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DEVELOPMENT OF A THREE-DIMENSIONAL NEUTRON DIFFUSION CODE SERIES BY LEAKAGE ITERATIVE METHOD

May 1979

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Development of a Three-Dimensional Neutron Diffusion Code Series by Leakage Iterative Method

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In nodal and coarse mesh methods, one of the most difficult problems is to estimate neutron leakages from a subregion because spatial mesh widths are too wide to evaluate the neutron current at the boundary of the subregion accurately. To eliminate this difficulty, a leakage iterative method is proposed and several computer codes have been developed. The contents of these codes are briefly shown in this article.

Keywords: Coarse Mesh Method, Neutron Diffusion Code, Leakage Iterative Method, Three-Dimensional Calculation, Neutron Leakage

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漏洩量繰返し法による3次元中性子拡散コード・シリーズの開発

日本原子力研究所東海研究所動力試験炉部

内藤俶孝

（1979年4月14日受理）

ノード法や粗メッシュ法において最も困難な問題の1つは部分領域からの中性子漏洩量を推定することである。なぜなら，部分領域の境界における中性子流を高精度で求めるには上記計算手法における空間メッシュの巾は広すぎる。この困難さを取除くために，漏洩量繰返し法が提案され，いくつかの計算コードが開発された。この報告書では，これ等の計算コードの内容を簡単に示す。

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1. Outline of Leakage Iterative Method

In the leakage iterative method (LIM), the reactor is divided into several layers along the z axis and into several rectangular channels across the xy plane as shown in Fig.1. A parallelepiped formed by a channel and a layer is called a block. To start the iterative procedure for solving the diffusion equation, the neutron source and the radial leakage coefficients are assumed as shown in Fig.2. A one-dimensional neutron flux calculation is performed for each channel with the radial leakage coefficient. A two-dimensional neutron flux calculation is then made for each layer with the axial leakage determined from the one-dimensional calculation. The one- and two-dimensional leakage calculations are iterated until the consistency is attained between both. At each step of the iteration the neutron source distribution and the eigenvalue are recalculated. For obtaining the balance of the neutron population within a block, it is important to evaluate the neutron leakage from the block as precisely as possible. For this purpose, a block is subdivided into fine meshes, and the fine-mesh difference approximation method is applied to solve the one- and two-dimensional neutron diffusion equations for each channel and layer, respectively.

The present method has the following characteristics:

1) A fine-mesh difference approximation technique is applied only to the channels and layers. Therefore it is not necessary to calculate the neutron fluxes at all fine-mesh points in the core and thus the computer time is reduced. If the block is a 12 cm cube and the mesh width is 2 cm, the number of fine-mesh points is $6 \times 6 \times 6 = 216$. In the present method, however, the number of mesh points used for the calculations is $6 + (6 \times 6) = 42$, that is about one-fifth of the former. The physical

quantities connecting the channel and layer calculations are only the neutron leakage and the neutron source, and the fact reduces the computer memory required.

2) Since the neutron leakage from a block is calculated by a fine-mesh difference approximation, the numerical error due to discretization is minimized.

3) When only one fine-mesh point is located in each block, this approach becomes the same as the usual method in a fine-mesh difference approximation. In this case, the iterative scheme coresponds to one of the variants of the Peaceman-Rachford iterative method. Therefore, it is possible to establish the condition, under which the consistency is achieved between the axial and radial leakages in the same manner as ADI (alternating direction implicit iterative methods of Peaceman and Rachford). In addition, it is easy to compare the results with those obtained from conventional fine-mesh difference approximation methods. The computer code can thus be used for calculating both the collapsed flux and the fine-mesh flux.

2. Fundamental Equation

The iterative process to recalculate the source distribution with use of the previously obtained neutron flux is the same as that utilized in the conventional power-iteration method. This iterative process is called as the outer iteration or source iteration, as usual.

The problem with which are concerned here is how to calculate the neutron flux distribution for a fixed neutron source distribution. This process is called as the inner iteration.

The fundamental equation to be solved is

$$\nabla D \nabla \phi - \Sigma_T \phi + \theta = 0 \quad , \tag{1}$$

where ϕ is the neutron flux and θ the neutron source at a fine-mesh point, and D and Σ_T are the diffusion coefficient and the macroscopic removal cross section, respectively. Integrating Eq.(1) over a block results in

$$D\int_{\Delta x}\int_{\Delta y} dxdy \int_{\Delta z} \frac{\partial^2 \phi}{\partial z^2}\, dz$$

$$+ D\int_{\Delta z} dz \int_{\Delta x}\int_{\Delta y} \left(\frac{\partial^2}{\partial x^2} + \frac{\partial^2}{\partial y^2}\right)\phi dxdy$$

$$+ \Sigma_T \int_{\Delta x}\int_{\Delta y}\int_{\Delta z} \phi dxdydz + \int_{\Delta x}\int_{\Delta y}\int_{\Delta z} \theta dxdydz = 0 \quad , \tag{2}$$

where Δx, Δy, and Δz are the widths of the block. Using the following notations:

$$-\ell_z \equiv \frac{\int_{\Delta x}\int_{\Delta y} dxdy \int_{\Delta z} D \frac{\partial^2 \phi}{\partial z^2}\, dz}{\int_{\Delta x}\int_{\Delta y}\int_{\Delta z} \phi dxdydz} \quad ,$$

$$-\ell_{xy} \equiv \frac{\int_{\Delta x}\int_{\Delta y}\int_{\Delta z} D\left(\frac{\partial^2}{\partial x^2} + \frac{\partial^2}{\partial y^2}\right)\phi dxdy}{\int_{\Delta x}\int_{\Delta y}\int_{\Delta z} \phi dxdydz} \quad ,$$

$$\psi_z \equiv \int_{\Delta x}\int_{\Delta y} \phi dxdy \quad , \qquad \psi_{xy} \equiv \int_{\Delta z} \phi dz \quad ,$$

$$\theta_z \equiv \int_{\Delta x}\int_{\Delta y} \theta dxdy \quad , \qquad \theta_{xy} \equiv \int_{\Delta z} \theta dz \quad , \tag{3}$$

Equation (2) is rewritten as

$$\int_{\Delta z} D \frac{\partial^2 \psi_z}{\partial z^2} dz - \int_{\Delta z} (\Sigma_T + \ell_{xy})\psi_z dz + \int_{\Delta z} \theta dz = 0 \quad , \tag{4}$$

$$\int_{\Delta x} \int_{\Delta y} D(\frac{\partial^2}{\partial x^2} + \frac{\partial^2}{\partial y^2})\psi_{xy} \, dxdy - \int_{\Delta x} \int_{\Delta y} (\Sigma_T + \ell_z)\psi_{xy} \, dxdy$$

$$+ \int_{\Delta x} \int_{\Delta y} \theta_{xy} \, dxdy = 0 \quad , \tag{5}$$

where ψ_z and ψ_{xy} are obtained respectively by solving the following one- and two-dimensional fine-mesh neutron diffusion equations:

$$D \frac{\partial^2 \psi_z}{\partial z^2} - (\Sigma_T + \ell_{xy})\psi_z + \theta_z = 0 \quad , \tag{6}$$

$$D(\frac{\partial^2}{\partial x^2} + \frac{\partial^2}{\partial y^2})\psi_{xy} - (\Sigma_T + \ell_z)\psi_{xy} + \theta_{xy} = 0 \quad . \tag{7}$$

One-dimensional fine-mesh neutron flux distribution calculations are performed along the channels, and two-dimensional calculations are made over the layers. The axial and radial leakages from each block, ℓ_z and ℓ_{xy}, are obtained as follows:

$$-\ell_z = \frac{\int_{\Delta z} D \frac{\partial^2 \psi_z}{\partial z^2} dz}{\int_{\Delta z} \psi_z \, dz} \quad ,$$

$$-\ell_{xy} = \frac{\int_{\Delta x} \int_{\Delta y} D(\frac{\partial^2}{\partial x^2} + \frac{\partial^2}{\partial y^2})\psi_{xy} \, dxdy}{\int_{\Delta x} \int_{\Delta y} \psi_{xy} \, dxdy} \quad . \tag{8}$$

The neutron flux distribution in the core is now determined by solving Eqs.(6) and (7) alternately.

The neutron flux convergence criterion is given by

$$\left| \frac{\int_{\Delta z} \psi_z \, dz - \int_{\Delta x} \int_{\Delta y} \psi_{xy} \, dxdy}{\int_{\Delta z} \psi_z \, dz} \right| < E \quad . \tag{9}$$

If the neutron flux distribution satisfies the above condition, the leakage coeffieients also satisfy the following condition, as is readily seen from Eqs.(4), (5) and (8):

$$\left| \ell_{xy}^{new} - \ell_{xy}^{old} \right|$$

$$= \frac{\int_{\Delta x} \int_{\Delta y} \theta_{xy} \, dxdy}{\int_{\Delta x} \int_{\Delta y} \psi_{xy} \, dxdy} \cdot \left| \frac{\int_{\Delta z} \psi_z \, dz - \int_{\Delta x} \int_{\Delta y} \psi_{xy} \, dxdy}{\int_{\Delta z} \psi_z \, dz} \right|$$

$$< \frac{\int_{\Delta x} \int_{\Delta y} \theta_{xy} \, dxdy}{\int_{\Delta x} \int_{\Delta y} \psi_{xy} \, dxdy} \cdot E \quad . \tag{10}$$

The suffix *new* indicates the result of this iteration step and *old* the result of the preceding iteration step. After the convergence criterion (9) is satisfied, the neutron sources, θ_{xy} and θ_z, are recalculated by using the converged neutron fluxes ψ_z and ψ_{xy}. The source iteration process is repeated until the source distribution is converged.

The above explanation gives the general feature of the computer code DIFFUSION-ACE, which is a standard of our code series of LIM. Equation (7) can be solved not only by two-dimensional finite difference approximation method but by finite element method (FEM) or by two one-dimensional calculations as follows,

$$\left.\begin{aligned} D\frac{\partial^2}{\partial x^2}\psi_x - (\Sigma_T + \ell_y + \ell_z)\psi_x + \theta_x = 0 \\ D\frac{\partial^2}{\partial y^2}\psi_y - (\Sigma_T + \ell_x + \ell_z)\psi_y + \theta_y = 0 \end{aligned}\right\} \quad (7)'$$

Such solution algorithms are also included in our code series.

3. Development of Computer Codes with LIM

In Fig.3 and TABLE I are shown the relation between computer codes in our system. Program language used is FORTRAN IV for all codes, and check calculations are done by solving the three energy groups neutron diffusion equation. The contents of these codes are summarized as follows:

(1) DIFFUSION-ACE

A) General calculational flow

This code was developed in 1972 and reported in Nucl. Sci. Eng. 58 (1975)[1]. The general calculational flow diagram is shown in Fig.2. As the figure shows, one-dimensional z axial calculation and two-dimensional x-y layer calculation are repeated alternately until the consistency is attained.

B) Characteristics

1) One- and two-dimensional neutron fluxes are obtained by fine mesh finite difference approximation method (FDM).

2) Neutron flux in reflector region can be calculated analytically and combined with the neutron flux in the core region calculated by FDM. The boundary condition for core calculation is determined by this analytic solution, so that the neutron flux in reflector

region is not necessary to be calculated.

3) The boundary condition on the surface of a cruciformed control rod can be given by logarithmic derivative value of the neutron flux.

C) Check calculation

The calculational results by DIFFUSION-ACE are compared in Fig. 4 with those by the CITATION code which adopts the standard fine mesh difference approximation method. The analysis of JPDR-II (BWR) core performance was carried out by this code and one of the results was reported in J. of Nucl. Sci. and Tech[2].

(2) STEADY-ACE

A) General calculational flow

This code was developed in 1973 with the general calculational flow diagram shown in Fig.5. As the figure shows, this code is made of two subroutines,a neutron diffusion calculation routine (DIFFUSION-ACE) and a thermal hydraulics calculation routine (HYDRO-ACE). These two routines are combined by a subroutine CROSS-ACE. The nuclear group constants for the diffusion equation are expressed as functions of the thermal hydraulic condition at each block in the core. The calculation of thermal power distribution is repeated until the consistency is attained between thermo-hydraulics and thermal power distribution.

B) Characteristics

1) The subroutine to deal with the neutron diffusion is the same as DIFFUSION-ACE.

2) The subroutine HYDRO-ACE is for calculating the multi-channel thermo-hydraulics in a BWR core. So STEADY-ACE is a code for analysing the core performance of a BWR.

C) Check calculation

The analysis of JPDR-II (BWR) core phenomena was carried out by this program and the some of the results were reported orally at JNS committee in October, 1977.

(3) STEADY-SHIP

A) General calculational flow

This code was developed in 1977 with the general calculational flow diagram shown in Fig.6. This program is developed by modifying STEADY-ACE for calculating the core performance of a ship reactor. The nuclear group constants are expressed as functions of the thermal hydraulic condition as shown in TABLE II. After the gross neutron flux and thermal hydraulics in the core are obtained, a more precise calculation is carried out for the block appointed by a user with a subroutine LOCAL-FINE. For the neutron flux calculation in this block, boundary conditions arround the selected block are obtained by the gross calculation.

B) Characteristics

1) The subroutine to treat the neutron diffusion is named as DIFFUSION-SHIP. In this program, mesh and channel widths are arbitrarily given by input data to evaluate the effect of heterogeneous structure in a block as shown in Fig.7.

2) The subroutine HYDRO-SHIP is a program to calculate multi-channel thermo-hydraulics in a ship reactor (PWR). So, this code is applied to analysis of a PWR core performance with cruciformed control rods.

3) Taking into consideration of the fine structure in an appointed block as shown in Fig.8, the local peaking factor can be calculated more precisely.

C) Check calculation

1) The power distribution in a ship reactor "MUTSU" is calculated and compared with the experimental data. One example of them is given in Fig.9, which shows a good agreement with the experimental data.

2) The computational results by LOCAL-FINE are compared with those by CITATION. One example is shown in Fig.10, where it is seen the results of gross calculation exert influence on the local power peaking.

(4) WHITE-HORSE

A) General calculational flow

The general calculational flow diagram is shown in Fig.11. This program was developed in 1975 for a PWR type commercial electric power plant. The outline of this code was reported orally at JNS committee in November, 1975.

B) Characteristics

1) In two-dimensional layer calculations, not only the reflector but also the pressure vessel can be taken into consideration. Geometry of the pressure vessel is cylindrical and that of the core is rectangular, so that some regions are treated by R-θ coordinate meshes and the others are by X-Y coordinate meshes.

2) This code is programed for analysing the core performance of a PWR. Therefore, critical search routine by a boron concentration and thermo-hydraulics calculation routine are included also in this code.

C) Check calculation

The check calculation is performed by using the experimental data of the critical boron contents and power distribution in the IKATA-I (PWR)

of Shikoku electric power company, Inc.

(5) FEDM

A) General calculational flow

In this code developed in March, 1978, two-dimensional neutron diffusion layer calculations are performed by using the finite element method (FEM) and one-dimensional channel calculations are done by using the finite difference method. The general calculational flow diagram is almost the same as DIFFUSION-ACE.

B) Characteristics

1) Due to two-dimensional FEM layer calculations, this code is powerful to analyse the core performance with complex geometry.

C) Check calculation

The check calculation is performed by using the experimental data of the high flux reactor (HFR) of Kyoto University. An example for the neutron flux calculation is shown in Fig.12.

(6) JUMP-FDM

A) General calculational flow

In the above mentioned five codes, the layer calculation is performed by the two-dimensional finite difference or finite element method. In this code, however, the calculation is done by repeatedly using the one-dimensional finite difference method. The fundamental treatment of the neutron diffusion equation is shown in Appendix A-I. By this treatment, coarse mesh points are combined with each other by one-dimensional fine mesh finite difference calculations. The general calculational flow is shown in Fig.13.

B) Characteristics

1) Since coarse mesh points are combined with each other by the one-dimensional fine mesh finite difference scheme, the fundermental difference equation easily corresponds with the coarse mesh seven points difference equation. That is, with the exception of the coefficients of flux vector, the calculational method is almost the same as the coarse mesh defference method.

2) As shown in A-I, if the neutron source distribution in a block is assumed to be fixed, the coefficients of flux vector do not change with each iteration step. In this case the one-dimensional calculation between coarse mesh points must be done only once before performing the coarse mesh calculation.

3) As coarse mesh points are combined with one-dimensional fine meshes, the intervals between coarse mesh points can be selected much larger than the node length used in the ordinary node method.

4) Since the three-dimensional block geometry is expressed only three one-dimensional lines, it is not appropriate to perform the core calculation with complex geometry.

Acknowledgment

This code series has been developed by many engineers. The author wishes to express his appreciation to them for permitting him to write this report. Thanks are also rendered to T. Asaoka and S. Matsuura of the Japan Atomic Energy Research Institute for their valuable discussions.

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(1) NAITO, Y., et al.: A leakage iterative method for solving the three-dimensional neutron diffusion equation, Nucl. Sci. Eng., 58, 182 (1975).

(2) NAITO, Y.,: Moderator temperature coefficient in BWR core, J. of Nucl. Sci. and Tech., 14, 826 (1977).

Appendix A-1

Fundamental Treatment of Neutron Diffision Equation in JUMP-FDM

A reactor is divided into many blocks which are called coarse meshes. If intervals between coarse mesh points are too wide to express neutron diffusion by difference equation, the intervals must be divided into finer meshes as shown in Fig.A1. The neutron leakage from a block is obtained by using fine mesh flux ψ_B arround the block boundary.

The one-dimensional fine mesh difference equation between coarse mesh points I and I + 1, is expressed as follows:

$$\begin{bmatrix} r_1 & -\ell_1 & & & & \\ -\ell_1 & r_2 & -\ell_2 & & & \\ & \ddots & \ddots & \ddots & & \\ & & -\ell_{p-1} & r_p & -\ell_p & \\ & & & \ddots & \ddots & \ddots \\ & & & & -\ell_{k-1} & r_k \end{bmatrix} \begin{bmatrix} \psi_1 \\ \psi_2 \\ \vdots \\ \psi_p \\ \vdots \\ \psi_k \end{bmatrix} = \begin{bmatrix} S_1 + \ell_0 \Phi^I \\ S_2 \\ \vdots \\ S_p \\ \vdots \\ S_k + \ell_k \Phi^{I+1} \end{bmatrix} \quad (1)$$

where

$$\left.\begin{aligned} \ell_p &\equiv 2 \cdot \frac{D_p \cdot D_{p+1}}{D_p \cdot \Delta X_{p+1} + D_{p+1} \cdot \Delta X_p} \quad , \\ \ell_o &\equiv 2 \cdot \frac{D_1}{\Delta X_1} \quad , \qquad \ell_k \equiv 2 \cdot \frac{D_k}{\Delta X_k} \quad , \\ r_p &\equiv (\Sigma_{ap} + D_p B^2) * \Delta X_p + \ell_{p-1} + \ell_p \quad . \end{aligned}\right\} \quad (2)$$

In the above expressions, Φ^I is the neutron flux at the center point in a block I and ΔX_p is a fine mesh width.

The neutron source vector in Eq.(1) is divided into three terms as follows:

$$\vec{S} = \begin{bmatrix} S_1' \\ S_2' \\ \vdots \\ S_p' \\ \vdots \\ S_k' \end{bmatrix} S^I_{I+1} + \begin{bmatrix} \ell_o \\ 0 \\ \vdots \\ 0 \\ \vdots \\ 0 \end{bmatrix} \Phi^I + \begin{bmatrix} 0 \\ 0 \\ \vdots \\ 0 \\ \vdots \\ \ell_k \end{bmatrix} \Phi^{I+1}$$

$$= \vec{S}^E \cdot S^I_{I+1} + \vec{S}^I \cdot \Phi^I + \vec{S}^{I+1} \Phi^{I+1} \quad . \tag{3}$$

Now, three vectors $\vec{\psi}^E$, $\vec{\psi}^I$ and $\vec{\psi}^{I+1}$ are defined as solutions of the following equations:

$$M\vec{\psi}^E = \vec{S}^E \quad , \quad M\vec{\psi}^I = \vec{S}^I \quad , \quad M\vec{\psi}^{I+1} = \vec{S}^{I+1} \quad , \tag{4}$$

where M is the matrix shown in Eq.(1). With using above three vectors, the one-dimensional fine mesh neutron flux $\vec{\psi}$ is expressed as follows:

$$\vec{\psi} = \vec{\psi}^E \cdot S^I_{I+1} + \vec{\psi}^I \cdot \Phi^I + \vec{\psi}^{I+1} \cdot \Phi^{I+1} \quad . \tag{5}$$

The neutron current between blocks is expressed with the fine mesh neutron flux near the boundary,

$$J(I|I+1) = \ell_B(I|I+1)*\{\psi_B(I|I+1) - \psi_{B+1}(I|I+1)\}*A(I|I+1) \quad , \qquad (6)$$

where $A(I|I+1)$ is the boundary area between the blocks I and I+1. The neutron current J can be divided into three terms, J^E, J^I and J^{I+1}:

$$\left.\begin{aligned} J^E(I|I+1) &= \ell_B(I|I+1)*\{\psi_B{}^E(I|I+1) - \psi^E_{B+1}(I|I+1)\}*A(I|I+1) \\ J^I(I|I+1) &= \ell_B(I|I+1)*\{\psi_B{}^I(I|I+1) - \psi^I_{B+1}(I|I+1)\}*A(I|I+1) \\ J^{I+1}(I|I+1) &= \ell_B(I|I+1)*\{\psi^{I+1}_B(I|I+1) - \psi_{B+1}^{I+1}(I|I+1)\}*A(I|I+1) \\ J(I|I+1) &= J^E(I|I+1)S^I_{I+1} + J^I(I|I+1)\Phi^I + J^{I+1}(I|I+1)\Phi^{I+1} \quad . \end{aligned}\right\} \qquad (7)$$

In the same manner as for the neutron current, the neutron absorption in a block is expressed as follows:

$$\left.\begin{aligned} P^E(I|I+1) &= \underset{p=1,B}{\text{Sum}}\ \psi_p{}^E \cdot \Sigma_{ap}\, V_p \\ P^I(I|I+1) &= \underset{p=1,B}{\text{Sum}}\ \psi_p{}^I \cdot \Sigma_{ap}\, V_p \\ P^{I+1}(I|I+1) &= \underset{p=1,B}{\text{Sum}}\ \psi_p{}^{I+1} \cdot \Sigma_{ap}\, V_p \end{aligned}\right\} \quad , \qquad (8)$$

$$P(I|I+1) = P^E(I|I+1) \cdot S^I_{I+1} + P^I(I|I+1)\Phi^I + P^{I+1}(I|I+1)\Phi^{I+1} \quad . \qquad (9)$$

With the above notations, the neutron balance equation in the block I is expressed as

$$\{J^{E,I+1} + J^{E,J-1} + J^{E,J+1} + J^{E,J-1} + J^{E,k+1} + J^{E,k-1}\}S(I,J,k\ ;\ g)$$

$$+ \{P^{E,I+1} + P^{E,I-1} + P^{E,J+1} + P^{E,J-1} + P^{E,k+1} + P^{E,k-1}\}S(I,J,k\ ;\ g)$$

$$+ [\{J^{I,I+1} + J^{I,I-1} + J^{J,J+1} + J^{J,J-1} + J^{k,k+1} + J^{k,k-1}\}$$

$$+ \{P^{I,I+1} + P^{I,I-1} + P^{J,J+1} + P^{J,J-1} + P^{k,k+1} + P^{k,k-1}\}]\phi(I,J,k\ ;\ g)$$

$$+ \{J^{I+1,I+1} + P^{I+1,I+1}\}*\phi(I+1,J,k\ ;\ g)$$

$$+ \{J^{I-1,I-1} + P^{I-1,I-1}\}*\phi(I-1,J,k\ ;\ g)$$

$$+ \{J^{J+1,J+1} + P^{J+1,J+1}\}*\phi(I,J+1,k\ ;\ g)$$

$$+ \{J^{J-1,J-1} + P^{J-1,J-1}\}*\phi(I,J-1,k\ ;\ g)$$

$$+ \{J^{k+1,k+1} + P^{k+1,k+1}\}*\phi(I,J,k+1\ ;\ g)$$

$$+ \{J^{k-1,k-1} + P^{k-1,k-1}\}*\phi(I,J,k-1\ ;\ g)$$

$$= \text{Source}. \qquad (10)$$

In the above expression, $J^{E}(I|I+1)$ is written as $J^{E,I+1}$ and the other terms are expressed in the same manner. Equation (10) is a seven point coarse mesh equation. The coefficients of the neutron flux depend only on the calculational system but do not change with iteration steps. These values are obtained by solving only once Eq.(4). The values of J^{E} and P^{E} depend on the source distribution in a block, and they must be obtained at every outer iteration steps. If the source distribution in a block is assumed to be fixed, however, also these values do not change with iteration steps. This suggests the possibility of developing a high speed calculation method.

TABLE I Summary of a code series by LIM

NAME	Purpose	Property	Author
DIFFUSION-ACE	Three dimensional neutron diffusion	JAERI	Y. Naito, M. Maekawa, K. Shibuya
STEADY-ACE	BWR core performance for JPDR	JAERI	Y. Naito, M. Maekawa, K. Abe
WHITE-HORSE	PWR core performance for IKATA-I	Shikoku Electric Power Company INC.	I. Tsujimoto, K. Kawanishi et al.
STEADY-SHIP	Ship reactor core performance for MUTSU	Japan Nuclear Ship Development Agency	Y. Naito, M. Itagaki, Y. Tokuno et al.
FEDM	High flux reactor core performance for HFR of Kyoto University	Kyoto University	Y. Naito, M. Hayashi, S. Tsuruta et al.
JUMP-FDM	Three dimensional neutron diffusion for LWR	JAERI and Central Research Institute of Electric Power Industry	Y. Naito, T. Matsumura

TABLE II The nuclear group constants fitting formulas

$$T_f = e_0 + e_1*P + e_2*P^2 + T_m$$

$$T_m = T_m - T_m^{STD}$$

$$\sqrt{T_f} = \sqrt{T_f} - \sqrt{T_f^{STD}}$$

$$1/D = d_0 + d_1*T_m + d_2*T_m^2$$

$$\Sigma_R = r_0 + r_1*T_m + r_2*T_m^2$$

$$(\delta\Sigma_a)_{xe} = \frac{P}{b_0 + b_1*P}$$

$$(\delta\Sigma_a)_{DOP} = a_0 + a_1*\sqrt{T_f} + a_2*(\sqrt{T_f})^2$$

$$\Sigma_a = f_0 + f_1*T_m + f_2*T_m^2 + F*\{a_0 + a_1*\sqrt{T_f} + a_2*(\sqrt{T_f})^2\} + G*\{P / (b_0 + b_1*P)\}$$

$$\nu\Sigma_f = h_0 + h_1*T_m + h_2*T_m^2 + G*\{P / (c_0 + c_1*P)\}$$

$$G = \delta\ (g - 3)\ , \qquad F = \delta\ (g - 2)$$

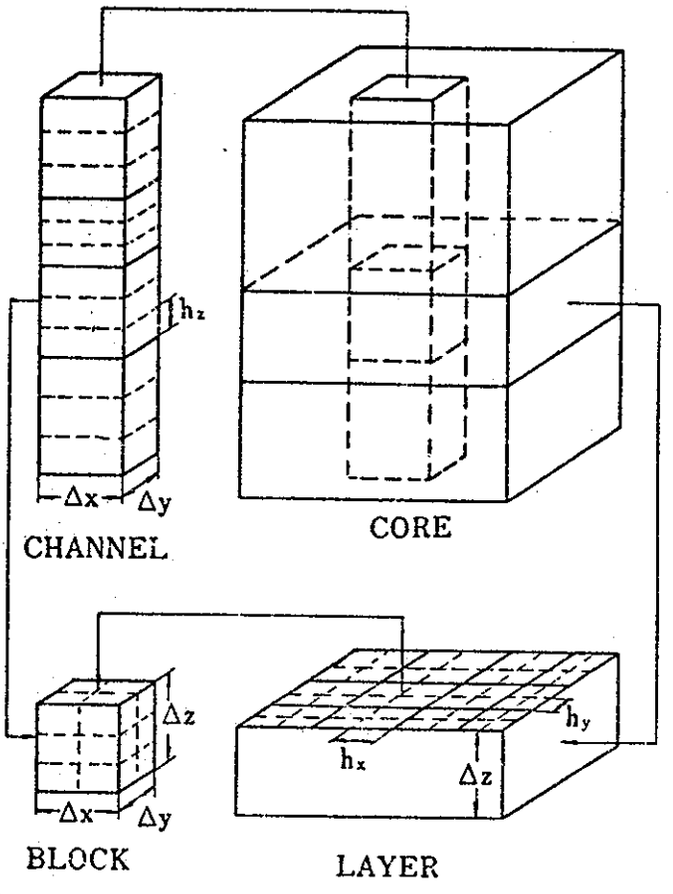


Fig. 1 Configuration of Channels, Layers, and Blocks.

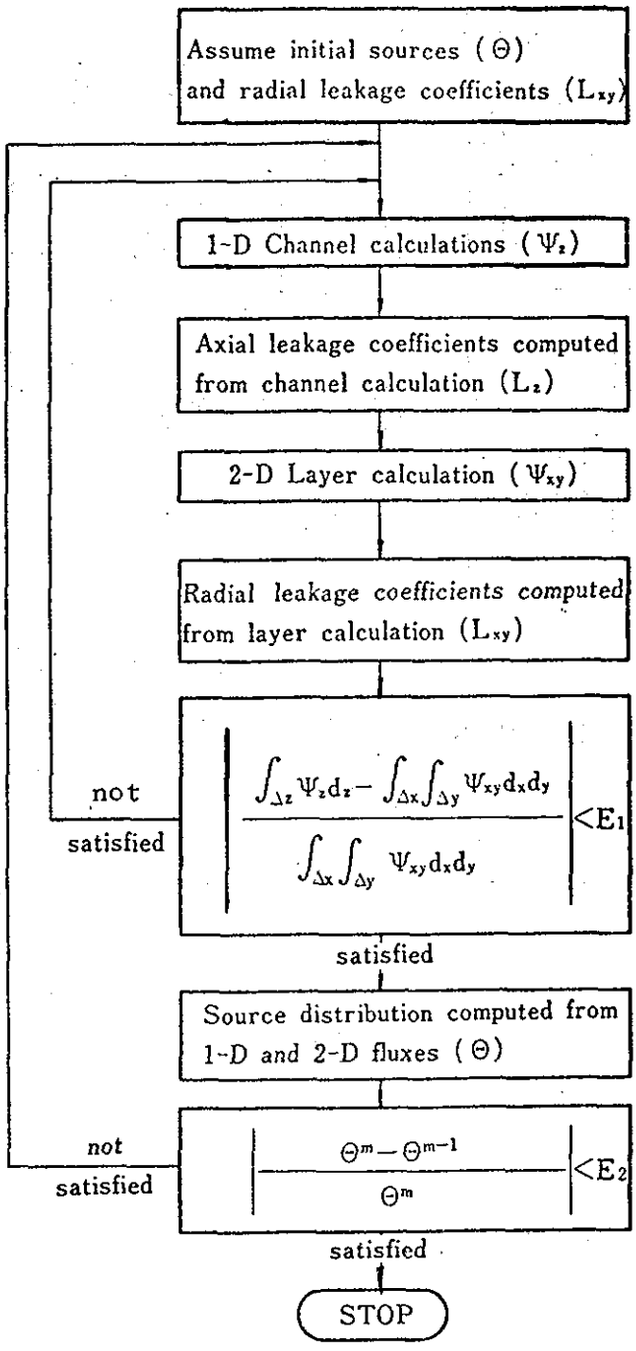


Fig. 2 Schematic diagram of the Leakage Iterative Method.

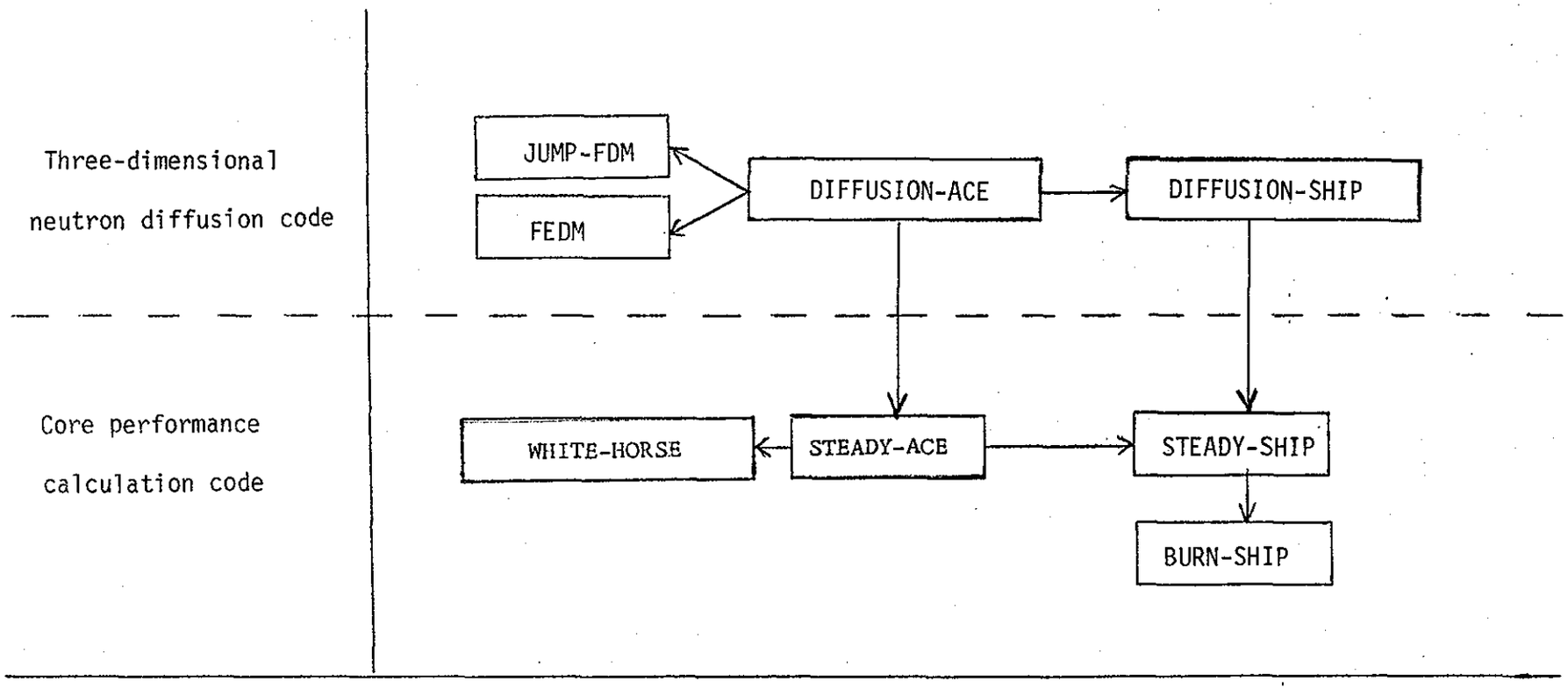


Fig. 3 A computer code series by leakage iterative method

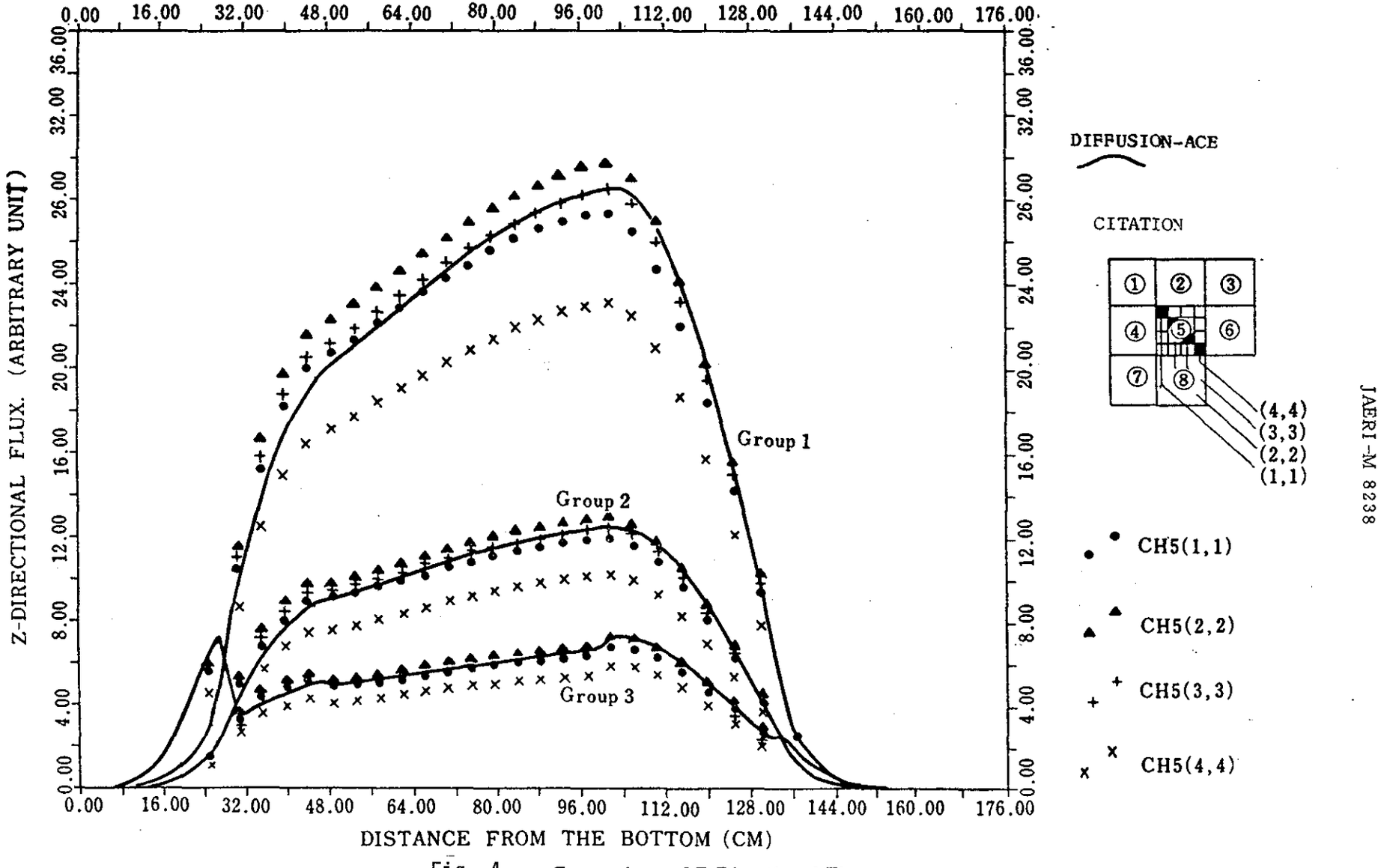


Fig. 4 Comparison of Z-Directional Flux.

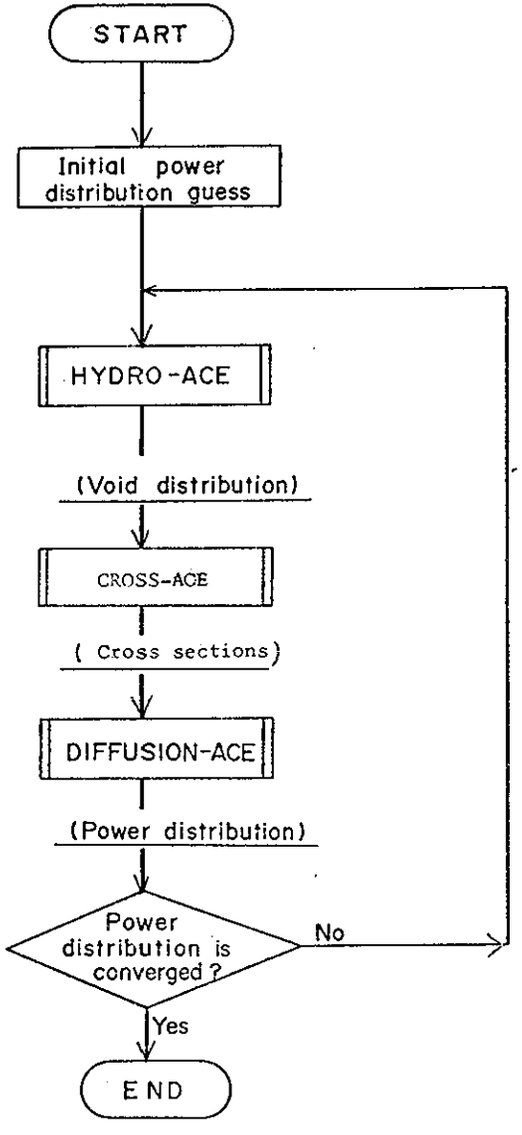


Fig. 5 Schematic diagram of STEADY-ACE

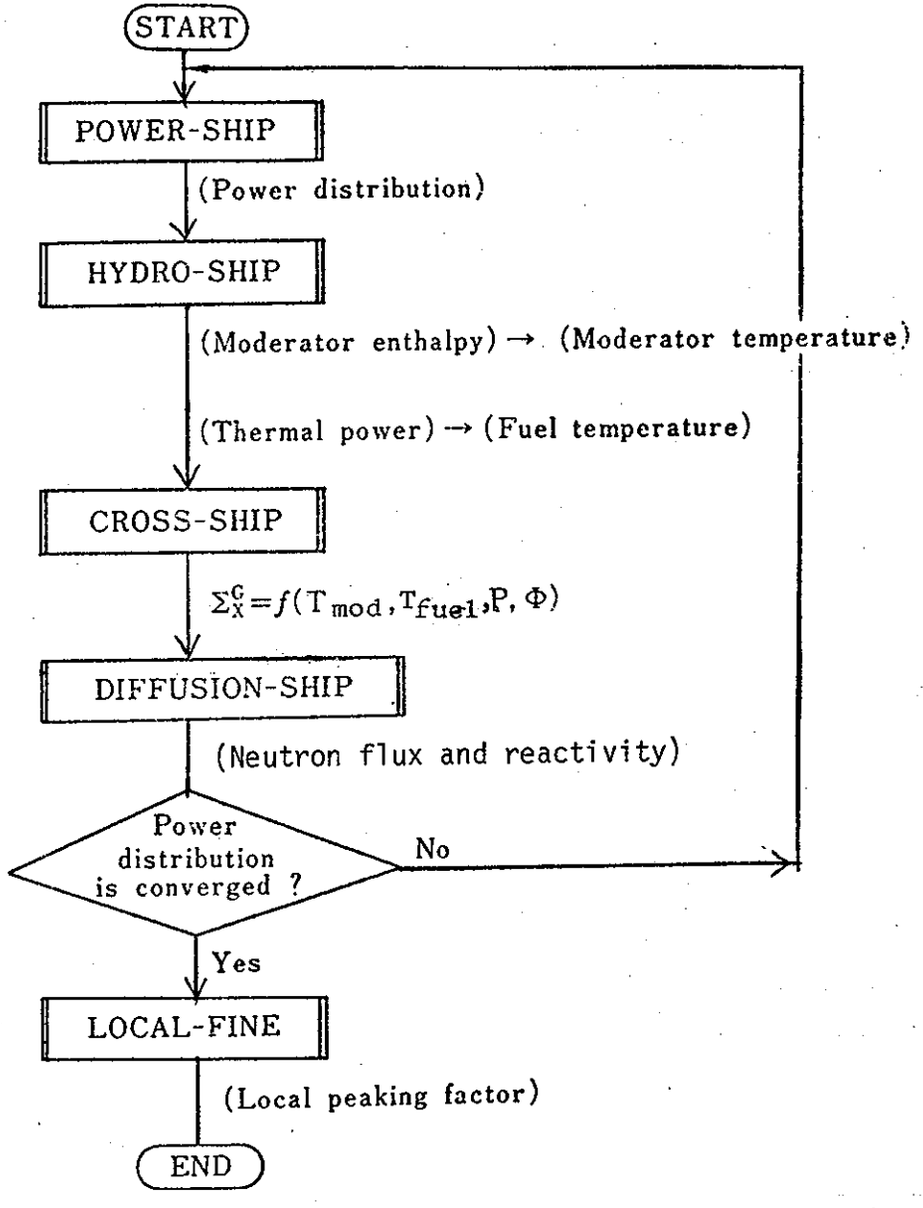
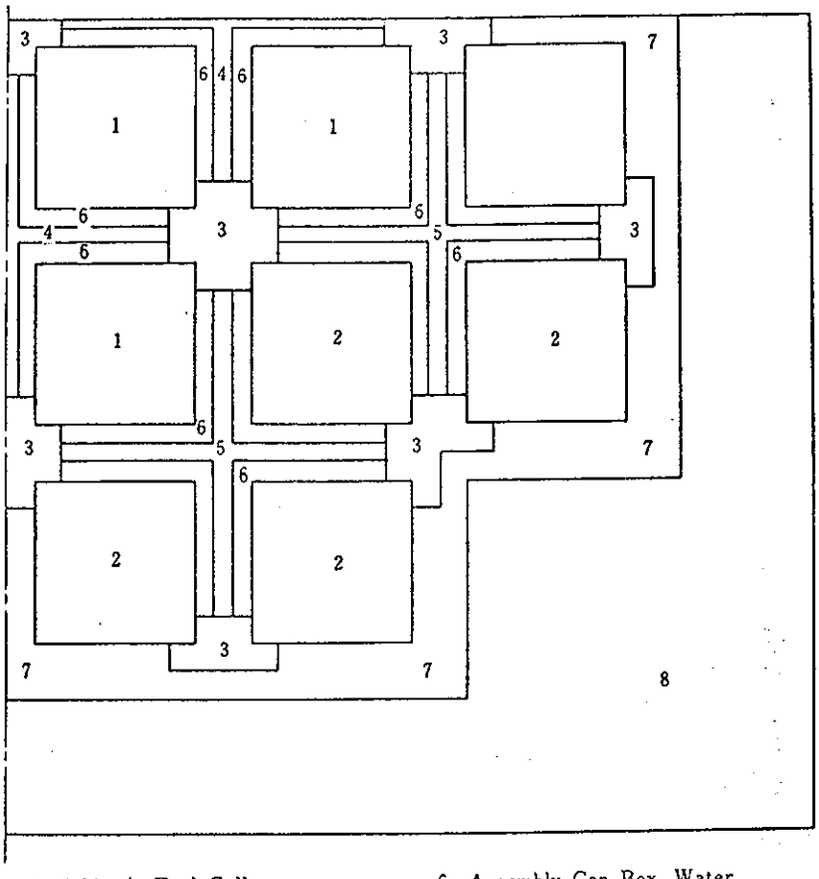


Fig. 6 Schematic diagram of the STEADY-SHIP calculation



1. 3.24 w/o Fuel Cell and BPR Region.
2. 4.44 w/o Fuel Cell and BPR Region.
3. Zr-2 Filler, Assembly Can Box and Water Homogenized Region.
4. G1, G2 Group Control Rod (C/R) or Zr-2 Follower Region.
5. G3, G4 Group C/R or Zr-2 Follower Region.
6. Assembly Can Box, Water and C/R Clad Homogenized Region.
7. SUS Buffle.
8. Water Reflector, Core Tank and Thermal Shroud Homogenized Region.

Fig. 7 The Full Core Three Dimensional Model. (Horizontal Section)

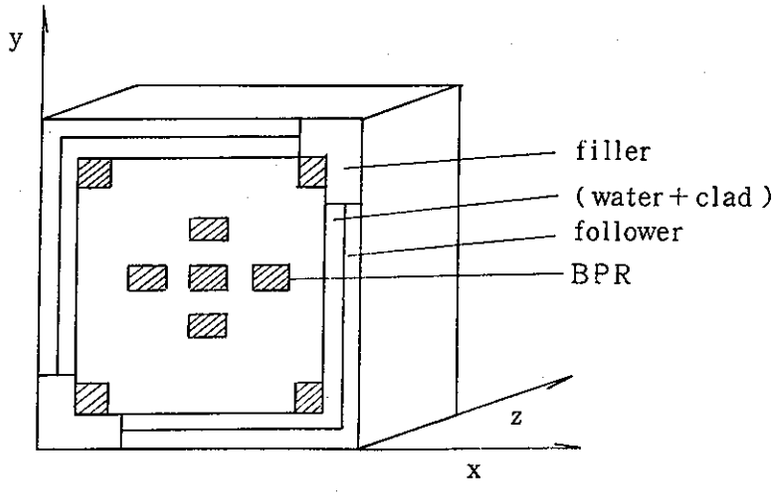


Fig. 8 Calculational model for LOCAL-FINE

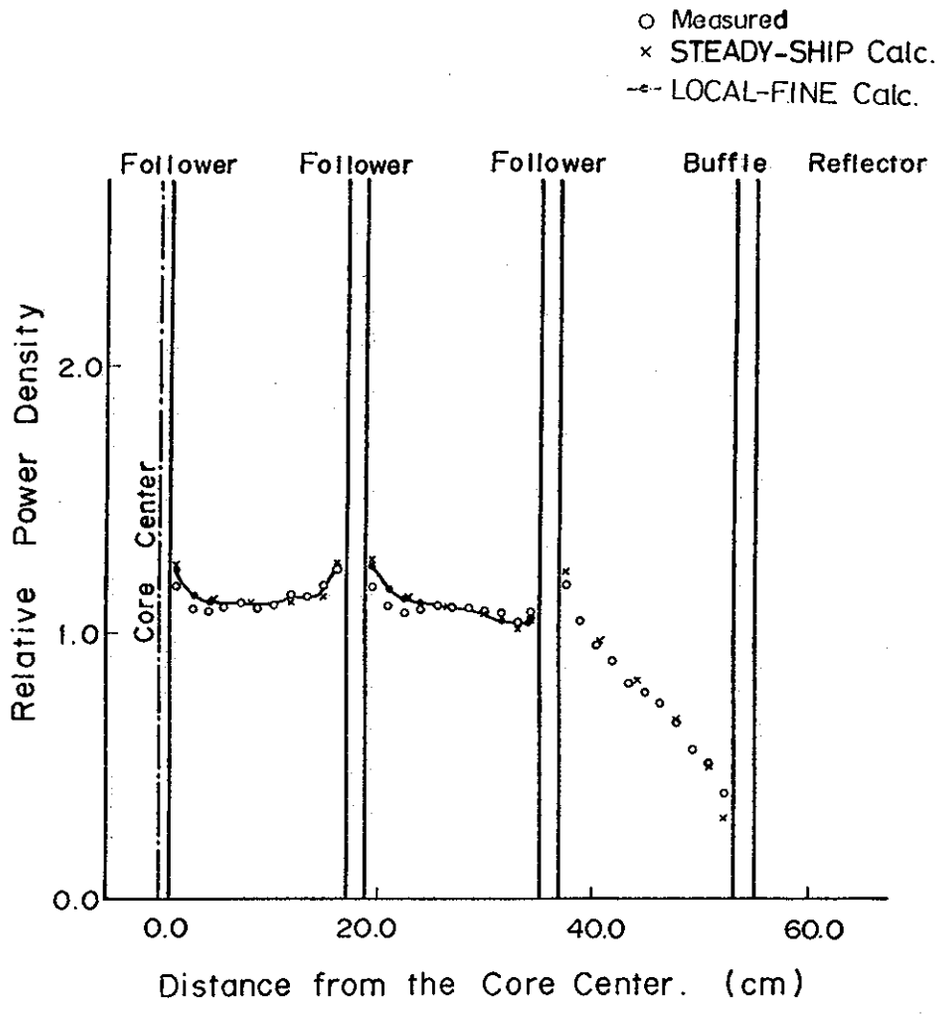
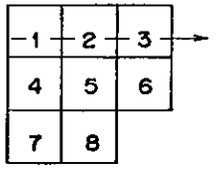


Fig. 9 Relative power distribution in the critical core of the ship reactor MUTSU



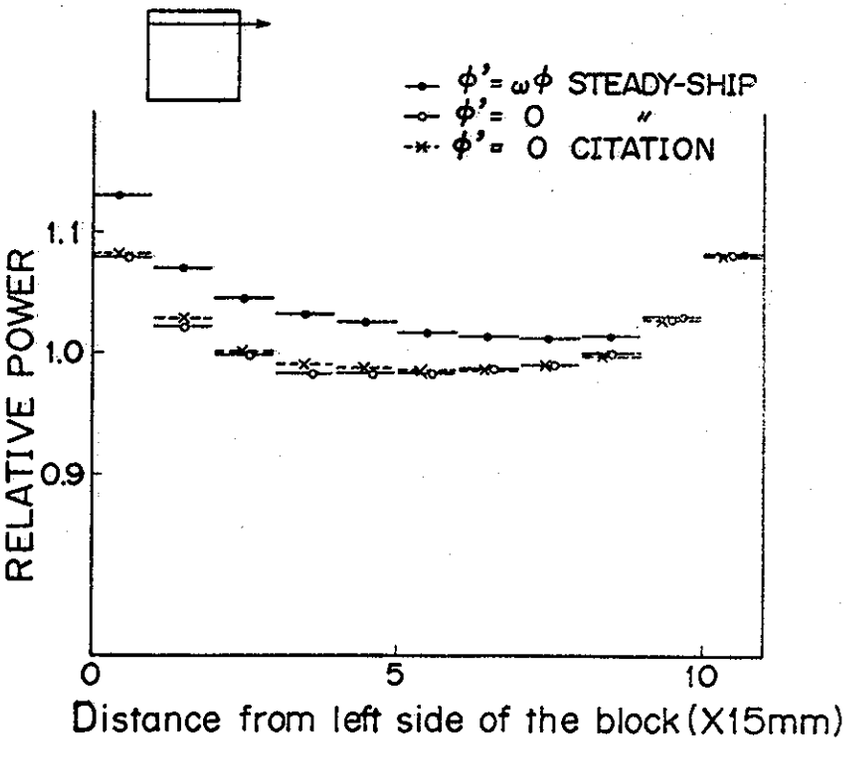


Fig. 10 Power distribution in an assembly calculated by the LOCAL-FINE code

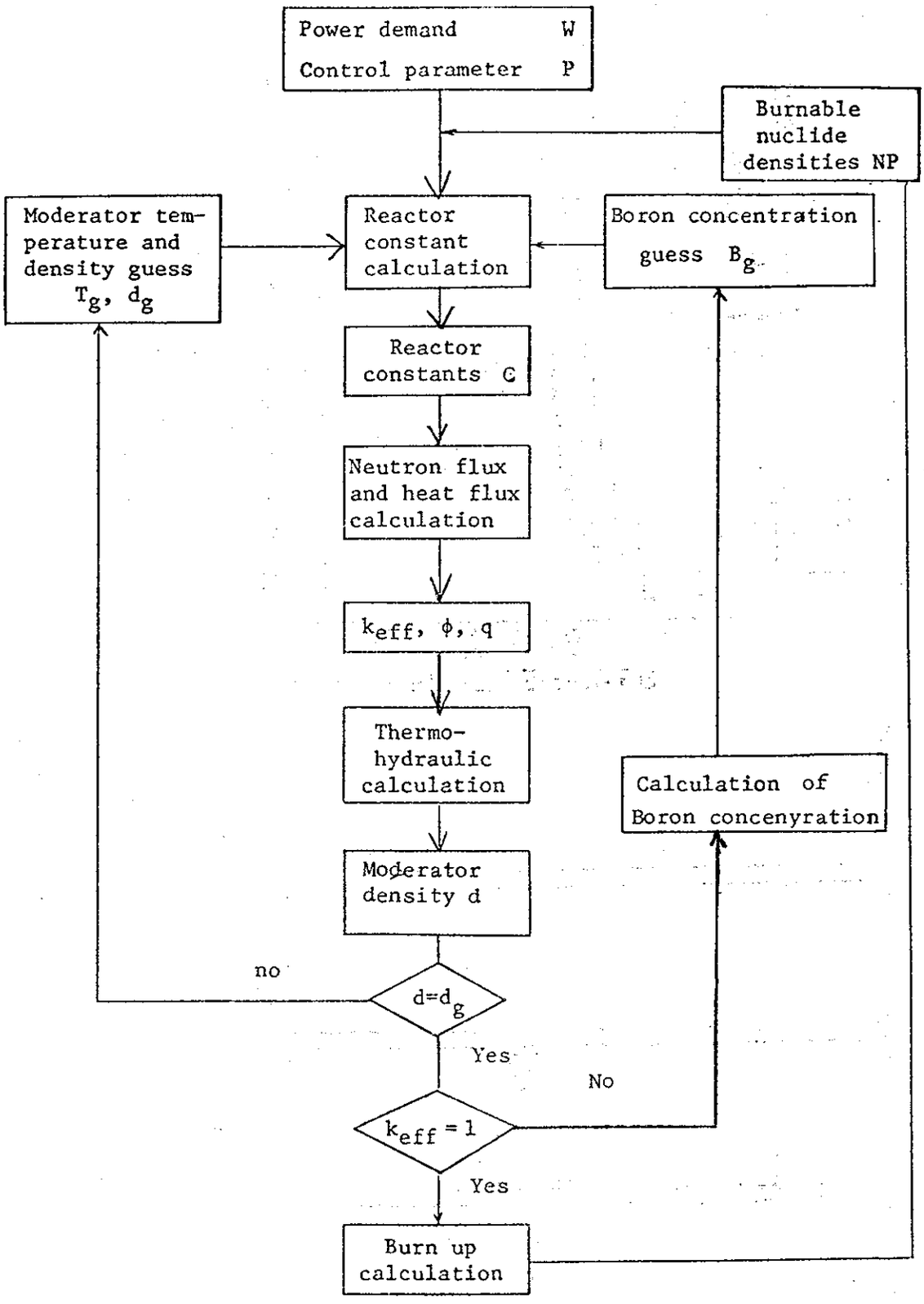


Fig. 11 Schematic diagram of WHITE-HORSE

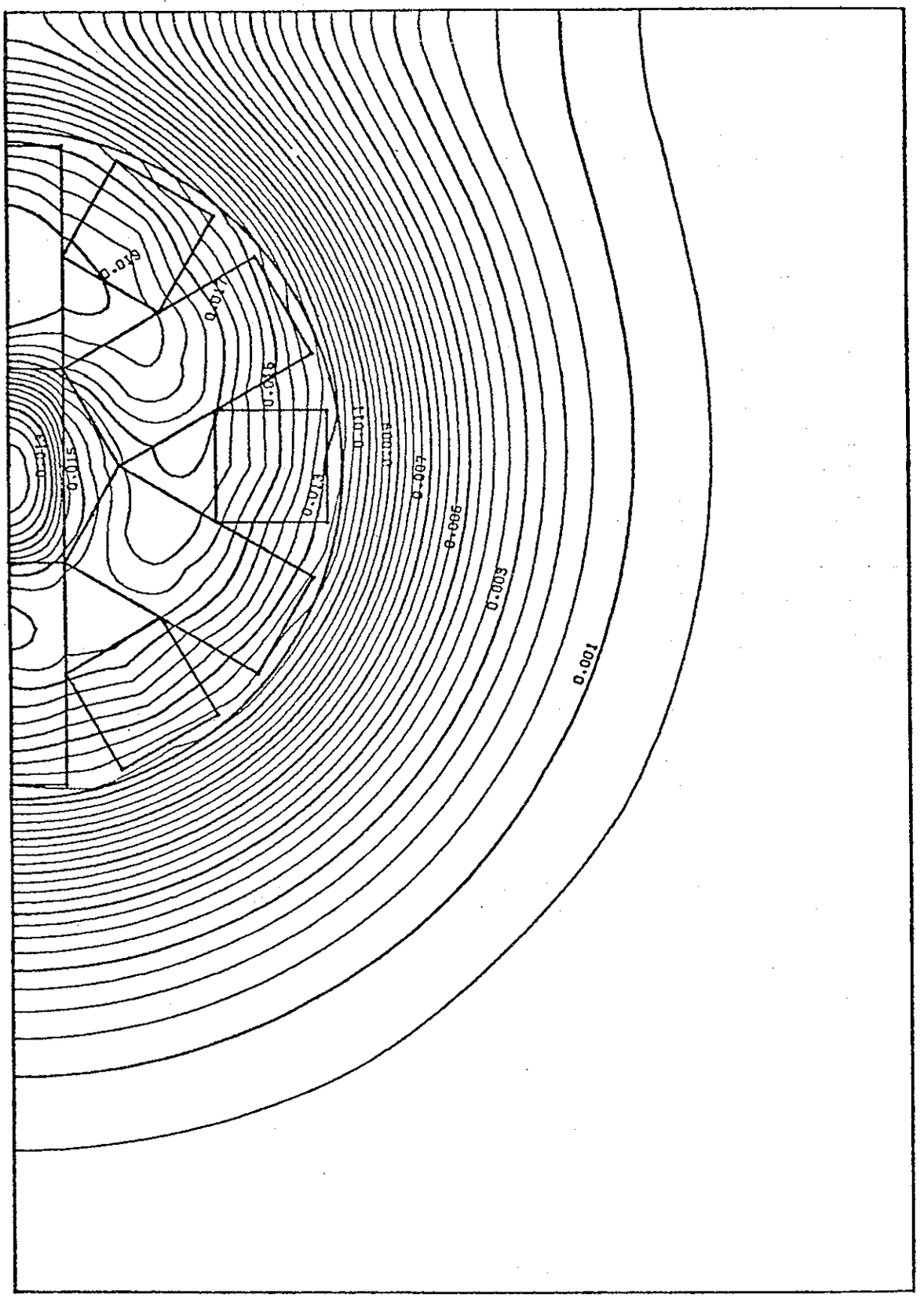


Fig. 12 Neutron flux distribution in HFR calculated by FEDM

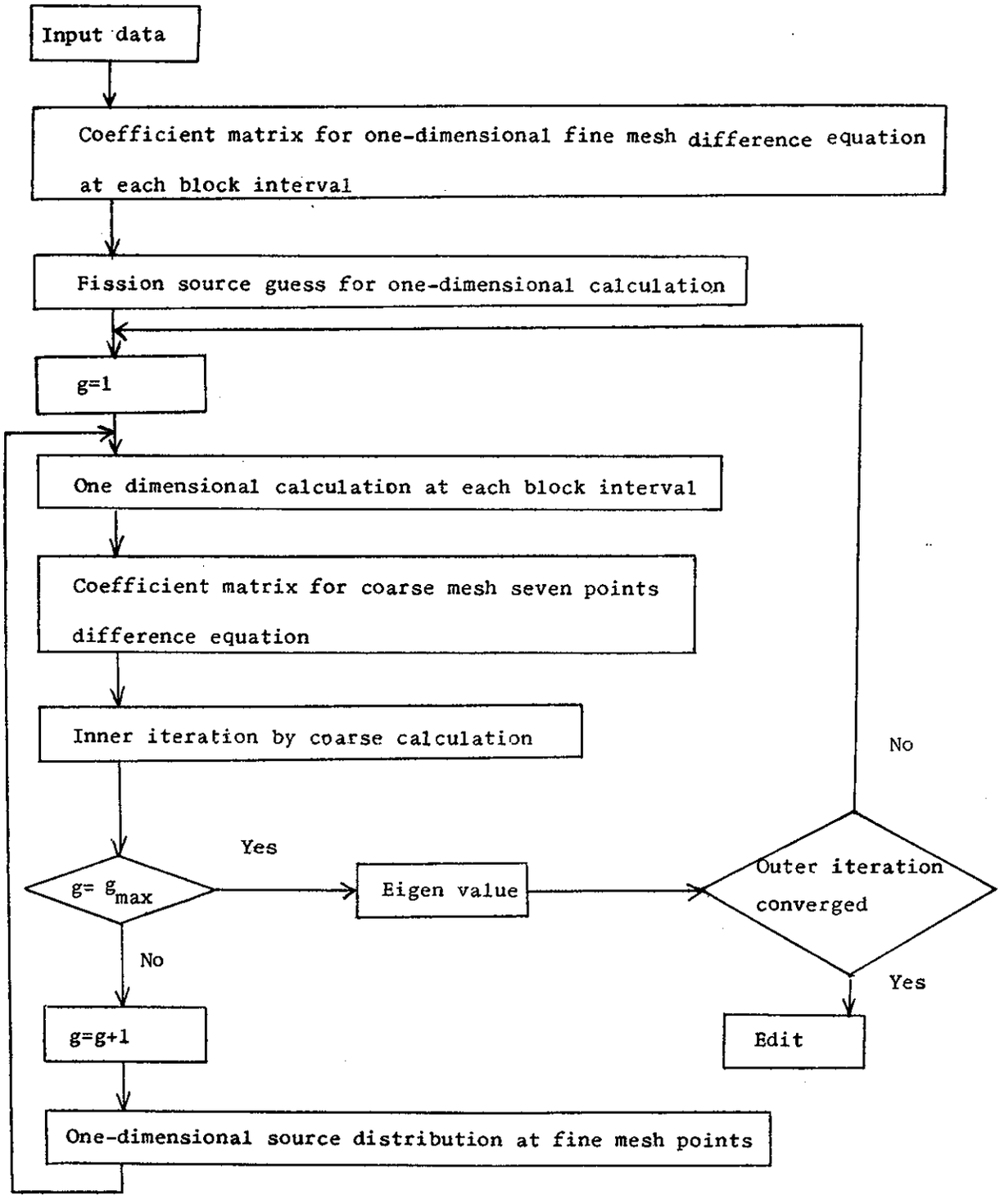


Fig.13 Schematic diagram of JUMP-FDM

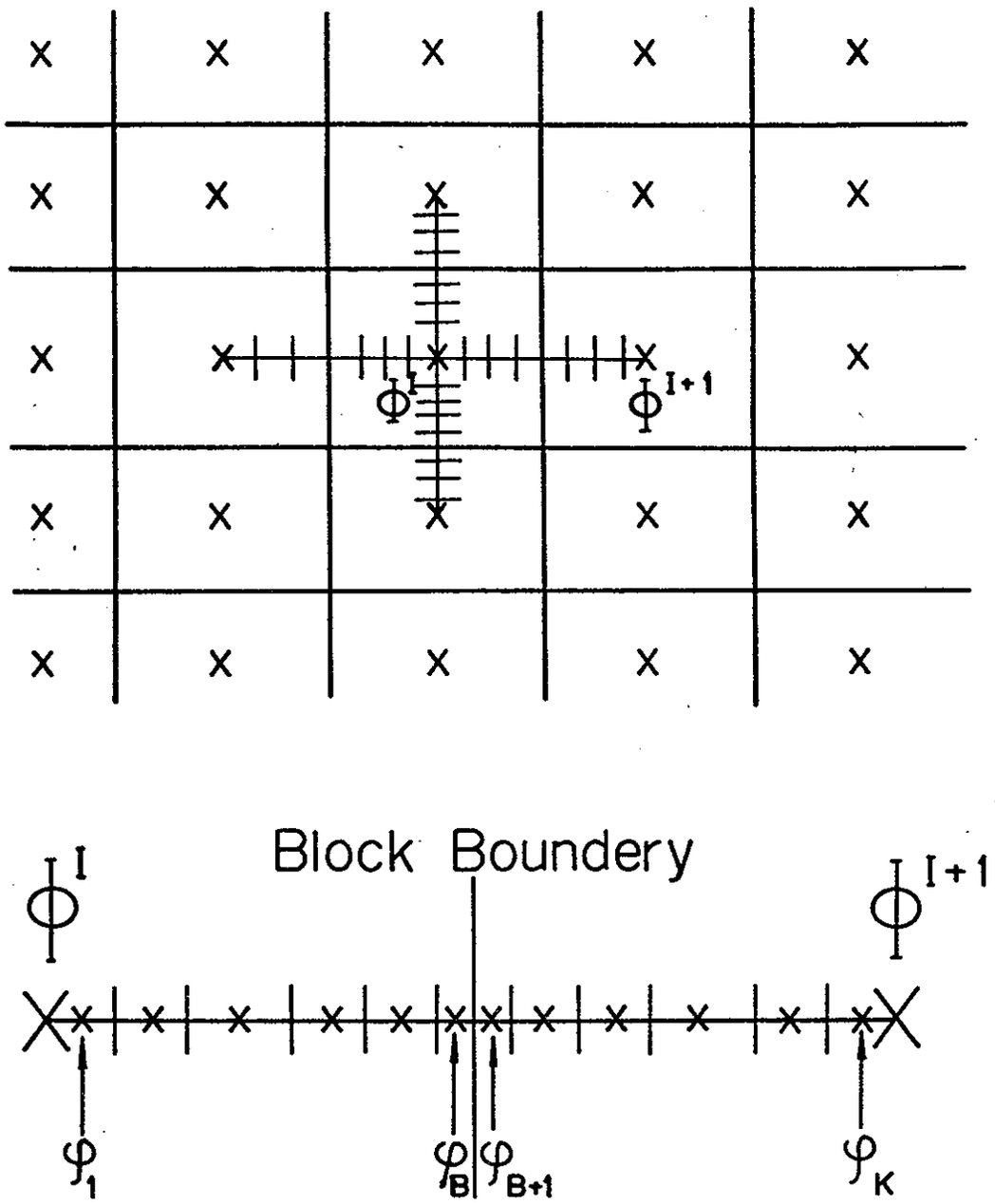


Fig.A1 Mesh description in JUMP-FDM