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COMMISSION OF THE EUROPEAN COMMUNITIES

TRACE: A FUEL CYCLE COMPUTER CODE FOR FAST REACTOR ANALYSIS

by

G. GRAZIANI

1971

Joint Nuclear Research Centre
Ispra Establishment - Italy

Nuclear Studies Division

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ABSTRACT

This report describes a two-dimensional computer program written for the IBM 360/65 which calculates the fuel input requirements and the neutron physics behaviour of the equilibrium fuel cycle of a fast reactor using a partial refuelling scheme.

KEYWORDS

FAST REACTORS
FUEL CYCLE
PROGRAMMING
IBM 360
2-DIMENSIONAL CALCULATIONS
BURNUP
SELF-SHIELDING
NEUTRON DIFFUSION EQUATION
NEUTRON SPECTRA
BUCKLING

INDEX

1. INTRODUCTION

In the process of designing a fast reactor, it is convenient, when fractional core loading is considered, to investigate the equilibrium fuel cycle before a large effort is spent in the project. Generally speaking it is possible to reach the equilibrium cycle following the so called "approach to equi--librium" procedure. In this case the calculation starts con--sidering the initial charge to be in the core and calculating the flux distribution and the depletion up to the time when the first reloading occurs. At that time a fraction of the burned fuel is replaced with fresh one. The calculation of the flux distribution and the depletion is repeated till the next refueling time. If this procedure is continued long en--ough, the feed fuel requirements will become stationary and the behaviour of the nuclear parameters will repeat at each cycle. The equilibrium cycle is so reached. This approach has the advantage to give all the informations on the way to the approach to the equilibrium, as well as on the equilibrium cycle itself. However this procedure is quite lenghty and com--puter time consuming. For much of the survey work on fast reactor fuel cycles, data on the approach to equilibrium are not necessary and the calculation of the running in period can be avoided. Actually the selection of a certain number of nuclear parameters from an equilibrium cycle survey is ad--vantageous because the equilibrium conditions of the fuel cycle represent the larger part of the reactor life.
The code TRACE is a two dimensional programme (R-Z) written to investigate the parameters of the equilibrium fuel cycle without generating the data for all the approach cycles. The consequent reduction of informations is counter-balanced by a large gain in computer time (in spite of the complexity of the problem). A further source of time-saving is the assump--tion that the spatial dependence of the flux can be well re--presented by one group diffusion calculation: this is accu--rate enough for this kind of fast reactor calculations. Computer times between 2 or 3 minutes can be obtained for each complete calculation on the IBM 360/65 machine.

2. THE BURN-UP CALCULATION

The code allows a depletion model in which the transmutation of a nuclide can happen by neutron capture and radioactive decay. Each nuclide may have up to two capture parents and one decay parent.

Fission yields may be specified for each individual fission product resulting from a fission of any heavy nuclide. Provision is made to take into account leakage out of the system of the volatile nuclides by a leakage constant (sec^{-1}). The restriction imposed is that no nuclide can be produced by another nuclide which is in a lower position in the list of isotopes. This implies that, for example, all fission products must follow the fissionable isotopes. All changes in nuclide densities are represented by the system of first order differential equations:

$$\frac{dN^i}{dt} = - A_{ii} N^i + \sum_{j=1}^{i-1} A_{ij} N^J \quad (i=1, NUCL) \qquad (1)$$

where NUCL is the total number of burnable elements.
The coefficients Aij of the system equations form a triangu-lar matrix and represent the transmutation rate (by decay or capture) of nuclide j into nuclide i. The diagonal ele-ments Aii are the total removal rate of the isotope i out of the system.
The code assumes that all these reaction rates are time in-dipendent. In fact they are not, because the flux spectrum varies slightly in the reload interval period. However this variation becomes negligible when the number of reloads in-creases, i.e. when the refueling interval becomes shorter. In the fast reactor core, where the conversion ratio is clo-sed to one this flux variation is in any case negligible.
In order to reduce the large matrix A, the programme, before solving the burn-up equations, investigates the structure of the matrix, separating the independent burn-up chains, i.e. splitting the matrix A in a certain number of small matrices. One of these will be the fission products chain. On the assum-ptions that the coefficients Aji are constant, the fission product source in the reactor will then be constant, i.e. pro-portional to the average fuel composition of the reactor. This is equivalent to approximate in each region of the reac-tor, the refueling scheme to the continuous reload scheme and therefore to assume that the space averaged concentrations in the zone are equivalent to the time averaged concentrations. This is in most of cases quite satisfactory.

3. THE SELF-SHIELDING CALCULATION

The space and energy distribution of the flux in the fuel cell of a single zone changes during irradiation, thereby affecting the effective cross-sections of the isotopes and the neutron balance. For an heterogeneous core, the calculation of the va--riation of the spatial form of the flux with the cell compo--sition is of great importance for the calculation of the iso--topes burn-up. For fast reactors the variation of multigroup cross-sections as a function of the composition is also impor--tant. Therefore a multigroup calculation appears to be neces--sary. However, in order to use time constant cross-sections in the depletion equations (1), without neglecting the spectrum variation in the fuel element, the code assumes that the flux variation in the cell is function of the time averaged composi--tion in the cell itself, by means of the self-shielding fact--ors.

The self-shielding factors are defined as the ratio bet--ween the true reaction rate of the isotope considered and the reaction rate which would be obtained, if the flux in the cell was everywhere equal to a reference flux, say the flux at the cell boundary or the average flux in a given part of the cell.

If the group structure is sufficiently fine, the flux shape in the fuel element can be supposed to depend only on the macroscopic absorption cross-section of the cell in the energy group considered. In this case of spatially heterogene--ous cores few previous transport calculations for the same cell geometry with different compositions will enable to ob--tain a fitting of the self-shielding factors as a function of the cell absorption cross-section. A good fitting is given for example with the formula:

$$SS_{IE,i} = \frac{T_{1IE,i}}{\sqrt{1 + \Sigma_{aIE}\left(T_{2IE,i} + \Sigma_{aIE} \cdot T_{3IE,i}\right)}} \qquad (2)$$

where $SS_{IE,i}$ is the self-shielding factor of the isotope i in the energy group iE

Σ_{aIE} is the total macroscopic absorpt--ion cross-section of the cell in the group iE

$T_{1IE,i}$; $T_{2IE,i}$; $T_{3IE,i}$ are the coefficients supplied into the programme and are different for each group and isotope.

4. THE DIFFUSION CALCULATION METHOD

A diffusion calculation procedure not too time consuming but still accurate is needed, due to the fact that the code has to perform many such calculations in a single run, taking into account the flux currents and the different flux levels in the various zones in which the reactor is subvided.
An analytical nodal approach for the solution of the diffusion equation in one energy group is employed. (Ref. 1) The basic idea is that the real spatial form of the flux within each region is uninteresting for the code purposes; only the average fluxes in each region are actually needed for the calculation, while the neutron currents between adjacent regions shallalso be correctly evaluated. The one energy group flux is the solution of the second order differential equation:

$$\nabla^2 \psi_R + B_R^2 \psi_R = 0 \qquad (3)$$

with

$$B_R^2 = (\nu \Sigma_{fR} - \Sigma a_R)/D_R \qquad (4)$$

The solution of such equation will require the determination of two functions of the boundary coordinates. In order to arrive to a simple solution, the approximation is made that the neutron currents are constant on each boundary of each region and equal to an average value. In this way the true two-dimension solution can be approximated by the superposition of two one-dimension solutions and a total of four coefficients have to be determined (two for each direction).
The two analytical functions which are solution of the equation (3) in each direction depend on the component of the buckling B^2 in either direction (say α^2 and β^2, where $\alpha^2+\beta^2 = B^2$)
If these quantities would be known, the four coefficients present in the flux solution for each zone could be expressed in terms of the average fluxes in the zones.
The solution of the equation (3) would then become an espression of the type:

$$\psi_{ij} Z_{ij} = E_{ij} \psi_{i+1,j} + F_{ij} \psi_{i-1,j} + H_{ij} \psi_{i,j+1} + G_{ij} \psi_{i,j-1} \qquad (5)$$

where ψij are the region average fluxes, i and j are the in--dices of the reactor regions. The programme starts guessing the quantities α^2 and β^2 and then solving the system (5); it ob--tains the average fluxes in each region and the flux curren--ts between adjacent regions. At this point the two componen--ts of the buckling α^2 and β^2 are recalculated and new coeffi--cients for the expressions (5) are obtained. This leads to a new solution for the average fluxes and the currents. When this procedure has converged, the reactivity of the system is calculated. If criticality is not achieved, the source term is adjusted and the calculation is repeated till the fluxes and the currents match with a critical system.
The procedure described assumes the separability of the flux--es within each region. The inaccuracy due to this approxima--tion has been verified to be of no major concern for the e--quilibrium fuel cycle calculations for the fast reactors, although improvements can be searched for in this connection. Finally the diffusion calculation in one energy group has been demonstrated to give quite accurate results for this kind of problems in power fast reactor. Figure 1. gives the flux distribution in the radial direction calculated in one energy group with the nodal approach and in 26 energy group with the finite difference method diffusion code SQUID (Ref. 2). The agreement is satisfactory.

5. SPECTRUM CALCULATION AND BUCKLING VECTORS

The flux spectrum in each of the region in which the reactor is subvided must correspond to a critical assembly. The spec--trum calculation will then be correct once proper values of multigroup bucklings are introduced into the spectrum routine for each region.

The buckling values supplied are the one deduced by the one group diffusion calculation. The direct use of these va--lues to describe the leakages can introduce an error into the spectrum, as the energy dependence of the current is not properly taken into account by a single value buckling. FOr this reason provision is made to introduce into the program--me a set of previously calculated buckling vectors V.
With these quantities the energy group bucklings have to satisfy the equality:

$$\sum_{IE} D_{IE} B^2_{IE} \phi_{IE} \Big/ \sum_{IE} \phi_{IE} = \bar{D}\, \bar{B}^2 \qquad (6)$$

where

$$B^2_{IE} = \bar{B}^2 V_{IE}$$

which imposes that the total leakage has to remain the same. The energy group bucklings obtained from equalities (6) are supplied into the spectrum routine.
When the calculation of the region spectra has converged, the programme uses the multigroup energy fluxes obtained to condense in one group the macroscopic quantities that have to be introduced into the one group diffusion equation (3).

6. THE CALCULATION PROCEDURE

The programme flow diagram is shown in fig. 2 and 3.
First the set of the library data is read in. These data consist in cross-section values, fission yields, the fission source spectrum, the convergence criteria, and the informations needed to generate the isotopes transmutation chains.
Next information are those necessary to describe the reactor. The number of burnable regions, the dimensions of each region, the region compositions and the one group constant for the reflector have to be provided.
The information include the total thermal power of the reactor, the burn-up values for the regions, together with a guess for the axial blanket burn-up, or in turn the burn-up averaged along an axial stripe, and a guess for the feed quantity in the certain region which has been pre-determined for the search.
To begin the calculation a flux distribution is guessed: flat in the radial direction and cosinus shaped in the axial direction. The burn-up values of the axial regions are re-adjusted to give equal residence time in every axial stripe, or in turn region burn-up are calculated from the flux cosinus distribution.

Region spectra are then computed from the fuel compositions given, taking also into account the fuel guess. Residence times are calculated from the one group flux distribution and the burn-up values obtained. Using these data programme estimates the average and final concentrations of each region. Region spectra and K-effective are then calculated using the more recent average composition. Up to this point the flux spatial distribution has not been altered.
Provision is made to recycle any fraction of any nuclide in the same or in another zone. If this is the case new initial concentrations are obtained adding the fraction of the recycled isotopes at the end of the burn-up to the original initial densities.
In any case the depletion and the spectrum calculation are repeated until the multiplication factors of the two consecutive iterations differ by less than the corresponding convergence criterion.

This is the flux spectrum recycle loop and it has been deduced mainly from the zero-dimensional burn-up code Gaffee (Ref. 3).
At this point the code enters into the nodal calculation of the flux spatial distribution. With the new flux levels the residence time values are re-adjusted in each zone. Next, the programme proceeds with the routine which calculates the reactivity at the beginning and at the end of the refueling interval. If the smaller of the two corresponding values of the multiplication factorsis within the specified convergence criterion from the searched value, the code controls the residence time of the axial stripes and, eventually after another burn-up calculation in the case these time values have to be re-adjusted, the computation stops.
If the convergency on the K-effective is not achieved, the programme re-enters into the flux spectrum recycle loop and the calculation is repeated.

7. OUTPUT DESCRIPTION

The programme prints first of all the input data: the library nuclear data, the compositions of the regions and the geometry of the reactor.
The iteration procedure can be easily followed in the output list. When convergence is achieved the initial average and the final isotopic compositions are printed together with spectra, macroscopic multigroup cross-section and neutron balance for each region.
Fraction of the power, power densities, specific power, average reactor conversion ratio and the doubling time of the system are also given.
Finally a number of data are edited and punched on cards, which may be used if a calculation of the fuel cycle cost has tobe performed. These are: initial, average and final, fuel isotopic composition of each region, the region volume (in cm^3) and the fuel residence time (in days), the fuel specific power (in watt/gr of fuel) and the power densities (in watt/$/cm^3$).

8. ACKNOWLEDGEMENTS

Thanks are due to Dr. Rinaldini for his constant support and his valuable suggestions.

9. REFERENCES

Ref. 1 C. Rinaldini - A nodal approach to solve few region neutron diffusion problems.

Energia Nucleare - Vol. 17 N.7 - Luglio 1970.

Ref. 2 SQUID - A multigroup program with criticality searches for the IBM - 360 - EUR 388 2C.

Ref. 3 GAFFEE - A G.G.A. zero dimensional equilibrium burn-up code.

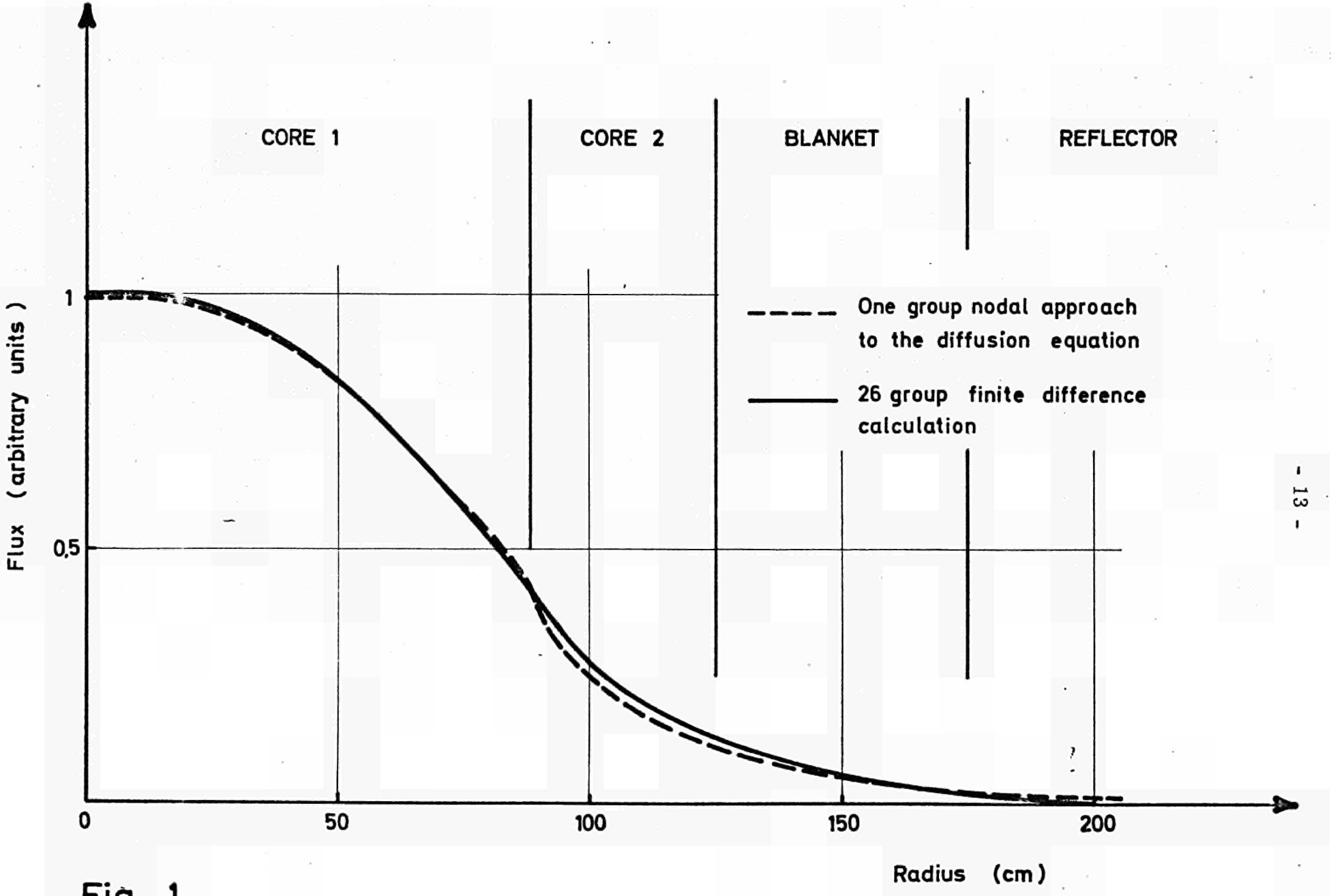

CORE 1
CORE 2
BLANKET
REFLECTOR
One group nodal approach to the diffusion equation
26 group finite difference calculation
Flux (arbitrary units)
1
0,5
0
50
100
150
200
Radius (cm)

Fig. 1

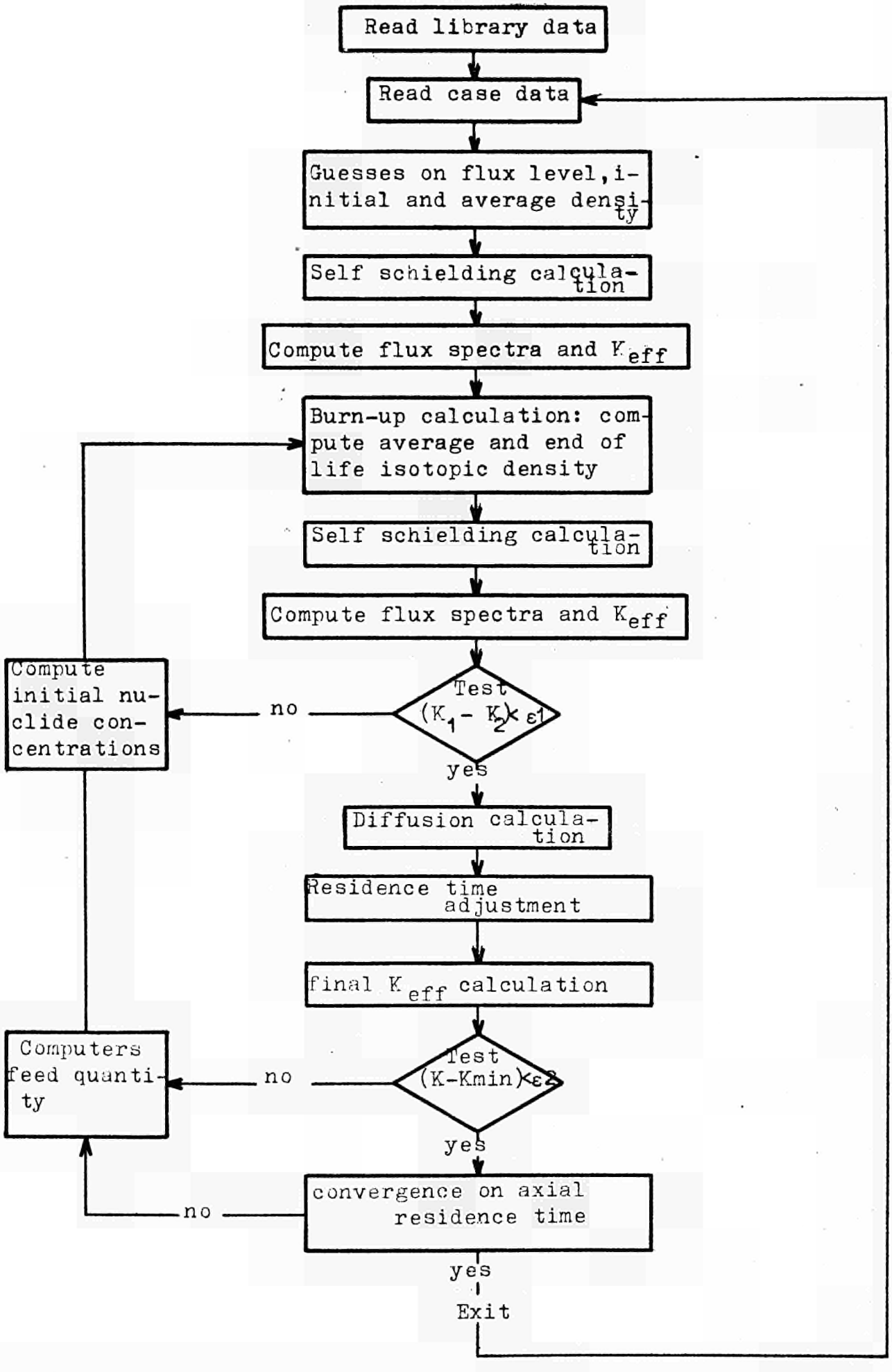


Fig. 2 - Programme flow diagram

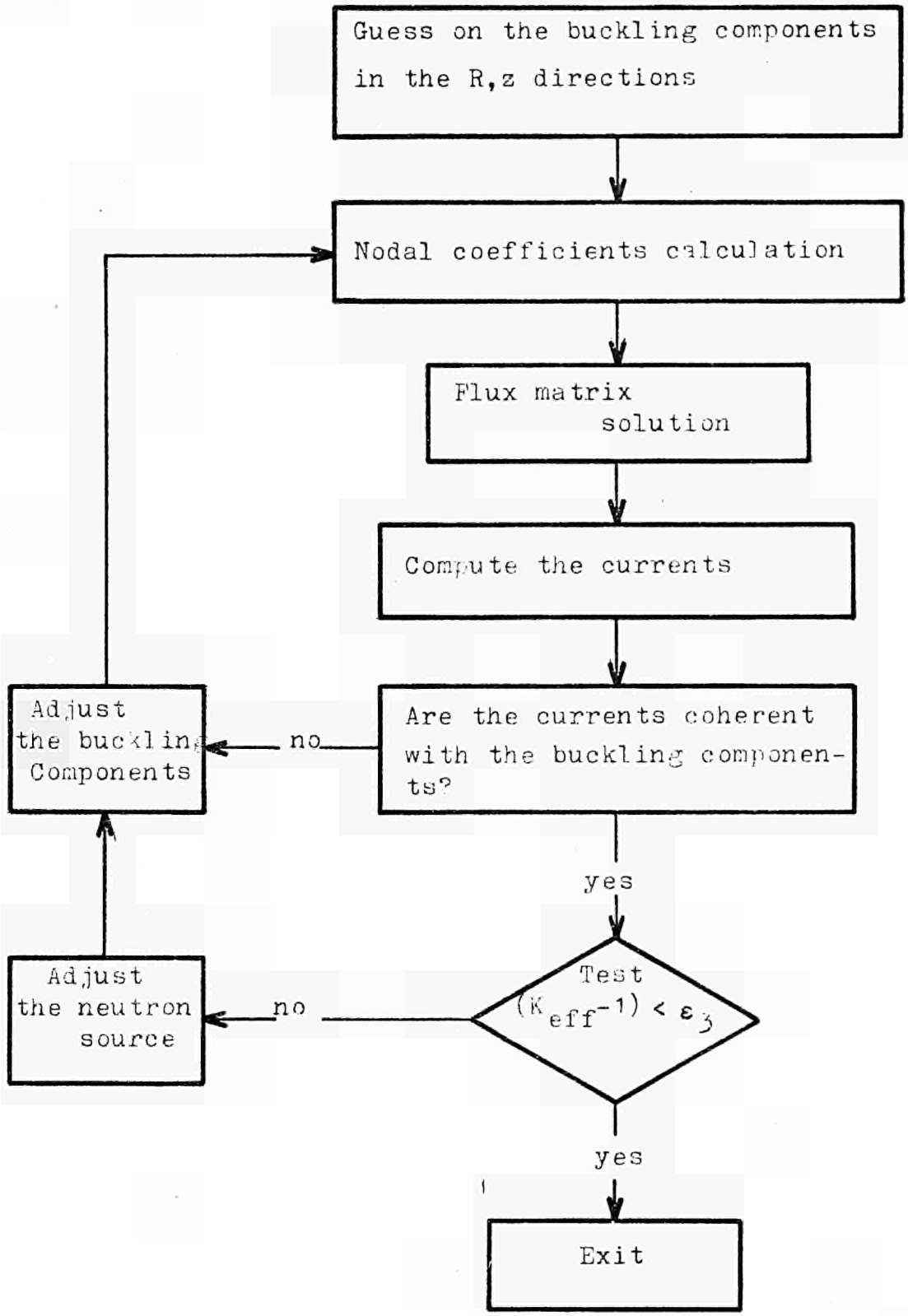


Fig. 3 - Diffusion calculation flow diagram

Word	1	2	3	4	5	6
Column	1-4	5-8	9-12	13-16	17-20	21-24
Format	Integer	Integer	Integer	Integer	Integer	Integer
Card Nº A1	Number of energy groups	Number of fast groups	Number of cross section blocks	Number of heavy nuclides	Number of moderator nuclides	Dummy
	26	24	40		3	
Symbol	N26	N23	NLB	NHEV	NLM	NLT

Word	7	8	9			
Column	25-28	29-32	33-36			
Format	Integer	Integer	Integer			
Card Nº A1 (cont.)	Number of burnup step described in the print out (usually=1)	id. number of the con--trol (should be zero if no Xe-over. calculation is desired)	id. number of the Xe-135 in the library			
Symbol	NCOST	NBORON	NXE5			

Word	1	2	3	4	5	6	Comment
Column	1-4	5-8	9-12	13-16	17-20	21-24	
Format	Integer	Integer	Integer	Integer	Integer	Integer	
Card Nº A2	Nuclide number	Nuclide number of 1st capture parent	Nuclide number of 2nd capture parent	Nuclide number of N, 2N parent	Nuclide number of decay parent	Nuclide has non-zero of? 0 - No 1 - Yes	Supply one card for each nuclide.
Symbol	L	NCAP1 (L)	NCAP2 (L)	NN2NN (L)	NBETA (L)	KFISS (L)	

Word	7	8	9	10	11	12	
Column	25-28	29-32	33-36	37-48	49-60	61-72	
Format	Integer	Integer		Decimal	Decimal	Decimal	
Card Nº A2 (cont.)	Nuclide is a fission product? 0 - No 1 - Yes	Nuclide has non-zero ? n, 2n 0 - No 1 - Yes	Blank	Decay constant	Leakage constant	Atomic weight	
Symbol	KFP (L)	KN2N (L)		XLAM (L)	XLEAK (L)	AWT (L)	

Word	1	2	3	4	5	
Column	1-4	5-8	9-12	13-16	17-20	
Format	Integer	Integer	In-teger	Integer		
Card Nº A3	1st nuclide is primary fissile? 0 - No 1 - Yes	2nd nuclide is primary fissile? 0 - No 1 - Yes	3rd nuclide is primary fissile? 0 - No 1 - Yes	etc.		
Symbol	NFA (1)	NFA (2)	NFA (3)	etc.		

Comment
Supply one word of data for each heavy nuclide. See card A1 word 4. Continue on additional cards if necessary.

Word	1	2	3	4	5	
Column	1-4	5-8	9-12	13-16	17-20	
Format	Integer	Integer	Integer	Integer		
Card Nº A4	1st nuclide is prim.fiss. precursor? 0 - No 1 - Yes -1 Neg contrib	2nd nuclide is prim.fiss. precursor? 0 - No 1 - Yes -1Neg contrib	3rd nuclide is prim.fiss. precursor? 0 - No 1 - Yes 1 Neg contrib	etc.		
Symbol	NCR (1)	NCR (2)	NCR (3)	etc.		

Comment
Supply one word of data for each heavy nuclide. See card A1, word4 Continue on additional cards if necessary.

Word	1	2	3	4	5	
Column	1-12	13-24	25-36	37-48	49-60	
Format	Decimal	Decimal	Decimal	Decimal		
Card A 5	Fiss. Yield from 1st heavy	Fiss. Yield from 2nd heavy	Fiss. Yield from 3rd heavy	etc.		
Symbol	YIELD (I,1)	YIELD (I,2)	YIELD (I,3)	etc.		

Comment
Supply a yield value from each heavy nuclide. See card A1, word 4. Supply a set of yields for each fiss product. Begin each set on a new card. See card A2,word7.

Word	1	2	3	4	5	
Column	1-6	7-60	60-64	65-68	69-72	
Format	Alphanumeric	Alphanumeric	Integer	Integer	Integer	
Card A6	Nuclide Identification	Other cross-block identification	0 - Not a moderator 1 - moderator	Read transfer matrix? 0 - Yes 1 - No	Read self-shielding factors 0 - No 1 - Yes	
Symbol	CLOG	CNAME	NMOD	MATRIX	NSHLD	

Comment
Repeat cards A6, A7, A8, A9, and A10 in sets for each cross section block. See card A1, word 3.

Word	1	2	3	4	5	6	Comment
Column	1-12	13-24	25-36	37-48	49-60	61-72	Supply one card for each energy group.
Format							
Card A7	vc_f	σ_{tr}	σ_a	$\sigma_{g,g+1}$	ν	$\sigma_{n,2n}$	See card A1, word 1.
Symbol	FISIG	TOSIG	ABSIG	OUSIG	XNU	XNSIG	

Word	1	2	3	4	5	6	Comment
Column	1-12	13-24	25-36	37-48	49-60	61-72	Fast group transfer matrix: Start a new card for each group. Continue on another card if necessary. Supply this data for all fast groups except the last.
Format	Decimal	Decimal	Decimal	Decimal	Decimal	Decimal	
Card A 8	$\sigma_{g,g+1}$	$\sigma_{g,g+2}$	$\sigma_{g,g+3}$	etc.	....	g, last fast group	
Symbol	OUSIG						

Word	1	2	3	4	5	6	Comment
Column	1-12	13-24	25-36	37-48	49-60	61-72	Transfer into thermal region for moderators only: Start a new card for each group. Supply this data for all fast and thermal groups.
Format	Decimal	Decimal	Decimal	Decimal	Decimal	Decimal	
Card A9	σ_g, 1st thermal group	σ_g, 2nd thermal group	σ_g, 3rd thermal group	etc.	....	σ_g, last thermal group	
Symbol	OUSIGM					OUSIGM	

Word	1	2	3	4	5	6	Comment
Column	1-12	13-24	25-36	37-48	49-60	61-72	Supply a value for each group, including thermal groups. Continue on additional cards if necessary.
Format	Decimal	Decimal	Decimal	Decimal	Decimal	Decimal	
Card A10	Fission source fraction for 1st group	Fission source fraction for 2nd group	Fission source fraction for 3rd group	etc.	....	Fission source fraction for last group	
Symbol	CHI(1)	CHI(2)	CHI(3)			CHI(N26)	

Word	1	2	3	4		
Column	1-12	13-24	25-36	37-48		
Format	Decimal	Decimal	Decimal	Decimal		
Card A11	Flux convergeance criterion	Flux-recycle loop convergence criterion	Final K_{eff} convergence criterion	Fraction of power reduction for Xe-overr. calculation		
Symbol	CONK	EPS1	EPS2			

Comment
CONK < EPS1 < EPS2 .00001 .0003 .0005 for example

Word	1					
Column	1-4					
Format	Integer					
Card 1 B	Number of self shielding set to be supplied (if zero skip to card 1)					
Symbol	NSET					

Word	1	2	3	4	5	6	Comment
Column	1-4	5-8	7-12	13-16	17-20	21-24	only if NSET > 0 (as many) (as NSET)
Format	Integer	Integer	Integer	Integer	Integer	Integer	
Card 2 B	set number	=0 fitting on concentration formula (1) =1fitting on concentr. formula (2) >1 see next word	=0 fitting on a formula (1) =1 fitting on a formula (2)	id. number of the 1st isotope referred by NØPT=0 or NØPT=1	id. number of the 2nd isotope referred by NØPT=0 or NØPT=1	id. number of the 3rd isotope referred by NØPT=0 or NØPT=1	
Symbol	I	NØPT(I)	NØPI(I)	IS(I,1)	IS(I,2)	IS(I,3)	

Word	1	2	3	4	5	6	Comment
Column	1-12	13-24	25-36	37-48	42-60	61-72	Only if NØPT = 0 or NØPT 1 and NØPI = 0 continue on another card if necessary.
Format	Decimal	Decimal	Decimal	Decimal	Decimal	Decimal	
Card 3 B	First coefficient group (1) formula (1)	Second coefficient group (1) formula (1)	Third coefficient group (1) formula (1)	First coefficient group (2) formula (1)	Second coefficient group (2) formula (1)	Third coefficient group (2) formula (1)	
Symbol	T1(I,1)	T2(I,1)	T3(I,1)	T1(I,2)	T2(I,2)	T3(I,2)	

Word	1	2	3	4	5	6	Comment
Column	1-12	13-24	25-36	37-48	49-60	61-72	Only if NØPT = 1 or NØPT 1 NØPI = 1 continue: 1 card each group
Format	Decimal	Decimal	Decimal	Decimal	Decimal	Decimal	
Card 4 B	First coefficient group(1) formula (2)	Second coefficient group (1) formula (2)	third coefficient group (1) formula (2)	4th _____ _____ _____	5th _____ _____ _____	6th _____ _____ _____	
Symbol	S1 (I,1)	S2 (I,1)	S3 (I,1)	S4 (I,1)	S5 (I,1)	S6 (I,1)	

Word	1	2	3	4	5	6	Comment
Column	1-12	13-24	25-36	37-48	49-60	61-72	Only if NSET < 0 continue on another card if necessary.
Format	Decimal	Decimal	Decimal	Decimal	Decimal	Decimal	
Card 5 B	Constant _____ self shielding - - - - - - set i - - - - - - group 1	_____ - - - - - - - - - - - - group 2	- - - - - - - - - - - - - - - - group 3	- - - - - - - - - - - - - - - - group 4	- - - - - - - - - - - - - - - - group 5	- - - - - - - - - - - - - - - - group 6	
Symbol	SS(i,1)	SS(i,2)					

Word	1	2	3			
Column	1-4	5-6	7-8			
Format	Integer	Integer	Integer			
Card 6 B	Id. number of the self shielding set for isotope 1	Id. number of the self shielding set for isotope 2	etc.			
Symbol	ISET (1)	ISET (2)	ISET (3)			

Comment
Supply as many words as word 3 card A1; continue on another card if necessary

Word	1					
Column	1-72					
Format	Alphanumeric					
Card N° 1	Case identification					
Symbol						

Word	1	2	3	4	5	6
Column	1-4	5-8	9-12	13-16	17-20	21-24
Format	Integer	Integer	Integer	In-teger	Integer	Integer
Card N° 2	Number of nuclide cards to be read (card 6)	0-No effect N-Punch weights for N time intervals	Search option 0-Feed quantity 1-burn-up (MWD/T)	Number of reload intervals during life on one batch	Maximum total iterations	Maximum iterations in a single flux-recycle loop
Symbol	NREAD	NCOST	NTYPE	IRELO	JNSTOP	JNNSTP

Word	7					
Column	25-28					
Format	Integer					
Card N° 2	0-No effect 1-Search for recycle factor if necessary					
Symbol	JNESTP					

Word	8	9	10		18	
Column	29-32	33-36	37-40		69-72	
Format	Integer	Integer			Integer	
Card N° 2 cont.)	0-No effect 1-punch atom densities	Library for next case 0-Same 1-Read AL-A12 1-Read All--A12	0-No effect 1-Poison or fert. mat. search			
Symbol	NPUN	NLIB	NFIXP(1)	....	NFIXP(9)	

Word	1	2	3	4	5	6
Column	1-4	5-8	7-12	13-16	17-20	21-24
Format	Integer	Integer	Integer	Integer	Integer	Integer
Card N° 3	Number of radial burnable zones	Number of axial burnable zones	is the search to be performed in zone 1? 1 yes 0 no	supply as many words as word 1		buckling option if > 0 the one group B will be shared among the groups
Symbol	NZØNE	NZETA	ICHANG(1)	ICHANG(2)	ICHANG....	NVECT

Word	7					
Column	25-28					
Format	Integer					
Card N° 3 (cont.)	has the calculation to be performed with an average axial burn-up? 1 no 0 yes					
Symbol	NBR					

Word	1	2	3	4	5	6
Column	1-12	13-24	25-36	37-48	49-60	61-72
Format	Decimal	Decimal	Decimal	Decimal	Decimal	Decimal
Card N° 4	= 0 axial symmetry = 1 no symmetry inaxial direction	1 group diffusion coefficient of the reflector	macroscopic absorption x-section of the reflector	fission per watt sec	core power (watts)	shut-down K_{eff}
Symbol	HEIGHT	DREFL	SAREFL	FIWATT	PØWER	ZKMIN

Word	1	2	3				Comment
Column	1-12	13-24	25-36				
Format	Decimal	Decimal	Decimal				
Card N° 5	Minimum feed quantity or burn-up value	Maximum feed quantity or burn-up value	IF = 0.0 DELTFD =1.5 (IF x 0) MINIF (x,1.5)				See card2,word3.
Symbol	FMIN	FMAX	DELTFD	FLOATP(1)	FLOATP(2)	FLOATP(3)	

Word	1	2	3			
Column	1-12	13-24	25-36			
Format	Decimal	Decimal	Decimal			
Card N° 6	radius (cm) of the first zone	supply as many words as NZØNE+1	etc.			
Symbol	RZØNE(1)	RZØNE(2)	RZØNE(3)			

Word	1	2	3			
Column	1-12	13-24	25-36			
Format	Decimal	Decimal	Decimal			
Card N° 7	axial thickness of the first zone (cm)	supply as many words as NZETA+1	etc.			
Symbol	ZZ(1)	ZZ(2)	ZZ(3)			

Word	1	2	3			
Column	1-12	13-24	25-36			
Format	Decimal	Decimal	Decimal			
Card	burn-up (MWD/T) of the first radial and axial zone	second radial zone first axial zone	supply as many words as NZØNE times NZETA	etc.		
Symbol	BRUP(1,1)	BRUP(2,1)				

N.B.

The burn-up values of the first axial region are taken as average b.u. in axial direction when NBR=0.

Word	1	2	3			
Column	1-12	13-24	25-36			
Format	Decimal	Decimal	Decimal			
Card N° 9	residence time guess (days) first radial and axial zone	second radial first axial zone	as many words as NZØNE times NZETA			
Symbol	DELDAY(1,1)	DELDAY(2,1)	DELDAY(3,1)			

Word	1	2	3			
Column	1-12	13-24	25-36			
Format	Decimal	Decimal	Decimal			
Card N° 10	feed guess 1st zone		as amny words as NZØNE times NZETA			
Symbol	FEED(1,1)	FEED(2,1)	FEED(3,1)			

Word	1	2	3	4	5	6	Comment
Column	1-12	13-24	25-36	37-48	49-60	61-72	
Format	Decimal	Decimal	Decimal	Decimal	Decimal	Decimal	
Card N° 11	Vector for radial buckling 1st group	Vector for radial buckling 2nd group	....	....	....	Vector for radial buckling last group	Continue on another card if necessary. As many vector sets as NZØNE. Only if word 5 card 3 is not 0.
Symbol	VECTØR(1)	VECTØR(2)				VECTØR(N26)	

Word	1	2	3	4	5	6	Comment
Column	1-4	5-8	9-12	13-24	25-36	37-48	
Format	Integer	Integer	Integer	Decimal	Decimal	Decimal	
Card N° 12	Nuclide identification number	0-No effect N-add recycled part into nuclide N	the recycled nuclide is in the zone NSHU	Recycle fraction	Feed fraction	Constant feed quantity (atoms: barn-cm)	Supply as many nuclide cards as required. See card 2, word 1. (As many sets as the total number of burnable zones)
Symbol	L	NSWIT(L)	NSHU	QR(L)	Z(L)	CDEN(L)	

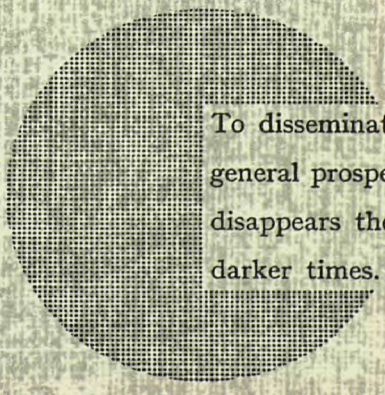
Word	1	2	3	4	5	6	Comment
Column	1-4	5-8	9-12	13-16	17-20	21-24	
Format	Integer	Integer	Integer	Integer	Integer	Integer	
Card N° 13	Number of Th-232 nuclides	Number of fed U-233 nuclides	Number of fed U-235 nuclides	Number of fed Pu-239, Pu-241, and Np-239 nuclides	Number of bred Pa-233 and U-233 nuclides	Number of bred U-235 nuclides	Supply a set of cards 7 and 8 if fuel wts. are to be punched. See card 2, word 2. Supply a second set of cards 7 and 8 if there are two fuel particle types.
Symbol	NW(1)	NW(2)	NW(3)	NW(4)	NW(5)	NW(6)	

Word	7	8	9	10	11	
Column	25-28	29-32	33-36	37-40	41-44	
Format	Integer	Integer	Integer	Integer	Integer	
Card N° 13 (cont.)	Number of bred Np-239, Pu-239, and Pu-241 nuclides	Number of all U-232 nuclides	Number of all Pa-233 and uranium nuclides	Number of all Np-239 and plutonium nuclides	Number of fuel particles 0-1 particle 1-2 particles	
Symbol	NW(7)	NW(8)	NW(9)	NW(10)	NPT	

Word	1	2	3	4		
Column	1-4	5-8	9-12	13-16		
Format	Integer	Integer	Integer	Integer		
Card Nº 14	Ident. no. of 1st nuclide referred to by card 7	Ident. no. of 2nd nuclide referred to by card 7	Ident. no. of 3rd nuclide referred to by card 7	etc.		
Symbol	LIST(1)	LIST(2)	LIST(3)	etc.		

Comment
List the identification number of each nuclide referred to by card 7, words 1 to 10. Continue on another card if necessary. (As many sets as the total number of burnable zones)

Word						
Column						
Format						
Card						
Symbol						



To disseminate knowledge is to disseminate prosperity — I mean general prosperity and not individual riches — and with prosperity disappears the greater part of the evil which is our heritage from darker times.

Alfred Nobel

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