NEUTRONIC CHARACTERISTICS AT IRRADIATION POSITIONS OF THE DALAT NUCLEAR RESEARCH REACTOR AFTER REFUELING

Presented

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Content

Introduction

- After more than 9 years since the core conversion using 92 LEU FAs, the Dalat Nuclear Research Reactor (DNRR) has been operated safely and efficiently for over 18,000 hours.

- According to the reactor operational data, the total operating time of the reactor was about 383 MWD and the excess reactivity was decreased to be $2.98 until the end of January, 2021.

- In the limiting operation conditions of the reactor, the required excess reactivity value is about 2.8$ to compensate the reactivity causing by temperature effects, xenon poisoning and irradiation samples. Therefore, the refueling is necessary to increase the excess reactivity for safety operation of the reactor.

Dalat Research Reactor and its parameter

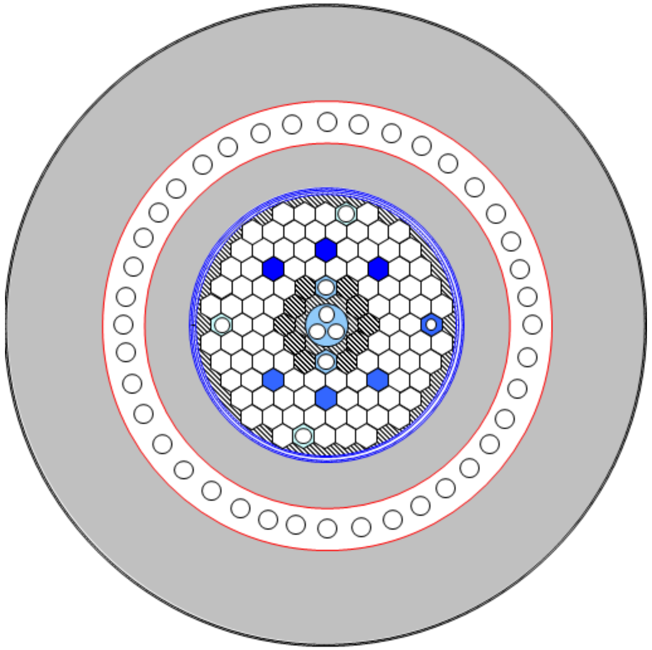
Summary description of the DNRR

Parameter	Description
Nominal power	500 kW
Neutron flux (thermal, max.)	2×10^{13} neutrons/cm^2.s
Fuel	VVR-M2 type, tube form
Fuel meat	Mixed UO_2-Al, 19,75% enrichment
Moderator	Light water
Reflector	Graphite, beryllium and water
Core cooling	Natural convection
Control rods	2 safety, 4 shim and 1 regulating

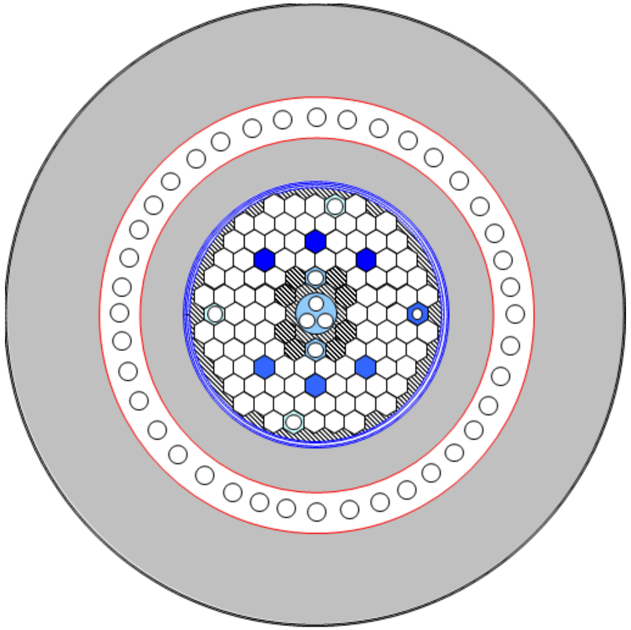
The refueling steps

The detail steps of fuel loading are following :

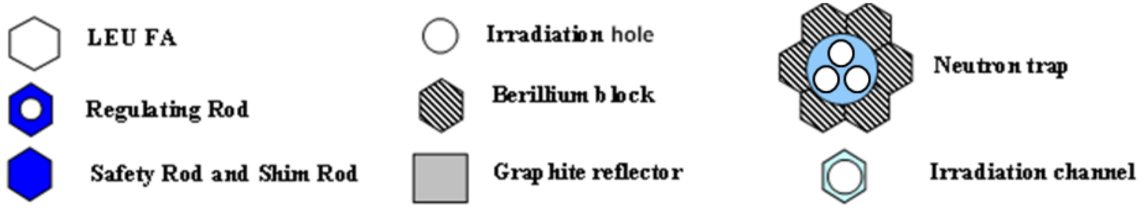
- Taking out two beryllium blocks at the positions of cell No.7-4 and cell No.7-8.
- Moving two used LEU FAs from cell No.4-5 to cell No.7-4 and from cell No.10-6 to cell No.7-8.
- Moving two used LEU FAs from cell No.7-3 to cell No.4-5 and from cell No.7-9 to cell No.10-6.
- Putting two new LEU FAs into cell No.7-3 and cell No.7-9.
- Completing the loading configuration of 94 LEU FAs



Core of 92 LEU FAs



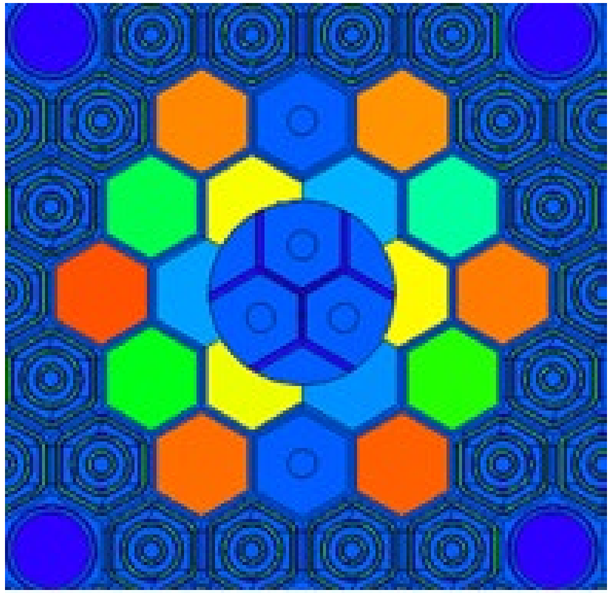
Core of 94 LEU FAs



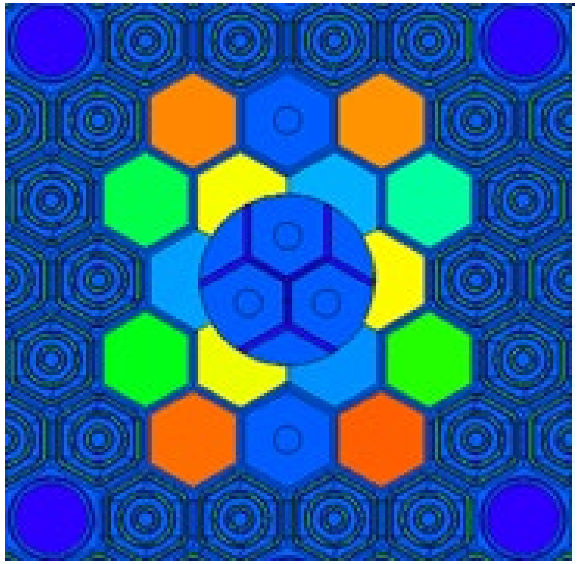
(cell No.X-Y means grid row number X – grid cell number Y)

Calculation and Experiment

The MCNP5 computer code was used for calculation of neutron flux distribution, neutron spectrum before and after refueling of the reactor. This computer code was developed by the Los Alamos National Laboratory, USA. MCNP5 has been being officially used for core management of DNRR with the ENDF/B-VII library .



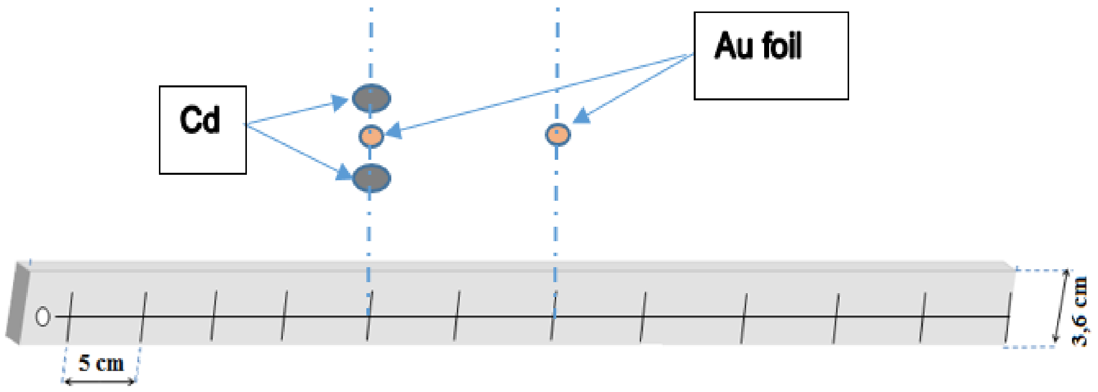
Neutron trap in core configuration of 92 LEU FAs



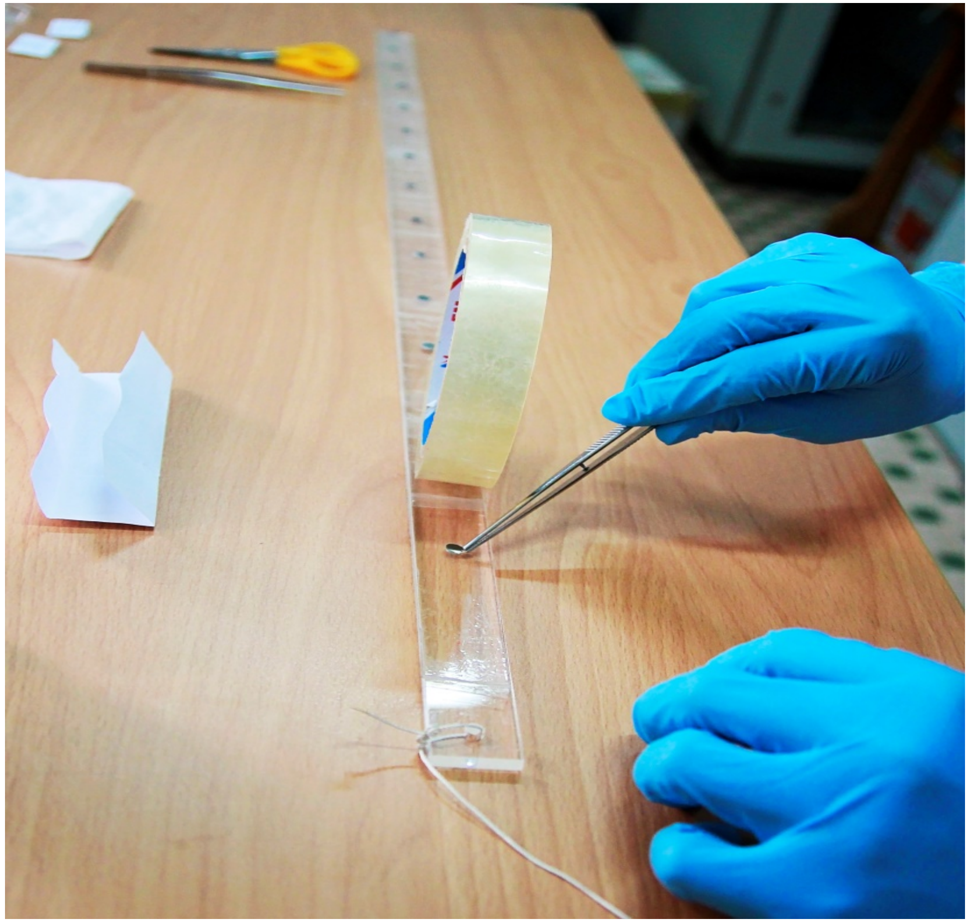
Neutron trap in core configuration of 94 LEU FAs

Calculation and Experiment

- Determination of the **relative distribution of thermal neutron flux**, Lutetium foils are used.
- For the **thermal neutron flux measurement**, two identical Au foils are used. One Au foil is covered with the Cd and the other Au foil is uncovered. The Cd-covered Au foil is fixed at least 10 cm away from the uncovered foil to avoid neutron flux depression.



Calculation and Experiment



Calculation and Experiment

To calculate the thermal neutron flux in the core, the following equations are used:

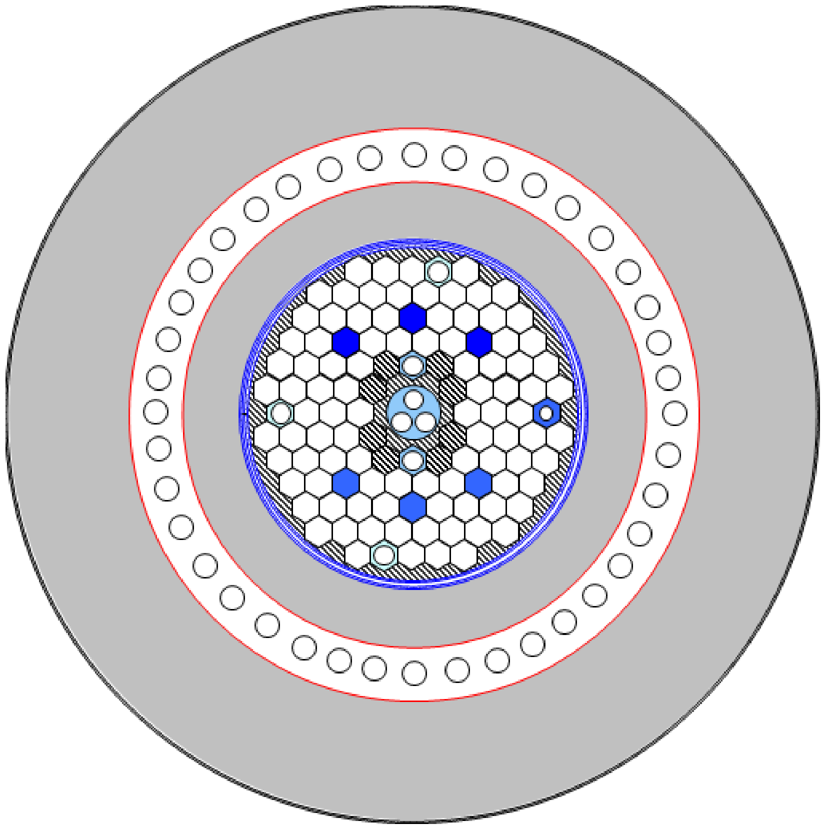
$$R_i = \frac{P_i}{\alpha . G_i . m . L / A_i . \varepsilon_{abs} . G_{th} . \gamma_i . t_{m,eff}} \times \frac{e^{\lambda_i t_c}}{(1 - e^{-\lambda_i t_{irr}})} \times \frac{\lambda_i t_m}{(1 - e^{-\lambda_i t_m})}$$

$$\Phi_{th} = \frac{\left(R^b - R^{Cd}\right)}{\frac{\sqrt{\pi}}{2} \sqrt{\frac{T_o}{T_n}} \sigma_0}$$

RESULTS AND DISCUSSIONS

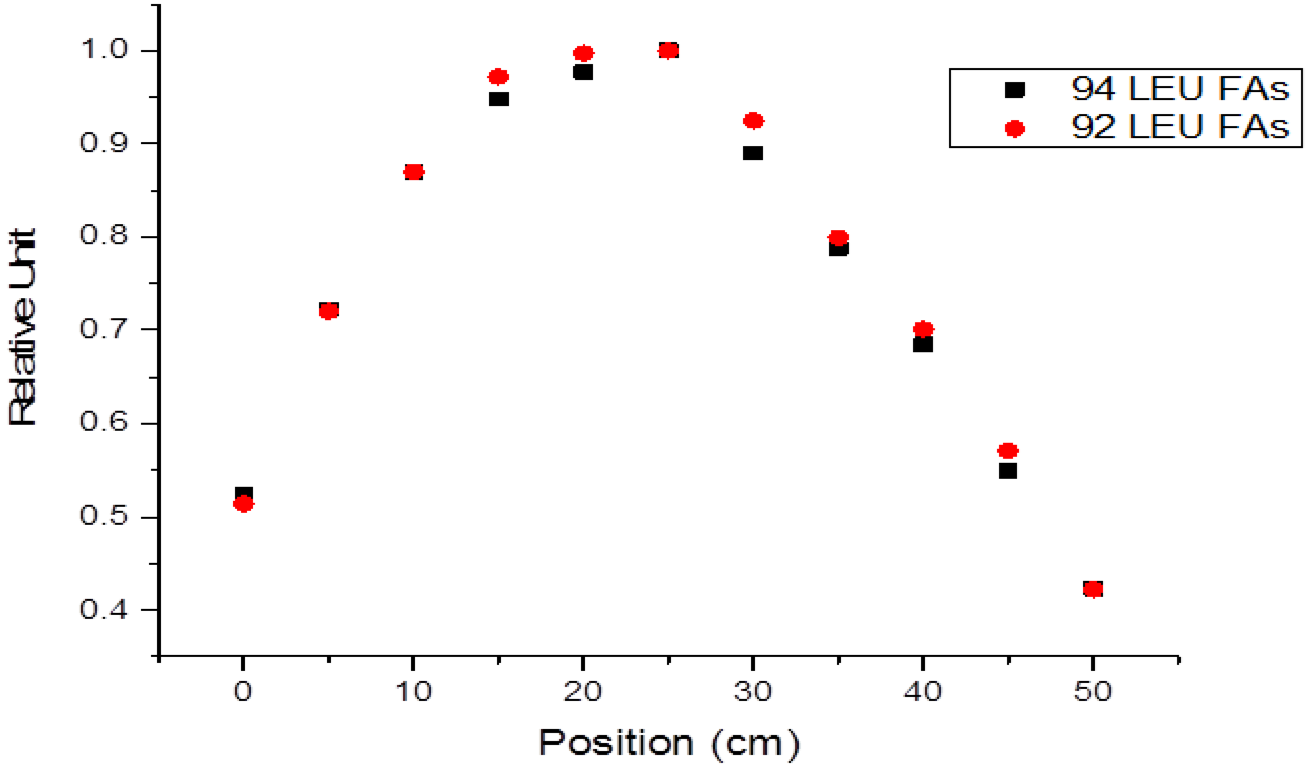
Comparison of thermal neutron flux measurement of the wet irradiation channels before and after refueling ($n/cm^2.sec$)

Position	Measurement Data (error ±7%)	
	92 LEU FAs	94 LEU FAs
3 inner holes of neutron trap	1.98×10^{13}	1.90×10^{13}
Irr. channel No.5-6	1.83×10^{13}	1.81×10^{13}
Irr. channel No.1-4	1.04×10^{13}	1.02×10^{13}



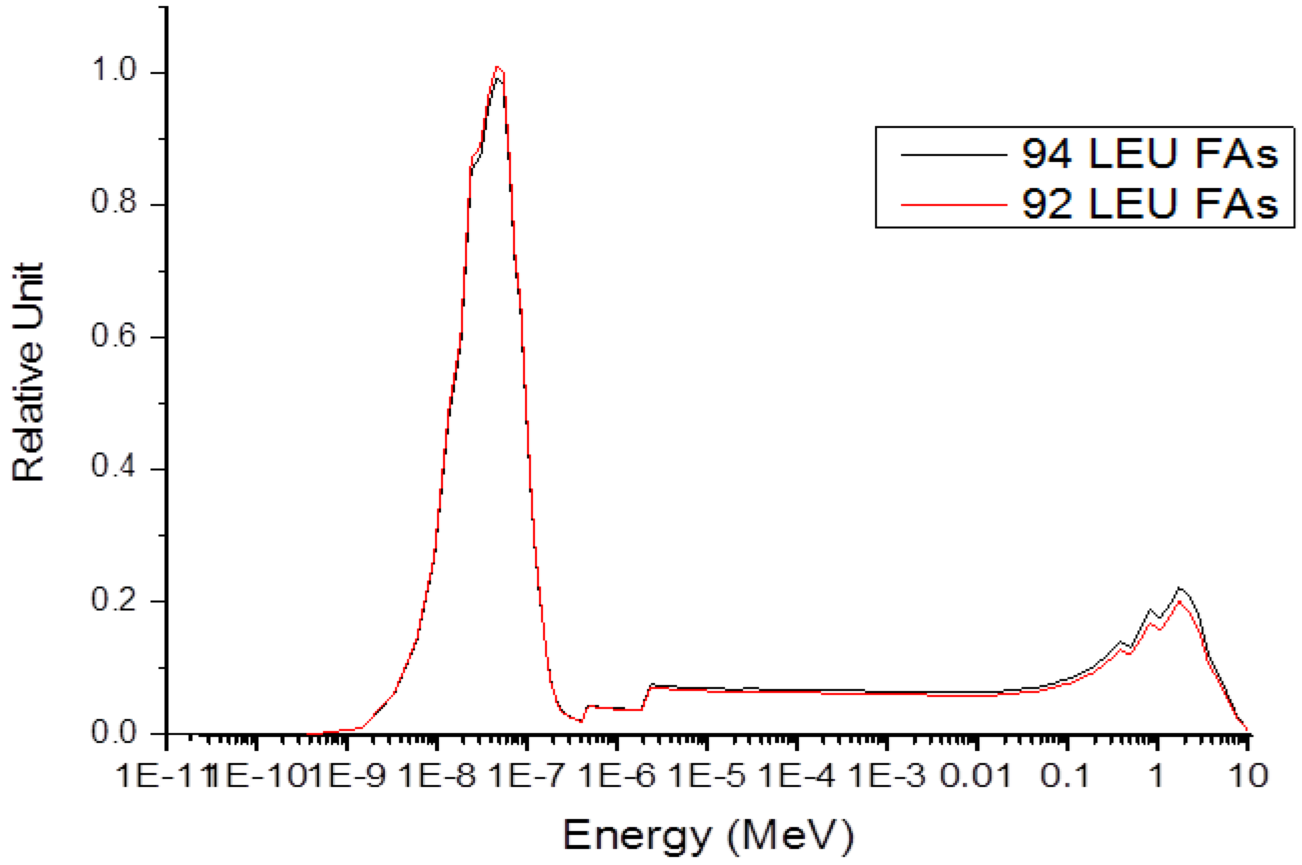
RESULTS AND DISCUSSIONS

Experimental data of thermal neutron flux distribution of the neutron trap before and after refueling



RESULTS AND DISCUSSIONS

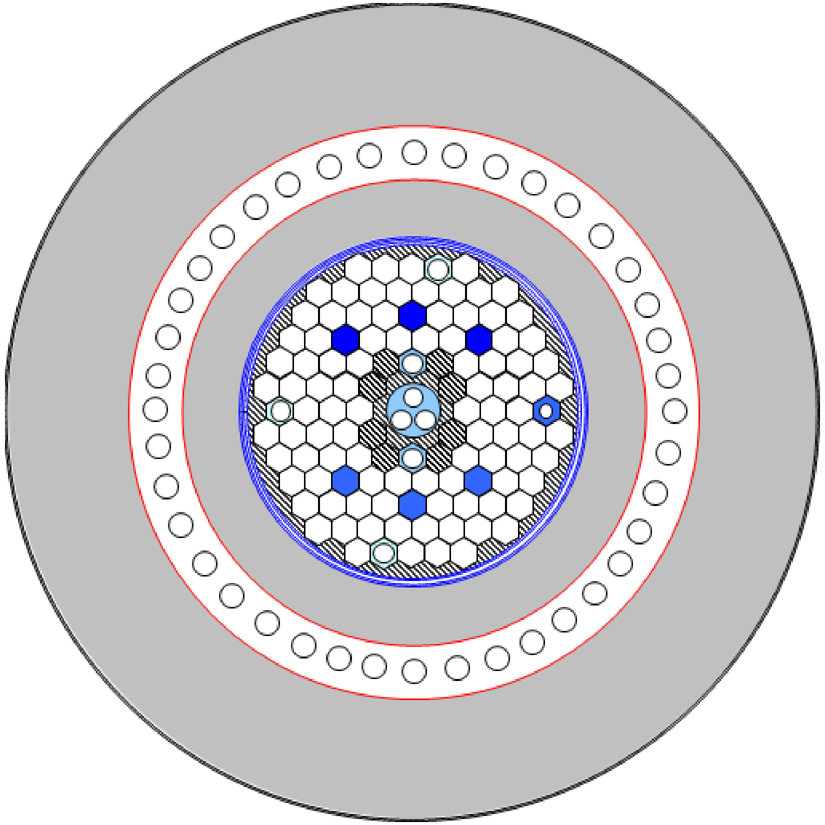
Neutron spectrum at the neutron trap before and after refueling



RESULTS AND DISCUSSIONS

The average thermal neutron flux in the irradiation samples at the irradiation channels

Position	Calculation of average thermal neutron flux (n/cm².sec)	
	92 LEU FAs	94 LEU FAs
3 inner holes of neutron trap	8.26×10^{12}	7.87×10^{12}
Irr. channel No.5-6	7.28×10^{12}	7.17×10^{12}
Irr. channel No.1-4	4.17×10^{12}	4.16×10^{12}



Conclusion

- From the experimental and computational results of neutron characteristics, it can be seen that the values of thermal neutron flux in the irradiation channels tend to be a bit decreased after refueling.

- The maximum thermal neutron flux value in the axial direction is found located at the same position in both 92 and 94 LEU FAs configurations.

- The average thermal flux in the irradiation channels during normal operation of the DNRR with full irradiation samples are also insignificantly changed after refueling.

- Thus, the continued effective exploitation of the reactor is ensured.

THANK YOU
FOR YOUR ATTENTION!