Recent Progress on Fast Reactor Analysis in UNIST CORE Laboratory

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1. Introduction

The STREAM and RAST-K codes were designed by the Computational Reactor Physics and Experiment (CORE) laboratory and they were used as the accurate computational codes for light water reactor analysis. The fast reactor (FR) plays a very important role in the next generation reactor design and most of them are designed with hexagonal fuel assembly. In order to continuously develop our code system, the capability of solving neutron transport/diffusion equation with hexagonal geometry is implemented in STREAM and RAST-K codes. In addition, the discussion of different calculation methods for FR analysis is made.

2. Methods and Verifications

2.1 Hexagonal geometry in STREAM

The neutron transport equation along the trajectory of direction $\Omega_{n,m}$ can be written as:

$$\sin\theta_n \frac{d}{ds}\phi_g(s) + \Sigma_g(s)\phi_g(s) = Q_g(s) \qquad (1)$$

where, subscript g, n, and m stand for energy group, polar angle and azimuthal angle, respectively. s is projection of the trajectory on X-Y plane, ϕ is angular flux, Σ is macroscopic total cross section, and Q is neutron source.

The out angular flux of flat source region *i* and characteristics track *k* can be solved as:

$$\phi^{out}_{n,m,i,k} = \phi^{in}_{n,m,i,k} e^{-\Sigma_i s_{m,i,k}/\sin\theta_n} + \frac{Q_i}{\Sigma_i}(1 - e^{-\Sigma_i s_{m,i,k}/\sin\theta_n}) \qquad (2)$$

Therefore, the average angular flux can be obtained as:

$$\bar{\phi}_{n,m,i,k} = \frac{Q_i}{\Sigma_i} + \frac{\phi^{in}_{n,m,i,k} - \phi^{out}_{n,m,i,k}}{s_{m,i,k}\Sigma_i}\sin\theta_n \qquad (3)$$

In order to solving hexagonal problem, the modular ray tracing is implemented. The detailed information of developed algorithm to determine the modular ray parameters can be found in the reference [1].

2.2 Hexagonal geometry in RAST-K

The triangle-based polynomial expansion nodal method[2] was used in RAST-K code. Totally six radial and one axial equations are defined for the transverse-integrated neutron diffusion equations:

$$-D_g\left(\frac{\partial^2}{\partial x^2} + \frac{\partial^2}{\partial y^2}\right)\phi^{R,m}_g(x,y) + \Sigma_{r,g}\phi^{R,m}_g(x,y)$$
$$-\sum_{g'<g}\Sigma_{s,g'\to g}\phi^{R,m}_{g'}(x,y) = Q^{R,m}_g(x,y) - L^{Z,m}_g(x,y), \qquad (4)$$
$$m = 1\ldots6$$

$$-D_g \frac{\partial^2}{\partial z^2}\phi^Z_g(z) + \Sigma_{r,g}\phi^Z_g(z)$$
$$-\sum_{g'<g}\Sigma_{s,g'\to g}\phi^Z_{g'}(z) = Q^Z_g(z) - L^R_g(z) \qquad (5)$$

2.3 Verifications

To verify the accuracy of STREAM code, the C5G7 hexagonal variation problem is selected. **Figure 1** and **Figure 2** show the assembly and core information of verification test. The cross sections set is same as the original C5G7 benchmark.

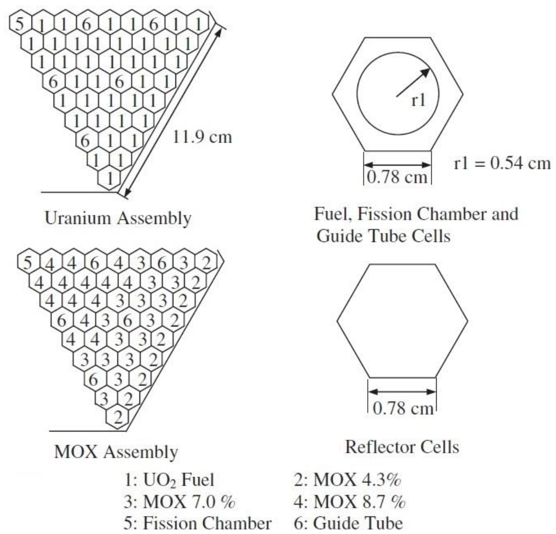

Figure 1 Assembly information of C5G7 hexagonal problem

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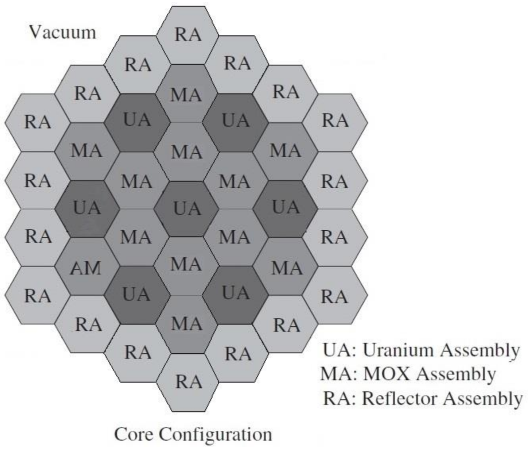

Figure 2 Core configuration of C5G7 hexagonal problem

Table 1 and Figure 3 show the results of verification. The reference calculation is performed with McCARD Monte Carlo code. In the STREAM calculation, 48 azimuthal angles are used and the ray distance is 0.05 cm. Compared with reference solutions, the STREAM code obtains accurate results. The differences of k_{eff} value between STREAM and McCARD is only 8 pcm. The maximum difference of assembly power distribution is only 0.43%.

Table 1 k_{eff} result of C5G7 hexagonal problem

	McCARD	STREAM	Diff., [pcm]
k_{eff}	1.16243±0.00009	1.16251	8

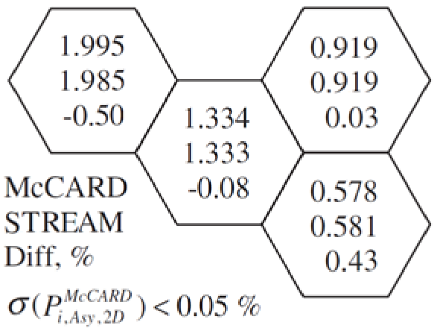

Figure 3 1/6 assembly power distribution and its differences

The OCED-NEA SFR numerical benchmarks[3] are selected to verify the accuracy of upgraded RAST-K code. In order to only perform the verifications of diffusion solver, the results are compared with the PARCS code by using same cross sections sets. The multigroup homogenized cross sections are generated by TULIP[4] code, which implemented in SARAX[5] code for fast reactor analysis.

Table 2 summarize the k_{eff} results of different core designs. For both cases, the differences are quite small. The largest one is only 6 pcm and other cases have only 1 or 2 pcm differences.

Table 2 k_{eff} comparison of RAST-K and PARCS

Benchmark	RAST-K	PARCS	Diff., [pcm]
MET-1000	1.00745	1.00747	-2
MOX-1000	1.01349	1.01350	-1
MOX-3600	1.00141	1.00147	-6
CAR-3600	0.99438	0.99439	-1

3. Application

In this section, we would like to discuss the impact of different method during the FR numerical simulation. To perform this discussion, several calculations of a 2-D pseudo core based on MET-1000 core design are analyzed. Figure 4 shows the layout of the 2-D core. The geometry and composition are same as that of the benchmark report.

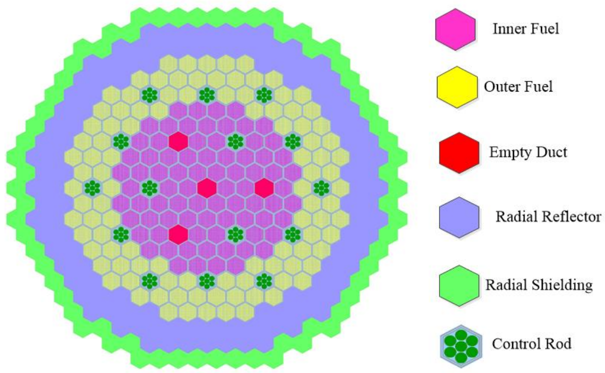

Figure 4 Layout of 2-D pseudo core problem

During the calculation, both Un-Rodded and Rodded cases are considered. For Un-rodded case, all the control rod assemblies are replaced by empty duct assembly. For Rodded case, only the primary control rod system is inserted in the core. In addition, the sodium voided (SV) case and Doppler case are also calculated. In the SV case, the sodium density of fuel region was changed to very small number to simulate sodium void. In the Doppler case, the fuel temperature was doubled. The reference state of SV case and Doppler case is Rodded case.

Four different ways to get the final k_{eff} results are discussed: 1) MCS Monte Carlo continuous energy calculation as the reference; 2) TULIP/STREAM calculation, cross sections are prepared with pin-wise, direct 2-D core calculation with explicit heterogeneous geometry description; 3) TULIP/SARAX calculation, cross sections are prepared by assembly homogenization, 2-D core calculation is performed by S_N transport nodal code; and 4) TULIP/RAST-K calculation, cross sections are still prepared by assembly homogenization, 2-D core calculation is performed by diffusion nodal code. 33-group and 195-group cross sections set under out-flow transport correction are prepared for each calculation. In the TULIP/STREAM calculation, the pin-wise cross sections of fuel, cladding, coolant and duct are calculated based on equivalent 1-D cylindrical assembly model. The k_{eff} and pin-wise power distribution results of each case are summarized in the Table 3 and Table 4.

Table 3 k_{eff} results of each calculation and its difference

Case	Energy Group	Code	k_{eff}	Diff., [pcm]
Un-rodded	CE	MCS	1.23895 ± 0.00003	
	195	TULIP/STREAM	1.23687	-135
		TULIP/SARAX	1.23816	-51
		TULIP/RAST-K	1.23577	-188
	33	TULIP/STREAM	1.23418	-312
		TULIP/SARAX	1.23532	-237
		TULIP/RAST-K	1.23304	-387
Rodded	CE	MCS	1.05816 ± 0.00003	
	195	TULIP/STREAM	1.05889	65
		TULIP/SARAX	1.05961	129
		TULIP/RAST-K	1.04773	-941
	33	TULIP/STREAM	1.05772	-39
		TULIP/SARAX	1.05797	-17
		TULIP/RAST-K	1.04561	-1134
SV	CE	MCS	1.09700 ± 0.00003	
	195	TULIP/STREAM	1.09735	29
		TULIP/SARAX	1.09910	174
		TULIP/RAST-K	1.08316	-1164
	33	TULIP/STREAM	1.09585	-96
		TULIP/SARAX	1.09707	6
		TULIP/RAST-K	1.08158	-1299
Doppler	CE	MCS	1.05657 ± 0.00003	
	195	TULIP/STREAM	1.05659	2
		TULIP/SARAX	1.05731	66
		TULIP/RAST-K	1.04549	-1003
	33	TULIP/STREAM	1.05538	-106
		TULIP/SARAX	1.05559	-87
		TULIP/RAST-K	1.04418	-1123

Table 4 The differences of pin-wise power distribution between TULIP/STREAM and MCS

Case	Energy Group	Max., [%]	Min., [%]	RMS, [%]
Un-rodded	195	1.96	-2.64	0.96
	33	3.44	-4.71	2.09
Rodded	195	3.01	-4.76	1.76
	33	4.67	-7.31	3.26

According to the results of Table 3, some conclusions can be drawn. First, compared with TULIP/STREAM and TULIP/SARAX, the homogenization shows the positive contribution on k_{eff} about 100 pcm. It is not sensitive to the case. Second, the reduction of energy group number shows the negative impact on k_{eff} value. The largest underestimation is about 200 pcm and at least 100 pcm differences can be found between 195-group and 33-group calculation. Third, the diffusion theory always underestimates k_{eff} value compared with transport theory. Obviously, the underestimation is sensitive to the case. For the Un-rodded case, the differences between diffusion and transport theory is less than 200 pcm. However, for the other cases which has the control rod assembly inserted in the active region, the differences become larger. The maximum number is 1338 pcm of SV 195-group calculation. In the fast reactor, the mean free path is bigger than 10 cm, the neutron can be generated in one assembly and disappeared in another assembly. Therefore, the non-absorption assumption of Fick's law is not satisfied in those cases. It should be noticed that the angular dependency of total cross sections is considered in TULIP code. If we don't consider the angular dependency of cross sections and apply diffusion theory for core calculation, we can obtain good results due to an interesting error cancellation.

From Table 4 we can find that 195-group calculation improves the accuracy of pin-wise power distribution so much. Although in some cases 33-group calculation obtains better k_{eff} results than that of 195-group calculation, it is believed that error cancellation occurs. The 195-group calculation will always give better results.

4. Conclusions

In this paper, the capabilities of dealing with hexagonal geometry are implemented in the current STREAM and RAST-K code. The modular ray tracing algorithm and the triangle-based polynomial expansion nodal method are used in those codes separately. Verifications show the accuracy of developed solver.

In addition, the evaluation of different methods in fast reactor analysis is performed based on the new code. The impact of energy group number, transport/diffusion theory, and modeling are quantified. It indicates that better results can be obtained by using appropriate method.

ACKNOWLEDGEMENT

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