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Modeling of the Dalat Nuclear Research Reactor (DNRR) with the Serpent 2 Monte Carlo code

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Abstract: This paper presents a model development of the Dalat Nuclear Research Reactor (DNRR) loaded with low enriched uranium (LEU) fuel using the Serpent 2 Monte Carlo code. The purpose is to prepare the DNRR Serpent 2 model for performing fuel burnup calculations of the DNRR as well as for generating multi-group neutron cross sections to be further used in the kinetics calculations of the DNRR with a 3D reactor kinetics code. The DNRR Serpent 2 model was verified through comparing with the MCNP6 criticality calculations under different reactor conditions. The parameters to be compared include the effective neutron multiplication factor, radial and axial power distributions, and thermal neutron flux distributions. The comparative results generally show a good agreement between Serpent 2 and MCNP6 and thus indicate that the DNRR Serpent 2 model can be used for further calculations of the DNRR.

Keywords: DNRR, neutronics, Monte Carlo simulation, Serpent 2, MCNP6.

I. INTRODUCTION

The 250 kW TRIGA Mark-II reactor, a pool-type multipurpose research reactor designed and manufactured by General Atomics [1], was installed in the Dalat Nuclear Research Institute, Vietnam in early 1960s. It was upgraded to 500 kW in early 1980s and named the Dalat Nuclear Research Reactor (DNRR). In this upgrade, the main structures of the TRIGA Mark-II reactor were remained whereas the reactor core was loaded with the Russian VVR-M2 fuel type and cooled by natural convection. The first criticality of the DNRR was achieved on November 1st, 1983 and the operation with full power was achieved in March 1984. During the period from 1984 to

2007, the core was loaded with highly enriched uranium (HEU) fuel with ²³⁵U enrichment of 36%. In the framework of the program on Russian Research Reactor Fuel Return (RRFR) and the program on Reduced Enrichment for Research and Test Reactor (RERTR), the DNRR core was partly converted to low enriched uranium (LEU) fuel with ²³⁵U enrichment of 19.75% in September 2007 and the full core conversion to LEU fuel was performed during the period from November 24th, 2011 to January 13th, 2012 [2]. Presently, the operation of DNRR is mostly for radioisotope production, neutron activation analyses, education and researches. The DNRR is operated mainly in a continuous run of about 100-130 hrs in a period of 3-4

weeks and then the reactor is shut down for maintenance activities during which experimental physics researches can also be conducted [3].

The DNRR, i.e., a TRIGA Mark-II reactor loaded with the Russian VVR-M2 fuel, consists of various components such as a neutron trap, water gaps, irradiation channels, horizontal beam ports, beryllium blocks, control rods, rotary specimen rack and graphite reflector, which make the reactor geometry become complicated and difficult to be simulated with computer codes. In certain applications such as in the field of in-core fuel management, deterministic codes like WIMSD/CITATION and SRAC can be used for modeling and analysis of the DNRR [4][5][6]. Nevertheless, Monte Carlo codes like MCNP [5][6][7], Serpent 2 [8] are often preferred to simulating complicated reactor geometries like the DNRR. It is because an arbitrary reactor geometry can be only simulated without any approximation by using a Monte Carlo code, which also allows using continuous neutron energy cross sections, eliminating the need for energy groups collapsing as required with the deterministic ones. In particular, spatial homogenization and fuel burnup calculation can also be performed with the Serpent 2 Monte Carlo code.

In the present study, a model of the DNRR loaded with LEU fuel was developed and analyzed using the Serpent 2 Monte Carlo code. The goal is to prepare the DNRR model with Serpent 2 for performing fuel burnup calculations of the DNRR and for generating multi-group neutron cross sections to be further used in the kinetics analysis of the DNRR with a 3D reactor kinetics code. The DNRR core loaded with 92 LEU fuel bundles and its surrounding structures were simulated with high detail using Serpent 2. The DNRR Serpent 2 model was then verified through comparing with the MCNP6 criticality calculations under different reactor conditions.

The effective neutron multiplication factor, k_{eff} , was calculated and analyzed using Serpent 2 and MCNP6 in the cases corresponding to the complete withdrawal and the full insertion of the four shim rods and the automatic regulating rod as well as under the criticality condition of the core. In addition, power and thermal neutron flux distributions were also examined throughout the radial core and along several typical axial channels.

II. CALCULATION METHODOLOGY

The horizontal cross section view and main specifications of the DNRR are shown in Fig. 1 and Table I, respectively [2]. The reactor core consists of hexagonal cells including fuel bundles, control rods, in-core irradiation channels, beryllium blocks, and a neutron trap. The fuel bundle is the Russian VVR-M2 type made of $\text{UO}_2\text{-Al}$ dispersion clad in aluminum. The total ^{235}U mass in each LEU fuel bundle is about 49.7 g distributed in three coaxial fuel tubes as illustrated in Fig. 2. The outer tube has a hexagonal shape and the two inner tubes are cylindrical. The core is controlled by seven control rods: two safety rods, four shim rods and one automatic regulating rod. The safety and shim rods are made of boron carbide (B_4C), while the automatic regulating rod is made of stainless steel. The absorption length of each control rod is 65 cm, ensuring to completely cover the 60 cm long active core. The dry and wet channels are covered by aluminum cylinders with the thickness of 0.5 mm. The neutron trap located at the core center is a water cylinder with 6.5 cm in diameter and 60 cm in length surrounded by the beryllium blocks. The beryllium blocks have the same outer shape and dimension with the fuel bundle. At several peripheral cells, if no fuel bundle is loaded, the beryllium blocks are loaded for providing supplementary neutron reflection. A ring of serrated beryllium blocks is also located between the active core and the graphite reflector to act as an additional

reflector. This beryllium ring and the core are placed in a cylindrical aluminum shell, which is the lower section of the supporting structure. The thickness of the graphite reflector is 30.5 cm. The core and the graphite reflector are placed in the reactor pool.

In this work, the DNRR configuration with 92 LEU fuel bundles (Fig. 2) was simulated using the Serpent 2 Monte Carlo code [8]. The DNRR model with Serpent 2 was then verified through comparing with the MCNP6 criticality calculations under different reactor conditions including the critical condition and the cases of complete withdrawal and full insertion of the four shim rods and the regulating rod. The radial layout of the DNRR core modeled in MCNP6 and Serpent 2 was shown in Fig. 3. The ENDF/B-VII.0 data library [9] was used in both the Serpent 2 and MCNP6 calculations. The parameters to be calculated and compared include the k_{eff} ; radial power distributions; axial power distributions in cell 4-5 and cell 7-10; and axial thermal neutron flux distributions in cell 4-5, cell 7-10 and neutron trap.

The DNRR model with MCNP6 was developed similarly to that reported with MCNP5 in Ref. [5]. It is noted that the DNRR model with MCNP had been well validated against experiments and other calculation results obtained with different codes [2][5]. By using MCNP, the complex geometry of the DNRR including the fuel bundles, control rods, in-core irradiation channels, beryllium rods, horizontal beam tubes, graphite reflector, rotary specimen rack, thermal column and thermalizing column was simulated with high accuracy and detail. This calculation model covers from the active core to reactor tank following the radial direction with 198.72 cm in diameter and the axial direction with 187 cm in height. Tallies 4 and 7 were used for calculating the neutron flux and power distributions, respectively. Then a DNRR model was developed with Serpent 2 using the same input data for the reactor geometry and materials with the MCNP6 model as described below.

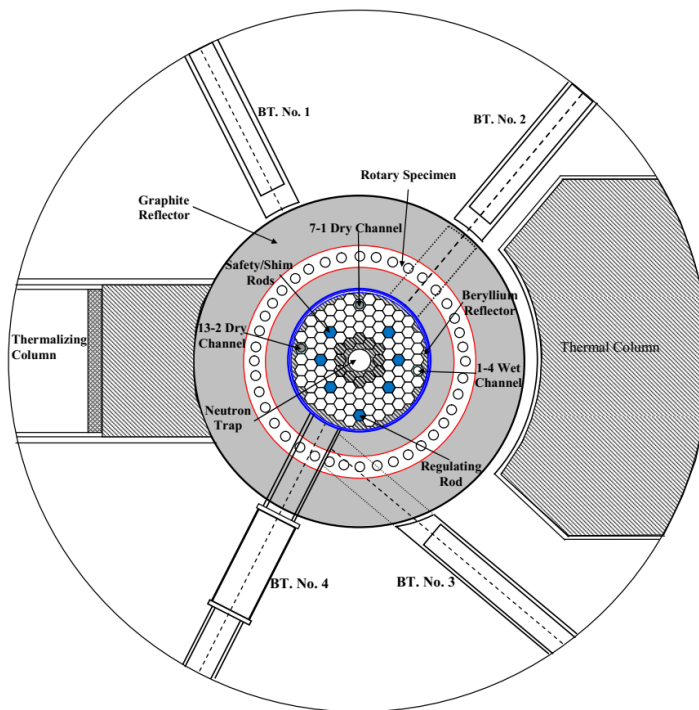


Fig. 1. Horizontal cross section view of the DNRR.

Serpent is a three-dimensional continuous-energy Monte Carlo reactor physics burnup calculation code developed at the VTT Technical Research Centre of Finland [8]. Similar to MCNP, the code allows easily modeling of complicated reactor geometries for criticality calculations, fuel cycle studies, etc. Furthermore, it has also various powerful capabilities such as automated burnup sequence for spatial homogenization, coupled multi-physics calculations, transient simulations, sensitivity calculations, reactor geometry pre-implementation and fast running time.

Therefore, Serpent 2 has been widely used in the calculations of nuclear reactors in general and of TRIGA reactors in particular [10][11]. To simulate the DNRR geometry with high detail in the Serpent 2 model, we adopted all the data that were input to the DNRR MCNP6 model as above mentioned. The only simplification is that the rounded corners of the outer hexagonal tubes of the VVR-M2 fuel bundles were not modeled in the Serpent 2 simulation. It is worth noting that the delta tracking method was used in the Serpent 2 calculations as compared to the track length method used in the MCNP6 calculations.

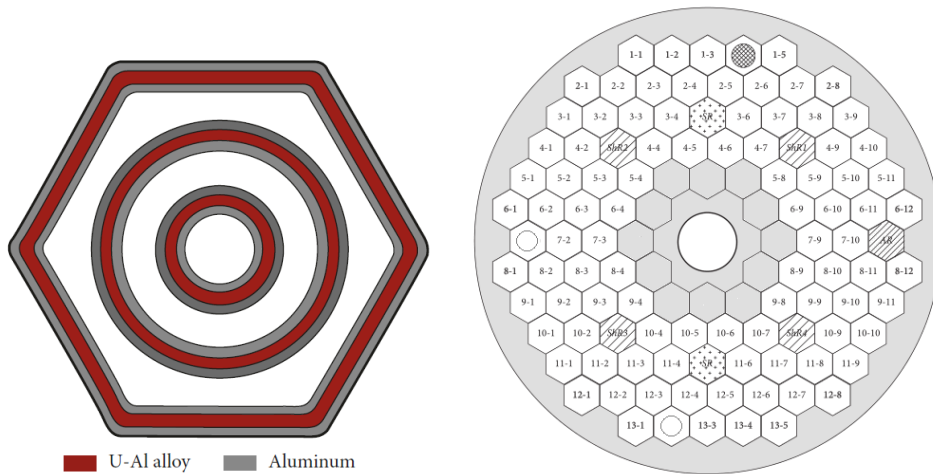


Fig. 2. Radial layout of the VVR-M2 fuel bundle (left) and configuration of the DNRR core with 92 LEU bundles (right). SR: safety rod, ShR: shim rod, and AR: automatic regulating rod.

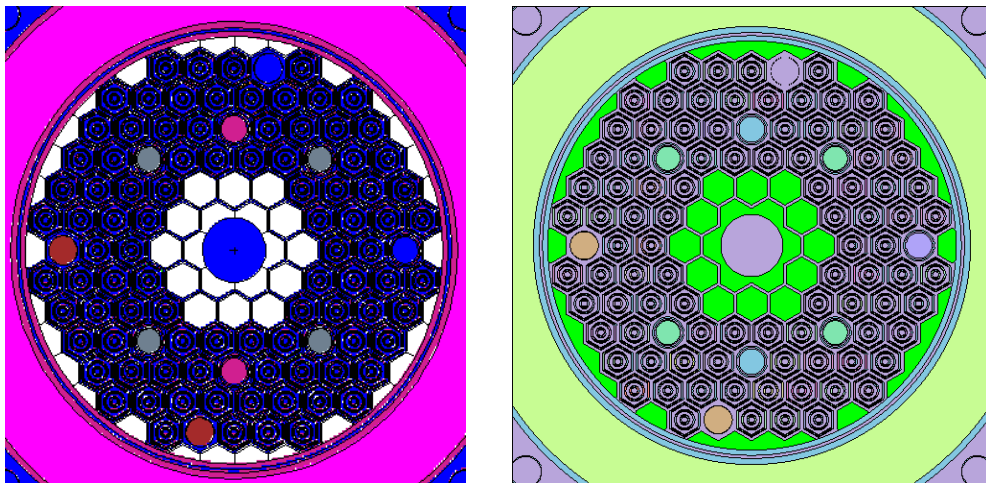


Fig. 3. The DNRR core with 92 LEU bundles modeled in MCNP6 (left) and Serpent 2 (right).

Table I. Main specifications of the DNRR.

Reactor type	Pool type
Nominal thermal power	500 kW
Coolant and moderator	Light water
Core cooling mechanism	Natural convection
Reflector	Graphite, beryllium and light water
Active core height	60 cm
Core equivalent diameter	44.2 cm
Fuel pitch	3.5 cm
Fuel type	VVR-M2 type, dispersed UO ₂ -Al with 19.75% enrichment, aluminium cladding
Number of control rods	7 (2 safety rods, 4 shim rods, 1 regulating rod)
Material of safety and shim rods	B ₄ C
Material of automatic regulating rod	Stainless steel
Vertical irradiation channels	4 (1 neutron trap, 1 wet channel, 2 dry channels)
Horizontal beam ports	4 (1 tangential, 3 penetrant)

III. RESULTS AND DISCUSSION

A. Effective neutron multiplication factor

The k_{eff} values obtained by Serpent 2 and MCNP6 in the cases of (a) criticality condition (four shim rods were inserted 40.5 cm in the core and the regulating rod was inserted 40 cm in the core), (b) complete withdrawal of the four shim rods and the regulating rod from the core, and (c) full insertion of the four shim rods and the regulating rod in the core were shown in Table II. In all the calculations, the two safety rods were assumed completely withdrawn from the core as they are used only for emergency shutdown. In this investigation, the four shim rods were inserted in the core and then the

position of the regulating rod was adjusted correspondingly to search for the criticality of the core. It can be seen that there was a good agreement within 71 pcm in the k_{eff} values calculated using Serpent 2 and MCNP6. Also, the statistic errors of the k_{eff} values in the Serpent 2 and MCNP6 calculations were almost identical to each other with the same number of neutron history used. The difference in the k_{eff} values of at most 71 pcm might be mainly related to the two reasons as follows: (1) the rounded corners of the outer hexagonal tubes of the VVR-M2 fuel bundles were not included in the Serpent 2 model, and (2) different neutron tracking methods used in the Serpent 2 (delta tracking method) and MCNP6 (track length method) calculations.

Table II. Comparison of k_{eff} calculated using Serpent 2 and MCNP6.

Position of control rods (cm)		k_{eff}		
Four shim rods	Regulating rod	Serpent 2	MCNP6	Deviation (pcm)

Full insertion	Full insertion	0.97103 ± 0.00007	0.97140 ± 0.00006	-37
Complete withdrawal	Complete withdrawal	1.07864 ± 0.00007	1.07793 ± 0.00005	71
40.5	40	1.00073 ± 0.00007	1.00018 ± 0.00006	55

B. Power and thermal neutron flux distributions

In this Section, power and thermal neutron flux distributions were calculated and analyzed using Serpent 2 in comparison with the MCNP6 calculations. The two cases including the criticality of the core and complete withdrawal of the shim rods and the regulating rod as shown in Table II were considered. Fig. 4 displays the relative radial power distributions calculated by Serpent 2 and MCNP6. It can be seen that there was a good agreement between the two codes. The maximum difference in the

radial power distributions calculated by Serpent 2 and MCNP6 in the cases was 6.2% (critical condition) and -5.7% (complete withdrawal of control rods), which appeared at the fuel bundles located in the periphery of the core. The maximum value of the relative radial power distribution was found at cell 4-5 in both cases. This value was 1.424 under the critical condition and 1.374 in case of complete withdrawal of control rods. Hence, the insertion of control rods under the critical condition increased the maximum value of the radial power distribution ~3.6%.

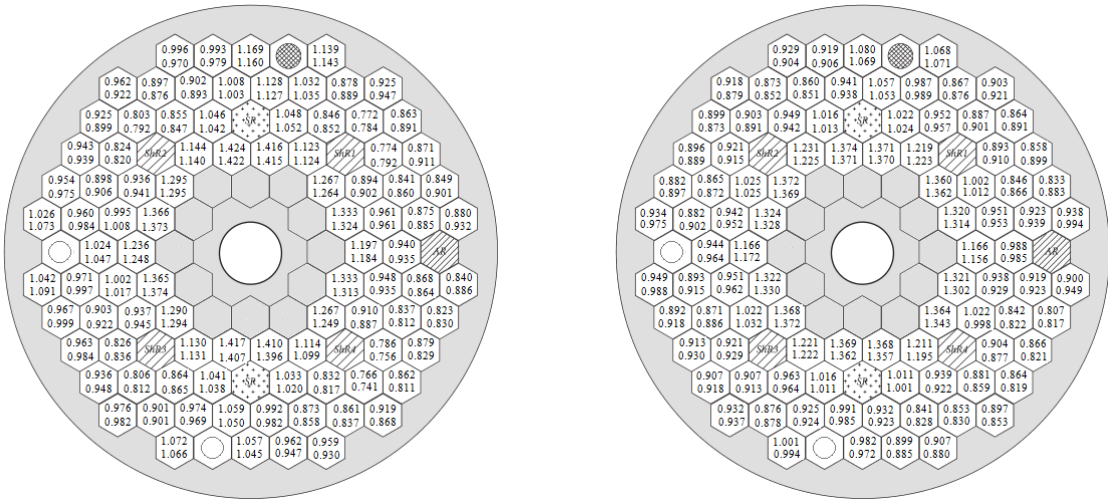


Fig. 4. Relative radial power distribution in the core under the critical condition (left) and in the case of complete withdrawal of control rods (right). The upper and lower values were obtained with Serpent 2 and MCNP6, respectively.

The axial power and thermal neutron flux distributions in cell 4-5 and cell 7-10 were shown in Figs. 5-6. Cell 4-5 is the position at which the relative radial power was highest and cell 7-10 is the position next to the regulating rod, which can show clearly the effect of control rod insertion. Fig. 5 shows a good agreement between the axial power distributions obtained with Serpent 2 and

MCNP6 in these two cells. The respective maximum difference between the codes was 1.5% (cell 4-5) and 4.9% (cell 7-10). Similarly, the axial thermal neutron flux distributions in cell 4-5 and cell 7-10 calculated by Serpent 2 compared well with those calculated by MCNP6 as shown in Fig. 6. The respective maximum difference between the codes in this case was 3.3% (cell 4-5) and -4.2% (cell

7-10). Additionally, the axial thermal neutron flux distribution in the neutron trap calculated by Serpent 2 agreed well within 2.8% with the MCNP6 calculations as shown

in Fig. 7. Also, the maximum thermal neutron flux in the neutron trap was found consistent with that reported in the Safety Analysis Report [2].

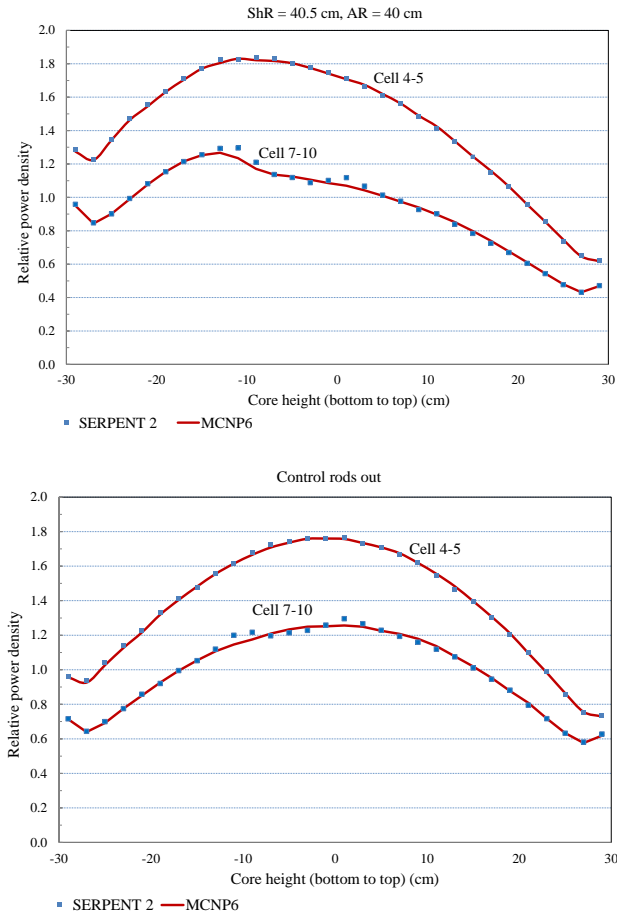
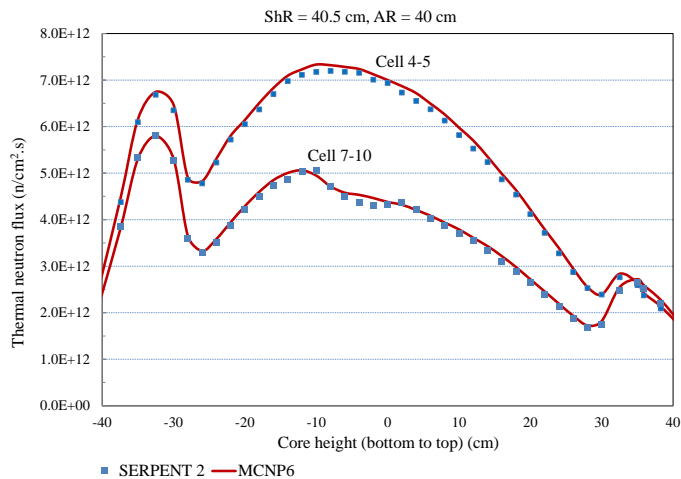


Fig. 5. Axial power distribution along the fuel bundles 4-5 and 7-10 under the critical core condition (left) and in the case of complete withdrawal of control rods (right).



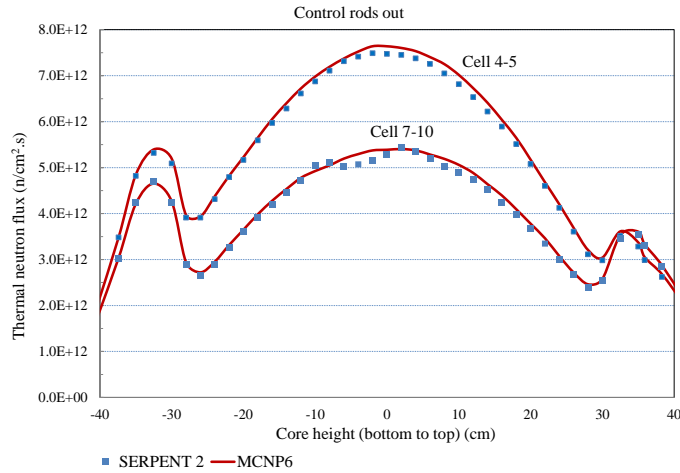


Fig. 6. Thermal neutron flux distribution along the fuel bundles 4-5 and 7-10 under the critical core condition (left) and in the case of complete withdrawal of control rods (right).

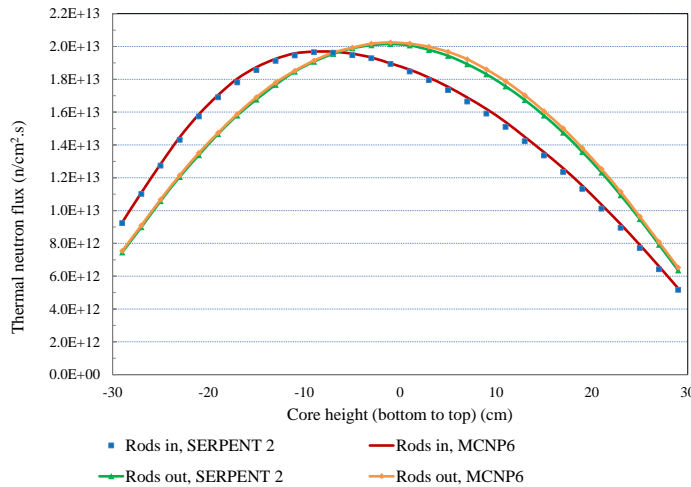


Fig. 7. Thermal neutron flux distribution along the neutron trap under the critical core condition (Rods in) and in the case of complete withdrawal of control rods (Rods out).

IV. CONCLUSIONS

In this study, the Serpent 2 Monte Carlo code was applied to model the DNRR. The DNRR Serpent 2 model was verified through comparing with the MCNP6 criticality calculations under different reactor conditions. These conditions include the critical condition of the core, full insertion of the four shim rods and the regulating rod in the core, and complete withdrawal of the four shim rods and the regulating rod from the core. It is found that there was a good agreement within 71 pcm

in the k_{eff} values obtained with Serpent 2 and MCNP6. In addition, the comparison of power and thermal neutron flux distributions in the core generally shows a good agreement between the two codes.

Consequently, the results indicate that the DNRR model developed herein with Serpent 2 is reliable and can be used for further analyses of the DNRR. The DNRR Serpent 2 model is now being used for generating multi-group neutron cross sections to be used in the kinetics analysis of the DNRR with a 3D

reactor kinetics code. Also, this model with Serpent 2 is being planned for performing fuel burnup calculations of the DNRR.

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