Feasibility Study of Core Design with a Monte Carlo Code for APR1400 Initial core

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1. Introduction

Monte Carlo transport codes are used for calculating the effective multiplication factor of the core in reactor physics area, dose rates in shielding problem for developing, and the design of a detector and also for verifying the results of deterministic transport codes. The main advantage of Monte Carlo transport codes is the capability of modeling the complete core geometry and treating neutron interaction physics with essentially no approximations. However, Monte Carlo code imposes high computational burden to model the sophisticated system, thus, puts a barrier to the various applications in reactor engineering such as PWR full-core Monte Carlo calculation. The Monte Carlo calculation becomes more popular and useful nowadays due to the rapid progress in computing power and parallel calculation techniques [1]. There have been many attempts to analyze a commercial core by Monte Carlo transport code using the enhanced computer capability, recently. In this paper, Monte Carlo calculation of APR1400[7] initial core has been performed and the results are compared with the calculation results of conventional deterministic code to find out the feasibility of core design using Monte Carlo code. SERPENT[1], a 3D continuous-energy Monte Carlo reactor physics burnup calculation code is used for this purpose and the KARMA-ASTRA code system [8],[5] is used for a deterministic code of comparison.

2. Methods and Results

2.1 Monte Carlo code SERPENT

SERPENT (version 1.1.19) is used as a Monte Carlo code for this study and it was first developed at VTT Technical Research Centre of Finland in 2004 and has been evolved continuously. SERPENT is widely applicable to the calculation of homogenized multi-group constants for a deterministic reactor simulator, fuel cycle studies involving a burnup calculation, validation of deterministic lattice transport codes, full-core modeling of research reactors, small modular reactors involving burnup calculations. SERPENT also can be used for the educational understanding of reactor physics phenomena. SERPENT reads fission yield data from ACE format cross section libraries. This code has been distributed by the OECD/NEA Data Bank and RSICC since 2009, and has wide spectrum of users including 112 universities and research organization over 30 countries around the world.[2]

2.2 Core configuration

The geometry and material composition of the reactor core for this study utilize the information of NEA Benchmark specifications for PWR core [3] except for the fuel assembly type and loading pattern. The core design is based on the APR1400 initial core reference design. The reactor core consists of 241 PLUS7^{TM} fuel assemblies arranged as shown in Fig. 1 and the geometrical information is also indicated in the figure.

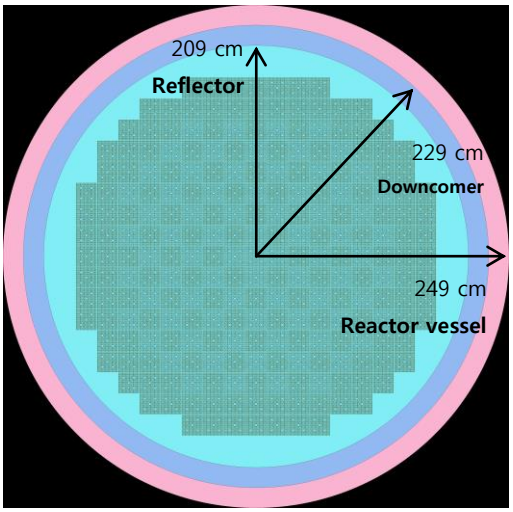

Fig. 1 APR1400 reference core configuration

Although the core is typically bounded by baffle plates and a core barrel, these regions have been homogenized into water reflector to simplify the modeling. The downcomer filled cold water is modelled by cylindrical surface with inner radius of 209cm, outer radius of 229cm. The reactor vessel has 20cm thickness. The core design parameters for this research are given in Table 1.

Table 1. Parameters of the APR1400 Cycle 1

Parameter	Value
Reactor thermal power	3983 MW(thermal)
Operating pressure	15.513 Mpa(155.13bar)
Core inlet/outlet temperature	290.56/325℃
No. fuel assembly	241
Active length	381cm
Fuel assembly geometry	PLUS7^{TM} 16 x 16
Fuel enrichment	3.64/3.14/2.64/1.72 ^{235}U w/o

The vertical cross section of the reactor core is shown in Fig. 2. For the simplicity, the moderator is modeled

as two regions of a low temperature region below the mid plane and a high temperature region above the mid plane. The space above the active fuel was homogenized into three regions as 20cm of top fuel assembly (FA) region, 8cm of top nozzle region, and 12cm of upper core plate region. Similar homogenization was made for the region below the active fuel into regions of 10cm of bottom FA region, 6cm of bottom nozzle region, and 30cm of lower core plate region.

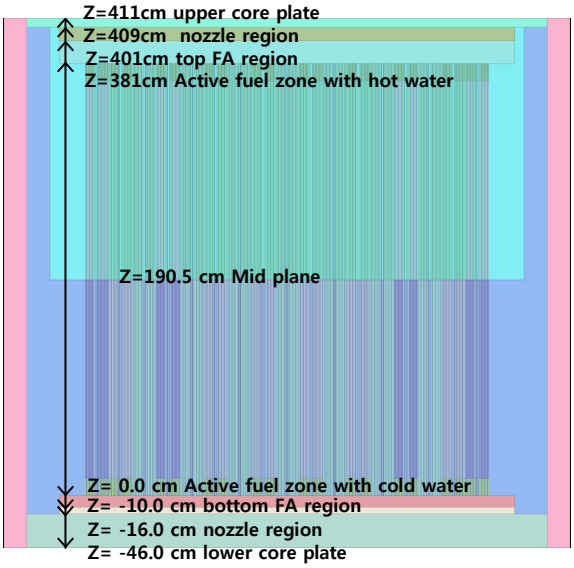

Fig. 2 Vertical cross section of the core

2.2 Lattice configuration

PLUS7^{TM} fuel assembly consists of typically 16 x 16 array of 236 rods and 4 large size guide tubes and 1 large size instrumentation as shown in Fig. 3. There are lower enriched fuel rods around water holes where the thermal neutron peaks are encountered and burnable absorber rods are placed symmetrically in the assembly as shown in the Fig. 3.

The structure materials and space grid are not modeled in this study. The basic geometry and properties of the PLUS7^{TM} fuel are listed in Table 2.

Burnable absorber rods are composed of 8 w/o gadolinia (Gd_2O_3) admixed with UO_2 of 2.0 w/o U-235 enrichment. Because of the rim-effects caused by self-shielding, burnable absorber rods are divided into 10 rings for depletion calculation in this study.[4]

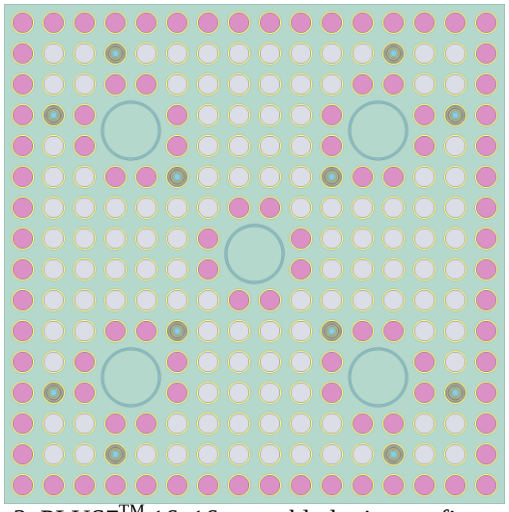

Fig. 3. PLUS7^{TM} 16x16 assembly lattice configuration

Table 2 PLUS7^{TM} basic geometry and properties

Fuel rod	
Fuel rod pitch	1.28774 cm
Number of fuel rods	236
Clad material	Zircaloy-2
Cladding, inner diameter	0.83746 cm
Cladding, outer diameter	0.95200 cm
Cladding, thickness	0.05727 cm
Fuel material	UO_2
Pellet, diameter	0.81916 cm
Fuel pellet density	10.313 g/cm3
Active length	381 cm

The axial assembly span of 381cm is divided into 26 axial nodes of non-uniform mesh in this calculation. Burnable absorber fuel rods have axial cutbacks of 15.24cm both at the top and bottom of active region and normal fuel rods also have axial blankets of 15.24cm with 2.0 w/o U-235 enrichment both at the top and bottom of active region..

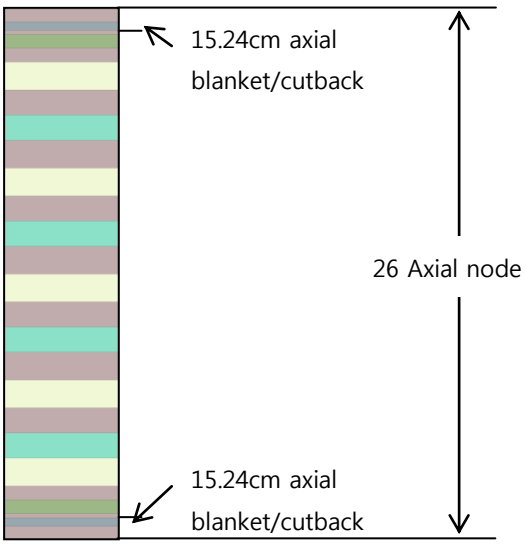

Fig. 4 Axial modeling of PLUS7^{TM} assembly

The lower enriched(zoned) fuel pins are utilized to decrease the power peaking within an assembly and to have more even power distribution. Due to the usage of various fuel enrichment patterns and burnable absorber rods, the fuel assemblies for initial core comprise 9 assembly types as listed in Table 3.

Table 3. Assembly types for APR1400 initial core

Assembly type	No. of Fuel Assemblies	Fuel enrichment (w/o U-235)	No. of Gd rods	Gd enrichment (w/o Gd2O3)
A0	77	1.71	-	-
B0	12	3.14	-	-
B1	28	3.14/2.64	12	8
B2	8	3.14/2.64	12	8
B3	40	3.14/2.64	16	8
C0	36	3.64/3.14	-	-
C1	8	3.64/3.14	12	8
C2	12	3.64/3.14	16	8
C3	20	3.64/3.14	16	8

2.3 Results(SERPENT vs KARMA/ASTRA)

The SERPENT burnup calculations were performed using the Message Passing Interface (MPI)[4] on a 3.0 GHz Intel Xeon pc linux platform with 64GB. The ENDF/B-VII library was used to generate cross section in this study. The depletion calculation is performed with CRAM method which is based on the advanced matrix exponential solution [6]. Equilibrium xenon condition and boron concentration of boron letdown results by ASTRA were assumed for the full core depletion calculation. The default calculation mode in SERPENT is the k-eigenvalue criticality source method, in which the simulation is run per cycles and the source distribution of each cycle is formed from the fission reaction distribution of the previous cycle. For the reliable statistical accuracy of SERPENT run, 500 active cycles with 5,000 neutrons per cycle, 20 inactive cycles run and 8 MPI for lattice calculation, whereas 1,000 active cycles with 100000 neutrons per cycle, 200 inactive cycles run and 10 MPI for full core depletion calculation. Inactive cycles are required in order to allow the initial fission source distribution to converge before starting to collect a result. The SERPENT calculation results were compared with the results of KARMA-ASTRA deterministic code system.

KARMA(Kernel Analyzer by Ray-tracing Method for fuel Assembly) which is a two-dimensional multi-group transport theory code for burnup calculation was used as a tool of cross section generation for ASTRA, a multi-group 3-D nodal code for the nuclear design of commercial reactor core [5]. The lattice calculation of KARMA and depletion calculation of ASTRA were performed with conditions to be consistent with SERPENT.

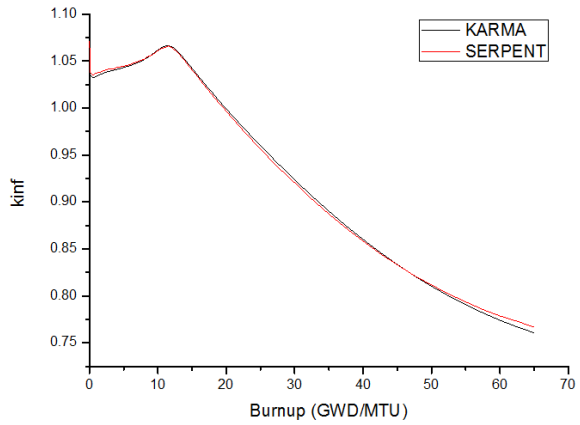

Fig. 5 Result of reactivity comparison (KARMA vs SERPENT)

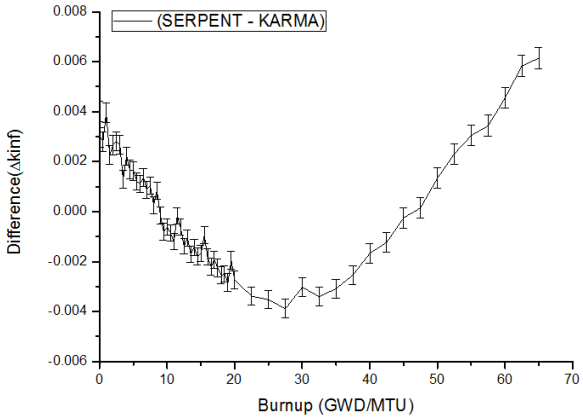

Fig. 6 relative difference kinf (KARMA vs SERPENT)

Fig. 5 shows the result of reactivity comparison for the lattice calculation of PLUS7^{TM} fuel assembly based ENDF/B-VII cross section library using KARMA and SERPENT. The reactivity behavior of each code is quite similar during depletion. Fig. 6 shows the result of difference in kinf comparison. The differences of multiplication factor (kinf) between KARMA and SERPENT are 0.0035 at 0 GWD/MTU and about -0.002 at 40 GWD/MTU. The difference slightly decreases after 0 GWD/MTU and increases toward the last step of depletion from about 25 GWD/MTU. This result of difference is slightly larger compared to that of CASMO-4E which shows about 0.0015 at 0 GWD/MTU and about 0.0031 at 40 GWD/MTU [9].

The result of reactivity comparison for depletion results of ASTRA and SERPENT based on the critical boron of ASTRA. While the thermal hydraulic feedback was not modeled in this simulation (SERPENT Version 1.1.19) and the simulation result included inherent modeling errors, the reactivity behavior of each code is similar except the first two step of depletion reflecting the difference in xenon buildup process of each code. However, the result showed a considerable difference in absolute reactivity and the main reason of reactivity difference is the thermal-hydraulics feedback effects in SERPENT calculation which include changes in the dimensions of the geometry, material density and

temperature effects on cross section data. In order to confirm this assertion, the SERPENT simulation was performed applying axial temperature profile obtained from ASTRA output at BOC 100% power, ARO, equilibrium xenon. With soluble boron concentration of 794.9 ppm to be consistent with ASTRA, 500,000 neutrons per cycle runs with 3 MPI were performed for this calculation. The reactivity difference between ASTRA and SERPENT is 180 ± 9 pcm, which is a reasonable difference.

Also, the reactivity is compared at BOC HZP, ARO no xenon condition which shows uniform axial temperature profile and it is the reference condition at which measurement is made at the start of every reactor cycle. Boron is set to zero and 500,000 neutrons per cycle runs with 10 MPI were performed. The reactivity calculated by ASTRA at this condition is 13402 pcm and that of SERPENT is 13670 ±8 pcm to show the difference of 268 ±8 pcm.

Although the radial coolant temperature profile is not considered, the Monte Carlo simulation shows reasonably consistent results with those from the deterministic code. As well known, it is clear from this simulation that the thermal hydraulic model linkage is essential for the successful application of Monte Carlo code to a commercial core design. SERPENT 2 code is now available and under development to have this functionality of sophisticated moderator temperature modeling. The next step of study would be SERPENT 2 modeling of this calculation. Also, The comparison of peaking factor, power distribution, reactivity coefficients, and the shutdown margin will be made in the further study.

Computer running time for the full core depletion calculation is 22.7hr which is performed by 100,000 neutrons per cycle runs for SERPENT and 0.083hr for ASTRA, respectively. Also, for the axial temperature modeling it took 2.19hr which is performed by 500,000 neutrons per cycle runs to simulate just 1 step of calculation. Parallel computing is used for SERPENT only. According to this result, the computer running time would still be one of the barriers for the core design based on Monte Carlo code.

3. Conclusions

The preliminary investigation for the feasibility of commercial core design with Monte Carlo code was performed in this study.

Simplified core geometry modeling was performed for the reactor core surroundings and reactor coolant model is based on two region model. The assembly geometry was modeled completely with axial nodes same as those of a deterministic code except for the spacer grids. The reactivity behavior according to burnup depletion condition indicates that the current method of modeling in Monte Carlo code should be improved for the depletion methods and moderator temperature modeling. The reactivity difference at HZP ARO condition between Monte Carlo code and the deterministic code is consistent with each other and the reactivity difference during the depletion could be reduced by adopting the realistic moderator temperature. The reactivity difference calculated at HFP, BOC, ARO equilibrium condition was 180 ±9 pcm, with axial moderator temperature of a deterministic code.

The computing time will be a significant burden at this time for the application of Monte Carlo code to the commercial core design even with the application of parallel computing because numerous core simulations are required for actual loading pattern search. One of the remedy will be a combination of Monte Carlo code and the deterministic code to generate the physics data.

The comparison of physics parameters with sophisticated moderator temperature modeling and depletion will be performed for a further study.

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