Vol 17 No 3, March 2008
1674-1056/2008/17(03)/0896-05
Chinese Physics B
© 2008 Chin. Phys. Soc.
and IOP Publishing Ltd

The influence of reactor core parameters on effective breeding coefficient $K_{\rm eff}$*

Liu Li-Po(刘立坡)[a)b)], Liu Yi-Bao(刘义保)[a)b)†], Wang Juan(王 娟)[b)], Yang Bo(杨 波)[b)], and Zhang Tao(张 涛)[b)]

[a)] *Key Laboratory of Nuclear Resources and Environment (East China Institute of Technology), Ministry of Education, Nanchang 330013, China*

[b)] *School of Nuclear Engineering and Technology, East China Institute of Technology, Fuzhou 344000, China*

(Received 20 May 2007; revised manuscript received 21 September 2007)

The values of effective breeding coefficient $K_{\rm eff}$ in a reactor core of nuclear power plant are calculated for different values of parameters (core structure, fuel assembly component) by using the Monte Carlo method. The obtained values of $K_{\rm eff}$ are compared and analysed, which can provide theoretical basis for reactor design.

Keywords: Monte Carlo method, reactor core parameter, effective breeding coefficient $K_{\rm eff}$
PACC: 2841, 2850

1. Introduction

The reason why the nuclear power application develops so slowly lies in difficulty in choosing design parameters but not in lacking appropriate design schemes. We cannot judge which scheme is best due to our insufficient practical experience on this matter, although there are several schemes of reactor design that are available for reference. Generally, deterministic method is applied to reactor design. However, when a reactor structure is quite complex and asymmetric, the deterministic method cannot offer an accurate value of effective breeding coefficient $K_{\rm eff}$. At this point, the Monte Carlo (MC) method is used frequently to solve the problems in areas of reactor such as shield calculation, radiation destruct and so on, but rarely to solve the problems involved in reactor core.[1] In the present paper we calculate the values of $K_{\rm eff}$ at different values of parameters such as core structure, fuel assembly component and so on in a reactor core of nuclear power plant by using the MCNP4C code of Monte Carlo method. The obtained values of $K_{\rm eff}$ are compared and the results of comparison are also analysed.

2. Reactor core framework

A reactor core of nuclear power plant is comprised of fuel assemblies (see Fig.1), control elements, lumped burnable poison (LBP), neutron source and the resistance plug. The fuel assemblies are pivotal parts within reactor core framework, and they are located in a high temperature and high pressure region, where they are irradiated by both intense neutrons and γ-rays and suffer from boracic water eroding, scouring and hydraulic vibration as well. The function of fuel assemblies directly influences the safety and economy of reactor and can also improve the performance of the reactor.

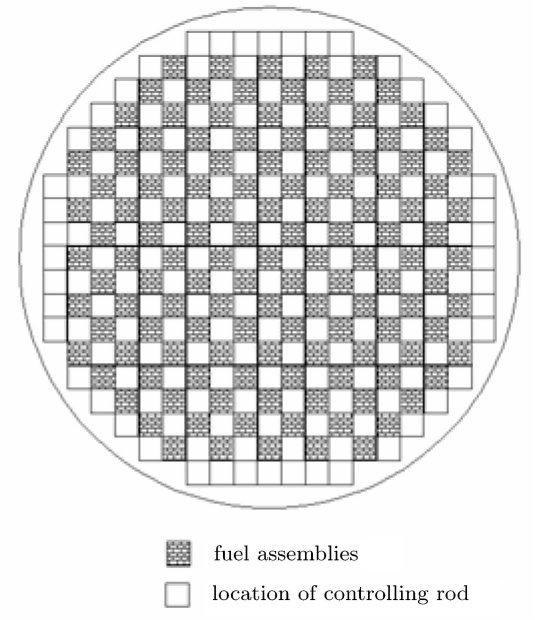


Fig.1. Schematic of reactor core structure.

*Project supported by the National Natural Science Foundation of China (Grant No 10647109).
†Corresponding author. E-mail: liuyb01@mails.tsinghua.edu.cn

The fuel assemblies include fuel rods (Fig.2), assembly framework, and expansion spring. The fuel rod is designed as a four-square assembly by 15×15, comprised of 20 control rod guide tubes, a neutron flux measurement tube, and 204 fuel rods.

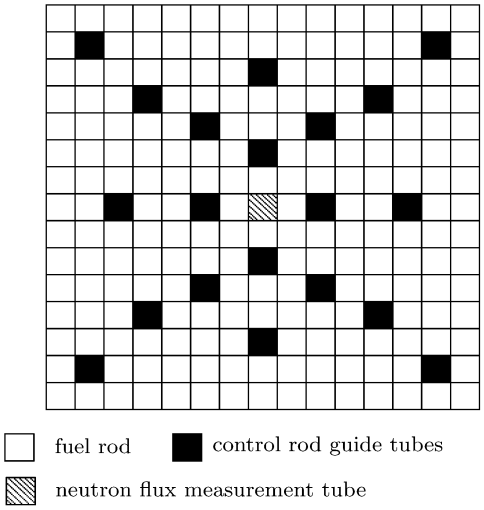


Fig.2. Schematic of fuel assemblies.

The fuel rods are important parts that engender fission and release energy. In the fuel rods, the UO_2 fuel pins, Al_2O_3 adiabatic slice and expansion spring are packed in Zirconium-4 alloys tube, and the both ends are welded by Zirconium-4 sealing alloys.

The fuel pins are of small porcelain agglomeration, which are made of low enriched powder of UO_2. It is 95±1.5% (10.4g/cm^3) in theoretical density, 8.43mm in diameter and 10mm in height. The clad is 10mm in diameter, 3176mm in length, 0.7mm in thickness, with the gap being 0.0085mm. Its section is shown in Fig.3.

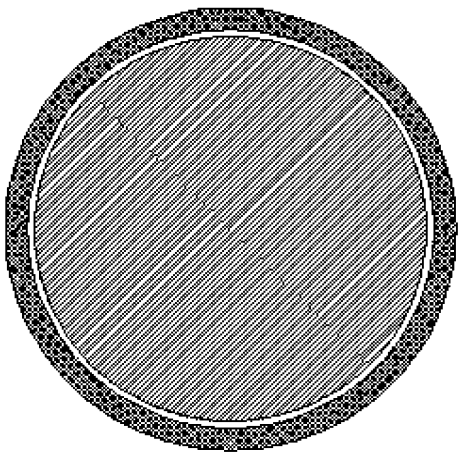


Fig.3. Schematic of fuel rod.

In order to improve its ability to absorb hydrogen, the clad is made of Zirconium-4 alloys. The top of the fuel rods reaches 250mm axially for accommodating fission gas which is released by UO_2 fuel, and compensating the axial inflation that is caused by hot environment.

3. The Monte Carlo method

3.1. MCNP4C

MCNP4C is one of Monte Carlo transport codes, of which the full name is Mont Carlo Neutron and Photo MCNP Transport Code. The MCNP4C is a general-purpose Monte Carlo transport code for neutrons, photons and electrons developed by American Los Alamos national laboratory. This code contains pointwise cross-section data and treats the geometry of MCNP as an arbitrary three-dimensional configuration, and it has many variant reduction techniques, so it is called the 'super-Monte Carlo program'. The MCNP4C can be used in several transport modes: neutron, photon, electron, or coupled neutron, photon, electron transport, including the capability of calculating eigenvalues for critical systems.[1−7] It does not solve an explicit equation, but obtains answers by simulating individual particles and recording some aspects of their average behaviour.

3.2. The geometry

Figures 1 and 2 show that the geometric configuration in reactor is very complex, so it is very difficult to describe. According to the array character of fuel module, the workload can be reduced to a great degree by using the repeated structure in the MCNP to describe the reactor.

The first thing to do is to establish a critical calculating model in the MCNP code, in which the fuel assemblies, control elements and neutron flux measure assemblies are described as cells in input file. All surfaces of the cells are assigned in the surface cards of input file to make sure that the cells can describe each part of the reactor. Each cell is filled with the relevant material, and it describes this material in material card according to actual component. Each cell is assigned an importance of neutron (IMP:N). Because critical system calculation does not consider the sample of particle basically, the importance of cells of reactor: N=1, and others: N=0, means that once the

neutrons enter into that cell, they will be killed. In order to describe the complex structure of the reactor, it is needed to refer to the universe cards and repeat structure cards.

Figure 1 shows that the reactor includes many same fuel modules; in Fig.2, each fuel assembly includes 201 identical fuel rods and 20 identical control rod holes and one neutron flux measure hole. These cells and surfaces which have strong repetition can be described only once. The repeated structures serve the function of automatically filling a reactor.

3.3. The influence of reactor parameters on the value of K_{eff}

3.3.1. The effects of the distribution of fuel module locations and the controlling rod module on the value of K_{eff}

According to the arrangement in Fig.1 (the fuel module borders on controlling rod module), the value of breeding effective coefficient K_{eff} is calculated, and the parameters of Source Points for KCODE Calculation, which is called KSRC card, and Criticality Source information Card, which is called KCODE card, are as follows.

The number of neutrons for each recycle source (NSRCK) is 5000; the value of original K_{eff} (RKK) is 1; the circulation number 10 should be skipped over before the marking number; the total circulation number (KCT) of the calculation is 110; the original source point which the KSRC card appointed is the reactor core midpoint, namely (0, 0, 0) point. Then, the value of K_{eff} for another kind arrangement of reactor core can be calculated, and the specific distribution is shown in Fig.4. The parameters of KSRC card and KCODE card are the same as the previous ones, and the enrichment degree of U-235 is 2%. The calculation results are shown in Table 1.

Table 1. The effect of fuel module distribution in reactor on the value of K_{eff}.

arrangement	number of fuel stick	K_{eff}	statistical error
1	113	0.99342	0.00084
2	105	1.02653	0.00092

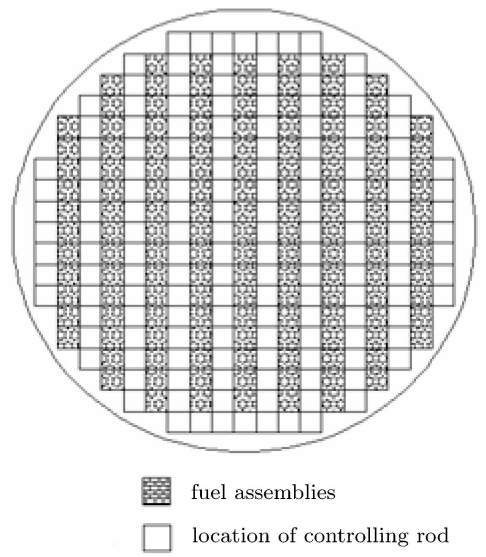


Fig.4. Schematic of reactor core structure.

It follows from Table 1 that although the number of fuel modules is less than that in the first arrangement, the value of K_{eff} is bigger than the first one in the calculation of second arrangement. The first arrangement makes each fuel assembly located around the location installing the control rods, but this location is replaced by water. It is obvious that the absorbability of moderator light water to neutron is higher than the neutron resonance absorption with the nonfission nucleus in moderate process.

3.3.2. The effect of aluminium material on the value of K_{eff}

Fuel rods contain Al_2O_3 adiabatic slice and ceramic fuel UO_2. The states are simulated, with or without the aluminium contained in the fuel rods and with the other parameter values kept the same as above.

Table 2. The effect of aluminium material on the value of K_{eff}.

aluminium state	fuel assembly number	K_{eff}	statistical errors
with aluminium	113	1.01052	0.00094
without aluminium	113	0.99342	0.00084

The data in Table 2 show that the value of K_{eff} is larger when the fuel rods contain aluminium than when they do not contain the aluminium, which is perhaps due to the fact that the aluminium can push out part of water from the reactor containing aluminium, and the aluminium absorption cross section to neutrons is smaller than that of light water.

3.3.3. The effect of KSRC card and KCODE card parameters on the value of K_{eff}

The differences in the value of K_{eff} caused by the circular neutron number (NSRCK), the circular number of overleap before counting and the total circular number of counting (KCT) have been analysed in Ref.[4]. The different settings of KSRC card and KCODE card parameters can also influence the value of K_{eff}. Three different solutions of source at the values of initial source point parameters appointed in the KSRC cards are discussed in this paper and shown in Table 3.

Table 3. The solutions of source at the values of initial point parameters.

solution	the position of initial source point
1	centre of core
2	the centre of fuel assembly above the x-axis
3	the centre of overall fuel assembly

Without aluminium contained in the fuel rods and with the remaining parameters kept the same values as those in subsection 3.3.1, the value of K_{eff} are calculated and listed in Table 4.

Table 4. The values of K_{eff} for different initial source points.

solution	fuel assembly number	K_{eff}	statistical errors
1	113	0.99342	0.00084
2	113	0.98685	0.00091
3	113	0.98657	0.00086

As can be seen from this table, the value of K_{eff} is bigger when initial source is located at the centre of core (solution 1) than when the initial source is not located at the centre of core (solutions 2 and 3), and the values of K_{eff} calculated in the other two solutions (solution 2 and 3) are almost the same. Thermal neutrons from the peripheric moderator outside the fuel may be absorbed by the outer covering of fuel container. Probably, that is the reason why the value of K_{eff} calculated in solution 1 is bigger than those calculated in solutions 2 and 3.

3.3.4. The effect of fuel rod compositions on the value of K_{eff}

The fuel rods are mainly composed of core fuel material UO_2. We calculate the value of K_{eff} without aluminium contained in the fuel rods and with the fissile fuel (U-235) having degrees of enrichment of 2.4%, 2.7% and 3.0% separately. The initial sources are allocated at the centre of core. The obtained results are listed in Table 5.

Table 5. The effects of the degree of enrichment of U-235 on the value of K_{eff}.

U-235 enrichment	fuel assemble number	K_{eff}	statistical errors
2.4%	113	1.04283	0.00092
2.7%	113	1.07121	0.00090
3.0%	113	1.09755	0.00095

As can be seen from Table 5, the value of K_{eff} increases gradually with the increase of U-235 concentration, which means that the sum of U-235 nonfission absorption ability and U-238 absorption ability is much less than U-235 fission absorption ability.

4. Conclusion

The variation in the value of K_{eff} with the values of core parameters is studied by changing the input file in this paper. The input file is the most important file in the MCNP4C code, in which the variation in the value of K_{eff} can be analysed. The setting of parameters in input file is vital to the whole model under the assumption of these data. The cause of variation in the value of K_{eff} is analysed, which can serve as the theoretical basis for the reactor design.

References

[1] Li S, Xie Z S, Deng L and Sun S H 2003 *Chin. J. Nucl. Sci. Engng.* **23** 188 (in Chinese)

[2] Xu S Y 1996 *The Applications of Monte Carlo Methods in Experimental Nuclear Physics* (Beijing: Atomic Energy Publishing Company) p267 (in Chinese)

[3] Zhong Z P, Shi G and Hu Y M 2003 *Nucl. Power Engng.* **24** 8 (in Chinese)

[4] Jing X Q, Yang Y W and Xu Y L 2005 *Nucl. Power Engng.* **26** 28 (in Chinese)

[5] Qiu X P, Li X C and Wang J H 2005 *J. Nanhua University* (Science and Technology) **19** 42 (in Chinese)

[6] Liu T X and Cao X R 2006 *Appl. Sci. Technol.* **33** 68 (in Chinese)

[7] Hao F H, Hu G C, Liu S P, Gong J, Xiang Y C, Huang R L, Shi X M and Wu J 2005 *Acta Phys. Sin.* **54** 3523 (in Chinese)