M&C 2001 Salt Lake City, Utah, USA, September 2001.

Application of an Advanced MCNP Technique to Analysis of Nuclear Characteristics in Reactor Core

Chi Young HAN, Chang Ho SHIN, and Jong Kyung KIM
Hanyang University
17 Haengdang, Sungdong, Seoul, 133-791, Korea
charlie@nural.hanyang.ac.kr, gemini@nural.hanyang.ac.kr, and
jkkim@nural.hanyang.ac.kr

Won Seok PARK
Korea Atomic Energy Research Institute
150 Dukjin, Yusong, Taejon, 305-353, Korea
wonpark@kaeri.re.kr

Keywords: Monte Carlo, Advanced MCNP Technique, Core Characteristics

ABSTRACT

An advanced MCNP simulation technique has been suggested to reduce computing time in Monte Carlo calculations for a reactor core analysis. This technique is performed in two steps using the SSW and SSR cards options, geometry cut-off, fission multiplying option, etc., in MCNP4B code, and is able to represent explicitly albedo effects in specified surfaces. In this paper, the technique was applied to calculating pin power distributions in the peak power assembly of APR1400 and HYPER cores, and the results were compared with the values calculated by the conventional method. The relative pin power distributions of both cores yielded the root mean square errors of 1.86% and 2.80%, respectively, as compared with the reference values. The results showed that the method suggested could construct an equivalent reactor physics model reducing computer time remarkably.

1. INTRODUCTION

In the analyses of neutron behavior within reactor core, deterministic methods are widely used. Those involve fundamental uncertainties because of the approximations in group cross-sections, finite mesh size, cell homogenization, resonance self-shielding factor, and so on.

On the other hand, Monte Carlo method is almost free from the approximations such as the above and can give an accurate solution of transport equation by simulating with continuous energy cross-section data without any discretization of space, energy, and angle. At the same time, its general geometry modeling capability permits easy modeling of core experiments or facilities. Based on this reason, Monte Carlo method is currently getting more attention in some cases of reactor core calculations in spite of enormous computing time. The MCNP Monte Carlo code has offered the capability of performing complicated reactor physics calculations not possible with most deterministic methods (Redmond II, 1991) (Redmond II, 1994) (Peirre, 1999) (Joneja, 2001) and proved superior in calculating some core physics characteristics to deterministic methods (Redmond II, 1991).

The MCNP was also applied to the reactor pressure vessel neutron fluence calculation in order to solve a problem more accurately (Laky, 1995) (Wagner, 1996), and to full-scope explicit simulation of a commercial reactor core by a pin-by-pin model (Kim, 1998) (Seo, 2001). The Monte Carlo technique, however, requires significant amount of computing time in order to obtain statistically reliable results. This makes it difficult to apply Monte Carlo method to complex and large-scale system such as a commercial reactor core system. To put Monte Carlo calculations for a reactor core to practical use, remarkable reduction in computing time is necessary.

In this relation, this work intended to develop an advanced technique to reduce computing time in the Monte Carlo calculation for a reactor core. The attention was given to the detailed calculation in a specific local important region of reactor. As an example, a design optimization of LLFP (Long-Lived Fission Product) assembly is currently being investigated to maximize the transmutation capability of Tc-99 and I-129 in the HYPER (HYbrid Power Extraction Reactor) core at KAERI (Korea Atomic Energy Research Institute). It is required to analyze the neutronic effects of the LLFP assembly on the reactor core, and particularly the pin power peaking within the assemblies adjacent to the LLFP assembly in safety point of view. Though an interest in nuclear characteristics within only a local region of reactor core (e.g., a fuel assembly) is taken, the whole core geometry model is required. Particle transport simulations over this whole core geometry would require undesirable huge computing time even if various variance reduction techniques would be used. If we can construct an equivalent reactor physics model to simulate intensively neutron transport only within an interesting local region and to simulate roughly within the other whole core regions, the computing time would be reduced.

For constructing the equivalent reactor physics model to reduce computing time, an advanced simulation technique using the SSW (Surface Source Write) and SSR (Surface Source Read) cards options, geometry cut-off, fission multiplying option, etc. in MCNP4B code is introduced. The technique suggested is applied to calculating power peaking factors and pin power distributions of peak power assembly for APR1400 (Advanced Power Reactor 1400) and HYPER cores in order to investigate the reduction in the computing time and the credibility in the results.

2. AN ADVANCED MCNP SIMULATION TECHNIQUE

The SSW card of MCNP code is used to write a surface source file or KCODE (criticality calculation) fission volume source file for use in a subsequent MCNP calculation using the SSR card. It is, sometimes, desirable to solve a problem with the transformation of a multiplying problem of power reactor core into the fixed source problem. For example, an operating reactor power distribution could be specified as a fixed source distribution in the core by writing fission volume sources from KCODE calculation to a source file with the SSW card. In the subsequent non-KCODE calculation using the SSR card, a NONU card that allows turning off of fission event in any cell and then treats the fission as simple capture so that fission neutrons and photons should be not counted twice is employed.

The procedure of the advanced MCNP simulation technique is perform in the two steps, differently with the conventional procedure that carries out one step KCODE calculation for the whole core MCNP model. The procedures for the conventional calculation, here, named as reference calculation and the advanced calculation are simply compared in Figure 1.

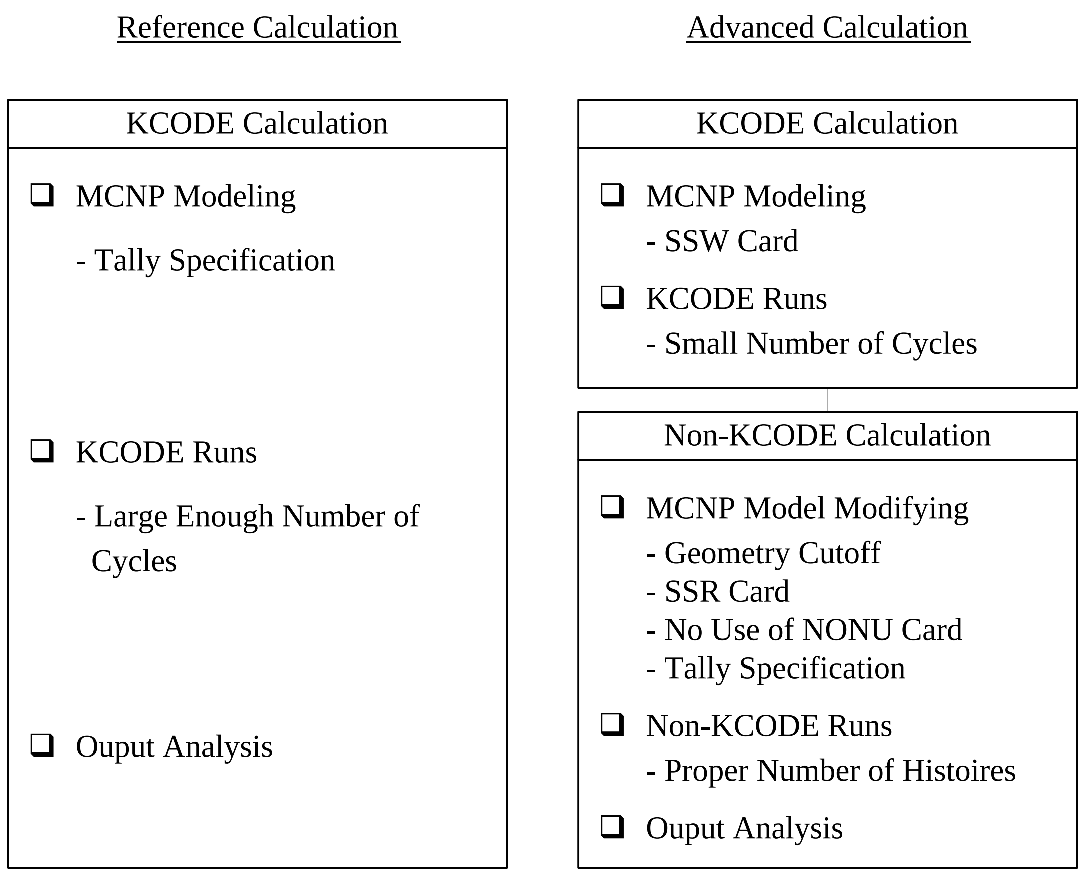

Fig. 1 Comparison of Procedures between the Reference and Advanced Calculations

In the first step (KCODE calculation) of the advanced technique, the technique introduces the use of SSW card for surfaces constructing geometrically one fuel assembly instead of fission volume source obtained in KCODE calculation for a whole core. The full-scope MCNP model for a reactor core as the reference calculation is constructed, but it is used only to generate a surface source file with a small number of KCODE cycles. The SSW card can record tracks of particles that cross the surfaces in the correct direction and enter the fuel assembly. Since the recorded surface source file contains the position, direction, energy, weight, and all other information necessary to run a subsequent job, it can account for albedo effects of the specified surfaces such as a boundary condition if the reactor core model includes enough geometry far beyond the surfaces. In the second step (non-KCODE calculation), only the interesting fuel assembly cell is chosen and the use of NONU card recommended strongly in MCNP code manual is not employed for neutron balance. Therefore, the source particles sampled in the surface source file using the SSR card are transported to either inward or outward the fuel assembly and all particles leaving the cell are killed immediately upon entering the other zero importance cells acting as a geometry cut-off.

3. APPLICATION TO APR1400 CORE

3.1 Core Description

The APR1400 is an advanced PWR being planned to produce thermal core power of 3,983 MW. The general core description is given in Table 1. The reactor core is composed of 241 fuel assemblies arranged with an effective core diameter of 365.8 cm and an active fuel length of 381 cm. Each fuel assembly consists of a 16×16 array of 241 fuel rods and 5 guide tubes (water holes), which each displace 4 fuel rod positions.

Table 1 APR1400 Core Description

Total Power	3983 MWth
Active Core Height	381 cm
No. of Fuel Assemblies	241
Lattice in Assemblies	16×16
Effective Core Diameter	365.8 cm
Fuel Assembly Pitch	20.88 cm
Fuel Rod Pitch	1.285 cm
Fuel Rod Diameter	0.826 cm
Clad Inner Diameter	0.843 cm
Clad Outer Diameter	0.970 cm

Many assemblies contain zones of lower enrichment fuel pins. Due to the use of various enrichment zoning patterns and burnable poison shim loadings, the fuel assemblies in the initial core are classified into 6 different sub-batches as listed in Table 2, although only 4 different fuel pin enrichments (1.60, 2.78, 3.28, 3.78 w/o) are used. The shim is comprised of gadolinia (Gd_2O_3) admixed in natural urania (UO_2) and the axial cut back regions where no gadolinia is admixed are introduced in the top and bottom 5% of the shim rod.

Table 2 APR1400 Fuel Assemblies Description

FA Type	No. of FAs	Fuel Rod Enrichment	No. of Fuel Rods per Assembly	No. of BP Rods per Assembly	Gd_2O_3 Weight %
A0	81	1.60	236	–	–
B0	28	3.28	236	–	–
B1	48	3.28/2.78	172/52	12	8
B2	4	3.28/2.78	124/100	12	8
C0	20	3.78/3.28	184/52	–	–
C1	60	3.78/3.28	120/100	16	8

3.2 MCNP Model

The APR1400 was modeled in full three-dimensional detail to minimize the number of approximations. The fuel elements (i.e., fuel, clad, gap, and coolant) were explicitly modeled to eliminate inaccuracy contribution resulting in homogenization. The model was extended to the upper and lower core supports, which are sufficient to account for the returning neutrons. The reactor core was modeled on 1/8 core in the azimuthal direction using symmetrical property and divided into 20 segments in the axial direction. Figure 2 is cross-sectional views of the APR1400 cycle 1 core as modeled and displayed using MCNP. Both figures show center and periphery regions of the 1/8 core including 38 fuel assemblies in respective.

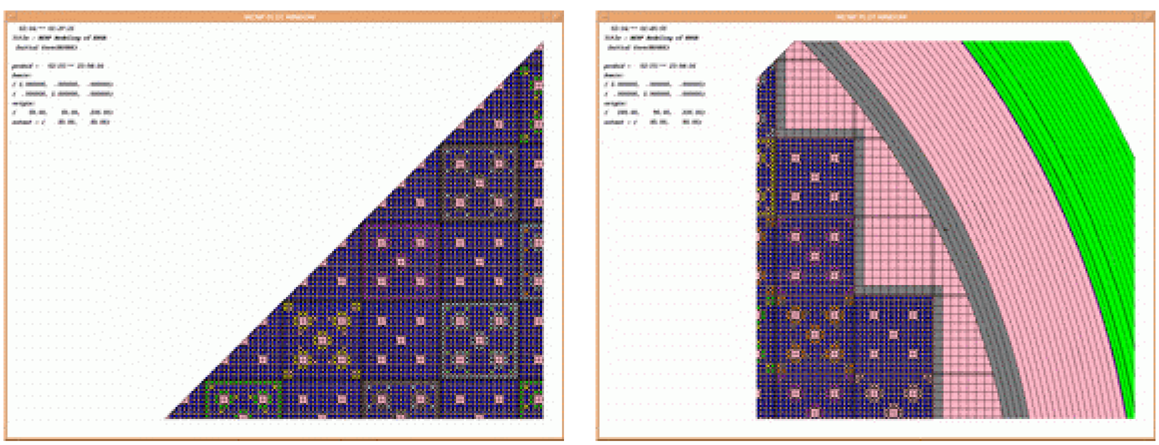

Fig. 2 Cross-sectional Veiws of the APR1400 MCNP Model (Center and Periphery)

The atomic number densities including uranium, actinides, and fission products were calculated by running CASMO-3 code at BOC (50 MWD/MTU) and xenon equilibrium state for all fuel assemblies. The densities were specified as the fuel composition input data of MCNP code. The coolant density was calculated by considering the coolant temperature and pressure and then the atomic number density within the coolant was calculated at borated water (893 ppm) condition. 10 spacer grids

(Zircarloy-4) and 1 bottom spacer grid (Inconel 625) were considered as the fuel assembly structures and then the grids were smeared into the neighboring moderator region. The coolant average temperature of 310.6 °C was used.

In this work, cross sections for different in-vessel core temperatures were generated by using ENDF/B-VI (release 4) library and NJOY 97 code, and named as APR1400 cross section library (APR1400XS). For all fuel materials, the average fuel temperature, 701.4 °C, was used and for the coolant materials including boron and homogenized grids, the average coolant temperature was used. It was assumed that the cladding temperature was 339 °C and the temperature of other structure materials except the cladding was equal to the coolant temperature.

3.3 Calculation and Comparison

To verify the MCNP model for APR1400 core, the criticality calculation was performed for cycle 1, BOC, ARO (All Rods Out), HFP (Hot Full Power), and equilibrium xenon configuration. The predicted value of k_{eff} was 1.00582±0.00014 (1σ) and indicates the correct MCNP model of the APR1400.

Core RPD (Relative Power Distribution) can be calculated by using F7 tally in the KCODE mode. In the reference calculation, the F7 tally was performed for all pins of 38 fuel assemblies and the statistical error for each pin could be reduced less than 2% through KCODE runs of 2,000 cycles. The approximate running time was 2,991 CPU minutes on a LINUX workstation (PIII-1GHz, 512MB). The RPD for initial core (BOC) of APR1400 cycle 1 calculated is within 2.53% of RMS (Root Mean Square) error, compared with the values predicted by KNFC (KEPCO Nuclear Fuel Company) and pictured in Figure 3. The radial pin peaking factor (F_r) was also calculated. The F_r is 1.4359 and appeared on 28th fuel assembly (B1 type). The F_r predicted by KNFC calculation is 1.4491 and is within 0.91% error as compared with present result.

In the advanced calculation, the surface source writing was performed with KCODE runs of 200 cycles and took about 113 CPU minutes on the same machine. The pin power distribution was calculated for the 28th fuel assembly with the source and took about 245 CPU minutes. Therefore, total 358 CPU minutes were taken in the advanced calculation of two steps and the result showed the computing time reduction of about 8.4 times.

Both results are compared with some integral parameters in Table 3 and pin power distributions at the 28th fuel assembly are depicted in Figure 4. The advanced calculation produced the fuel assembly power and the average statistical error similar to those of the reference calculation. The relative pin power distribution of the advanced calculation yielded RMS error of 1.86% compared with the values calculated in the reference. This indicates that, if using the advanced technique, an equivalent reactor physics model can be constructed with a shorter computing time.

The pin peaking location within the fuel assembly appears to be different with relative peaking factor of 1.226 and 1.213 respectively

AA NN X.XXXX X.XXXX X.XXX%	AA: FA Type NN: FA No. MCNP Design Error(%)*					C0 38 0.7061 0.6974 1.247%		
					A0 35 0.8565 0.8624 -0.684%	C1 36 0.9327 0.9541 -2.243%	B0 37 0.6558 0.6600 -0.636%	
				A0 31 0.9213 0.9036 1.959%	B1 32 1.1618 1.1503 1.000%	B1 33 1.1005 1.1143 -1.238%	C0 34 1.0407 1.0971 -5.141%	
			A0 25 0.9044 0.8725 3.656%	C1 26 1.1766 1.1519 2.144%	A0 27 0.9248 0.9177 0.774%	B1 28 1.1619 1.1727 -0.921%	B1 29 1.0182 1.0522 -3.231%	B0 30 0.6913 0.6829 1.230%
		A0 18 0.9169 0.8850 3.605%	C1 19 1.1668 1.1343 2.865%	A0 20 0.8958 0.8741 2.483%	C1 21 1.1714 1.1611 0.887%	A0 22 0.9169 0.9242 -0.790%	B1 23 1.1200 1.1561 -3.123%	C0 24 0.9701 1.0165 -4.565%
	A0 10 0.9254 1.0052 -7.939%	B1 11 1.1699 1.1342 3.148%	A0 12 0.8972 0.8749 2.549%	C1 13 1.1473 1.1308 1.459%	A0 14 0.8845 0.8786 0.672%	C1 15 1.1623 1.1722 -0.845%	A0 16 0.9083 0.9355 -2.908%	B0 17 0.9906 1.0060 -1.531%
A0 1 0.9332 0.8957 4.187%	B0 2 1.1439 1.1142 2.666%	A0 3 0.9136 0.8924 2.376%	C1 4 1.1553 1.1342 1.860%	A0 5 0.8785 0.8659 1.455%	C1 6 1.1464 1.1394 0.614%	A0 7 0.8950 0.8943 0.078%	C1 8 1.1501 1.1841 -2.871%	B0 9 0.9725 0.9778 -0.542%

*** Difference = (MCNP - Design) / Design**
RMS Error = 2.53%

Fig. 3 Comparison of RPDs for 1/6 APR1400 Core

Table 3 Comparison of APR1400 MCNP Calculations

	Histories	CPU Time (min.)	Avg. Err.$^{a)}$ (%)	PD $^{b)}$ (MeV/g-n)	Diff. $^{c)}$ (%)
Reference Calculation	19,998,342	2,991	1.27	1.207E-02	–
Advanced Calculation	4,242,252	358	1.30	1.180E-02	-2.2
KCODE Mode	(2,001,396)	(113)	–	–	–
Non-KCODE Mode	(2,240,856)	(245)	(1.30)	(1.180E-02)	(-2.2)

a) Average Pin-wise Statistical Error within of the 28th Fuel Assembly
b) Normalized Power Density of the 28th Fuel Assembly
c) (Advanced Calculation PD – Reference Calculation PD) / Reference Calculation PD

0.985	0.957	1.121	1.192	1.183	1.147	1.171	1.174	1.147	1.192	1.181	1.165	1.176	1.135	0.970	1.061
0.988	0.938	1.150	1.198	1.187	1.198	1.171	1.161	1.180	1.164	1.174	1.190	1.155	1.127	0.959	1.047
-0.34%	2.11%	-2.49%	-0.48%	-0.30%	-4.19%	0.05%	1.12%	-2.73%	2.38%	0.57%	-2.05%	1.80%	0.67%	1.16%	1.35%
0.922	0.101	1.035	1.113	1.115	1.091	1.071	1.049	1.074	1.064	1.099	1.125	1.119	1.038	0.100	0.998
0.885	0.101	1.043	1.117	1.127	1.093	1.081	1.046	1.052	1.060	1.090	1.132	1.114	1.037	0.100	0.985
4.14%	0.02%	-0.79%	-0.28%	-1.07%	-0.22%	-0.96%	0.26%	2.04%	0.46%	0.80%	-0.63%	0.44%	0.09%	0.54%	1.30%
1.046	0.993	1.097	1.062	1.062	1.078	0.974	0.975	0.980	0.989	1.082	1.057	1.069	1.119	1.040	1.202
1.041	0.990	1.095	1.073	1.052	1.091	0.980	0.977	0.981	0.977	1.091	1.059	1.061	1.098	1.038	1.145
0.49%	0.32%	0.21%	-1.03%	0.98%	-1.14%	-0.63%	-0.14%	-0.04%	1.25%	-0.89%	-0.17%	0.72%	1.85%	0.22%	4.99%
1.055	1.077	1.041			0.978	0.102	0.914	0.919	0.100	0.989			1.058	1.108	1.193
1.092	1.048	1.045			0.989	0.101	0.945	0.920	0.100	0.976			1.054	1.109	1.195
-3.38%	2.74%	-0.35%			-1.07%	0.77%	-3.22%	-0.12%	-0.08%	1.31%			0.40%	-0.07%	-0.16%
1.048	1.083	1.033			1.017	1.002	1.001	0.990	0.992	1.025			1.071	1.134	1.224
1.084	1.086	1.040			1.023	1.026	0.972	1.007	0.987	1.006			1.061	1.140	1.187
-3.33%	-0.32%	-0.71%			-0.56%	-2.33%	2.97%	-1.73%	0.58%	1.86%			0.92%	-0.53%	3.15%
1.024	1.039	1.056	0.983	1.015	1.091	1.061	1.084	1.070	1.079	1.097	1.026	0.979	1.084	1.097	1.185
1.081	1.043	1.074	1.004	1.014	1.114	1.057	1.077	1.083	1.050	1.128	1.010	0.995	1.076	1.099	**1.213**
-5.25%	-0.35%	-1.76%	-2.03%	0.16%	-2.09%	0.39%	0.66%	-1.23%	2.72%	-2.74%	1.58%	-1.70%	0.75%	-0.15%	-2.30%
1.027	0.990	0.943	0.100	1.003	1.060	1.129	1.076	1.078	1.132	1.078	1.015	0.098	0.973	1.071	1.165
1.066	1.028	0.967	0.102	0.993	1.071	1.128	1.061	1.089	1.157	1.067	1.010	0.100	0.964	1.077	1.170
-3.62%	-3.71%	-2.44%	-2.35%	1.05%	-0.98%	0.07%	1.47%	-0.97%	-2.10%	1.01%	0.47%	-1.68%	0.94%	-0.48%	-0.42%
1.058	0.985	0.943	0.920	0.995	1.066	1.074			1.081	1.078	0.998	0.914	0.978	1.061	1.185
1.072	1.032	0.961	0.927	0.963	1.067	1.077			1.053	1.065	0.992	0.937	1.005	1.046	1.188
-1.29%	-4.52%	-1.89%	-0.71%	3.34%	-0.09%	-0.29%			2.66%	1.26%	0.65%	-2.46%	-2.60%	1.49%	-0.20%
1.035	1.009	0.959	0.914	0.985	1.072	1.074			1.066	1.063	0.987	0.930	0.983	1.061	1.218
1.064	0.987	0.959	0.909	1.003	1.097	1.076			1.075	1.073	0.983	0.927	0.995	1.059	1.188
-2.69%	2.29%	0.03%	0.56%	-1.80%	-2.26%	-0.22%			-0.83%	-0.98%	0.42%	0.32%	-1.14%	0.23%	2.54%
1.082	1.020	0.960	0.102	1.005	1.054	1.158	1.082	1.067	1.138	1.074	1.005	0.099	0.990	1.082	1.223
1.066	1.006	0.954	0.102	0.995	1.061	1.160	1.091	1.057	1.154	1.082	0.985	0.102	0.993	1.071	1.188
1.50%	1.37%	0.58%	0.14%	0.96%	-0.67%	-0.17%	-0.91%	0.91%	-1.40%	-0.71%	2.04%	-2.93%	-0.37%	1.02%	2.91%
1.077	1.027	1.071	0.989	1.017	1.101	1.061	1.080	1.086	1.066	1.109	1.014	0.990	1.092	1.114	**1.226**
1.053	1.041	1.062	0.984	1.040	1.138	1.054	1.087	1.050	1.071	1.113	1.013	0.984	1.098	1.127	1.201
2.24%	-1.31%	0.83%	0.45%	-2.14%	-3.25%	0.70%	-0.72%	3.45%	-0.46%	-0.28%	0.09%	0.59%	-0.54%	-1.14%	2.13%
1.081	1.078	1.044			1.019	1.003	1.004	0.985	1.000	1.026			1.061	1.150	1.214
1.073	1.091	1.065			1.028	0.995	0.979	0.987	0.988	1.024			1.045	1.122	1.188
0.68%	-1.19%	-2.01%			-0.92%	0.85%	2.49%	-0.30%	1.26%	0.20%			1.50%	2.47%	2.20%
1.087	1.061	1.036			0.974	0.100	0.903	0.908	0.099	0.989			1.080	1.122	1.193
1.050	1.032	1.039			0.974	0.101	0.912	0.903	0.099	0.987			1.061	1.095	1.179
3.51%	2.82%	-0.29%			0.06%	-0.86%	-0.99%	0.54%	-0.47%	0.22%			1.77%	2.47%	1.23%
1.024	0.989	1.062	1.048	1.052	1.058	0.954	0.938	0.960	0.949	1.066	1.043	1.036	1.083	1.021	1.109
1.007	0.963	1.088	1.039	1.051	1.057	0.955	0.957	0.949	0.933	1.079	1.025	1.039	1.104	1.044	1.124
1.68%	2.74%	-2.36%	0.91%	0.13%	0.04%	-0.14%	-1.96%	1.13%	1.71%	-1.17%	1.72%	-0.28%	-1.88%	-2.21%	-1.36%
0.853	0.100	0.968	1.047	1.070	1.040	0.991	0.983	0.993	1.018	1.052	1.093	1.065	0.990	0.100	0.961
0.864	0.101	0.989	1.082	1.097	1.043	1.027	1.004	0.975	0.987	1.049	1.066	1.080	0.967	0.100	0.932
-1.29%	-0.38%	-2.03%	-3.29%	-2.49%	-0.22%	-3.47%	-2.12%	1.84%	3.21%	0.23%	2.53%	-1.32%	2.35%	0.07%	3.07%
0.888	0.814	0.965	1.056	1.061	1.041	1.037	1.072	1.036	1.037	1.082	1.077	1.062	1.030	0.895	0.994
0.899	0.871	1.010	1.069	1.077	1.045	1.046	1.022	1.027	1.031	1.022	1.073	1.057	1.000	0.865	0.969
-1.21%	-6.56%	-4.49%	-1.23%	-1.49%	-0.42%	-0.89%	4.91%	0.85%	0.53%	5.87%	0.38%	0.48%	2.96%	3.46%	2.53%

A	Advanced Calculation
R	Reference Calculation
D	Difference = (A – R) / R

* RMS Error = 1.86 %

Fig. 4 Comparison of Relative Pin Power Distributions in the 28th Fuel Assembly

4. APPLICATION TO HYPER CORE DESIGN

4.1 Core Description

The HYPER based on an accelerator-driven system is a subcritical fast reactor for TRU (transuranics) transmutation. The reactor uses the hard spectrum neutron produced from nuclear spallation reaction between accelerated proton and lead-bismuth target in core center as an external source for transmutation. The general core description is presented in Table 4. The reactor is being designed to produce 1,000MWth power and to maintain a subcriticality of $k_{\mathrm{eff}} = 0.97$. The loading fuel takes the form of TRU-Zr and the coolant is lead-bismuth. The core is divided into two zones by the content of TRU and is composed of 192 assemblies with an active height of 120 cm. Each fuel assembly consists of a hexagonal array of 331 fuel rods and six assemblies include LLFPs. Figure 5 shows 1/6 core model of the HYPER.

Table 4 HYPER Core Description

Total Power	1000 MWth
Core Multiplication Factor	0.97
Active Core Height	120 cm
Effective Core Diameter	380.0 cm
No. of Fuel Assemblies / LLFP Assemblies	186 / 6
Fuel Assembly Pitch	19.96 cm
No. of Fuel Pins per Assembly	331
Fuel Rod Diameter	0.668 cm
Fuel Rod Pitch/Diameter	1.5
Clad Thickness	0.051 cm

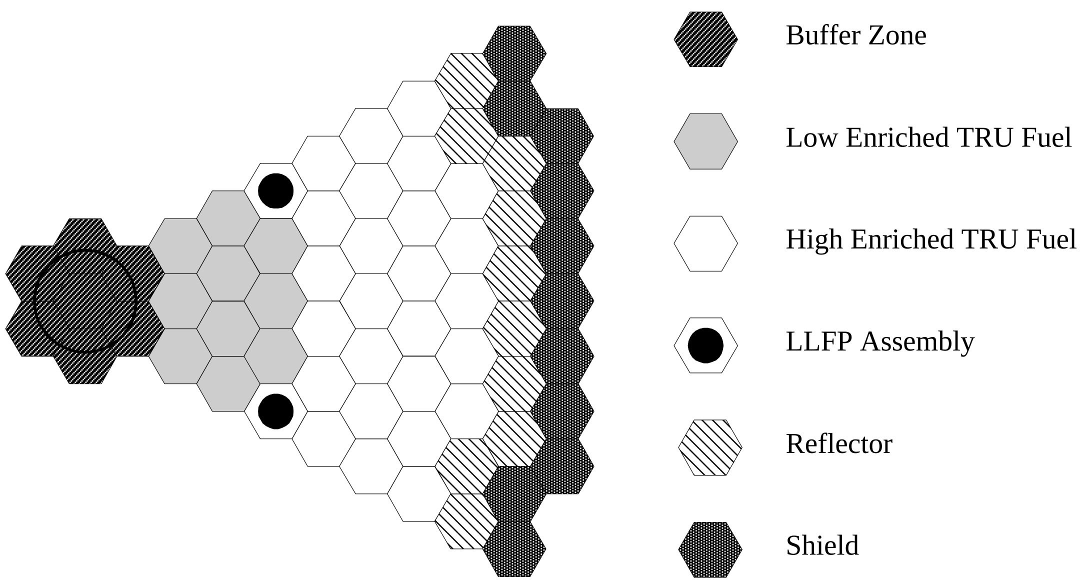

Fig. 5 HYPER 1/6 Core Model

4.2 MCNP Model

An optional MCNP model of HYPER obtained from previous work (Chung, 2000) was used. A design optimization of LLFP assembly to maximize the transmutation of Tc-99 and I-129 in the HYPER system was performed with the MCNP model. It indicated that the localized thermal flux was required to transmute fission products effectively and could lead to increase a local peak power. A moderation method by analyzing the peaking factor around the LLFP assembly as transmutation amount and moderator loading pattern was decided. In this procedure, the MCNP model was focused on only a fuel assembly adjacent to the LLFP assembly. All assemblies were homogenized except a target assembly, which was represented in a pin-by-pin model.

Additional works to the MCNP model were carried out using the F7 tally option for all pins and modifiyng some cell/surface description for the use of the SSW and SSR cards.

4.3 Calculation and Comparison

The pin power distribution within a fuel assembly adjacent to the LLFP assembly was calculated with KCODE runs of 2,000 cycles using the MCNP model for the reference solution. The computer time was 4,124 CPU minutes.

In the advanced calculation, it took about 195 CPU minutes in the KCODE calculation of 100 cycles and 738 CPU minutes in the non-KCODE calculation. The time saving in HYPER calculation is somewhat smaller as compared with APR1400 calculation. It can be explained to result from tally specifications and homogenized assembly model of HYPER. The results are compared with some integral parameters in Table 6 and show trend similar to the previous APR1400 calculation.

Table 6 Comparison of HYPER MCNP Calculations

	Histories	CPU Time (min.)	Avg. Err.[a] (%)	PD [b] (MeV/g-n)	Diff. (%)
Reference Calculation	19,999,810	4,124	1.22	2.577E-03	–
Advanced Calculation	4,190,523	933	1.47	2.447E-03	-4.3
KCODE Mode	(998,833)	(195)	–	–	–
Non-KCODE Mode	(3,191,690)	(738)	(1.47)	(2.447E-03)	(-4.3)

a) Average Pin-wise Statistical Error within the Target Fuel Assembly
b) Normalized Power Density of the Target Fuel Assembly

The relative pin power distribution of the advanced calculation showed RMS error of 2.80% compared with the values calculated in the reference and is not figured here due to a large amount of data. The results gave the peaking factor of 6.00 and 6.20 in the same local position. This advanced technique can be applied effectively in the purpose of producing the parameters only a local region not in the whole region.

5. CONCLUSIONS

This paper introduced an advanced MCNP technique to reduce computing time for only a local region calculation in reactor core analyses and the technique was applied to APR1400 and HYPER core analyses. The results showed that the advanced technique suggested here could construct an equivalent reactor physics model to obtain almost equivalent results with the remarkable reduction of computing time in the calculation of nuclear characteristics within a specific local region. It is expected that this technique can be widely applied to the problems for obtaining the parameters only in a local region not in the whole region.

ACKNOWLEDGMENTS

The authors wish to acknowledge the financial support from the Innovative Technology Center for Radiation Safety at Hanyang University, Korea.

REFERENCES

Chung, J.S., Kim, C.H., Kim, Y.H., Park, W.S., 2000. Transmutation of Tc-99 and I-129 in HYPER system. Proc. of the Korean Nuclear Society Autumn Meeting, Taejon, Korea, to appear.

Joneja, O.P., Plaschy, M., Jatuff, F., Lüthi, A., Murphy, M., Seiler, R. Chawla, R., 2001. Validation of an MCNP4B whole-reactor model for LWR-PROTEUS using ENDF/B-V, ENDF/B-VI and JEF-2.2 cross-section libraries. *Annals of Nuclear Energy* **28**, 701-713.

Kim, J.O, Kim, J.K., 1998. A new uncertainty arising in reactor pressure vessel fluence calculation. *Annals of Nuclear Energy* **25**, 963-982.

Laky, P.G., Tsoulfanidis, N., 1995. Neutron fluence at the pressure vessel of a pressurized water reactor determined by the MCNP code. *Nucl. Sci. Eng.* **121**, 433-447.

Pierre, M.J.R., Bonin, H.W., 1999. Monte Carlo simulation of the LEU-fueled SLOWPOKE-2 nuclear reactor using MCNP 4A. *Nuclear Technology* **125**, 1-11.

Redmond II, E.L., Ryskamp, J.M., 1991. Monte Carlo methods, models, and applications to the advanced neutron source. *Nuclear Technology* **95**, 272-286.

Redmond II, E.L., Yanch, J.C., Harling, O.K., 1994. Monte Carlo simulation of the Massachusetts Institute of Technology Research Reactor. *Nuclear Technology* **106**, 1-14.

Seo, B.K., Kim, J.K., Shin, C.H., Kwon, T.J., 2001. A modification of the inner and outer core for reactor pressure vessel lifetime extension. *Nuclear Technology* **133**, 325-337.

Wagner, J.C., Haghighat, A., Petrovic, B.G., 1996. Monte Carlo transport calculations and analysis for reactor pressure vessel neutron fluence. *Nuclear Technology* **114**, 373-398.