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RINA
A COMPUTER CODE FOR THE RUNNING IN EVALUATION

by

G. GRAZIANI and M. PARUCCINI

1973

Joint Nuclear Research Centre
Ispra Establishment - Italy

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ABSTRACT

This report describes a 1-dimensional computer programme, which calculates the approach to equilibrium cycle for thermal reactors, using a partial refueling scheme.

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KEYWORDS

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FUEL CYCLE
EQUILIBRIUM
R—CODES
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1) Introduction

For much of the survey work on reactor fuel cycles, only the equilibrium fuel cycle is usually considered, as it represents the larger part of the reactor life. The contribution of the approach to equilibrium period is usually taken into account, in the total fuel cycle cost calculation, on the basis of the value of the fuel and fabrication inventory, which is deduced by the average composition of the equilibrium fuel cycle itself (Ref.1).

This approximation can introduce some errors due not only to the way in which the inventory is calculated, but also to the fact that, in this way, the fuel cycle cost is independent by the strategy followed in the approach to equilibrium phase.

These errors can be relatively more important when a comparison is carried out between different types of fuel cycles for a given reactor, where the cost differences result to be relatively small.

To avoid these difficulties the 1 dimensional code RINA has been written. In this code the calculation starts with the initial charge in the core, determining the flux distribution and the nuclear depletion up to the time when the first reloading occurs. At that time, a fraction of burned fuel is replaced with the fresh one.

The calculation of the flux distribution and the depletion is repeated till the next refueling time. If this procedure is continued long enough, the feed fuel requirements and the composition of the fuel discharged will not change any more with time and the behaviour of the nuclear parameters will repeat at each cycle. The equilibrium cycle is reached.

In this way the code RINA gives all the informations on the running-in period, as well as on the equilibrium cycle itself. In the following, a detailed description of the programme flow diagramme with the different calculation options is given, together with the description of the methods to solve the burn-up equations, the diffusion equation, the fuel cell self-shielding factors, the multigroup spectrum, and the burnable poison equations.

2) The calculation procedure

The programme flow diagram is shown in fig.1.
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. Up to 26 energy group and 45 different isotopes can be used.

Next information is that necessary to describe the reactor (case data). The information include the total thermal power of the reactor, the number of diffusion zones (=1 in the case of zero-dimensional calculation) the axial buckling, the height of the core, the radia of the diffusion zones and of the reflector, the one group neutron constants for the reflector.

In each of the diffusion zones different burn-up regions can be considered, which have, in general, different volumes and compositions. For a given refueling interval, the programme will search for a single value of the initial fissile concentration in one or more zones, necessary to obtain an imposed reactivity at the end of the interval, or searches for the refueling time when the initial composition is fixed for each nuclide in each region. A third searching option is also included. The code can search both for the concentration and for the refueling time, when the initial and final reactivity values are fixed.

In the first refueling interval, the code allows the possibility of introducing burnable poison. In this case the radius and the initial composition of the burnable poison pin have to be given in the input. The total number of pins is a result of the calculation if the poisoned Δ Keff between beginning and end of the first charge is imposed. The possibility also exists to give in input the number of burnable poison pins in the whole core.

The re-utilization of part of the fissile material discharged can be taken into account, both in the case in which the fissile is immediately available, as when the reactor considered is in an expanding system of similar reactors and in the case in which a certain delay time has to be applied due to the cooling, reprocessing and re-fabrication times. In this last case, the code compares the delay time with the time interval between the two discharges in question, in order to decide if, at that time, the recycling is feasable or not.

The code starts guessing, for example, the initial fissile composition. It calculates, then, the isotope self-shielding factors and the average zone spectra. An initial radial flux distribution is assumed as a guess. Using this flux, the programme normalizes the spectra to the total reactor power, and performs the depletion calculation, affecting the library

cross-sections with the self-shielding factors previously calculated. The average concentrations obtained with the burn-up routine are used to obtain new zone spectra and the data for the one-group diffusion calculation.

New self-shielding factors are also obtained as function of the average macroscopic obsorption cross-section.
Again the spectra are normalized to the reactor power and a new depletion calculation starts.

Where the difference in average composition reactivities of two successive iterations is below the convergency criterium fixed in the input, the code calculates the initial and the average K_{EFF} in the refueling interval considered. If the refueling period is the first and if burnable poisons have to be used, the code gives also the poisoned initial and final reactivity, and the number of burnable poison pins.

If the final reactivity is different from the imposed value the programme changes consequently the guess of the initial fissile composition and a new external iteration is performed. Where the external convergency is achieved the code discharges the indicated regions (or a fraction of them) and replaces them with fresh fuel.

A new calculation for the next refueling time then begins.

3) The depletion 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:

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

where NUCL is the total number of burnable elements. The coefficients Aij of the system equations form a triangular matrix and represent the transmutation rate (by decay or capture) of nuclide j into nuclide i. The diagonal elements A_{ii} are the total removal rate of isotope i out of the system.

The code assumes that all these reaction rates are time independent. 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 increases, i.e. when the refueling interval becomes shorter. 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. In order to make the coefficients A_{ij} to be constant, the fission product source in the reactor is assumed to be proportional to the average fuel composition of the reactor.

This is equivalent to approximate, in each region of the reactor, the refueling scheme to the continuous reload scheme and therefore to assume that the space average concentrations in the zone are equivalent to the time averaged concentrations. This is in most of the cases quite satisfactory.

4) 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 variation of the neutron flux spectrum within the cell is of great importance for the calculation of the isotopes burn-up. Therefore a multigroup calculation appears to be necessary. However, in oder 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 composition in the cell itself, by means of the self-shielding factors.

The self-shielding factors are defined as the ratio between 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.
Few previous transport calculations for the same cell geometry with different compositions will enable to obtain 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 absorption 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.

Two different possibilities of using these self-shielding factors are envisaged in the programme. The first one assumes that the average macroscopic absorption cross-section is the average over the diffusion region, not taking into account the fact that different elements with different burn-ups are present in each region.

In the second one, which is more accurate but also more computer-time consuming, the code considers as many macroscopic absorption cross-section as the number of burn-up regions in each of the diffusion zone, giving in such a way the correspondent self-shielding factors to elements with different irradiations.

5) The diffusion calculation

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 subdivided.

An analytical nodal approach for the solution of the diffusion equation in one energy group is employed. (Ref.2) 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 shall also be correctly evaluated. The one energy group flux is the solution of the second order differential equation:

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

with

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

The solution of equation (3) with the nodal method leads to the solution of equations of the type

$$\psi_i A_i = \psi_{i+1} X_i + \psi_{i-1} Y_i \qquad (5)$$

i being the index of the zone; A,X,Y being function of the zone characteristics.
The routine starts solving the matrix (5) and calculating the reactivity of the system. If criticality is not achieved, the source term is adjusted and calculation is repeated till the fluxes and the currents match with a critical system.

6) Spectrum calculation and buckling vectors

The spectrum employed in the burn-up calculation is a multigroup spectrum averaged over the refueling interval and over the burn-up regions belonging to the same diffusion zone. This assumption corresponds to a "scatter load" refueling and

it is a good approximation when the spectrum does not change rapidly within the burn-up interval, as it occurs (apart for the first charge) when the core fraction to be reloaded is sufficiently small.
The flux spectrum in each of the diffusion zone in which the reactor is subdivided must correspond to a critical assembly. For this reason, the spectrum calculation will be repeated, iterating on the buckling values.

The buckling values supplied are the one deduced by the one group diffusion calculation. The direct use of these values to describe the leakages can introduce a distortion 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 programme a set of previously calculated buckling vectors V.
With these quantities, the energy group bucklings have to satisfy the equality:

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

where

$$B^2_{IE} = \bar{B}^2 \; V_{IE} \qquad \text{(IE group index)}$$

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.

The solution of the spectrum matrix equation

$$\varphi_{IE} = (\chi_{IE} + {}_{IJ}\text{Sum} \; \Sigma_{IJ \to IE} \; \varphi_{IJ}) \; / \; (\Sigma_{a_{IE}} + \text{Sum}_{IK} \; \Sigma_{IE \to IK} + D_{IE} \bar{B}^2) \qquad (7)$$

is obtained by iteration.
In the zero dimensional scheme, the possibility of introducing an extra absorber into the core is envisaged in order to match the criticality requirements.

In the one dimensional scheme, 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). The final matching of the criticality requirements is made at this level, when solving for the spatial distribution.

7) The burnable poison routine

The calculation of the burnable poison depletion equations is based on two main assumptions.
The first is to consider an infinite solid cylinder in a flat flux distribution. The incoming neutrons are monoenergetic and enter uniformly over the entire surface of the absorber, every collision within the cylinder resulting in an absorption.
As a result, an analytical approximated formula for the pin self-shielding factor can be obtained.
The second assumption is to separate the burnable poison pin absorption into a surface contribution and into a volume contribution. The surface term has the effect of burning completely a thin layer of thickness dR in the time interval dt, whilst the volume term takes into account the depletion of the burnable poison concentration inside the pin.

A better description of the assumptions and of the calculation method is reported in the Appendix, together with a comparison between the results of the method with those of a more accurate calculation.

For a given geometry and initial density of the burnable poison rods, the programme calculates the number of pins necessary to reduce the reactivity difference between fresh and spent fuel in the first reloading interval to the desired value. In case of more than one diffusion zone, the numbers of pins are taken proportional to the zone volumes (i.e. no power flattening can be obtained with burnable poisons).
The resulting average absorption due to the presence of the absorbers is included in the calculations of the zone spectra.

8) The economic calculation

For practical reasons the economic calculation of the fuel cycle worked out with RINA has not been physically included in the main programme. However the natural way of using the results of RINA is to evaluate them in terms of cost, in order to look at the different influence of the running in period over the total fuel cycle cost and in order to check the validity of the assumption employed in the equilibrium fuel cycles.

The key to understand the nuclear fuel cycle cost evaluation is the investment-time diagram (figure 2).

In the figure a single refueling interval is reported.
The initial investments in fissile material and fabrication

are represented by the steps on the left side of the figure. The straight line with negative slope indicates that the reactor is producing energy, and then is producing return money.
The last positive step represents the expenditures due to reprocessing and shipping the spent fuel, and the last step going down to zero indicates that some of the reprocessed material is eventually sold.

In order to actualize all the investments to the same time, the energy was supposed to be produced at the time T_1 (at half a way of the refueling interval).
The fuel cycle cost is split into the following items:
fresh fuel consumption: a certain rate of the make-up fuel is introduced into the reactor.
The unit cost of the fresh fuel is given in the input, or it is calculated on the basis of the enrichment, according to the use of ideal Cascade theory (Ref.5). The delay time between the purchase of the fuel and the beginning of the irradiation has also to be given.
Revenue: The spent fuel is eventually sold after reprocessing. The input cost of the spent fuel and the delay time due to cooling and reprocessing have to be given.
Fabrication: on the basis of a fabrication cost per unit fuel weight and of a fabrication delay the contribution to the fuel cycle cost, inversely proportional to the fuel burn-up, can be calculated.
Reprocessing: the same as for the fabrication.

In order to calculate the average total fuel cycle cost, the calculation has to be performed along the total reactor life. All the expenditures and revenues have to be actualized at the same time at the numerator of the fuel cycle cost formula, as well for all the energy produced (at the denominator).

The difference, between the total fuel cycle cost and the fuel cycle cost for the equilibrium refueling interval, gives the contribution of the running in period to the fuel cycle cost and can be directly compared with the approximate evaluation resulting from the usual equilibrium economic calculations.

9) Output description and computer time

The programme, first of all, prints the input data. The library nuclear data, the geometry of the reactor and the compositions of the regions.

The iteration procedure can be easily followed in the output list. At each external iteration, the number of burnable fission pins, K_{eff} initial and final with and without burnable poisons are printed.

For each refueling interval, when convergence is achieved, the initial and final isotopic compositions are printed, together with zone spectra and neutron balance for each region. The power fraction of each zone, age factors (*), average region conversion ratios (**) 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 to be performed. These are, for each interval, the weight (grams) of heavy isotopes charged into the core and discharged, the zone power density (w/cc) times the maximum age factor in the zone and the refueling time (days).

Computer time depends very much on the complexity of the problem. It is function of the number of cycles, the number of zones and burnable regions, and depends also on the different options used for the criticality search, for the self-shielding calculation and for the burnable poison calculation.

As an example, for a case consisting of 12 refueling intervals with 20 energy groups, 45 isotopes and 16 burnable region, in one-dimensional approach, a computer time of about 3 minutes can be expected for the IBM 360/70.

(*) (defined as the ratio between initial and average fissions)
(**)(which is the absorptions in fertile materials over absorptions in fissile)

Appendix

The burnable poison absorption as a function of the irradiation

A method to calculate the self absorption of monoenergetic neutrons in an infinite cylinder was developed by W.J.C. Bartels (Ref.2). He derives an analytical expression of the self-shielding factor as a function of the parameter

$$\chi = R\Sigma$$

- R being the cylinder radius
- Σ being the macroscopic total cross-section of the cylindrical medium

The approximate solution of the integral derived by Bartels leads to two analytical expressions, one for small and the other for large values of x

Small x $$f_1(x)=1-\frac{4}{3}x + \sum_{n=0}^{\infty} a_n x^{2n+2} (\ln 2 - \gamma + a_n - \ln x) \qquad (8)$$

with $$a_n = \frac{(2n)!}{2^{2n}(n+2)\,(n+1)!\,n!^2}$$

Large x $$f_2(x)= \frac{1}{2x}\left(1- \frac{3}{16x^2}\left(1+\frac{5}{8x^2}\left(1+ \frac{105}{32x^2}\left(1+.....\right)\right)\right)\right) \qquad (9)$$

The behaviour of these two expressions is reported in figure 2. The figure shows that the two formulae give the same results for x being the interval (2.5,4). A value of x in this range must then be chosen to pass from one expression to the other in the calculation of the self-shielding factor. A break even point of 3.3 has been used in the following.

Employing the expressions (8) and (9), it is possible to calculate with a certain approximation the flux depression in each energy group in the burnable poison pin, and then to calculate the total reaction rate at the initial time.
To study the evolution of the total reaction rate with the irradiation, the following assumptions have been made

- The spectrum in the medium surrounding the burnable poison element is kept constant in time
- The absorption in the pin can be separated into two contributions: a volume and a surface term

If the volume absorption rate is defined as

$$\sum_c \phi_c = \sum_{IE} \phi_{0,IE} f_{c,IE} \Sigma_{a,IE}$$

(IE. being the group index)

and ϕ_c is the total flux at the center of the pin

ϕ_0 is the incoming flux

f is the total self shielding factor averaged over the groups,

we can define the parameter ξ as the excess of the absorption rate due to the presence of the surface absorption relative to the real average absorption rate.

$$\xi = \frac{\phi_0 f \bar{\Sigma}_a - \phi_c \Sigma_c}{\phi_0 f \bar{\Sigma}_a} = 1 - \frac{\phi_c \Sigma_c}{\phi_0 f \bar{\Sigma}_a} \qquad (10)$$

with

$$\phi_0 f \bar{\Sigma}_a = \sum_{IE} \phi_{0,IE} \bar{f}_{IE} \Sigma_{a,IE}$$

We can now assume that the excess absorption due to the surface term has the result of burning completely a thin layer of thickness dR in the time interval dt.

The fraction of neutrons which contribute to this radius depletion will be ξ , $(1-\xi)$ being the neutrons which enter the rod burning uniformely the burnable poison inside. In this case the two depletion equations for radius and burnable poison density can be written.

$$N\, 2\pi R\, dR = -\xi \left(\phi_0 f \bar{\Sigma}_a\right) \pi R^2\, dt \qquad (11)$$

$$dN = -(1-\xi)\, \sigma_a \phi_0 \bar{f} N\, dt \qquad (12)$$

The parameter ξ is time-dependent. In order to evaluate its behaviour with the irradiation, a result, obtained with the code MINOTAUR (Ref.4) and shown in the following table, can be used.

burn-up (MWd/Kg)	$(\Sigma_c \phi_c)/(\Sigma_c \phi_c)t=o$	total absorption
0	1	1
4	1.035	0.84
13	1.08	0.595
17	1.22	0.406

From the table, it can be deduced that the quantity

$$N_c(t) \underset{iE}{Sum} \phi_{c,iE} \sigma_{a,iE} = \phi_c \Sigma_c$$

is roughly constant. Employing the definition of ξ (10) the expression

$$1 - \xi = \text{const} / \phi_o \overline{f \Sigma_a}$$

is obtained for the dependence of ξ on the irradiation.

In order to check the validity of our assumptions a certain number of calculations has been carried out using the code MINOTAUR (Ref.4) in which a burn-up routine has been added in order to perform the depletion of the burnable isotopes present in the pin.

The cases studied involved different rod radia from 0.07 to 0.28 cm.. The pin composition was chosen to be natural Gadolinium oxide with a density of 7.4 gr/cc. The incoming spectrum was a typical HTR spectrum in 15 energy groups. The mesh partition has been chosen sufficiently fine at pin edge, at the beginning, and has been changed at about half of the irradiation considered.

The initial values of ξ for different radii have been calculated with MINOTAUR and are reported in figure 4.
The validity of the assumption of separating the total absorption in two terms, the volume contribution and the surface contribution, is justified by the resulting behaviour of the absorption rate with the pin radius as a function of the irradiation.

The comparison of the results of MINOTAUR with the approximate anlytical method are shown in fig. 5,6,7,8 where the total reaction rate taking place in the pin is reported versus burn-up for the different pin radii considered.

The small discrepancy at low irradiation values are due to the approximate calculation of the self-shielding factor. The slope of the curves is, however, almost the same in all the cases and justify the assumption made on the dependence of ξ on the irradiation.

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5) A.J.Snyder - Methods of Calculating U-235 Outputs and changes by use of Ideal Cascade Theory - T.I.D./8522, Feb. 1960

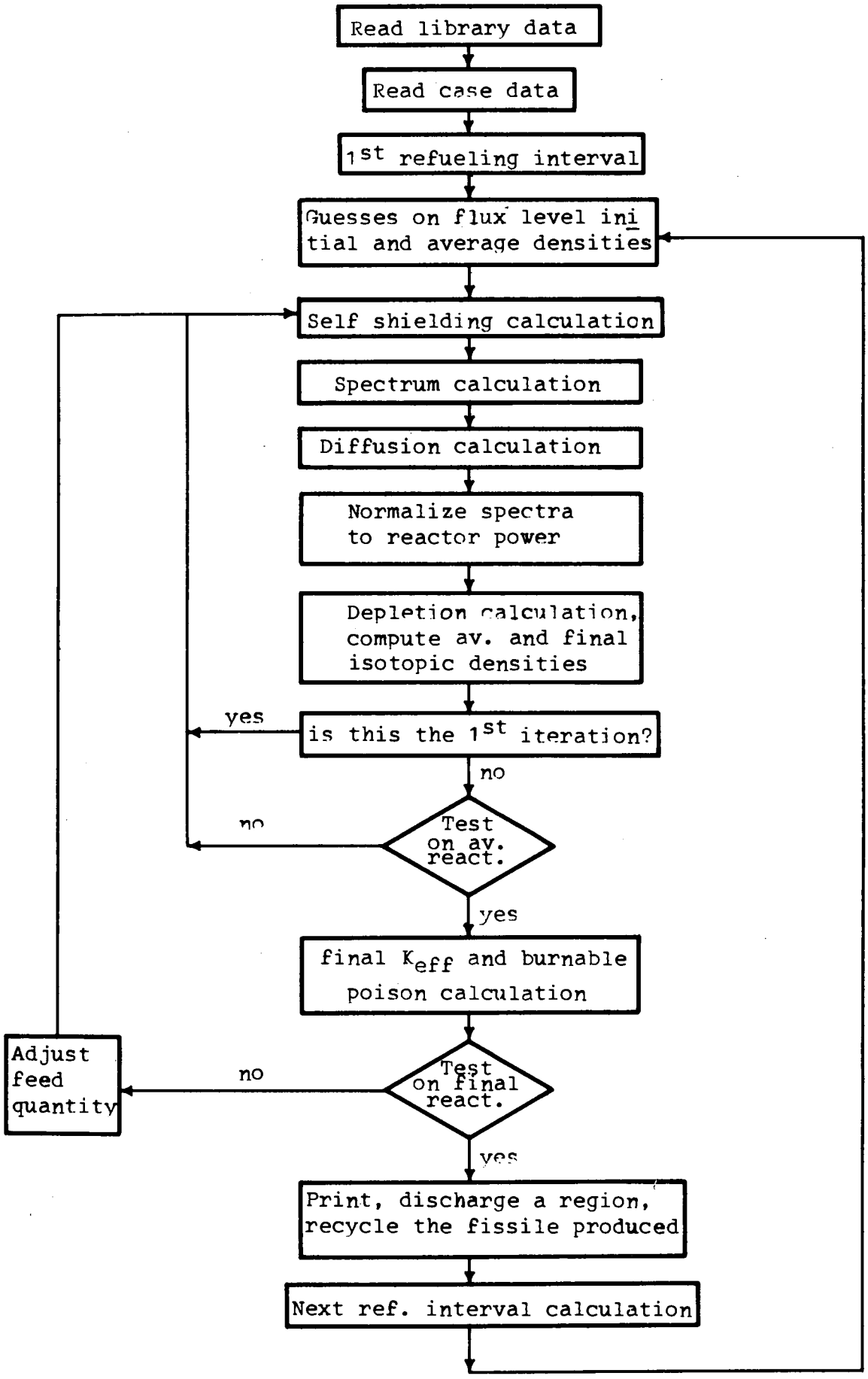

Fig. 1 - FLOW DIAGRAM

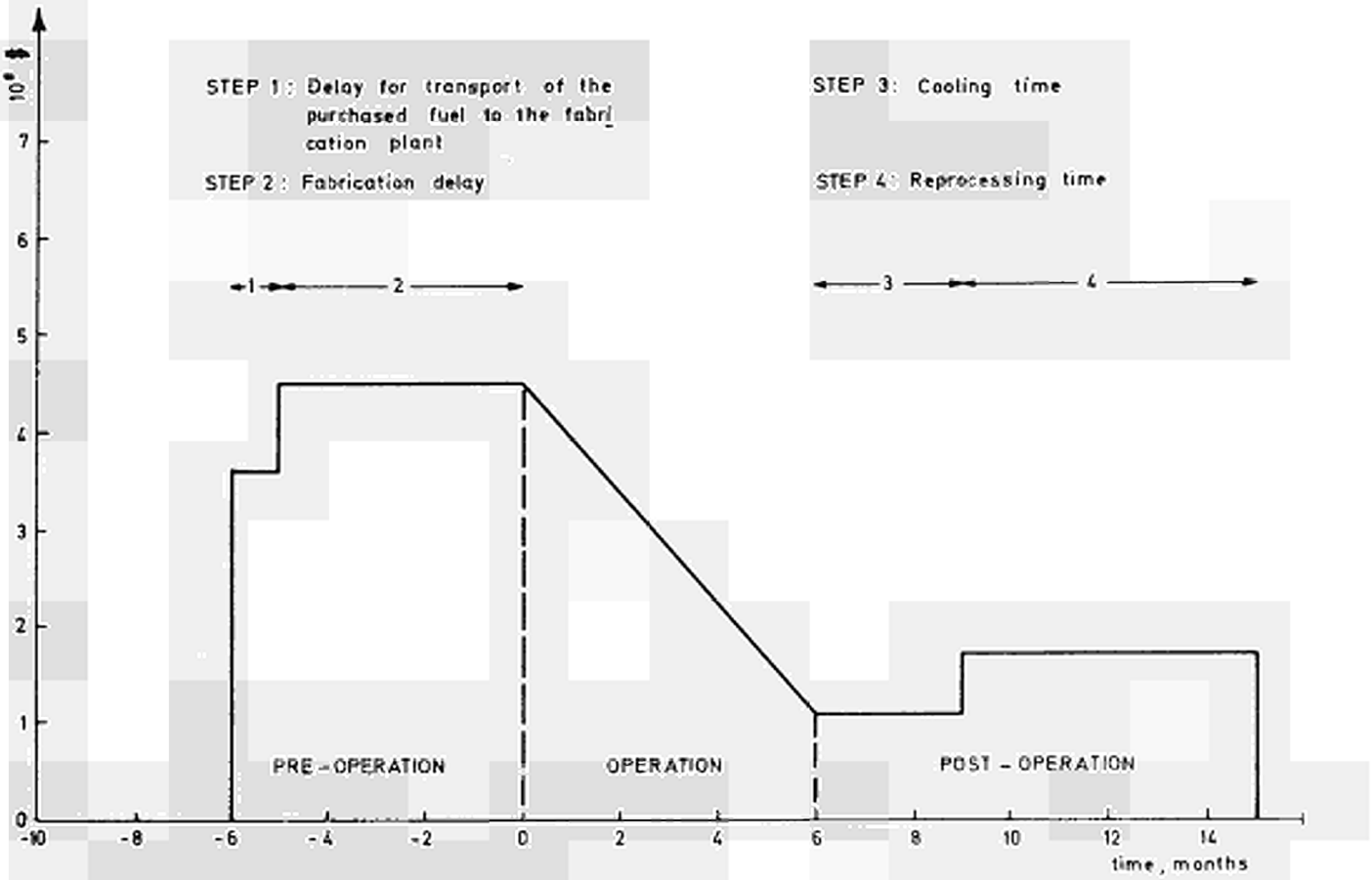

Fig. 2 - THE INVESTIMENT - TIME DIAGRAM

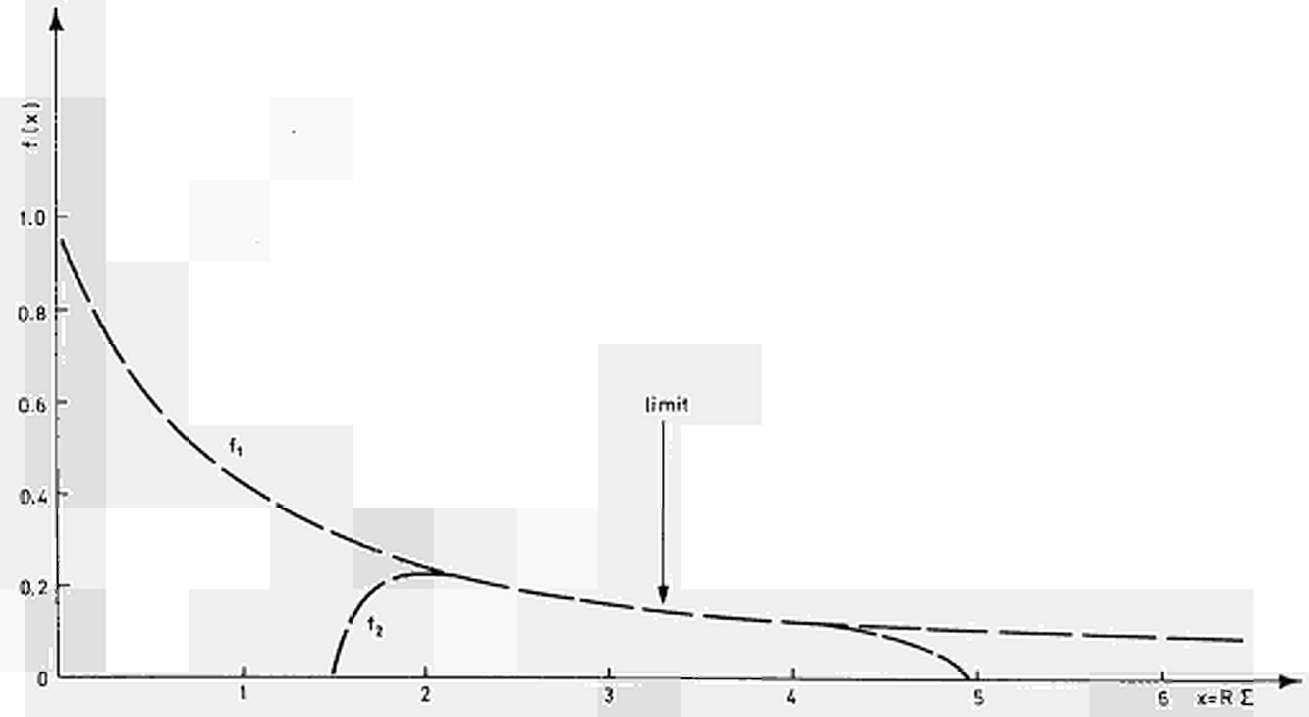

Fig. 3 - BURNABLE POISON SELF - SHIELDING

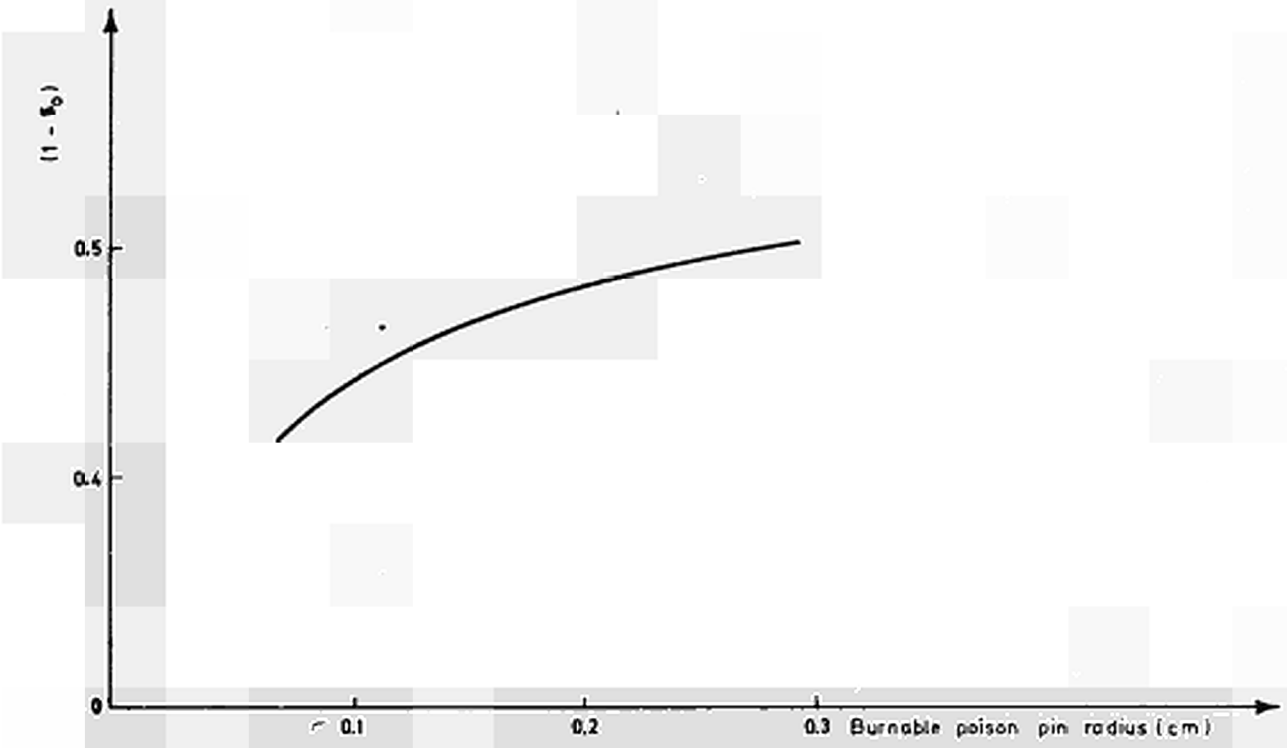

Fig. 4 - VARIATION OF THE INITIAL VALUE OF ε WITH THE PIN RADIUS

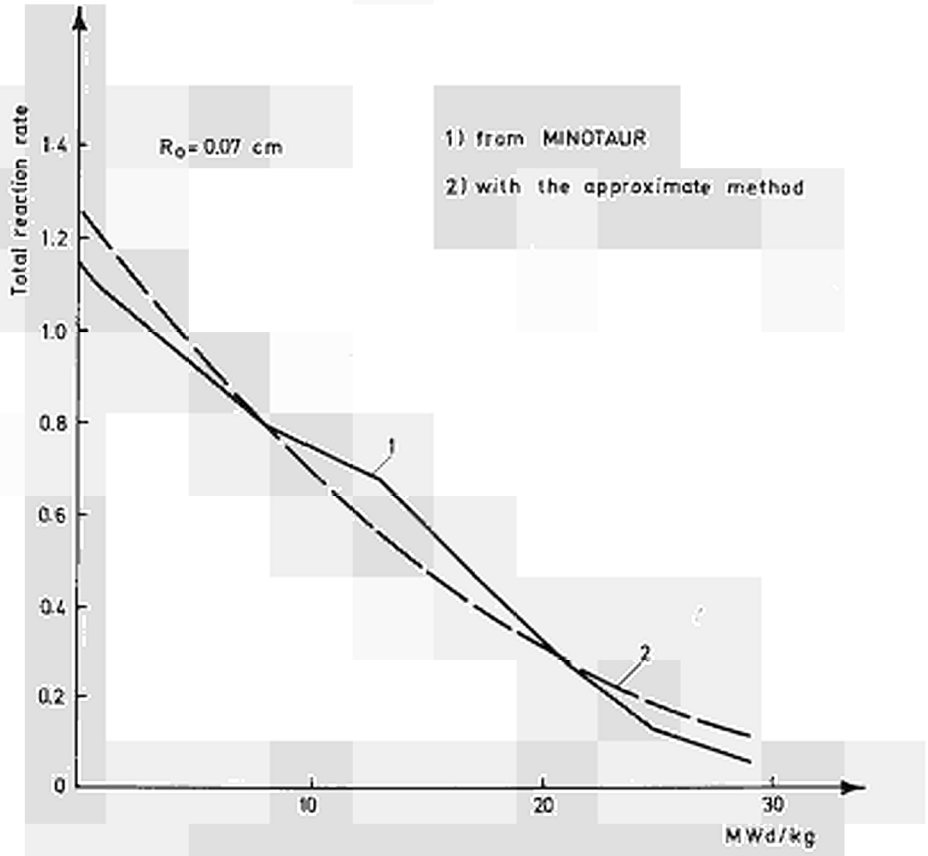

Fig. 5 - BURNABLE POISON REACTION RATE AS A FUNCTION OF THE IRRADIATION

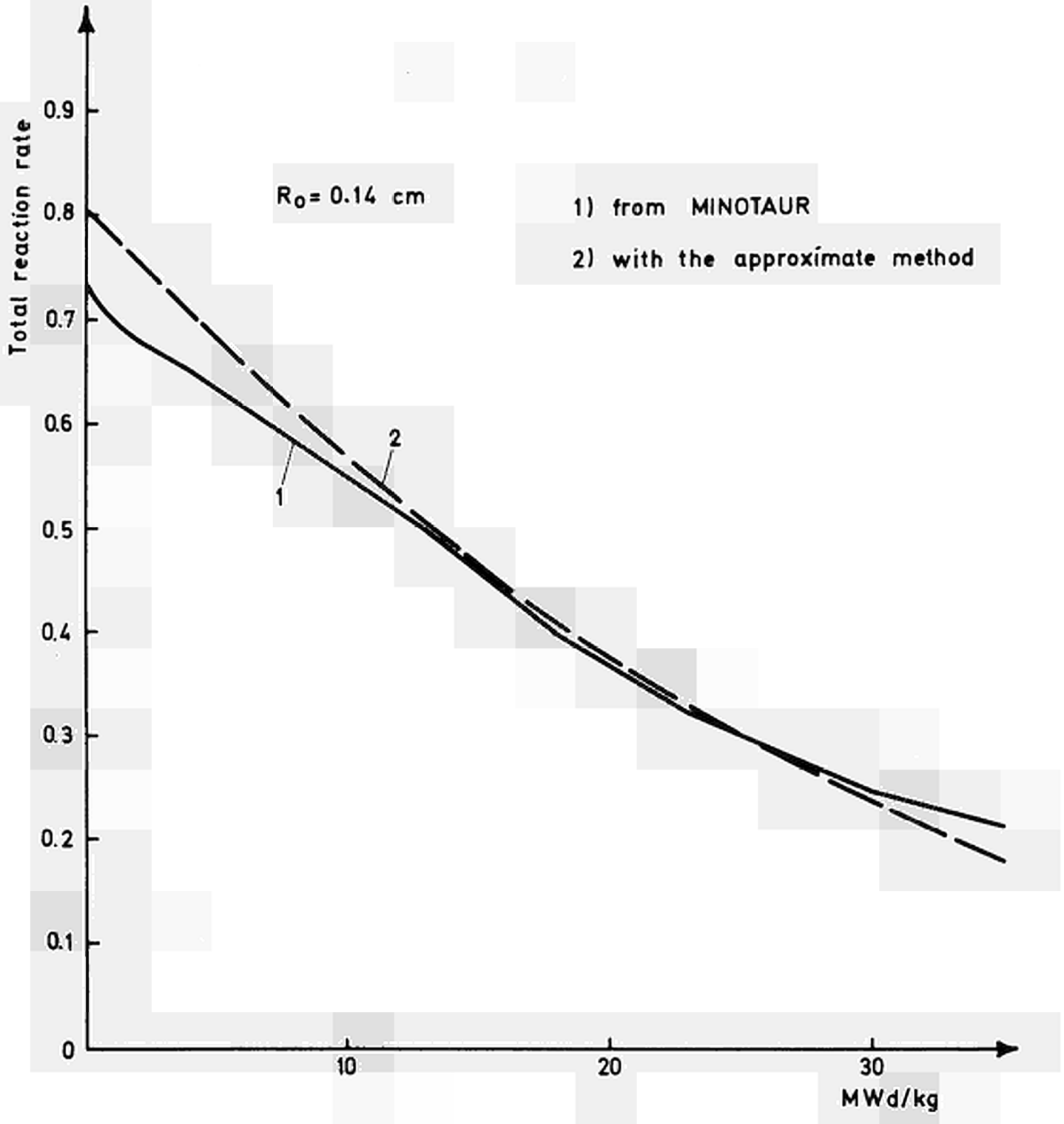

Fig. 6 - BURNABLE POISON REACTION RATE AS A FUNCTION OF THE IRRADIATION

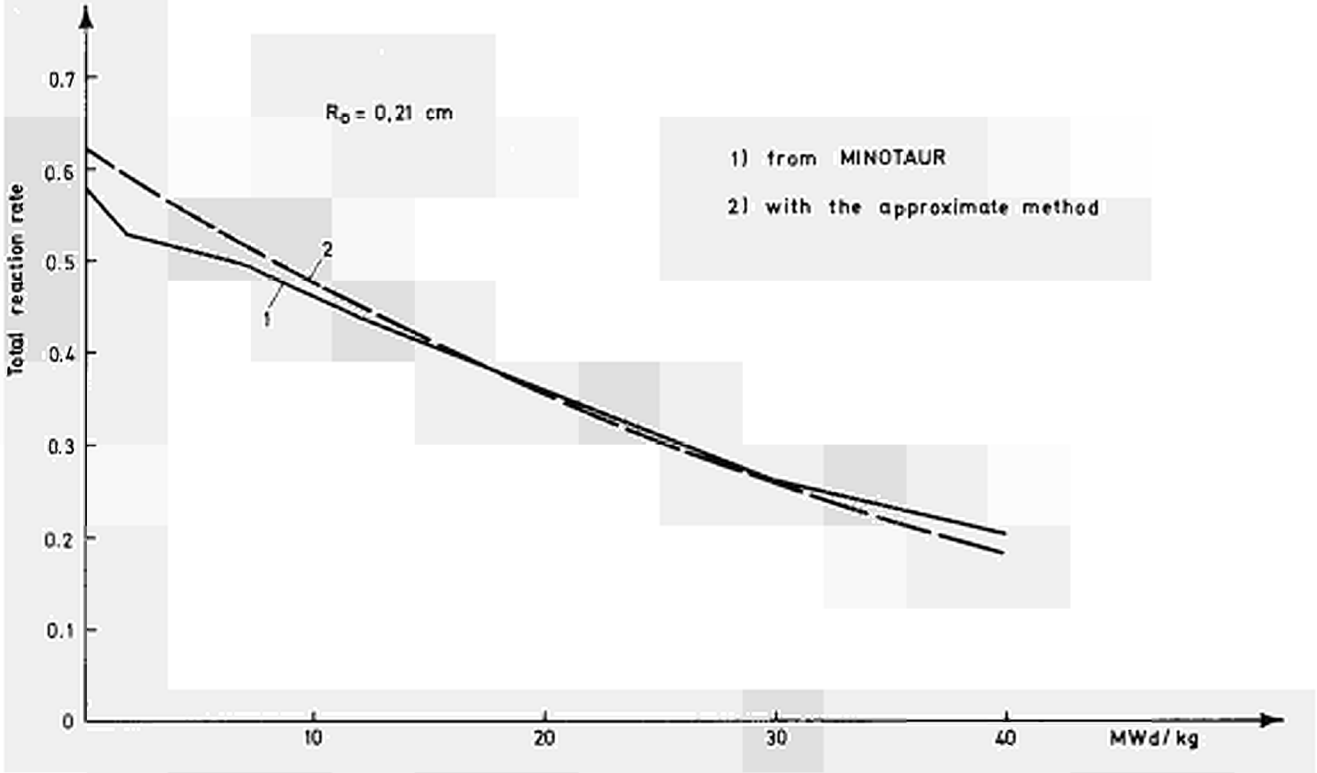

Fig. 7 - BURNABLE POISON REACTION RATE AS A FUNCTION OF THE IRRADIATION

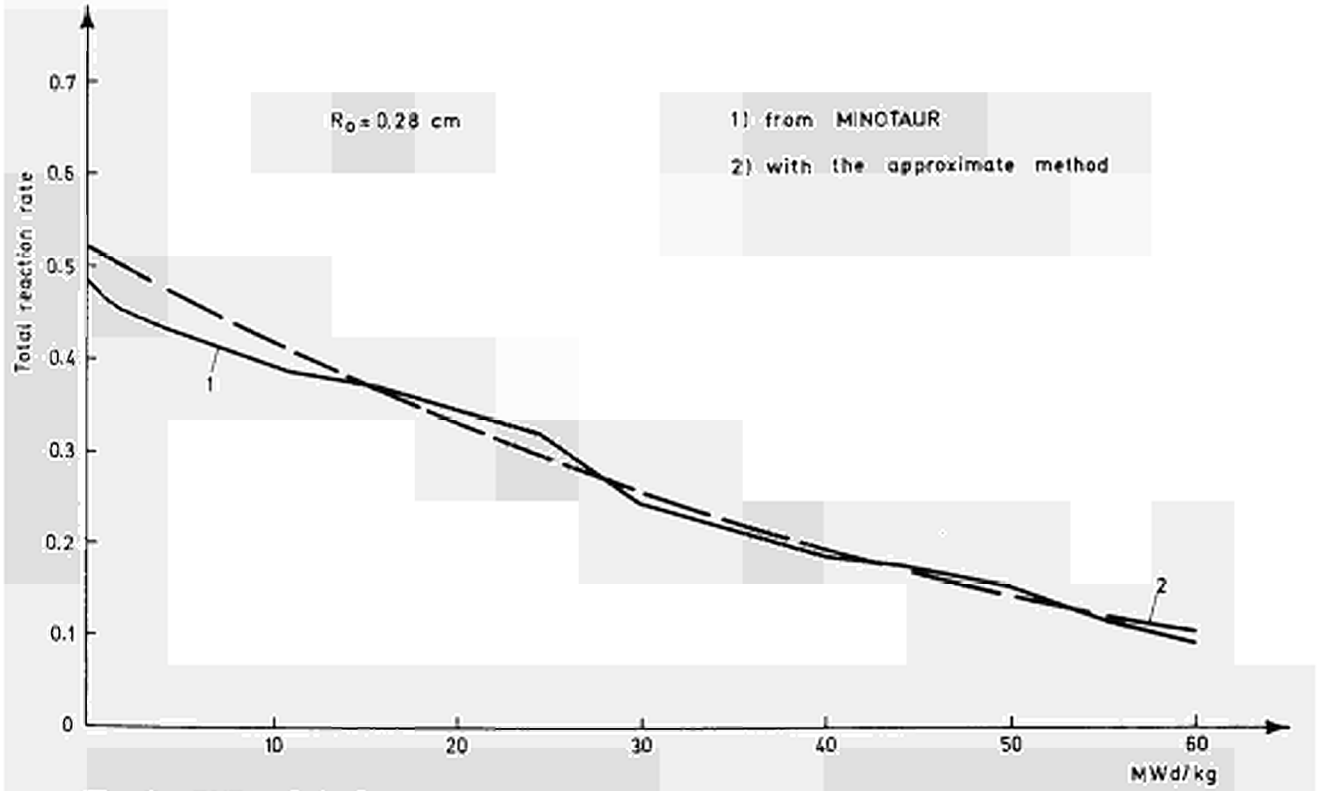

Fig. 8 - BURNABLE POISON REACTION RATE AS A FUNCTION OF THE IRRADIATION

LIBRARY DATA

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 N° A1	Number of energy groups (less or equal 26)	Number of fast groups (less or equal 20)	Number of cross section blocks (less or equal 45)	Number of heavy nuclides	Number of moderator nuclides (less or equal 3)	Dummy
Symbol	N26	N23	NLB	NHEV	NLM	NLT

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° A1 (cont.)	If larger than 0 prints neutron balance and spectrum in each diffusion region	id.number of the control (should be zero if no Xe-override calculation is desired)	id.number of the Xe-135 in the library	id.number of the unit from which the code reads the library data	the id. number in the library of the spectrum calculation control poison. (if greater than 0)	
Symbol	NCOST	NBORON	NXE5	NI	NPOIS	

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°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 σ_f ? 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 N° A2 (cont.)	Nuclide is a fission product? 0 - No 1 - Yes	Nuclide has non-zero $\sigma_{n,2n}$? 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	Integer	Integer		
Card N° 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 N° A4	1st nuclide is prim. fiss. precursor? 0 - No 1 - Yes -1 Neg contrib	2nd nuclide is prim. fiss. precursor? 0 - No 1 - Yes -1 Neg 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, word 4.

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 N° A5	Fiss.Yield from 1st heavy nuclide	Fiss. Yield from 2nd heavy nuclide	Fiss.Yield from 3rd heavy nuclide	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, word 7

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

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
Column	1-12	13-24	25-36	37-48	49-60	61-72
Format						
Card N° A7	$\nu\sigma_f$	σ_{tr}	σ_a	$\sigma_{g,g+1}$	ν	$\sigma_{n,2n}$
Symbol	FISIG	TOSIG	ABSIG	OUSIG	XNU	XNSIG

Comment

Supply one card for each energy group.

See card A1, word1.

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° A8	$\sigma_{g,g+1}$	$\sigma_{g,g+2}$	$\sigma_{g,g+3}$	etc.		σ_g, last fast group
Symbol	OUSIG					

Comment

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.

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° 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

Comment

Transfer into thermal region for moderators only:
Start a new card for each group. Supply this data for all fast and thermal groups.

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° 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)

Comment

Supply a value for each group, including thermal groups.

Continue on additional cards if necessary

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

Comment

CONK < EPS1 < EPS2

.00001 .0003 .0005

for example

Word						
Column						
Format						
Card						
Symbol						

SELF SHIELDING DATA

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

Word	1					
Column	1-4					
Format	Integer					
Card 2B	Set number					
Symbol	I					

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 3B	First coefficient group (1)	Second coefficient group(1)	Third coefficient group(1)	First coefficient group (2)	Second coefficient group (2)	Third coefficient group (2)	Continue on another card if necessary.
Symbol	T1(I,1)	T2(I,1)	T3(I,1)	T1(I,2)	T2(I,2)	T3(I,2)	

Word	1	2	3				Comment
Column	1-4	5-6	7-8				
Format	Integer	Integer	Integer				
Card 4B	Id. Number of the self shielding set for isotope 1	Id. number of the self shielding set for isotope 2	etc.				Supply as many words as word 3 card A1. Continue on another card if necessary.
Symbol	ISET(1)	ISET (2)	ISET (3)				

CASE DATA

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	Integer	Integer	Integer
Card 2	Number of refueling intervals in the calculation (less or equal 15)	Number of total burn-up regions (less or equal 16)	Number of initial isotopic densities different from zero	Number of refueling intervals having a complete print-out	Maximum number of iterations (usually 100)	Maximum iterations in a single flux loop
Symbol	NCYCLE	NREG	NREAD	NCOS	JNSTOP	JNSTP

Word	7	8	9	10	11	12
Column	25-28	29-32	33-36	37-40	41-44	45-48
Format	Integer	Integer	Integer	Integer	Integer	Integer
Card 2 (cont.)	Iteration control number (as usually equal to word 5)	Number of diffusion regions	Buckling option; if greater than 0 the one-group buckl. will be shared among the groups	= 1 the number of burnable poison pin is searched. If less than zero NPIN=-NFIXP (1)*100	Number of buckling iterations	= 1 the fissile produced is recycled
Symbol	JNESTP	NZONE	NVECT	NFIXP(1)	NFIXP(2)	NFIXP(3)

Word	13	14	15	16		
Column	49-52	53-56	57-60	61-64		
Format	Integer	Integer	Integer	Integer		
Card 2 (cont)	self-shielding calc.option =0 average s.s.f. per diffusion zone. =1 average s.s.f. per burn-up region	Identification number of 1st burnable poison in the library	Identification number of the moderator (if any) in the burnable poison pin	Number of burnable poisons (less or equal 4)		
Symbol	NFIXP(4)	NFIXP(5)	NFIXP(6)	JPOIS		

Word	1	2				
Column	1-4	5-8				
Format	Integer	Integer				
Card 3	If less or equal zero, no burnable poison in this zone	if less or equal zero no burnable poison in this zone				
Symbol	NFIXP1(1)	NFIXP1(2)				

Comment
only if JPOIS is larger than 0. As many words as NZONE

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 4	Moderator density (if any) in the burable poison pins	Radius of burnable poison pins	Initial density of the 1st burnable poison isotope in the 1st diffusion region			Initial density of the last burnable poison isotope in the 1st diffusion region
Symbol	ZOX	RZERO	BP1(1,1)	BP1 (2,1)	BP1(3,1)	BP1(JPOIS,1)

Comment
only if JPOIS larger than 0. Continue on another card, if necessary

Word	1	2	3				Comment
Column	1-4	5-8	7-12				
Format	Integer	Integer	Integer				Supply as many words as the number of refueling intervals
Card 5	Type of search in the 1st interval = 0 feed search =-1 time search =+1 feed plus time search	Type of search in the 2nd interval	Type of search in the last refueling interval				
Symbol	NTYPE(1)	NTYPE(2)	NTYPE(NCYCLE)				

Word	1	2	3				
Column	1-4	5-8	7-12				
Format	Integer	Integer	Integer				
Card 6	If greater than zero the feed search is performed in this diffusion region						
Symbol	ICHANG(1)	ICHANG(2)	ICHANG(NZONE)				

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 7	Number of fission per watt.sec	Total reactor power(w)	Minimum value for the feed search	Maximum value for the feed search	Minimum value for the time search	Maximum value for the time search
Symbol	FIWATT	POWER	FMIN	FMAX	TMIN	TMAX

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 8	Feed (or time) initial change in the search (if 0 the code puts it equal to 1.5)	Diffusion coefficient of the reflector	Macroscopic absorption cross-section of the reflector	Core height	Cooling plus reprocessing and re-fabrication times (in the recycling cases)	Maximum quantity of fuel that can be re-cycled in each re-fueling interval
Symbol	DELTFD	DI(NZONE+1)	SA(NZONE+1)	HEIGHT	TRFUEL	ZDMX

Word	1	2	3	4		
Column	1-12	13-24	25-36	37-48		
Format	Decimal	Decimal	Decimal	Decimal		
Card 9	Radius of the first diffusion region(cm)	Radius of the second diffusion region(cm)	Radius of the last diffusion region(cm)	External radius of the reflecto (cm)		
Symbol	RZONE(1)	RZONE(2)	RZONE(NZONE)	RZONE(NZONE+1)		

Word	1	2	3			
Column	1-4	5-8	9-12			
Format	Integer	Integer	Integer			
Card 10	The id.Number of the isotope in which the first heavy metal is recycled (if=0 no recycling)		The id.number of the isotope in which the last heavy metal is recycled (if=0 no recycling)			
Symbol	NSWIT(1)	NSWIT(2)	NSWIT(NHEV)			

Comment
Only if word 12, card 2 is larger than 0

Word	1	2	3			
Column	1-12	13-24	25-36			
Format	Decimal	Decimal	Decimal			
Card 11	Fraction of the first isotope to be recycled		Fraction of the last isotope to be recycled			
Symbol	QR(1)	QR(2)	QR(3)			

Comment
Only if word 12, card 2 is larger than 0.

Word	1	2	3			
Column	1-4	5-8	9-12			
Format	Integer	Integer	Integer			
Card 12	If greater than zero the 1st burn-up zone belongs to the 1st diffusion region		If greater than zero, the last burn-up zone belongs to the 1st diffusion region			
Symbol	KZID(1,1)	KZID(2,1)	KZID(NREG,1)			

Comment
As many cards 12 as the number of the diffusion regions

Word	1	2	3			
Column	1-12	13-24	25-36			
Format	Decimal	Decimal	Decimal			
Card 13	The feed guess in the 1st diffusion region (1st cycle)		The feed guess in the last diffusion region (1st cycle)			
Symbol	FEED(1,1)	FEED(1,2)	FEED(1,NZONE)			

Comment
As many cards 13 as the number of the refueling intervals

Word	1	2	3			
Column	1-12	13-24	25-36			
Format	Decimal	Decimal	Decimal			
Card 14	Time of the first refueling interval (days) (fixed or guess)		Time of the last refueling interval (days)			
Symbol	DELDAY(1)	DELDAY(2)	DELDAY(NCYCLE)			

Comment
As many words as the number of refueling intervals

Word	1	2	3			
Column	1-4	5-8	9-12			
Format	Integer	Integer	Integer			
Card 15	Number of total burn-up regions in the first refueling interval		Number of total burn-up regions in the last refueling interval			
Symbol	NRG (1)	NRG(2)	NRG(NCYCLE)			

Word	1	2	3			
Column	1-12	13-24	25-36			
Format	Decimal	Decimal	Decimal			
Card 16	Fraction of the 1st burn-up region to be discharged at the end of the 1st refueling interval		Fraction of the last burn up region to be discharged at the end of the last refueling interval			
Symbol	XFRACT(1,1)	XFRACT(1,2)	XFRACT(1,NRG)			

Comment

Continue on another card, if necessary; as many cards 16 as the number of the refueling intervals

Word	1	2	3			
Column	1-12	13-24	25-36			
Format	Decimal	Decimal	Decimal			
Card 17	Final K-eff value (1st refueling interval)		Final K-eff value (last refueling interval)			
Symbol	ZKMIN(1)	ZKMIN (2)	ZKMIN(NCYCLE)			

Comment

As many words as the number of refueling intervals.

Continue on another card if necessary.

Word	1	2	3			
Column	1-12	13-24	25-36			
Format	Decimal	Decimal	Decimal			
Card 18	Difference between initial and final reactivity of 1st refueling interval		Difference between initial and final reactivity of the last refueling interval			
Symbol	DK(1)	DK(2)	DK(NCYCLE)			

Comment

As many words as the number of ~~refueling~~ intervals.

Continue on another card, if necessary.

Word	1	2	3			
Column	1-12	13-24	25-36			
Format	Decimal	Decimal	Decimal			
Card 19	Axial buckling of the first energy group and the first diffusion region		Axial buckling of the last energy group and the first diffusion region			
Symbol	TRANS(1,1)	TRANS(2,1)	TRANS(N26,1)			

Comment
Continue on another card, if necessary. As many cards 19 as the number of diffusion regions.

Word	1	2	3			
Column	1-12	13-24	25-36			
Format	Decimal	Decimal	Decimal			
Card 20	Vector for radial buckling first energy group and first diffusion region		Vector for radial buckling last energy group and first diffusion region			
Symbol	VECTOR(1,1)	VECTOR(2,1)	VECTOR(N26,1)			

Comment
Only if word 9 in card 2, is greater than 0. Supply as many cards 20, as the number of diffusion regions.

Word	1	2				
Column	1-4	5-16				
Format	Integer	Decimal				
Card 21	Id. number of the isotope in the library	Feed fraction in the isotope				
Symbol	II	Z(II)				

Comment

Supply as many cards 21, 22, 23, 24 as indicated by word 3 in card 2.

Word	1	2	3			
Column	1-12	13-24	25-36			
Format	Decimal	Decimal	Decimal			
Card 22	Initial concentration of the isotope in the 1st burn-up zone of the fresh core		Initial concentration of the isotope in the last burn-up zone of the fresh core			
Symbol	CDEN(II,1)	CDEN(II,2)	CDEN(IINREG)			

Comment

As many words as the initial number of the burn-up zones

Word	1	2	3			
Column	1-12	13-24	25-36			
Format	Decimal	Decimal	Decimal			
Card 23	Initial isotopic concentrat. to be re-fueled at the end of the 1st charge		Initial isotopic concentrat. to be re-fueled at the end of the last charge			
Symbol	BDEN(II,1)	BDEN(II,2)	BDEN(II,NCYCLE)			

Comment

As many words as the number of refueling intervals

Word	1	2	3			
Column	1-12	13-24	25-36			
Format	Decimal	Decimal	Decimal			
Card 24	Isotopic density multiplier in the 1st diffusion region		Isotopic density multiplier in the last diffusion region			
Symbol	ODEN(II,1)	ODEN(II,2)	ODEN(II,NZONE)			

Comment

As many words as the number of diffusion regions.

If II is less than NREAD, return to card 21.

ECONOMIC DATA

Word	1					
Column	1-4					
Format	integer					
Card C1	if zero, read new economic data; if greater than 0, employes the previous economic data; if less than 0, calculation stops					
Symbol	ICHG					

Word	1					
Column	1-72					
Format	alphanumerical					
Card C2	Case title					
Symbol	TITLE					

This card is the 1st of each RINA punch

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 C3	Case number	number of the refueling intervals	number of the heavy materials	number of cost data (effective only if ICHG is equal 0)	id. number of Uranium 235	id. number of Uranium 238
Symbol	ICASE	NCYCLE	NHEV	NCOST	NU5	NU8

Word	7	8	9			
Column	25-28	29-32	33-36			
Format	integer	integer	integer			
Card C3 (cont.)	number of refueling fractions	number of the refueling intervals belonging to the Running-In	number of diffusion zones			
Symbol	NREG	NEND	NZONE			

Word	1	2	3			
Column	1-4	5-8	9-12			
Format	integer	integer	integer			
Card C4	If larger than 0, the 1st isotope is charged into the core	.....	If larger than 0, the last isotope is charged into the core			
Symbol	NFED(1)	NFED(2)	NFED(NHEV)			

Word	1	2	3			
Column	1-4	5-8	9-12			
Format	integer	integer	integer			
Card C5	If larger than 0, the 1st isotope is discharged from the core	.....	if larger than 0, the last isotope is discharged from the core			
Symbol	NREC(1)	NREC(2)	NREC(NHEV)			

Word	1	2	3	4	5	
Column	1-12	13-24	25-36	37-48	49-60	
Format	decimal	decimal	decimal	decimal	decimal	
Card C6	Reactor thermal power (w)	Core volume. (cm^3)	Thermal efficiency	Load factor	Reactor life (y)	
Symbol	POWER	VOLUME	ETA	HOUR	TLIFE	

After this, the cards of the RINA punch have to be introduced (apart the title card)

Word	1	2	3			
Column	1-12	13-24	25-36			
Format	decimal	decimal	decimal			
Card C7	value of the 1st heavy metal ($/Kg)	.....	value of the last heavy metal ($/Kg)			
Symbol	V(1)	V(2)	V(NHEV)			

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 C8	Cost of separative work ($/Kg)	tail enrichment	Cost of natural Uranium	Annual interest rate	Additional cost in the Uranium gaseous phase	Fabrication dependence on the initial enrichment
Symbol	SWRK	TAIL	CU38	AIR	ALFA9	ALFA8

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 C9	fabrication unit cost $/Kg	Reprocessing unit cost $/Kg	Graphite fabrication cost $/Kg of graphite	Delay time from the fissile purchase to the reactor in (y)	Fabrication time (y)	Reprocessing time (y)
Symbol	FABR	REPR	FAGR	TA	TF	TR

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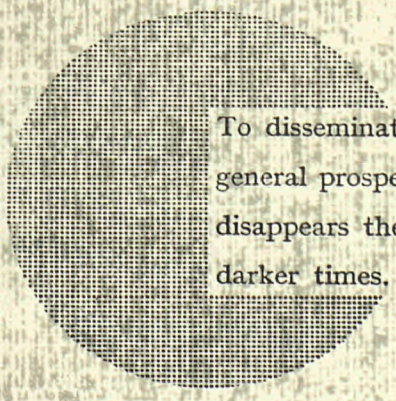

To disseminate knowledge is to disseminate prosperity — I mean general prosperity and not individual riches — and with prosperity disappears the greater part of the evil which is our heritage from darker times.

Alfred Nobel

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