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Kalpakkam Multigroup Cross Section Set for Fast Reactor Applications - Status and Performance

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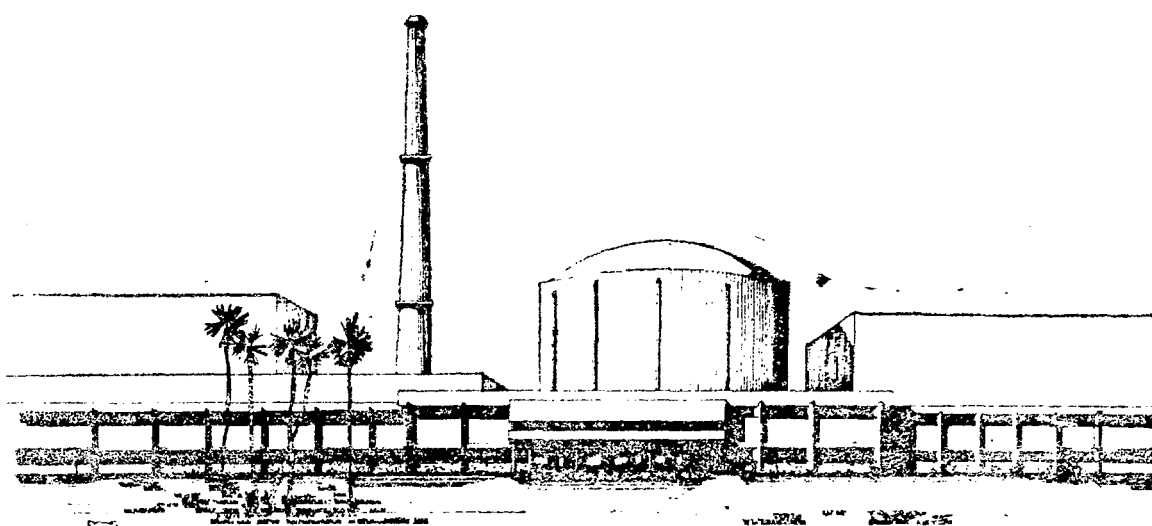

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KALPAKKAM MULTIGROUP CROSS SECTION SET FOR FAST REACTOR APPLICATIONS - STATUS AND PERFORMANCE

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ABSTRACT

This report documents the status of the presently created set of multigroup constants at Kalpakkam. The list of nuclides processed and the details of multigroup structure are given. Also included are the particulars of dilutions and temperatures for each nuclide in the multigroup cross section set for which self shielding factors have been calculated.

Using this new multigroup cross section set, measured integral quantities such as K-eff, central reaction rate ratios, central reactivity worths etc. were calculated for a few fast critical benchmark assemblies and the calculated values of neutronic parameters obtained were compared with those obtained using the available Cadarache cross section library and that published in literature for ENDF/B-IV based set and Japanese evaluated nuclear data library (JENDL). The details of analyses are documented along with the conclusions.

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11. (a) Central reactivity worths of ^{235}U normalised to those of ^{239}Pu.

    (b) Central reactivity worths of ^{238}U normalised to those of ^{239}Pu.

    (c) Central reactivity worths of Chromium normalised to those of Pu-239.

    (d) Central reactivity worths of Iron normalised to those of ^{239}Pu.

    (e) Central reactivity worths of Nickel normalised to those of ^{239}Pu.

12. Calculated net K_{eff} using ENDF/B-IV non-adjusted multigroup set and the 1969 adjusted French set for ZPR-3-56 assemblies.

13. A Sample out of a 25 group cross section set generated at Kalpakkam.

1. Introduction

Using the nuclear data processing codes developed at Kalpakkam[1-3], multigroup cross sections were generated from the basic general purpose nuclear data library ENDF/B-IV for various elements (Table 1). For the element Ga, the data set used was ENDL-84, since this element was not available in ENDF/B-IV library. Similarly wherever the element was not available in ENDF/B-IV, other libraries were chosen. The multigroup structure followed given in (Table 2) is the same as the Cadarache 25 group structure to enable compatibility with the reactor mixture cross section preparation code, and the various neutronic codes at present available with us for fast reactor design calculations. Table 3 gives the dilutions and temperatures for which self shielding factors are calculated in our multigroup set for the resolved and unresolved resonance regions.

The Kalpakkam multigroup cross section set is dynamic in the sense that it has the flexibility for updating cross sections for any given isotope and also for addition of a full cross section set for any new isotope. For this purpose a code LCAT[4] was indigenously developed and is in use.

To evaluate the usefulness of the newly created multigroup cross section set, analysis of various fast critical assemblies from other countries was undertaken, in the absence of a programme of fast critical experiments in India. The integral parameters investigated in this comparative study were K-eff, central reaction rate ratio, central reactivity worths etc. A detailed analysis of a fast critical assembly ZPR-9-31[5] simulating the advanced plutonium carbide fuelled large fast reactor using both the 1969 adjusted French multigroup set[6] and the Kalpakkam non-adjusted multigroup set was reported by us recently. This

study showed that the Kalpakkam non-adjusted ENDF/B-IV based set performed reasonably well in predicting various neutronic integral parameters, as in the case of other non-adjusted sets derived from ENDF/B-IV data base.

2. Description of Procedure and Preparation of Multigroup Cross Section Set

The multigroup cross section preparation from a basic data library such as ENDF/B by itself is a complex procedure since it involves the choosing of various algorithms corresponding to the nature of applications for which the multigroup cross section is meant. The complex situation in this processing can be gauged from Figure.I where the flow chart details the vital note played by processing methods in the nuclear data evaluation, processing and testing.

The availability of our processing code has enabled us to perform some detailed and specific sensitivity studies starting from the differential cross sections. This also enables us to have a better understanding of the effect of certain changes in fast neutron cross sections on neutronic parameters of fast reactor systmes. The RAMBHA code system developed by our group was used in 1983 to generate the Kalpakkam multigroup cross section set. Subsequently the codes REX1 and REX2 which represent improved form of RAMBHA, were used to process the preprocessed ENDF/B libraries. The preprocessing of ENDF/B library itself was performed by using the codes LINEAR[7], RECENT[8], FIXUP[9] AND SIGMA1[10].

The computer code EFFCROSS[11] was used to generate the region and composition dependent cross sections for core and reflector regions. This code takes the multigroup cross section library as input and generates, composition and temperature dependent mixture cross sections for each region of the reactor.

The one-dimensional diffusion-cum-perturbation theory code MUDE[12] was used to calculate the K-eff values, fluxes, adjoint, β effs, central material worths for the various benchmark assemblies.

3. Selection and Description of Benchmark Assemblies

Testing of Kalpakkam multigroup cross section set (containing 25 group infinite dilution cross sections with self shielding factors and fine group elastic removal data) was done for the following assemblies[14]:

(a) ZPR-3-48

(b) ZPR-3-56B

(c) ZPR-6-7

We had earlier analysed one benchmark assembly ZPR-9-31, and in the present analysis, the assemblies chosen were on the basis of system simplicity, extent of experiments carried out, an adjudged precision of experimental results, and covering a wide spectrum range, typical of a 500 MWe fast reactor. A listing of key characteristics of all critical assemblies considered in this study is presented in Table 4. The compositions and dimensions for the one dimensional spherical models of these assemblies appear separately in Table 5.

The physical characteristics of these assemblies vary considerably from a reactor core volume of 410 litres capacity in the case of ZPR-3-48 to 3100 litres capacity as in the case of ZPR-6-7, from a predominantly nickel reflected assembly to a mixed (Pu-U) fuel plus carbon, typical of a mixed carbide fuelled LMFBR design. Of course, it has to be considered that the bigger the reactor size, the spectrum becomes softer and the uncertainties in the prediction of integral parameters may not remain the same.

4. Analysis and Results of Benchmark Calculations

4.1 Eigenvalue (K-eff)

The value of calculated K-effs for the series of benchmark fast critical assemblies are summarised in Table 6 & 7. The basic reactor model with which the critical assemblies were analysed was a one dimensional diffusion theory claculation in spherical geometry. Therefore, one to two dimensional, diffusion to transport theory and heterogeneity correction factors were applied to the calculated K_{effs}. It is seen from the results given in Table 7 that the non-adjusted Kalpakkam set gives K-eff about 0.3% lower than that of 1969 adjusted French set in most cases. It may be just a fortuitous coincidence that K-eff for the Kalpakkam set is closer to the experimental value of 1.000 $\pm$ .0015 as compared to other multigroup cross section sets derived from ENDF/B-IV. In the recent past Japanese[15] have reported for ZPR-6-7 assembly that their ENDF/B-IV based multigroup set gave 1% lower K_{eff} compared to the value reported for the same assembly by a US team. This can be accounted as due to the processing methods which can introduce an uncertainty of as much as 1% in K-eff. This uncertainty includes effects due to different multigroup structures (fine or broad), methods of collapsing etc. The subject of validation and verification of nuclear cross section processing code and the data file through analyses of fast critical assemblies is a complex task and ivolves taking into account uncertainties in integral parameters and cross sections including correlations among them. The consideration of K_{eff} was meant to illustrate that different countries and laboratories arrive at significantly different values for K_{eff} though the starting is from the same ENDF/B-IV data base and same descriptions of benchmark specifications are used. This is an interesting result which vouchsafes and stresses the importance of the recently proposed IAEA Code. Verification Project[16] which aims at understanding and removing descrepancies and errors in processing algorithms.

4.2 Central Reaction Rates

Calculation of reaction rate ratio were performed using central fluxes of one dimensional homogeneous spherical model. These one dimensional homogeneous calculated values were transformed to calculated two dimensional hetrogeneous values by the application of model correction factors. This factor is defined as,

$$\text{Model Correction Factor} = \frac{\text{Calculated Two Dimensional hetrogeneous}}{\text{Calculated One Dimensional homogeneous}}$$

For the critical assembly, the central fission rates in ^{238}U, ^{239}Pu and ^{240}Pu (Tables 8a, 8b, 8c) and the central neutron capture rate in ^{238}U, relative to the central fission rates in ^{235}U and ^{239}Pu (Tables 9a, 9b) were determined by both calculations and experiment. It may be possible that the discrepancy between calculation and experiment could be reduced by a more consistent treatment of heterogeneity effects. The ratio of the capture rate in ^{238}U to the fission rate in ^{239}Pu is an important component of the breeding ratio.

4.3 Central Reactivity Worths

In recent years the discrepancy between calculated and experimental small-sample central ractivity worths has received much attention.

Tables 10a to 10g give the results with regards to prediction of absolute central reactivity worths in the various fast critical assemblies for ^{235}U, ^{238}U, ^{239}Pu, Fe, Cr, Ni and Na. The model correction factors taken from Ref.5 was used in the comparison of reacticity worths obtained with the homogeneous one dimensional spherical model with the experimental values in units of $(\Delta K/K)/10^{24}$ atoms. Tuttle's[17] delayed neutron data was used in the calculation for the in hour conversion factors for both Kalpakkam and French multigroup sets.

In general the adjusted multigroup French sets are closer to experiments in most cases, though even in these there are large overpredictions. It cannot be said that nuclear data alone is responsible for these descrepancies since there could be several other reasons such as scattering in light elements, sample size correction, ultra cell adjoint heterogeneity etc.

4.4 Normalised Worths

The central reacticity worths normalised to those of ^{239}Pu are given in Tables 11a to 11e. The normalised calculated worths are obtained for ^{235}U, ^{238}U, Cr, Fe and Ni.

The Kalpakkam multigroup set predictions for ^{235}U are consistent with those of US values and are superior to the French multigroup set. In the case of ^{238}U except for ZPR-3-56 assembly where there is an overestimation to about 7%, for the other assemblies, the values obtained by Kalpakkam multigroup set are fairly consistent.

As for the structural material Fe, Cr, Ni it is rather difficult to well predict the normalised worths. Though there are large overestimations, the values obtained from the Kalpakkam set are closer to those reported for US set.

5. Scope for Future Work

(1) The ENDF/B-IV base is what we have got from U.S.A. This reflects 1974 status and if Version V or VI becomes available, every effort must be made to process such later versions. Towards this objective processing of recent basic data libraries such as JENDL-2 (1979) INDL/A-85, ENDL/84, parts of ENDF/B-V procured from IAEA and other laboratories is already in progress.

(2) Creation of a hybrid multigroup set by updating the French set with recent non-adjusted multigroup sets for specific isotopes may alleviate the problems in this generation of best data set to a certain extent. For example as seen in Table 12 a hybrid set consisting of adjusted French set in which the cross sections of Fe, Cr and Ni are replaced by the non-adjusted ENDF/B-IV based Kalpakkam set has been found to predict K-eff of ZPR-3-56B better than ENDF/B-IV based set or the French set. Such a hybrid set should be obtained by modifying the French set only to a certain limited extent without disturbing the advantages characteristic of the 'adjustment' effects of the French set.

6. Conclusions

Based on our present analysis of the few benchmark fast critical assemblies the following conclusions and recommendations are drawn:

(a) The indigenously developed processing code RAMBHA, (REX1 and REX2) and the generations of non-adjusted ENDF/B-IV based Kalpakkam multigroup cross section set and its validation have been quite successful. The availability of our own processing codes has enabled us to generate the Kalpakkam multigroup set in a detailed and systematic manner. The K_{eff} predicted for various assemblies by our set has been as close to the experimental values as those predicted by other non-adjusted ENDF/B based sets.

(b) The overprediction of central worths by the Kalpakkam multigroup set and the US sets are higher than those predicted by the adjusted French set. For reducing these overpredictions of central worths of various elements, emphasis has to be laid on improving the status of structural materials like Fe, Cr and Ni. Studies on this have been taken up and are progressing.

(c) For isotope like ^{241}Pu the non-adjusted ENDF/B-IV based Kalpakkam multigroup set has proved superior to the 1969 adjusted French set due to major improvement in relatively recent (1974) evaluations and measurements of cross sections.

(d) In most of the assemblies taken up for analysis it is observed that the predictional performance for reaction rates of the non-adjusted Kalpakkam multigroup set lies in between the adjusted French multigroup set and the non adjusted American and Japanese multigroup sets.

(e) There is a lot of scope for further work in this area dealing with processing and analyses.

Acknowledgement

The authors thank Dr. S. Ganesan for participation and guidance in this work and valuable suggestions in preparation of this report. We are also thankful to Shri. R. Shankar Singh for his constant encouragement.

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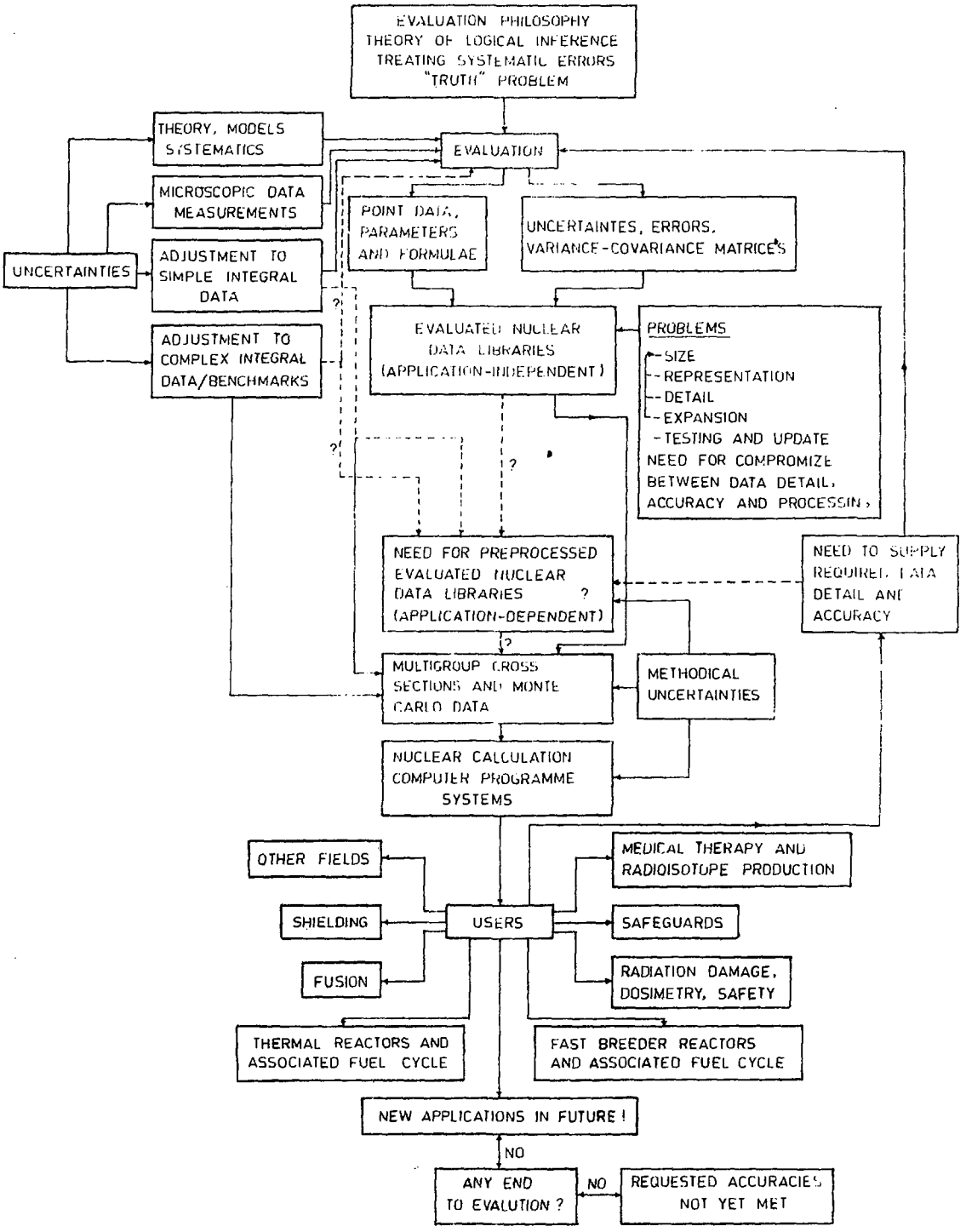
EVALUATION PHILOSOPHY
THEORY OF LOGICAL INFERENCE
TREATING SYSTEMATIC ERRORS
"TRUTH" PROBLEM
THEORY, MODELS
SYSTEMATICS
EVALUATION
MICROSCOPIC DATA
MEASUREMENTS
POINT DATA,
PARAMETERS
AND FORMULAE
UNCERTAINTES, ERRORS,
VARIANCE-COVARIANCE MATRICES
UNCERTAINTIES
ADJUSTMENT TO
SIMPLE INTEGRAL
DATA
ADJUSTMENT TO
COMPLEX INTEGRAL
DATA/BENCHMARKS
EVALUATED NUCLEAR
DATA LIBRARIES
(APPLICATION-INDEPENDENT)
PROBLEMS
-SIZE
-REPRESENTATION
-DETAIL
-EXPANSION
-TESTING AND UPDATE
NEED FOR COMPROMIZE
BETWEEN DATA DETAIL,
ACCURACY AND PROCESSING
NEED FOR PREPROCESSED
EVALUATED NUCLEAR
DATA LIBRARIES ?
(APPLICATION-DEPENDENT)
NEED TO SUPPLY
REQUIRED DATA
DETAIL AND
ACCURACY
MULTIGROUP CROSS
SECTIONS AND MONTE
CARLO DATA
METHODICAL
UNCERTAINTIES
NUCLEAR CALCULATION
COMPUTER PROGRAMME
SYSTEMS
OTHER FIELDS
MEDICAL THERAPY AND
RADIOISOTOPE PRODUCTION
SHIELDING
USERS
SAFEGUARDS
FUSION
RADIATION DAMAGE,
DOSIMETRY, SAFETY
THERMAL REACTORS AND
ASSOCIATED FUEL CYCLE
FAST BREEDER REACTORS
AND ASSOCIATED FUEL CYCLE
NEW APPLICATIONS IN FUTURE !
NO
ANY END
TO EVALUTION ?
NO
REQUESTED ACCURACIES
NOT YET MET

TABLE 1

List of Isotopes/Elements for which Complete Data is available in Kalpakkam Multigroup Set - 85

S.N	IGC No.	Material	ENDF Mat No.
1	101	U-235	1261
2	102	U-238	1262
3	103	Pu-239	1264
4	104	Pu-240	1265
5	105	Pu-241	1266
6	106	Pu-242	1161
7	107	Ni*	1190
8	108	Cr*	1191
9	109	Fe*	1192
10	110	Na*	1156
11	111	Al*	1193
12	112	Si*	1194
13	113	Mn*	1197
14	114	Mo*	1287
15	115	C*	1274
16	116	N*	1275
17	117	O*	1276
18	118	Th-232	1296
19	119	B-10	1273
20	120	B-Nat	1160
21	121	Be	1289
22	122	U-235	1260
23	123	U-234	1043
24	124	Ba-Nat	7840 (ENDL-84)
25	125	U-236	1163
26	126	U-237@	8237 (ENDF/B-V actinide)
27	127	U-239@	7872 ENDF -78
28	128	Np-237@	1263
29	129	Np-238@	8338 (ENDF/B-V actinide)
30	130	Np-239@	1931 JENDL-1
31	131	Pu-238@	1050
32	132	Pu-243@	8443 (ENDF/B-V actinide)
33	133	Am-241@	1361
34	134	Am-242M@	8542 (ENDF/B-V actinide)
35	135	Am-242 @	1369
36	136	Am-243@	1363
37	137	Cm-242@	8642 (ENDF/B-V actinide)
38	138	Cm-243@	1343
39	139	Cm-244@	1162

* includes Part II fine group elastic scattering data

@ infinite dilute cross section data only

T A B L E 2

Energy Boundaries of the Kalpakkam Multigroup Set

Group No.	Energy-Boundaries Upper		Lower		Standard Weighting Ø (E)
1.	14.5	MeV	3.68	MeV	
2.	3.68	MeV	2.23	MeV	∫E*exp(-E/1.4*10**6)
3.	2.23	MeV	1.36	MeV	
4.	1.36	MeV	822.13	KeV	E**-2
5.	822.13	KeV	498.65	KeV	
6.	498.65	KeV	302.45	KeV	
7.	302.45	KeV	183.44	KeV	
8.	183.44	KeV	111.26	KeV	E**-1
9.	111.26	KeV	67.48	KeV	
10.	67.48	KeV	40.93	KeV	
11.	40.93	KeV	24.83	KeV	
12.	24.83	KeV	15.60	KeV	
13.	15.60	KeV	9.13	KeV	
14.	9.13	KeV	5.54	KeV	E**-0.5
15.	5.54	KeV	3.36	KeV	
16.	3.36	KeV	2.04	KeV	
17.	2.04	KeV	1.24	KeV	
18.	1.24	KeV	749.68	eV	
19.	749.68	eV	454.71	eV	
20.	454.71	eV	275.79	eV	flat
21.	275.79	eV	101.46	eV	
22.	101.46	eV	22.64	eV	
23.	22.64	eV	3.06	eV	
24.	3.06	eV	0.414	eV	E
25.	0.414	eV	0.025	eV	E*exp(-E/0.025)

TABLE 3

Dilutions (Background cross section) in barns for the Various Isotopes available for Self Shielding in Kalpakkam Multigroup cross section set

Sl. No.	IGC No.	Material	UR	RR
1.	101	U-235	10^4, 10^3, 10^2	10^5, 10^4, 10^3, 10^2, 10, 0
2.	102	U-238	10^3, 300, 50, 10	10^5, 10^4, 10^3, 10^2, 10
3.	103	Pu-239	600, 100	10^5, 10^4, 10^3, 10^2, 10, 0
4.	104	Pu-240	2×10^4, 10^4, 4×10^3, 10^3	10^5, 10^4, 10^3, 10^2, 10, 0
5.	105	Pu-241	2×10^5, 2×10^4, 10^4, 10^3	- do -
6.	106	Pu-242	10^5, 10^4	- do -
7.	107	Ni	-	- do -
8.	108	Cr	-	- do -
9.	109	Fe	-	- do -
10.	110	Na	-	- do -
11.	113	Mn	-	- do -
12.	114	Mo	6×10^3, 700	- do -
13.	118	Th-232	10^4, 10^3, 10^2, 10, 0	- do -
14.	122	U-233	-	10^5, 10^4, 10^3, 10^2, 10, 1
15.	123	U-234	-	- do -

UR - Unresolved Resonance Region

RR - Resolved Resonance Region

TABLE 4

Characteristics of Critical Assemblies

Assembly	Fissile Fuel	Fertile to Fissile Ratio	Approximate core volume (litres)	Remarks
ZPR-3-48	Pu	4.5	410	C added to soften spectrum
ZPR-3-56B	Pu	4.6	610	Predominantly Ni reflector
ZPR-6-7	Pu	6.5	3100	

TABLE 5

Composition of Critical Assemblies atom densities (atom/barn-cm)

	ZPR-3-48		ZPR-3-56B		ZPR-6-7	
	Core	Spherical Reflector	Core	Spherical Reflector	Core	Blanket
U-235	.000016	.000083	.000014	-	.0000126	.0000856
U-238	.007405	.03969	.006195	-	.00578036	.0396179
Pu-239	.001645	-	.001358	-	.00088672	-
Pu-240	.000106	-	.000181	-	.00011944	-
Pu-241	.000011	-	.001358	-	.0000133	-
Pu-242	.000004	-	.000181	-	-	-
O	-	-	.015	-	.01398	.000024
C	.02077	-	.00103	-	-	-
Na	.006231	-	.008669	.007879	.009204	-
Al	.000109	-	-	-	-	-
Cr	.002531	.001225	.0025	.001941	.002709	.001295
Fe	.01018	.004925	.0137	.007824	.01297	.004637
Ni	.001119	.000536	.00109	.042261	.001240	.0005653
Mo	.000206	-	.000343	-	.0002357	.0000038
Mn	.000106	.000051	.00022	.0003	.000212	.0000998
Si	.000124	.000060	.00022	.0003	-	-

TABLE 6

Correction Factors Applicable to K_{eff}

Assembly	1-2D	*Diffusion* to transport	Heterogeneity	Total Correction factor
ZPR-3-48	-.0009	.0064	.0183	.0238
ZPR-3-56B	-.0166	.0065	.0102	.0001
ZPR-6-7	-.0020	.0016	.0166	.0162

TABLE 7

Comparison of Calculated values of K_{eff} of Experimental Critical Assemblies

Assembly	JENDL-1	JFS-2	ENDF/B-IV JAERI	ENDF/B-IV Hardie et al.	ENDF/B-IV Kalpakkam Set	Adjusted French Set
ZPR-3-48	1.0005	1.0031	0.9885	1.0015	0.9946	1.0069
ZPR-3-56B	0.9957	0.9967	0.9768	0.9882	0.9979	1.0164
ZPR-6-7	0.9983	1.0033	0.9810	0.9917	0.9867	1.0049

TABLE 8a

Ratio of ^{238}U fission rate to ^{235}U fission rate at core centre

Assembly	Experimental	C/E					
		ENDF/B-IV Hardie et al	ENDF/B-IV JAERI	ENDF/B-IV KALPAKKAM	FRENCH	JENDL-1	JFS-2
1	2	3	4	5	6	7	8
ZPR-3-48	.0326	1.026	1.027	0.935	0.934	0.997	1.019
ZPR-3-56B	.0308	0.940	0.963	0.868	0.860	0.934	0.947
ZPR-6-7	0.230	0.914	0.933	0.869	0.871	0.911	0.916

TABLE 8b

Ratio of ^{239}Pu fission rate to ^{235}U fission rate at core centre

1	2	3	4	5	6	7	8
ZPR-3-48	0.976	0.993	0.984	0.976	0.991	0.973	0.983
ZPR-3-56B	1.028	0.994	0.942	0.932	0.939	0.927	0.936
ZPR-6-7	0.953	0.961	0.957	0.956	0.969	0.946	0.954

TABLE 8c

Ratio of ^{240}Pu fission rate to ^{235}U fission rate at core centre

1	2	3	4	5	6	7	8
ZPR-3-48	0.243	1.040	1.051	0.969	0.993	0.988	1.048
ZPR-3-56B	0.282	0.824	0.846	0.764	0.779	0.793	0.849
ZPR-6-7	-	-	-	-	-	-	-

T A B L E 9a

Ratio of ^{238}U Capture rate to ^{235}U fission rate at core centre

Assembly	Experimental	C/E					
		ENDF/B-IV Hardie et al	ENDF/B-IV JAERI	ENDF/B-IV KALPAKKAM	FRENCH	JENDL-1	JFS-2
1	2	3	4	5	6	7	8
ZPR-3-48	0.138	0.963	0.974	0.988	0.930	0.953	0.953
ZPR-3-56B	-	-	-	-	-	-	-
ZPR-6-7	0.136	1.044	1.046	1.053	0.997	1.016	1.015

T A B L E 9b

Ratio of ^{238}U Capture rate to ^{239}Pu fission rate at Core Centre

1	2	3	4	5	6	7	8
ZPR-3-48	0.141	0.970	0.993	1.015	0.941	0.983	0.972
ZPR-3-56B	-	-	-	-	-	-	-
ZPR-6-7	0.143	1.086	1.091	1.099	1.027	1.072	1.063

T A B L E 10a

Central Reactivity worths of Nickel

Assembly	INHOUR/KG			Factor		Calculated C/E		
	Experimental	Kalpakkam	French	1D	2D	ENDF/B-IV Hardie et al	ENDF/B-IV Kalpakkam	French set
1	2	3	4	5		6	7	8
ZPR-3-48	- 18.2	- 22.3	- 16.6	0.9914		1.391	1.215	0.904
ZPR-3-56B	- 16.8	- 19.5	- 13.5	1.0285		1.300	1.194	0.826
ZPR-6-7	- 6.5	- 7.9	- 5.6	1.0039		1.284	1.220	0.865

T A B L E 10b

Central Reactivity Worths of Chromium

1	2	3	4	5	6	7	8
ZPR-3-48	- 12.3	- 18.4	- 13.0	0.9890	1.529	1.479	1.045
ZPR-3-56B	- 12.7	- 15.5	- 10.3	1.0308	1.271	1.258	0.836
ZPR-6-7	- 4.5	- 6.8	- 4.6	1.0040	1.489	1.517	1.026

T A B L E 10c

Central Reactivity Worths of Iron

1	2	3	4	5	6	7	8
ZPR-3-48	- 12.2	- 15.1	- 13.4	0.9921	1.250	1.228	1.090
ZPR-3-56B	- 12.3	- 12.1	- 10.0	1.0277	1.038	1.011	0.836
ZPR-6-7	- 4.3	- 5.2	- 5.5	1.0042	1.207	1.214	1.051

TABLE 10d

Central Reactivity Worths of ^{235}U

Assembly	INHOUR/KG			Factor		Calculated C/E				
	Experimental	Kalpakkam	French	1D	2D	ENDF/B-IV Hardie et al	ENDF/B-IV Kalpakkam	French	JENDL-1	JFS-2
1	2	3	4	5		6	7	8	9	10
ZPR-3-48	334	405	374	0.9941		1.189	1.205	1.113	1.239	1.174
ZPR-3-56B	295	321	287	1.0635		1.239	1.157	1.035	1.254	1.217
ZPR-6-7	133	158	144	1.0042		1.204	1.193	1.087	1.233	1.180

TABLE 10e

Central Reactivity Worths of ^{238}U

1	2	3	4	5	6	7	8	9	10
ZPR-3-48	- 23.6	- 27.8	- 24.3	0.9913	1.029	1.168	1.020	-	-
ZPR-3-56B	- 18.4	- 23.1	- 20.2	1.0175	1.221	1.277	1.117	-	-
ZPR-6-7	- 10.9	- 11.5	- 10.4	0.9970	1.013	1.052	0.951	-	-

T A B L E 10f

Central Reactivity Worths of ^{239}Pu

Assembly	INHOUR/KG			Factor		Calculated C/E				
	Experimental	Kalpakkam	French	1D	2D	ENDF/B-IV Hardie et al	ENDF/B-IV Kalpakkam	French	JENDL-1	JFS-2
1	2	3	4	5		6	7	8	9	10
ZPR-3-48	445	531	502	0.9941		1.178	1.186	1.121	1.202	1.177
ZPR-3-56B	372	418	383	1.0644		1.290	1.196	1.096	1.275	1.277
ZPR-6-7	158	190	178	1.0041		1.222	1.207	1.131	1.228	1.214

T A B L E 10g

Central Reactivity Worths of Sodium

1	2	3	4	5	6	7	8	9	10
ZPR-3-48	- 6.3	- 11.6	- 13.35	0.9880	2.053	1.819	2.094	-	-
ZPR-3-56B	- 8.9	- 20.1	- 20.8	0.9825	1.905	2.219	2.296	-	-
ZPR-6-7	- 6.8	- 8.6	- 9.0	0.9895	1.128	1.251	1.310	-	-

TABLE 11a

Central Reactivity Worths of ^{235}U normalised to those of ^{239}Pu

Assembly	*Experimental	Calculated C/E					
		ENDF/B-IV Hardie et al	ENDF/B-IV JAERI	ENDF/B-IV KALPAKKAM	FRENCH	JENDL-1	JFS-2
1	2	3	4	5	6	7	8
ZPR-3-48	0.738	1.009	1.025	1.016	0.993	1.030	0.997
ZPR-3-56B	0.779	0.960	0.975	0.967	0.944	0.985	0.956
ZPR-6-7	0.827	0.985	0.999	0.988	0.961	1.005	0.973

TABLE 11b

Central Reactivity Worths of ^{238}U normalised to those of ^{239}Pu

1	2	3	4	5	6	7	8
ZPR-3-48	-0.0528	0.874	0.882	0.985	0.910	0.970	0.907
ZPR-3-56B	-0.0493	0.947	0.983	1.068	1.019	1.099	1.020
ZPR-6-7	-0.0686	0.829	0.833	0.872	0.841	0.913	0.864

TABLE 11c

Central Reactivity Worths of Chormium normalised to those of ^{239}Pu

1	2	3	4	5	6	7	8
ZPR-3-48	-0.00602	1.298	1.391	1.247	0.932	1.033	1.328
ZPR-3-56B	-0.00745	0.985	1.106	1.052	0.763	0.852	1.065
ZPR-6-7	-0.00623	1.218	1.284	1.257	0.907	1.002	1.250

TABLE 11d

Central Reactivity Worths of Iron normalised to those of ^{239}Pu

Assembly	*Experimental	Calculated C/E					
		ENDF/B-IV Hardie et al	ENDF/B-IV JAERI	ENDF/B-IV KALPAKKAM	FRENCH	JENDL-1	JFS-2
1	2	3	4	5	6	7	8
ZPR-3-48	-0.00646	1.061	1.092	1.035	0.972	0.924	1.017
ZPR-3-56B	-0.00772	0.805	0.866	0.845	0.763	0.766	0.818
ZPR-6-7	-0.00630	0.988	0.977	1.006	0.929	0.869	0.930

TABLE 11e

Central Reactivity Worths of Nickel normalised to those of ^{239}Pu

1	2	3	4	5	6	7	8
ZPR-3-48	-0.01006	1.181	1.185	1.025	0.806	1.125	1.161
ZPR-3-56B	-0.01109	1.008	1.047	0.998	0.754	1.013	1.038
ZPR-6-7	-0.01003	1.051	1.026	1.011	0.765	0.975	1.026

* Ratio of Reactivity worths in unit of I/H /mol.

TABLE 12

Calculated net * K_{eff} using ENDF/B-IV non adjusted multigroup sets and the 1969 adjusted French set for ZPR-3-56B assembly

S.No.		K_{eff}
1.	Experimental (BNL, 1983)	1.000 ± 0.0014
2.	Japan JAERI (Kikuchi et al, 1982)	0.9768
3.	U.S.A. (Hardie et al, 1975)	0.9882
4.	IGCAR, Kalpakkam Present Calculations	0.9991
5.	Hanford Engineering Development Laboratory (Roussin 1978)	0.9901
6.	Oak Ridge National Laboratory (Roussin 1978)	0.9919
7.	French (1969) adjusted set present calculations	1.0163
8.	French set (case 7) with Cr, Fe, Ni cross sections replaced from non adjusted Kalpakkam Set (Case 4) for	
	(a) both reflector and core	1.0017
	(b) reflector only	1.0003

* See foot note in Table 3 for a meaning of 'net k_{eff}'