A DETERMINISTIC MODEL FOR CRITICALITY AND ROD WORTH ANALYSIS OF THE DNRR RESEARCH REACTOR WITH HEU FUEL

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Abstract: This paper presents the development and verification of a deterministic model using the SRAC code system for analysing the criticality and control rod worth of the Dalat Nuclear Research Reactor (DNRR). Numerical calculations were performed based on seven core configurations established experimentally during the startup period of the DNRR. Comparison of the results with MCNP5 calculations and with the measurement data was conducted. The impact of various nuclear data libraries (ENDF/B and JENDL) on the criticality and control rod worth analysis was also investigated.

Keywords: *DNRR, SRAC, criticality, control rod worth*

1. INTRODUCTION

Various simulation codes are available to solve the neutron transport equation for predicting the kinetics and neutronics characteristics of reactor cores [1] base on two classification methods: stochastic and deterministic methods. A number of Monte Carlo simulation codes have been developed such as MCNP, KENO, MCBEND, Serpent, TRIPOLI [2-5]. The advantage of Monte Carlo codes is the possibility to simulate complex geometries of reactor cores with less simplification. However, even in Monte Carlo simulations, possible modifications are usually made if the neutronic characteristics of the core are not affected significantly. On the other hand, deterministic codes take an advantage in computational time. Satisfactory agreement between Monte Carlo and deterministic codes was found in many neutronics analysis results [6-8].

The Dalat nuclear research reactor (DNRR) is a 500 kW pool-type research reactor upgraded from a 250 kW TRIGA Mark II reactor in the 1980s. The main structures of the TRIGA Mark II were maintained, except the active core. The TRIGA fuel elements were converted to VVR-M2 fuel assemblies. The new core is a cylinder with 44.2 cm in diameter and 60 cm in length consisting of 121 hexagonal cells of fuel bundles, control rods, irradiation channels, beryllium blocks, and aluminum chocks. The control system of the DNRR consists of seven control rods: two Safety rod (SR), four Shim rods (ShR) and one Automatic Regulating rod (AR).

With the aim of implementing reliable simulation models for analyzing the neutronics and thermal hydraulics, several deterministic and Monte Carlo models were developed using a number of computational codes such as MCNP, TRIPOLI, WIMSD/CITATION, APOLO [9-11]. In a previous work, taking into account the advantage of the deterministic method, an analysis model of the DNRR was developed using the SRAC code system [12]. Calculations performed based on the core configuration loaded with 88 highly enriched uranium (HEU) indicated a good agreement in comparison with the results obtained from MCNP5 calculations and the experimental values. In this work, the model is extensively benchmarked based on the criticality and rod worth analysis of the seven core configurations of the DNRR established during the startup period. Comparison with the results obtained from MCNP5 calculations and the experimental data has also been conducted. The impact of several evaluated nuclear data libraries (ENDF/B, JENDL) on the neutronics characteristics of the DNRR was also investigated.

2. NEUTRONICS MODELING

Neutronics model of the DNRR was developed using the SRAC code system [13]. The SRAC code system developed by Japan Atomic Energy Agency consists of three transport and two diffusion codes for neutronics calculations. The PIJ code based on the collision probability method was used for unit cell calculations. Macroscopic cross sections of fuel bundles, neutron trap, control rods and many other non-fuel lattice cells were then prepared for full core analysis using the CITATION module. Figure 1 shows the calculation flow chart of SRAC calculation model.

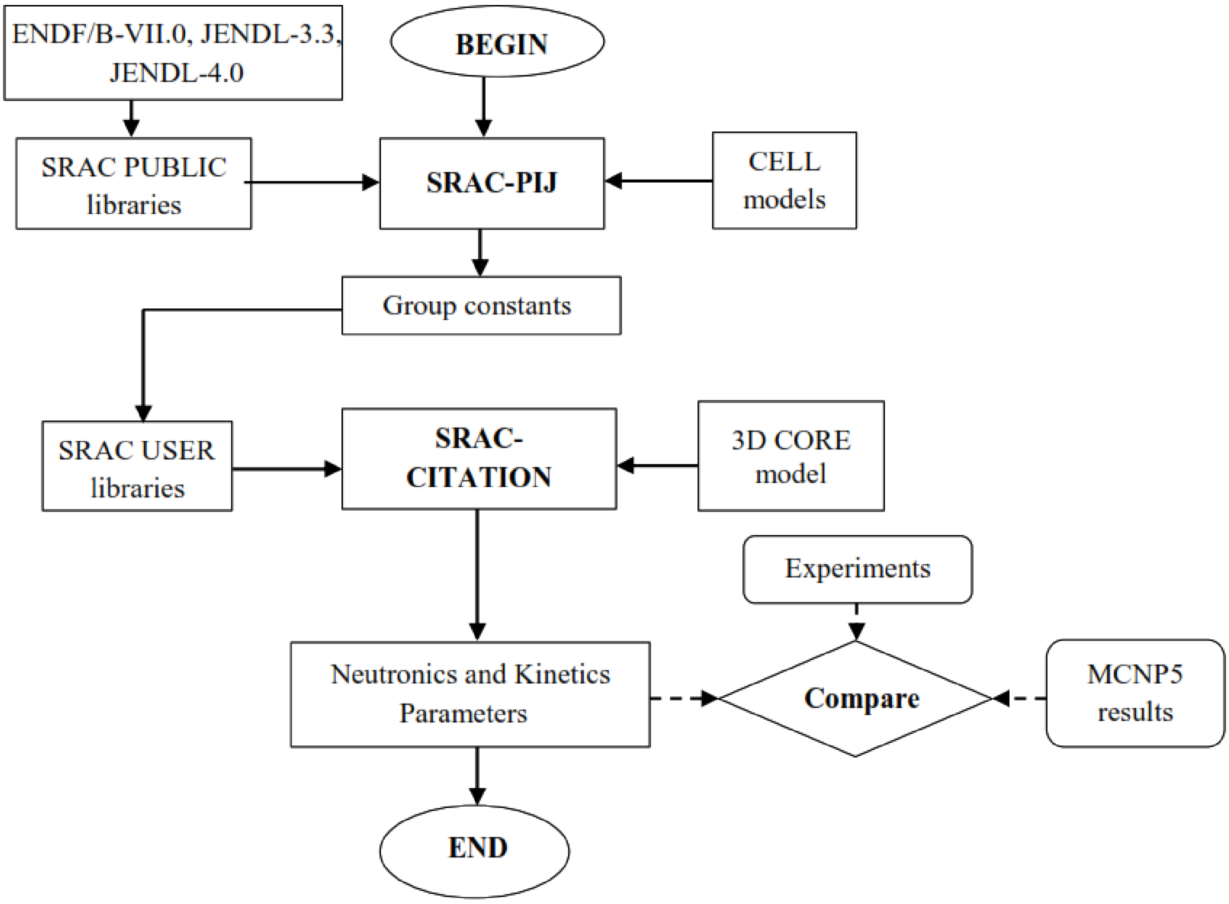

Fig. 1. Calculation flow chart of the DNRR

Since the PIJ code covers only 16 lattice geometries, some simplification and homogenization were carried out for fuel and non-fuel lattice cells of the DNRR. However, homogenization can affect to neutronics characteristics of control rods (SR and ShR) because of their strong absorption to thermal neutrons.

In a three-dimensional core model using the CITATION module, hexagonal grids of 37x37 in the X-Y coordinate plane were used to describe the core and graphite reflector. The total height of the model was 184.5 cm divided into a number of layers so that each axial layer in a cell at a horizontal position has the same material. The active core is 60 cm divided into 60 layers with the mesh size of about 1 cm but there could be slightly changed depending on the position of control rods. For the case of control rods (ShR) where the diffusion theory cannot be applied, the internal black absorber may be introduced. The extrapolated boundary constants corresponding to each neutron energy should be taken into account.

The energy structure consisting of 107 groups was collapsed into a seven energy-group cross-section set. The fast energy range (71 fine groups) was collapsed into 4 groups, and the thermal energy (36 fine groups) was collapsed into 3 groups. The energy boundary of each group is shown in Table 1.

Table 1. Seven energy group structure of cross-section data set in SRAC code.

Group number	Energy (eV)		Flux type
	Upper	Lower	
1	10^{7}	8.2085×10^{5}	Fast
2	8.2085×10^{5}	7.5308×10^{3}	
3	7.5308×10^{3}	3.9279	Epithermal
4	3.9279	6.0236×10^{-1}	
5	6.0236×10^{-1}	2.5683×10^{-1}	Thermal
6	2.5683×10^{-1}	5.4520×10^{-2}	
7	5.4520×10^{-2}	10^{-5}	

3. RESULTS AND DISCUSSION

Criticality and control rod worth analysis was performed based on seven core configurations established during the startup period of the DNRR. Fig. 2 shows the seven core configurations of the DNRR loaded with 69, 72, 74, 75, 83, 86 and 88 HEU fuel bundles, respectively. Core 7 loaded with 88 HEU fuel bundles was chosen to perform the analysis of the worth of the automatic regulating rod (AR) in comparison with the measurement data.

3. 1. Criticality analysis

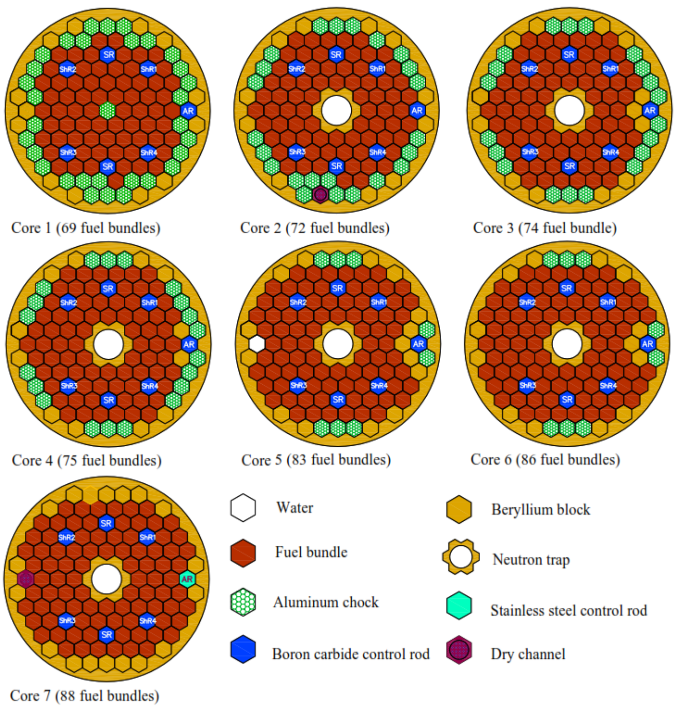

Fig. 2. Criticality configurations of the DNRR with HEU fuel.

Criticality calculations were performed based on 49 criticality conditions of the seven core configurations of the DNRR with three nuclear data libraries: ENDF/B-VII.0, JENDL-3.3 and JENDL-4.0. Whereas, the MCNP5 calculations were only performed with the ENDF/B-VII.0 library for comparison. Fig. 3 shows the k_{eff} values of the 49 core conditions in comparison with the MCNP5 calculations and the experimental data.

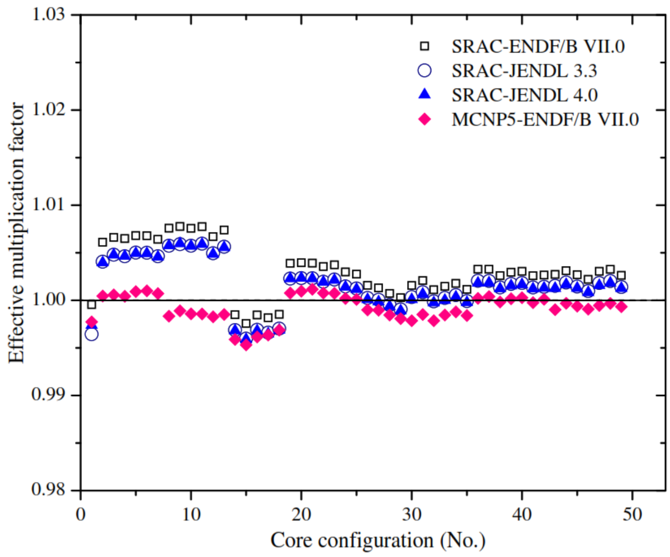

Fig. 3. The effective multiplication factor of criticality configurations obtained from the SRAC and MCNP5.

In the first criticality configuration with 69 HEU fuel bundles without neutron trap, the k_{eff} value obtained from SRAC calculations was underestimated of -45, -358 and -261 pcm with ENDF/B-VII.0, JENDL-3.3 and JENDL-4.0 libraries compared to the experiment, respectively. Whereas, MCNP5 calculation predicted with the discrepancy of -227 pcm with ENDF/B-VII.0 library. The criticality prediction from SRAC models for Cores 2, 3, 4 shows the discrepancy less than 770 pcm with the three nuclear data libraries. The largest discrepancy for Cores 5, 6 and 7 was about 405 pcm. For most of the calculated configurations, the criticality prediction using MCNP5 code was less than 471 pcm. It is found that the MCNP5 model showed a better agreement with the experiments than the SRAC model. Besides the errors of the data libraries and the codes, the specification of some structural components which are not available or provided accurately enough could contribute to the sources of discrepancy between the calculations and the experiments. The criticality analysis for 24 configurations of the working core has a good agreement with the experiment for both the SRAC and MCNP5 codes within 330 pcm.

In a little more detail, the simulation by MCNP5 model in this work was performed on a Windows server with 80 CPUs. The history number of 210 x 10^6 was chosen to guarantee the statistic error of k_{eff} below 0.006%. Each run took about 156 minutes when the SRAC model took about half a minute for the full core calculation on a personal computer with Intel Core i5-4460 of CPU 3.3 GHz. In the computational time view, the SRAC code is advantageous with the acceptable results compared to MCNP5 code.

3. 2. Reactivity worth of the AR rod

Table 2. The measurement of the AR rod worth depends on the insertion patterns of four shim rods.

Group	Shim rod insertion (cm)			
	ShR1	ShR2	ShR3	ShR4
1 (a, b, c)	65			65
2 (d, e, f)		65	65	

3 (g)	45	45	45	45
	45.7	45.7	45.7	45.7
	46.7	46.7	46.7	46.7

Measurements of the control rod reactivity worth were carried out by the asymptotic period method based on the doubling time measurement at the very low power to reduce the influence of temperature effect [14]. The measurement of the AR rod worth divided into three groups was performed with seven cases, denoted as (a), (b), (c), (d), (e), (f) and (g), corresponding to different insertion patterns of four shim rods (Table 2). In the first group, the insertion of the ShR1 and ShR4 was fixed at 65 cm while the ShR2 and ShR3 were adjusted to achieve criticality with the insertion of the AR respectively. The second group was measured with the position of ShR2 and ShR3 fixed at 65 cm, while the last group was the case with all four Shim rods were inserted uniformly in the core. Figures 4-6 show the integral reactivity of the AR rod in core 7 corresponding to these groups. The delayed neutron fraction of the DNRR loading HEU fuel is about 0.81% [15].

Fig 4 displays the comparison of the reactivity worth of the AR rod between the experiment and the calculations in group 1. The integral reactivity of the AR rod obtained from the calculations in this group is around 0.489 – 0.510 \$. The largest discrepancy between the calculations and the experiment is 17%. An agreement within 3% was found among the calculated results using the SRAC model with three nuclear data libraries.

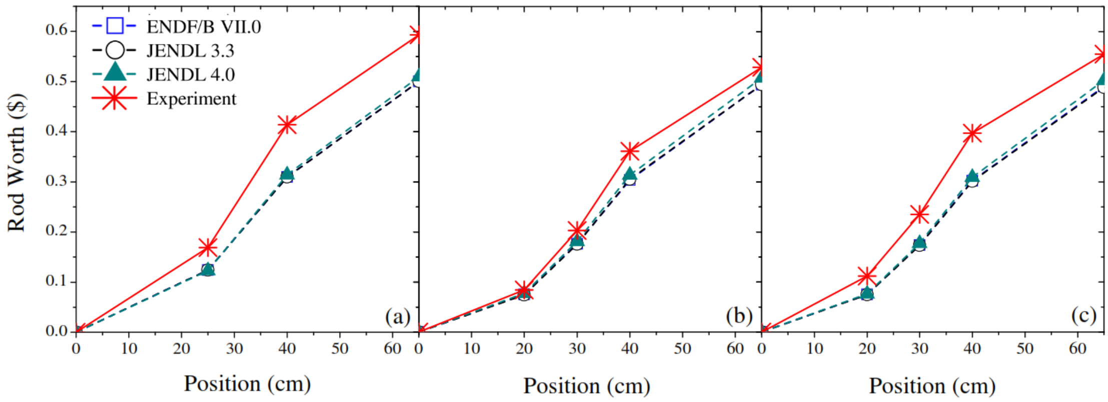

Fig. 4. The integral reactivity of the AR rod in group 1.

The reactivity worth of the AR rod obtained from group 2 was showed in figure 5. The deviation among the calculations with ENDF/B-VII.0, JENDL-3.3 and JENDL-4.0 libraries was found less than 4%.

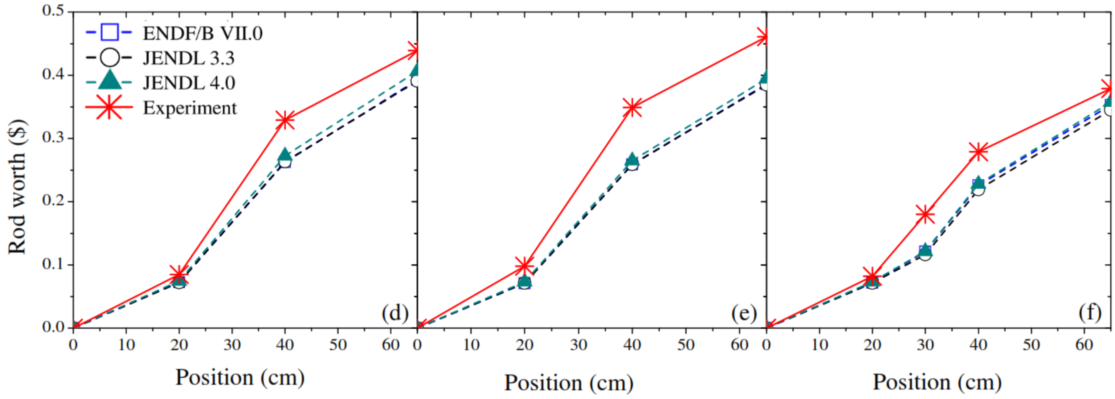

Fig. 5. The integral reactivity of the AR rod in group 2.

Fig. 6 displays the comparison of the reactivity worth of the AR rod between the calculations and the experiment of group 3. It can be seen that the calculation data has a good agreement with the experimental values with a deviation of 3% or 0.021$ respectively.

Besides the errors of the modeling, the analysis code and data libraries, the experiment of rod worth always has the inevitable errors caused by the control rod shadowing effects. As reported by Liem et al. in the measurements of the rod worth based on the method of power period of the MPR-30 reactor [16], the shadowing effect of the control rod caused by the change of the insertion pattern of other control rods could be reached up 19 % or 32 % for the effect between two control rods or among eight control rods, respectively. This may contribute to a larger discrepancy of groups 1, 2 compared to group 3.

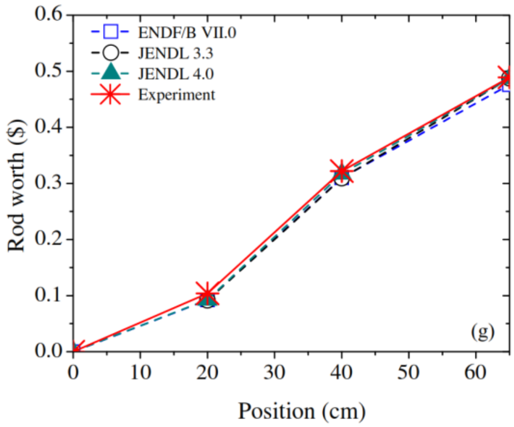

Fig. 6. The integral reactivity of the AR rod in group 3.

4. CONCLUSIONS

A deterministic model using SRAC code system was developed for criticality and control rod worth analysis of the DNRR reactor. Calculations were performed for 49 criticality conditions and control rod worth of the AR rod of the DNRR. The results show that criticality prediction of the 49 criticality conditions gives the discrepancy less than 770 pcm. For the configuration of 88 HEU fuel bundles, the k_{eff} values achieved a better agreement with experimental data with the deviation less than 330 pcm. The reactivity worth of the AR rod was predicted with the deviation less than 17% compared to the measurement. A good agreement in the AR rod worth obtained with the different data libraries was found with the discrepancy less than 4%. The results imply that the ENDF/B-VII.0, JENDL-3.3 and JENDL-4.0 nuclear data libraries are suitable for analyzing the criticality and rod worth of the DNRR with VVR-M2 HEU fuel.

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MÔ HÌNH TẤT ĐỊNH TRONG PHÂN TÍCH TỚI HẠN VÀ ĐẶC TRƯNG TÍCH PHÂN CỦA THANH ĐIỀU KHIỂN CỦA LÒ PHẢN ỨNG HẠT NHÂN ĐÀ LẠT SỬ DỤNG NHIÊN LIỆU HEU

Tóm tắt: Nghiên cứu này phát triển và kiểm chứng mô hình tất định sử dụng chương trình SRAC trong phân tích tới hạn và đặc trưng tích phân của thanh điều khiển của lò phản ứng hạt nhân Đà Lạt. Các tính toán được thực hiện trên bảy cấu hình vùng hoạt được thiết lập bằng thực nghiệm trong quá trình khởi động lò Đà Lạt. Kết quả tính toán được so sánh với số liệu từ thực nghiệm và tính toán MCNP5. Cùng với đó, ảnh hưởng của các thư viện số liệu hạt nhân (ENDF/B-VII.0, JENDL) cũng được phân tích.

Từ khóa: *SRAC, phân tích tới hạn, tích phân thanh điều khiển*