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CORE BURN-UP ANALYSIS OF THE RSG-GAS RESEARCH REACTOR USING DETERMINISTIC AND STOCHASTIC METHODS

Tukiran Surbakti[a]*, Surian Pinem[a], Wahid Luthfi[a], Donny Hartanto[b,c]

[a]Research Center for Nuclear Reactor Technology, Research Organization for Nuclear Energy, National Research and Innovation Agency (BRIN), Kawasan PUSPIPTEK Serpong Gd.80, Tangerang Selatan, Banten 15314, Indonesia
[b] Department of Mechanical and Nuclear Engineering, University of Sharjah, P.O. BOX 27272, Sharjah, United Arab Emirates
[c]Nuclear Energy System Simulation and Safety Research Group, Research Institute of Sciences and Engineering, University of Sharjah, P.O. BOX 27272, Sharjah, United Arab Emirates

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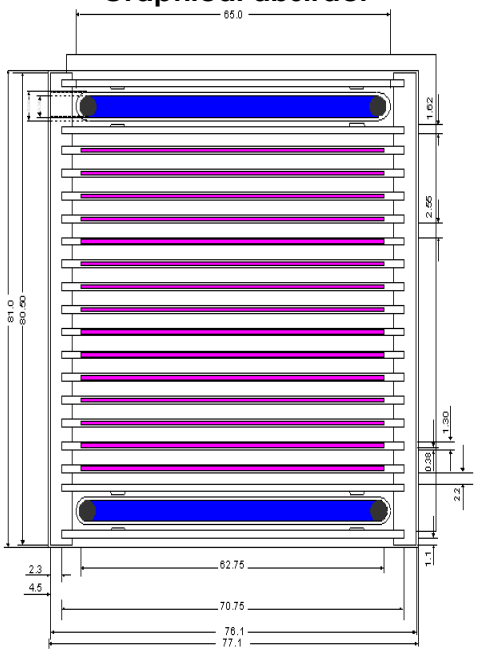
*Corresponding author
tukiran@brin.go.id

Graphical abstract



Abstract

Due to several characteristics, such as geometry, compact core, high coolant flow, and high neutron flux, the burn-up study of the RSG-GAS multi-purpose reactor provides challenges when employing a neutronic calculation. For the burn-up analysis, two calculating methodologies are used in the RSG-GAS: deterministic and probabilistic methods. The deterministic codes such as WIMSD-5B and Batan-FUEL are utilized, whereas the continuous-energy Monte Carlo code Serpent 2 is used for the stochastic method. WIMSD-5B is being used to produce a four-group cross-section that is needed by Batan-FUEL to do full core diffusion calculations. Burn-up calculations were performed at the whole fuel assemblies inside the core to see if the deterministic code, WIMSD-5B/Batan-FUEL, could accurately replicate the burn-up behavior of the RSG-GAS research reactor. The Serpent 2 calculation was also done with the exact models to provide a comparison. The results show that both Serpent 2 and WIMSD-5B/Batan-FUEL can perform the RSG-GAS burn-up analysis if appropriate treatments are made to the deterministic codes at both the assembly and core levels. There is a 5% difference in calculated fuel burn-up between deterministic and stochastic approaches.

Keywords: RSG-GAS, WIMSD-5B, Batan-FUEL, Serpent 2, burn-up, core analysis

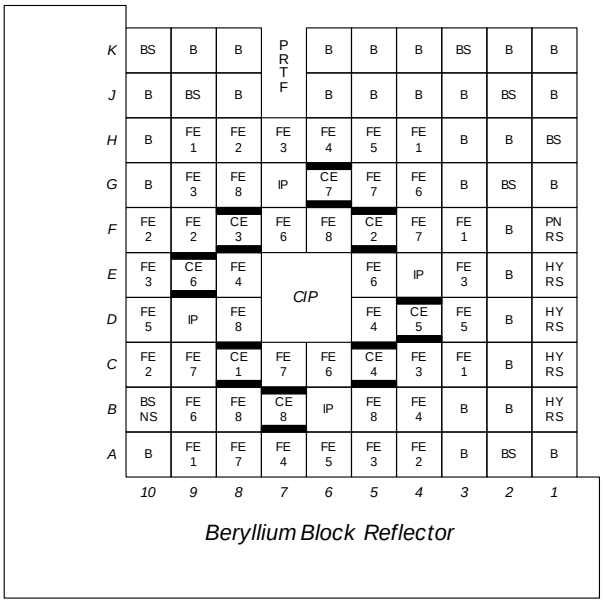
1.0 INTRODUCTION

Research reactors play a significant role in the advancement of nuclear technology, neutron application, and radioisotopes production. According to the IAEA's Research Reactor Database (RRDB), research reactor technology is still evolving to increase its utilization for humanity. This database is also containing technical information concerning research reactors in 67 nations [1]. The research reactors are designed to create neutrons for a variety of applications, including neutron scattering, neutron activation analysis, neutron radiography, post-irradiation material testing, and isotope production. Because of the presence of strong neutron absorbers and the compact reactor design, these research reactors have significant heterogeneity and considerable leakage. It is crucial to estimate the burn-up behavior of the RSG-GAS research reactor to conduct safety analysis, and to improve its efficiency of reactor operation and strategic fuel management planning.

However, to accurately predict the neutronic behavior of research reactors such as RSG-GAS [2] using a full core diffusion calculation, several modifications must be made to the traditional three-level calculation (fuel cell and fuel assembly-level transport calculation and full core diffusion calculation). Additionally, the Monte Carlo approach previously shows that it could do extremely accurate burn-up calculations, with its comprehensive core model. Serpent 2 is one of the most extensively used Monte Carlo programs which use ACE format continuous-energy cross-sections data library. For large-scale core, the Serpent 2 has great burn-up capability with its algorithms and excellent parallel efficiency. WIMSD-5B was used to homogenize the core material in the first step into 4-group cross-sections. Verification and validation of WIMS-D5 [3], as well as Batan-FUEL [4], have been done by comparing them to the experimental data and also Monte Carlo code (MCNP) [5]. WIMSD-5B/Batan-FUEL has been utilized to simulate and solve practical challenges in research reactors, including RSG-GAS reactor core [6], RSG-GAS fuel loading pattern optimization [7], and RSG-GAS neutronic and thermal-hydraulic safety analyses [8]. With relevant modeling characteristics employed, deterministic programs have the advantages of fast execution time with adequate accuracy. In this study, the deterministic code WIMSD-5B/Batan-FUEL is compared to the Serpent 2 stochastic code in simulating the burn-up behavior of RSG-GAS. The ENDF/B-VII.1 library is used to ensure that nuclear data is consistent between each code Serpent uses a continuous-energy data library, and Batan-FUEL uses a multi-group data library.

2.0 METHODOLOGY

The RSG-GAS is a pool-type research reactor with a maximum thermal output of 30 MW. It uses low enriched uranium (LEU) plate-type fuels, beryllium as a reflector, and light water as moderators and coolants [9]. The RSG-GAS equilibrium core is depicted in Figure 1, with 40 standard fuel elements, 8 control fuel elements and neutron absorbers, 4 irradiation positions (IP), 1 center of irradiation position (CIP), and 37 beryllium reflectors. The core is surrounded by light water with its standard fuel element and control element are shown in Figures 2 and 3. RSG-GAS core has a beryllium reflector, L-shape, and a light water reflector. Due to its complexity, several modifications have been done to the geometry modeling to ensure the consistency of geometry in both Batan-FUEL and Serpent 2. The RSG-GAS research reactors modeling scheme has been described in the previous study [10].



Note : FE = Fuel Element, CE = Control Element, BE = Be Reflector Element, BS = Be Reflector Element with plug, IP = Irradiation Position, CIP = Central Irradiation Position, PNRS = Pneumatic Rabbit System, HYRS = Hydraulic Rabbit System

Figure 1 Core configuration of RSG-GAS research reactor [11]

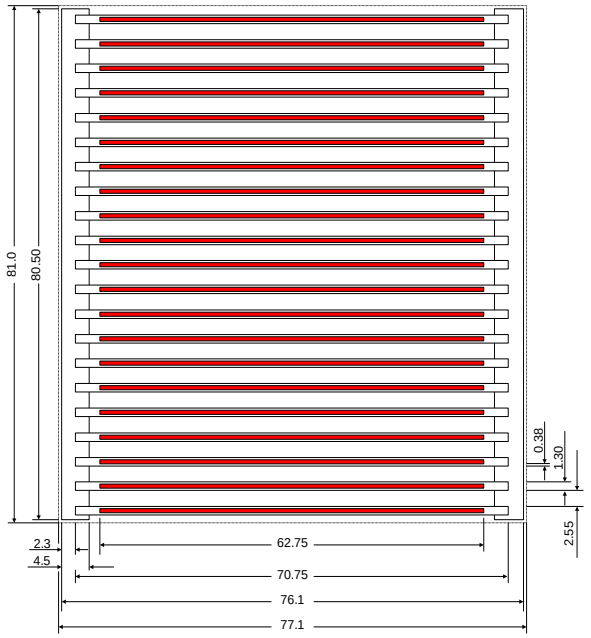


Figure 2 RSG-GAS Standard fuel element [12]

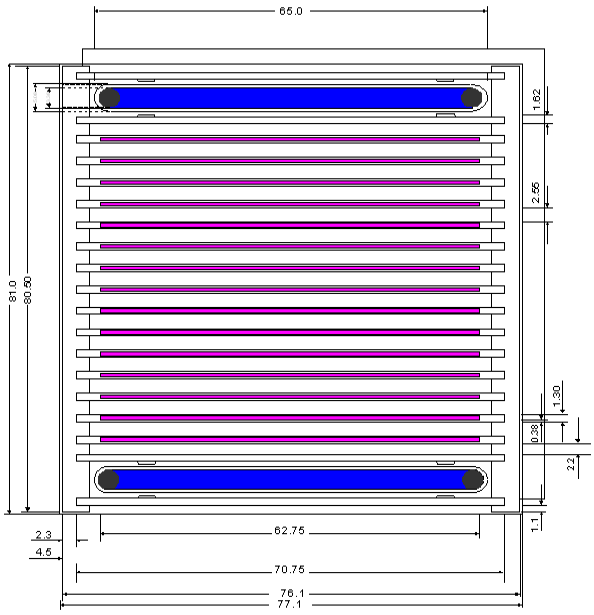


Figure 3 RSG-GAS standard control element [13]

The Winfrith Improved Multigroup Scheme Version D-5B (WIMSD-5B) calculation system is a general deterministic code for solving cell and lattice transport calculations that are commonly used in neutronic simulations of thermal reactors [14]. The WIMSD code determines the flux as a function of space and energy in the cell by numerically solving the Boltzmann neutron transport equation for a range of geometries, including homogeneous, slab, rod clusters, and finite cylinders [15]. There are numbers of algorithms to solve the neutron transport equation, and which discrete ordinate SN-method is being used for all WIMSD calculations in this study. WIMSD-5B is an improved version of the code that was first released by Winfrith in 1998 and distributed by the OECD/NEA Data Bank [16].

The burn-up-dependent few group cross-sections were constructed to do the full core burn-up calculations using the Batan-FUEL diffusion code. The macroscopic cross-sections for a comprehensive core diffusion calculation of Batan-FUEL are generated using WIMSD-5B. The WIMSD-5B is also used to calculate depletion over an assembly at various burn-up stages. The geometric modeling of each component of the RSG-GAS reactor was presented in the previous section.

The whole core neutron flux and multiplication factors are then solved using Batan-FUEL. The macroscopic cross-section from the WIMSD-5B code may directly access the group constants provided by Prenox1. In the radial plane, 31 meshes are utilized, each with the size of a fuel element. The active core is surrounded by light water as a coolant and reflector. 75 meshes with 5 cm per mesh are employed in the axial direction. The void boundary conditions are used in axial and radial directions.

The standard fuel element and control element few group cross-sections are constructed using one-dimensional single lattice models in detailed geometry with the reflecting boundary condition. The standard fuel element has 21 fuel plates, while the control element has 15 fuel plates. The RSG fuel has the same geometric parameters as many other MTR types as described before.

Except for the fuel element, a two-dimensional whole core model of the RSG-GAS research reactor is utilized to build a few group cross-sections of the control rod element, beryllium element, beryllium reflector, light water reflector, and so on. Because all components such as fuel meat, cladding, and coolant are blended proportionally to volume, the fuel element is classified as a single material.

Our earlier study [17] described the comprehensive geometry models of the standard fuel element, follower element, and whole core model. The temperature of the fuel, coolant, and reflector is set to 293.6 K in Batan-FUEL and Serpent 2 calculations to ensure the consistency of the temperature while using the ENDFB-VII.1 continuous-energy nuclear data utilized by Serpent 2. In this study, the thermal-hydraulic feedback was not taken into account.

The transport calculations were carried out in WIMSD-5B with a 69-group evaluated nuclear data, ENDFB-VII.1. The 69 groups (fast, resonant, and thermal) are then condensed into four groups of macroscopic cross-sections which resemble the fast neutron energy group ranging from 10 MeV to 0.821 MeV (1-5), the first epi-thermal group's energy ranges from 0.821 MeV to 5.530 eV (5-15), the second epi-thermal group's energy ranges from 5.530 MeV to 0.625 (15-45), and the thermal group's energy ranges from 0.625 eV to 0.00001 eV (45-69) [18]. The B1 leakage model employs the buckling search model but there is also an option to account for the influence

of heterogeneous buckling. The alternative is proving to be more practical.

In Serpent code, three-dimensional models are employed for core calculations. Figure 4 depicts the radial cross-section of Serpent's core model, whereas Figure 5 depicts its axial cut-away of the active core. Each fuel element is sized 7.71 cm x 8.10 cm in the radial direction with the active core made up of 8-8 array meshes. The width of the beryllium water reflector and the light water reflector is equivalent to 8 and 4 fuel element pitches, respectively. The overall height of the reactor is 135 cm, with a 75 cm fuel element (60 cm is for active fuel height) and a 30 cm light water reflector in both the upper and lower halves. As said before, the void boundary conditions are used as axial and radial outer boundaries.

The transport calculations were carried out in Serpent 2 with the ENDFB-VII.1 continuous-energy library, for its consistency with WIMSD-5B's 69-group library. Serpent 2's burn-up calculation was carried out with its built-in calculation procedures, with no external solvers. The number of depletion zones is unrestricted, yet when the quantity of burnable elements is considerable, memory utilization may limit the optimization. Fission and activation products, as well as actinide nuclides, are automatically picked for the calculation, and burnable materials can be subdivided into depletion zones. The whole burnup or irradiation calculation is measured based on time or burn-up units.

The normalization of reaction rates to total power, specific power density, flux, fission, or source rate can be altered by dividing the irradiation cycle into multiple depletion intervals. By separating the calculation into numerous segments, the restart option allows you to do fuel shuffling or make any changes to the input. For basic geometries, such as 2D fuel pin lattices, the volumes, and masses required for normalization are determined automatically.

The data for radioactive decay and fission yield is taken from standard ENDF format data libraries. Under meta-stable conditions, the decay library data comprises 4000 nuclides, all of which may be calculated. In general, the total number of nuclides created by fission, transmutation, and decay is fewer than 1500. Burn-up calculations use nuclide concentration data from fission and non-fission reactions in the reactor core, with specific cross-sections typically ranging from 200 to 300 nuclide. For all significant actinides (31 nuclides in the ENDF/B-VII.1 data), the results of fission reactions are highly dependent on the neutron energy available in the library; by default, the library provides the option to read energy-dependent data from the ENDF format file.

The Serpent's program reads energy-dependent nuclides and employs a pre-set ratio for particularly significant nuclides like Am-241 and Pm-147. The average volume flux for a reaction rate is computed using the flux spectrum gathered on a particular energy grid or by collecting the neutron from the fission reaction after the calculation is done. The computation will be sped up by a factor of 3-4 if the collapsed neutron energy group spectrum is used, and inaccuracies in the calculation results can be overlooked due to the high energy resolution of the flux spectrum. Other coupled Monte Carlo burn-up computation algorithms [19] have adopted a similar methodology.

The Bateman fuel depletion equation can be solved in two ways using the serpent program. The TTA (Transmutation Trajectory Analysis) approach [20] that used in the linearization of the nuclear fuel depletion reaction and relies on analytical solutions. Second, an exponential matrix analytic solution was constructed specifically for the Serpent program [21] using the CRAM approach (Chebyshev's Rational Approach Method). The study related to fuel depletion estimations is obtained using this method [22, 23]. Both the traditional Euler method and predictor-corrector used in the burn-up calculation, and Serpent 2 have a number of approaches and solutions for burn-up computations [24]. Implicit techniques can increase the accuracy and consistency of 3D burn-up estimations [25]. During a neutron transport simulation, the poisons Xe-135 and Sm-149 can be estimated separately from other nuclides and adjusted for their equilibrium concentrations which made the iteration can be done in neutron transport mode without burn-up calculations. The Chebyshev Rational Approximation Method (CRAM) approach was used in this study with linear extrapolation for both the predictor and corrector [26].

3.0 RESULT AND DISCUSSIONS

Each typical fuel element has 21 fuel plates with a fuel meat thickness of 0.038 cm. The Al uses as a matrix combined with U3Si2 in the fuel meat with a uranium density of 2.96 g/cm^3. Table 1 lists the atomic densities of each component of the standard fuel element. It should be noted that the fresh fuel contains no xenon. Validations on plate-type fuel burnup calculations were done prior to the homogenization of cross-sections and depletion calculation using WIMSD-5B. The k-inf value is greatly dependent on the fuel burn-up, as seen in Table 2. With higher fuel burn-up, then smaller the k-inf value, which is caused by depletion of fuel. on 75 percent fuel burn-up, the k-inf value still exceeds 1.0 while on 80 percent, the k-inf value has dropped below 1.0, as shown in Table 2. The k-inf has a typical steep gradient at first, but later the evolution becomes smoother. Before the burn-up approaches 50%, the relative difference of each k-inf on the burnup step is less than 0.03. The difference grows after 50 percent burnup and reaches 0.03 at 150,000 MWd/tU.

Table 1 Atomic densities of the RSG-GAS fuel element

Component	Nuclide	The density of atom (Atom/b*cm)
Meat of fuel	^{235}U	1.50025×10^{-3}
	^{238}U	6.01895×10^{-3}
	^{29}Si	5.01267×10^{-3}
	^{27}Al	4.30311×10^{-2}
Cladding	^{24}Mg	1.36127×10^{-3}
	^{29}Si	1.72395×10^{-4}
	^{63}Cu	1.26989×10^{-5}
	^{55}Mn	8.81320×10^{-5}
	^{56}Fe	1.15597×10^{-4}
	^{52}Cr	9.31187×10^{-5}
	^{48}Ti	3.37079×10^{-5}
	^{27}Al	5.76030×10^{-2}
Coolant	^{1}H	6.65620×10^{-2}
	^{16}O	3.32810×10^{-2}
Extra region	^{24}Mg	5.60411×10^{-4}
	^{29}Si	3.34402×10^{-4}
	^{63}Cu	9.36022×10^{-5}
	^{55}Mn	1.21957×10^{-4}
	^{56}Fe	9.56977×10^{-5}
	^{52}Cr	4.12492×10^{-5}
	^{48}Ti	2.35808×10^{-5}
	^{27}Al	4.01204×10^{-2}
	^{16}O	2.02507×10^{-2}
	^{1}H	1.01253×10^{-2}

Table 2 k-inf of WIMSD-5B for fuel assembly calculations

Steps	Atomic density (Atom/b-cm x 10^{-3})	Burn-up (%)	k-inf
1	1.5003	0.0	1.60279
2	1.4988	0.1	1.59797
3	1.4913	0.6	1.53878
4	1.4240	5.0	1.50614
5	1.3345	11.0	1.48238
6	1.2447	17.0	1.45748
7	1.1543	23.0	1.43035
8	1.0648	29.0	1.40099
9	0.9752	35.0	1.36864
10	0.8848	41.0	1.33259
11	0.7944	47.0	1.29242
12	0.7037	53.0	1.24715
13	0.5983	60.0	1.18624
14	0.4797	68.0	1.10352
15	0.3753	75.0	1.01249
16	0.2702	82.0	0.89488
17	0.0149	90.0	0.70449

The 4 groups macroscopic cross-section obtained from the WIMSD-5B program was then used to calculate the RSG-GAS reactor fuel burn-up using the Batan-FUEL program. The burn-up calculation results can be seen in Figure 5 which consist of 3 core conditions, namely BOEC (Beginning of Equilibrium Cycle) at 0 MWd, MOEC (Middle of the Equilibrium Cycle) at 315 MWd, and, EOEC (End of Equilibrium Cycle) at 630 MWd, all under 15 MWth operating power. The results of this calculation can be grouped into 8 classes of average burn-up in BOC conditions, namely fresh fuel element with 0% burnup fraction, and fuel element with burnup fraction close to 7%, 14%, 21%, 28%, 35%, 42%, and 49%. In the MOEC condition, there were also 8 averages burn-up classes, namely 3.5%, 10.5%, 17.5%, 24.5%, 31.5%, 38.5%, 45.5%, and 52.5%. In the condition of the EOC also 8 classes of burn up averaged 7%, 14%, 21%, 28%, 35%, 42%, 49%, and 56%. There are five standard fuel elements and one control fuel element in each class. Each standard fuel element and control fuel element is designed to achieve maximum burn-up through a reshuffling scheme, for example, in the 8th class at the end of the cycle, the highest control element burn-up is in the B-7 position achieved 57.53% burnup. It is still lower than the operational limit (59.56%).

For Serpent 2 simulations, 50,000 neutron histories per cycle were used with a total cycle of 300, the first 100 as inactive cycles. Each fuel assembly is divided into 5 axial regions that are treated as different material compositions, with its composition taken from 2D calculations on fuel assembly. Figure 4 shows that the core k-eff value decreases with the passage of time from the Serpent 2 calculation. The calculation results show that the core k-eff value is close to unity (1.0) after it operates for 43 days at 15 MW power. These results are consistent with the results of the RSG-GAS working core which is always operated for about 43 days at 15 MW. After that, a new core configuration is formed by removing 5 standard fuel elements and 1 control fuel element at the 8th burn-up class, reshuffling the remaining fuel element, and insertion of 5 new standard fuels and a control fuel element.

Figure 6 shows the calculated fuel burn-up of RSG-GAS core with Serpent 2 code at BOEC, MOEC, and EOEC conditions. The calculation results show that there is a consistency between Serpent 2 and Batan-FUEL, with its 8 burnup classes as shown in Figure 4. When comparing the calculation results for each fuel element, there is an average difference of ± 5%. At the MOC, the highest error between WIMS-5B/Batan-Fuel and Serpent 2 is at location D-5 which provides an error of 10.95%, followed by positions-9, H-9, and F-10 with a difference of 7.43%, 6.11%, and 5.66%, respectively. However, in another position, there is a relative error of less than 6%. It is noticed that this higher error is for the fuel elements near the left side reflector blocks. Meanwhile, at the EOEC, the highest difference is located at B-9 with about 8.19% difference. In addition, it is shown that at position B-7, the highest burn-up at EOEC has a difference of about 0.9%.

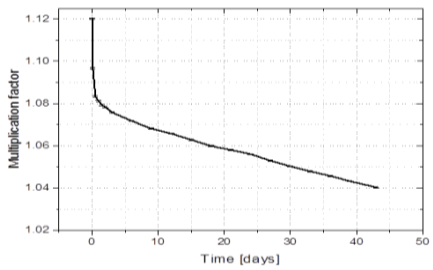


Figure 4 Multiplication factor value as a function of time

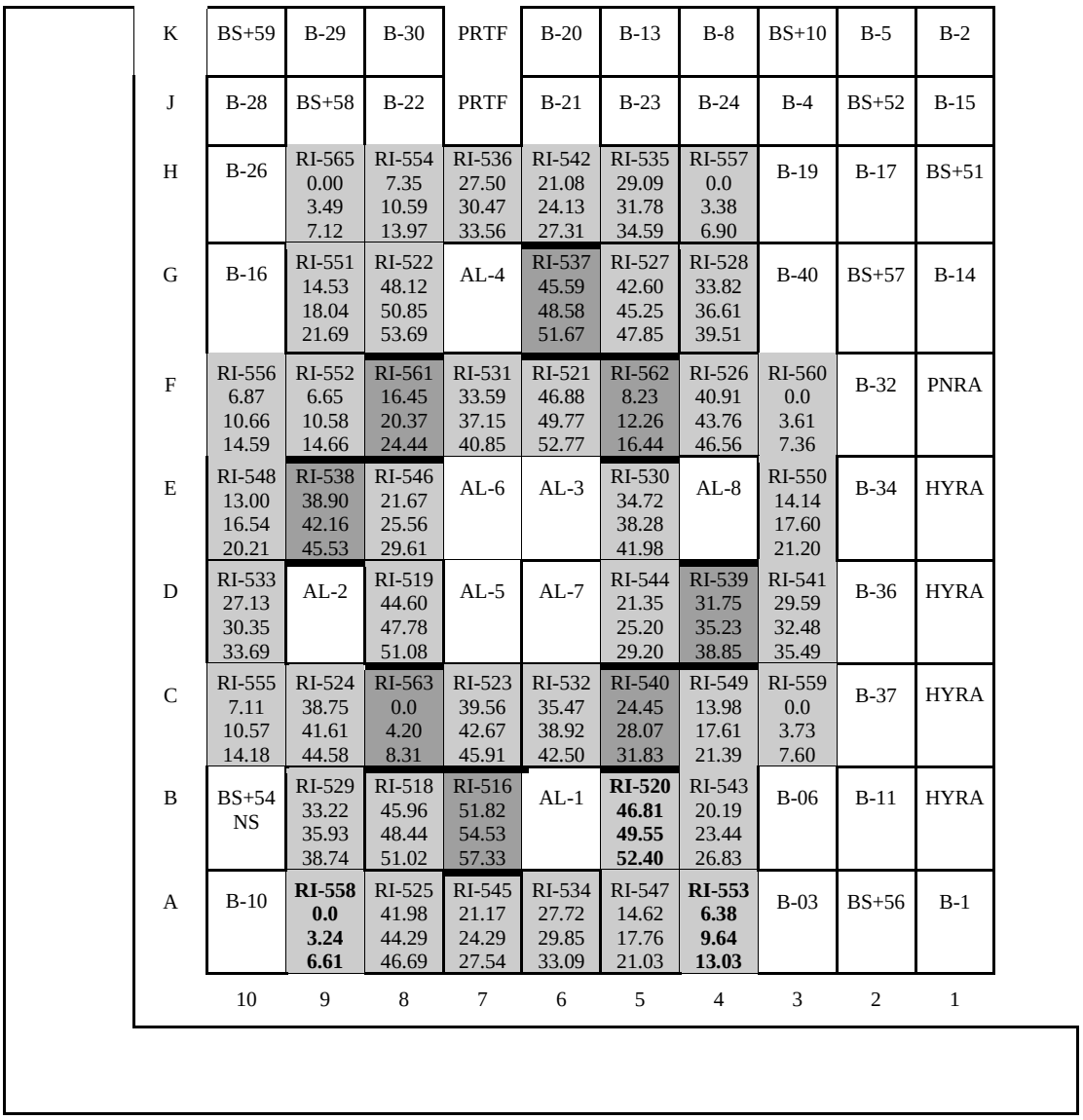


Figure 5 Calculated fuel burns up at BOEC (top), MOEC (middle), and EOEC (bottom) from Batan-FUEL code

The difference between the deterministic method and the Monte Carlo method lies in the neutron energy data used. The deterministic method uses neutron energy in the form of a fixed variable. In the simulation using the Monte Carlo method, continuous neutron energy is in the form of random variables and is probabilistic in nature. The calculation time is much longer than the simulation using Monte Carlo in terms of burn-up calculation and the input model is more detailed.

	10	9	8	7	6	5	4	3	2	1
K	BS+59	B-29	B-30	PRTF	B-20	B-13	B-8	BS+10	B-5	B-2
J	B-28	BS+58	B-22	PRTF	B-21	B-23	B-24	B-4	BS+52	B-15
H	B-26	RI-565 0.0 3.7 7.6	RI-554 7.2 10.6 14.1	RI-536 27.6 30.5 33.7	RI-542 21.4 24.5 27.7	RI-535 29.1 31.8 34.7	RI-557 0.0 3.6 7.3	B-19	B-17	BS+51
G	B-16	RI-551 14.4 18.0 21.8	RI-522 48.5 51.1 53.7	AL-4	RI-537 45.7 48.5 51.3	RI-527 42.6 45.2 47.8	RI-528 33.6 36.5 39.4	B-40	BS+57	B-14
F	RI-556 7.2 11.3 15.4	RI-552 6.4 10.4 14.5	RI-561 16.7 20.5 24.5	RI-531 33.6 36.9 40.4	RI-521 47.0 49.7 52.4	RI-562 8.0 11.8 16.0	RI-526 41.1 43.9 46.7	RI-560 0.0 3.7 7.7	B-32	PNRA
E	RI-548 12.8 16.5 20.3	RI-538 39.0 42.1 45.4	RI-546 22.2 25.8 29.5	AL-6	AL-3	RI-530 14.4 17.8 21.5	AL-8	RI-550 14.14 17.60 21.20	B-34	HYRA
D	RI-533 26.8 30.1 33.5	AL-2	RI-519 44.8 47.7 50.7	AL-5	AL-7	RI-544 21.4 25.0 28.8	RI-539 31.4 34.8 38.3	RI-541 25.3 28.3 31.5	B-36	HYRA
C	RI-555 7.2 10.9 14.8	RI-524 38.9 41.7 45.6	RI-563 0.0 4.1 8.4	RI-523 39.6 42.6 45.6	RI-532 35.2 38.4 41.8	RI-540 24.5 28.0 31.7	RI-549 14.4 18.0 21.8	RI-559 0.0 3.9 8.1	B-37	HYRA
B	BS+54 NS	RI-529 32.9 35.7 38.6	RI-518 46.3 48.7 51.2	RI-516 51.6 54.2 56.8	AL-1	**RI-520** **46.3** **49.0** **51.7**	RI-543 19.8 23.2 26.8	B-06	B-11	HYRA
A	B-10	**RI-558** **0.0** **3.5** **7.2**	RI-525 41.9 44.3 46.8	RI-545 21.4 24.6 28.1	RI-534 26.8 30.1 33.5	RI-547 14.4 17.7 21.3	**RI-553** **6.4** **10.0** **13.8**	B-03	BS+56	B-1

Figure 6 Calculated burn- at BOEC (top), MOEC (middle), and EOEC (bottom) from Serpent 2 calculation

The power distribution is also an important parameter for core analysis in which Figure 7 shows the normalized power distributions. The fuel assembly between two control rods has the lowest power, whereas the two neighboring outside fuel assemblies near the reflector have the highest power, as seen in this figure. In addition, the fuel assembly between two control rods has the greatest relative divergence, 0.0907.

The average normalized axial power distributions as illustrated in Figure 7 from BOEC (Beginning of Equilibrium Cycle and EOEC (End of Equilibrium Cycle). At the EOEC, the axial power is higher than BOEC in positions close to the top and bottom light water reflector, while in the center of the active core, the BOEC has the higher axial power. The largest relative difference is 0.099, which appears when the control rods are put between the upper core and the axial reflector.

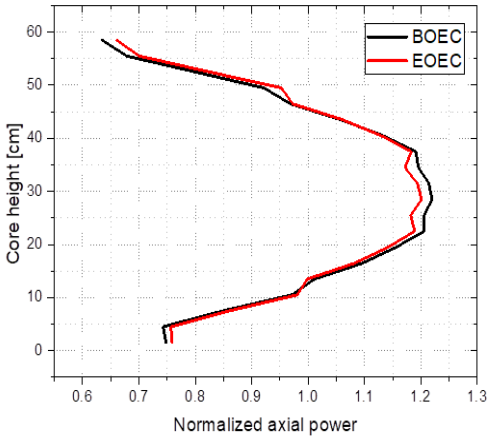


Figure 7 Normalized axial power distribution at BOEC and EOEC from Serpent 2 calculation

4.0 CONCLUSION

The neutronic calculation of the RSG-GAS research reactor has been done using two separate approaches, deterministic and stochastic methods. In the first stage, the WIMSD-5B is used to generate few-group cross-sections for a diffusion code Batan-FUEL which is then compared to Serpent 2 (Monte Carlo) simulations. Serpent 2 was chosen as the reference since Serpent's depletion module can handle detailed depletion chains with thousands of isotopes. The Batan-FUEL code is used to calculate the burn-up of the fuel element inside the RSG-GAS core. The WIMSD-5B/Batan-FUEL burn-up dependent calculations are in good agreement with Serpent2. The relative fuel burn-up differential in each burnup class is within 5.0 percent. Aside from the different calculation methods, the deviation reported when comparing WIMSD-5B/Batan-FUEL to Serpent 2 could be caused by the differences in modeling the Be block reflector, which will be examined in the future study.

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