Molten Salt Breeder Reactor Analysis Based on Unit Cell Model

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1. Introduction

For the research of Molten Salt Breeder Reactor (MSBR), we are developing computer code systems. [1] Contemporary computer codes like the MCNP6 [2] or SCALE [3] are only good for solving a fixed solid fuel reactor. However, due to the molten-salt fuel, MSR analysis needs some functions such as online reprocessing & refueling, and circulating fuel. J.J. Power of Oak Ridge National Laboratory (ORNL) suggested in 2013 a method for simulating the Molten Salt Breeder Reactor (MSBR) with SCALE, which does not support continuous material processing. [4] In order to simulate MSR characteristics, the method proposes dividing a depletion time into short time intervals and batchwise reprocessing & refueling at each step. We are applying this method by using the MCNP6 & PYTHON and NEWT-TRITON-PYTHON & PYTHON code systems to MSBR. This paper contains various parameters to analyze the MSBR unit cell model such as the multiplication factor, breeding ratio, change of amount of fuel, amount of fuel feeding, and neutron flux distribution.

2. Methods and Results

This section contains some of the reprocessing method used to develop two code systems for MSR, MSBR specifications and the various parameters that shows characteristics of MSR.

2.1 Reprocessing Method

The MSBR is operated by removing fission products and actinides, and adding fertile material (^{232}Th) continuously. It also has characteristics for in dealing with materials during reprocessing & refueling; it can remove all volatile gases (e.g., ^{135}Xe) and noble metals in 20 seconds. The MSBR separates Protactinium-233 (^{233}Pa) from the molten-salt fuel over 3 days, and allows it to decay to Uranium-233 (^{233}U) and be recovered. The reactor prevents ^{233}Pa from absorbing neutrons, and increases the efficiency of breeding ^{233}U in the core. Other fission products have specific removal rates. If the depletion time intervals are very short owing to the material being instantly removed, it is hard to calculate long depletion times of MSBR by computer code. For that reason, the unit time interval is set to 3 days stand for the removal time of ^{233}Pa. ^{232}Th is added to breed ^{233}U and to keep the initial amount of

^{232}Th. The total depletion time is 20 years. Figure 1 shows the depletion procedure.
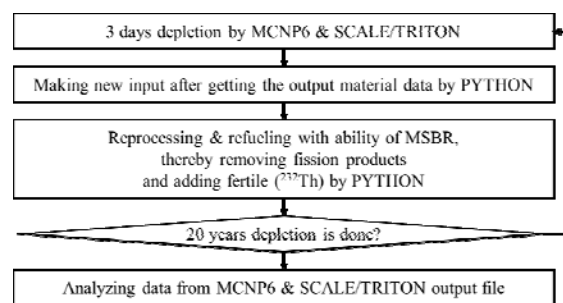

Fig. 1. A depletion calculation procedure of the MSBR

2.2 MSBR Specification

The Molten Salt Breeder Reactor project by ORNL provides the characteristics of the MSBR reactor concept. The MSBR is a thermal spectrum reactor with a power level of 2250MWth. The MSBR uses molten-salt fuel (LiF-BeF$_2$-UF$_4$-ThF$_4$) and bare graphite moderator. Its reactor core consists of two zones one flow. It means the same composition of molten-salt fuel is divided into two zones in the reactor core, and the graphite to fuel ratio is different in each zone. In this calculation, the two zone design is replaced by a single zone maintaining a volume of molten-salt fuel and graphite. The size of the square shaped graphite cell of Zone I and Zone II are the same. The fuel channel radius is determined by the volume fraction of molten-salt fuel in the unit cell (20.6%) that represents the average volume fraction of molten-salt fuel in Zone I (13%) and Zone II (37%). Fig 2 shows the geometry for the unit cell calculation. Initial fuel loading composition is roughly the same as MSBR, 71.8 LiF – 16 BeF$_2$ – 12 ThF$_4$ – 0.2 UF$_4$. The boundary condition of this unit cell is reflective. The density of the molten-salt is 3.28g/cm^3 and that of graphite is 1.84g/cm^3 at 900K. The temperature of fuel and graphite is fixed to 900K.
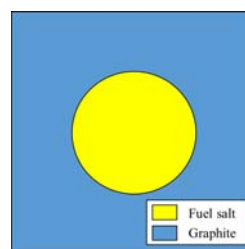

Fig. 2. MSBR unit cell geometry for MCNP6 & SCALE/TRITON depletion

2.3 Result Parameters

Fig 3 represents the infinite multiplication factor calculated by MCNP6 and SCALE. The value denoted on the figure is the initial infinite multiplication factor of every depletion time interval. The infinite multiplication factor of MCNP6 is fluctuating because of a high standard deviation (around 200pcm). An infinite multiplication factor calculated by a small standard deviation is the mid-point of fluctuating K-infinite because the reaction rate for the depletion calculation is already converged. This means that the fluctuating K-infinite is bounded by a designated value that is determined by material composition. K-infinite of the two code systems show differences. It will be explained by Fig 5.
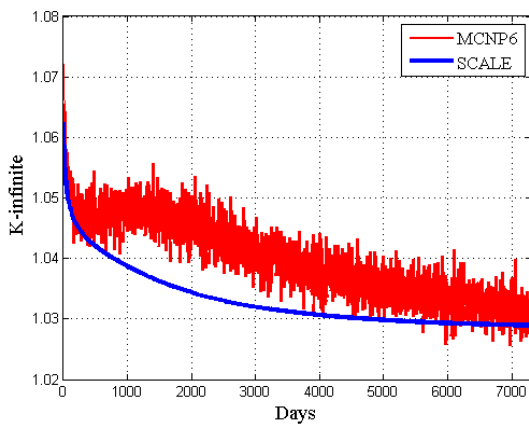

Fig. 3. Multiplication factor of MCNP6 and TRITON for 20 years depletion

In fig 4, the breeding ratio is calculated from the characteristics of isotopes in the thorium cycle and the very fine amount of material information according to short depletion time interval. The average breeding ratio calculated from the result of MCNP6 is 1.0045, and from the result of SCALE is 1.0027. The breeding ratios are slightly higher than one.
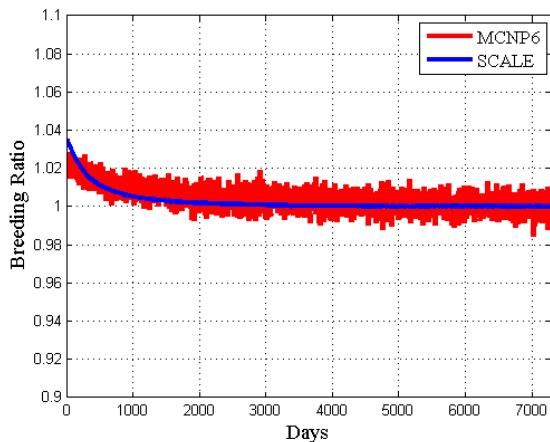

Fig. 4. Breeding ratio of MCNP6 and SCALE for 20 years depletion

Fig 5 shows the number density of major isotopes that have a strong impact on the states of the core. Every isotope represents the beginning of each depletion time interval, except for ^{233}Pa because it is removed in these intervals, so it represents the end of the depletion time intervals. Every isotope denoted in Fig 5, except ^{235}U, is well matched with each other. However, MCNP6 has twice an amount of ^{235}U than SCALE. ^{235}U mainly originates from the ^{234}U neutron capture. It has a large influence on other parameters. Table I shows the total amount of ^{232}Th feeding during 20 years depletion.
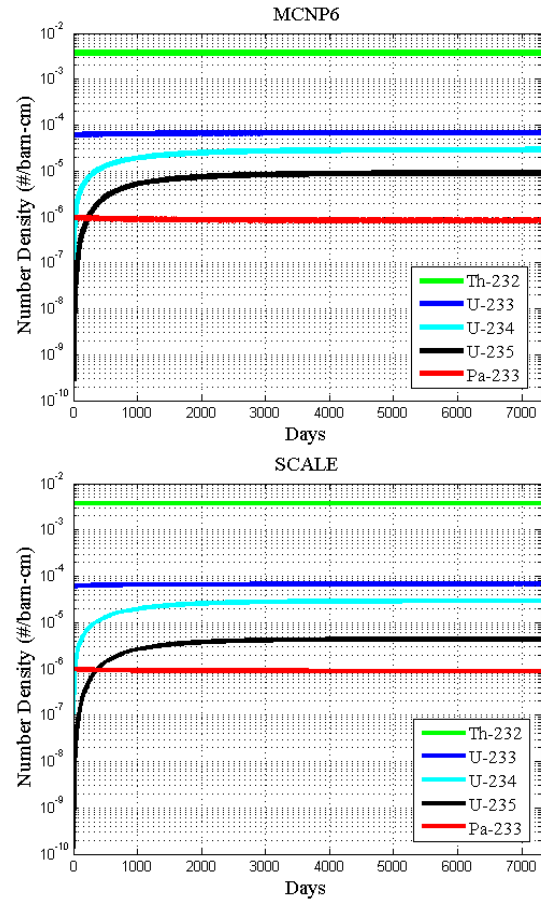

Fig. 5. Material number density for 20 years depletion of MCNP6 and SCALE

Table I. Total amount of feeding per unit cell in each computer code system during 20 years

Code system	MCNP6	SCALE
Thorium-232 (kg)	4.45	4.71

Fig 6 shows the normalized neutron flux distribution of the initial and equilibrium states; from the results, MCNP6 is slightly more moderated in the initial state than SCALE. On the contrary, MCNP6 is slightly less moderate than SCALE in the equilibrium state. The difference of the initial state is negligible, but that of the equilibrium state is considerable. At the equilibrium, MCNP6 has twice the amount of ^{235}U than SCALE, so it consumes a large amount of thermal neutrons. ^{235}U mainly causes the difference in the normalized neutron flux distribution.

Fig. 6 Initial and equilibrium state of normalized neutron flux of MCNP6 and SCALE

3. Conclusions

The result of MCNP6 and NEWT module in SCALE show some difference in depletion analysis, but it still seems that they can be used to analyze MSBR. Using these two computer code system, it is possible to analyze various parameters for the MSBR unit cells such as the multiplication factor, breeding ratio, amount of material, total feeding, and neutron flux distribution. Furthermore, the two code systems will be able to be used for analyzing other MSR model or whole core models of MSR.

ACKNOWLEDGMENTS

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[4] J. J. Powers, T. J. Harrison, and J. C. Gehin, "A new approach for modeling and analysis of molten salt reactors using scale",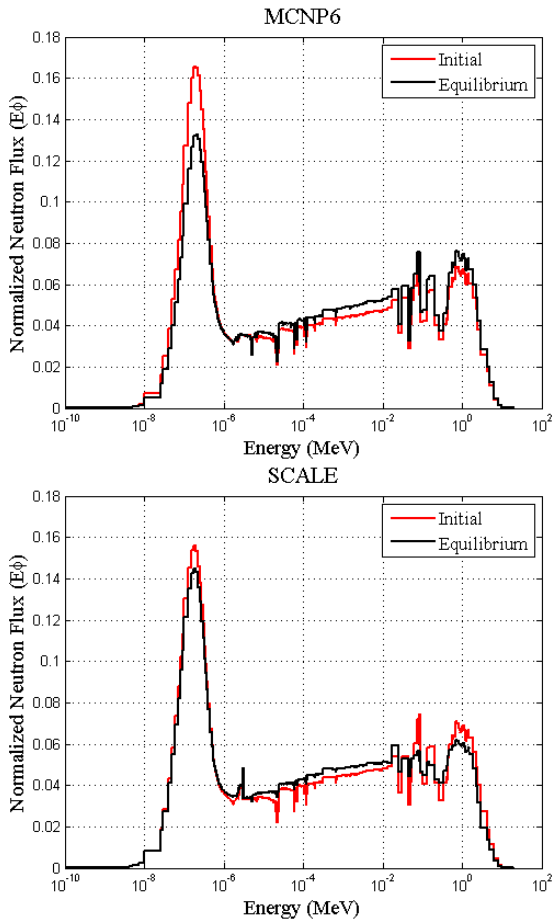 M&C, Sun Valley, Idaho, May 5-9, vol 2, pp. 803 – 815, 2013