nuclear Physics Division

1986

B.A.R.C. - 1297

GOVERNMENT OF INDIA
ATOMIC ENERGY COMMISSION

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Descriptors

NEUTRON REACTIONS

FISSION

FISSION YIELD

CROSS SECTIONS

FAST NEUTRONS

MULTIGROUP THEORY

EXPERIMENTAL DATA

COMPUTER CALCULATIONS

RESEARCH PROGRAMS

INDIA

PREFACE

The present progress report on Nuclear Data Activities in India is the fourth in the current new series of progress reports, the first of which was brought out in 1981. This report covers the period from July, 84 to December, 85.

It contains information about nuclear data measurements, compilations, evaluations and other related works being carried out at B.A.R.C., Bombay and R.R.C., Kalpakkam, the two nuclear and reactor research centres in this country.

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

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

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MEASUREMENT OF THE FAST NEUTRON INDUCED FISSION CROSS-SECTION OF TH-232 RELATIVE TO U-238

R.P.Anand,K.N.Iyengar,N.N.Ajitanand
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Neutron induced fission cross-sections for ^{232}Th are of interest in the energy region of fission neutrons as they provide important Nuclear Data. Some measurements of these cross-sections have been made in the past /1-4/ using thin targets. With a view to carry out measurements of low cross-sections in future, we have employed thick targets to measure fission cross-sections.

EXPERIMENTAL METHOD

As shown schematically in Fig.1, the method consists of exposing a sandwich of 1 cm diameter, 100 mg/sq.cm. thick thorium and depleted uranium (i.e. 99% U238) discs with a 100 μm thick lexan plastic in between, to neutrons from a 1 Ci Tritium target bombarded by protons from a Van de Graaff accelerator. The distance between the neutron source and the sandwich is 2.0 cm. Irradiations were carried out with typical proton beam currents of one μA on target. The neutron energy spread at the target was estimated to be about 100 keV. The lexan pieces were etched by a standard procedure to reveal fission tracks on both surfaces of the lexan /5/. The counting was done to include only those tracks which had diameters larger than a background discriminating value. The ratio of cross sections was then given by the ratio of track densities on the two surfaces of the lexens.

RESULTS AND DISCUSSION

The results of the present measurements are shown in table I along with those of the measurements of Blons /4/ averaged over the energy resolution of the present experiment in converting the ratios to ^{232}Th cross-sections, the ^{238}U cross-sections were taken from a standard data file /5/. The statistical error of each measurement is about 10 percent. From the present results it is seen that the thick target method has an accuracy of about 30 percent. With this limitation it can well be used for measurements of very low fission cross-sections.

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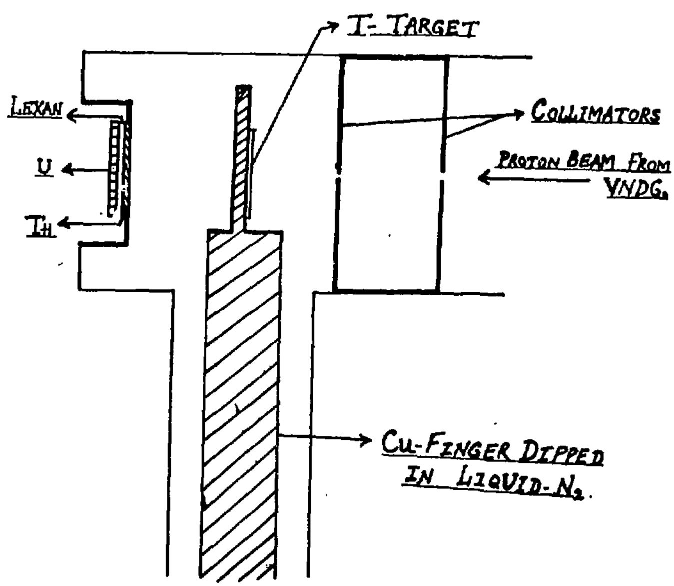
TABLE I

En (MeV)	σ(n,f) present (barns)	σ(n,f) Blons /4/ (barns)
1.38	0.046	0.057
1.52	0.088	0.085
1.59	0.123	0.097
1.65	0.127	0.092
1.71	0.105	0.073
1.85	0.121	0.091
1.95	0.086	0.119

Fig.1 Experimental arrangement



MULTIPARTICLE REACTION CROSS SECTIONS IN THE ENERGY RANGE OF 10 TO 50 MeV

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Neutron induced multiparticle reaction cross-sections for several elements such as V,Cr,Mn,Fe,Ni,Zr,Co,Nb,Pb and Bi in the energy range of 10 MeV to 50 MeV are of direct interest to fusion,fission-fusion and spallation based applications.The cross-section data are also needed to carry out the shielding and radiation damage studies. The important reactions in the above mentioned energy range are (n,2n),(n,3n),(n,4n),(n,np),(n,npn),(n,2np),(n,p), (n,pn),(n,p2n),(n,α),(n,αn),(n,nα),(n,nαn) and (n,α2n). The measured cross-sections on these reactions are generally sparse.To meet the needs of the data users, the cross-sections are generated with several appropriate nuclear models.

We have used the code ALICE/85/300 /1/ based on the Geometry Dependant Hybrid Model /2/ and the Weisskopf-Ewing Evaporation Model /3/ to compute the cross-sections for the above listed reactions and nuclides.The energy spectra and angular distributions of the emitted neutrons and protons are also calculated.

A study of these reactions based on the multistep Hauser- Feshbach statistical model with the precompound corrections taken into account is also being carried out in order to understand the salient points of agreement between these various approaches.

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PRE-EQUILIBRIUM EFFECTS IN NEUTRON,PROTON AND ALPHA EMISSION OF Nb-93 REACTION+

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Pre-equilibrium emission of nucleons and alpha particles during a nuclear reaction induced by nucleons of energy 10 MeV or more is considered to be one of the main mechanisms of the interaction. Pre-equilibrium theory with its closed form and master equation approaches has been employed to explain the energy spectra and angular distributins of the emitted particles.

In the present study we have used the multistep direct and multistep compound mechanisms of Kalbach /1/ and modified exciton model of Manzouranis et al /2,3/ to describe the proton,neutron and alpha emission together with their spectra and angular distributions in the ineraction of neutrons with Nb-93.The level density parameters obtained with the modified Gilbert-Cameron formulation have been used for the target and other involved nuclides. The inverse reaction cross-sections for neutrons,protons and alpha particles have been evaluated with appropriate optical model parameters using SCAT2 /4/ code.

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MULTIGROUP CROSS-SECTIONS OF MINOR ACTINIDE ELEMENTS

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Growing nuclear power programmes around the globe have generated a lot of interest in the isotopes of several minor actinide elements such as Np, Am, Cm, Cf etc. The interest arises from the potential application of some of these nuclides in fusion reactor blankets as well as in studies pertaining to the transmutation of these long lived nuclides to short lived ones in radioactive wastes generated in nuclear reactors. The criticality aspects of some of these nuclides are relevant from considerations of nuclear safety /1,2/ as well as from safeguards /3/ point of view.

In order to carryout the criticality studies based on the odd-neutron isotopes of the above mentioned elements 35-group cross-sections with P_3-anisotropic scattering matrices have been generated from the recent basic point cross-section library ENDL /4/ using the code MINX /5/ which has been modified for operation on CDC-Cyber computer.

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ADAPTATION OF MULTIGROUP CROSS-SECTIONS FOR NEUTRONICS STUDIES IN NUCLEAR REACTORS

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Neutron multigroup cross-sections are the basic inputs to carry out the neutronics studies of reactor systems. 35-group cross-sections for several elements /1/ have been generated using the basic point cross-section libraries ENDF/B-IV/2/ and recent ENDL /3/ using the MINX /4/ code.

In order to make use of the generated multigroup cross-sections in neutron diffusion and transport theory based computations the cross-sections have to be transformed into specially formatted tables. To accomplish this task a code 12D /5/ has been modified for operation on CDC-Cyber computer. Using this code 35-group cross-sections with P_3-anisotropic scattering matrices have been converted into DTF4/6/ and ANISN /7/ formats so that the data can be used in the reactor neutronics and safety studies.

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SENSITIVITY STUDIES OF NEUTRON MULTIPLICITY SPECTRUM IN THE SPALLATION OF LEAD TARGETS+

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The number of neutrons produced per incident proton in the spallation of lead targets is of direct relevance to the design of accelerator breeders.The nuclear cascade initiated by high energy protons in spallation targets is usually described by an IntraNuclear Cascade Evaporation (INCE) /1/ model.Eventhough this model describes various average nuclear properties of spallation targets fairly well,differential quantities such as energy spectra,angular spectra etc. are not reproduced within the limits of experimental uncertainty /2/.One of the reasons for this is the uncertainty in the magnitude of parameters involved in the model,notably the level density parameter B_0 whose magnitude is quoted by different workers /1/ to be in the range of 8 to 20 MeV.The accuracy of B_0 could be improved if we could experimentally determine a quantity which is much more sensitive to B_0 than the average neutron yield.In this paper we discuss one such quantity namely the neutron Multiplicity Spectrum (MS). We compute MS due to the spallation of lead targets of different sizes at oton energies of 1.5 GeV, 1.0 GeV, 0.59 GeV using the Monte Carlo code HETC /3/. It is noticed that for the 1.5 Gev proton case the probability $p(\nu)$ for leakage of ν neutrons in the range of 60 to 65 changes by about 70% when B_0 is varied from 8 to 20 MeV.The corresponding change in the average neutron yield is less than 20%. It is therefore suggested that an accurate measurement of MS can serve as a useful tool to narrow down the range of uncertainty in the B_0 parameter. The Multiplicity Spectrum (MS) of leakage neutrons can be deduced from a statistical analysis of the correlated coincidence counts /4/ using a bank of high efficiency neutron detectors located around the spallation target assembly.A similar approach has been suggested by Srinivasan /5/ for the study of multiplication of 14 MeV neutrons in Be assembly.

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+ Paper to be presented at the International Conference On Emerging Nuclear Energy Systems,Madrid,Spain,1986.

ANAMOLOUS BEHAVIOUR OF SUB-COULOMB PROTON INDUCED FISSION CROSS-SECTION OF U-235 AND U-238

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Sub-coulomb fusion has been studied extensively /1/ and it has been established that for heavy projectiles the cross-section is enhanced with respect to the predictions based on a static barrier. Such enhancement is absent in reactions with very light projectiles /2/. With a view to investigating the situation existing at very low energies we have carried out the present measurements.

EXPERIMENTAL METHOD

Targets of 1.22 mgm cm^{-2} ^{235}U and 490 μgm cm^{-2} ^{238}U were used to make the measurements. As shown schematically in Fig.1, each target was bombarded with the proton beam from a Van de Graaff accelerator and confronted with an annular lexan detector of thickness 75 μm which detected fission fragments due to fission induced by both, protons and background neutrons generated by the proton beam. The beam was stopped in 2mm thick aluminium after which another target(uranium target)of similar dimensions, faced by a similar annular lexan detector, was used to measure the background fission due to the neutrons alone. The assembly containing the two targets also served as the Faraday cup to measure the total number of protons seen by the target. All the lexan pieces were etched by a standard procedure to reveal fission tracks /3/. By appropriately subtracting the track counts in the second lexan from the first and dividing by the geometrical detection efficiency, the total fissions and hence the cross-section due to proton induced fission was obtained.

RESULTS AND DISCUSSION

The results obtained are shown in Table 1. It is clear from the optical model calculation of the reaction cross-section /4/ that fission following proton capture cannot account for the large observed cross-section below 4 MeV. The distance of closest approach is so large (see table) that only coulomb excitation of low lying levels in the target is likely to take place. The coulomb excited states have to face the fission barrier of ^{235}U which is known to be double humped and the bottom of the second well is about 3 MeV above the ground state. However, if the potential energy diagram were to have an additional shallow pocket whose bottom is a little above the ground state, fission would be greatly enhanced due to the coupling of states in the first and third wells. Some indication of this is seen in the potential energy surface calculated for some of the actinides /5/.

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TABLE I

Ep (MeV)	σ_R Optical Model (mb)	σ_f measured (mb) ^{235}U	^{238}U	closest approach (fm)
4.0	1.0 (10^{-4})	5.6(10^{-5})		33
3.5	6.5 (10^{-6})	1.4(10^{-5})		38
3.0	1.7 (10^{-7})	8.0(10^{-6})	3.5(10^{-5})	44
2.5	1.7 (10^{-9})	1.7(10^{-6})	1.0(10^{-6})	53
2.0	2.8 (10^{-12})	4.5(10^{-6})		66

NEUTRON RADIATIVE CAPTURE CROSS SECTION OF Th-232

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The neutron capture cross section of Th-232 was calculated by computer code FISPRO-II in the energy range 0.2 to 1.6 MeV. This code is based on the statistical model - Hauser-Feshbach (H.F.) theory. We have made calculations using both Weisskopf's and Axel's prescriptions for the energy dependence of $\overline{\Gamma_\gamma}$ and used the following parameters set :

$$\overline{\Gamma_\gamma}(B_n) = 20 \text{ meV}, \quad a = 28 \text{ MeV}^{-1}, \quad S = 1.223$$

The optical model parameters set :

$$V_R = 47 \text{ MeV}; \quad V_I = 5 \text{ MeV}; \quad a_R = a_I = 0.60 \text{ fm}; \quad R_R = R_I = 1.25 \text{ fm}$$

We have investigated the sensitivity of the various parameters to the calculated cross section by varying them systematically. In Fig.1 we have compared the calculated values with the measured and evaluated data.

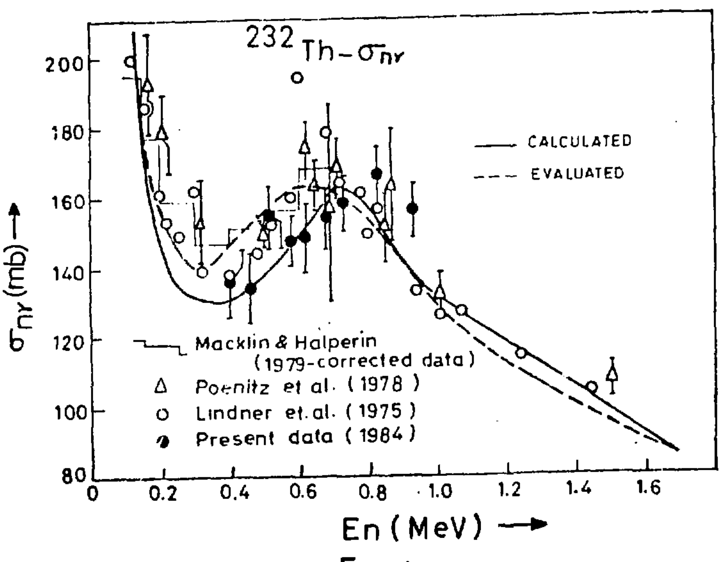

Fig. 1

Nuclear Data activities of Nuclear chemistry section

Naik H., Guin R., Das S.K., Nair A.G.C., Singh R.J., Srivastava B.K., Chakravarty N., Rattan S.S., Goswami A., Datta T., Ramaswami A., Dange S.P., Sahakundu S.M., Manohar S.B., Satya Prakash. and Ramaniah M.V.
Radiochemistry Division. BARC.

1. Nuclear Fission.

For charge distribution studies in spontaneous, neutron and helium ion induced fissioning systems, fractional independent yield 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 were determined for certain fission products in spontaneous, neutron and helium ion induced fissions. The values are reported in Table 2. Mass yields for certain fission products were determined for 30 MeV helium ion induced fission of ^{232}Th are given in Table 3. Experimental cross sections for the formation of certain fission products in helium ion induced fission of ^{209}Bi are reported in Table 4.

2. Helium ion induced reaction cross section measurements.

Cross section measurements for the production of certain reaction products in helium ion induced reactions on ^{27}Al, ^{63,65}Cu, ^{209}Bi, ^{244}Cm are reported in Table 5 8.

Table 1. Fractional and cumulative Yield measurments in spontaneous, neutron and Helium ion induced fissions.

Sr.No.	Fissioning Nucleus	Fission Product	F.I.Y.	F.C.Y.
1	$^{239}Pu(n_{th}, f)$	^{101}Tc	0.0018 ± 0.0012	0.0018 ± 0.0012
		^{102}Tc	0.0063 ± 0.0033	0.0081 ± 0.0035
		^{103}Tc	0.0301 ± 0.0131	0.0382 ± 0.0136
		^{104}Tc	0.067 ± 0.0152	0.1052 ± 0.0204
		^{105}Tc	0.1358 ± 0.0389	0.2410 ± 0.0439
2	$^{241}Pu(n_{th}, f)$	^{105}Mo		0.499 ± 0.011
		^{105}Tc		0.929 ± 0.034
		^{91}Sr		0.9677 ± 0.0009
		^{92}Sr		0.9212 ± 0.0011
		^{134}Te		0.655 ± 0.016
3	$^{252}Cf(SF)$	^{140}Ba		0.998 ± 0.002
		^{142}Ba		0.915 ± 0.034
		^{146}Ce		0.989 ± 0.006
4	$^{232}Th(\alpha_{30\,MeV}, f)$	^{135}I		0.766 ± 0.02
		^{138}Cs		0.813 ± 0.03
		^{140}Ba		0.991 ± 0.004

Table 2. Isomeric yield Ratios of fission products

Sr.No.	Fissioning Nucleus	Fission product	Isomeric yield ratio
1.	$^{252}Cf(SF)$	^{111}Pd	0.586 ± 0.070
		^{117}Cd	0.732 ± 0.088
		^{138}Cs	0.582 ± 0.068
2.	$^{233}U(n_{th},f)$	^{134}I	0.429 ± 0.028
3.	$^{235}U(n_{th},f)$	^{134}I	0.412 ± 0.046
4.	$^{239}Pu(n_{th},f)$	^{134}I	0.394 ± 0.047
5.	$^{241}Pu(n_{th},f)$	^{134}I	0.362 ± 0.032
6.	$^{232}Th(\alpha_{30}$ MeV, f)	^{131}Te	0.516 ± 0.050
		^{133}Te	0.593 ± 0.080
7.	$^{232}Th(\alpha_{40}$ MeV, f)	^{131}Te	0.506 ± 0.061
		^{133}Te	0.632 ± 0.070

Table 3. Mass yields* in ^{232}Th (α_{30} MeV, f)

Sr.No.	Nuclide	Yield (%)
1.	^{78}As	0.225
2.	^{85m}Kr	1.448
3.	^{87}Kr	2.347
4.	^{88}Kr	2.794
5.	^{89}Rb	3.33
6.	^{91}Sr	3.379
7.	^{92}Sr	3.412
8.	^{93}Y	4.335
9.	^{97}Zr	3.97
10.	^{99}Mo	4.52
11.	^{103}Ru	3.62
12.	^{104}Te	3.00
13.	^{105}Rh	2.717
14.	^{107}Rh	1.832
15.	^{111}Ag	1.294
16.	^{112}Pd	1.325
17.	^{113}Ag	2.124
18.	^{115}Cd	1.78
19.	^{117}Cd	0.887
20.	^{123m}Sn	1.754
21.	^{127}Sb	2.550
22.	^{129}Sb	3.256
23.	^{131}I	4.743
24.	^{133}I	5.706
25.	^{138}Cs	5.408
26.	^{139}Ba	4.794
27.	^{140}Ba	5.083
28.	^{141}Ce	3.877
29.	^{142}La	2.240
30.	^{143}Ce	2.371
31.	^{151}Pm	0.375

* Precision of detertrmination: ±10 to 15 %

Table 4. Experimental cross section in μb for the formation of fission products in the alpha induced fission of ^{209}Bi.

Fission Product	49.1 Mev alpha	44.9 Mev alpha
^{85m}Kr	173 ± 48	
^{91}Sr		395 ± 38
^{92}Sr	450 ± 158	
^{95}Zr	793 ± 72	
^{97}Zr	715 ± 150	
^{99}Mo	918 ± 95	497 ± 35
^{103}Ru	1109 ± 170	653 ± 95
^{104}Tc	804 ± 80	
^{105}Ru	990 ± 67	
^{112}Pd	657 ± 66	483 ± 68
^{115}Cd	382 ± 77	

Table 5. Experimental cross sections (mb) for the production of ^{22}Na, ^{24}Na and ^{28}Mg by irradiation of alpha particles.

Sr.No.	Eα (MeV)	^{22}Na	^{24}Na	^{28}Mg
1.	44.3	28.81 ± 1.64	1.52 ± 0.05	0.198 ± 0.055
2.	45.4	31.88 ± 0.46	1.77 ± 0.05	0.206 ± 0.003
3.	48.6	43.36 ± 2.09	3.73 ± 0.02	0.310 ± 0.011
4.	49.6	48.79 ± 0.84	4.35 ± 0.08	0.299 ± 0.003

Table 6. experimental cross sections (mb) for production of reaction products by irradiation of $^{63,65}Cu$ with alpha particles of 47.0 MeV.

Reaction Products	Cross section (mb) for		
	$^{63,65}Cu$	Only ^{63}Cu	Only ^{65}Cu
^{57}Co	1.738 ± 0.065	2.51 ± 0.09	
^{58}Co	22.96 ± 0.25	33.2 ± 0.4	
^{61}Cu	57.66 ± 1.27	83.4 ± 1.8	
^{64}Cu	98.08 ± 7.57		318.1 ± 24.6
^{65}Zn	1.987 ± 0.023		6.45 ± 0.07
^{66}Ga	83.96 ± 0.34		272.0 ± 1.1
^{67}Ga	26.88 ± 0.84		87.2 ± 2.7

Table 7. Experimental cross sections (mb) for production of ^{208}At, ^{209}At and ^{210}At by alpha irradiation of ^{209}Bi.

Sr.No.	Eα (MeV)	^{208}At	^{209}At	^{210}At
1.	44.9	0.628	891 ± 78	408 ± 48
2.	49.1	68.9 ± 3.2	968 ± 122	197 ± 35

Table 8. Cross section for ^{244}Cm (α,2n) ^{246}Cf

Sr.No.	Eα (MeV)	Cross Section (mb)
1.	33.2	8.4 ± 0.6
2.	44.4	7.1 ± 0.5

HALF LIFE OF ^{241}Pu

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Beta decay half-life of ^{241}Pu is of great importance in nuclear technology. In view of large variation in the values (13-15 yr) reported till 1974 in literature, efforts have been made in different international laboratories to determine this half-life with high precision and accuracy. In our laboratory, it has been determined by different methods which may be classified in two categories viz. (i) Parent decay method and (ii) Daughter growth method. In the parent decay method, change in isotope ratios $^{241}Pu/^{239}Pu$, $^{241}Pu/^{240}Pu$ and $^{241}Pu/^{242}Pu$ was studied periodically by a thermal ionisation mass spectrometer. Single as well as double ratio method was used to calculate the half-life. In the daughter growth method, the half-life was obtained in four independent ways, by determining the ingrowth of ^{241}Am. These were (i) Alpha spectrometry taking ^{239}Pu and ^{242}Pu separately as reference isotopes and studying periodically the increase in alpha activity ratios: $(^{238}Pu + ^{241}Am) / (^{239}Pu + ^{240}Pu)$ and $(^{238}Pu + ^{241}Am) / ^{242}Pu$ (ii) Alpha proportional counting for observing periodically the change in total alpha activity (iii) Isotope dilution alpha spectrometry using ^{243}Am as a spike and measuring $^{241}Am/^{243}Am$ alpha activity ratios (iv) Isotope dilution mass spectrometry using ^{243}Am as a spike and determining $^{241}Am/^{243}Am$ atom ratios. In these methods, synthetic mixtures were prepared for achieving high precision and accuracy in different measurements. In the alpha spectrometric work, the synthetic mixtures were prepared with an objective of obtaining 100% change in the alpha activity ratio after a period of about 30 days. Further, the tail contribution at low energy peaks, (^{239}Pu + ^{240}Pu) and ^{242}Pu, due to energy degradation of high energy peak (^{238}Pu + ^{241}Am) was accounted for by using a method based on the geometric progression decrease for the far tail of the spectrum. In mass spectrometry, isotope ratios were determined with high precision (better than 0.1%) and the double ratio method takes care of isotope fractionation effects which can lead to variable systematic error.

Based on the six independent measurements from this laboratory from 1980-84, a mean value of 14.42 $\pm$ 0.07 yr is obtained. The Advisory Group on Transactinium Isotope Decay Data in IAEA has recommended values of 14.7 $\pm$0.4 yr in 1979 and 14.4 $\pm$ 0.2 yr in 1983. The data obtained on the half-life values have been published at various times and are given in the Table.

S.No.	Reference	Half-life (yr)
1.	Seaborg et al, The Transuraium elements, McGraw-Hill, 1949 p.22.1Div.IV, Vol.14B	10
2.	Thompson et al, Phys.Rev.,80,1108 (1950)	15.37
3.	Mackenzie et al,Phys.Rev.,90,327 (1953)	14.2±0.2
4.	Rose et al, J.Nucl.Energy,2,264 (1956)	13.87±0.28
5.	Brown et al, J.Inorg.Nucl.Chem.,13,192(1960)	14.12±0.24
6.	Smith, J.Inorg.Nucl.Chem.,17, 178 (1961)	13.3±0.3
7.	French et al, WCAP-6082 (1966)	13.59±0.46
8.	Nisle and Stepan, Nucl.Sci.Eng.,39,257 (1970)	14.63±0.27
9.	Shields, NBS Tech.Note 546, 25 (1970)	14.6±0.4
10.	Whitehead et al, AERE-PR/NP 18 (1972)	14.96±0.15
11.	Zeigler and Ferris, J.Inorg.Nucl.Chem., 35,3417 (1973)	14.89±0.11
12.	Strohm and Jordan, Trans.Am.Nucl.Soc., 18,185 (1974)	14.355±0.00
13.	Wilkins, AERE-R 7906 (1974)	15.02±0.10
14.	Whitehead, UKNDC (76) P86, 41 (1977)	14.56±0.10
15.	Crouch, UKNDC (78) P88, 97 (1978)	14.24-14.53
16.	Garner, Trans.Am.Nucl.Soc, 33,3 (1979)	14.38±0.07
17.	Vaninbroukx, Int.Conf. Neutron Phys. and Nucl. Data for Reactors, OCDE Nucl. Energy Agency,Paris, 235 (1978)	14.30±0.14 14.60±0.10
18.	Aggarwal and Jain,Phys.Rev.,C21,2033 (1980)	14.42±0.09
19.	Marsh et al,Int.J.Appl.Rad.Isot.,31,629(1980)	14.32±14.43
20.	Aggarwal et al, Phys.Rev.C23, 1748 (1981)	14.37±0.09 14.50±0.08
21.	Aggarwal et al, Radiochim.Acta, 29,65 (1981)	14.52±0.08
22.	Aggarwal et al, Radiochem.Radioanal.Lett. 54, 83 (1982)	14.32±0.11
23.	Takashi, PNCT-N-831-82-01 (1982)	14.29±0.15
24.	DeBievre et al, Int.J.Mass Spectrom. Ion Phys., 51, 111 (1983)	14.33±0.02
25.	Aggarwal et al, Phys.Rev.,C31, 1885 (1985)	14.43±0.08 14.38±0.02

FISSION YIELDS IN THE THERMAL NEUTRON INDUCED FISSION OF ^{233}U, ^{235}U, ^{239}Pu AND ^{241}Pu

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Fission yields in the thermal neutron fission of ^{239}Pu

Fission yields for 27 mass numbers were determined in the thermal neutron fission of ^{239}Pu using high resolution gamma ray spectrometry and radiochemical method. In the radiochemical method, the total number of fissions were calculated without considering the neutron temperature correction to the ^{239}Pu fission cross-section. A recalculation of the fission yeild data of ^{239}Pu incorporating the (g+rs) correction for the neutron temperature was carried out. This has resulted in resolving the difference between the earlier published high values for ^{99}Mo fission yield in ^{239}Pu (which was used as a reference nuclide) and other recently reported values. Details of these calculations and the results obtained along with the comparison with the experimentally determined values and with the yields given in two recent compilations are published in Radiochim. Acta, Vol.37, p.63(1984).

Fission yields in the thermal neutron fission of ^{233}U, ^{235}U, ^{239}Pu and ^{241}Pu

The fission yields for 40 mass numbers in each of the fissioning systems $^{233}U(n_{th},f)$, $^{235}U(n_{th},f)$, $^{239}Pu(n_{th},f)$ and $^{241}Pu(n_{th},f)$ were determined employing mass spectrometric, radiochemical and gamma spectrometric methods.

In $^{233}U(n_{th},f)$, fission yields for ^{113}Ag and ^{123}Sn were determined for the first time. These yields along with those for ^{111}Ag, ^{112}Ag, ^{115}Cd, ^{121}Sn and ^{125}Sn determined in the present work have led to sharp dips at mass numbers 112 and 123 in the symmetric region of the mass yield curve resulting in the appearance of a third peak.

In $^{239}Pu(n_{th},f)$, fission yields for ^{98}Mo and ^{100}Mo were found to be 10-12% higher than the only one experimental data reported in literature. Fission yields for these nuclides and that for ^{103}Ru are seen as spikes in the light wing of the mass yield curve.

In $^{241}Pu(n_{th},f)$, fission yields of ^{98}Mo, ^{100}Mo, ^{121}Sn and ^{123}Sn were determined for the first time. A peak to valley ratio of 600 was obtained compared to a value of 230 arrived at from estimated values of fission yields in the symmetric region reported in literature.

Adequacy of Indian Cross-section Libraries in the Reactivity Prediction of Thorium System

by

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1.1 Introduction

Reactor physics design in India is normally done with the help of cross-section libraries acquired from advanced countries. These libraries, created by processing ENDF/D data, tend to emphasize the requirements of uranium systems. This makes them somewhat less desirable for thorium systems; more so in the resolved resonance energy range. The adequacy of these libraries for thorium systems has been assessed by using them for predicting the effective multiplication factor of experimentally measured lattices.

1.2 The reactor physics design methods used here i India are on three different level : the two group, the few group, and the multigroup treatments. One representative code from each level has been used for the present analysis.

1.2.1 Dumlac - This code uses Westcott corss-sections in the thermal region, Hellstrand's measured resonance integrals in the resonance region, and cross-sections integrated over the fission spectrum in the fast region.

1.2.2 Rhea - This is a few group code which uses Ombrellaro cross-sections in the fast range and Amster corss-sections in the thermal range.

1.2.3 Hygea - This multigroup code makes use of the Muft cross-section set in the fast region and the Thermas library in the thermal region.

2. Measurements and Calculations

Literature survey for thorium D_2O systems, where the primary interest of India lies, revealed some lattice measurements conducted at AECL and some at BNL.

The result are summarised in the following table.

Keff values as calculated

Pitch (cms)	DUMLAC	RHEA	HYGEA
AECL Experiments D_2O cooled. (ThO_2 + 1.5% U235 O_2)			
22.0	0.985	0.996	
24.0	0.987	0.997	
28.0	0.988	0.998	
BNL Experiments (Th, U233)O_2 2.5% U233			
2.17	1.111	0.973	1.005
3.76	0.995	0.994	0.997
4.34	0.996	1.000	0.996
5.74	0.985	1.003	0.988
6.51	0.988	0.996	0.993
7.82	0.973	1.006	0.981
9.46	0.975	1.016	0.985
11.48	0.971	0.998	0.980
AECL Experiments (Th, U233) H_2O cooled 1.5% U233			
22.0		1.004	
24.0		1.007	
28.0		1.008	
32.0		1.010	

3. Conclusions

It would appear that at least as far as reactivity predictions go, these libraries are acceptable. For thorium reactors however, the most important as well as the most sensitive quantity is the conversion ratio. Due to the paucity of experimental information, it has not yet been possible to evaluate the adequancy of our libraries for this quantity. Efforts are being made to acquire relevant data in this regard.

KALPAKKAM MULTIGROUP CROSS SECTION SET FOR FAST REACTOR APPLICATIONS - STATUS AND PERFORMANCE

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The newly generated Kalpakkam multigroup set was used to predict integral quantities such as k_{eff}, central reaction rates, central reactivity worths[1] for three fast critical assemblies viz. ZPR-3-48, ZPR-3-56B and ZPR-6-7[2] using the ENDF/B-IV as the basic file.

The ratios of calculated to experimental values were compared for the ENDF/B-IV based American multigroup set[3], JENDL based Japanese multigroup set[4], ENDF/B-IV based Japanese multigroup set and the adjusted French cross section set[5].

The values of calculated K_{eff}'s for the three assemblies are summarised below

Assembly →	ZPR-3-48	ZPR-3-56B	ZPR-6-7
JENDL-1	1.0005	0.9957	0.9983
JFS-2	1.0031	0.9967	1.0033
ENDF/B-IV JEARI	0.9885	0.9768	0.9810
ENDF/B-IV Hardie et al.	1.0015	0.9882	0.9917
ENDF/B-IV Kalpakkam	0.9946	0.9979	0.9867
French set	1.0069	1.0164	1.0049
δ K* ENDF	.0130	.0211	.0107

* Maximum spreads on K_{eff} calculated by different sets all based on ENDF/B-IV only.

The results obtained for central reactivity worths for ^{235}U, ^{238}U, ^{239}Pu, Fe, Cr, Ni and Na and those normalised to ^{239}Pu, ratio of capture reaction rate of ^{238}U to the fission rates of ^{238}U, ^{239}Pu and ^{240}Pu to the fission rate of ^{235}U are given in Ref.6.

<u>References</u>

1. M.M. Ramanadhan, V.Gopalakrishnan and S.Ganesan, Indian Journal of Physics, Vol.58A Number 6 (Nov. 1984).

2. Cross Section evaluated working group Benchmark Specification BNL 19302, ENDF-202.

3. R.W.Hardie, et al., Nucl. Sci. Eng. 9, 57 222-238 (1975).

4. Yasuyuki Kikuchi et al.,Benchmark Tests of JENDL-1, JEARI 1275, 1982.

5. J.Ravier and T.M.Chaumont, Presentation of the Multigroup cross section set prepared at Cadarache, Proceedings of the Conference on Fast Critical Experiments and their analysis, USAEC report ANL-7320, 1966.

6. M.M.Ramanadhan et al.,to be published (1985).

MULTIGROUP CROSS SECTIONS FOR ALKALI ISOTOPES AND ACTINIDES

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Average cross sections of all available neutron induced nuclear reaction processes were calculated in the SETR 25 group structure from available basic data files in the ENDF/B Format obtained from IAEA Nuclear Data Section for the alkali isotopes and actinides given in Table 1. The basic data was preprocessed by LINEAR[1], resonance reconstructed by RECENT[2] and multigrouped by REX1[3]. Only infinite dilution cross sections were obtained, using a typical fast reactor spectrum. The multigroup cross sections and collapsed one-group cross sections for alkali metals[4] were generated on a request from the Radio Chemistry Programme of our Centre, and for actinides[5] on a request from Shielding Group of our Section.

1. D.E. Cullen, 'Program LINEAR', UCRL-50400, Vol. 17, Part A, Lawrence Livermore Laboratory (1979).
2. D.E. Cullen, 'Program RECENT', UCRL-50400, Vol. 17, Part A, Lawrence Livermore Laboratory (1979).
3. V. Gopalakrishnan and S. Ganesan, 'A Note on the Program REX1.....', Internal Note REDG/RP-243 (1983).
4. V. Gopalakrishnan and M.M. Ramanadhan, 'Multigroup data for alkali Metals', Internal Note REDG/RP-267 (1984).
5. V. Gopalakrishnan and M.M. Ramanadhan, 'Multigroup data for Actinides', Internal Note to be published (1985).

Table 1

ISOTOPES PROCESSED AND THE BASIC DATA FILES FROM WHERE THEY WERE CHOSEN.

Nuclides	Basic Data File
Li-6, Li-7, Na-23, K(nat), Cs-133, Th-232, U-233, U-234, U-235, U-236, U-238, Np-237, Pu-238, Pu-239, Pu-240, Pu-241, Pu-242, Cm-244.	ENDF/B-IV General purpose.
Rb-85.	ENDF/B-IV Fission Products.
U-237, Np-238, Pu-243, Am-241, Am-242, Am-242 Am-243, Cm-242, Cm-243.	ENDF/B-V Actinides.
Np-239.	JENDL-1
U-239	ENDL-78.

INTEGRAL VALIDATION OF (N, 2N) CROSS SECTIONS OF TH-232 BY ANALYSES OF MEASURED CENTRAL REACTION RATE RATIOS IN THOR ASSEMBLY

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The (n, 2n) cross sections of Th-232 plays an important role in predicting the production of U-232 which leads to hard gamma rays in thorium cycle. The present work attempts to intercompare and validate in an integral sense the (n, 2n) cross sections of various available basic data libraries for Th-232.

Experimental values of spectral ratios given are as follows :

$$\langle \sigma_{n,\ 2n}\ (\text{U-238}) \rangle / \langle \sigma_f (\text{U-238}) \rangle = 0.053 \pm 0.003$$

$$\langle \sigma_f\ (\text{Th-232}) \rangle / \langle \sigma_f (\text{U-238}) \rangle = 0.26 \pm 0.01$$

$$\langle \sigma_{n,\ 2n}\ (\text{Th-232}) \rangle / \langle \sigma_{n,\ 2n}\ (\text{U-238}) \rangle = 1.04 \pm 0.03$$

The direct interference from (n, 2n) and fission cross sections of U-238 while interpreting the effects of changes in cross sections for Th-232 is eliminated as follows. We cross multiply and eliminate the cross sections of U-238 and obtain the ratio of experimental value of effective one group (n, 2n) cross section to one group effective fission cross section for Th-232 at the centre of THOR assembly as:

$$\left[\frac{\langle \sigma_{n,\ 2n} \rangle}{\langle \sigma_f \rangle} \right]_{\text{EXP.}} = 0.212 \pm 0.0157 = \mathcal{E}$$

This ratio is calculated using various data files and fluxes at core centre of THOR assembly using

$$\left[\frac{\langle \sigma_{n,\ 2n} \rangle}{\langle \sigma_f \rangle}\right]_{CALC.} = \frac{(\bar{\sigma}_{n,\ 2n})_1\ \bar{\phi}_1}{\sum\limits_{g=1}^{4} \bar{\sigma}_{fg}\ \bar{\phi}_g}$$

where g is the energy group index. The fluxes $\bar{\phi}_g$ are calculated using one dimensional transport theory code DTF-IV and the group cross sections $\bar{\sigma}_{n,2n}$ and $\bar{\sigma}_f$ are derived from various basic nuclear data files.

The error in "C" is taken as 4% as the uncertainties in cross sections of Pu-239 and those of Th-232 influence "C" by as much as 4%, on the average, through corresponding uncertainty in calculated neutron energy spectrum. This coupled with the error in"E" of 7.4% gives a net error of 8.4% in C/E. The results are given below :

File	Calculated $\frac{\langle \sigma_{n,2n} \rangle}{\langle \sigma_f \rangle}$	C/E Error : + 8.4%	$\frac{\langle \sigma_{n,\ 2n} \rangle}{\langle \sigma_{f(JENDL-2)} \rangle}$	$\frac{1}{E} \cdot \frac{\langle \sigma_{n,\ 2n} \rangle}{\langle \sigma_{f(JENDL-2)} \rangle}$ Error $\pm$ 10%
ENDF/B-IV	0.283	1.335	0.257	1.21
ENDF/B-V	0.255	1.202	0.243	1.15
JENDL-2	0.225	1.063	0.225	1.06
ENDL-84	0.259	1.221	0.257	1.21
INDIAN	0.258	1.22	0.247	1.17
JNDL/A-83	0.270	1.275	0.250	1.18

Except JENDL-2 file all other data files overestimate the $\langle \sigma_{n,2n} \rangle / \langle \sigma_f \rangle$ ratio. The entire (n, 2n) contribution comes only from the first group in our present study. From the results presented in the above table we find that JENDL-2 is consistent within the interpretational error bar and the other files show an overprediction of (n, 2n) cross section by about 5 to 11% outside the interpretational uncertainty which is assessed to be about 10% for the last column.

AN ATTEMPT OF INTEGRAL VALIDATION OF U-233 FISSION AND CAPTURE CROSS SECTIONS IN KEV-MEV ENERGY REGION USING IRRADIATION IN RAPSODIE FORTISSIMA FAST TEST REACTOR

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RAPSODIE FORTISSIMA reactor at Cadarache has been used to irradiate some of the actinides and fission products with a view to determine the integral neutron cross sections of the irradiated nuclides and fission yields of the actinides in a fast reactor. This experiment was known as TACO (TAUX de COMBUSTION) experiment, in which the irradiation took place from February 1971 to June 1972 giving a total irradiation equivalent to 263.5 days at full power.

Cricchio et al.[1] have analysed the above experiment and have calculated the integral capture and fission cross sections for U-233, Np-237, Am-241, Am-243 and a few fission products. They have also compared[1] these (experimental) values with those calculated from the basic cross section data library KEDAK. The multigroup structure they have used is the SETR structure which is extensively used in RRC. This gives us an opportunity for comparing and validating the integral cross sections calculated from various data libraries. Presently we restrict our calculations to U-233 which is of interest to us for the RRC-IAEA Co-ordinated Research Programme (CRP)[2] on integral validation of actinide nuclear data. The following data sources are considered; (1) ENDF/B-IV, (2) JENDL-2, (3) CADARACHE,

(4) ENDL 84/V and (5) MEADOWS[3]. Of these the Cadarache nuclear data are already in multigroup form in the SETR structure. The basic data were multigrouped by REX1 and then collapsed to one group using the experimental RAPSODIE spectra[1]. Table 1 gives a comparison of the integral cross sections calculated using data from KEDAK[1], Cadarache and ENDF/B-IV with the experimental values for fission and capture processes. Calculations are under way for the other data libraries. The assessment of errors and interpretations are in progress although the C/E values given in Table 1 would give an indication of the extent of agreement.

1. A. Cricchio et al., The TACO experiment for the Determination of Integral Neutron Cross Sections in a Fast Reactor, Nuclear Data for Science and Technology, D.Reidel Publishing Company (1983), page 175.

2. S.Ganesan, Status Report on RRC-IAEA Co-ordinated Research Programme on Validation and Benchmark Testing of Actinides Nuclear Data presented at the Second Research Co-ordination Meeting held in Varna, Bulgaria during 14-16, October 1985.

3. J.W.Meadows, The Fission Cross Sections of Some Thorium, Uranium, Neptunium and Plutonium Isotopes Relative to U-235, ANL/NDM-83 (1983).

Table 1 Comparison of Integral Cross Sections for U-233

Process	Experimental (RAPSODIE) (E)	Calculated Values (C) KEDAK	CADARACHE	ENDF/B-IV
FISSION	2.31 ± 6%	2.12 (0.918)	2.28 (0.987)	2.11 (0.913)
CAPTURE	0.155 ± 0.2%	0.155 (1.0)	0.182 (1.174)	0.156 (1.006)

The values in paranthesis give C/E values.

GENERATION OF DISPLACEMENT DAMAGE CROSS SECTIONS FOR VARIOUS LOCATIONS IN A FAST TEST REACTOR

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A programme of obtaining the map of dpa/sec for any given material as a function of position in FBTR has been taken up. The code RECOIL[1] which generates displacement damage cross sections in 105 fine groups was successfully commissioned and tested[2]. The usefulness of RECOIL and its data base is not restricted to just FBTR needs. By using the calculational method and data base of RECOIL damage effects of structural materials like gas production, nuclear displacements etc. can be simulated, studied, evaluated and corrected and correlated in different neutron environments. The mathematical formulae are available in Ref.1.

Table 1 gives the displacement damage cross section calculated in 25 groups for SS-316 stainless steel by collapsing the fine group values produced by RECOIL code. The values of σ_a in Table 1 were utilised to obtain the effective dpa/sec in the central core location of FBTR-65. The fluxes in 25 groups obtained by two dimensional diffusion theory calculations were supplied to us by Reddy[4]. Table 2 compares this displacement rate with those reported for a typical fast reactor like EBR-II in the literature[2,3]. Case 3 differs from Case 1 in the methods of calculation of dpa.

1. T.A.Gabriel et al., ORNL/TM-5160 (1976)
2. M.M.Ramanadhan and S.Ganesan, 'Adaptation of Code RECOIL System', RRC-64, p.13 (1984).
3. G.L.Kulcinski, D.G.Doran and M.A.Adou UWFDM, University of Wisconsin (1974).
4. C.P.Reddy, Private Communication (1984).

Table 1

Calculated Displacement Damage Cross Sections for Stainless Steel Type 316

Grp No.	Upper Energy (in ev)	Displacement cross section	Grp No.	Upper Energy (in ev)	Displacement cross section (in barns)
1	.14500E 08*	.19887E 04	14	.91300E 04	.36590E 02
2	.36800E 07	.14393E 04	15	.55400E 04	.19128E 02
3	.22300E 07	.10037E 04	16	.33600E 04	.85325E 01
4	.13500E 07	.61490E 03	17	.20400E 04	.59907E 01
5	.82213E 06	.46065E 03	18	.12400E 04	.59030E 01
6	.49865E 06	.37088E 03	19	.74968E 03	.15222E 01
7	.30245E 06	.25254E 03	20	.45471E 03	.27158E 00
8	.18344E 06	.19500E 03	21	.27579E 03	.37140E 00
9	.11126E 06	.15791E 03	22	.10146E 03	.79885E 00
10	.67480E 05	.89191E 02	23	.22640E 02	.67811E 00
11	.40930E 05	.14236E 03	24	.30500E 01	.17891E 01
12	.24830E 05	.34056E 01	25	.41400E 00	.10584E 02
13	.15600E 05	.32288E 02			

* Read as $.145 \times 10^{8}$

Table 2

Calculated DPA rate/sec $\times 10^{7}$ for Type 316 Stainless Steel

Sl.No.	Facility	dpa/s $\times 10^{7}$
1	EBR II Row 2 (62.5 MWth) (Ref.3)	14.00
2	FBTR-65 (50.0 MWth) core centre (axially averaged flux)	15.26
3	EBR-II Row 2 (62.5 MWth) (Ref.1)	16.00

STATUS OF ACTIVITIES UNDER IAEA's CO-ORDINATED RESEARCH PROGRAMME ON VALIDATION AND BENCHMARK TESTING OF ACTINIDE NUCLEAR DATA

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The programme of work agreed by RRC within the scope of IAEA-CRP, to be carried out at RRC, as stated in the Research Contract (No.3690/R1/RB), is as follows:

(a) Continuation of producing best evaluated data files for Th-232 and U-233 in ENDF/B format

(b) Detailed validation of these data files, including intercomparisons with similar data files from different origins and analysis of the discrepancies encountered.

(c) Compilation of related data from integral experiments and review of uncertainties for benchmark testing purposes'

The period of the research contract was commenced on 1st March 1985 and will cover period upto 28th February 1986 presently.

The summary presented at the recent CRP[1] meeting in Bulgaria in October 1985 is given below:

- The capture cross sections evaluated in Trombay by Jain and Mehta (1983) along with Meadows (1983) fission cross sections for Th-232 are consistent with the integral value of alpha deduced from measurements of spectral indices measured at core center of THOR assembly. JENDL-2 file for Th-232 is, in itself, also consistent in the same way.
- Discrepancies in inelastic cross sections for Th-232 need to be resolved.
- The discrepancies in Keff by themselves for THOR assembly do not help to validate any data set in a conclusive manner.

- The ratio of (n,2n) to fission cross sections seem to be O.K. only in JENDL-2 file as per preliminary studies for Th-232.
- The CFRMF assembly for which the central core flux with associated error matrix is given will be taken up for analysis
- The suitability of using JEZEBEL-23, a bare sphere of U-233 (98.13%) metal will be assessed by analysis.
- The analysis of KUR-RPI experiment for spectra in THORIA and the LLL pulsed thorium sphere may be taken up only after the relevant neutronic codes are developed to the required sophistication, at Kalpakkam.
- Concerning the Th-232 and U-233 data files, the energy range below 50 KeV will be borrowed from JENDL-2 file. The inelastic cross sections will be critically examined. The purpose is to identify the best data and create new files for Th-232 and U-233 in ENDF/B format only if the available existing recent evaluated files like JENDL-2 show sufficient deficiency to warrant such a creation of new data file. However, the partial data evaluated in India by itself has also been coded already in ENDF/B format for possible dissemination by IAEA, after scrutiny.
- Further critical evaluation and compilation of benchmarks for data testing purposes for Th-232 and U-233 will be continued.

1. S.Ganesan, Status Report on RRC-IAEA Co-ordinated Research Programme on Validation and Benchmark Testing of Actinide Nuclear Data' presented at the second Research Coordination Meeting held in Varna, Bulgaria during 14-16, October 1985 .

INTEGRAL VALIDATION OF NEUTRON INDUCED CAPTURE AND FISSION CROSS SECTIONS FOR TH-232 IN FISSION SOURCE ENERGY RANGE

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As part of the IAEA-NDS sponsored Co-ordinated Research Programme on the intercomparison of evaluations of Actinide Neutron Nuclear Data, a programme[1] is in progress at RRC for Th-232 and U-233. We obtained[2] interesting results highlighting the extent to which evaluations in JENDL-2, INDIAN, ENDF/B-V, ENDL-84, FRENCH SET (1969), INDL/A-83 (RUMANIAN), ENDF/B-IV and JENDL-1 files are consistent with the measured value of σ_c/σ_f for Th-232 at the center of THOR critical assembly which emphasizes transport of neutrons in the fission source energy range. THOR assembly, in equivalent spherical model has a core of 5.310 cm radius centered in a reflector of 29.88 cm outer radius. Experimental values of σ_f(U-238)/ σ_f(U-235), σ_f(TH-232/ σ_f(U-238), σ_γ(U-238)/ σ_f(U-235) and σ_γ(TH-232)/ σ_γ(U-238) have been published alongwith the associated uncertainties. We deduced[1] the value of alpha from these ratios as 1.9645 ± 0.146 for Th-232 at core center of THOR assembly.

Shown in Figure 1 are our results for "C/E" (calculated to experimental ratio) of the value of alpha at the center of THOR assembly. The error in "C" is taken as 4% as the uncertainties in cross sections of Pu-239 and those of Th-232 influence "C" by as much as 4%, on the average, through corresponding uncertainty in calculated neutron energy spectrum. It is clearly seen that the evaluations in ENDF/B-IV, INDL/A-83 (RUMANIAN), JENDL-1 and the FRENCH set overpredict alpha for Th-232 by 25 to 46%. The relative error (in C/E) of 8.43% (4% in "C" and 7.43% in "E") could be as large as 11.43% if we treat the error in "C" as systematic. Then INDIAN and JENDL-2 files are consistent and ENDL-84 and ENDF/B-V would appear, within 5%, consistent with integral measurement of alpha at core center of THOR assembly.

The uncertainty in σ_f for Th-232 in recent files introduce an uncertainty of about 5% in the interpretations of "C/E" for alpha. The use of recent data of Meadows[5] for σ_f of Th-232 brings the C/E value to 1.08 for INDIAN file. We therefore conclude[3] that the INDIAN evaluation by Jain and Mehta[4] for σ_c with σ_f from Meadows[5] and the JENDL-2 for σ_c and σ_f are consistent with the integral value of σ_c/σ_f deduced by us from the measured values of spectral indices at the core center of THOR assembly. Since JENDL-2 uses Lindner's data[6] for σ_c our analysis validates integrally Lindner's data. New and accurate measurements of σ_c are needed to reduce further the existing discrepancies seen in Figure 2. **Further details are given in Ref.7.**

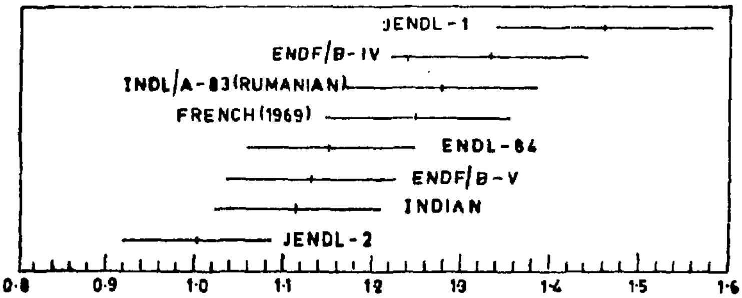

FIGURE 1 Calculated to Experimental ratio for Alpha for Th-232 at core centre of THOR Assembly

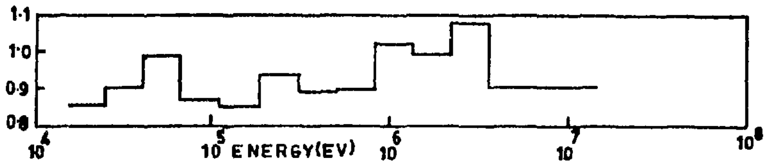

FIGURE 2 JENDL-2 file is compared with INDIAN file in fission source energy range for c of Th-232.

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4. H.M.Jain and M.K.Mehta, in Nuclear Data for Science and Technology, Edited by K.H. Bockhoff (D. Reidel Publishing Company, 1983) pp.657-660.

5. J.W.Meadows, ANL-NDM-83 (1983).

6. M.Lindner et al., RRC-43 (1981).

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INTERPRETATION OF DOPPLER EXPERIMENTS USING THE NEWLY GENERATED KALPAKKAM MULTIGROUP CROSS SECTION SET

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The problem of interpretation of Doppler effect measurements is a challenge from research point of view and provides, a method for integral test of resonance cross section data and Doppler broadening techniques in resolved and unresolved resonance regions.

(1) SEFOR : The Doppler effect calculations for SEFOR[1,2] reactor which was mainly constructed for studying the Doppler effect in fast reactors, were performed using the newly generated Kalpakkam multigroup cross section set[3], and the 1969 Cadarache multigroup set available with us.

The diffusion calculations were performed in one dimension using the code 'MUDE'[4], while the code 'EFFCROSS'[5] was used for generating composition and temperature dependent group cross sections.

The full core configuration, the atom densities of materials and the material volume fractions in 2 dimensions are given in a report by Harris and Nelson[2], and in a BNL report[6].

The temperature rise ΔT was assumed to be the same throughout the core. Thus the temperature gradient and its effect were not considered explicitly. It was however clear that the value of the Doppler coefficient obtained by running a two K calculation corresponding to a uniform rise in temperature throughout the core will slightly underestimate the magnitude of the static Doppler Coefficient. The temperature derivative of the effective multiplication factor $\Delta K / \Delta T$ varies approximately as the inverse of temperature with the result that T is nearly a constant.

The calculated (C) to experimental (E) ratio C/E for the Doppler coefficient is found to be 0.85 for Cadarache set available with us and 0.79 for Kalpakkam set.

(2) <u>The U-233 Doppler Experiment</u> : The one Doppler experiment for which data exists to permit a re-analysis is the central doppler reactivity worth measurements in ZPR-6-7[6] assembly. The Doppler sample was of 1 inch diameter and 12 inch long cylinder at the centre of the ZPR-6-7[6] assembly. The sample composition was U-233:785.85 gm, U-234:9.67 gm, U-238:3.44 gm and O:138.15 gm.

No analyses of this experiment has ever been reported for U-233 Doppler worth using ENDF/B-IV or more modern cross section files.

The calculations using French (1969) and ENDF/B-IV based Kalpakkam set (1985) gives almost zero Doppler coefficient showing the complete absence of temperature dependence of cross section of U-233 isotope arising from absence of unresolved resonance data in these sets. The calculations using JENDL-2 and Kalpakkam cross section sets are under progress for this U-233 Doppler experiment.

1. R.A.Meyer et al, 'Design and Analysis of SEFOR Core I' GEAP 13398 (1970).

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3. M.M.Ramanadhan and V.Gopalakrishnan, 'Kalpakkam Multigroup Cross Section set - Status and Performance (RRC Report to be published).

4. T.M.John, MUDE:A 1 D Diffusion Theory Code : Note No.FRG/01100/RP-60 (1975).

5. Damiens, J and Ravier, J. 'Programme SETR 512 Exploitation des Constantes Multigroupe, SETR Cadarache, PNR/67/746.

6. ENDF-202:Cross Section Evaluation Working Group Benchmark Specifications BNL report USA.

EFFECT OF INTERPOLATION ERROR IN PRE-PROCESSING CODES ON CALCULATIONS OF SELF SHIELDING FACTORS AND THEIR TEMPERATURE DERIVATIVES

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The Doppler coefficient in fast reactors should be predicted to ± 10% accuracy and the portion of the error arising from interpolation error used in pre-processing codes should be less than 1%. Because of limitations in computer time, generally these codes are run only with relaxed error criteria. In addition, problems occur due to improper representation of evaluated data in resolved resonance region. We investigated the effect of interpolation error in pre-processing codes LINEAR, RECENT and SIGMA1 on calculations of self shielding factors and their temperature derivatives. We consider 2.0347 to 3.3546 keV energy region for U-238 capture, which is the NEACRP benchmark exercise on unresolved parameters. The calculated values of temperature derivatives of self shielding factors is significantly affected by interpolation error as shown in Table 1. The source of problems in both evaluated data and codes were identified by Cullen[1] and eliminated[2] in 1985 version of these codes. Our paper[1] helps to (1) inform code users to only use 1985 versions of LINEAR, RECENT and SIGMA1 and (2) inform designers of other code systems where they may have problems and what to do to eliminate their problems[(1)].

1. S. Ganesan, V. Gopalakrishnan, M.M. Ramanadhan and D.E. Cullen, "Effect of Interpolation Error in Pre-processing On Self Shielding Factors and Their Temperature Derivatives" Paper presented in the International Conference on Nuclear

Data for Basic and Applied Science during May 13-17, 1985 Santa Fe, New Mexico, U.S.A. (To appear in Radiation Effects).

2. D.E. Cullen, Private Communication (1985).

Table 1

Results with 1984 and previous versions of codes LINEAR/RECENT/SIGMA1

Dilution (barns)	$\tilde{\sigma}_{2a}/\tilde{\sigma}_{1a}$	$\tilde{\sigma}_{2b}/\tilde{\sigma}_{1b}$	$\Delta\tilde{\sigma}_2/\Delta\tilde{\sigma}_1$
0.0	1.027	1.063	1.159
1.0	1.021	1.061	1.180
10.0	1.020	1.052	1.167
100.0	1.012	1.037	1.218
1000.0	1.008	1.031	1.912
10000.0	1.007	1.031	11.333

$\tilde{\sigma}_{xy}$ = Self shielded cross section for case xy

$\Delta\tilde{\sigma}_x$ = $\tilde{\sigma}_{xb} - \tilde{\sigma}_{xa}$ a : 300K; b: 2100K

Case 1a = LINEAR (0.1%) - RECENT (0.1%) - SIGMA1 (0.0%; 300K); Error : 0.141%.

Case 1b = LINEAR (0.1%) - RECENT (0.1%) - SIGMA1 (0.0%, 2100K); Error : 0.141%.

Case 2a = LINEAR (0.1%) - RECENT (0.1%) - LINEAR (1.0%) - SIGMA1 (0.2%, 300K); Error : 1.03%.

Case 2b = LINEAR (0.1%) - RECENT (0.1%) - LINEAR (1.0%) - SIGMA1 (0.2%, 300K) - SIGMA1 (0.0%, 2100K); Error : 1.03%

ENDF/B-IV; NEACRP Benchmark energy region : 2.0347 to 3.3546 keV; Capture; U-238.

DEVELOPMENT AND ADAPTATION OF COMPUTER CODES RELATED TO EVALUATION AND PROCESSING OF NUCLEAR DATA

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The details of work done in the field of development and adaptation of computer codes related to nuclear data research at Kalpakkam are given below:

LINEAR-85 : This code replaces the 1984 version of LINEAR. This code processes the energy cross section table having any interpolation scheme and provides a table with linear-linear interpolation. In this new version all energies are treated in DOUBLE PRECISION. Report UCRL-50400, Vol.17, Part A (1979).

<u>RECENT-85</u> : This code replaces the 1984 version of RECENT. This code reconstructs the cross sections given in the form of resonance parameters into linearly interpolable point data table. In this new version all energies are treated in DOUBLE PRECISION. Report UCRL-50400, Vol.17, Part C (1979).

<u>FIXUP-85</u> : This code that can do useful consistency checks, deletion of one or more sections of the data, re-definition of a section as summation over other specified sections etc. has been adapted during thisperiod.

<u>SIGMA1-85</u> : This version replaces the 1984 version of

SIGMA1. This code provides cross sections at a specified higher temperature by broadening the point cross sections at a lower temperature using kernel broadening method. Apart from the treatment of all energies in DOUBLE PRECISION the major revision in this code is a better linearization procedure. Report UCRL-50400 Vol.17 Part B (1979).

REX2-85 : This code written at Kalpakkam calculates the shielding factors for various dilutions. The improvement in this version is more appropriate integration procedures for greater accuracy and better programming for significant reduction in computing time.

EVGRP : This code is to process the photonic cross section data. Originally written for CDC, has been modified for HONEYWELL computer. Originally written to take input from RESEND output has been modified for taking data from RECENT output. These two aspects requiring many changes to be made, the adaptation is in the last stages of completion.

SCAT2 : is a code written in Fortran langauge for spherical optical model calculations. This was recently commissioned in our inhouse computer.

Ref.Olivier Bersillon, SCAT2: A Programme of the optical Spherical model CEA-N -222 (1984)

RECOIL : This is a code for studies relating to radiation damage effects in materials, displacement rates, gas production rates in fission and fusion reactors. The code uses a heavy charged particle RECOIL data base derived from ENDF/B data.

Ref. J.A.Gabriel, J.D.Amburge N.H.Greene, Radiation Damage Calculations, Primary Recoil Spectra, Displacement Rates and Gas Production Rates, Report ORNL-TM-5160 (1976).

AN OVERVIEW OF PROBLEMS AND EXPERIENCES IN NUCLEAR DATA PROCESSING IN DEVELOPING COUNTRIES

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An overview has been published[1] on the details concerning our experiences during last one decade in nuclear data processing for fast reactor applications.

The preparation of multigroup cross section set from a basic data library such as ENDF/B is tending to be an art and involves selection of various algorithms corresponding to the nature of applications for which the multigroup cross section set is intended. Depending on the nature of the computing facility available in the laboratory, the processing procedures may vary widely to suit the memory, cpu time and accuracy restrictions. As is the case with many pre-processing and processing codes, the user input is indeed very simple but the successful processing of the data for certain materials is very difficult. This state of affairs is mainly because of the vast amount of computing time, input-output operations, large core memory needed for storage and handling of the data and the resulting complexities. For these straight-forward reasons, a novice is cautioned against use of such processing codes without a thorough knowledge of their inherent complexities.

We also recommend[1] extension of code Verification Project[2,3] to cover all modules of processing to help resolve existing discrepancies in the generation and use of multigroup constant sets by covering also the physical approximations in processing methods.

1. S. Ganesan, V. Gopalakrishnan, M.M. Ramanadhan, "Problems and Experiences in Nuclear Data Processing in Developing Countries", Progress in Nucl. Energy, Vol. 14, No. 3 (1984).

2. D.E. Cullen et al., INDC(NDS) - 134/G (1982), Vienna.

3. D.E.Cullen, Report on the IAEA Cross Section Processing Code Verification Project, INDC(NDS)-170/NI (1985).

REMARKS ON EXISTING ENDF/B FORMATS AND PROCEDURES

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On a suggestion from IAEA nuclear data section, a contribution was made to IAEA's Consultants Meeting on the 'Format' for exchange of evaluated neutron nuclear data held during 2-4 April 1984. The following remarks were made, based on our limited experience with processing of ENDF/B data and the writing of our processing code system RAMBHA[1].

1. We are convinced[2] that the representation of cross sections in the unresolved resonance region is presently not scientifically founded and the present ENDF/B convention of deriving the first moment and the distribution functions for resonance parameters have[3-5] large uncertainties. This makes the accurate prediction of self shielding factors and their temperature derivatives quite impossible at present[3,5]. The reader is referred to the overview[2] for the actual impact of these uncertainties on some fast reactor parameters.

2. The use of J* method of Hwang[6] and the use of Cullen's codes[7] in the resolved resonance region gives[8] different self shielding factors at low dilutions for ^{239}Pu and ^{235}U. We recommend abandoning the use of J* method in processing resolved resonance region assuming that the interpolation error in Cullen's codes LRSG[7] can be minimized within reasonable computing time. However the present method of using resonance parameters and floor

corrections will have to be retained because of memory constraints.

3. We should wait till the first few rounds of testing in IAEA Code Verification Project[9] on processing codes are completed, before freezing ENDF/B formats and procedures. It is very likely that more suggestions, some even concrete, come out regarding formats and procedures as a result of present and future rounds of the IAEA Code Verification Project covering all areas of processing modules.

4. A suggestion for a small change in the ENDF/B/IV format for the beginning of a section was also made[10].

1. S. Ganesan et al. 'Development of a New Fast Reactor Processing Code RAMBHA at RRC' in Proceedings of the Workshop on Nuclear Data Evaluation, Processing and Testing INDC(IND)-30 (1981), IAEA, Vienna.

2. S. Ganesan, 'On the Need for Changing the ENDF/B Convention for the Representation of Cross Sections in the Unresolved Resonance region of Fertile and Fissile Nuclei', Annals of Nuclear Energy 9, 481-487 (1982).

3. S. Ganesan, Atomkernenergie 27, 14-16 (1977).

4. S. Ganesan, Nucl.Phy. and Solid State Phys. 23B, 142-145(1980).

5. S. Ganesan, Nucl. Sci. Eng., 74, 49 (1980).

6. R.N. Hwang, Nucl. Sci. Eng., 52, 157 (1973).

7. D.E. Cullen, UCRL 50400 Series 20, 157 (1973).

8. S. Ganesan, M.M. Ramanadhan and V. Gopalakrishnan, 'Comparison of Calculated Self Shielding Factors with Measured Values for ^{239}Pu, ^{235}U, Fe and Na', pp. 32-35 in Nuclear Data for Science and Technology, D. Reidel Publishing Company (1983).

9. D.E. Cullen, W.L. Zijp and R.E. MacFarlane, 'Verification of Nuclear Cross Section Processing Codes', INDC(NDS)-134/G (May 1982). See also INDC(NDS)-170/NI. (1985)

10. S. Ganesan, Note distributed at IAEA Consultants Meeting on 'Format for Exchange of Evaluated Neutron Nuclear Data' 2-4 April, 1984, IAEA, Vienna.

Data and Data needs for Status of XRF and PIXE based element analysis.

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For quantitative elemental analysis of samples, by XRF and PIXE methods, knowledge of a number of fundamental quantities such as the ionisation cross-sections, fluorescence yields, relative line intensity ratios, x-ray attenuation coefficients, detector response functions, matrix correction factors, etc., are required. There have been many measurements of these quantities and compilations of these are available in literature. The accuracies in these quantities affect directly the accuracy in the quantitative analysis of samples. Apart from the practical importance of these data from the point of elemental analysis, some of them are quite useful in order to test the predictions of various theoretical models. A review has been carried out on the data related to XRF and PIXE analysis in order to identify areas where more accurate and detailed information are required to be obtained.

In both XRF and PIXE methods, the x-rays of elements present in the sample are excited which are then measured by the energy dispersive method using a high resolution Si(Li) detector. The energy of each x-ray peak is associated with the presence of a particular element and the intensity of the peak is related to the elemental concentration.For quantitative analysis of the samples, the observed intensity of the various peaks is transformed to the concentration of the corresponding elements and this transformation involves many physical quantities as mentioned above. In many applications,methods involving the use of standards have been developed for accurate quantitative analysis to minimise errors in the results due to uncertainties in the fundamental x-ray related data or in the quantities dependent on a particular experimental arrangement. However, since usally the calibration of the experimental set up is done with only a few element standards, errors in the knowledge of the various physical quantities still affects the accuracy of analysis.

In many trace element analysis using the PIXE method, one is quite often interested not only in the accuracy of the analysis but also in its sensitivity as determined by the minimum detectable limit (MDL) of an element. The MDL depends on the relative yields of the fluorescent x-rays and the background x-rays in the region of interest. The data related to background continuum and its angular distribution and dependence on the particle beam energy are quite important.

Recently there have been evidences of the influence of the chemical states of the elements on the x-ray cross-sections and intensity ratios of the K and L lines. The limit on the accuracy in the trace element analysis involving sample preparation will therefore be ultimately governed by these chemical effects. More systematic data on these are required in order to understand the magnitude and importance of these effects.

Since the elemental analysis is carried out using both the thin and thick samples, data related to the matrix correction factors such as absorption and enhancement of x-rays and slowing down of the charged particles in the sample matrix are quite important. While analysing thick samples, these matrix corrections are usually done by computer calculations and accuracies achieved in such calculations have been reported in the literature to be between 10 to 15 percent for some typical samples. The accuracy in the analysis of thick samples depends on the extent of corrections applied to the data and systematic investigations in these lines are needed to be carried out.

The report therefore aims at reviewing the status of the data and enumerating the areas where more experimental data are required with regard to improving the accuracy in PIXE and XRF analysis and also understanding the various theoretical models for explaining these data.

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