A Comparative Depletion Analysis using MCNP6 and REBUS-3 for Advanced SFR Burner Core

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1. Introduction

The code system validation for fast reactor design is one of the important research topics. In our previous studies, depletion analysis and physics parameter evaluation of fast reactor core were done with REBUS-3 code [1] and DIF3D code [2], respectively. In particular, the depletion analysis was done with lumped fission products. However, it is need to verify the accuracy of these calculation methodologies by using Monte Carlo neutron transport calculation coupled with explicit treatment of fission products [3,4]. In this study, the accuracy of fast reactor design codes and procedures were evaluated using MCNP6 code [5] and VARIANT nodal transport calculation [6] for an initial cycle of an advanced sodium-cooled burner core loaded with uranium-free fuels.

2. Core Design and Results

2.1 Specification of SFR burner core

The SFR burner core rates 300MWe (762MWth). The core uses uranium-free metallic fuels of TRU-W-10Zr to maximize TRU burning rate. In particular, tungsten was added into the fuel to enhance the Doppler coefficient because uranium free fueled cores can have too small or positive Doppler coefficient. Table I summarizes the design parameters of the core. The core is loaded with two different type fuel assemblies for power flattening under a single feed fuel composition. The normal 271 rod assemblies are loaded into the outer core region while new 217 rod assemblies with thick duct are loaded into the inner core. The duct thickness for normal and the thick duct assemblies are 3.7mm and 11.5mm, respectively. Also, each fuel assembly has 12 moderator rods ($ZrH_{1.8}$) to improve Doppler coefficient and to reduce sodium void worth. Fig. 1 compares the normal and new type fuel assemblies. The radial core configuration is shown in Fig. 2. Also, axially central B_4C regions are considered in order to reduce burnup reactivity swing by increasing initial heavy metal loading [7]. The active fuel and B_4C absorber lengths are 72.8cm and 18.2cm, respectively. The axial cutview of the core is shown in Fig. 3.

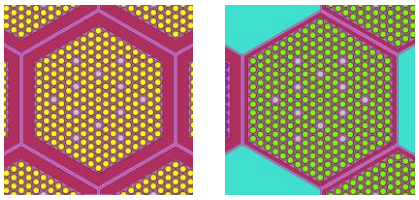

Fig. 1. Configurations of normal (right) and new fuel (left) assemblies having thick duct

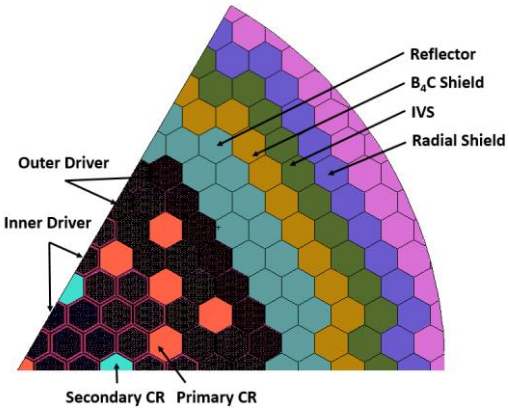

Fig. 2. Configuration of the reference core (1/6)

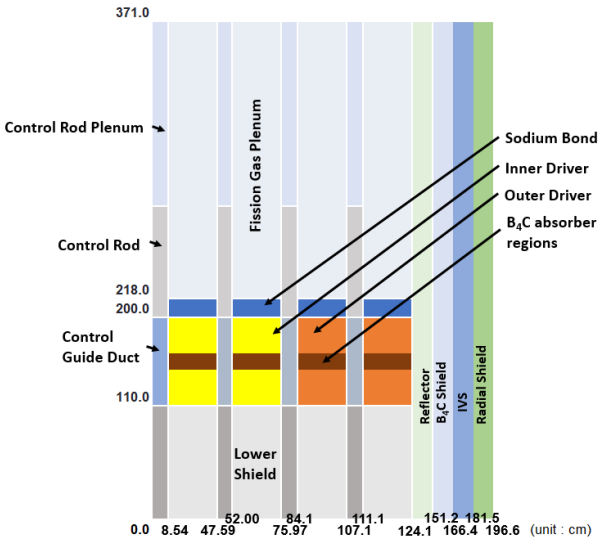

Fig. 3. The axial cutview of the reference core

2.2 Calculation Methods

Our previous work on the SFR burner core design [7,8,9,10] was done with REBUS-3 equilibrium cycle model and 9 group cross sections for depletion analysis. The core physics parameters were analyzed with 80 group cross section and DIF3D HEX-Z nodal diffusion option. In this work, it was determined to analyze the initial core characteristics by using REBUS-3 non-equilibrium cycle model and MCNP6 depletion with a fixed initial fuel composition and depletion time for clarity of the composition. The MCNP6 depletion analysis was performed with the separated considerations of heterogeneous and homogeneous fuel assembly models to show the heterogeneity effects. The point-wise cross-sections for MCNP6 were generated based on ENDF/B-VII.R0. The tier3 option for the fission product treatment was applied to consider the detailed depletion chain of the fission products. The core was divided into 20 material regions (4 radial regions and 5 axial regions) to consider the dependency of one-group cross sections used in depletion calcualtion. The initial TRU content in fuel is 38.25wt% and the TRU composition is from the LWR spent fuel having 50MWD/kg and 10 years cooling. The following different methods with the same 80 group cross sections were considered and their results were inter-compared : 1) HEX-Z nodal diffusion option, 2) VARIANT SP_3 nodal option (REBUS-PC 1.4) [11], and 3) VARIANT nodal P_3 transport option(REBUS-PC 1.4) with isotropic scattering option. The 80 group cross sections for REBUS-3(or REBUS-PC 1.4) depletion analysis were produced with TRANSX[12] and ENDF/B-VII.R0 based MATXS library. The core region-wise spectra estimated with TWODANT[13] were used for group collapsing. These 80 group cross sections were also used in the estimation of sodium void worth and control rod worth.

Table I Design specifications

Design parameters	Specification
Power(MWe/MWt)	300/761.7
Fuel type	TRU-W-10Zr
Number of fuel rods per FA(inner/outer)	205/259
Number of moderator rods per FA(inner/outer)	12/12
Smear density of fuel	75%
Duct wall thickness(mm) (inner/outer)	11.5/3.7
Assembly pitch(cm)	16.22
Rod outer diameter(mm)	7.5
Number of fuel assemblies	174
Active driver height(cm, cold)	72.8
Equivalent core radius(cm)	112.8
Average linear power density(W/cm)	254
Volume fraction(fuel/coolant/structure/moderator)	
Inner core	28.95 / 30.80 / 38.56 / 1.69
Outer core	36.57 / 36.92 / 24.82 / 1.69

Table II Fuel assembly data for MCNP6 modeling

Parameters	Homo (Inner/Outer)	Hete (Inner)	Hete (Outer)
Assembly pitch (cm)		16.22	
Duct outer flat-to-flat distance (cm)	-	15.82	
Duct wall thickness (cm)	-	1.15	0.37
Duct inside flat-to-flat distance (cm)	-	13.52	15.08
Fuel pin data			
Number of pins	-	217	271
Fuel pin pitch (cm)	-	0.9	
Outer radius of clad (cm)	-	0.375	
Inner radius of clad (cm)	-	0.320	

2.3 Results of Comparative Analysis

Table III compares the depletion results calculated using REBUS-3(nodal diffusion method), REBUS-PC 1.4(VARIANT Simplified Spherical Harmonics option (SP_3) and VARIANT nodal transport option (P_3)), and MCNP6. This table shows that REBUS-3 nodal diffusion calculations give a large discrepancy of

2131pcm in the initial excess reactivity in comparison with MCNP6 depletion calculation using heterogeneous fuel assembly model, which means that the depletion calculations with nodal diffusion method for the core having large heterogeneities can cause considerable errors. On the other hand, the REBUS-PC 1.4 calculation (80 group) using VARIANT SP_3 option, and VARIANT nodal transport give the smaller discrepancies in the initial reactivity of 987pcm and 784pcm respectively, than the REBUS-3 nodal diffusion in comparison with the heterogeneous MCNP6 calculation. It is noted that the MCNP6 depletion calculation using homogeneous fuel assemblies give a discrepancy of 662pcm in the initial excess reactivity in comparison with the MCNP6 heterogeneous fuel assembly model. Also, it is noted that the VARIANT and MCNP6 using homogenous model have very similar initial effective multiplication factors to each other (i.e. 122pcm). Fig. 4 compares the eigenvalue evolutions obtained with different codes and options. As shown in Fig. 4, the effective multiplication factors linearly decrease as depletion time and the discrepancies in the effective multiplication factors were maintained to be nearly constant during the depletion. So, all the calculations give very similar burnup reactivity swings (see Table IV).

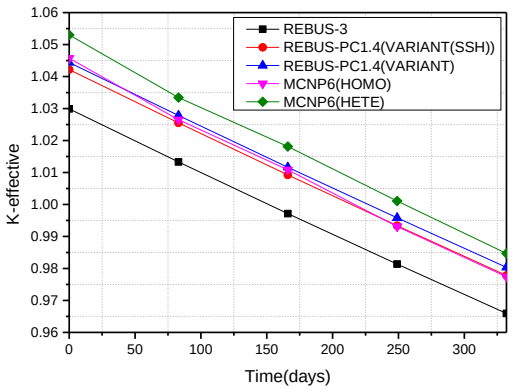

Fig. 4. Comparison of evolutions of burnup

Table III Comparison of the depletion calculations evaluated with REBUS-3, VARIANT, and MCNP6

	REBUS-3 (Nodal diffusion, 80 Group)	REBUS-PC 1.4 (VARIANT(SSH), 80 Group)	REBUS-PC 1.4 (VARIANT, 80 Group)	MCNP6 (HOMO)	MCNP6 (HETE)
Day			K-effective(ENDF7.R0)		
0	1.029911	1.042188	1.044404	1.04574	1.05303
83	1.013301	1.025504	1.027831	1.02652	1.03343
166	0.9971328	1.009241	1.011651	1.01085	1.01813
249	0.9813589	0.993362	0.995841	0.9931	1.00109
332	0.9659184	0.977805	0.97738	0.97738	0.9847
Burnup(MWD/kg)	56.95	56.95	56.95	56.80	56.84

Table IV compares the control rod worth and sodium void reactivity worth calculated using DIF3D (nodal diffusion method, 80group), DIF3D10.0 (VARIANT SP_3 option, 80group), and MCNP6 for the reference core. The reactivity worth of the primary control assemblies at BOC were accurately estimated and the discrepancies were estimated to be ~10% in comparison with the reference heterogeneous MCNP6 calculation. For sodium void reactivity worth at BOC, the homogeneous MCNP6 calculation gave quite accurate results whose discrepancy is ~2% but DIF3D nodal diffusion and VARIANT nodal transport option gave larger discrepancies of 12% and 17%, respectively. The assembly-wise normalized power distributions at BOC for reference core using REBUS-3 and MCNP6 are inter-compared in Fig. 5. For MCNP6 code, F7 tally were used to obtain normalized power distribution in assemblies power. As shown in Fig. 5, the percentage difference between REBUS-3 nodal diffusion option and MCNP6 heterogeneous assembly model is less than 2.1%.

Table IV Comparison of the core performances evaluated with DIF3D (nodal diffusion), DIF3D10.0 (VARIANT), and MCNP6 (HOMO,HETE)

	DIF3D (Nodal diffusion, 80 Group)	DIF3D10.0 (VARIANT, 80 Group)	MCNP6 (HOMO)	MCNP6 (HETE)
Burnup reactivity swing (pcm)	6432.6 (2.38%)	6565.96 (0.36%)	6688.28 (1.49%)	6589.71
Control assembly worth (pcm)				
Primary	10625.5 (11.1%)	10419.0 (8.9%)	10517.1 (9.9%)	9562.2
Secondary	3358.3 (9.0%)	3171 (2.9%)	3312.7 (7.5%)	3081
Sodium void worth(pcm)	444.6 (12.2%)	592.7 (16.9%)	516.5 (1.9%)	506.6

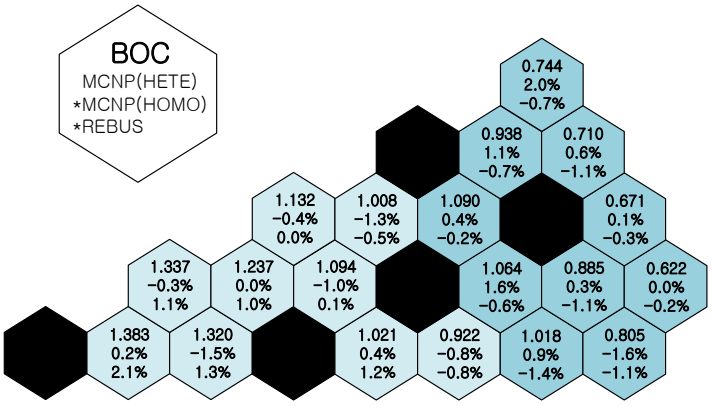

Fig. 5. Comparison of the assembly-wise power distribution at BOC

3. Conclusions

In this paper, we evaluated the accuracy of fast reactor design codes by comparing with MCNP6-based Monte Carlo simulation and REBUS-3-based the nodal transport theory for an initial cycle of an advanced uranium-free fueled SFR burner core having large heterogeneities. It was shown that the nodal diffusion calculation in REBUS-3 gave a large difference in initial k-effective value by 2132pcm when compared with MCNP6 depletion calculation using heterogeneous model. However, the differences between MCNP6 depletion calculation using heterogeneous fuel assembly model and the REBUS-PC 1.4 calculations (80 group) using VARIANT SP_3 and VARIANT nodal transport option were observed to be less than 1000pcm. Therefore, it was considered that the REBUS-3 nodal diffusion option can not be used to accurately estimate the depletion calculations and VARIANT nodal transport or VARIANT SP_3 options are required for this purpose for this kind of heterogeneous burner core loaded with uranium-free fuel. The control rod worths with nodal diffusion and transport options were estimated with discrepancies less than 12% while these methods for sodium void worth at BOC gave large discrepancies of 12.2% and 16.9%, respectively. It is considered that these large discrepancies in sodium void worth are resulted from the inaccurate consideration of spectrum change in multi-group cross section.

ACKNOWLEDGMENT

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