PRELIMINARY STUDY ON MODELING THE DALAT NUCLEAR RESEARCH REACTOR AND GENERATING THE MULTI-GROUP CROSS-SECTION FOR THREE DIMENSIONAL REACTOR KINETICS CALCULATIONS

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Abstract: Nowadays nuclear kinetic codes coupled with thermal-hydraulics codes are necessary in analyzing transients/accidents scenarios to ensure the safety of the reactor. The 3D kinetic code PARCS is used for the DNRR using homogenized macroscopic cross-section to be prepared by other physics lattice codes such as MCNP, SCALE, SERPENT. In this paper, the Da Lat Nuclear Research Reactor (DNRR) core has been modeled by SCALE/TRITON code to generate two-group homogenized cross-sections for 3D kinetics calculations. In the calculation, plate-type model has been applied in self-shielding structure while the fuel assemblies have been grouped for the cross-section generation. The preliminary calculation results not only include the cross-sections for in-core region but also for the reflective graphite region for kinetic calculations.

Keywords: SCALE, TRITON, homogenized cross-section, DNRR.

NGHIÊN CỨU SƠ BỘ VỀ MÔ HÌNH HÓA VÙNG HOẠT LÒ ĐÀ LẠT VÀ PHÁT DỮ LIỆU TIẾT DIỆN BẰNG CHƯƠNG TRÌNH SCALE/TRITON SỬ DỤNG CHO TÍNH TOÁN ĐỘNG HỌC LÒ PHẢN ỨNG BA CHIỀU.

Tóm tắt:

Ngày nay việc sử dụng các chương trình tính toán động học kết hợp với các chương trình tính toán thủy nhiệt trở nên cần thiết đối với các bài toán phân tích trạng thái chuyển tiếp nhằm đảm bảo sự an toàn của vùng hoạt lò phản ứng. Chương trình tính toán động học lò phản ứng PARCS sử dụng các dữ liệu tiết diện vĩ mô đồng nhất được tính toán bởi các chương trình tính toán lưới mạng như MCNP, SCALE, SERPENT. Trong báo cáo này, lò phản ứng hạt nhân Đà Lạt được mô hình hóa bời chương trình SCALE/TRITON nhằm đưa ra các tiết diện hai nhóm sử dụng cho tính toán động học. Trong tính toán này, một mô hình bản phẳng được sử dụng trong tính toán tự che chắn đối với bó nhiên liệu, các bó nhiên liệu được phân loại theo đặc trưng cho các dữ liệu tiết diện đồng nhất được tính toán. Các kết quả sơ bộ trong báo cáo này được tính toán cho cả bên ngoài vùng hoạt đến lớp vành phản xạ graphit của lò phản ứng Đà Lạt.

Từ khóa: SCALE, TRITON, homogenized cross-section, DNRR.

1. INTRODUCTION

Nowadays, the coupled calculations between neutronics and thermal-hydraulics codes become popular in nuclear reactor calculations to consider the thermal-hydraulics feedbacks [1-2]. Along with the development of the computer and the computational tools, the kinetic codes are also taking into account in coupling calculation with the system codes to analyze transient or accident conditions as well as incidents [3-5] which can occur in a nuclear reactor.

Among the nuclear reactor kinetics calculation codes, the PARCS code which was developed by Purdue University, USA and used by USNRC in analyzing the transient states in the reactor, based on nodal method and diffusion theory for the three-dimensional multi-group kinetics calculation is one of the most popular tools in nuclear reactor safety analysis. The PARCS code, which has the ability to couple with thermal hydraulics code like TRACE, RELAP5 are widely used in analyzing the transients and accidents in both the nuclear power reactors [3-5] and nuclear research reactors [6-7]. In PARCS code, the homogenized macroscopic cross-sections for the kinetic calculations are provided by other lattice codes such as SCALE/TRITON, SERPENT or MCNP, etc. The TRITON code in the SCALE code system, which is a powerful tool for lattice physics calculations [8] is normally used in generating cross-section for PARCS by its accuracy in lattice physics calculation and the capability to couple with PARCS [9]. The DNRR, which based on TRIGA Mark II reactor, is the unique reactor of Vietnam until present. So far, kinetic calculations for the DNRR have been performed using RELAP5 code with point kinetic model [10-12], while the 3D kinetic calculations for the DNRR to compare with the experimental results have been still under studying. This research on homogenized cross-section generating will be used as the first step for the 3D kinetic model calculation using PARCS for analyzing both the steady and the transient conditions for the DNRR. In this paper, the DNRR fuel assemblies and reflector structures are modeled by SCALE/TRITON. The fuel assemblies are grouped based on their positions to acquire the average cross-section of each group, the structure materials are also grouped according to their material compositions. Due to the limitation of self-shielding structure in TRITON for the DNRR fuel assembly geometry, a plate-type model [7] which has been applied for a research on VR-1 reactor will be used. Then, the homogenized macroscopic cross-sections of the fuel assemblies and other structure cells like beryllium at neutron trap, auxiliary beryllium block reflector around the reactor core and graphite reflector are generated.

2. THE DALAT NUCLEAR RESEARCH REACTOR MODEL IN SCALE/TRITON

In this research, the Low Enriched Uranium (LEU) core of the DNRR is considered. The fuel assembly of the DNRR is VVR-M2 fuel type, consist 3 UO_2-Al alloy fuel elements with 0.94 mm of fuel meat thickness and two cladding layers with the thickness of one layer 0.78 mm, as shown in Fig 1.

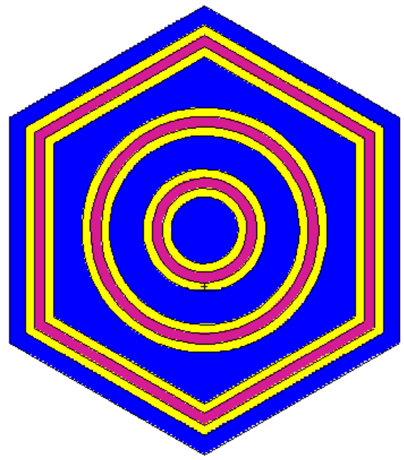


Fig 1: The DNRR fuel assembly

The two inner fuel elements of the DNRR fuel assembly are cylinder shape while the outer fuel element has the hexagonal geometry. The active length of the fuel assembly is 600 mm. The material specification of the UO_2-Al fuel using in DNRR core is shown in Table 1.

Table 1: Material specification of LEU fuel of the DNRR

Nuclide	Density (Atom/barn/cm)
^{234}U	1.34219E-05
^{235}U	1.19978E-03
^{238}U	4.80027E-03
^{16}O	1.20269E-02
Al	4.16117E-02

The DNRR core has diameter of 44.2 cm and consists of 92 fuel assemblies with 19.75 wt% of ^{235}U. The neutron trap is located at the center of the reactor core, surrounded with 2 layers of Beryllium block, each block has the same size as the fuel assembly. There are 3 irradiation channels (1-4, 7-1 and 13-2), 4 shim rods, 2 safety rods and an automatic rod in the reactor core. The reflective graphite region around the reactor core has the inner and outer radius are 23.75 cm and 54.25 cm, respectively. The rotary specimen consists of 40 irradiation holes with the diameter of 31.75 cm, located at the reflective graphite region and it is also taken into account in the calculation model. For 3D kinetic calculation with PARCS code, a 3D homogenized cross-section must be prepared by using SCALE/TRITON. Due to the limitation of 2D model in TRITON, the 3D cross-section is prepared by using a sequence of 2D models at different axial layers of the DNRR. In this paper, a 2D model of DNRR is presented at the axial layer near the center of the reactor core, in which the rotary specimen is located at the graphite reflector region. The 2D DNRR core geometry and the graphite reflector region are shown in Fig 2, the details are given in the Safety Analysis Report (SAR) of the Da Lat Nuclear Research Reactor [13].

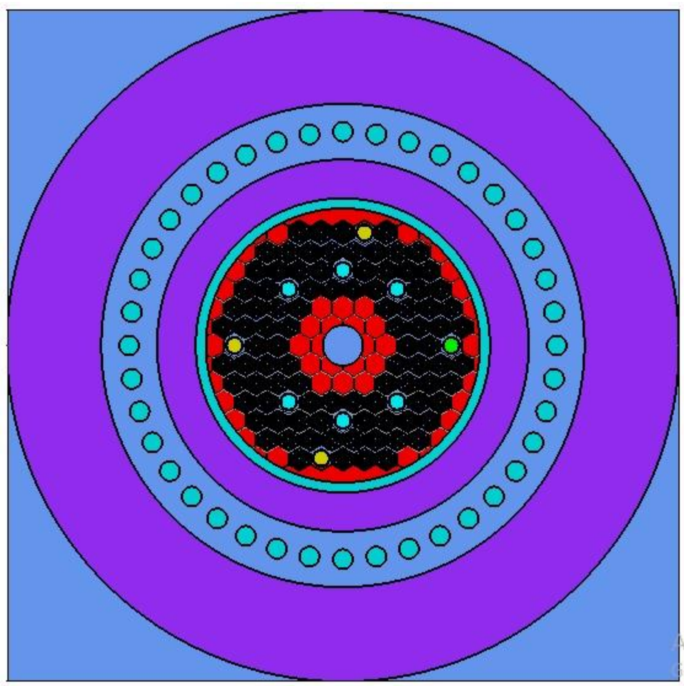


Fig 2: The DNRR core and reflective structure

Due to the limitation of the self-shielding structure with the geometry similar to the DNRR fuel assembly, the plate-type model with the conservation of the fuel and cladding thickness was applied when the half-pitch (HPITCH) was interpolated based on the results of SRAC2006 calculation for the actual model of the fuel assembly [7]. The infinite multiplication factor (k-inf) for different half-pitches of plate-type model are calculated by SRAC2006 and compared with the result of the 2D actual fuel assembly model to determine the half-pitch which can be used in self-shielding specification of SCALE/TRITON. The results of the infinite multiplication factor versus the change on the half-pitch of the plate-type model are shown in Table 2. Comparing with the k-inf value of actual model, k-inf = 1.63557, the half-pitch value for the self-shielding structure was chosen with the value of HPITCH=0.592 cm.

Table 2: k-inf versus half-pitch for the plate-type model calculated by SRAC

HPITCH	0.59	0.591	**0.592**	0.593	0.594	0.595	0.596	0.597	0.598	0.599
k-inf	1.63618	1.63594	**1.63569**	1.63543	1.6352	1.63494	1.63471	1.63446	1.63422	1.63397

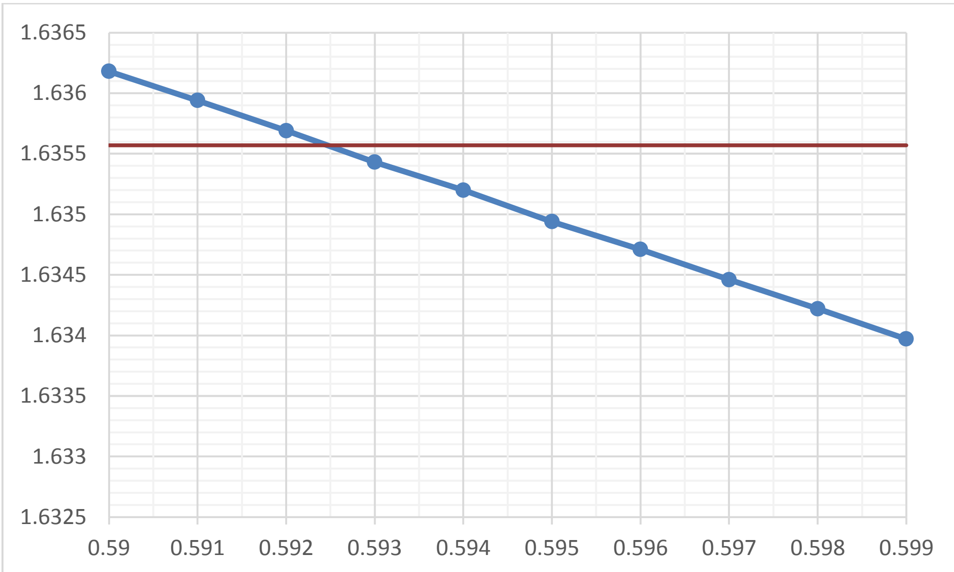


Fig 3: k-inf versus half-pitch for the plate-type model calculated by SRAC

In kinetic calculation within PARCS code, the homogenized macroscopic cross-sections of fuel assemblies to be prepared by TRITON code are used as the input parameters. To simplify the model in neutron kinetic calculations, 92 fuel assemblies in the DNRR reactor core were divided into 7 groups by their different positions in the reactor core [14]. In details, fuel assembly groups are surrounded by:

- Group 1: Surrounded by shim rods or safety rods, beryllium block and other fuel assemblies
- Group 2: Surrounded by beryllium block and other fuel assemblies
- Group 3: Surrounded by shim rods or safety rods, irradiation channel and other fuel assemblies
- Group 4: Surrounded by shim rods or safety rods and other fuel assemblies
- Group 5: Surrounded by automatic rod (AR) and other fuel assemblies
- Group 6: Surrounded by irradiation channel and other fuel assemblies

- Group 7: Surrounded by only other fuel assemblies

For preparing input file of PARCS code, the homogenized macroscopic cross-sections of other structures as non-fuel are also taken into account in SCALE/TRITON code including: Neutron trap, irradiation channels, shim rods and safety rods, automatic rod, beryllium blocks around the reactor core and graphite reflector. These homogenized cross-sections are calculated for both fast and thermal neutron energy groups.

3. PRELIMINARY CALCULATION RESULTS

As mentioned in the previous section, the DNRR has been modeled by using TRITON code in the SCALE code system. The extending of DNRR calculation model to the graphite reflector region is shown in Fig 4a while the details of the reactor core are shown in Fig 4b.

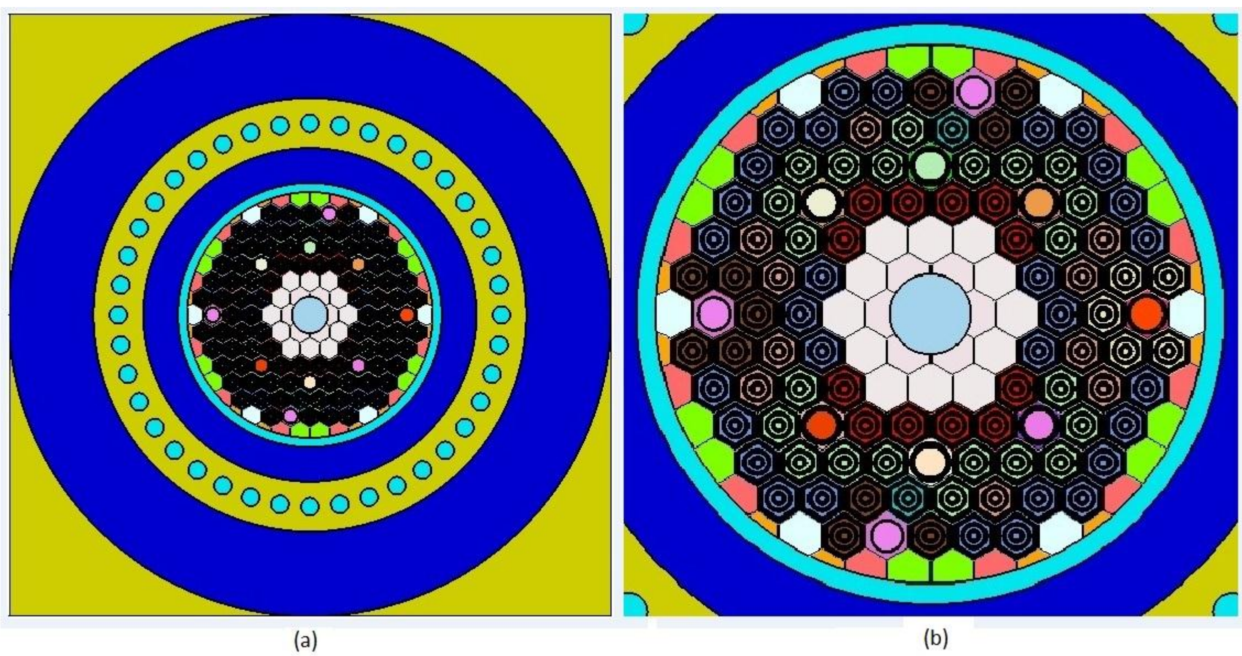
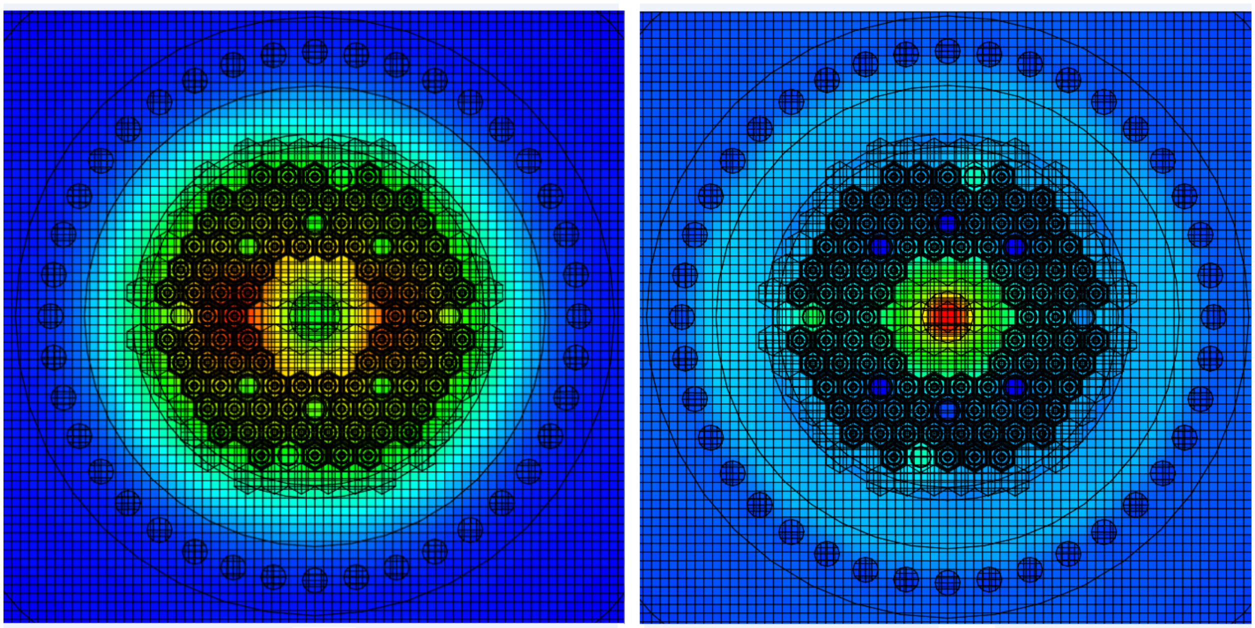


Fig 4: The DNRR model in SCALE/TRITON (a) and the core in details (b)

In calculation model, all the shim rods and safety rods are fully inserted in the reactor core and the effective multiplication factor (k-eff) calculated by SCALE/TRITON has value of k-eff = 0.95780. The fast and thermal neutron flux of the DNRR with LEU core are also calculated by SCALE/TRITON and shown in Fig 5. In Fig 5a, it can be seen that the fast neutron flux has the higher value in the fuel regions around the center beryllium trap, at the positions where the shim rods and safety rods are not presented. Because of slowing down ability of the beryllium in the center beryllium trap, the fast neutron flux inside the center hole of the reactor are significant reduced compared to the fast neutron flux in the outer beryllium layer of the center trap. The fast neutron flux outside the reactor core is reduced by the beryllium blocks located around the fuel assembly in the reactor core. The thermal neutron flux distribution is shown in Fig 5b. The highest value of the thermal neutron flux is located at the neutron trap of the core, where water hole is located and surrounded by beryllium blocks and beryllium rods. The thermal flux is significant higher than the average at the position of the irradiation channels and has the lowest value at the position of the shim rods and safety rods. Both the fast and thermal neutron fluxes have rather low value outside the reactor core in radial direction.



(a) (b)

Fig 5: Fast neutron flux (a) and thermal neutron flux (b) distribution in the DNRR

Table 3 presents the homogenized macroscopic cross-sections by using the SCALE/TRITON calculation. These cross-sections include transport, absorption, scattering cross-sections of fast and thermal energy as well as the fission cross-sections within two energy groups. For the kinetic calculation with PARCS code, the cross-sections are calculated for fuel assemblies and also the other structure regions inside the reactor core. However, in the paper, only one axial layer of the DNRR has been calculated, the remaining axial layers for generating homogenized cross-sections for 3D kinetic calculations by using TRITON are under carried out.

Table 3: The homogenized macroscopic cross-section generated by SCALE/TRITON for the DNRR

Cell regions	Homogenized macroscopic cross-section										
	Fast neutron energy group					Thermal neutron energy groups				Fission	
	Transport	Absorption	nu-fission	kappa-fission	down-scattering	Transport	Absorption	nu-fission	kappa-fission	Fast	Thermal
Fuel group 1	2.07E-01	5.00E-03	4.13E-03	5.20E-14	2.29E-02	1.03E+00	9.22E-02	1.64E-01	2.10E-12	1.67E-03	6.75E-02
Fuel group 2	2.07E-01	5.11E-03	4.22E-03	5.32E-14	2.39E-02	1.01E+00	8.99E-02	1.60E-01	2.05E-12	1.71E-03	6.58E-02
Fuel group 3	2.04E-01	5.04E-03	4.18E-03	5.26E-14	2.37E-02	1.01E+00	9.03E-02	1.61E-01	2.05E-12	1.69E-03	6.61E-02
Fuel group 4	2.04E-01	4.91E-03	4.10E-03	5.16E-14	2.27E-02	9.86E-01	8.76E-02	1.56E-01	1.99E-12	1.66E-03	6.40E-02
Fuel group 5	2.07E-01	5.17E-03	4.27E-03	5.38E-14	2.45E-02	9.91E-01	8.85E-02	1.58E-01	2.01E-12	1.73E-03	6.48E-02
Fuel group 6	2.05E-01	5.12E-03	4.23E-03	5.33E-14	2.42E-02	1.02E+00	9.16E-02	1.64E-01	2.09E-12	1.71E-03	6.71E-02
Fuel group 7	2.03E-01	4.94E-03	4.14E-03	5.21E-14	2.34E-02	9.90E-01	8.79E-02	1.57E-01	2.00E-12	1.67E-03	6.43E-02
Automatic rod	2.73E-01	2.09E-03	0.00E+00	0.00E+00	1.80E-02	1.17E+00	7.83E-02	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Irradiation channel	2.31E-01	3.50E-04	0.00E+00	0.00E+00	4.90E-02	1.62E+00	1.51E-02	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Outer neutron trap layer	3.89E-01	1.18E-03	0.00E+00	0.00E+00	6.48E-03	7.62E-01	1.35E-03	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Inner neutron trap layer	3.67E-01	7.60E-04	0.00E+00	0.00E+00	3.24E-02	1.39E+00	9.39E-03	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Shim and safety rods	2.41E-01	5.41E-02	0.00E+00	0.00E+00	1.38E-02	1.69E+00	4.19E-01	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Beryllium block and reactor core tank	2.88E-01	1.27E-03	0.00E+00	0.00E+00	5.41E-03	6.47E-01	5.23E-02	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Graphite Reflector region	2.56E-01	2.78E-05	0.00E+00	0.00E+00	2.69E-03	3.82E-01	2.40E-04	0.00E+00	0.00E+00	0.00E+00	0.00E+00

4. CONCLUDING REMARKS

In the paper, the calculation model and results of the DNRR for generating homogenized macroscopic cross-sections using SCALE/TRITON have been presented. For preparing cross-section data to the kinetic calculations, fuel assemblies inside reactor core of the DNRR are divided into 7 groups based on their position specifications. The other structure region cross-sections are also calculated for using in the kinetic code. Because of the limitation of self-shielding specification in SCALE with the geometry of the DNRR fuel assembly, plate-type model for self-shielding structure in SCALE/TRITON is used for analyzing the VR-1 reactor, is also applied for the DNRR fuel assembly in cross sections calculation.

The calculation results of SCALE/TRITON code include multiplication factor, neutron flux distribution and homogenized cross-sections that are calculated within two neutron energy group for transport, absorption, scattering and fission cross sections. These cross sections will be used for the kinetic calculations with PARCS code. However, for using in the 3D kinetic calculation, the homogenized cross-sections are need to be calculated for all axial layers of the DNRR and also need to be verified by experiment of calculation results from other code as MCNP5.

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