



**AUSTRALIAN ATOMIC ENERGY COMMISSION  
RESEARCH ESTABLISHMENT  
LUCAS HEIGHTS**

**PERTURBATION THEORY APPROACH TO FUEL MANAGEMENT  
STUDY OF THE PEBBLE BED REACTOR**

by

**A. BICEVSKIS**

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ABSTRACT

A method is proposed, based on perturbation theory, to assess the relative merits of fuel management schemes for the pebble bed reactor. The method is suitable for multi-group treatment and could be applied to other continuously fuelled reactors.

As a specific example the computer code N.I.B. is developed based on spectrum weighted one-group treatment. Preliminary reactivity lifetime investigations are described. Some power density calculations are also reported for a random distribution of the pebbles.



## CONTENTS

	Page
1. INTRODUCTION	1
2. PRINCIPLE OF THE METHOD	1
3. NEUTRON IMPORTANCE BALANCE CODE (N.I.B.)	2
3.1 General Description	2
3.2 Scheduling of Pebble Flow	2
3.3 Flux Distribution	3
3.4 Burnup Calculation	3
3.5 Reactivity Calculation	4
3.6 Iteration for Reference Reactivity	5
4. ASSESSMENT OF FUEL MANAGEMENT	5
4.1 Data Used	5
4.2 Possible Management Schemes	6
4.3 Sequence of Pebble Passes	6
4.4 Number of Passes	7
4.5 Decay Outside Core	7
5. POWER DENSITIES FOR RANDOM FEED	8
6. EXTENSION OF THE METHOD TO SPACE DEPENDENT CALCULATIONS	8
7. REFERENCES	9

Figure 1 Neutron Importance Balance Code – Functional Block Diagram

Figure 2 Sample Output of N.I.B. Code

Figure 3 Power Density of a Pebble Started at the Centre in Chopped Cosine Flux Field

Figure 4 Power Density of a Pebble Started at the Core Periphery in Chopped Cosine Flux Field

Figure 5 Maximum to Average and Maximum to Minimum Power Density Ratios for Random Feed



## 1. INTRODUCTION

The investigations covered in this report relate to the  $\text{PuO}_2\text{-ThO}_2\text{-BeO}$  pebble bed reactor study undertaken by the Australian Atomic Energy Commission and described by Ebeling and Hayes (1966). The burnup and fuel management problems for the project in order of increasing complexity, are:

### (A) Operation at Equilibrium

- (a) Fuel selection by zero-dimensional burnup studies.
- (b) Fuel management.

### (B) Approach to Equilibrium

It is reasonable to base preliminary comparisons on studies carried out at equilibrium because a reactor will operate for a large fraction of its life under conditions which approach steady state operation. For the final assessment, the approach to equilibrium must be investigated as a time dependent problem (involving a large increase in computing costs).

Any realistic fuel management study of a pebble bed reactor (P.B.R.) must include both radial and axial dependence. In general, the criterion of any management scheme is the optimization of fuel costs which depend on both the obtainable burnup and average core power density. It is clear that any study of two-dimensional fuel management at equilibrium is time-consuming but the approach to equilibrium would be beyond the capabilities of most currently available computers. Consequently there is an obvious need to develop methods that can provide approximate but still useful results at relatively low computing cost.

In the course of the P.B.R. study, the fuel composition was selected by means of the zero-dimensional burnup methods based on the equilibrium spectrum (Bicevskis 1966; Bicevskis and Hesse 1966). Before starting a large scale survey of the many possible schemes of fuel management, it was decided to make a preliminary investigation to select the most promising schemes. The perturbation theory approach was developed for this purpose in 1964. Only the reactivity lifetime aspect is considered and a constant average power density is assumed. The numerical results are subject to the same uncertainties as any predictions based on the perturbation theory.

## 2. PRINCIPLE OF THE METHOD

In the general case of a multi-group treatment one assumes that the following parameters have already been obtained:

- (1) Fuel composition.
- (2) Cross sections derived from the average properties and spectrum at reference burnup.
- (3) Core and reflector size which define the reference reactivity of the system.
- (4) Distribution of the flux and adjoint flux in the specified number of groups.

The method of calculation proceeds by the following steps:

- (1) Specify a definite pebble path through the reactor.
- (2) Calculate the stage of burnup for each position.
- (3) Calculate the effect of reactivity for the resultant distribution of properties in the core.
- (4) Change the burnup, keeping the average core power density constant, until the reactor returns to the reference reactivity. The departure from reference burnup serves as a figure of merit for assessing various fuel management schemes.

To provide preliminary reactivity lifetime estimates a computer code Neutron Importance Balance (N.I.B.) was developed adopting a spectrum weighted one-group treatment.

### 3. NEUTRON IMPORTANCE BALANCE CODE (N.I.B.)

#### 3.1 General Description

The code N.I.B. was written in FORTRAN 4 and its functional arrangement is shown in Figure 1 which is largely self-explanatory if taken in conjunction with the remarks made above. Provision was made for ageing the fuel outside the core at the end of each pass to check whether the decay of Pa233 to U233 is valuable. A synthetic fission product treatment was used without decay reactions except in the case of Xe135.

A sample of N.I.B. output is reproduced (Figure 2) listing the input data and first iteration for reference reactivity.

Besides its main objective to calculate F.I.F.A. (fissions per initial fissile atom) as a function of fuel management, N.I.B. may be used to obtain pebble power density for any specified core flux distribution, as well as for zero-dimensional burnup calculations.

E.W. Hesse (1966) has shown that for the large pebble bed cores considered in our study, one-group methods with reflector savings predict critical core sizes with good accuracy, provided the one-group effective cross sections are prepared for the correct spectrum. N.I.B. adopts the condensed one-group treatment. Extension to a multi-group scheme is discussed in Section 6.

#### 3.2 Scheduling of Pebble Flow

There are many possible ways of arranging an ordered pebble flow. For preliminary studies, undivided pebble flow is assumed, that is, all pebbles follow the same route depending on the stage of burnup. The inference is that more sophisticated schemes may involve branching the flow, that is, adding half of the new balls at the centre and the other half at the outside diameter of the reactor core as discussed below.

With undivided flow the pebble path through the reactor may be represented by a continuous thread passing downwards through the core and looping back outside the core, traversing all the volume elements in the selected sequence.

In a reactor at equilibrium, the total number of balls in the core does not change and thus the feed rate (new balls) would be equal to the discharge rate and also to the flow rate in any pass. Assuming a constant core density, (there would be no difficulty in incorporating a variable voidage) we also get the same volumetric flow rate per pass.

Our model represents a two dimensional study of a cylindrical reactor in  $r - z$  geometry (radial and axial dependence) excluding angular dependence. It is therefore reasonable to assume annular pass cross sections and we then get:

$$F = \int_{R_1}^{R_2} 2\pi r V(r) dr,$$

where  $F$  = volumetric flow rate per pass (ft<sup>3</sup>/hr)

$R_1, R_2$  = radial boundaries of the pass (ft)

$V(r)$  = pebble axial velocity profile (ft/hr).

We have assumed here that pebble flow is "laminar", that is, there is no radial velocity component (a tangential component would not affect our argument). The above expression gives a simple rule for determining the pass boundaries for any experimentally obtained pebble velocity distribution.

For our calculation, it was assumed that the pebble flow takes place in piston or slug form, that is, with a flat velocity distribution across the core. It appears that this type of flow is achieved with certain configurations of the core bottom hoppers. With a constant pebble velocity, all pass areas must be equal, in order to satisfy the above equation of continuity. Thus the widths of the annuli would decrease with an increasing radius.

There is no difficulty, in principle, in extending the above model to cater for more complicated pebble path schedules. Let us assume that all pebbles make  $M$  passes through the core. If one provides  $N \times M$  equal flow areas, it follows that one may split the total pebble flow in  $N$  equal mass flows. If each of the  $N \times M$  annular elements is traversed once, the equation of continuity is automatically satisfied irrespective of the sequence of the passes. If on the other hand one provides  $M/N$  annular elements with equal volumetric flow ( $M/N$  must be an integer, of course), any  $N$  passes may traverse a volume element.

Further combinations of the above schemes are possible. For the branching of the total flow in  $N$  parts, each may have a completely independent existence, that is, there need not be any mixing for subsequent passes.

There is no difficulty, in principle, in accounting for the reactivity balances of such schemes and they would constitute typical digital computer applications.

### 3.3 Flux Distribution

The programme is designed for a cylindrical reactor and the core is subdivided into  $N \times M$  volume elements by the intersection of equidistant planes, which are at right angles to the axis, with cylindrical radial boundaries. Thus, one may consider the core as consisting of  $N$  equal axial slices, or, alternatively, as  $M$  (hollow) concentric cylinders, the intersections of these boundaries forming the volume elements. The sectional areas of all the cylindrical elements are equal, as stated above, and therefore all the annular elements have equal volumes. The volume elements are numbered in the axial direction, starting from the top, and radially, starting from the centre. A pebble pass is represented by a hollow cylinder, the movement taking place downwards from one volume element to the next.

Using the same notation as in the programme:

$$\text{FLUX}(N,M) = \text{FXA}(N) \times \text{FXR}(M),$$

where  $\text{FLUX}(N,M)$  = flux in element with axial position  $N$  and radial position  $M$ .

$$\text{FXA}(N) = \text{shape factor for axial position } N,$$

$$\text{FXR}(M) = \text{shape factor for radial position } M.$$

As shown in Figure 1, the flux shape factors may be either read in or calculated by the programme for cosine distribution. The absolute values of the flux are normalized to give the required average flux,  $\text{FXAV}$ (read in).

$$\sum \sum \text{FLUX}(N,M) = \text{ANAX} \times \text{ANRD} \times \text{FXAV},$$

where

$$\text{ANAX} = \text{total number of axial regions},$$

$$\text{ANRD} = \text{total number of radial regions}.$$

### 3.4 Burnup Calculation

Any burnup calculation involves, in general:

(1) flux level, and

(2) residence time.

The flux level for each volume element is determined as shown above.

The time per pebble pass is specified as the input of the code. The residence time per volume element follows with the specified number of axial elements.

The burnup mechanism is the same as used in the ORVOP code developed by the author (unpublished). The total residence time for each volume element is subdivided in M time steps for the finite difference scheme used in ORVOP.

The average properties of a volume element are then calculated as follows:

- (1) macroscopic absorption cross section

$$\Sigma_{ae} = \sum_i \Sigma_{ai} / M ,$$

- (2) macroscopic neutron production cross section

$$\nu \Sigma_{f,e} = \sum_i \nu_i \Sigma_{fi} / M ,$$

and from these values:

- (3) material buckling

$$B_e^2 = \frac{\chi \nu \Sigma_{f,e} - \Sigma_{ae}}{D} ,$$

where  $\chi$  denotes the net beryllium enhancement factor defined in one-group treatment as follows:

$$k_{inf} = \frac{\chi \nu \Sigma_f}{\Sigma_a^*} .$$

The infinite medium multiplication constant  $k_{inf}$  may be obtained from a spectrum calculation and the net macroscopic absorption cross section is defined in turn as:

$$\Sigma_a^* = \Sigma_a - (\Sigma_{n,\alpha} + \Sigma_{n,2n}) ,$$

where  $\Sigma_{n,\alpha}$  and  $\Sigma_{n,2n}$  are, respectively, the beryllium (n, $\alpha$ ) and (n,2n) macroscopic cross sections.

Calculations were made with several fuel compositions and BeO moderator and in all cases a value for  $\chi$  of approximately 1.05 was obtained. The (n, $\alpha$ ) and (n,2n) reactions in beryllium are restricted to an energy range above 1 MeV. It is reasonable to expect that all fission neutrons would experience the same net enhancement in BeO, slowing down past this energy threshold.

The fact that  $\chi$  is spectrum independent is of great advantage for perturbation studies as the same value may be used for burnups departing from the reference case, which is used to produce the fixed cross section library. It has also been shown (Bicevskis and Hesse 1966) that the burnup predictions for the equilibrium core at F.I.F.A. > 1 are not sensitive to 20 per cent. variations in the cross sections of Pu239 and Pu240.

### 3.5 Reactivity Calculation

The above calculations would define the material properties (concentrations, macroscopic cross sections, etc.) for all volume elements of the core.

The perturbation in reactivity, compared with a uniform core, will be the change in neutron leakage at the same total core power.

Starting with the neutron balance equation and introducing the neutron importance  $\phi^*$  one obtains by integrating over the volume of the reactor:

$$\int_V \phi^* \nabla \cdot D \nabla \phi \, dV + \int_V \phi^* (\chi \nu \Sigma_f - \Sigma_a) \phi \, dV = 0 .$$

With constant  $D$  (our results indicate that the diffusion coefficient changes little during burnup), making use of the fact that one-group operators are self-adjoint ( $\phi = \phi^*$ ) one obtains:

$$D \int_V \phi \nabla^2 \phi \, dV + D \int_V B_m^2 \phi^2 \, dV = 0 \quad ,$$

where  $B_m^2$  is variable and depends on the local properties.

One can define an equivalent  $B^2$  for the reactor, in terms of the N.I.B. scheme of equal volume elements, the buckling for each volume element being  $B_e^2$ :

$$B^2 = \frac{\sum_i \phi_i^2 B_{ei}^2}{\sum_i \phi_i^2}$$

This can be expressed in the form:

$$B^2 = \sum_i W_i B_{ei}^2 \quad ,$$

where

$$W_i = \frac{\phi_i^2}{\sum_i \phi_i^2} \quad ,$$

is commonly called the statistical weight of the volume element.

The great advantage of the N.I.B. scheme is that the local isotopic compositions and  $B_e^2$  need not be stored for all the volume elements because the sum of  $B_e^2 W$  is formed as the burnup calculation proceeds.

### 3.6 Iteration for Reference Reactivity

To illustrate the method adopted, let us assume that any particular path schedule, operating with standard flux and pebble velocity (time per pass), departed from the reference reactivity or  $B^2$ . If the departure gave an increase in reactivity, it could be returned to standard by decreasing the pebble velocity (increasing residence time). This would result in a larger value of F.I.F.A. and lower core reactivity as the average fissile content of the core would decrease (with conversion ratios less than one). The latter, however, would also reduce the average core power density with the reference flux. It follows that one must also change the flux level (increase in our example) in conjunction with the residence time. A Regula Falsi routine was developed with the integrated flux as the independent variable and the departure from reference reactivity as the dependent variable. Keeping the ratio of flux to residence time constant at a pre-determined value, the average core power density could be kept constant.

## 4. ASSESSMENT OF FUEL MANAGEMENT

### 4.1 Data Used

The investigations were based on a fuel composition 1:16.5:1650 (fissile:fertile:moderator). Plutonium as  $\text{PuO}_2$  was specified as the fissile material and thorium as  $\text{ThO}_2$  as the fertile material,  $\text{BeO}$  being the fuel dispersant-moderator. The current reference core diameter of 14.1 ft and height of 11.8 ft were used.

In the tables given below, a chopping ratio of 0.0 implies a flat flux distribution (point model), and a chopping ratio of 1.0 a full cosine shape. For most calculations, a chopping ratio of 0.74 was selected which was the value used for power density distribution (private communication, P.A.E. Hawker).

Preliminary nuclear data were used for the investigations and only percentage variations in the obtainable reactivity lifetime are quoted.

#### 4.2 Possible Management Schemes

In general, the following main factors may be "managed" in the pebble bed concept with fuel elements containing both  $\text{PuO}_2$  and  $\text{ThO}_2$ :

- (i) Position of feed at the top of the core.
- (ii) Number of passes.
- (iii) Fuel velocity distribution across the core.
- (iv) Decaying (ageing) of the irradiated fuel outside the core.

Item (i) is the most important factor. The feed position would, in general, include both radial and angular dependence. Only the radial dependence (cylindrical symmetry), which is governed by the sequence of passes, is considered in our investigations.

The number of passes affects the axial distribution of the burnup stages and it also affects the "definition" of the radial distribution. This item is investigated below.

Item (iii) was not considered but it could be included easily as experimentally determined velocity distributions become available.

Item (iv) was investigated with both space dependent and independent (flat flux) models.

Further management schemes would be possible by using separate fissile and fertile balls. The P.B.R. lends itself readily to introduction of the fissile balls at the positions of highest importance (centre). On the other hand, the fertile balls could be fed initially to positions of low importance (periphery of the core) and then, as the fissile material builds up, transferred in subsequent passes nearer to the centre. However, to treat this problem adequately, methods are required that account for the variations in the local spectrum.

#### 4.3 Sequence of Pebble Passes

For this investigation, six pebble passes were assumed as reasonable for the current reference design and fuel composition. With six passes, 720 permutations of the pass sequence are possible. Only a few representative cases were selected for this preliminary study and the results are reproduced in the following table:

TABLE 1  
EFFECT OF PASS SEQUENCE ON REACTIVITY LIFETIME

Run No.	Sequence of Passes	Chopping Ratio	Decay After Each Pass, Days	Departure from Reference F.I.F.A., %
30	1,2,3,4,5,6	0.74	0	+ 6.9
48	2,1,3,4,5,6	0.74	0	+ 3.4
49	3,1,2,4,5,6	0.74	0	- 0.1
50	4,1,2,3,5,6	0.74	0	- 3.2
51	5,1,2,3,4,6	0.74	0	- 5.9
52	6,1,2,3,4,5	0.74	0	- 7.8
31	6,5,4,3,2,1	0.74	0	-18.2

The reference value of F.I.F.A. is based on a random distribution of the pebbles in the core.

Run No. 30 was the best arrangement as regards reactivity lifetime; new fuel was introduced at the centre with successive passes moving outwards. It gave the highest improvement of 6.9 per cent.

Run No. 31 was the worst arrangement with new fuel introduced at the periphery and the successive passes moving towards the centre. It gave a value of F.I.F.A. which was 18.2 per cent. smaller than the reference figure. Thus the maximum difference obtainable by path scheduling alone was 25.1 per cent. The intermediate schemes (Run Nos. 48, 49, 50, 51, and 52) gave, as expected, results between the two extremes.

It follows that no large gains in F.I.F.A. can be expected even for the most favourable management scheme which would necessitate fairly elaborate fuel distribution and collection hoppers. On the other hand, extreme management schemes attempting to flatten the power distribution by introducing new fuel at the periphery may have to accept a larger reduction in F.I.F.A.

#### 4.4 Number of Passes

With the other assumptions unchanged the effect of changing the number of passes is as follows for the two extreme schemes:

TABLE 2

EFFECT OF NUMBERS OF PASSES ON REACTIVITY LIFETIME

Run No.	Sequence of Passes	Chopping Ratio	Decay After Each Pass, Days	Departure from Reference F.I.F.A., %
38	1,2	0.74	0	- 0.6
40	1,2,3,4	0.74	0	+ 4.3
30	1,2,3,4,5,6	0.74	0	+ 6.9
42	1, . . .,8	0.74	0	+ 8.4
44	1, . . .,12	0.74	0	+10.2
46	1, . . .,18	0.74	0	+11.5
39	2,1	0.74	0	-18.0
41	4,3,2,1	0.74	0	-18.6
31	6,5,4,3,2,1	0.74	0	-18.2
43	8, . . .,1	0.74	0	-17.8
45	12, . . .,1	0.74	0	-17.4
47	18, . . .,1	0.74	0	-17.2

As shown in Table 2, starting the sequence from the centre of the top of the core, the reactivity lifetime improved progressively as the number of passes increased.

Starting the new fuel on the outside, the reactivity lifetime was not significantly affected by the number of passes. One may infer that in this case some mixing of the passes would have little effect also. These comments are relevant to management schemes where an attempt is made to flatten the radial power distribution.

#### 4.5 Decay Outside Core

Some runs were repeated with a decay period of 60 days outside the core after each pass (Table 3). In a similar manner for the space independent case (No. 35) the "ageing" was carried out after each sixth of the total core residence time.

TABLE 3

EFFECT OF DECAY OUTSIDE THE CORE ON REACTIVITY LIFETIME

Run No.	Sequence of Passes	Chopping Ratio	Decay After Each Pass, Days	Departure from Reference F.I.F.A., %
35		0.0	60	1.4
32	1,2,3,4,5,6	0.74	60	0.3
33	6,5,4,3,2,1	0.74	60	0.6

The last column in the table represents the difference in F.I.F.A. for otherwise similar runs except for decay outside the core. It is interesting to note that the space independent method (Run 35) gave a gain of 1.4 per cent. whereas for the space dependent calculations the gain was less than half. This is not surprising as the variable flux encountered by the pebble would approximate an on-and-off schedule and achieve a large fraction of the total gain possible. The fission product treatment did not allow for decay except in the case of Xe135 so that the above results should be attributed to the changes in the decay of Pa233 to U233 only.

**5. POWER DENSITIES FOR RANDOM FEED**

Calculation of the thermal stresses in the pebbles requires determination of the power density ratio of the least spent pebble to the average pebble. N.I.B. was used to supply this information for the reference case based on a random feed to the core.

It was assumed that both the radial and axial power distribution could be represented by a chopped cosine function with a chopping ratio of 0.74. For a uniform core composition based on the average pebble properties, this would also lead to a similar distribution of the equivalent one group flux. The flux level was normalized to give the required average core power density of  $9.7 \text{ W cm}^{-3}$ . Assuming, further, six passes through the core, a pebble was started at the centre and the power density was obtained as shown in Figure 3. For comparison, a pebble was started at the periphery in the same flux field and the resultant power density distribution is shown in Figure 4. The first pass in both Figures 3 and 4 represents the power density of the least spent pebble or the maximum power density. With the maximum power densities from Figures 3 and 4 and the average power density from the normalized chopped cosine distribution the ratio of maximum to average power density shown in Figure 5 was obtained. Of most interest for the calculation of thermal stresses are the ratio values on the core axis which start with 2.35 at the top and gradually fall to 1.17 at the bottom. The limiting value or "design point" normally is nearly half-way up the core. This value is approximately 1.6, as shown in Figure 5, which also displays the maximum to minimum ratios.

The ratios of power densities are much higher than the ratios of fissile concentrations for new and average pebbles. This unfavourable feature is produced by the much larger fission cross section of the initial plutonium fuel compared with the bred fuel U233 dominating the average composition.

**6. EXTENSION OF THE METHOD TO SPACE DEPENDENT CALCULATIONS**

A continuously fuelled reactor at equilibrium is characterized by a time independent distribution of flux and core properties, the space dependence being a function of the management scheme adopted. Such a system must be calculated in an iterative manner, alternating between, first, the burnup calculations which define the material properties for the given flux distribution and management schedule and, second, calculation of the flux distribution and criticality for the given distribution of core properties.

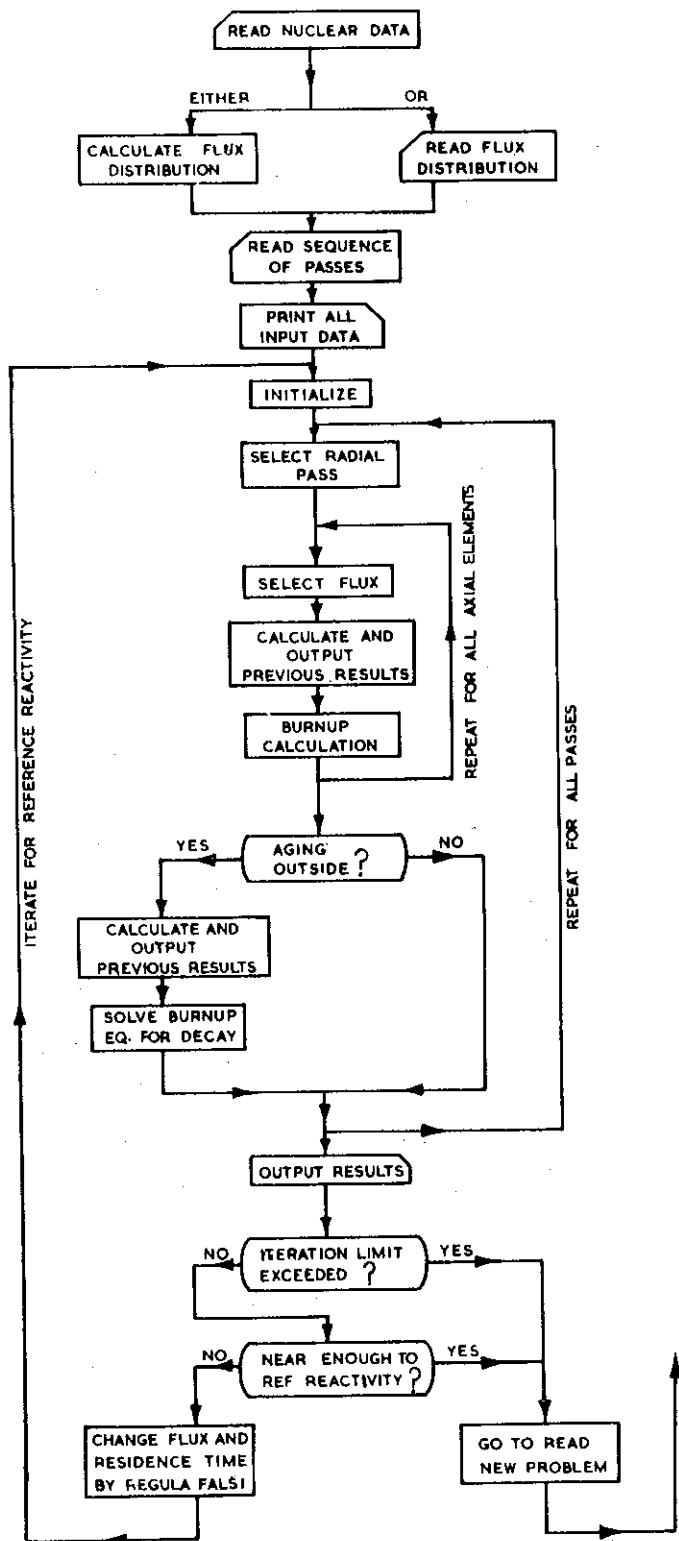
The burnup calculation for each step is relatively inexpensive as it is equivalent to as many zero-dimensional calculations as there are separate fuel element flows. However, the space dependent flux and criticality calculation would be much more time consuming because most practical problems necessitate a two-dimensional treatment. It appears that the perturbation method developed in the N.I.B. code could be used in iterative fuel management schemes to reduce the number of space dependent criticality calculations.

There are no difficulties, in principle, in adopting a multi-group treatment, and suitable expressions for the perturbation of reactivity are available. The method described could be applied to other reactors besides pebble bed reactors, with regular movement of fuel in the core such as axial shuffling.

## 7. REFERENCES

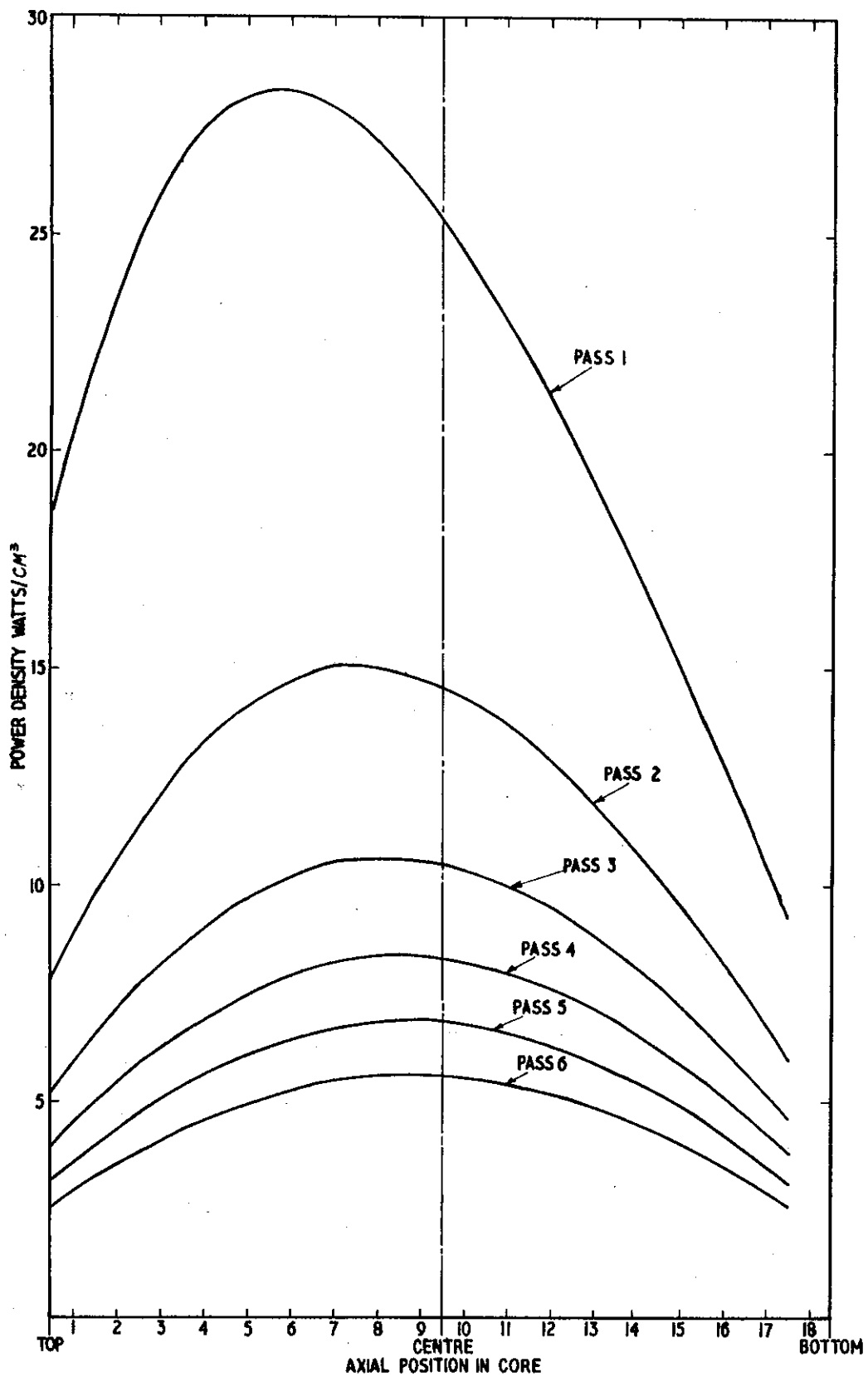
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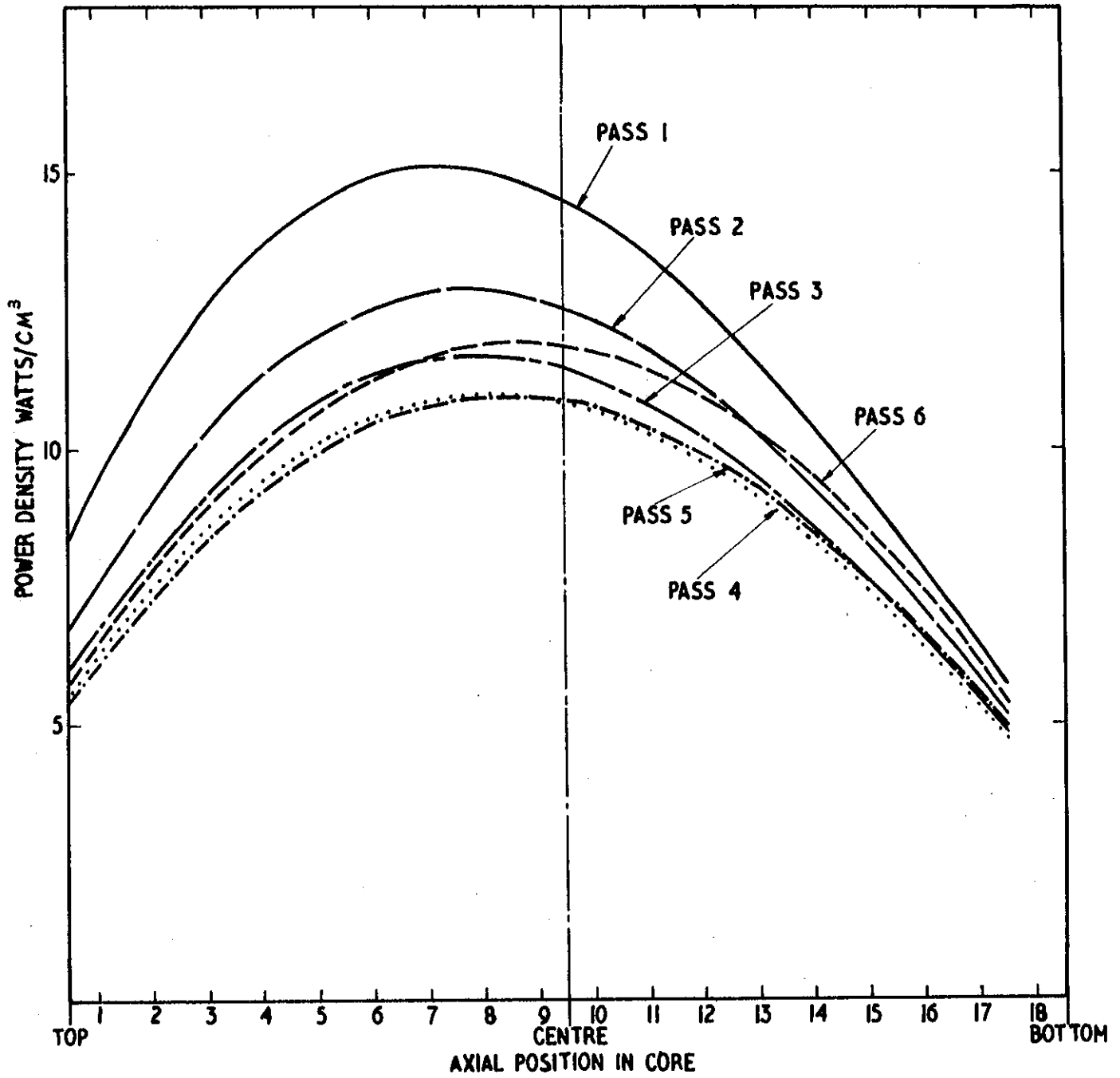


**FIGURE 1. NEUTRON IMPORTANCE BALANCE CODE —  
FUNCTIONAL BLOCK DIAGRAM**

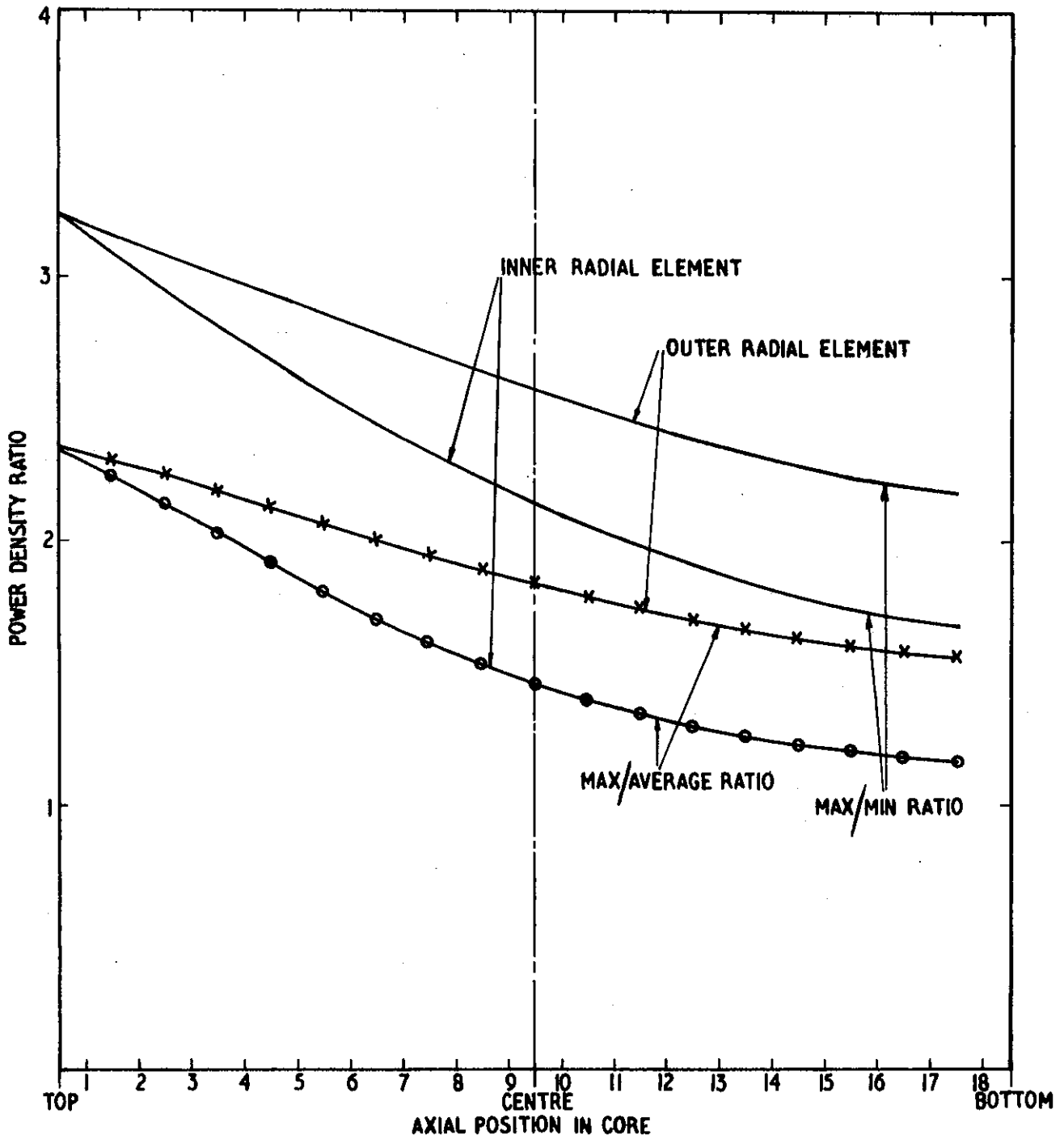




**FIGURE 3. POWER DENSITY OF A PEBBLE STARTED AT THE CENTRE IN CHOPPED COSINE FLUX FIELD**



**FIGURE 4. POWER DENSITY OF A PEBBLE STARTED AT THE CORE PERIPHERY IN CHOPPED COSINE FLUX FIELD**



**FIGURE 5. MAXIMUM TO AVERAGE AND MAXIMUM TO MINIMUM POWER DENSITY RATIOS FOR RANDOM FEED**

