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ATOMIC ENERGY ESTABLISHMENT

REACTOR AND NEUTRON PHYSICS DEPARTMENT

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CONTINTS

ABSTRACT

Study and calculations of physical characteristics and various parameters of large fast reactor have been carried out to explore the capabilities and confirm the validation of the methods used. These investigations were based on the well known model of "Baker" for standard large sodium cooled type of fast reactors. The main features of this model are: spherical core of radius ≈ 84 cm, with spherical annular blanket of ≈ 46 cm thick and appropriate vacuum boundary. Multigroup diffusion theory is used to formulate the problem under consideration.

Comparative study was performed to emphasize the effects of using different nuclear data systems and methods on the various parameters of the fast reactor. Multigroup libraries as 11 (ANL-5800) and 26 (BNAB-64) energy group systems of nuclear data constants were used in the present work. The calculations were carried out for both infinite dilution (self-shielding factor $F = 1$) and self-shielded cross sections. Various computer codes were elaborated and derived to meet the conditional requirements for such calculations. The important output of these calculations are the neutron spectra, neutron balance, fission and capture rate distributions, critical mass, breeding ratio in each region and total breeding ratio of the reactor.

Five different cases of study were considered employing two systems of constants, infinite dilution and self-shielded cross-sections and treating stainless steel of the reactor as to be substituted by iron. Calculations were performed, for these five different cases to evaluate the differences in the value of parameters, e.g., critical mass of plutonium-239. In case of using infinite dilution group cross sections, critical mass decreases when Ni and Cr (of stainless steel) were substituted by iron. However, in case of employing self-shielded group cross-sections critical mass increases.

Moreover, calculations have been concerned for averaged one group nuclear data constants which were condensed from the 11 and 26 group systems. Comparisons of the multigroup results with those of the one group were made. The condensation process for averaging to one group was done to estimate the effect of such physical simplification on the calculated parameters.

The present work results have been compared with many published works. Fair agreements are obtained, which varified the consistance and completeness of the methods implemented and used.

INTRODUCTION

In reactor designs the multigroup methods of neutron calculations are widely used. Within the last few years both the theory of these methods and multigroup calculation techniques on high speed computers have been developed.

Multigroup method simplifies neutron slowing down analysis in complex media. A detailed representation of calculation methods given by Weinberg and Wigner [1], Marchuk [2], Galanin [3] and others are devoted to the theory and methods of fast reactor calculations.

The methods of calculations of the critical mass, flux distribution, breeding ratio and other fast reactor physics parameters are reviewed. Results of calculations depend on the method and the type of nuclear data used. The meeting of the working group on the physics of fast reactors [4] and onward [5,6,7] reviewed the importance of using "Baker" model in present work.

A comparison has been published [8] for the selfshielded group cross-sections calculations which were obtained using different nuclear data systems such as BNAB-70, BNAB-M, KFKINR, and ENDF/B-4.

The Abagyan system of constants (BNAB) [9] is being used for reactor calculations. The energy interval in this system of constants (BNAB) assures the possibility of its use for different applications. In order to use this system of group constants it was necessary to extend the energy from 11 up to 26 groups. The theoretical concepts used in RAM-1 [10] code lead to the formulation of the calculation method.

The code RAM-1 performs calculation of effective multiplication factor (K_{eff}) and neutron flux distribution [10]. An improvement of the developed code RAM-1 was done to accelerate obtaining the eigenvalue in REPO-4 code [11].

The code REPO-5 was developed and is written in FORTRAN-IV for ICL-1905/H Computer. It contains SUBROUTINE SIGMA which performs the group constants calculation for each region. Also, it is used to calculated some parameters of plutonium fast reactor of spherical geometry. Comparisons are made between 11-group constants and that of 26-group constants. Algorithms has been introduced and programmed to RAM-1 and REPO-4 codes. Each of the 11-group [12] and 26-group nuclear data [9] for the cases under consideration were condensed to an average one group. Comparison was made between the multigroup and the averaged one group calculation.

The resonance self-shielded group cross section calculation is considered. The present work contains the results of critical mass neutron balance and breeding ratio using the followings:

(1) Infinite dilution cross-sections, BNAB-64.

(2) Self-shielded nuclear data system, using the shielding factors given in BNAB-64.

The method of the calculation, Bondarenko method [9] for the self-shielded cross-section has been used. To perform such calculations three SUBROUTINES were derived. These SUBROUTINES are briefly described as:

(1) DILUT is derived to obtain the dilution cross section $\sigma_{o,i}^{l,m}$ for element l, energy group i and reactor reactor region m.

(2) SCF2 is to perform calculation of the resonance self-shielded capture cross-section $\sigma_{c,i}^{l,m}$ and the resonance self-shielded fission cross-section $\sigma_{f,i}^{l,m}$.

(3) SETR2 is made to calculate the followings:

(a) The resonance self-shielded elastic group cross-section $\sigma_{e,i}^{l,m}$.

(b) The resonance self-shielded elastic removal group cross-section $\sigma_{er,i}^{l,m}$.

(c) The resonance self-shielded total group cross section $\sigma_{t,i}^{l,m}$.

These SUBROUTINES are written in FORTRAN-IV language, and combined together with RAM-1, REPO-4, REPO-5 and SIGMA codes to obtain the flux distribution, critical mass, neutron balance, breeding ratio certain fuel enrichment used in the reactor.

PHYSICAL CONCEPTS OF THE CALCULATIONS

The Calculation of Spectral Characteristics:

In fast reactor physics, calculations are required of the K_{eff}, and some other physical characteristics, e.g., heat release, neutron balance, the number of born neutrons, the number of fissions and captures, plutonium-239 build-up...etc. In this part the basic algorithmic formulation is done which was used to compute the corresponding characteristics. These are based on the spatial distribution of neutron fluxes and are obtained from REPO-5 REPO-4 and RAM-1 codes.

Neutron Energy Spectrum:

Neutron spectrum is the most important reactor characteristic. The fast reactor spectrum formation is mostly due to inelastic neutron scattering in fuel and structural material nuclei. Neutron energy spectrum calculation over separate reactor regions is carried out using the following formulae [7]

In core $$I_C^{(k)} = \frac{\int_{V_C} \phi^{(k)}(r)\, dV}{\sum_k \left[\int_{V_C} \phi^{(k)}(r)\, dV \right]}$$

In blanket
$$I_B^{(k)} = \frac{\int_{V_B} \phi^{(k)}(r)\, dV}{\sum_k \left[\int_{V_B} \phi^{(k)}(r)\, dV \right]} \qquad (2)$$

In blanket normalized by core flux:

$$I_{B'}^{(k)} = \frac{\int_{V_B} \phi^{(k)}(r)\, dV}{\sum_k \left[\int_{V_C} \phi^{(k)}(r)\, dV \right]} \qquad (3)$$

Where V_C and V_B are volumes the core and blanket (or reflector). Certain normalizations are introduced either for unit area in the core or at each region to make comparison convenient.

Fission Capture Rate Distribution and Total Neutron Flux:

In order to calculate the heat release it is necessary to obtain various rate distributions of neutron interactions as well as total neutron flux distribution. The total neutron flux is obtained by summing the fluxes of separate groups, therefore:

$$\phi_t(r) = \sum_k \phi^{(k)}(r) \qquad (4)$$

For convenience, the following normalization was used:

$$\phi_{nor}(r) = \frac{\phi_t(r)}{\phi_t(0)} \qquad (5)$$

where $\phi_t(0)$ is the value of the total flux in the centre of the core. Spatial distributions of fission and capture rate in RFPO-5 was calculated respectively, as follows [13].

$$N_{f,i}(r) = \sum_k \gamma_i\, \sigma_{f,i}^{(k)}\, \phi^{(k)}(r) \qquad (6)$$

$$N_{c,i}(r) = \sum_{k} \rho_i \, \sigma_{c,i}^{(k)} \, \phi^{(k)}(r) \qquad (7)$$

where: i is for the isotope,

k is the energy group,

ρ_i is the nuclear density of the isotope i, and

$\sigma_{f,i}$ & $\sigma_{c,i}$ are the fission and capture cross section, respectively.

Neutron Balance:

For neutron balance the following has to be calculated [8]:

(a) Number of neutrons born at region "m" for every isotope i:

$$Q_{m,i} = \sum_{k} \left[\int_{V_m} \rho_{m,i} \, (\nu \sigma_f)_i^{(k)} \phi_m^{(k)}(r) \, dV \right] \qquad (8)$$

(b) Number of capture at region "m" for every isotope "i"

$$N_{c,m,i} = \sum_{k} \left[\int_{V_m} \rho_{m,i} \sigma_{c,i}^{(k)} \phi_m^{(k)}(r) dV \right] \qquad (9)$$

(c) Number of fissions at region "m" for every isotope "i"

$$N_{f,m,i} = \sum_{k} \left[\int_{V_m} \rho_{m,i} \, \sigma_{f,i}^{(k)} \phi_m^{(k)}(r) \, dV \right] \qquad (10)$$

(d) Total number of neutrons born in the whole of reactor volume:

$$Q_t = \sum_{m} \sum_{i} Q_{m,i} \qquad (11)$$

(e) Total number of fissions:

$$N_{f,t} = \sum_{m} \sum_{i} N_{f,m,i} \tag{12}$$

(f) Total number of captures:

$$N_{c,t} = \sum_{m} \sum_{i} N_{c,m,i} \tag{13}$$

Therefore, the neutron balance will be as follows:

$$\frac{Q_t}{K_{eff}} - N_{f,t} - N_{c,t} = L \tag{14}$$

where L is the neutron leakage from the reactor.

All the constituents of neutron balance are normalized in such a way that the total number of neutrons born in the reactor is 100.

For the sake of convenience in the present analysis the following values are calculated:

$$P_{m,i} = \frac{100\ Q_{m,i}}{Q_t} \tag{15}$$

$$F_{m,i} = \frac{100\ N_{f,m,i}}{Q_t} \tag{16}$$

$$C_{m,i} = \frac{100\ N_{c,m,i}}{Q_t} \tag{17}$$

From these expression it is, directly, obtained, the fission and capture numbers (in percentage to the number of born neutrons) for each isotope at each reactor region.

Plutonium Build-up:

Taking into account the neutron balance components as the basis, one may obtains the expressions which determine plutonium build-up in the reactor.

The quantity obtained of plutonium has been defined by the conversion coefficient for uranium-fueled reactors and by breeding ratios for plutonium-fueled reactors. These values have been determined as the number of plutonium nuclei obtained for one of the fuel nucleus (Uranium-235 or Plutonium-239) disappeared as a result of fission and capture.

Thus, to compute the conversion coefficient (breeding ratio) it is necessary to know the absorption density distribution of Uranium-238 through the reactor, fission and capture densities of the fuel materials (Uranium-235 or Plutonium-239). The above-mentioned parameters are obtained during neutron balance calculation, which facilitates the calculation of plutonium build-up.

One may writes down the mathematical expressions for breeding ratio in the following way [14]:

$$BR = \frac{N_{c,8}}{N_{f,fuel} + N_{c,fuel}} \tag{18}$$

where:

$$N_{c,g} = \sum_{m} N_{c,m,g}$$

$$N_{f,fuel} = \sum_{m} N_{f,m,g(9)}$$

$$N_{c,Fuel} = \sum_{m} N_{c,m,g(9)}$$

The expression for breeding ratio in separate regions may be of the form:

$$BR_{m} = \frac{N_{c,m,g}}{N_{F,Fuel} + N_{C,Fuel}} \qquad (19)$$

<u>Group Averaging and Dilution Calculation of Cross Sections:</u>

Many have reviewed the dilution cross-section and the self-shielding factor [9,13,15,16,17]. The effect of the self-shielding factor in the calculation of the critical mass and radiation shielding are considered. In the present work use is made of the Bondarenko [9] self-shielding factors. The Bondarenko concept is useful and it has increasing favour. This is due to its application to narrow group cross-sections (and even to broad group cross-section) which are valid for large range of compositions.

In this method the neutron spectrum is assumed as the product of two parts. The first part is $\phi_0(E)$ a gross one, while the second one is a fine distribution F(E). The fine distribution F(E) is assumed to be inversely proportional to the total macroscopic cross section, $\sum_t$ so that, one can write:

$$\phi(E) = \phi_o(E) \cdot \frac{1}{\sum_t (E)}$$

As $\phi(E)$ within a group varies practically as 1/E, then:

$$\phi(E) = \frac{1}{\sum_t (E)}$$

Introducing the lethargy; $\frac{E_o}{E}$, $\frac{du}{dE} = 1/E$, then :

$$\phi(u) = \frac{1}{\sum_t (u)}$$

The average cross sections of type X for region m can be written as:

$$\sigma_{x,i}^{m} = \frac{\int_{u_{i-1}}^{u_i} \frac{\sigma_x^{m}(u)}{\sum_t(u)} du}{\int_{u_{i-1}}^{u_i} \frac{1}{\sum_t(u)} du} \qquad (20)$$

Introduce the dilution cross section σ_0^{m}, which it may be written as:

$$\sigma_0^{m} = \frac{1}{\rho_m} \sum_{m \neq l} \sigma_t^{l} \rho^{l} \tag{21}$$

where ρ_m is the nuclear density.

Then, equation (20) can be written in detail as:

$$\sigma_c^{m}(\sigma_0) = \frac{\left\langle \dfrac{\sigma_c^{m}}{\sigma_t^{m} + \sigma_0^{m}} \right\rangle}{\left\langle \dfrac{1}{\sigma_t^{m} + \sigma_0^{m}} \right\rangle} \tag{22}$$

$$\sigma_t^{m}(\sigma_0) = \frac{\left\langle \dfrac{1}{\sigma_T^{m} + \sigma_0^{m}} \right\rangle}{\left\langle \dfrac{1}{(\sigma_t + \sigma_0^{m})^{2}} \right\rangle} \tag{23}$$

$$\sigma_e^{m}(\sigma_0) = \frac{\left\langle \dfrac{\sigma_e^{m}}{\sigma_t^{m} + \sigma_0} \right\rangle}{\left\langle \dfrac{1}{\sigma_t^{m} + \sigma_0^{m}} \right\rangle} \tag{24}$$

where σ_c, σ_t and σ_e are the effective capture, total and elastic cross-sections, respectively of that nuclide as energy-dependent variables, and σ_o is energy-independent. When the isotope m present in the medium has a low concentration (i.e., $\sigma_o \longrightarrow \infty$), equation (20) is for the "infinite dilution cross-section" and that is for constant Γ_{t}. Therefore, one can have the following:

$$\sigma^{m}_{x,\infty,i} = \frac{\int_{u_{i-1}}^{u_i} \sigma_x{}^{m}(u)\, du}{u_i - u_{i-1}} \qquad (25)$$

Equations (22), (23) and (24) can be written as follows:

$$\sigma_c(\infty) = \langle \sigma_c \rangle \qquad (26)$$

$$\sigma_t(\infty) = \langle \sigma_t \rangle \qquad (27)$$

$$\sigma_e(\infty) = \langle \sigma_e \rangle \qquad (28)$$

If one would like to neglect the effect of the resonance self-shielding, then one can use the infinite dilution cross-sections. These cross-sections correspond to the infinite dilution concentration of the isotope, i.e., $\sigma_o = \infty$. On the other side, when it is necessary to take the resonance self-shielding into account, a correction factor is given for different values of σ_o. Interpolation can be used to obtain the required value of the correction factor for certain dilution cross-section σ_o. The correction factor can be defined as:

$$F_c(\sigma_o) = \frac{\bar{\sigma}_c(\sigma_o)}{\sigma_c(\infty)} = \frac{1}{\langle \sigma_c \rangle} \cdot \frac{\left\langle \frac{\sigma_c}{\sigma_t + \sigma_o} \right\rangle}{\left\langle \frac{1}{\sigma_t + \sigma_o} \right\rangle} \tag{29}$$

$$F_t(\sigma_o) = \frac{\bar{\sigma}_t(\sigma_o)}{\sigma_t(\infty)} = \frac{1}{\langle \sigma_t \rangle} \cdot \left[\frac{\left\langle \frac{1}{\sigma_t + \sigma_o} \right\rangle}{\left\langle \frac{1}{(\sigma_t + \sigma_o)^2} \right\rangle} - \sigma_o \right] \tag{30}$$

and

$$F_e(\sigma_o) \quad \frac{\bar{\sigma}_e(\sigma_o)}{\sigma_e(\infty)} = \frac{1}{\langle \sigma_e \rangle} \cdot \frac{\left\langle \frac{\sigma_e}{\sigma_t + \sigma_o} \right\rangle}{\left\langle \frac{1}{\sigma_t + \sigma_o} \right\rangle} \tag{31}$$

The correction factors are defined as:

$F_c(\sigma_o)$ = The resonance self-shielding factor for capture cross-section.

$F_t(\sigma_o)$= The resonance self-shielding factor for total cross-section.

$F_e(\sigma_o)$= The resonance self-shielding factor of elastic cross-section.

Also, for the fissile element as U-235 and Pu-239, there is the resonance self-shielding factor for the fission cross section, $F_f(\sigma_o)$:

$$F_f(\sigma_o) = \frac{\bar{\sigma}_f(\sigma_o)}{\bar{\sigma}_{f(\infty)}} = \frac{1}{\langle \sigma_f \rangle} \cdot \frac{\left\langle \frac{\sigma_f}{\sigma_t + \sigma_o} \right\rangle}{\left\langle \frac{1}{\sigma_t + \sigma_o} \right\rangle} \tag{32}$$

Clearly from equations (29) to (32) for $\sigma_o = \infty$;

$$F_c(\infty) = F_t(\infty) = F_e(\infty) = F_f(\infty) = 1$$

To obtain the average cross-section of certain group for a particular isotope,one should multiply the effective cross section by the self-shielding factor as:

$$\left.\begin{aligned} \bar{\sigma}_c &= \sigma_c F_c(\sigma_o,T) \\ \bar{\sigma}_f &= \sigma_f F_f(\sigma_o,T) \\ \bar{\sigma}_e &= \sigma_e F_e(\sigma_o,T) \\ \bar{\sigma}_t &= \sigma_t F_t(\sigma_o,T) \end{aligned}\right\} \tag{33}$$

where T is the temperature of the medium.

Note that for inelastic cross-section: $\bar{\sigma}_{in} = \sigma_{in}$.

Also, when considering the resonance self-shielding into account, the transport cross-section can be obtained from the following

$$\sigma_{tr,i} = (\bar{\sigma}_{t,i} - \bar{\sigma}_{f,i} - \bar{\sigma}_{c,i} - \bar{\sigma}_{in,i})(1-U_{e,i}) + \bar{\sigma}_{c,i} + \bar{\sigma}_{f,i} + \bar{\sigma}_{in,i} \tag{34}$$

CALCULATION, RESULTS AND DISCUSSIONS

The present calculations of a plutonium fast reactor were carried out using REPO-5, REPO-4 and RAM-1 codes. Baker [4] has developed reactor model and its initial data. All the calculation were carried out for the spherical geometry. These initial data available [4] and are tabulated in Table (1). Table (2) is a representation of the nuclear densities of reactor composition in $\frac{nucl.}{cm^3} \cdot 10^{-24}$.

The present calculations are done, firstly, for three cases:

Case A: 11 energy group calculation.

Case B: 26 energy group calculation with component of stainless steel (Cr, Ni, Fe).

Case C: 26 energy group calculation, where stainless steel components are substituted by Fe.

Due to the absence of Nickel and Chromium data in the system of constants [12], stainless steel is substituted by iron, when calculation is carried out for case A. In case B all the components of the stainless steel are taken into account [9]. In case C components of stainless steel are substituted by iron to evaluate the resulting effect.

The plots of K_{eff} dependance on Pu-239 nuclear concentrations, obtained from the above calculations, are represented in Fig.(1). Nuclear concentrations of Pu-239, for K_{eff} = 1 were obtained from the graphs. Table (3) contains the results about the nuclear concentration of Pu-239 and U-238.

Table (1): Elements of "Baker" Model.

NO	Parameters	Composition
1	Fuel in the core	$PuO_2 + UO_2$
2	Coolant	Sodium
3	Structural material	Stainless steel
4	Blanket	Uranium oxide with a slight Plutonium contents
5	Radius of the core	84.196 cm
6	Blanket thickness	45.72 cm
7	The number of calculation:	
	Points in the core	14
	in the blanket	8

Table (2): Nuclear Densities (in, $\frac{\text{nucel}}{cm^3} 10^{-24}$).

Isotope	Core	Blanket
Pu + U	0.0072	-
Pu-239	-	0.00012
U-238	-	0.012
Ni	0.00088	0.00088
Fe	0.00814	0.00814
Cr	0.00198	0.00198
Na	0.0123	0.0069
O	0.0144	0.024

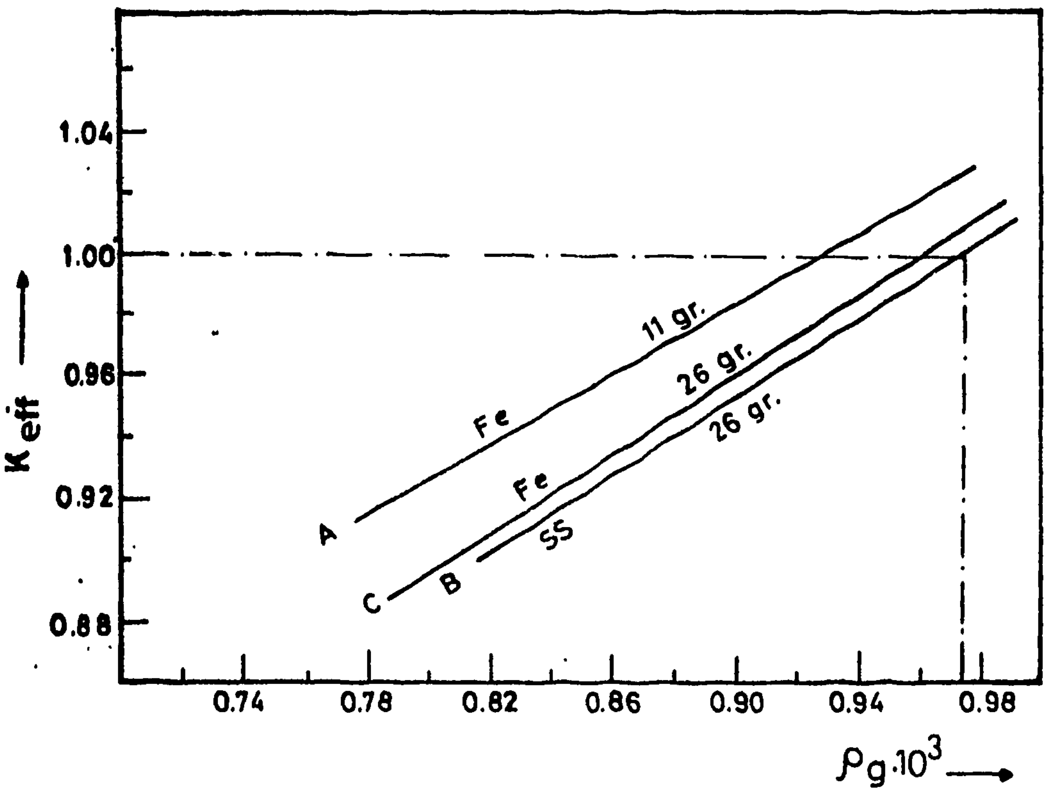

Fig.(1) K_{eff} versus Pu - 239 concentration (cases A, B and C).

The calculations with the system of constants [12] give less Plutonium-239 enrichment in comparison with those of system of constants [9].

Using the data from Tables (2) and (3) the calculations were carried out from which reactor physical characteristics were obtained. Table (4) is a representation of the reactor critical parameters.

The substitution of stainless steel by iron reduces the critical mass by 1.7 %. The use of recent system of constants "BNAB" [7] of better accuracy shows a critical mass increase.

Figure (2) shows the total flux distribution. The maximum flux difference in the core was 4.35%. Neutron energy spectrum hystograms in the core (solid line) for 26-group shown in Figs. (3) and (4) for case B and C, respectively. It is clear that the maximum value of the spectra is in the 7th group (0.2 MeV $\leqslant$ E $\leqslant$ 0.4 MeV) for the core and in the 8 th group for the blanket (0.10 MeV $\leqslant$ E $\leqslant$ 0.20 MeV). The maximum of neutron flux shifts slightly despite the substitution of the stainless steel by iron. It must be noted, however, that the ratio of the neutron flux values to the flux, for instance, in the 1 st group varies for the core and blanket spectra by about two times.

Table (3): Fuel Nuclear Concentration .. (10^{-24} nucl/cm^3) ,for K_{eff} = 1.

Element	Case A	Case B	Case C
Pu-239	0.0009281	0.0009735	0.0009605
U-238	0.0062719	0.0062265	0.0062395

Note that: $\zeta_9 + \zeta_8 = 0.0072$.

Table (4): Reactor Critical Parameters

Parameter	Case A	Case B	Case C
K_{eff}	1.0000	1.0000	0.99983
M_{cr}(in kg)	918	966	950
R_{core}(in cm)	84.196	84.196	84.196

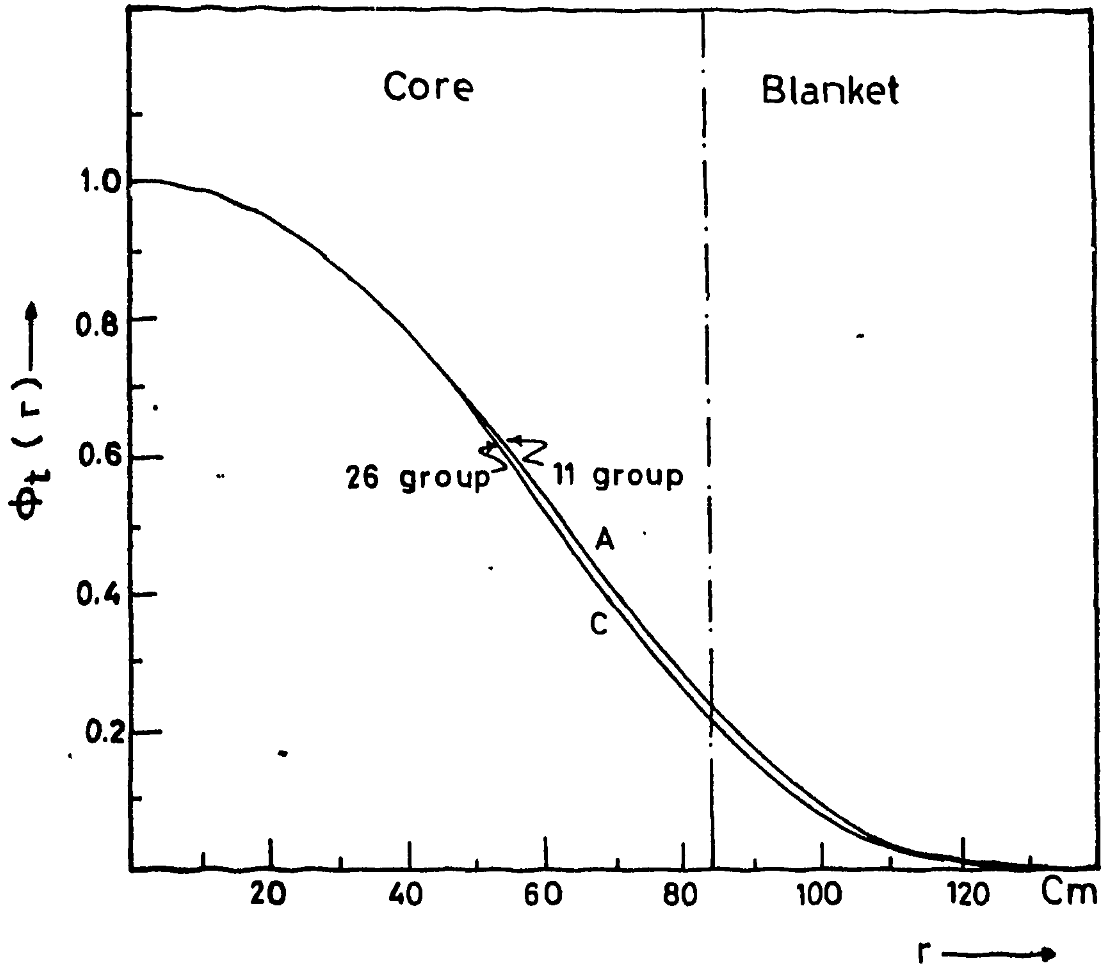

Fig. (2) Distribution of total neutron flux through the reactor radius (cases A and C).

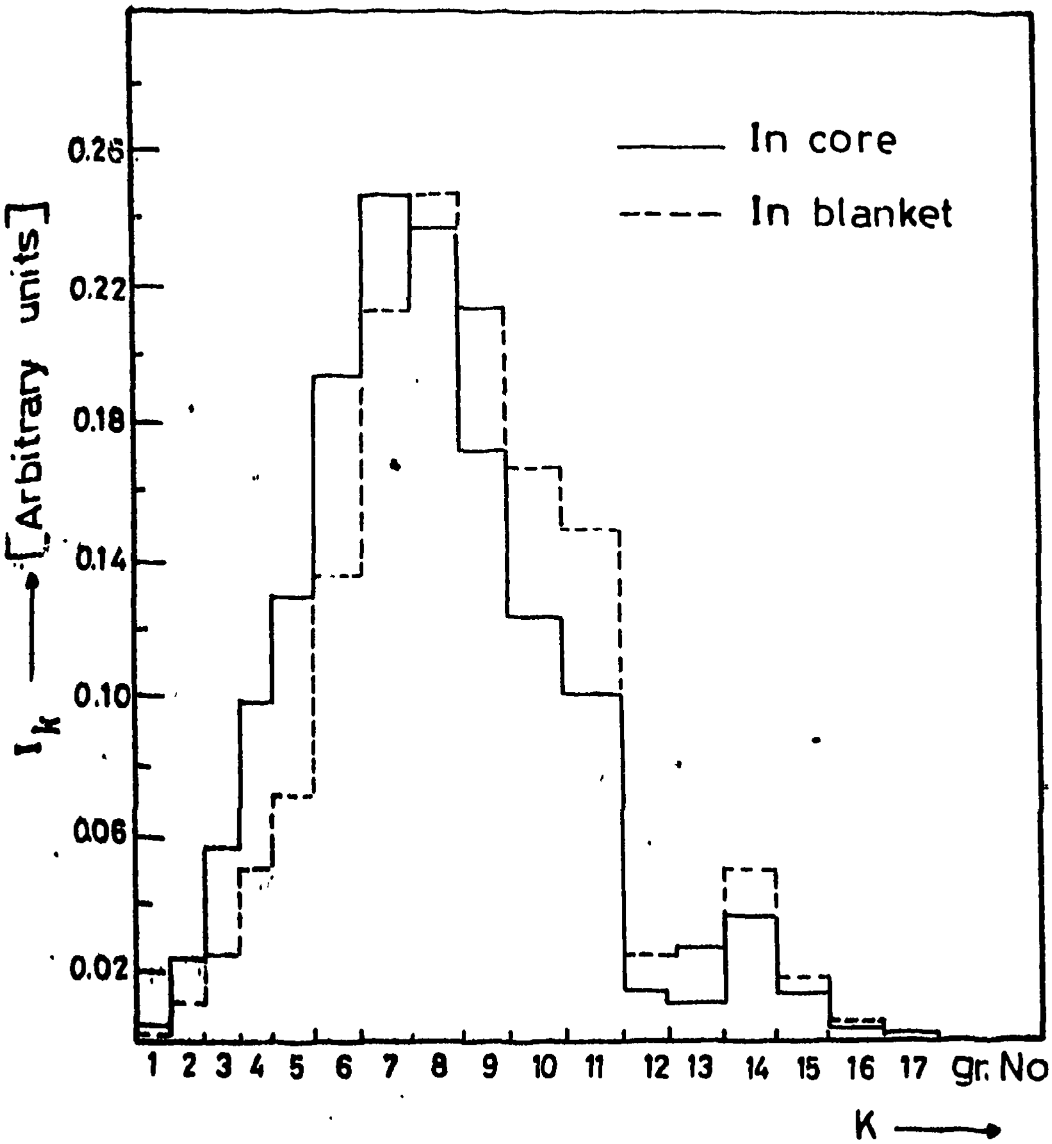

Fig. (3) Neutron spectra for case B

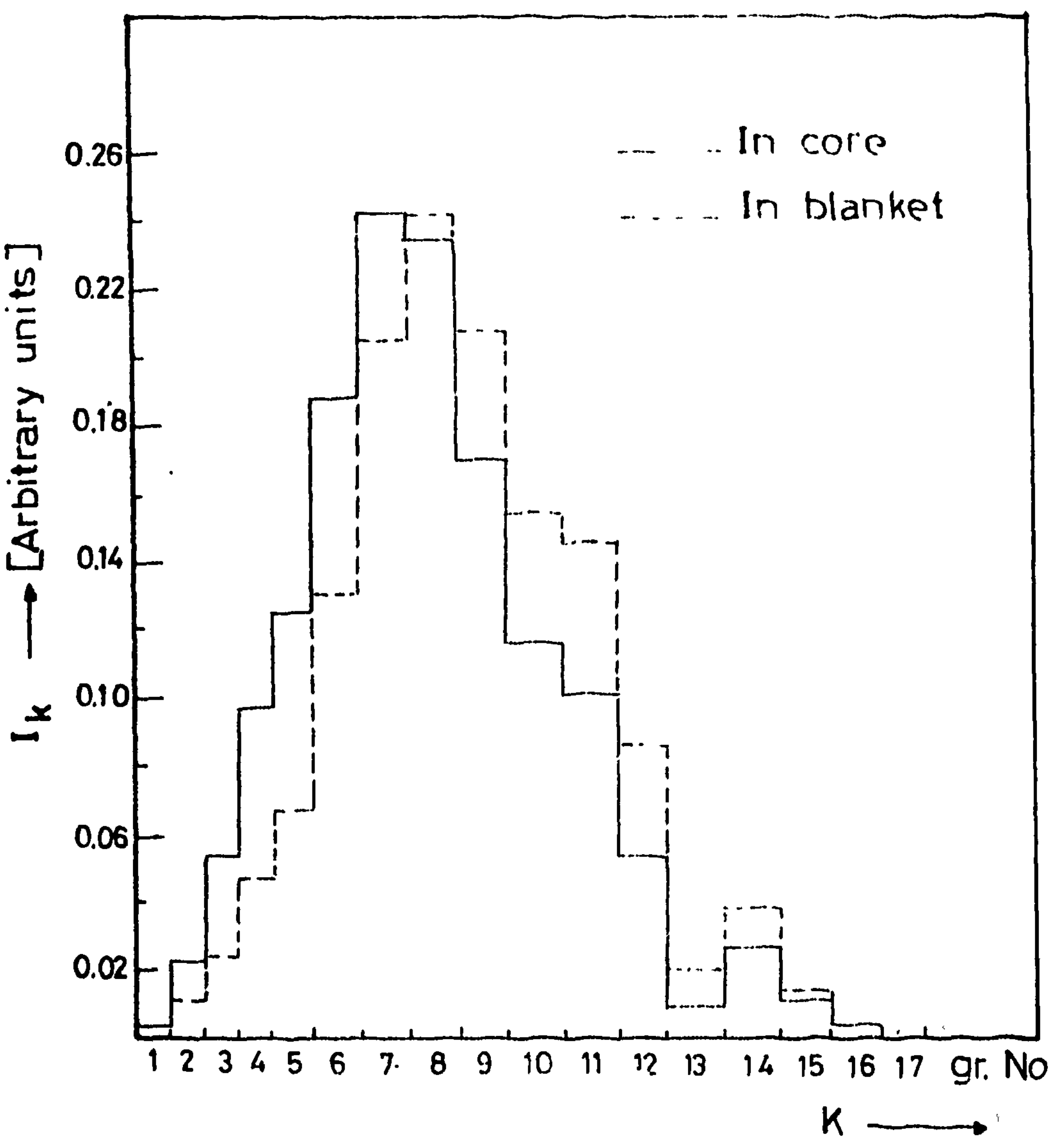

Fig. (4) Neutron spectra for case C.

The data showed that neutron spectrum in the blanket is considerably "softened" . This "softening" of the spectrum is due to the inelastic neutron scattering of U-238. Fission and capture rate distributions are shown in Figs. (5) and (6) for cases A and C. Neutron flux distributions through the reactor radius for separate groups are represented in Figs. (7) and (8) for cases A and B respectively. Neutron balance components are given in Tables (5),(6) and (7) for the cases under consideration. The data brings out explicitly that neutron leakage from the reactor with regard to the neutrons born is: 3.48%, in case A, 2.45%, in case B, and 2.60%, in case C.

The nuclear data constants of 11 group , case A, is averaged to a one group, case A : The neutron balance of case A' is given in Table (8). It is clear that the averaging of nuclear data constants from 11 group to one group decreases the breeding ratio of the reactor under consideration by an amount of -1.77%. Similarly the nuclear data constants of 26 group for case B are averaged to one group [19], case B'. The neutron balance of case B'. The neutron balance of case B' calculation is given in Table (9). By comparison between the results obtained from case B and case B', it is found a decrease in breeding ratio with an amount of -1.39% which is due to averaging. Also, the nuclear data constants of 26 group for case C are averaged to one group, case C'. The neutron balance for case C' is given in Table (10). Comparing the results obtained from case C and case C', a decrease in breeding ratio with an amount of -1.39% is noticed. The percentage difference between K_{eff} in case A and in case A' is 0.63%. While the percentage difference of K_{eff} between case B and case B' is 0.08%. For case C and case C' the percentage difference for K_{eff} is the same as that between case B and case B'.

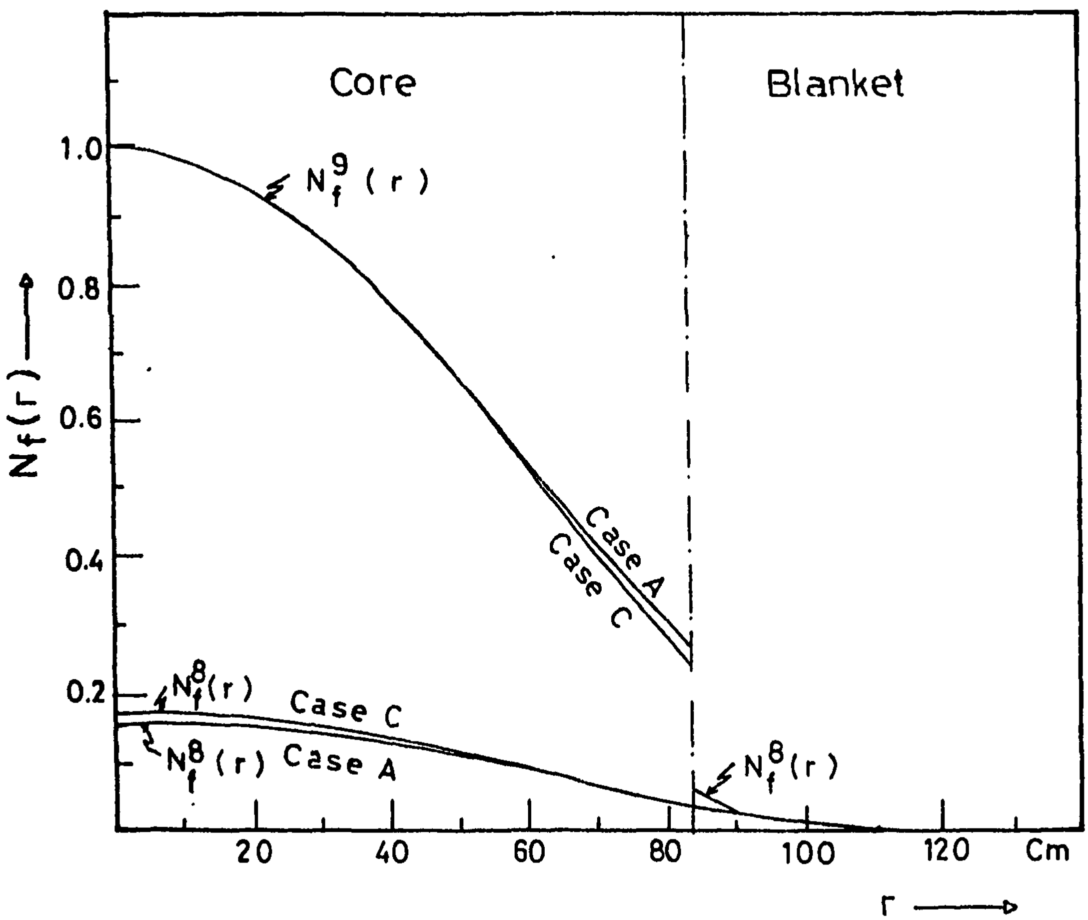

Fig. (5) Fission rate, $N_f(r)$, distribution for Pu-239 and U-238 through the reactor radius ; cases (A and C).

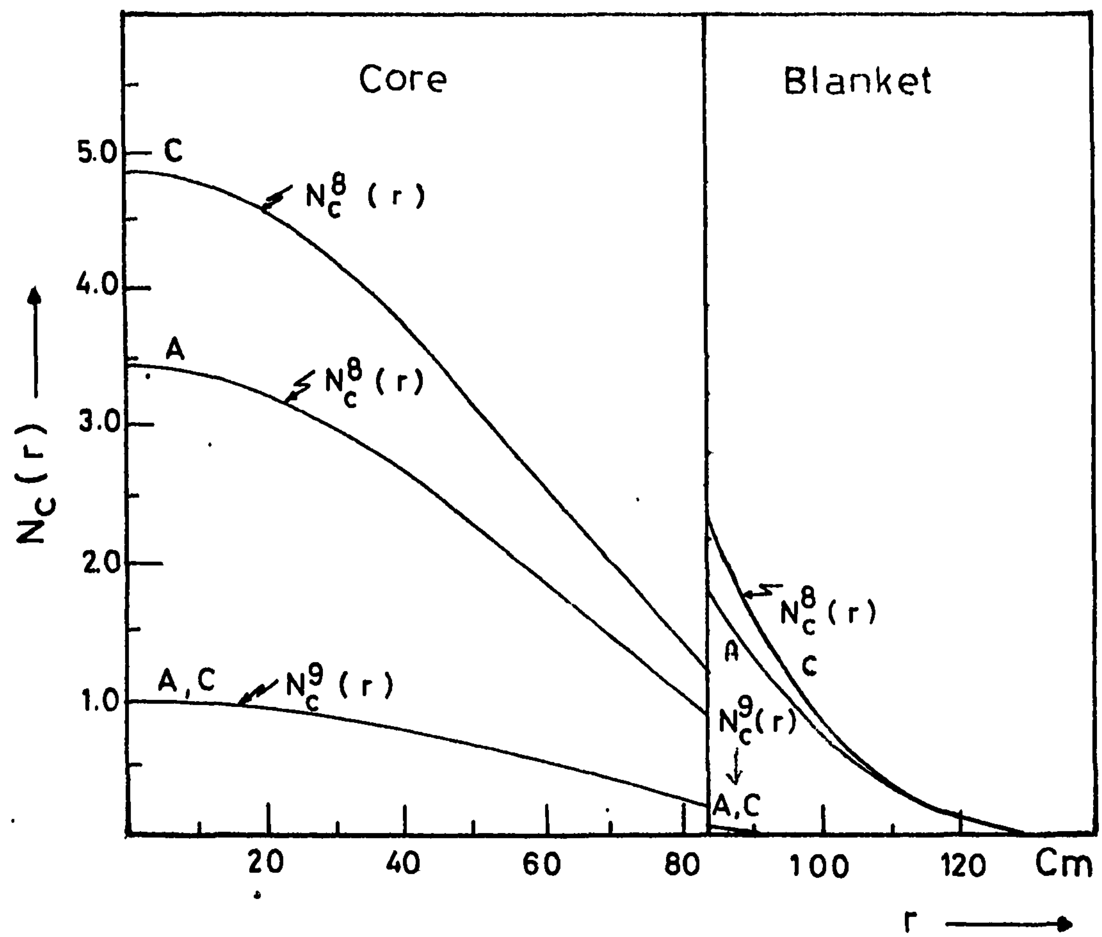

Fig. (6) Capture rate , $N_c(r)$, distribution for U-238 and Pu-239 through the reactor radius (cases A and C).

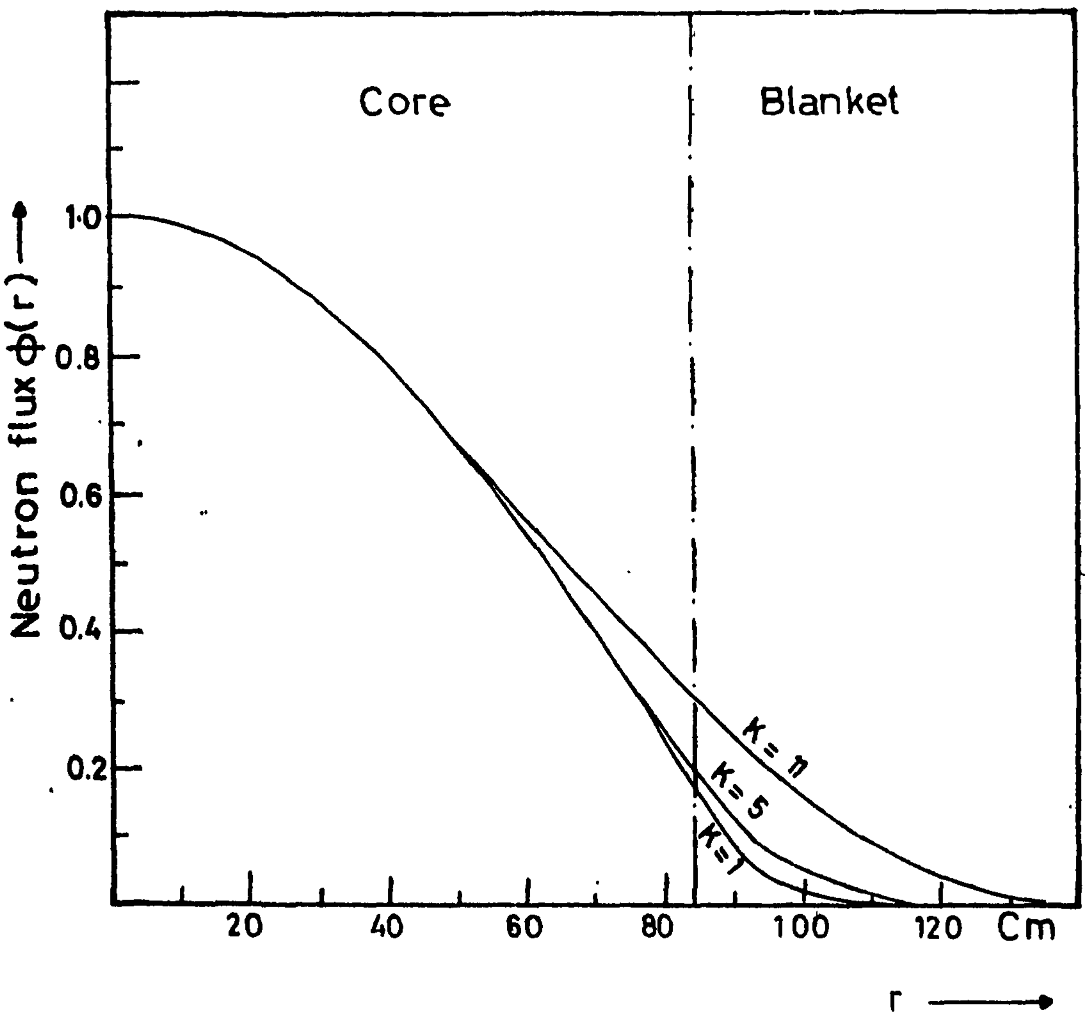

Fig.(7) Flux distribution through the reactor radius for certain groups (case A).

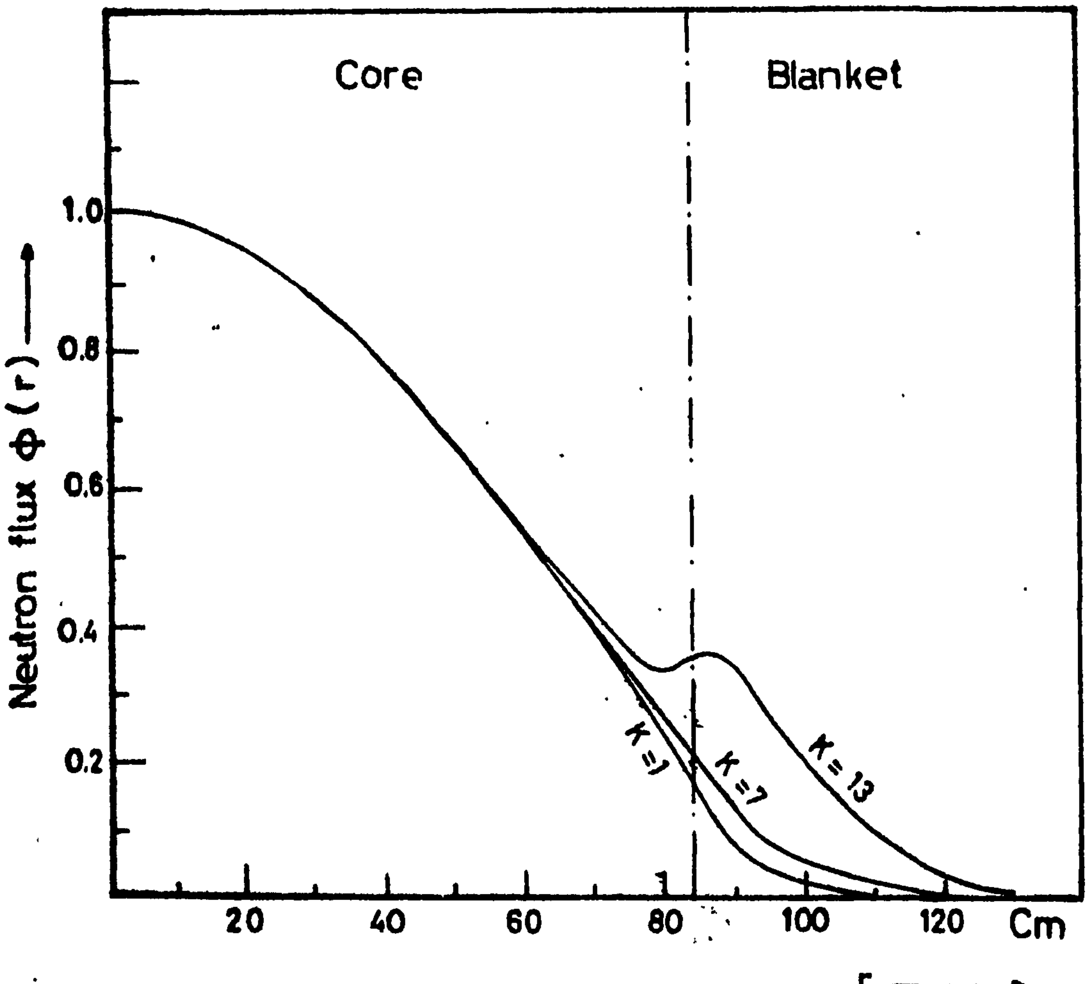

Fig. (8) Flux distribution throught the reactor radius for certain groups (case B).

Table (5): Neutron Balance for Case A

Region	Isotope	P_i	C_i	F_i
Core	Pu-239	82.367	8.2869	27.967
	U-238	11.052	28.491	4.2503
	Fe	-	1.6706	-
	Na	-	0.28600	-
Blanket	Pu-239	3.6054	0.46074	1.2315
	U-238	2.9762	22.049	1.1459
	Fe	-	0.61271	-
	Na	-	0.065689	-
Sum		100.0000	61.922	34.595
Leakage		3.4823		
Breeding Ratio		1.3319		
K_{eff}		1.0000		

Table (6)
Neutron Balance for Case B.

Region	Isotope	P_i	C_i	F_i
	Pu-239	80.767	6.5469	27.568
	U- 238	12 . 871	31.523	4.5858
	Ni	-	0.30391	-
Coer	Fe	-	1.0793	-
	Cr	-	0.24337	-
	Na	-	0.38088	-
	0	-	0.27230	-
	Pu-239	2.9611	0.30269	1.0197
	U-238	3.4013	21.831	1.2141
	Ni	-	0.073711	-
Blanket	Fe	-	0.36328	-
	Cr	-	0.083701	-
	Na	-	0.097605	-
	0	-	0.061105	-
Sum		100.0000	63.162	34.388
Leakage		2.4495		
Breeding Ratio		1.5056		
K_{eff}		1.0000		

Table (7)
Neutron Balance for Case C

Region	Isotope	P_i	C_i	F_i
Core	Pu-239	80.578	6.5074	27.510
	U-238	12.934	31.470	4.6086
	Fe	-	1.3704	-
	Na	-	0.34620	-
	0	-	0.2723	-
Blanket	Pu-239	3.0445	0.31059	1.0487
	U-238	3.4436	22.132	1.2292
	Fe	-	0.46082	-
	Na	-	0.086963	-
	0	-	0.061755	-
Sum		100.0000	66.016	34.397
Leakage		2.6046		
Breeding Ratio		1.5153		
K_{eff}		0.99983		

Table (8)
Neutron Balance for Case A'

Region	Isotope	P_i	C_i	F_i
Core	Pu-239	82.633	8.3107	28.057
	U-238	11.069	28.509	1.[illegible]
	Fe	-	1.6725	-
	Na	-	0.28626	-
Blanket	Pu-239	3.45	0.44086	1.1784
	U-238	2.848	21.098	1.0966
	Fe	-	0.58632	-
	Na	-	0.062855	-
Sum		100.0000	60.966	34.589
Leakage		3.8208		
Breeding Ratio		1.3059		
K_{eff}		1.0063		

Table (9)
Neutron Balance for Case B'

Region	Isotope	P_i	C_i	F_i
Core	Pu-239	80.957	6.5623	27.633
	U-238	12.901	31.587	4.5865
	Ni	-	0.36163	-
	Fe	-	1.0819	-
	Cr	-	0.24395	-
	Na	-	0.38177	-
	C		0.27294	
Blanket	Pu-239	2.8587	0.29223	0.98447
	U-238	3.2837	21.076	1.1722
	Ni	-	0.071162	-
	Fe	-	0.35072	-
	Cr	-	0.080807	-
	Na	-	0.004231	-
	C	-	0.058992	-
Sum		100.0000	62.469	34.380
Leakage		3.0628		
Breeding Ratio		1.4849		
K_{eff}		1.0008		

Table (10)

Neutron Balance for Case C'

Region	Isotope	P_i	C_i	F_i
Core	Pu-239	80.770	6.5199	27.576
	U-238	12.965	31.545	4.6196
	Fe	-	1.3797	-
	Na	-	0.34702	-
	O	-	0.27294	-
Blanket	Pu-239	2.9300	0.20992	1.0127
	U-238	3.3253	21.371	1.1870
	Fe	-	0.44499	-
	Na	-	0.083976	-
	O	-	0.059634	-
Sum		100.0000	62.319	34.395
Leakage		3.2225		
Breeding Ratio		1.4945		
K_{eff}		1.0000		

Table (11) gives a comparison between the present results and those previously published [4,8]. This comparison is for the percentage difference of the critical mass and the breeding ratio of our results using infinite dilution group cross-sections from BNAB-64. The critical mass obtained is near to that obtained by BNL [4], (percentage difference -0.5%) and Jaeri [4] (percentage difference 0.7%) while our breeding ratio is close to that obtained from Obninsk [4] (percentage difference 5.6%).

In the present calculations, using the self-shielded group cross-sections, the critical mass decreases and the percentage difference for the critical mass between infinite dilution group cross-sections and shielded group cross sections are 14.0%.

Let us discuss some results given by Baker and Hammond [4]. The maximum critical mass in their review is that obtained at Argonne, USA (1012.1 kg), while the smallest critical mass was obtained at Obninsk, USSR (877.2 kg). The percentage difference between the maximum and the minimum value of the critical mass is 13.3%. However, the percentage difference for the critical mass between our result (831.12 kg) and that of Winfrith, Uk (929 kg) is 10.5% which is fair to the above percent. Concerning the the breeding ratio, the percentage difference for our $Br_{s,sh}$ and that of Winfrith, UK [8] is 4.0% and with that obtained at Obninsk, USSR is 0.16%. While for the breeding ratio in Ref. [8]; the maximum value was obtained at Obninsk, USSR (1.422) and the lower one which was obtained at Cadarache, France (1.246). The percentage difference of both values is 12.4%.

Therefore, the present results for the breeding ratio is in good agreement with other authors; Table (12) gives the neutron balance for case $B_{s,sh}$ using the self-shielded group cross sections. Table(13) contains the neutron balance for case $C_{s,sh}$. In this case Ni and Cr are substituted by Fe, and the self-shielded group cross sections are used. The critical mass was 841.8 kg. It is clear that using stainless steel is better than using iron alone, because it saves fuel. When self-shielded group cross sections are used less fuel is required . This change is due to resonance self-shielding factor consideration.

Due to the differences in codes and methods employed, nuclear data and fission spectra used, acceptable differences are, clearly, observed . Also, condensation of multigroup systems of constants to one group systems were considered using self-shielded cross-sections. Calculations of the various factors and parameters were carried out, and comparative study was fruitful.

Table (11)

Percentage Difference for the Breeding Ratio and Critical Mass from Different Sources.

Organization	Breeding Ratio, BR	$\frac{(BR^{*}-BR)}{BR^{*}} \times 100$	Critical Mass, M_c Kg	$\frac{(M_c^{*} - M_c)}{M_c^{*}} \times 100$
Brookhaven USA ENDF/B (1971)	1.404	6.8	970.8	-0.5
Casaccia Italy (1971)	1.406	6.6	1002.5	-3.8
Tromby India (1971)	1.388	7.8	931.3	3.6
Obninsk USSR BNAB-64 (1971)	1.422	5.6	877.2	9.2
Jaeri Japan (1971)	1.35	10.4	958.9	0.7
Obninsk USSR (1971)	1.335	11.4	974.5	-0.9
Argonne USA (1971)	1.406	6.6	977.9	-1.2
Obninsk USSR (1978)	1.373	8.8	991.0	-2.6
Karlsruhe FRG (1978)	1.319	12.4	945.0	2.2
Present Calculation (B)* EGYPT (1983)	1.506	0.0	966.0	0.0

Table (1.2)

Neutron Balance for Case B (Self-Shielded)

(ρ_9=0.0008375x10^{-24}Nucl./cm^3, BNAB-64).

Region	Isotope	P_i	C_i	F_i
Core	Pu-239	77.781	7.3677	26.617
	U-238	12.701	27.947	4.5254
	Ni	-	0.31127	-
	Fe	-	0.89165	-
	Cr	-	0.27048	-
	Na	-	0.32794	-
	O	-	0.25446	-
Blanket	Pu-239	5.3988	0.84801	1.8666
	U-238	4.11[illegible]	22.524	1.4694
	Ni	-	0.1661	-
	Fe	-	0.41357	-
	Cr	-	0.12741	-
	Na	-	0.16395	-
	O	-	0.070253	-
Sum		100.0000	61.557	34.479
Leakage		3.9663		
Breeding Ratio		1.3752		
K_{eff}		0.99998		

Table (13):

Neutron Balance for Case C (Self-Shielded)

(ρ_9=0.00084824x10^{24}Nucl./cm^3,BNAB-64)

Region	Isotope	P_i	C_i	F_i
Core	Pu-239	77.986	7.1370	26.681
	U-238	12.766	28.231	4.5486
	Fe	-	1.0977	-
	Na	-	0.29807	-
	O	-	0.25557	-
Blanker	Pu-239	5.1584	0.77212	1.7830
	U-238	4.0897	22.962	1.4593
	Fe	-	0.49817	-
	Na	-	0.093215	-
	O	-	0.069508	-
Sum		100.0000	61.415	34.472
Leakage		4.1153		
Breeding Ratio		1.4075		
K_{eff}		0.99998		

CONCLUSION

The results presented in this report showed that the calculation technique, numerical computation and elaborated codes can be used to fairly calculate the fast physics parameters of a fast reactor. These results compete with those of international laboratories employed different techniques and computer codes.

The condensation of multigroup to an average one group of infinite dilution and self-shielded cross-sections provided a fruitful insight to reactor analysis with respect to physical simplification. However, each condition of the calculation procedure has its impact on the accuracy of computed parameter.

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