# Development of Burnup Analysis System for Rotational and Spiral Fuel-Shuffling Scheme in Breed and-Burn Fast Rectors

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**Abstract**

In a previous study, we reported that it is possible to establish a rotational fuel-shuffling breed-and-burn fast reactor (RFBB) using metallic fuel and lead-bismuth coolant. In the rotational shuffling scheme, fresh fuel assemblies composed of natural uranium are loaded at the edge of core and approach the center with each shuffling move, then rotate back out again; or, fresh fuel assemblies are loaded from the core periphery, moved toward the center along a spiral path, and are eventually discharged from the core center. The purpose of this study is to develop a burnup analysis system for an RFBB using Serpent code, and to compare analysis results in the equilibrium cycle to those using the MVP/MVP-BURN code. The results compared included the effective multiplication factors, power density distribution, neutron flux distribution during the equilibrium cycle, and average burnup of discharged fuel. The reported values of the effective multiplication factors, k-eff, at equilibrium cycle showed a discrepancy of < 0.78%. The power density distribution, neutron flux distribution during the equilibrium cycle, and average burnup of discharged fuel were also in good agreement with values in the simulation using the MVP/MVP-BURN code.

## INTRODUCTION

The breed-and-burn (B&B) mode of operation, first proposed by Feinberg in 1958, is made self-sustaining by producing fissile materials such as Pu-239 (or Pu-233) from U-238 (or Th-232) during burnup [1]. In a B&B reactor, the fissile elements are first bred into the fuel by neutrons produced in the reactor. The resulting fuel is then directly burned in the same reactor to sustain the chain reaction and supply neutrons for breeding additional fuel. Therefore, B&B reactors can operate using primarily fertile fuel but also reload fuel, such as natural or depleted uranium, low-enriched uranium, used light-water reactor fuel, or thorium. Since a B&B core contains fertile materials and fission products, the neutron economy in the core must be improved to maintain criticality in operation, unlike other types of reactors with the same core size. To maintain critical operation, the feed fuel must be burned to a sufficient level such that the rate of neutron production increases with burnup and overtakes that of neutron absorption.

In a burning wave reactor, there are two types of B&B wave movement: radial-direction [2], [3]; and axial-direction, as in the CANDLE reactor [4]. In the CANDLE reactor, the burnup region propagates in the core axial direction at a constant speed without changing the shapes of neutron flux and power density distributions. In the CANDLE reactor concept, high-reactivity fuels are continuously placed in a high-neutron-importance region. However, fuels need to be exchanged in the axial direction, and such a method has not been established. If the radial-direction wave movement is considered, fuels can be exchanged easily per unit of assembly. However, the wave movement in the radial direction causes changes in neutron flux and power density distributions. One solution to this problem is to keep the B&B wave stable in the same region by optimized fuel shuffling, meaning a stationary wave reactor, [6], [7].

Recent B&B reactors studied at the Tokyo Institute of Technology are similar in concept to the reactors at the University of California, Berkeley (UCB) [2], [3], the Tokyo Institute of Technology CANDLE [4], and the Terra Power Traveling Wave [5]. In order to achieve criticality with the B&B mode of operation, and a stable power profile in the core, spiral and rotational-fuel-shuffling concepts have been proposed [6], [7]. In the fuel-shuffling concept, fresh fuel is first loaded at the periphery of the core and moved toward the center. It is subsequently moved outward again and discharged close to the periphery of the core.

In previous reports [6], [7], [8], spiral and rotational fuel-shuffling concepts were applied to small lead-bismuth cooled 450-MWt nuclear reactors. High-reactivity fuels were stably positioned at the core center where the high-neutron-importance region was located. As a result, the B&B region remained stationary, and the power profile and neutron flux distribution were almost unchanged during operation in the equilibrium state. The reactor concept was named the rotational fuel-shuffling breed-and-burn fast reactor (RFBB).

The main objective of the current study is to present verification results for the equilibrium cycle of the RFBB with spiral fuel shuffling using the Serpent code, a three-dimensional (3-D) continuous-energy Monte Carlo particle transport code [9], [10]. The Serpent code, in addition to being available for general use, allows modeling of complex 3-D geometries, uses the ENDF/B-VII.0 cross-section library, has fuel depletion capability, and an efficient and fast running algorithm [9], [10].

## Reactor core data and calculation methods

### General description of the reactor core and spiral fuel shuffling

The main core parameters are available in our previous work [8], and are listed in Table 1.

TABLE 1. MAIN CORE PARAMETERS.

|  |  |
| --- | --- |
| **Parameter** | **Value** |
| Thermal power (MW) | 450 |
| Core height (cm) | 220 |
| Core equivalent radius (cm) | 137.1 |
| Number of assemblies | 168 |
| Number of coolant channels | 1 |
| Number of fuel pins in an assembly | 271 |
| Inner cladding radius (mm) | 4.5 |
| Cladding thickness (mm) | 0.6 |
| Fuel pin pitch (mm) | 12 |
| Smeared density | 75% |
| Fuel volume fraction | 51.00% |
| Cladding volume fraction | 14.50% |
| Coolant volume fraction | 34.50% |

We adopted a small reactor core in a 44.5%Pb-55.5%Bi reactor with metallic fuel. The reactor core was composed of 168 fuel assemblies (FA), with one coolant channel located at the core center, surrounded by a 1.0-m thick reflector in each direction. The core was designed to generate a total power of 450 MWth. It used binary metallic fuel natural U-10 wt% Zr. The structural and cladding material was 9Cr-ODS ferritic-martensitic alloy [11], and the bond material was sodium. The FAs had a hexagonal geometry with inner and outer flat-to-flat distance of assembly of 20.1 and 23.2 cm, respectively. The FAs were made of 271 fuel rods with a pin pitch of 12. More details on the core design can be found in [8]. All fuels were composed of natural uranium at the start of the simulation. Through the iteration calculations, the core achieved an equilibrium state. When the change of effective multiplication factors (k-eff) during the burnup cycle was within range of the statistical error, we considered the core to have reached an equilibrium state. We confirmed that the changes of major fissile nuclides converged in the equilibrium state.

In the spiral fuel-shuffling scheme, as shown in Fig. 1, fuels are loaded from the periphery and moved toward the center region step by step along a spiral path, until being discharged at the core center. Once the reactor could achieve an equilibrium state, high-reactivity fuels (those in a neutron-producer state) could be continuously positioned at the center region, maintaining a stable burning region, and retaining a high neutron importance region at the core center [8].



FIG. 1. Schematic diagram of spiral fuel shuffling.

### Calculation method

We used the Monte Carlo Serpent neutronic code [9] with the ENDF-B/VII nuclear data library [12] for the neutronic calculations, with 5000 neutron histories per cycle and 200 active cycles, skipping the first 20 cycles for statistical treatment to obtain a target statistical error in k-eff of ~150 pcm. The cross-section library was used with average temperatures of fuel, cladding, and coolant set at 800, 700, and 700 K, respectively, in the operating condition. The burnup calculation was carried out for one-sixth of the core region in the azimuthal direction. The axial direction was symmetrically divided into 11 burnup regions.

Fresh FAs were loaded from outside the core. The fuels were moved inward, reached and the core center, and were discharged after 28 fuel-shuffling steps. Figure 1 shows the fuel-shuffling pattern and position ID. Each FA position was assigned a position ID and the fuel-shuffling flowed from FA position ID 28 to 1. In this paper, the duration from FA loading to discharge from the core was defined as one fuel cycle length (FCL). The average fuel burnup in the equilibrium state was estimated as follows:

Average fuel burnup in MWd/t = (Core power in MWt) x (FCL in days) / (Consumed uranium during FCL in tons).

To calculate the fuel movement to achieve the equilibrium state of the RFBB, a computer interface program was developed to automatically conduct core burnup calculations from cycle to cycle. In this procedure, the equilibrium condition was determined based on the convergence of the k-eff, neutron flux, and nuclide number density values of certain isotopes, such as Pu-239, U-235, and U-238. The calculation procedure was performed as follows:

(1) Design parameters such as core size, FAs, fuel-shuffling pattern, operational conditions, and fuel cycle length were prepared in the input file.

(2) Core burnup calculations were performed to obtain end-of-cycle (EOC) fuel compositions, The end of cycle is the end of burnup before fuels are shuffled.

(3) The input file was updated with the new fuel compositions obtained in the previous calculational cycle along with the fuel-shuffling pattern. The core burnup calculation for the next cycle was performed using the new input file.

(4) Steps (2) and (3) were repeated until the core reached the equilibrium condition.

## Serpent code results and discussion

In this study, we analyzed and compared the detailed burnup characteristics of the equilibrium cycle in the case of a 620-day interval with the results using the MVP code in a previous work [8]. We found that the average discharged burnup was 114.95 GWd/t-HM, about a 0.02% difference compared to the 115 GWd/t-HM determined with the MVP code.

TABLE 2. CHANGE OF K-EFF DURING THE EQUILIBRIUM CYCLE IN 620-DAY SHUFFLING INTERVAL.

|  |  |  |  |  |  |
| --- | --- | --- | --- | --- | --- |
| Burnup step | MVP | ±∆MVP | Serpent | ±∆Serpent | (Serpent-MVP)/MVP, [%] |
| 0.00 | 1.0009 | 0.0004 | 0.9956 | 0.0007 | -0.53 |
| 0.25 | 1.0058 | 0.0005 | 1.0006 | 0.0005 | -0.51 |
| 0.50 | 1.0085 | 0.0005 | 1.0033 | 0.0009 | -0.51 |
| 0.75 | 1.0110 | 0.0004 | 1.0069 | 0.0008 | -0.41 |
| 1.00 | 1.0149 | 0.0004 | 1.0092 | 0.0007 | -0.56 |
| 1.00 | 1.0023 | 0.0004 | 0.9969 | 0.0009 | -0.54 |
| 1.25 | 1.0064 | 0.0004 | 0.9986 | 0.0007 | -0.78 |
| 1.50 | 1.0080 | 0.0004 | 1.0033 | 0.0009 | -0.46 |
| 1.75 | 1.0104 | 0.0003 | 1.0080 | 0.0006 | -0.24 |
| 2.00 | 1.0155 | 0.0006 | 1.0114 | 0.0008 | -0.41 |
| 2.00 | 1.0014 | 0.0004 | 0.9971 | 0.0010 | -0.43 |
| 2.25 | 1.0057 | 0.0002 | 0.9999 | 0.0007 | -0.58 |
| 2.50 | 1.0083 | 0.0005 | 1.0041 | 0.0009 | -0.42 |
| 2.75 | 1.0122 | 0.0007 | 1.0053 | 0.0008 | -0.69 |
| 3.00 | 1.0143 | 0.0004 | 1.0100 | 0.0008 | -0.42 |

Table 2 presents the k-eff and respective standard deviation for the equilibrium core. The Monte Carlo calculations with the Serpent code produced standard deviations smaller than 10 pcm. The results in Table 2 also indicate the k-eff values obtained in the Serpent calculation were smaller those of the MVP calculation. The percentage difference of k-eff was < 0.78% compared to that of the MVP simulation. The comparison of effective multiplication factors for full core at equilibrium condition in Table 2 show acceptable agreement. The possible reasons for the discrepancy include using different nuclear data libraries and burnup chains in the Serpent and MVP calculations. The comparison of radial average power density distribution and linear power density distribution at the beginning of the equilibrium cycle (BOEC) and end of the equilibrium cycle (EOEC) are shown in Tables 3 and 4, respectively.

The BOEC marks the beginning of burnup after the fuels are shuffled into an equilibrium burning condition, while the EOEC is the end of burnup before fuels are shuffled in the equilibrium burning condition. The maximum radial power is located in the peripheral region. These fuels are in the breeding state, with much smaller power than the neutron producer state fuels [8]. We also found high discrepancies corresponding to the FAs near the periphery of the core due to the low power values in this region.

TABLE 3. AVERAGE POWER DENSITY DISTRIBUTION AT THE BOEC AND EOEC.

|  |  |  |  |
| --- | --- | --- | --- |
|   | Serpent | MVP | (Serpent-MVP)/MVP, [%] |
| FA | BOEC | EOEC | BOEC | EOEC | BOEC | EOEC |
| 28 | 275.1 | 261.0 | 271.2 | 254.2 | 1.4 | 2.7 |
| 27 | 232.4 | 221.8 | 242.8 | 231.6 | -4.3 | -4.2 |
| 26 | 242.3 | 233.0 | 225.2 | 217.2 | 7.6 | 7.3 |
| 25 | 161.3 | 158.0 | 181.9 | 178.5 | -11.3 | -11.5 |
| 24 | 178.6 | 176.8 | 176.2 | 174.6 | 1.4 | 1.2 |
| 23 | 170.3 | 170.5 | 149.5 | 149.9 | 13.9 | 13.8 |
| 22 | 89.4 | 90.0 | 109.3 | 110.5 | -18.2 | -18.6 |
| 21 | 105.3 | 107.7 | 111.0 | 113.3 | -5.1 | -5.0 |
| 20 | 103.6 | 107.9 | 97.2 | 100.4 | 6.5 | 7.5 |
| 19 | 91.1 | 95.0 | 74.6 | 77.7 | 22.2 | 22.2 |
| 18 | 37.2 | 38.5 | 50.0 | 52.1 | -25.5 | -26.2 |
| 17 | 47.6 | 49.8 | 53.3 | 56.4 | -10.7 | -11.6 |
| 16 | 48.6 | 52.4 | 48.7 | 52.4 | -0.1 | 0.2 |
| 15 | 43.6 | 47.5 | 38.5 | 42.1 | 13.3 | 12.7 |
| 14 | 35.8 | 38.7 | 27.3 | 29.6 | 31.4 | 30.7 |
| 13 | 12.1 | 12.5 | 17.0 | 18.6 | -29.0 | -32.6 |
| 12 | 16.2 | 17.5 | 19.6 | 21.5 | -17.0 | -18.5 |
| 11 | 17.4 | 19.1 | 18.5 | 20.5 | -5.9 | -6.8 |
| 10 | 16.1 | 18.1 | 15.4 | 17.5 | 4.6 | 3.3 |
| 9 | 13.2 | 15.1 | 11.3 | 13.1 | 16.4 | 15.5 |
| 8 | 10.3 | 11.7 | 7.6 | 8.6 | 35.4 | 36.5 |
| 7 | 4.3 | 4.8 | 5.3 | 6.2 | -18.7 | -21.5 |
| 6 | 5.2 | 5.6 | 6.1 | 7.1 | -15.7 | -21.3 |
| 5 | 5.5 | 6.2 | 6.1 | 7.1 | -10.9 | -12.8 |
| 4 | 5.1 | 6.0 | 5.1 | 6.1 | -1.0 | -0.6 |
| 3 | 4.3 | 5.2 | 4.1 | 4.8 | 6.0 | 8.2 |
| 2 | 3.3 | 3.9 | 2.7 | 3.4 | 22.5 | 16.1 |
| 1 | 2.4 | 3.0 | 2.0 | 2.5 | 20.9 | 20.0 |

TABLE 4. LINEAR POWER DENSITY DISTRIBUTION VALUE AT THE BOEC AND EOEC.

|  |  |  |  |
| --- | --- | --- | --- |
|   | Serpent | MVP | (Serpent-MVP)/MVP, [%] |
| FA | BOEC | EOEC | BOEC | EOEC | BOEC | EOEC |
| 28 | 175.0 | 166.0 | 172.5 | 161.7 | 1.4 | 2.7 |
| 27 | 147.8 | 141.1 | 154.5 | 147.4 | -4.3 | -4.3 |
| 26 | 154.1 | 148.2 | 143.3 | 138.2 | 7.6 | 7.2 |
| 25 | 102.6 | 100.5 | 115.7 | 113.6 | -11.3 | -11.5 |
| 24 | 113.6 | 112.5 | 112.1 | 111.1 | 1.4 | 1.2 |
| 23 | 108.3 | 108.5 | 95.1 | 95.3 | 13.9 | 13.8 |
| 22 | 56.8 | 57.2 | 69.5 | 70.3 | -18.2 | -18.6 |
| 21 | 67.0 | 68.5 | 70.6 | 72.1 | -5.1 | -5.0 |
| 20 | 65.9 | 68.6 | 61.8 | 63.9 | 6.6 | 7.4 |
| 19 | 58.0 | 60.4 | 47.7 | 49.5 | 21.5 | 22.1 |
| 18 | 23.7 | 24.5 | 31.8 | 33.1 | -25.5 | -26.1 |
| 17 | 30.3 | 31.7 | 33.9 | 35.9 | -10.7 | -11.7 |
| 16 | 30.9 | 33.4 | 31 | 33.3 | -0.2 | 0.2 |
| 15 | 27.8 | 30.2 | 24.5 | 26.8 | 13.3 | 12.7 |
| 14 | 22.8 | 24.6 | 17.3 | 18.8 | 31.7 | 31.1 |
| 13 | 7.7 | 8.0 | 10.8 | 11.8 | -29.0 | -32.4 |
| 12 | 10.3 | 11.1 | 12.5 | 13.7 | -17.3 | -18.8 |
| 11 | 11.1 | 12.2 | 11.8 | 13 | -6.2 | -6.4 |
| 10 | 10.2 | 11.5 | 9.8 | 11.1 | 4.5 | 3.7 |
| 9 | 8.4 | 9.6 | 7.2 | 8.3 | 16.5 | 15.6 |
| 8 | 6.5 | 7.5 | 4.8 | 5.5 | 36.2 | 35.7 |
| 7 | 2.7 | 3.1 | 3.4 | 3.9 | -19.2 | -21.1 |
| 6 | 3.3 | 3.6 | 3.9 | 4.5 | -15.8 | -20.9 |
| 5 | 3.5 | 3.9 | 3.9 | 4.5 | -11.0 | -12.7 |
| 4 | 3.2 | 3.8 | 3.3 | 3.9 | -2.4 | -1.6 |
| 3 | 2.7 | 3.3 | 2.6 | 3 | 5.1 | 9.2 |
| 2 | 2.1 | 2.5 | 1.7 | 2.2 | 24.6 | 14.2 |
| 1 | 1.5 | 1.9 | 1.3 | 1.6 | 18.3 | 19.9 |

TABLE 5. NEUTRON FLUX DISTRIBUTION VALUE AT THE BOEC AND EOEC.

|  |  |  |  |
| --- | --- | --- | --- |
|   | Serpent | MVP | (Serpent-MVP)/MVP, [%] |
| FA | BOEC | EOEC | BOEC | EOEC | BOEC | EOEC |
| 28 | 2.03E+15 | 1.91E+15 | 2.03E+15 | 1.90E+15 | 0.0 | 0.5 |
| 27 | 1.72E+15 | 1.63E+15 | 1.82E+15 | 1.73E+15 | -5.4 | -5.5 |
| 26 | 1.82E+15 | 1.73E+15 | 1.72E+15 | 1.64E+15 | 5.9 | 5.7 |
| 25 | 1.24E+15 | 1.20E+15 | 1.42E+15 | 1.37E+15 | -12.8 | -12.5 |
| 24 | 1.42E+15 | 1.36E+15 | 1.41E+15 | 1.36E+15 | 0.1 | 0.0 |
| 23 | 1.40E+15 | 1.35E+15 | 1.24E+15 | 1.20E+15 | 12.7 | 12.5 |
| 22 | 7.64E+14 | 7.49E+14 | 9.45E+14 | 9.26E+14 | -19.2 | -19.1 |
| 21 | 9.35E+14 | 9.19E+14 | 1.00E+15 | 9.83E+14 | -6.8 | -6.5 |
| 20 | 9.77E+14 | 9.63E+14 | 9.28E+14 | 9.14E+14 | 5.3 | 5.5 |
| 19 | 9.14E+14 | 8.97E+14 | 7.56E+14 | 7.48E+14 | 20.9 | 19.9 |
| 18 | 3.95E+14 | 3.93E+14 | 5.31E+14 | 5.33E+14 | -25.6 | -26.3 |
| 17 | 5.26E+14 | 5.24E+14 | 6.04E+14 | 6.04E+14 | -12.8 | -13.3 |
| 16 | 5.79E+14 | 5.83E+14 | 5.92E+14 | 5.92E+14 | -2.1 | -1.6 |
| 15 | 5.69E+14 | 5.69E+14 | 5.11E+14 | 5.14E+14 | 11.4 | 10.7 |
| 14 | 5.03E+14 | 5.02E+14 | 3.89E+14 | 3.94E+14 | 29.4 | 27.2 |
| 13 | 1.80E+14 | 1.82E+14 | 2.58E+14 | 2.66E+14 | -30.3 | -31.4 |
| 12 | 2.57E+14 | 2.60E+14 | 3.15E+14 | 3.22E+14 | -18.6 | -19.3 |
| 11 | 2.97E+14 | 3.04E+14 | 3.27E+14 | 3.34E+14 | -9.2 | -9.0 |
| 10 | 3.05E+14 | 3.13E+14 | 3.00E+14 | 3.07E+14 | 1.9 | 2.0 |
| 9 | 2.82E+14 | 2.89E+14 | 2.44E+14 | 2.52E+14 | 15.6 | 14.8 |
| 8 | 2.38E+14 | 2.44E+14 | 1.78E+14 | 1.85E+14 | 33.8 | 31.8 |
| 7 | 8.90E+13 | 9.31E+13 | 1.24E+14 | 1.30E+14 | -28.0 | -28.1 |
| 6 | 1.23E+14 | 1.26E+14 | 1.58E+14 | 1.64E+14 | -21.7 | -23.4 |
| 5 | 1.50E+14 | 1.53E+14 | 1.73E+14 | 1.82E+14 | -13.5 | -16.0 |
| 4 | 1.59E+14 | 1.65E+14 | 1.69E+14 | 1.75E+14 | -5.9 | -5.9 |
| 3 | 1.54E+14 | 1.62E+14 | 1.46E+14 | 1.53E+14 | 5.4 | 5.7 |
| 2 | 1.36E+14 | 1.44E+14 | 1.15E+14 | 1.21E+14 | 17.8 | 18.8 |
| 1 | 1.13E+14 | 1.17E+14 | 8.85E+13 | 9.32E+13 | 27.6 | 25.6 |

TABLE 6. AXIAL DISTRIBUTION OF BURNUP VALUE IN THE DISCHARGED FUEL.

|  |  |
| --- | --- |
| Axial region [cm] | Burnup [GWd/t-HM] |
|  | MVP | Serpent | (Serpent-MVP)/MVP, [%] |
| 0–20, 200–220 | 37.5 | 37.92 | 1.12 |
| 20–40, 180–200 | 63.3 | 62.89 | -0.65 |
| 40–60, 160–180 | 105.1 | 106.45 | 1.28 |
| 60–80, 140–160 | 148.8 | 149.62 | 0.55 |
| 80–100, 120–140 | 181.2 | 181.15 | -0.03 |
| 100–120 | 192.5 | 193.8 | 0.68 |

Table 5 presents the results of a comparison of the neutron flux distribution at the BOEC and EOEC. The change in neutron flux distribution was small during the equilibrium burning condition, and a high neutron flux region was retained at the core center. This is because a high neutron flux region can be approximated as a region of high neutron importance; such regions are retained at the core center in this equilibrium cycle. Table 5 shows that the results obtained here were in good agreement with those from our previous study using the MVP code. The possible reasons for any discrepancy are the same as those discussed above, i.e., the different nuclear data libraries and burnup chains in the two codes.

Table 6 shows the discharged burnup value in the axial direction. The burnup calculations of the Serpent code were again in good agreement with the reported values of the MVP codes. From these comparisons, we concluded that the current core model with the Serpent code showed analysis results in acceptable agreement with those of the MVP code. This indicates that the Serpent code model can be used to design studies of RFBB.

## Conclusions

This study sought to develop a burnup analysis system for RFBB with a shuffling scheme using the Serpent code, and to then compare analysis results in the equilibrium cycle to those obtained using the MVP/MVP-BURN code. The RFBB was the chosen core model, with a spiral fuel-shuffling scheme. The burnup characteristics at the equilibrium cycle were selected for comparison with those in the MVP calculation. The comparative results included the effective multiplication factors, power density distribution, neutron flux distribution during the equilibrium cycle, and average burnup of discharged fuel. The reported values of the effective multiplication factors, k-eff, at the equilibrium cycle showed a discrepancy of < 0.78%. The power density distribution, neutron flux distribution during the equilibrium cycle, and average burnup of discharged fuel were in good agreement with the values in the simulation using the MVP/MVP-BURN code. The Serpent code model will work well in design studies of B&B reactors.

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