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Code System Development for Analysis of the Fast Transmutation Reactors

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Abstract

In this paper, research efforts to develop computer code systems for analysis of the transmutation reactors at KAIST are described. Especially the computer code HANCELL for assembly calculation of fast reactors is mainly described. Features and functions of the code are identified and current status of the code development is provided.

I. Introduction

To resolve the burden due to radionuclides (reduced volume of radioactive waste to be disposed), various concepts of transmutation systems have been proposed, which include existing power reactors (LWRs or FBRs) and dedicated systems such as actinide burner reactor(ABR) or accelerator-driven hybrid system[1, 2]. From the neutronics point of view, a reactor core of a transmutation system based on the fast reactor concept is very similar to that of the conventional power reactors, except some differences in fuel composition and core structure. Therefore, a code system for the analysis of the transmutation cores using fast neutrons should be essentially the same as that for the fast reactors. In general, calculational procedure for the fast reactor analysis consists of preparation of multigroup cross sections, assembly calculation, core calculation, and depletion calculation. These three codes are being developed at KAIST.

The computer code under development for assembly calculation is based on the collision probability method[3] (CPM, 1-D and 2-D) and a prototypical code, which is named HANCELL[4], is briefly described in this paper.

The codes for core calculation are in two categories: one is to develop a fine-mesh code utilizing the finite difference method (FDM) and the other is based on a nodal method. With regard to the fine-mesh code, a multi-dimensional (2-D and 3-D) code PRISM[4] was developed and tested. Development of a multi-dimensional nodal code for hexagonal geometry is in progress, where the AFEN[5] methodology is exploited.

Microscopic depletion is essential for fast reactor analysis. A computer code which calculates the change of nuclide concentration in each region of a core is also being developed.

II. Assembly Calculation Code HANCELL

Since the assembly geometry is hexagonal in fast reactors, the multigroup transport equations should be solved in the hexagonal geometry. However, it is impractical to treat the explicit geometry from the computing time point of view. Therefore, various types of geometrical approximations have been adopted and one-dimensional calculation is often used in fast reactor assembly calculation: Fig. 1 shows a typical example.

A prototypical computer code for assembly calculation HANCELL is being developed, which has the following features and capabilities: 1) macroscopic multigroup cross sections are used as input, 2) one-dimensional collision probability method for slab and cylindrical geometries, 3) fundamental mode calculation/one-dimensional diffusion calculation for group condensation, 4) control assembly homogenization.

HANCELL has no its own cross section library at the moment. The necessary multigroup library will be prepared by processing ENDF/B. Note that cross section library of Bondarenko[6] type is often used in fast reactor calculations. Extension to two-dimensional collision probability method is also in progress. In the HANCELL code, group collapsing can be done via either fundamental mode calculation or one-dimensional diffusion calculation. One-dimensional transport or diffusion calculation is usually used in fast reactor analysis for condensation of energy groups. The one-dimensional transport option will be added.

II.1 Collision Probability Method

Collision probability method is usually used for the unit cell calculation in reactor analysis since it can treat a sophisticated geometry and provide accurate solution for highly heterogeneous medium. Assuming that a unit cell is divided into N nodes, the transport equation can be written in the following algebraic form

$$\Sigma_i^i \phi_i V_i = \sum_{j=1}^{N} P_{ij} V_j \psi_j , \quad i = 1, 2, \ldots, N \tag{1}$$

where V_i and ϕ_i indicate volume and average flux of the ith node and ψ_j represents emission density in the jth node.

In Eq. (1), P_{ij} denotes the probability that a neutron isotropically and uniformly emitted in node j makes its first flight collision in node i and it is defined as

$$P_{ij}=\frac{1}{4\pi V_j}\int_{V_i}dr\int_{V_j}dr'\frac{\Sigma_t(r)}{|r-r'|^2}\exp[\ -\tau(r,r')]\ , \tag{2}$$

where $\tau(r,r')$ represents optical thickness between r and r'.

For slab and cylindrical geometries, P_{ij} can be written in terms of the exponential function and Bickley-Naylor function, respectively. For example, in cylindrical geometry (Fig. 2), P_{ij} can be written as

$$\begin{aligned}P_{ij}=\ &\frac{2}{\Sigma_j V_j}\int_0^{\min(r_i,r_j)}dR[\ K_3(\tau_{ij1})+K_3(\tau_{ij2})-K_3(\tau_{ij1}+\tau_i)-K_3(\tau_{ij2}+\tau_i)\\ &-\ K_3(\tau_{ij1}+\tau_j)-K_3(\tau_{ij2}+\tau_j)+K_3(\tau_{ij1}+\tau_i+\tau_j)+K_3(\tau_{ij2}+\tau_i+\tau_j)]\ ,\end{aligned} \tag{3}$$

where $\tau_{ij2}=\sum_{k=1}^{i-1}\tau_k+\sum_{k=1}^{j-1}\tau_k$ and $\tau_{ij1}=\sum_{k=l}^{m}\tau_k$.

In the above equation, $l=i+1$, $m=j-1$ if $r_i<r_j$ and $l=j+1$, $m=i-1$ if $r_i>r_j$. The self collision probability P_{ii} is

$$P_{ii}=1-\frac{2}{\Sigma_i V_i}\int_0^{r_i}dR[\ K_3(\tau_{ii})+2K_3(\tau_i)+K_3(\tau_{ii}+2\tau_i)-2K_3(\tau_{ii}+\tau_i)-2K_3(0)] \tag{4}$$

where $\tau_{ii}=2\sum_{k=1}^{i-1}\tau_k$. Numerical integrations required to evaluate collision probabilities are peformed using the Gauss quadrature scheme in the HANCELL code.

II.2 Control Assembly Homogenization

Typical homogenization of a control assembly in a fast core utilizes the supercell calculation, where the control assembly and adjacent fuel assemblies are transformed into an equivalent cylindrical geometry (Fig. 3). In general, a heterogeneous medium is homogenized by using the space-dependent flux as weighting function. However, the flux-weighting method for a control assembly preserves neither the neutron current at the interface between control assembly and fuel assembly nor the reaction rate in the control assembly. To overcome this drawback, Takeda[7] proposed a nonlinear iterative homogenization.

In the iterative homogenization, homogeneous cross sections are obtained in the following way:

$$\overline{\Sigma}_{ag}^{(n)}=\overline{\Sigma}_{ag}^{(0)}\int dv\Phi_g^{het}(r)/\int dv\Phi_g^{\hom(n-1)}\ , \tag{5}$$

where $\overline{\Sigma}_{ag}^{(0)}$ is a-type homogenized cross section calculated by the flux-weighting scheme. In Eq. (5), Φ^{het} and $\Phi^{\hom}$ represent heterogeneous and homogeneous fluxes, respectively.

III. Numerical Results

To validate the HANCELL code, a benchmark problem[4] was considered, which is a typical rectangular pincell consisting of fuel and coolant: radius of fuel r_1 = 0.46482 cm, side length = $\sqrt{2\pi}r_1$, total cross section of coolant σ_2 = 1.4916 cm^{-1}. In this two-region problem, the escape probability P_{21} was evaluated for several σ_1 (total cross section of fuel). Table 1 compares three escape probabilities, the first column represents exact value calculated by Fukai[8], the second one is from Pennington[9]. Pennington's escape probability is also calculated by using one-dimensional collision probability method as in the HANCELL code. Table 1 indicates that the collision probability calculation in HANCELL is accurate.

Table 1 Comparison of escape probability in a pincell

$\sigma_1 r_1$	Fukai	Pennington	HANCELL NG : order of Gauss quadrature		
			NG=10	NG=20	NG=40
0.1	0.80745	0.81364	0.81367	0.81354	0.81352
0.3	0.57988	0.58855	0.58864	0.58852	0.58852
0.5	0.44745	0.45695	0.45704	0.45694	0.45693
0.8	0.32973	0.33839	0.33848	0.33839	0.33837
1.0	0.27918	0.28694	0.28703	0.28694	0.28693
2.0	0.15490	0.15912	0.15920	0.15912	0.15911

The iterative homogenization method was applied to a typical control assembly[4] of fast reactors, where 4 energy groups were considered. Table 2 shows absorption reaction rates in a control assembly. Clearly, the iterative homogenization preserves reaction rates, while the conventional flux-weighting method does not.

Table 2 Absorption reaction rate in a control assembly

energy group (g)	heterogeneous control assembly	homogenized control assembly	
		flux-weighting	iterative homogenization
g=1	6.208733E-03	6.213842E-03	6.208731E-03
g=2	1.998284E-01	2.000695E-01	1.998283E-01
g=3	7.535474E-02	7.558379E-02	7.535496E-02
g=4	1.787372E-02	1.797007E-02	1.787632E-02

VI. Summary

At present, the main computer codes for the analysis of the transmutation reactors utilizing fast neutrons are being developed at KAIST.

Ongoing research efforts related to code system development are addressed and a prototypical assembly calculation code HANCELL is mainly described in this paper. Major features of the current HANCELL code are as follows:

- one-dimensional collision probability method for slab and cylindrical geometries,
- fundamental mode calculation/one-dimensional diffusion calculation for group condensation,
- control assembly homogenization using a nonlinear iterative method.

For more efficient and useful code, further refinement of HANCELL is in proress. Through some preliminary calculations, we have presented part of the capabilities of the HANCELL code.

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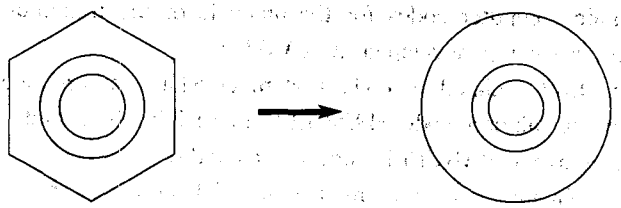

Fig.1 1-D equivalent model of a hexagonal cell

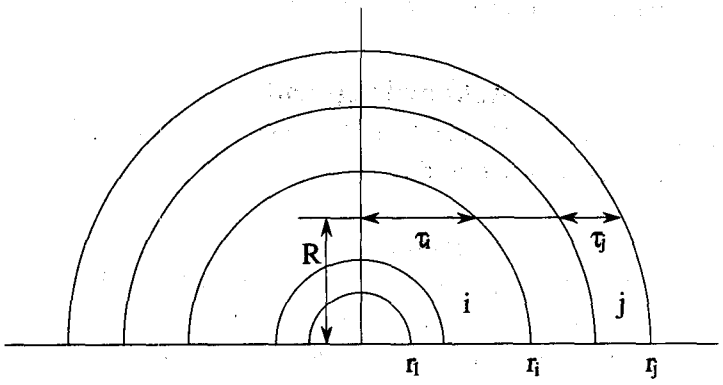

Fig. 2 Cylindrical Coordinate for collision probability calculation

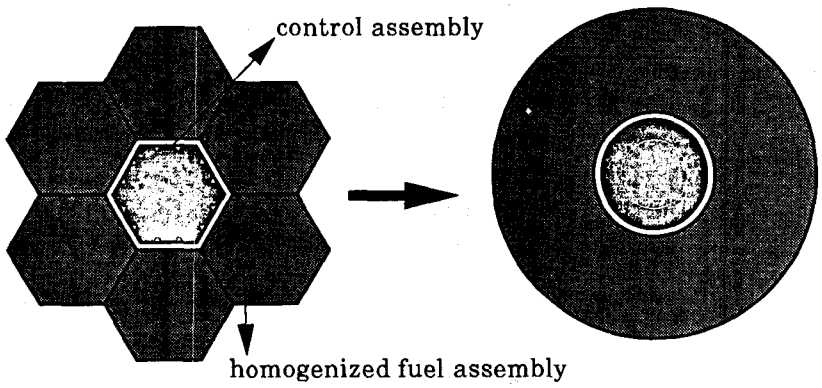

Fig. 3 Supercell model for control assembly homogenization