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REALIZATION OF THE CORE CONVERSION AND PRELIMINARY STUDY ON CORE CONFIGURATION WITH ONLY LEU FUEL ASSEMBLIES FOR THE DALAT NUCLEAR RESEARCH REACTOR

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ABSTRACT

The research on core conversion for Dalat Nuclear Research Reactor (DNRR) has been performed. Contracts for reactor core conversion between USA, Russia, Vietnam and the International Atomic Energy Agency for Nuclear fuel manufacture and supply for DNRR and Return of Russian-origin non-irradiated highly enriched uranium fuel to the Russian Federation have been realized. The 35 fresh HEU fuel assemblies (FA) were sent back to Russian Federation. The 36 new LEU FAs from Russian Federation have been received. According to the results of design and safety analyses performed by the joint study between RERTR Program at Argonne National Laboratory and Vietnam Atomic Energy Commission the mixed core of irradiated HEU and new LEU fuel assemblies has been created on 12 September, 2007. The preliminary study on new configuration with only LEU fuel assemblies for the DNRR has been performed. The codes MCNP, REBUS and VARI3D are used to calculate neutron flux performance in irradiation positions and kinetics parameters. This paper presents the results of reactor core conversion and preliminary study on new configuration with only LEU fuel assemblies for the DNRR.

1. Introduction

The DNRR is a light water moderated and cooled, pool-type reactor that is currently fueled with WWR-M2-type fuel assemblies. The reactor can be operated at a maximum power of 500 kW. The research on core conversion for DNRR has been performed. Contracts for reactor core conversion between USA, Russia, Vietnam and the International Atomic Energy Agency for Nuclear fuel manufacture and supply for DNRR and Return of Russian-origin non-irradiated highly enriched uranium fuel to the Russian Federation have been realized. The mixed core of irradiated HEU and new LEU fuel assemblies has been created on 12 September, 2007. Each HEU fuel assembly contains about 40.2 g of U-235 distributed inside three coaxial fuel tubes (elements), of which the outermost one is hexagonal shaped and the two inner ones are circular (see Figure 1). Each fuel element is composed of three layers; the fuel meat has a thickness of 0.7 mm and is cladded by two aluminum alloy layers with thickness of 0.9 mm as shown in Figure 1. The spaces between fuel elements are channels for coolant water. The total length of the fuel assembly is 865 mm, of which the fuelled part is

600 mm. The core is primarily reflected by graphite with a small amount of beryllium surrounding the core periphery inside the core vessel and in the central irradiation channel (neutron trap). Reactor control and protection rely on six control rods composed of boron carbide in stainless steel sheath (two are safety rods and the other four are shim rods), and a regulating rod, composed only of stainless steel. Each rod hangs from a flexible cable connected to its own electric motor drive. The core cooling is maintained by natural convection. A circular shell is installed right above the core to intensify water flow through the core by providing a 'chimney' effect. To keep the temperature of pool water at the core inlet below the operational limit, hot pool water is withdrawn from the upper part of the pool and circulated through a primary cooling loop. Heat rejection is achieved through a secondary cooling loop. The heat exchanger removes heat from primary coolant to secondary coolant, from which heat is discharged into the outside atmosphere through a fan-forced air-cooling tower. Each LEU (19.75% enriched) fuel assembly contains an average of 49.7 g of U-235 ^{with} UO_2-Al dispersion fuel meat. Each of the fuel elements in the HEU and LEU fuel assemblies has the same overall thickness of 2.5 mm, but the LEU fuel meat and cladding thickness are 0.94 mm and 0.78 mm, respectively. The geometries and materials of the HEU VVR-M2 fuel assemblies and LEU VVR-M2 fuel assemblies are described in Table 1. A cross sectional view of the VVR-M2 HEU and LEU fuel assembly is shown in Figure 2. Each LEU (19.75%) fuel assembly contains an average of 49.7 g ^{235}U with UO_2-Al dispersion fuel meat. Each of the fuel elements in the HEU and LEU fuel assemblies has the same thickness of 2.50 mm, but the fuel meat and cladding thickness are different.

Table 1. Characteristics of the HEU and LEU Fuel Assemblies

Parameter	**VVR-M2 HEU**	**VVR-M2 LEU**
Enrichment, %	36	19.75
Average Mass of ^{235}U in FA, g	40.20	49.70
Fuel Meat Composition	U-Al Alloy	UO_2+Al
Uranium Density of Fuel Meat, g/cm^3	1.40	2.50
Cladding Material	Al alloy (SAV1)	Al alloy (SAV1)
Fuel Element Thickness (Fuel Meat and Cladding), mm	2.50	2.50
Fuel Meat Thickness, mm	0.70	0.94
Cladding Thickness, mm	0.90	0.78

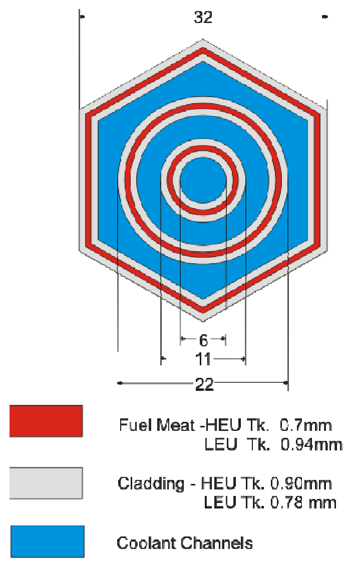

Fig. 1. Cross Sectional View of the VVR-M2 HEU and LEU Fuel Assembly

The preliminary study on new configuration with only LEU fuel assemblies for the DNRR has been performed. A working core configuration with 92 LEU fuel assemblies has been proposed.

2. Realization of the Core Conversion

Contracts for reactor core conversion between Russia, Vietnam, USA and the International Atomic Energy Agency for Nuclear fuel manufacture and supply for DNRR and Return of Russian-origin non-irradiated highly enriched uranium fuel to the Russian Federation have been realized. The 35 fresh HEU FAs were sent back to Russian Federation. We have received 36 new LEU FAs from Russian Federation. Fuel reloading has been executed by using LEU FAs on 12 September, 2007. The 8 HEU FAs with highest burnup were removed from the core periphery positions (P 1-3, P 1-5, P 1-2, P 2-1, P 13-3, P 13-1, P 13-4 and P 12-8). The 8 HEU FAs from second ring counted from neutron trap (P 6-9, P 5-8, P 5-4, P 6-4, P 8-4, P 9-4, P 9-8 and P 8-9) were moved to previous FA positions. The 2 HEU FAs from the core periphery positions (P 1-4 and P 13-2) were moved to 2 positions in second ring (P 6-9 and P 8-4). The 6 new LEU FAs were added in 6 positions in second ring (P 5-8, P 5-4, P 6-4, P 9-4, P 9-8 and P 8-9). The 2 wet irradiation channels were added in 2 positions of core periphery (P 1-4 and P 13-2). After reloading the working configuration of reactor core consisted of 104 FAs (98 HEU FAs and 6 new LEU FAs). We had first 8 spent HEU FAs. Figure 2 and 3 show reloading schema of DNRR and working configuration of DNRR. The value of 0.68 $ was increased to the reactor excess reactivity after reloading operation. Figure 4 and 5 show measured neutron spectrum at neutron trap and measured neutron flux distribution at neutron trap. Table 2 presents measured thermal and fast neutron flux at irradiation positions.

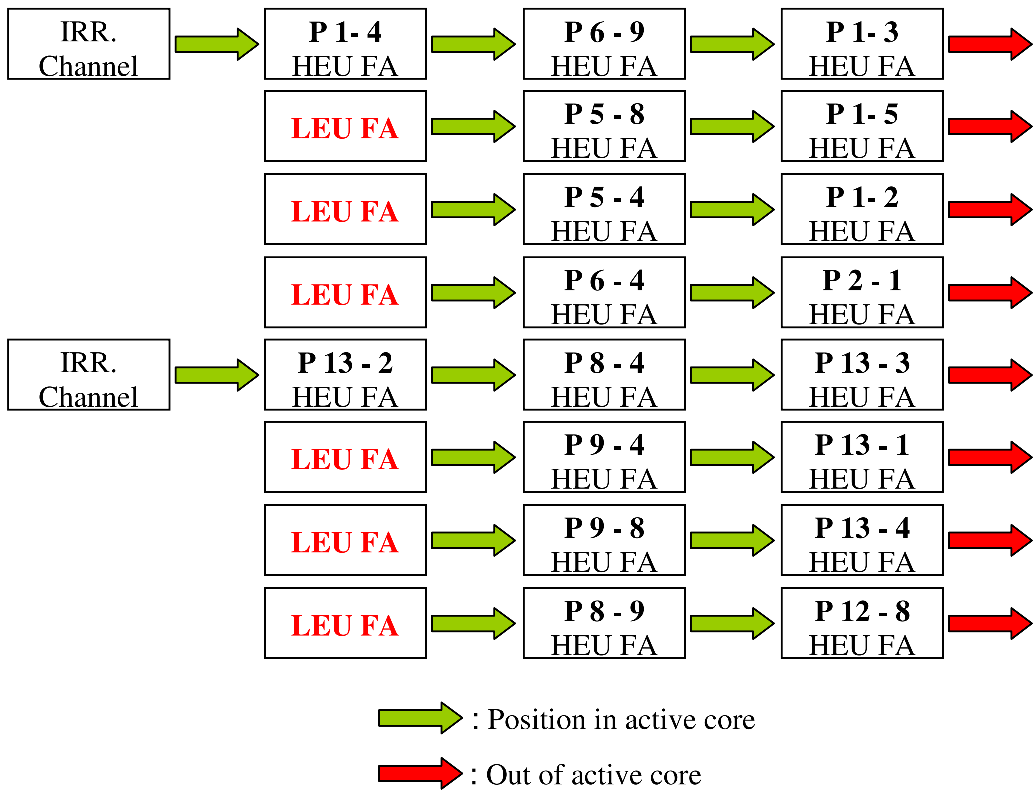

Fig. 2. Reloading schema of DNRR

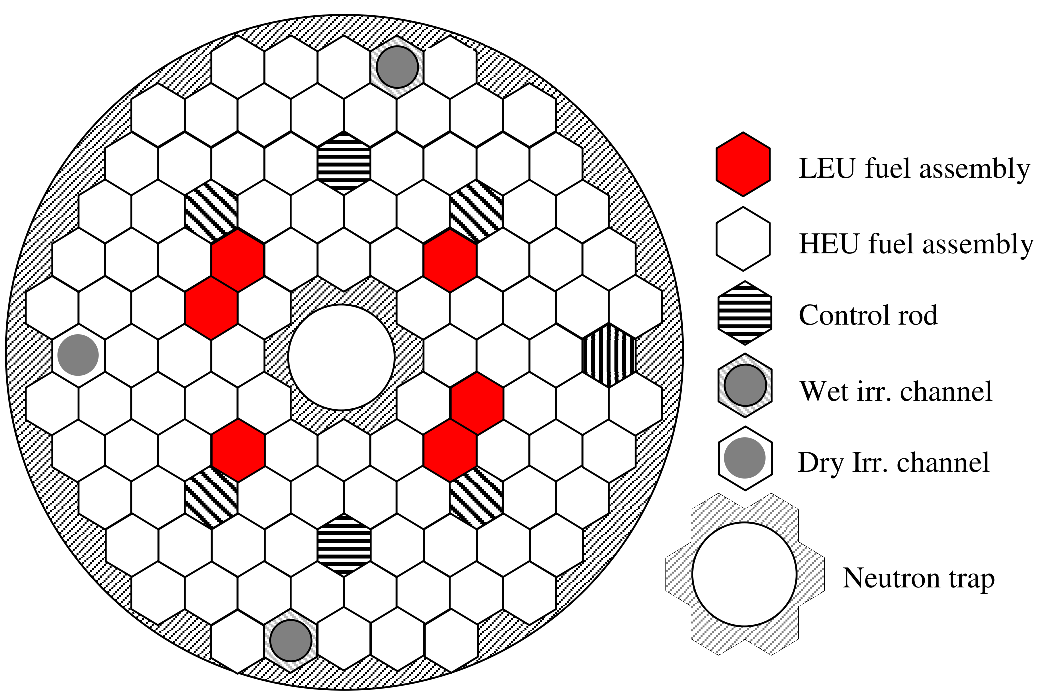

Fig. 3. Working Configuration of DNRR

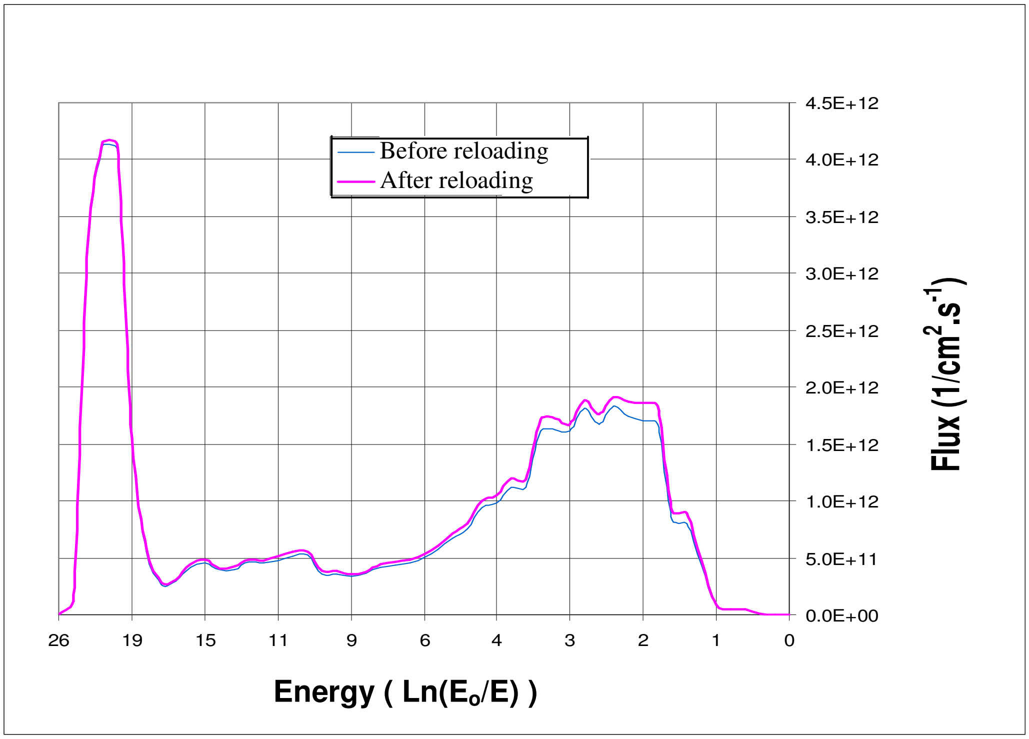

Fig. 4. Measured Neutron Spectrum at Neutron Trap

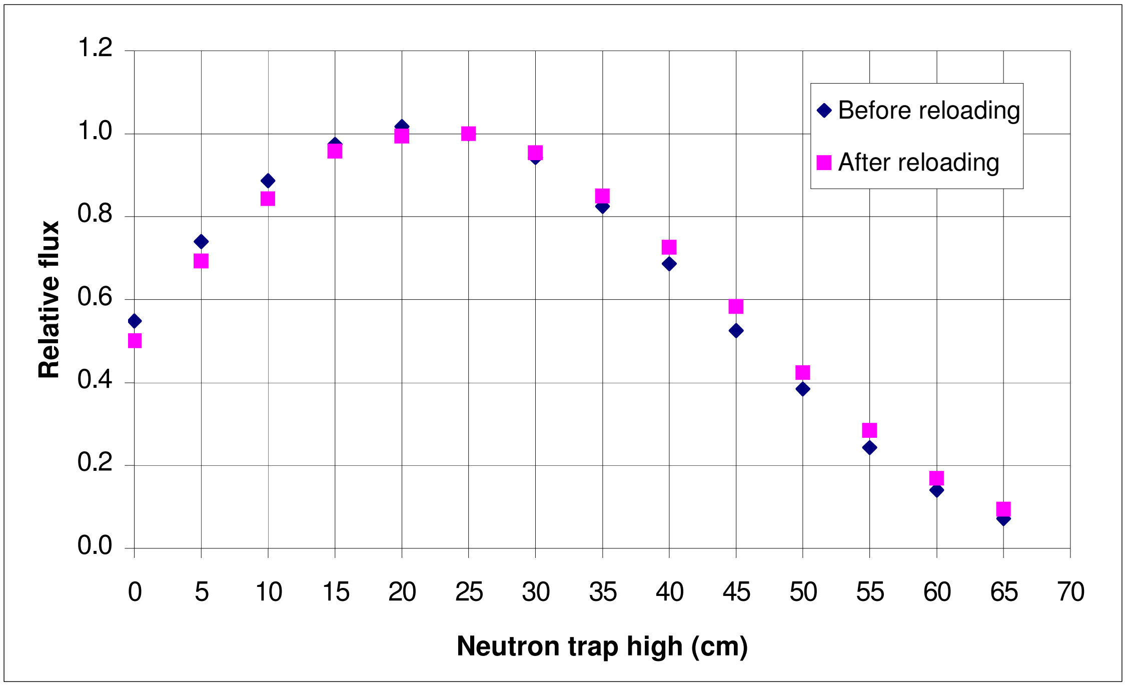

Fig. 5. Measured Neutron Flux Distribution at Neutron Trap

Table 2. Measured Thermal and Fast Neutron Flux at Irradiation Positions

Position	Before reloading		After reloading	
	$\phi_{th}^{(*)}$	$\phi_f^{(**)}$	$\phi_{th}^{(*)}$	$\phi_f^{(**)}$
Neutron Trap	$1,99 \times 10^{13}$	$4,45 \times 10^{12}$	$1,98 \times 10^{13}$	$4,61 \times 10^{12}$
Wet channel 13-2	-	-	$4,61 \times 10^{12}$	-
Rotary specimen rack	$4,28 \times 10^{12}$	-	$4,67 \times 10^{12}$	-

$^{(*)}$ Thermal neutron flux, $n.cm^{-2}.s^{-1}$.

$^{(**)}$ Fast neutron flux, $n.cm^{-2}.s^{-1}$

Measurement of maximum fuel cladding temperature after reloading has been carried out. In fact we measured fuel cladding temperature of instrumented FA. Because of instrumented FA has very low burnup compare to replaced FA then measured fuel cladding temperature of instrumented FA higher than value of replaced FA. Table 3 presents measured maximum fuel cladding temperature of instrumented FA placed at hottest position (P 9-6) near neutron trap. From table 3 we note that measured maximum fuel cladding temperature of instrumented FA with reactor power of 500 kW and inlet coolant temperature of 32 ^oC is less than 94 ^oC. This value is less than specified limit of 107 ^oC.

Table 3. Measured Maximum Fuel Cladding Temperature after Reloading

P (kW)	**T_{in} (^oC)**	**$T_{c, max}$ (^oC)**
2,5	20,3	21,9
250	20,8	61,2
400	22,0	79,0
500	23,1	90,2
	32,0	93,7

P: Reactor power

T_{in}: Inlet coolant temperature

$T_{c,max}$: Maximum fuel cladding temperature

3. Preliminary Study on New Configuration with only LEU Fuel Assemblies

MCNP Code was used to calculate critical core configuration, shutdown margin, neutron flux and power distribution. Kinetics parameters and reactivity coefficients were calculated by VARI3D and REBUS Codes. Neutron cross sections for use in the REBUS and VARI3D Codes were generated using WIMS-ANL code. The WIMS-ANL uses a 69 energy group library based on ENDF-B/VI data. The fuel assembly cross sections were generated in a radial geometry with each fuel element depleted based upon its unique neutron spectrum in the WIMS-ANL model. A separate WIMS-ANL model was created to generate cross sections for the control rods, irradiation positions, and reflector materials.

A detailed geometrical model of reactor components including all fuel assemblies, control rods, irradiation positions, beryllium and graphite reflectors, horizontal beam tubes and thermal column was made in the MCNP model, except in the axial reflectors above and below the fuel assembly where some materials were homogenized. Nuclear cross sections were based on ENDF-B/VI cross section library. Model of DNRR in MCNP and REBUS Codes is shown in Figure 6.

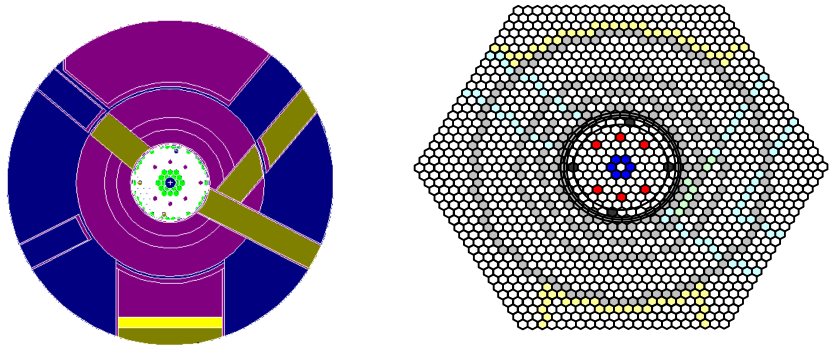

Fig. 6. Model of DNRR in MCNP and REBUS Codes

3.1. Critical core and working core with LEU fuel

DNRR was reached criticality without neutron trap and with neutron trap in 1983 contented 69 and 72 fuel assemblies respectively by using HEU fuel. If using LEU fuel, critical core without neutron trap has only 66 FAs. For critical core with neutron trap and using 72 LEU FAs, 12 Beryllium blocks will be put around neutron trap to create new neutron trap with 19 holes in the center of the core. Critical core configuration with 66 and 72 FAs LEU fuel is shown in Figure 7.

When designing for working core of DNRR in 1984, specialists have given a working core by loading 94 HEU FAs but in practice, the working core was only loaded 88 HEU FAs. The main reason is that loading more fuel assemblies excess reactivity will increase and shutdown margin will not meet requirements. But another problem was taken care related to high power density and it made low thermal hydraulics margin because of fewer number loaded FAs in the core. To LEU fuel, mass of U-235 and Uranium density is higher than HEU fuel so this problem became more serious. By creating new neutron trap with 12 Beryllium blocks to be arranged around and put FAs fully all holes remain with 92 LEU FAs so this working core with 92 LEU FAs will meet all safety requirements. Working core configuration with 92 LEU FAs type WWR-M2 is shown in Figure 8.

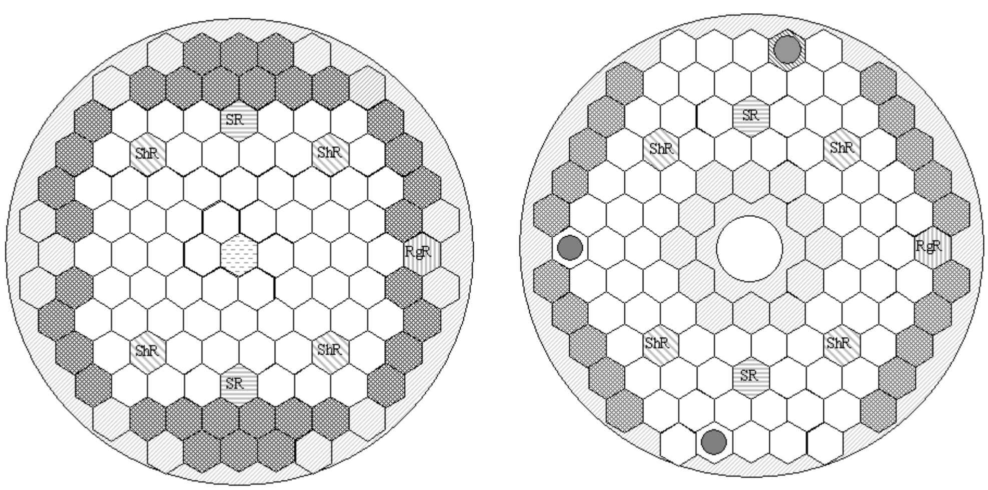

Fig. 7. Critical core configuration with 66 and 72 FAs LEU fuel

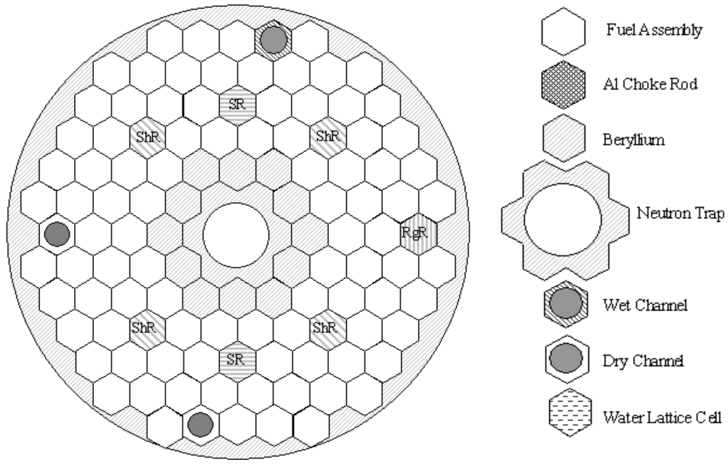

Fig. 8. Working core configuration with 92 LEU FAs type WWR-M2

3.2. Neutronics characteristics of working core

MCNP Code was used to calculate shutdown margin, power distribution and neutron flux at irradiation positions. In this configuration of the core with 92 LEU FAs, effective multiplication factor in some cases have values like this:

All control rods fully withdrawn: $k_{eff} = 1.07895 \pm 0.0008$
All control rods fully inserted except 2 safety rods: $k_{eff} = 0.97616 \pm 0.0009$
All control rods fully inserted: $k_{eff} = 0.92902 \pm 0.0008$

The shutdown margin with four shim rods and the regulating rod fully inserted, and the two safety rods fully withdrawn is -2.442% $\Delta k/k$. This value meets safety requirement.
If loading according to configuration of 92 HEU FAs (Beryllium blocks were put at periphery of the core the same in 1984) the effective multiplication factor has value 1.11238 ± 0.0009. It is clear that shutdown margin in this case is not reliable. If reducing number of FAs in the core than thermal hydraulics margin is not reliable because of increasing power density.
Power distribution was calculated by using MCNP Code and it showed in Figure 9. In this calculation, radial peaking factor is 1.40. This value will have value 1.41 if loading FAs and Beryllium blocks following old way (putting Beryllium blocks at periphery of the core). Fast (> 0.821 MeV) and thermal (< 0.625 eV) neutron flux in some irradiation positions were depicted in Table 4. Comparing with current core configuration, neutron flux at irradiation positions are not changed a lot. In neutron trap, thermal flux increase about 3% but fast neutron reduce about 40% while at irradiation channels 1-4, 7-1 and 13-2 thermal neutron flux reduce and fast neutron flux increase because neutron spectrum is harder when Uranium density increases. Maximum of thermal neutron flux in neutron trap has value 2.26E+13n/cm^2.s. It is show that this working core configuration meets all requirements about safety and exploiting.

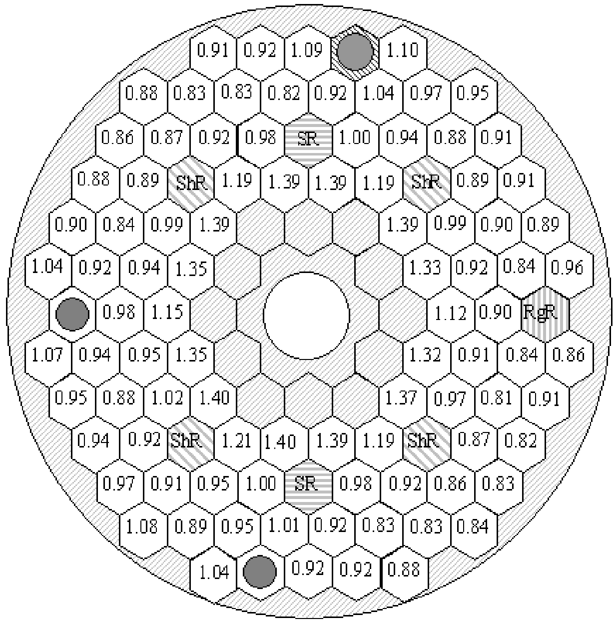

Fig. 9. Power Distribution of Working Core with 92 LEU

Table 4. Thermal and Fast Neutron Flux at Irradiation Positions of the Core with 92 LEU FAs and HEU Core

Position	**Thermal neutron flux ($n/cm^2.s$)**		**Fast neutron flux ($n/cm^2.s$)**	
	Core with 92 LEU	**Core with HEU**	**Core with 92 LEU**	**Core with HEU**
Neutron Trap	2.28E+13	2.22E+13	1.98E+12	3.23E+12
1-4 Channel	9.54E+12	9.88E+12	2.99E+12	2.63E+12
7-1 Channel	5.53E+12	6.27E+12	4.24E+12	3.38E+12
13-2 Channel	6.06E+12	6.74E+12	3.71E+12	3.27E+12

3.3. Kinetics parameters and reactivity coefficients

WIMS-ANL Code was used to create micro cross section from ENDF-B6.0 for VARI3D and REBUS Codes to calculate kinetics parameter, reactivity coefficients and void fraction of moderator. These parameters were also calculated by MCNP Code. Table 5. and Table 6. are showed detail obtained results from these Codes.

Table 5. Kinetics Parameters of Working Core with 92 LEU FAs

Beta	VARI3D:7,55094E-03; MCNP: 7,1612E-03		
Group (i)	Beta (i)	A (i)	Lambda (i)
1	2.64810E-04	3.50690E-02	1.33370E-02
2	1.36250E-03	1.80440E-01	3.27300E-02
3	1.31520E-03	1.74170E-01	1.20800E-01
4	2.90170E-03	3.84280E-01	3.02960E-01
5	1.20350E-03	1.59380E-01	8.50280E-01
6	5.03330E-04	6.66580E-02	2.85560E+00
Prompt lifetime	VARI3D: 8.79493E-05; MCNP: 8.78E-5 ± 6.4E-6		

Table 6. Reactivity and Void Coefficients of Working Core with 92 LEU

	REBUS	**MCNP**
Moderator Temperature Reactivity Coefficient $\Delta\rho(\%)/^oK$		
296^oK-350^oK	-0.00978	
350^oK-400^oK	-0.00995	
Fuel Temperature (Doppler) Reactivity Coefficient $\Delta\rho(\%)/^oK$		
300^oK-560^oK	-0.00172	
300^oK-400^oK		-0.00136
400^oK-500^oK		-0.00148
500^oK-700^oK		-0.00144
Moderator density (Void) Reactivity Coefficient $\Delta\rho(\%)/(\%void)$		
Void 0-5%	-0.213	-0.212
Void 5-10%	-0.233	-0.230
Void 10-20%	-0.267	-0.271

Obtained results show that it is slightly different of kinetics parameters and Reactivity and void coefficients in this working core compare with current core. So this working core will meet all criteria of safety analysis requirements.

4. Conclusion and Remarks

The contracts for reactor core conversion between Russia, Vietnam, USA and the International Atomic Energy Agency for Nuclear fuel manufacture and supply for DNRR and Return of Russian-origin non-irradiated highly enriched uranium fuel to the Russian Federation have been realized. Fuel reloading has been executed by using LEU FAs. Now DNRR mixed core consists of 98 HEU FAs and 6 LEU FAs. With an idea to arrange Beryllium blocks around neutron trap, it is permitted to create core with 92 LEU FAs type WWR-M2. This core still keeps the safety limit in operating and exploiting of DNRR.

5. References

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