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CALCULATION OF NEUTRON FLUX DISTRIBUTION IN THE HEXAGONAL FUEL ASSEMBLY AND THE PWR ASSEMBLY CODE TPFAP-HEX

Zhang Ying[(1)] Xie Zhongsheng[(2)]

(1) Northwest Institute of Nuclear Technology ,Xi'an,710024,P.R.China

Chen_nint@263.net.ifeng@nint.ac.cn

Chen Wei[(1)] Zhang Shaohong[(2)] Chen Lixin[(1)] Wang Daohua[(1)]

(2) Xi'an Jiaotong University Nuclear Energy Department,Xi'an,P.R.China

ABSTRACT

This paper develops a set of software package having engineering practical value to calculate PWR hexagonal fuel assembly homogenous parameters. It introduces the calculation method of neutron flux distribution in PWR hexagonal fuel assembly, solving the neutron integral transport equation in hexagonal geometry by the method combining the transmission probability method (TPM) with respond matrix method (RMM) and at the same time a rigorous and accurate method to deal with the water gap surrounding assembly is adopted. Base on above model the program TPHEX is encoded to calculate the neutron flux distribution in PWR hexagonal fuel assembly.

On the other hand, the PWR quadrate assembly software package TPFAP has been modified, the original function of TPFAP is kept, the program TPHEX is embedded in TPFAP, thus the software package TPFAP—HEX for PWR hexagonal fuel assembly homogenous calculation is formed. Through calculating the practical VVER—1000 hexagonal assembly, it proves that the software package has satisfactory calculation precision and can be used in engineering practices.

1. INTRODUCTION

The calculation of the homogenous parameters of reactor fuel assembly is an important step for reactor physical calculation and core design. And the calculation consists of three parts:

1. The cell homogenous parameters calculation,
2. The flux distribution in fuel assembly calculation,
3. The assembly burnup calculation.

In this paper a numerical method to calculate the flux distribution in hexagonal fuel assembly by solving multi-group neutron integral transport equation in 2D hexagonal

geometry is studied. This is a new method because it combines the transmission probability method (TPM) with the respond matrix method (RMM), and at the same time a rigorous and accurate method to deal with the water gap surrounding the assembly is adopted. Based on above mode the program TPHEX is encoded and embedded in the PWR quadrant assembly code TPFAP to form the software package TPFAP-HEX for hexagonal fuel assembly homogenization calculation. Through calculating the practical VVER—1000 hexagonal assembly, it proved that the software package has satisfactory calculation precision and can be used in engineering practices.

2. THE NEUTRON FLUX DISTRIBUTION CALCULATION IN THE HEXAGONAL ASSEMBLY

2.1 BASIC EQUATION

Due to the strong heterogeneous of the fuel assembly, in order to obtain accurate flux distribution in fuel assembly, it should solve neutron transport equation in fuel assembly. At first the hexagonal fuel assembly system can be divided into several homogenous hexagonal regions (usually one cell is divided as one region) and pentagon and ladder shape water gap regions (see Fig.1). And the multi-group neutron integral transport equation can be establish--ed in each region as equation (1),

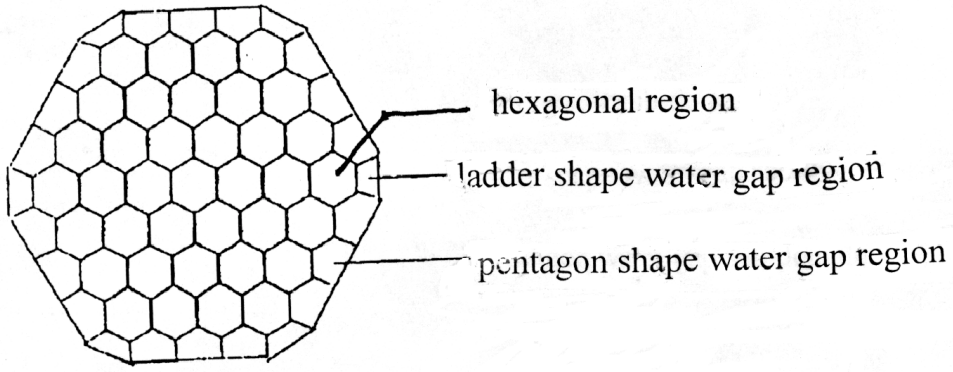


Fig1. Hexagonal assembly

$$\Phi_g(\vec{r}_s,\vec{\Omega}) = \int_0^{l_0} Q_g(\vec{r}',\vec{\Omega})\cdot\exp(-\Sigma_t l/\cos\theta)\frac{dl}{\cos\theta} + \Phi_g(\vec{r}_{s'},\vec{\Omega})\cdot\exp(-\Sigma_t l_0/\cos\theta) \qquad (1)$$

Where suffix g is neutron energy group number, $\Phi_g(\vec{r}_s,\vec{\Omega})$ $\Phi_g(\vec{r}_{s'},\vec{\Omega})$ are the neutron flux at position r_s and $r_{s'}$ following Ω direction respectively, $Q_g(\vec{r}',\vec{\Omega})$ is the neutron source at position r' following Ω direction, it can be written as equation (2) when neutron scattering source and independent outer neutron source are isotropic,

$$Q_g(\vec{r},\vec{\Omega}) = \frac{1}{4\pi}\sum_{g'=1}^{G}\left(\Sigma_{g'\to g} + \chi_g\frac{(\nu\Sigma_f)_{g'}}{k_{eff}}\right)\cdot\Phi_{g'}(\vec{r}) + \frac{1}{4\pi}S_g(\vec{r}) \qquad (2)$$

where $S_g(\vec{r})$ is the outer neutron source. When $S_g(\vec{r})$=0, equation (1) and (2) form an eigenvalue problem.

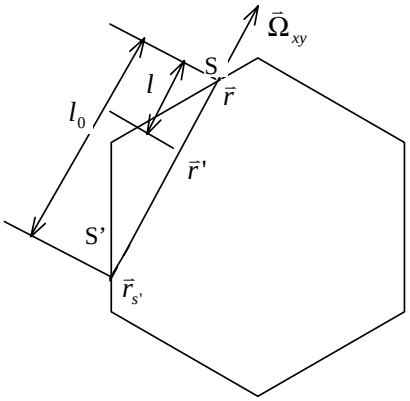


Fig2. Illustration of variables in equation (1)

Moreover we can establish the neutron balance equation in each region as equation (3),

$$\overline{\Phi}_{i,g} = \frac{\overline{Q}_{i,g}}{\Sigma_{t,i,g}} - \frac{J_{i,g}}{\Sigma_{t,i,g} \cdot V_i} \tag{3}$$

where suffix i is the region number, $\overline{\Phi}_{i,g}$ is the average neutron flux in the ith region, $\overline{Q}_{i,g}$ is the average neutron source in the ith region(including fission source, scatter source and outer source), $J_{i,g}$ is the total outgoing neutron current of the ith region and V_i is the volume of the ith region.

Now, equation (1),(2),(3) form the basic equations group to calculate the neutron flux distribution in hexagonal fuel assembly.

2.2 BASIC APPROXIMATION

2.2.1 APPROXIMATION FOR THE SPATIAL DISTRIBUTION OF NEUTRON SOURCE AND FLUX IN REGIONS

The approximation for the spatial distribution of neutron source and flux in regions includes two parts: the approximation for hexagonal region and the approximation for water gap region.

For hexagonal region, in order to reach higher calculation precision, the interior neutron source and flux within region are assumed to be quadratic in spatial distribution as equations (4),(5).

$$Q(x,y,\Omega) = \frac{1}{4\pi}\left(Q_0 + Q_x \cdot \frac{x}{a} + Q_y \cdot \frac{y}{a} + Q_{xy} \cdot \frac{xy}{a^2} + Q_{x^2} \cdot \left(\frac{x^2}{a^2} - \frac{5}{6}\right) + Q_{y^2} \cdot \left(\frac{y^2}{a^2} - \frac{5}{6}\right)\right) \tag{4}$$

$$\Phi(x,y,\Omega) = \frac{1}{4\pi}\left(\Phi_0 + \Phi_x \cdot \frac{x}{a} + \Phi_y \cdot \frac{y}{a} + \Phi_{xy} \cdot \frac{xy}{a^2} + \Phi_{x^2} \cdot \left(\frac{x^2}{a^2} - \frac{5}{6}\right) + \Phi_{y^2} \cdot \left(\frac{y^2}{a^2} - \frac{5}{6}\right)\right) \tag{5}$$

where Φ_v¡¢$Q_v (v = 0, x, y, xy, x^2, y^2)$ are expansion coefficients for neutron flux and source, a

is the half length of the hexagonal region side.

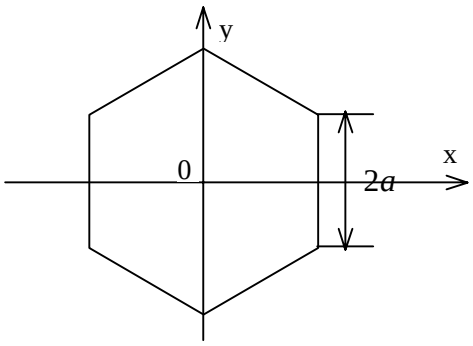


Fig3. The coordinate system for equation (4) (5)

For water gap region including pentagon and ladder shape region, because their shapes are complicated and their geometrical sizes are relatively small, the flat approximation is adopted for the neutron flux and source spatial distribution as equation (6) and (7).

$$Q_{i,g}(r,\Omega)=\frac{\overline{Q_{i,g}}}{4\pi} \tag{6}$$

$$\Phi_{i,g}(r,\Omega)=\frac{\overline{\Phi_{i,g}}}{4\pi} \tag{7}$$

where $\overline{Q}_{i,g}$ and $\overline{\Phi}_{i,g}$ are average neutron source and flux in the ith water gap region respectively.

2.2.2 APPROXIMATION FOR THE ANGULAR NEUTRON FLUX AT THE REGION SURFACE

The approximation for the angular flux at the region surface includes two parts too: the approximation for hexagonal region surface and the approximation for water gap region surface.

For hexagonal region surface, the flat approximate for spatial distribution and the simplified $6P_1$ approximate for angular distribution for neutron flux at region surface is adopted. As illustrated in figure 4, let the outer normal $\vec{n}_k^+$ of the surface k be the base axis,

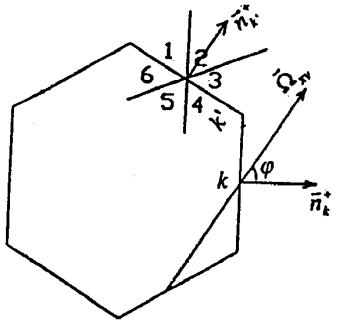


Fig4. The quadrant division on hexagonal region surface

the angular between the project of neutron flight direction $\vec{\Omega}$ on the X-Y plane $\vec{\Omega}_{xy}$ and $\vec{n}_k^+$ defines as φ, then according to the value of φ, the neutron flight direction on the every surface of hexagonal region can be divided into six quadrants averagely, and in each quadrant

the simplified P_1 approximation is adopted, then we can obtain the neutron flux in the q th quadrant on the k th surface as equation (8),

$$\Phi_{k,q}(\vec{r}_s,\bar{\Omega}) = \frac{1}{4\pi}\left[\phi_{k,q}^{(0)} + 3\cdot\phi_{k,q}^{(1)}\cdot\cos\theta\cdot\cos\varphi\right] \tag{8}$$

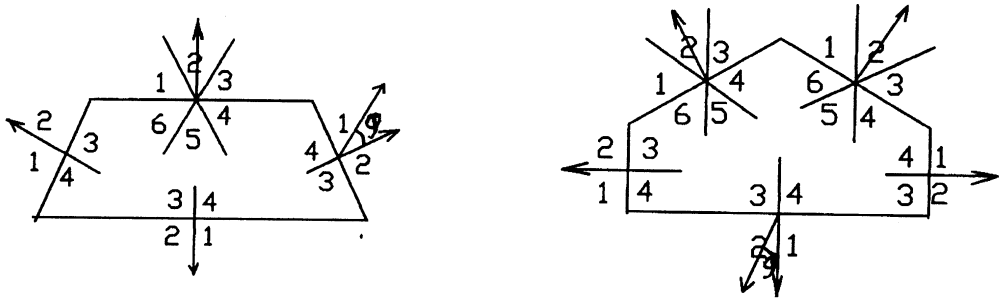
where $\phi_{k,q}^{(0)}$ and $\phi_{k,q}^{(1)}$ are expansion coefficients, they can be calculated in terms of the outgoing neutron flux $\Phi_{k,q}^{+}$ and the outgoing neutron current $J_{k,q}^{+}$ as equation (9) and (10)[2].

$$\begin{cases} \phi_{k,q}^{(0)} = \dfrac{6\Phi_{k,q}^{+}}{S_k} - \dfrac{9}{4}\cdot\dfrac{3\Phi_{k,q}^{+} - 8J_{k,q}^{+}}{\left(\dfrac{2\sqrt{3}}{\pi} - \dfrac{37}{24}\right)\cdot S_k} \\ \phi_{k,q}^{(1)} = \dfrac{6\Phi_{k,q}^{+} - 16J_{k,q}^{+}}{\left(\dfrac{2\sqrt{3}}{\pi} - \dfrac{37}{24}\right)\cdot S_k} \end{cases} \qquad q = 1,3 \tag{9}$$

$$\begin{cases} \phi_{k,q}^{(0)} = \dfrac{6\Phi_{k,q}^{+}}{S_k} - \dfrac{9}{2}\cdot\dfrac{3\Phi_{k,q}^{+} - 4J_{k,q}^{+}}{\left(-\dfrac{2\sqrt{3}}{\pi} + \dfrac{11}{12}\right)\cdot S_k} \\ \phi_{k,q}^{(1)} = \dfrac{6\Phi_{k,q}^{+} - 8J_{k,q}^{+}}{\left(-\dfrac{2\sqrt{3}}{\pi} + \dfrac{11}{12}\right)\cdot S_k} \end{cases} \qquad q = 2 \tag{10}$$

where S_k is the area of the k th surface.

For water gap region, the flat approximate for spatial distribution and the simplified $6P_1$



or simplified $4P_1$ approximate for angular distribution for neutron flux on region surface is adopted.

Fig5. The quadrant division on water gap region surfaces

For water gap region, the surfaces adjacent to hexagonal region must adopt simplified $6P_1$ approximation in order to maintain coherence with the hexagonal region surface approximation. And the other surfaces can adopt simplified $4P_1$ approximation for the sake of simplifying calculation. For simplified $4P_1$ approximation, the expansion coefficients in equation (8) can be calculate as following equation (11)[2],

$$\begin{cases} \phi_{k,q}^{(0)} = \dfrac{8\cdot(2\Phi_{k,q}^{+} - 3J_{k,q}^{+})}{S_k} \\ \phi_{k,q}^{(1)} = \dfrac{8\cdot(2J_{k,q}^{+} - \Phi_{k,q}^{+})}{S_k} \end{cases} \tag{11}$$

2.3 THE DERIVATION OF TRANSIMISSION PROBABILITY METHOD BASIC EQUATIONS

If we multiply $d\vec{\Omega}\cdot dS_k$ and $(\vec{\Omega}\cdot\vec{n}^{+})_k\, d\vec{\Omega}\cdot dS_k$ on the both side of equation (1), respectively, and integrate the equation in each surface and quadrant, and substitute equations (4) and (8) into equation (1), in the end, we can get regions coupling equations of outgoing neutron flux and current as follows(for the sake of simpleness, we omit the energy group suffix g),

$$\begin{cases} \Phi_{k,q}^{+} = V_i\cdot\overline{Q}_i\cdot E_{k,q}^{(0)} + \sum\limits_{\substack{k' \\ k'\neq k}} S_{k'}\cdot\left[\dfrac{1}{4}\cdot\phi_{k',q'}^{(0)}\cdot T_{k'k}^{(00)} + \dfrac{3}{4}\cdot\phi_{k',q'}^{(1)}\cdot T_{k'k}^{(10)}\right] \\ J_{k,q}^{+} = V_i\cdot\overline{Q}_i\cdot E_{k,q}^{(1)} + \sum\limits_{\substack{k' \\ k'\neq k}} S_{k'}\cdot\left[\dfrac{1}{4}\cdot\phi_{k',q'}^{(0)}\cdot T_{k'k}^{(01)} + \dfrac{3}{4}\cdot\phi_{k',q'}^{(1)}\cdot T_{k'k}^{(11)}\right] \end{cases} \tag{12}$$

where V_i is the volume of the i th region, S_k is the area of the k th surface, $\overline{Q}_i$ is the average neutron source in the i th region, $T_{k'k}^{(ll')}$ are the transmission probabilities of neutron transports from the k 'th surface to the k th surface, $E_{k,q}^{(j)}$ are the leakage probabilities of neutron in the i th region leakages from the q th quadrant on the k th surface, the expression of $E_{k,q}^{(j)}$ is,

$$E_{k,q}^{(j)} = (1-\frac{5}{6}\cdot Q_{x^2} - \frac{5}{6}\cdot Q_{y^2})\cdot E_{c,k,q}^{(j)} + Q_x\cdot E_{x,k,q}^{(j)} + Q_y\cdot E_{y,k,q}^{(j)} + Q_{xy}\cdot E_{xy,k,q}^{(j)} + Q_{x^2}\cdot E_{x^2,k,q}^{(j)} + Q_{y^2}\cdot E_{y^2,k,q}^{(j)} \tag{13}$$

where $E_{v,k,q}^{(j)}$ is the leakage probability of the v th term of neutron source. The detailed expressions for the above probabilities are given as follows,

$$E_{c,k,q}^{(j)} = \frac{1}{4\pi V_i}\int_{S_k} dS_k \int_q (\vec{\Omega}\cdot\vec{n}_k^{+})^j\, d\vec{\Omega}\int_0^{l_0} \exp(-\Sigma_t l/\cos\theta)\cdot\frac{dl}{\cos\theta} \tag{14}$$

$$E_{x,k,q}^{(j)} = \frac{1}{4\pi V_i}\int_{S_k} dS_k \int_q (\vec{\Omega}\cdot\vec{n}_k^{+})^j\, d\vec{\Omega}\int_0^{l_0} \frac{x}{a}\cdot\exp(-\Sigma_t l/\cos\theta)\cdot\frac{dl}{\cos\theta} \tag{15}$$

$$E_{y,k,q}^{(j)} = \frac{1}{4\pi V_i}\int_{S_k} dS_k \int_q (\vec{\Omega}\cdot\vec{n}_k^{+})^j\, d\vec{\Omega}\int_0^{l_0} \frac{y}{a}\cdot\exp(-\Sigma_t l/\cos\theta)\cdot\frac{dl}{\cos\theta} \tag{16}$$

$$E_{xy,k,q}^{(j)} = \frac{1}{4\pi V_i}\int_{S_k} dS_k \int_q (\vec{\Omega}\cdot\vec{n}_k^{+})^j\, d\vec{\Omega}\int_0^{l_0} \frac{xy}{a^2}\cdot\exp(-\Sigma l/\cos\theta)\cdot\frac{dl}{\cos\theta} \tag{17}$$

$$E_{x^2,k,q}^{(j)} = \frac{1}{4\pi V_i}\int_{S_k} dS_k \int_q (\vec{\Omega}\cdot\vec{n}_k^{+})^j\, d\vec{\Omega}\int_0^{l_0} \frac{x^2}{a^2}\cdot\exp(-\Sigma l/\cos\theta)\cdot\frac{dl}{\cos\theta} \tag{18}$$

$$E^{(j)}_{y^2,k,q} = \frac{1}{4\pi V_i}\int_{S_k} dS_k \int_q (\vec{\Omega}\cdot\vec{n}^{+}_{k})^{j} d\vec{\Omega} \int_0^{l_0} \frac{y^2}{a^2}\cdot \exp(-\Sigma l \big/ \cos\theta)\cdot \frac{dl}{\cos\theta} \tag{19}$$

$$T^{(jj')}_{k'k} = \frac{1}{\pi S_{k'}}\int_{S_k} dS_k \int_q (\vec{\Omega}\cdot\vec{n}^{-}_{k'})^{j}\cdot(\vec{\Omega}\cdot\vec{n}^{+}_{k})^{j'}\cdot \exp(-\Sigma_t l_0 \big/ \cos\theta) d\vec{\Omega} \tag{20}$$

Equations (2), (3), (9)~(20) form the basic equations of the transmission probability method (TPM) to calculate the neutron flux distribution in hexagonal fuel assembly.

2.4 THE DERIVATION OF RESPOND MATRIX METHOD BASIC EQUATIONS

In section 2.3, we have derived the basic equations of transmission probability method for calculation of the neutron flux distribution in hexagonal fuel assembly. But it isn't enough. Because there is a problem has to be resolved that how to determine the expansion coefficients in the expression (4), (5). As the hexagonal geometry is complicated and the power of expression (4) and (5) is high, this problem is very hard to be resolved. In this paper we adopt respond matrix method (RMM) to deal with this problem as equation (21).

$$\vec{\Phi} = [N]\cdot\vec{\phi} + [L]\cdot\vec{Q} \tag{21}$$

Expression (21) is a region neutron balance equation, it means that the inner neutron flux of the hexagonal region is composed of two parts: 1. The contribution of the incoming neutron flux on the region surfaces, 2. The contribution of the inner neutron source in the region. In equation (21) $\vec{\Phi} = (\Phi_0 \quad \Phi_x \quad \Phi_y \quad \Phi_{xy} \quad \Phi_{xy} \quad \Phi_{x^2} \quad \Phi_{y^2})^{-1}$ is the column vector of the expansion coefficients of neutron flux in the hexagonal region, $\vec{\phi} = (\phi^{(0)}_{1,4} \quad \phi^{(1)}_{1,4} \ldots \phi^{(l)}_{k,q} \ldots \phi^{(1)}_{6,6,})^{-1}$ is the column vector of the expansion coefficients of the incoming neutron flux on the surface of the hexagonal region, $\vec{Q} = (Q_0 \quad Q_x \quad Q_y \quad Q_{xy} \quad Q_{xy} \quad Q_{x^2} \quad Q_{y^2})^{-1}$ is the column vector of the expansion coefficients of neutron source in the hexagonal region, $[N]$ is the matrix of surface-region collision probability, $[N]\cdot\vec{\phi}$ stands for the contribution of incoming neutron flux on the region surfaces to the region inner neutron flux, $[L]$ is the matrix of region-region collision probability, $[L]\cdot\vec{Q}$ stands for the contribution of region inner neutron source to the region inner neutron flux. The material expression of $[N]$ and $[L]$ as follows,

$$[N] = \begin{pmatrix}[N_1] & [N_2] & [N_3] & [N_4] & [N_5] & [N_6]\end{pmatrix} \tag{22}$$

$$[N_k] = \begin{pmatrix} N^{0}_{0,4,k} & N^{0}_{1,4,k} & N^{0}_{0,5,k} & N^{0}_{1,5,k} & N^{0}_{0,6,k} & N^{0}_{1,6,k} \\ N^{x}_{0,4,k} & N^{x}_{1,4,k} & N^{x}_{0,5,k} & N^{x}_{1,5,k} & N^{x}_{0,6,k} & N^{x}_{1,6,k} \\ N^{y}_{0,4,k} & N^{y}_{1,4,k} & N^{y}_{0,5,k} & N^{y}_{1,5,k} & N^{y}_{0,6,k} & N^{y}_{1,6,k} \\ N^{xy}_{0,4,k} & N^{xy}_{1,4,k} & N^{xy}_{0,5,k} & N^{xy}_{1,5,k} & N^{xy}_{0,6,k} & N^{xy}_{1,6,k} \\ N^{x^2}_{0,4,k} & N^{x^2}_{1,4,k} & N^{x^2}_{0,5,k} & N^{x^2}_{1,5,k} & N^{x^2}_{0,6,k} & N^{x^2}_{1,6,k} \\ N^{y^2}_{0,4,k} & N^{y^2}_{1,4,k} & N^{y^2}_{0,5,k} & N^{y^2}_{1,5,k} & N^{y^2}_{0,6,k} & N^{y^2}_{1,6,k} \end{pmatrix} \text{(k=1~6)} \tag{23}$$

where matrix element $N^{v}_{l,q,k}$ means: the contribution probability of the l th term incoming neutron flux in the q th quadrant at the k th region surface to the v th term inner neutron flux of the region.

$$[L]=\begin{pmatrix} L_0^0 & L_x^0 & L_y^0 & L_{xy}^0 & L_{x^2}^0 & L_{y^2}^0 \\ L_0^x & L_x^x & L_y^x & L_{xy}^x & L_{x^2}^x & L_{y^2}^x \\ L_0^y & L_x^y & L_y^y & L_{xy}^y & L_{x^2}^y & L_{y^2}^y \\ L_0^{xy} & L_x^{xy} & L_y^{xy} & L_{xy}^{xy} & L_{x^2}^{xy} & L_{y^2}^{xy} \\ L_0^{x^2} & L_x^{x^2} & L_y^{x^2} & L_{xy}^{x^2} & L_{x^2}^{x^2} & L_{y^2}^{x^2} \\ L_0^{y^2} & L_x^{y^2} & L_y^{y^2} & L_{xy}^{y^2} & L_{x^2}^{y^2} & L_{y^2}^{y^2} \end{pmatrix} \tag{24}$$

where matrix element L_μ^ν means: the contribution probability of the μ th term inner neutron source of the region to the ν th term inner neutron flux of the region.

The derivation and calculation process of $[N]$ and $[L]$ is fairly complicated and troublesome. Limited by the length of this paper, we don't provided the detailed course in this paper, if someone is interested in it, can look out references [1],[2].

2.5 THE NEUTRON FLUX DISTRIBUTION CALCULATION PROGRAM TPHEX

Upwards, we study a new method combining the transmission probability method with the respond matrix method to solve neutron integral transport equation in 2-D hexagonal geometry. The total equations we have derived is gathered as follows,

$$\begin{cases} \Phi_{k,q}^{+} = V_i \cdot \overline{Q}_i \cdot E_{k,q}^{(0)} + \sum\limits_{\substack{k' \\ k' \neq k}} S_{k'} \cdot \left[\frac{1}{4} \cdot \phi_{k',q'}^{(0)} \cdot T_{k'k}^{(00)} + \frac{3}{4} \cdot \phi_{k',q'}^{(1)} \cdot T_{k'k}^{(10)} \right] \\ J_{k,q}^{+} = V_i \cdot \overline{Q}_i \cdot E_{k,q}^{(1)} + \sum\limits_{\substack{k' \\ k' \neq k}} S_{k'} \cdot \left[\frac{1}{4} \cdot \phi_{k',q'}^{(0)} \cdot T_{k'k}^{(01)} + \frac{3}{4} \cdot \phi_{k',q'}^{(1)} \cdot T_{k'k}^{(11)} \right] \\ \overline{\Phi}_{i,g} = \dfrac{\overline{Q}_{i,g}}{\Sigma_{t,i,g}} - \dfrac{J_{i,g}}{\Sigma_{t,i,g} \cdot V_i} \\ Q_{i,g} = \sum\limits_{g'=1}^{G} \left(\Sigma_{g' \to g} + \chi_g \dfrac{(\gamma \Sigma_f)_{g'}}{k_{eff}} \right) \cdot \Phi_{i,g'} + S_{i,g} \\ \bar{\Phi} = [N] \cdot \vec{\phi} + [L] \cdot \vec{Q} \end{cases} \tag{25}$$

Based on above model, the numerical calculation program TPHEX is encoded using FORTRAN langue to calculate the neutron flux distribution in hexagonal fuel assembly. To test the reliability of this program, we build some examples about neutron flux distribution in hexagonal fuel assembly, and compare the calculation results of TPHEX with that of MG-MCNP3B[3], they are all in good agreement, it proves that TPHEX has good calculation precision.

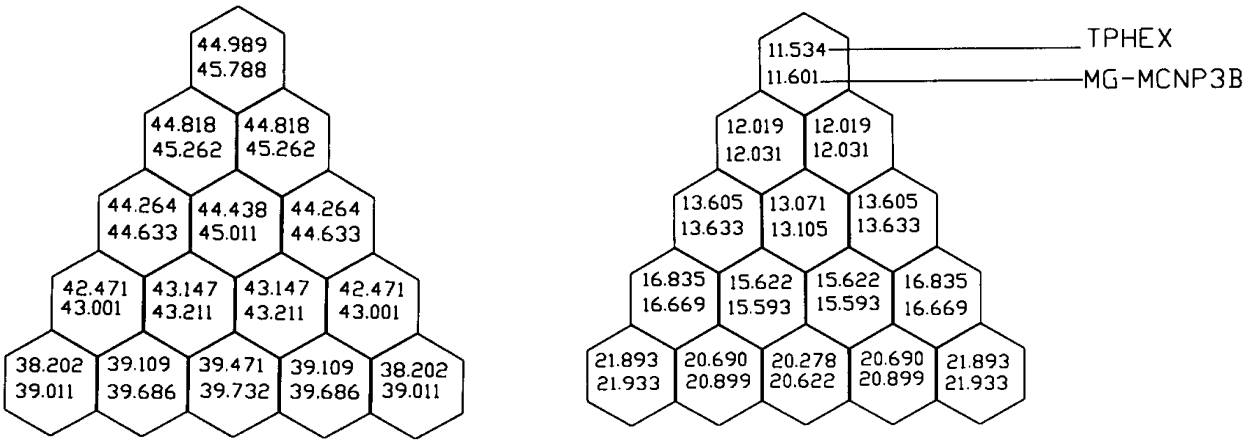
2.5.1 THE CHECK OF THE CALCULATION RESULT OF TPHEX

2.5.1.1 THE EXAMPLE ONE

In this example, the assembly consists of five rings of cells, and the cells in fifth ring(outer ring) are water cells, others are fuel cells. The pitch between cells is 1.73025 cm, the outer boundary condition is reflective condition. The cross section data of cells is in table 1, the calculation result is in figure6.

Tab1 The micro cross section data of example 1 (cm^{-1})

Energy group	Material	$\gamma\Sigma_f$	Σ_a	Σ_{1-1}	Σ_{1-2}	Σ_t
Fast group	Fuel cell	6.203e-3	8.627e-3	1.78e-1	1.002e-2	1.96647e-1
	Water cell	0.0	6.84e-4	1.995e-1	2.188e-2	2.22064e-1
		$\gamma\Sigma_f$	Σ_a	Σ_{2-1}	Σ_{2-2}	Σ_t
Thermal group	Fuel cell	1.101e-1	6.957e-2	1.089e-3	5.255e-1	5.96159e-1
	Water cell	0.0	8.016e-3	1.558e-3	8.783e-1	8.87874e-1



(a) The fast neutron flux distribution (b) The thermal neutron flux distribution

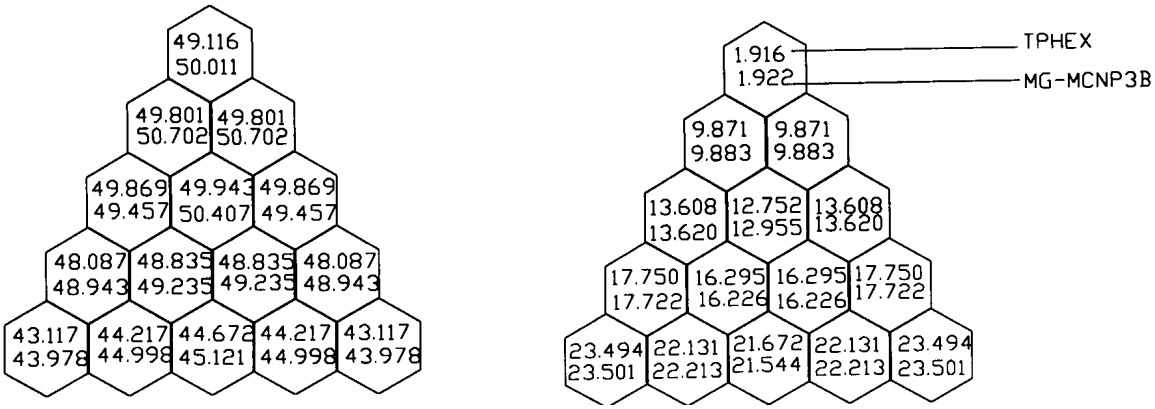
Fig6 The calculation result of example 1

2.5.1.2 THE EXAMPLE TWO

In this example, the assembly consists of five rings of cells, and the cells in fifth ring(outer ring) are water cells, the cell lies in the central position of the assembly is a poison cell, others are fuel cells. The pitch between cells is 1.73025 cm, the outer boundary condition is reflective condition. The cross section data of cells is in table 2, the calculation result is in figure7.

Tab2 The micro cross section data of example 2 (cm^{-1})

Energy group	Material	$\gamma\Sigma_f$	Σ_a	Σ_{1-1}	Σ_{1-2}	Σ_t
Fast group	Fuel cell Water cell Poison cell	6.2e-3 0.0 6.2e-3	9.0e-3 7.0e-4 9.0e-3	1.8e-1 2.0e-1 1.8e-1	1.0e-2 2.2e-2 1.0e-2	1.99e-1 2.227e-1 1.99e-1
		$\gamma\Sigma_f$	Σ_a	Σ_{2-1}	Σ_{2-2}	Σ_t
Thermal group	Fuel cell Water cell Poison cell	1.1e-1 0.0 1.1e-1	7.0e-2 8.0e-3 3.0	1.0e-3 2.0e-3 1.0e-3	5.3e-1 8.8e-1 5.3e-1	6.01e-1 8.9e-1 3.531



(a) The fast neutron flux distribution (b) The thermal neutron flux distribution

Fig7 The calculation result of example 2

Tab 3 The k_{eff} calculation result

	MG-MCNP3B	TPHEX	Relative error
Example 1	1.19730	1.19799	0.058%
Example 2	1.05011	1.05085	0.07%

Tab 4 The error analysis of flux calculation

	The biggest calculation error between TPHEX and MG MCNP3B	
Energy group	Fast group	Thermal group
Example 1	-2.074%	-1.668%
Example 2	-1.958%	-1.567%

3. THE HEXAGONAL FUEL ASSEMBLY CALCULATION SOFTWARE

PACKAGE TPFAP-HEX

Through above discuss, we obtain the program TPHEX for neutron flux distribution calculation in hexagonal fuel assembly. Now we can base on it to modify the software package TPFAP to form a new software package TPFAP-HEX for hexagonal fuel assembly homogenous calculation.

The TPFAP is a PWR quadrate assembly homogenous calculation software package, was developed by Xi'an Jiaotong University Nuclear Energy Department and China Nuclear Power Institute in 1990s[4]. Based on the cell homogenous calculation module and assembly burnup calculation module of TPFAP, some modifications aiming to TPFAP has been done. The modifications include: 1. The TPHEX is embedded in the TPFAP to replace the original calculation module of neutron flux distribution in quadrate assembly. 2. The other modifications of calculation module relating to geometry shape such as cell initialization calculation module, cell resonance absorption calculation module etc. Limited by the length of this paper, we don't provided the detailed introduction here, these contents can be looked out from references [2].

To verify TPFAP-HEX software, several IAEA benchmark problems of hexagonal fuel as--sembly of VVER-1000 type reactor are calculated and the results are shown in table 5 and table 6. In tables the calculation results of CASMO-HEX [5]are provide by the Technical Research Center of Finland and the calculation results of KASSETA[5] are provide by Kurchatov Institute of Russia. To calculate the neutron flux distribution in hexagonal fuel assembly, the CASMO-HEX solves the neutron transport equation and the KASSETA solves the neutron diffusion equation, the calculation precision of the former is better that that of the latter, and similar to that of the TPFAP-HEX, so the calculation results of TPFAP-HEX in table 5 and table 6 are more closer to CASMO-HEX calculation results than to KASSETA results.

Tab 5 k_{eff} of fuel assembly for the VVER-1000 (fuel temperature 120 ^{0} C,Xe=0,Sm=0 w=0 KW/l Bu=0Mwd/kg no control cluster, no burnable poison)

Boron concentration (ppm)		KASSETA	CASMO-HEX	TPFAP--HEX	Relative error (To CASMO-HEX)	Relative error (To KASSETA)
0	A	1.21020	1.21901	1.21642	-0.212%	0.514%
0	B	1.35036	1.33967	1.33632	-0.250%	1.04%
0	C	1.35515	1.36431	1.36075	-0.261%	0.413%
0	D	1.34996	1.35752	1.35458	-0.217%	0.342%
1000	A	1.01496	1.02059	1.02071	0.012%	0.567%
1000	B	1.15864	1.16532	1.16474	-0.049%	0.526%
1000	C	1.18978	1.19657	1.19574	-0.069%	0.501%
1000	D	1.18308	1.18779	1.18786	0.006%	0.404%

Tab6 k_{eff} as a function of burnup for C VVER—1000 fuel assembly(fuel temperature 830^0C clad temperature 302^0C moderator temperatture 302^0C power density 108KW/cm^3 equilibrium Xe and Sm boron concentration 600ppm)

Bu [Mwd/kg]	KASSETA	CASMO-HEX	TPFAP—HEX	relative error (To CASMO-HEX)	relative error (To KASSETA)
0	1.17712	1.19212	1.18747	-0.3901%	0.879%
1	----	1.17668	1.17190	-0.406%	-----
2	1.16265	1.16780	1.16324	-0.391%	0.05%
3	----	1.15794	1.15304	-0.423%	-----
4	1.14371	1.14759	1.14262	-0.433%	-0.09%
6	1.12419	1.12651	1.12129	-0.463%	-0.2585
8	1.10482	1.10594	1.10068	-0.476%	-0.3755
10	1.08653	1.08632	1.08107	-0.483%	-0.503%
12	1.06887	1.06764	1.06249	-0.482%	-0.5975
16	1.03496	1.03312	1.02799	-0.497%	-0.673%
20	1.00222	1.00096	0.99635	-0.461%	-0.5875
24	0.97055	0.97075	0.96691	-0.396%	-0.375%
28	0.94027	0.94204	0.93929	-0.292%	-0.104%
32	0.91127	0.91489	0.91331	-0.173%	-0.224%
36	0.88402	0.88923	0.88900	-0.026%	-0.563%
40	0.85802	0.86509	0.86636	-0.147%	-0.9725
44	0.83331	0.84258	0.84542	0.337%	1.45%
48	0.80856	0.82179	0.82617	0.533%	2.18%

The calculation results of TPFAP-HEX show good agreement with those of IAEA references, especially with the results of CASMO-HEX. It proves that TPFAP-HEX has satisfactory precision and can be used in engineering practices.

4. CONCLUSION

In this paper, aiming to the problems of neutron flux distribution calculation in PWR hexagonal fuel assembly and PWR hexagonal assembly homogenous calculation, two codes TPHEX and TPFAP-HEX were developed. Through calculation results checking, it proves that the code development is successful and the two codes have good calculation precision and can be used in engineering practices.

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