Real-Time Simulation of Reactor Physics for China Experimental Fast Reactor

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Abstract: China Experimental Fast Reactor (CEFR) is the first fast neutron breed reactor in China, which is different with PWR. In order to research the operational performance of CEFR, the real-time simulator was developed. The simulation of core physics is an important part of the simulator. The neutron dynamic model used in the simulator is three dimensions and four groups neutron diffusion model, which was solved by the improved quasi-static approximation node method. The neutron flux was divided into shape function and amplitude function. The shape function changes slowly with time, so a large time step is adopted. And the amplitude function changes quickly with time, so a small step is adopted. The calculation time can be saved, it is important for real-time simulation. According the character of CEFR, the core was divided into many nodes. The homogenization parameters of each node were calculated by HELIOS. Considering influences of fuel burnup, fuel temperature and coolant temperature on fuel assembly cross section, four-order polynomial is adopted for fitting. Because there are hexagonal fuel assemblies in CEFR core, the calculation of leakage term was modified based on the pressurized water reactor calculation program. The improved alternative direction implicit (ADI) algorithm is used to solve diffusion equations. The simulation result indicates that the improved algorithm is able to meet requirements for the real-time simulation. Two steady states (BOL and EOL) were simulated. And some dynamic operation cases were simulated, including reactor star-up and a control rod drawing out of core without control. Compared with the Final Safety Analysis Report for CEFR, the three-dimensional power distribution and control rod value are in good agreement. The core physics simulation program is able to use the operation research of CEFR.

Introduction

As the fast reactor has the phenomenon of transmutation, proliferation and other important features, the countries advanced in nuclear power have been carried out a Series of reactor technology researches. China Experimental Fast Reactor is China's first fast reactor, which reached the first critical on July 21, 2010. Therefore, China has also become one of the few countries in the world to master fast reactor technology. The simulator can verify and improve the reactor design, training operations personnel and so on. Therefore, each country are not only focusing on the development of fast reactor technology, but also the simultaneous development of fast reactor simulation. However, fast reactor simulation technology in China is still in the blank stage. At present, most of the fast reactor simulator are using the point pile equation to describe the core physical process, thus the reactor complex state can’t be fully described. Based on the present situation, it is not only necessary but also urgent to study the 3D physical simulation of fast reactor core.

1 Core physics simulation model

1.1 Diffusion equation

The reactor core physical simulation models adopt multi-group diffusion equations with six groups of delayed neutrons. The expression is as follows:

$$\begin{aligned}\frac{1}{\upsilon_g}\frac{\partial\phi_g(r,t)}{\partial t} &= \nabla D_g(r,t)\nabla\phi_g(r,t)-\Sigma_{ag}(r,t)\phi_g(r,t)-\Sigma_{sg}(r,t)\phi_g(r,t)\\ &+\sum_{g'=1}^{G}\Sigma_{sg'g}(r,t)\phi_{g'}(r,t)+(1-\beta)\chi_{pg}\sum_{g'=1}^{G}\nu_{g'}\Sigma_{fg'}(r,t)\phi_{g'}(r,t)\\ &+\chi_{dg}\sum_{d=1}^{D}\lambda_d C_d(r,t)+S_g(r,t)\end{aligned}\tag{1}$$

$$\frac{\partial C_d(r,t)}{\partial t}=-\lambda_d C_d(r,t)+\beta_d\sum_{g=1}^{G}\nu_g\Sigma_{fg}(r,t)\phi_g(r,t)\tag{2}$$

1.2 Improved quasi-static approximation

Before the introduction of the improved quasi-static approximation, neutrons flux is decomposed into the shape function $\Psi(r,t)$ and the amplitude function $n(t)$ [1], that the introduction of assumptions:

$$\phi(r,t)=\Psi(r,t)n(t)\tag{3}$$

Introduce the following constraints:

$$\sum_{V}\left[\sum_{i=1}^{g}\Psi_i(r,t)\right]\times V(r)=V_{core}\tag{4}$$

Equation (1) is simplified into the following form by using equation (3):

$$\frac{1}{\upsilon_g}\frac{\partial\Psi_g(r,t)}{\partial t}=\nabla D_g(r,t)\nabla\Psi_g(r,t)-f_g(r,t)\Psi_g(r,t)+g_g(r,t)\tag{5}$$

Where：

$$f_g(r,t)=\frac{\dot{n}(t)}{\upsilon_g n(t)}+\Sigma_{ag}(r,t)+\Sigma_{sg}(r,t)-(1-\beta)\chi_{pg}v_g\Sigma_{fg}(r,t)\tag{6}$$

$$\begin{aligned}g_g(r,t)&=(1-\beta)\chi_{pg}\sum_{g'\neq g}^{G}v_{g'}\Sigma_{fg'}(r,t)\Psi_{g'}(r,t)\\ &+\left(\chi_{dg}\sum_{d=1}^{D}\lambda_d C_d(r,t)+S_g(r,t)\right)/n(t)+\sum_{g'\neq g}^{G}\Sigma_{sg'g}(r,t)\Psi_{g'}(r,t)\end{aligned}\tag{7}$$

The amplitude function is solved by using the point reactor kinetics equations.

$$\frac{dn(t)}{dt}=\frac{\rho(t)-\beta}{\Lambda}n(t)+\sum_{k=1}^{6}\lambda_k C_k(t)+S\tag{8}$$

For the fast reactor using the hexagonal fuel assemblies, it can be seen that equation (5) is a four-dimensional parabolic partial differential equations with variable coefficients.

1.3 The handling of leakage items

The solution of the leakage term in the diffusion equation with six groups of delayed neutrons is solved by integrating in the beginning and then solving separately, only the neutrons to the adjacent radial 6 and axial 2 total of 8 nodes are considered in the leakage. The nodal division is shown in Figure 1. This assumption is reasonable and sufficient according to the geometry of the nodule and the size of the neutron diffusion length. The derivation of the leakage term coefficients is not repeated here. The expression for the integral of the leakage term in the neutron diffusion equation is given below:

$$\begin{aligned}\int_V \nabla \cdot D\nabla\phi dV &= \int_S \left(D\frac{\partial\phi}{\partial n}\right) dS = \sum_k \int_{S_k} D\frac{\partial\phi}{\partial n} dS \\ &= \left(T_1 + T_2 + T_3 + T_4 + T_5 + T_6 + T_7 + T_8\right)\end{aligned} \tag{9}$$

2 Alternating direction implicit method

After the improved quasi-static approximation is introduced, the diffusion equation is transformed into variable coefficient high-dimensional parabolic partial differential equations. The parabolic equations can be solved by using the typical method for solving parabolic partial differential equations. The typical method is Alternative Direction Implicit, in short, ADI. ADI method was put forward by the Peaceman and Rachford for solving parabolic equations and two-dimensional elliptic equations in as early as 1955[2]. ADI method now can be widely used to solve diffusion and convection-diffusion problems.

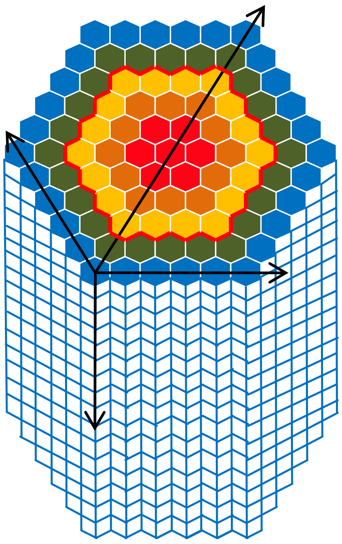

Fig 1 the nodal division

Because the instability exists in the traditional ADI method when solving three-dimensional and higher dimensional problems, Douglas format ADI method is usually used to solve higher dimensional problems. In this method, the simplified form of the neutron diffusion equation (5) is differenced using Douglas format. Omitting the subscript of the energy group and using u to represent ψ, the transformed Format is as follows:

$$\frac{1}{v}\frac{U-T}{\Delta t} = \frac{1}{2}\delta_x^2(U+T) + \delta_y^2 T + \delta_{y2}^2 T + \delta_z^2 T - fU + g \tag{9}$$

$$\frac{1}{v}\frac{V-U}{\Delta t}=\frac{1}{2}\delta_y^2(V-T)-f(V-U) \tag{10}$$

$$\frac{1}{v}\frac{V_2-U}{\Delta t}=\frac{1}{2}\delta_{y2}^2(V_2-T)-f(V_2-V) \tag{11}$$

$$\frac{1}{v}\frac{T^{'}-V_2}{\Delta t}=\frac{1}{2}\delta_z^2(T^{'}-T)-f(T^{'}-V_2) \tag{12}$$

Douglas format is valid for the three-dimensional parabolic partial differential equation, and is unconditionally stable. But on specific issues, only appropriate time step and spatial interval will get acceptable results [3]. In fact, in the core simulation, single assembly is usually selected as a grid, and this often leads to the inaccuracy of the results. So Douglas format does not have universal applicability.

2.2 Improved Alternative Direction Implicit method

In this paper, the traditional format ADI algorithm is adjusted so that the accuracy and stability of the calculation results can meet the requirements. The format after adjustment is as follows:

$$\frac{1}{v}\frac{U-T}{\Delta t}=\frac{1}{2}\delta_x^2(U+T)+\delta_y^2T+\delta_{y2}^2T+\delta_z^2T-fU+g \tag{13}$$

$$\frac{1}{v}\frac{V-T}{\Delta t}=\delta_x^2U+\frac{1}{2}\delta_y^2(V+T)+\delta_{y2}^2T+\delta_z^2T-fV+g \tag{14}$$

$$\frac{1}{v}\frac{V_2-T}{\Delta t}=\delta_x^2U+\delta_y^2V+\frac{1}{2}\delta_{y2}^2(V_2+T)+\delta_z^2-fV_2+g \tag{15}$$

$$\frac{1}{v}\frac{T^{'}-T}{\Delta t}=\delta_x^2U+\delta_y^2V+\delta_{y2}^2V_2+\frac{1}{2}\delta_z^2(T^{'}+T)-fT^{'}+g \tag{16}$$

3 Calculation of component homogenization section

For the diffusion equation used in the cross-section parameters, in this paper, the HELIOS[4] program is used to model the experimental fast reactor components, and the homogeneous section under specific working conditions is obtained. The effects of fuel temperature, coolant temperature and fuel consumption on the cross section are considered. Then the least squares method is used to fit the object, and the cross-section parameters of the core nodule under different working conditions are obtained, which are used for the diffusion equation. In order to ensure the correctness and accuracy of the simulation results, the fitting results are verified. The maximum error of the cross section calculated by the fitting formula and HELIOS is about 1%, and the rest is less than 0.5%. This is acceptable for real-time simulation of 3D core physics.

4 Simulation results

4.1 Begin of life simulation results

At full power, the power distribution at the beginning of service life of fuel assembly is shown in Fig 2. In Figure 2, the maximum difference between the reference value and the simulation value is 23KW, the error of which is 3% or less, it is satisfactory on the whole. Figure 3 shows the axial power distribution of the first row of fuel assemblies at the beginning of the service life. Here the first row of fuel assemblies refers to six components around the neutron source. Figure 4 shows the axial power distribution of the fifth row of fuel assemblies at the beginning of the service life. Here, the fifth row of fuel assemblies refers to the outermost fuel assembly of the reactor core. The difference between the upper and lower boundaries and the reference value is large, which may be due to the upper and lower boundary conditions have not been handled well.

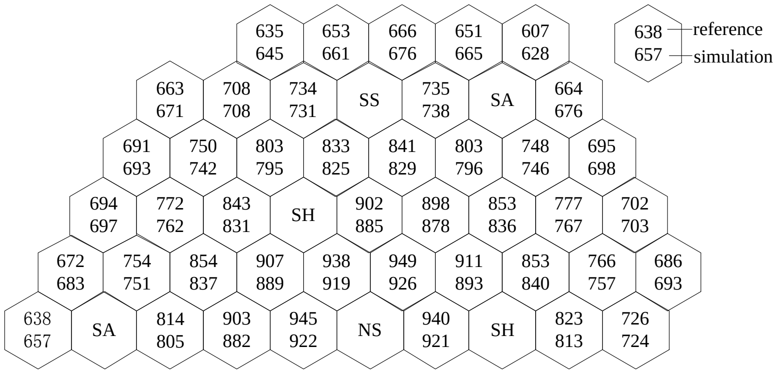

Fig 2 Radial power distribution

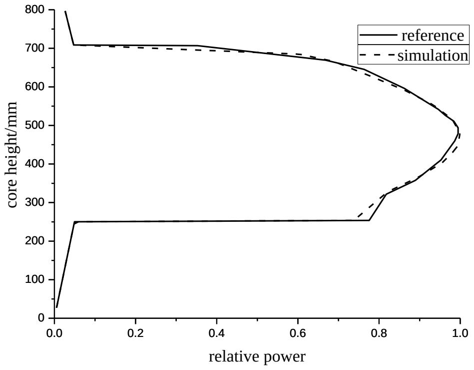

Fig 3 Axial power distribution of the first row of components

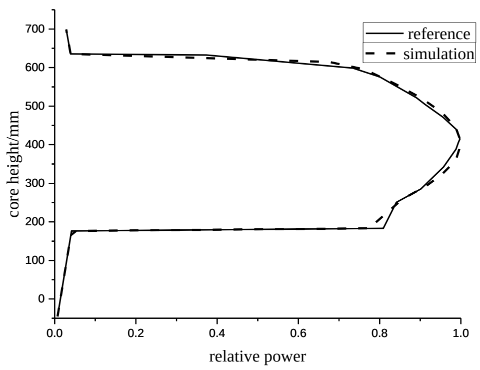

Fig 4 Axial power distribution of the fifth row of components

4.2 Simulation results of control rod uncontrolled lifting at rated power

During the normal operation of the reactor, both rods are located above the plane of the core. One of which is in the auto-tuning state and the other is in the standby state. It is assumed that the control rods in the automatic adjustment state are out of control to the top of the heap; the adjusting rod in the standby state is transferred to the automatic adjustment state and is then raised to the top of the heap. This corresponds to the reactivity of a rod from the bottom out of control to the top of the heap. The accident sequence is shown in Table 1[5].

In the shutdown process, in order to meet the stick standard, only two of the safety bar is assigned to insert to the core. The relative power change results are shown in Fig. It can be seen from the figure, that in the experimental fast reactor core physics simulation system built in this article grow faster when the control rod is raised to enhance the power, It is faster to reach the shutdown protection limit than the reference results, this is because the adjustment rod value in this system is greater than the reference value. During the whole process of the accident, the maximum relative power is 1.17, which is not much different from the reference value of 1.157.

Table 1 out of control rod to enhance the accident development sequence

event	time/s
An adjustment rod to 30mm/s speed out of control to enhance	0.0
The relative power of the reactor reaches the protection parameter setting value 1.14	7.9
Issued by the reactor relative to the power shutdown protection signal	8.9
Reactor emergency shutdown (after 0.2s)	9.1
The heap relative power reaches a maximum of 1.157	9.1
The safety rod falls to the bottom	10.5
Adjust the rod to the top of the heap	15.0

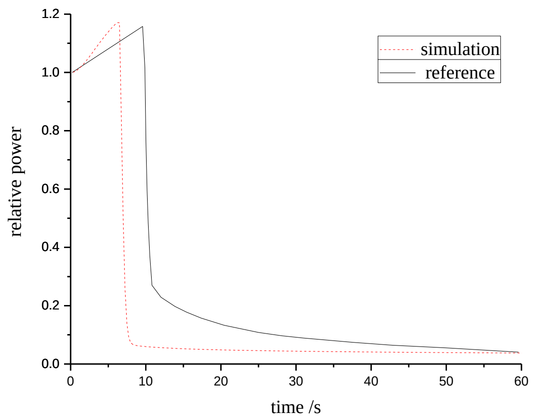

Fig 5 the control rod to enhance the accident
Control of the relative power changes

5 Results

In this paper, a real-time simulation system of fast reactor core physics is established. The ADI algorithm for solving the diffusion equation is improved by using three-dimensional four-group neutron dynamics model with six groups of delayed neutrons. The homogeneous cross is obtained and It is fitted by the least squares method. In this paper, the initial furnace core life and the accidently-raised condition is simulated, and the results are compared with the "China Experimental Fast Reactor Engineering Reactor System Manual", and good simulation effect is achieved. The real-time simulation system of experimental fast reactor core physics has been added to the simulation fast reactor simulation system to complete the commissioning work such as starting and stopping reactor, lifting power, accident simulation and so on.

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