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Development of Three Dimensional Kinetic Code for Real-Time Simulator

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Abstract

A new three-dimensional diffusion kinetic code CORE3D has been developed for PWR real-time simulator. A kinetic calculation is based on the modified quasi-static method. For three-dimensional neutron flux distribution calculation, the modified one-group method is applied to increase calculation speed. We have incorporated CORE3D into our real-time simulator and verified its performance. The calculation speed is sufficient for real-time simulation. And the calculation accuracy has been verified by comparing with our nuclear design code ANC. Both codes have a good agreement in various reactor characteristics calculation. We have confirmed the validity of CORE3D.

1. Introduction

Real time simulators are used for training of nuclear power plant operators. Up to date we have supplied PWR real-time simulators for Japanese utilities. Core model of conventional simulator was combined with one dimensional kinetics model in axial direction and simplified two-dimensional calculation model in horizontal direction because PWR core characteristics of these two directions have a good independence. In these days the three-dimensional effect on the core characteristics becomes larger because of the high burnup of the core and the needs of training for optimum operation arise, the core calculation model is to be improved based on the three dimensional model in order to be more accurate and the three-dimensional diffusion kinetic code CORE3D working in a real time has been developed.

2. Calculation theory

In this section, we would like to show the calculation theory. Figure 2-1 shows the interface between CORE3D and the other calculation models in the simulator. CORE3D receives input data e.g. control rod position from other models and performs transient calculation and sends the outputs e.g. neutron flux distribution as results of reactor kinetic

calculation.

Calculation flow of CORE3D is shown in Figure 2-2. First of all in the initial calculation it makes the state of the beginning of the simulation. If the simulation is in the state during nominal operation it makes burnup distribution of the core according to the nominal power burnup calculation. The burnup calculation is based on the three-dimensional diffusion calculation almost same as the method used in transient kinetic calculation mentioned in detail later.

After initial calculation, a transient calculation is performed repeatedly. In the transient calculation two-group constants are modified according to the core condition. Then, reactivity and neutron distribution are calculated in the three-dimension kinetic calculation.

These calculation methods are mentioned in detail as follows.

2.1 Calculation model

We would like to show the calculation model in CORE3D. In the horizontal direction the one calculation mesh is assigned to each fuel assembly and in the axial direction the effective core height is divided in 17 meshes. While in the axial direction the length of the mesh can be changed at the viewpoint of calculation accuracy and efficiency in order to make mesh a cube 17 mesh are usually chosen.

2.2 Nuclear group constant feedback

The assemblies consisting the core are divided into several groups as its nuclear characteristics. The each group has its own macroscopic cross section. It is a function of burnups as arranged in the table and used for each calculation mesh by fitting according to the burnup. As the constants are made based on a nominal condition, they are modified to adjust the change from nominal condition.

Macroscopic cross sections are the average value of the assembly. In order to remove this averaging error discontinuous factor prepared for each pseudo burnup is used, however in the assembly that has control rods the discontinuous factor prepared for the case of control rod insertion.

2.3 Three-dimensional kinetic calculation

In CORE3D, core kinetics calculated by time dependent three-dimensional diffusion equation which have two neutron energy groups; fast and thermal energy. The groups of the delayed neutron are divided in 6 groups according to the half-life time of the precursor.

$$\frac{1}{v_1}\frac{d}{dt}\phi_1(\vec{r},t) = D_1(\vec{r})\cdot\nabla^2\phi_1(\vec{r},t) - \Sigma_1(\vec{r})\phi_1(\vec{r},t) + (1-\beta)\cdot(\nu\Sigma_{f1}(\vec{r})\phi_1(\vec{r},t) + \nu\Sigma_{f2}(\vec{r})\phi(\vec{r},t)) + \sum_{i=1}^{6}\lambda_i C_i(\vec{r}) \quad (1)$$

$$\frac{1}{v_2}\frac{d}{dt}\phi_2(\vec{r},t) = D_2(\vec{r})\cdot\nabla^2\phi_2(\vec{r},t) - \Sigma_2(\vec{r})\phi_2(\vec{r},t) + \Sigma_m(\vec{r})\phi_1(\vec{r},t) \quad (2)$$

$$\frac{1}{dt}C_i(\vec{r},t) = \beta_i(\nu\Sigma_{f1}(\vec{r})\phi_1(\vec{r},t) + \nu\Sigma_{f2}(\vec{r})\phi_2(\vec{r},t)) - \lambda_i C_i(\vec{r}) \quad (3)$$

i = 1 - 6

To solve time dependent three-dimensional diffusion equation above in short time with good accuracy, it is treated as the modified quasi-static method[1)]. Time dependent neutron flux is described as follows.

$$\phi(\vec{r},t) = P(t)\cdot\Phi(\vec{r},t) \quad (4)$$

P(t) is an amplitude function and Φ(r, t) is a shape function. In such variable separation, P(t) is described as follows.

$$\frac{d}{dt}P(t) = \frac{\rho-\beta}{\Lambda}P(t) + \sum_{i=1}^{6}\lambda_i C_i(t) \quad (5)$$

$$\frac{d}{dt}C_i(t) = \frac{\beta_i}{\Lambda}P(t) - \lambda_i C_i(t) \quad (6)$$

$$\rho = keff - 1 \quad (7)$$

$$\Lambda = \lambda^* / keff \quad (8)$$

The amplitude function P(t) is solved by time integration of eq. (5) and (6) using Runge-Kutta-Gill's method[1)].

The solution of the shape function Φ(r, t) is similar to that of the general diffusion equation described as follows in a simple manner.

$$-D_1\nabla^2\Phi_1 + \Sigma_1\Phi_1 = \nu\Sigma_{f1}\Phi_1 + \nu\Sigma_{f2}\Phi_2 + S \quad (9)$$

$$-D_2\nabla^2\Phi_2 + \Sigma_2\Phi_2 = \Sigma_m\Phi_1 \quad (10)$$

S includes delayed neutron source and a time differential term.

We have applied the modified one group method to solve eq. (9) and (10). Eq. (10) becomes as follows.

$$\Phi_2 = \frac{\Sigma_m}{\Sigma_2}\Phi_1 + \frac{D_2}{\Sigma_2}\nabla^2\Phi_2 \quad (11)$$

Eq. (11) can be written as follows when ∇^2 operates eq. (11).

$$\nabla^2\Phi_2 = \frac{\Sigma_m}{\Sigma_2}\nabla^2\Phi_1 + \frac{D_2}{\Sigma_2}\nabla^4\Phi_2$$

Because $\nabla^4\Phi_2 << \nabla^2\Phi_1$ the second term can be neglected.

$$\nabla^2\Phi_2 = \frac{\Sigma_m}{\Sigma_2}\nabla^2\Phi_1 \quad (12)$$

Substituting Eq. (12) into eq. (22), modified one group equation (13) is obtained.

$$-D_1\nabla^2\Phi_1 + \Sigma_1\Phi_1 = \nu\Sigma_{f,feff}\Phi_1 + Seff \quad (13)$$

In Eq. (13) $\nu\Sigma_{f,feff}$ and *Seff* are effective one group neutron production cross section and neutron source respectively including thermal neutron effect. In effective one group neutron calculation spatial neutron flux distribution can be obtained.

As mentioned above, in horizontal direction the only one mesh is assigned to one assembly. In order to reduce the coarse mesh error, the modified coarse mesh method[2)] is applied to the differentiation of right hand of eq. (13).

In usual simulation shape function is calculated every time when CORE3D is called in the simulator program, as the shape function is not changed fast shape function calculation is skipped but effective multiplication factor is calculated.

3. Verification

We have incorporated CORE3D into our PWR real-time simulator and verified its performance.

(i) Calculation speed

CORE3D is performed on a COMPAQ workstation. CORE3D is called in interval of 0.1 second. The amplitude calculation is executed at each call but the shape function calculation is executed at every 0.4 second. The CPU load is about 40% in this condition. It shows that a double-speed simulation can be realized.

(ii) Verification of static reactor physics parameters

We have verified CORE3D static performance by comparing with our nuclear design code ANC[3)], which is the three-dimensional diffusion nodal calculation code. Table 3-1 shows the comparison of zero power reactor physics parameters. Both codes show good agreement within the criteria used in Japanese reactor physics tests.

Next we have compared the power distributions at several reactor conditions. CORE3D shows good agreement with ANC in all cases. Figure 3-1 shows the comparison in all control

rod withdrawn condition.

(iii) Verification of dynamic reactor physics parameters

To verify kinetic calculation of CORE3D, we have evaluated the relationship between reactor period and reactivity. Table 3-2 shows the results. The reactivity evaluated by reactor period shows a good agreement with added reactivity.

We have performed the CORE3D verification in several transient events and confirmed that CORE3D has shown good behavior. As an example, Table 3-3 shows the decrease of reactor power and reactor power tilts with the one control rod drop. The results of CORE3D show a good agreement with evaluated values with ANC.

We have confirmed the validity of CORE3D from these verifications.

4. Conclusion

We have developed the three-dimensional kinetic code CORE3D for PWR real-time simulators. CORE3D has shown a good performance in calculation speed and calculation accuracy.

5. References

1) Y.RONEN：CRC Handbook of Nuclear Reactor Calculations, Vol.II, "Reactor Dynamics" ,CRC Press, Inc.(1986)
2) T.TAKEDA, Y.KOMANO: J.Nucl.Sci.Technol.15,523 (1978).
3) T.Q.Nguyen, et al.: Qualification of the PHOENIX-P/ANC Nuclear Design System for Pressurized Water Reactor Cores,WCAP-11596(1987).

Table 3-1 Comparison of zero power reactor physics parameters
Beginning of cycle, Hot zero power condition

Parameter	CORE3D	ANC	Difference	Criteria
Critical Boron Concentration (ppm)				
All rod out	1807	1816	-9	+-50ppm
Bank D in	1695	1698	-3	
Bank D,Cin	1587	1593	-6	
Moderator Temperature Coefficient(pcm/degree C)				
All rod out	-2.4	-4.8	+2.4	+-5.4
Control Rod Worth(pcm)				
Control Bank D	755	814	-7.2%	+-10%
Control Bank C	722	720	0.3%	

Table 3-2 Verification of CORE3D Kinetic calculation

	Reactivity	
	Positive	Negative
Added Reactivity (pcm),A	37.1	-32.9
Reactor Period(sec)	158.9	-261.2
Period Reactivity (pcm),B	37.5	-32.1
Difference(%) (A-B)/B	-1.1%	-2.4%
Criteria	+-4%	

Table 3-3 Example of power decreases and power tilts during control rod drop
Beginning of cycle

	ANC		CORE3D	
Drop Rod :P-8				
Power decrease	6.2%		5.7%	
Power Tilt	0.957	1.043	0.962	1.037
	0.957	1.043	0.965	1.036
Drop Rod :P-12				
Power decrease	5.7		6.0	
Power Tilt	1.021	1.043	1.022	1.049
	0.903	1.033	0.892	1.037
Drop Rod: H-8				
Power decrease	4.9		5.4	
Power Tilt	1.000	1.000	1.000	1.000
	1.000	1.000	1.000	1.000

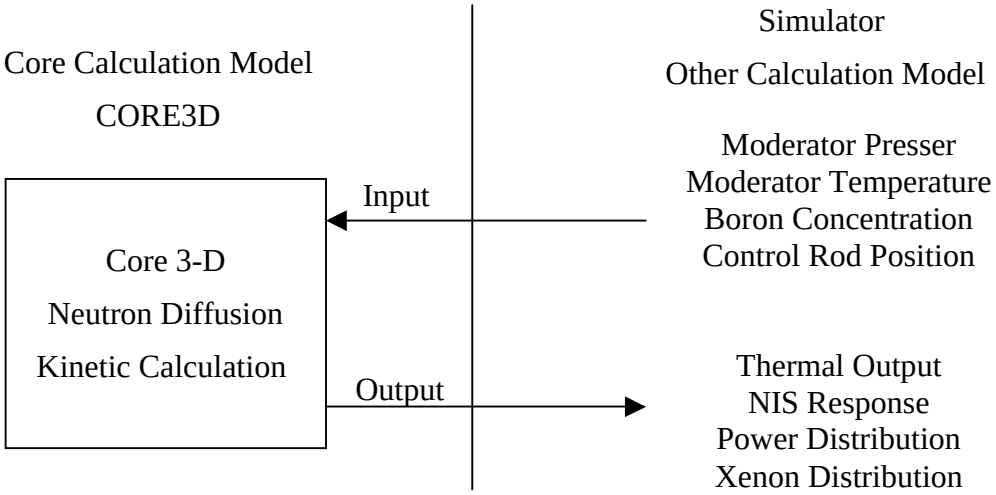

Figure 2-1 Interface between CORE3D and Other Calculation Model

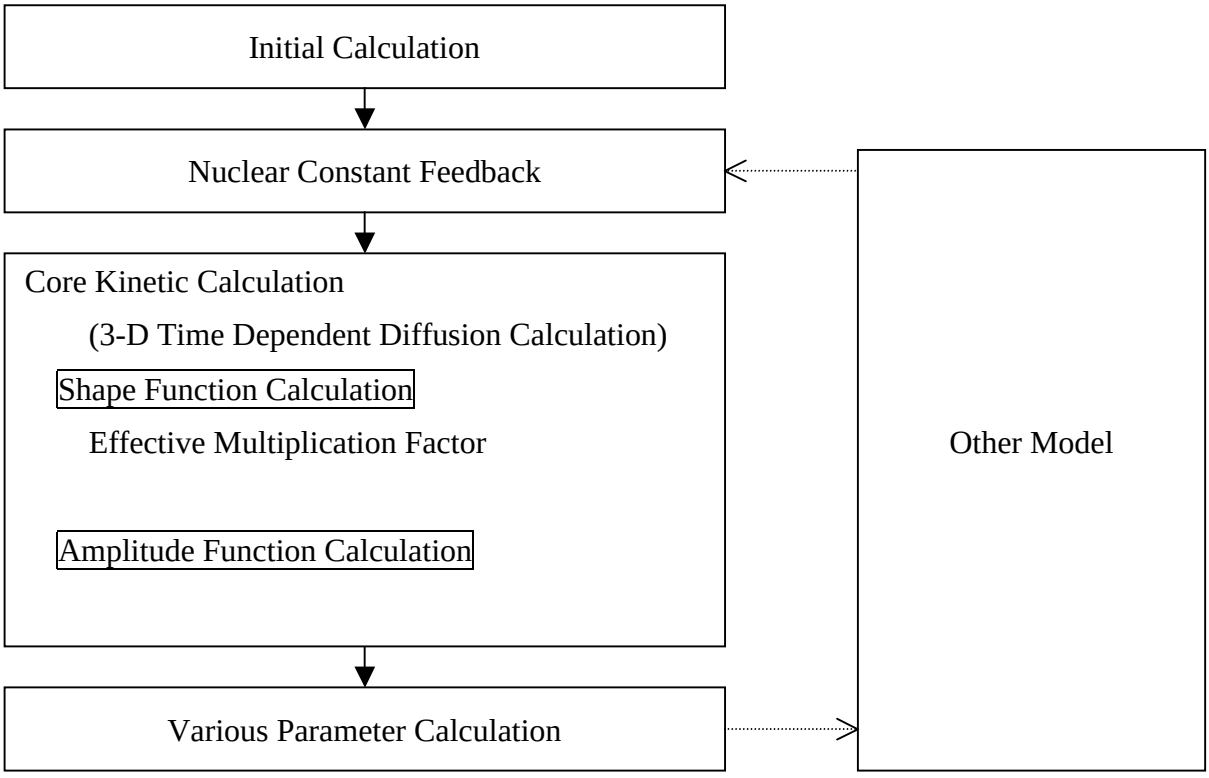

Figure 2-2 Calculation Flow of CORE3D

	R	P	N	M	L	K	J	H	G	F	E	D	C	B	A
1					3.8	1.8	8.4	9.4	5.8	-0.7	1.8				
2			-0.9	-2.9	0.4	1.7	5.3	1.9	1.4	-0.8	-1.7	-4.7	-2.6		
3		-1.3	-3.8	1.1	-2.4	2.8	0.7	-1.0	-1.5	0.5	-4.5	-0.7	-5.4	-2.4	
4		-3.2	1.0	3.4	2.6	-1.6	1.7	-0.4	-0.1	-2.9	0.8	1.2	-0.1	-4.6	
5	2.0	-0.3	-2.8	1.3	-3.1	3.3	-1.6	2.6	-2.9	1.6	-4.7	0.9	-4.7	-2.1	0.3
6	-0.4	-0.1	1.6	-2.1	2.6	-0.5	1.1	1.2	0.7	-1.2	1.6	-3.8	-0.8	-2.5	-2.6
7	5.6	1.4	-1.0	0.7	-2.4	0.2	-4.0	2.5	-3.8	0.2	-3.7	-2.3	-4.1	-1.5	2.9
8	8.0	0.2	-2.3	-0.9	1.9	0.6	2.8	-1.3	2.3	-0.3	0.5	-3.7	-4.7	-2.6	5.0
9	6.0	1.4	-1.1	0.4	-2.3	-0.5	-4.4	1.0	-5.3	-1.0	-4.3	-2.0	-3.2	-0.7	3.4
10	0.5	0.7	3.0	-0.8	3.2	-1.4	-0.7	-0.6	-0.8	-2.4	0.4	-4.0	-0.2	-1.9	-2.2
11	3.8	1.1	-1.2	4.3	-2.5	2.4	-3.2	1.0	-2.9	2.1	-4.2	1.4	-4.3	-1.7	0.3
12		-1.0	4.1	6.0	3.7	-1.5	0.1	-1.4	1.0	-1.5	2.7	2.4	0.0	-4.4	
13		2.1	-0.9	4.7	-0.6	3.4	0.0	-0.5	0.4	2.1	-3.0	0.5	-4.7	-2.4	
14			2.3	0.0	2.9	2.1	3.1	2.1	4.6	1.1	-0.3	-3.6	-1.8		
15					5.8	2.0	7.7	9.9	7.8	1.1	3.2				

Difference (%) = CORE3D - ANC

Figure 3-1 Comparison of assembly-averaged power distribution between CORE3D and ANC
Begging of cycle, Hot Zero power, All Rod out