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Feasibility Studies for LEU Conversion of the IRT MEPhI Reactor Using U-Mo Tubular Fuel

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ABSTRACT

The 2.5MW IRT MEPhI research reactor in National Research Nuclear University MEPhI (Moscow, Russia) currently uses HEU (90%) IRT-3M 8(6)-tube fuel assemblies (FA). This study compares the neutronic performance of the reactor and its experiments using LEU tube-type U-Mo fuel assembly IRT-3M with the performance of the current HEU (90%) reference fuel assembly and core. The parameters of operational performance, safety parameters and experimental performance parameters for the initial core (12 fresh FA) and operational core (16 FA) are defined. BNCT neutron beams parameters for existing HEU and LEU cores are considered.

1. Introduction

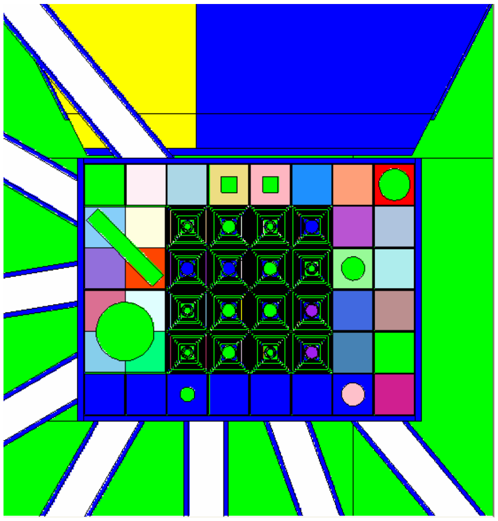
NNSA (USA) and Rosatom (Russia) have agreed to study the feasibility of converting six research reactors in Russia to LEU fuel. One of these is the IRT MEPhI research reactor at the National Research Nuclear University MEPhI.

The feasibility studies started in December 2010 and should be finished in November 2011. The feasibility studies are being carried out in cooperation with Argonne National Laboratory (USA) and at its funding support within the RERTR program under the contract No. 0J-30402 between ANL and MEPhI.

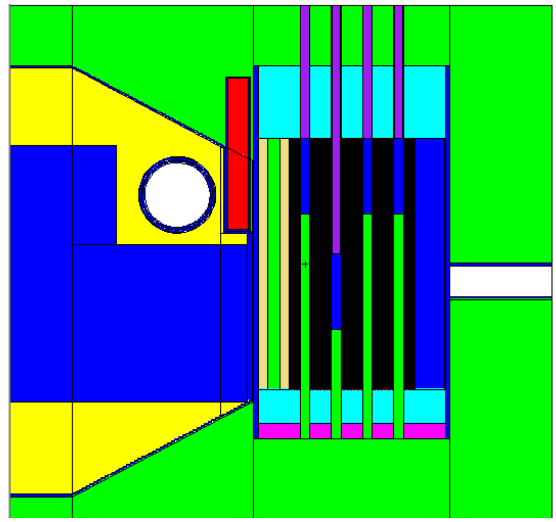
In the presented paper HEU and LEU core neutronics is analyzed.

2. Reactor description

The IRT MEPhI pool type research reactor (2.5MW) currently uses IRT-3M HEU (90%) fuel assemblies. The reactor first reached criticality in May 1967. IRT MEPhI began to operate with 4(3)- tube IRT-2M FA in 1975. The reactor is used for testing of the wide range neutron flux control channels for NPP reactors, ionization chambers and new types of radiation protective cables, neutron capture therapy investigations, neutron activation analyses, experiments in nuclear physics and personnel training. The reactor has 10 horizontal beam tubes, a graphite thermal column, and vertical irradiation channels in the reflector. Figure 1 shows a horizontal (a) and vertical (b) cross sections of the IRT MEPhI reactor current core with 16 fuel assemblies.



(a)



(b)

Figure 1 Current HEU core. Horizontal core cross section (a), vertical core cross section (b)

3. LEU and HEU fuel assemblies

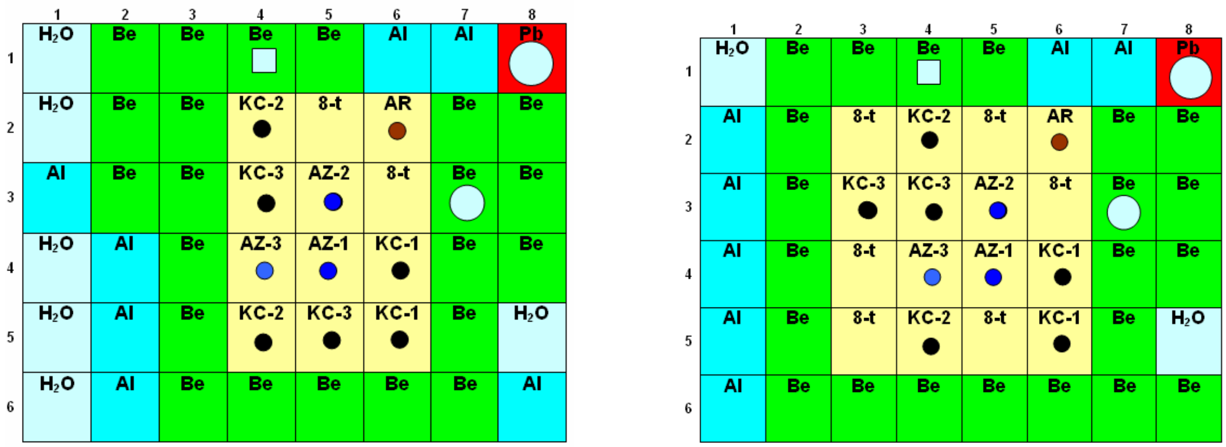
For feasibility studies of the IRT MEPhI reactor conversion fuel assembly (FA) IRT-3M with U9%Mo-Al fuel (enrichment 19.7%) was chosen as a LEU fuel [1]. Outer dimensions of the U-Mo IRT-3M FA are the same as those of HEU FA IRT-3M except of rounded corners radii.
In table 1 the data about FA IRT-3M with LEU and HEU fuel used in the study are presented.

Table 1 LEU and HEU FA parameters

Parameter	LEU FA		HEU FA	
	8-tube	6-tube	8-tube	6-tube
U density, g/cc	5.4	5.4	1.07	1.07
U^5 mass, g	405.1	355.1	300	263.7
Enrichment, %	19.7	19.7	90	90
U mass, g	2051.27	1803.07	333.7	293.3
Mo weight fraction in U-Mo	0.09	0.09	-	-
Meat thickness, cm	0.05	0.05	0.04	0.04
Clad thickness, cm	0.045	0.045	0.05	0.05
Clad material	SAV-1	SAV-1	SAV-1	SAV-1
Meat length, cm	58	58	58	58
Meat volume, cc	380.8	333.8	313.42	274.69

4. LEU and HEU cores

Within the framework of the feasibility studies initial (12 fresh FA) and operational (16 FA) cores with HEU and LEU fuel were analyzed. Initial and operational HEU/LEU core configurations are presented in figure 2. Burnup distributions for operational HEU and LEU core were obtained as a result cycle-by-cycle burnup analysis described further.



(a) (b)

Figure 2 Initial (a) and operational (b) LEU cores

The isotope concentrations for the MCNP calculation of the core at different stages of fuel burnup were obtained on the basis of ^{235}U burnup distribution for 6 axial layers calculated by TIGRIS code. The isotope concentrations are obtained from ^{235}U burnup using polynomial approximations. These polynomial approximations are defined on the basis of the results of GETERA calculation.

30 isotopes (actinides and fission products) with the largest capture cross sections were chosen. The other isotopes were excluded from calculation. The contribution of excluded isotopes to neutron multiplication factor was considered by introduction of surrogate (fictitious) product ^{101}Ru. The concentration of ^{101}Ru was chosen so that at all burnup stages the infinite neutron multiplication factor for 8-tube FA calculated by MCNP with 30 isotopes differs from that calculated by GETERA with all isotopes by approximately -0.3% (it is the difference between MCNP and GETERA results for the burnup 0%).

5. Models and codes

Calculations of criticality, CR worth, excess reactivity, shutdown margin, detail power density distribution and BNCT beams parameters were performed using MCNP code [2].

For the calculation of burnup process with reloads, criticality, CR worth, excess reactivity, shutdown margin code TIGRIS [3] was used with 4-group macro cross-sections (upper boundaries of energy ranges are equal to 10.5 MeV, 0.8 MeV, 4.6 keV and 0.63 eV, respectively). TIGRIS code is intended for 3-dimensional (in rectangular geometry) diffusion steady state neutronic calculation based on nodal or finite difference algorithm. The macro cross-sections library has been created by the code GETERA [4].

The burnup process was calculated using the following approximations. Fuel burnup process is analyzed under presumption that isotopic compositions of fuel assemblies at different stages of fuel burnup are unambiguously defined by amount of burnt-up ^{235}U. So, depletion of ^{235}U only is determined in the reactor burnup computations (in diffusion code). Then, macro cross-sections are calculated for current value of fuel burnup with application of appropriate polynomial dependencies preliminarily prepared by the lattice code GETERA.
In LEU case there is energy production from the plutonium created in the fuel; i.e. the consumption of ^{235}U is less than for an HEU core. The energy production from the plutonium created in the LEU fuel was taken into account in TIGRIS code by using the coefficient between the integral energy production and ^{235}U burnt-up mass:
K_w=(1.27-2.236E-3·b - 1.928E-5·b^2) [g/MWd], where b is ^{235}U burnup in %.
The dependence of the coefficient between energy production and ^{235}U burnt-up mass versus ^{235}U burnup was estimated from GETERA calculation.

Described burnup model is rather coarse but it enables to perform a large number in a short time and hence it enables to analyze the detail history of reactor operation during many years very quickly. Moreover, if the objects of calculation are integral reactivity characteristics (excess reactivity, CR worth) and ^{235}U burnup (as a criterion for FA discharge), then described burnup model does not lead to the large errors. It is unacceptable only if plutonium concentration itself is the object of calculation.

6. Calculation of burnup process with reloads

Calculation of burnup process with reloads of IRT MEPhI LEU/HEU core was performed by TIGRIS code. The initial core with 12 fresh FA was chosen as a start core. The cycle length was chosen so that at the end of cycle either excess reactivity without Xe was 3÷4$, either average burnup of one of the FA's has reached the value 55%. Fresh FA were loaded in peripheral cells of the core. 18 cycles were considered (total energy generation is 77100 MWh for HEU core and 106000 MWh for LEU core). At the beginning of each cycle CR positions were: AR=250 mm, KC-3 in critical position, KC-1,2=0 (criticality with accuracy up to 0.5$). During burnup process KC-3 rods were withdrawn to keep criticality approximately. The calculations were carried out for stationary poisoned state. As the calculation of burnup process with reloads was performed to compare fuel consumption of HEU and LEU fuel under the same conditions the beryllium poisoning was not considered in this calculation.
It should be noticed that the reloadings were the same for the LEU and HEU cores. So the difference was only in cycle length.
Excess reactivity (without Xe) versus integral energy generation for the first cycle is shown in figure 3 (β_{eff}=0.0077). The results calculated by TIGRIS code are presented. For energy generation 6000 MWh and 16000 MWh the excess reactivity calculated by TIGRIS code is also presented (with burnup distribution calculated by TIGRIS).

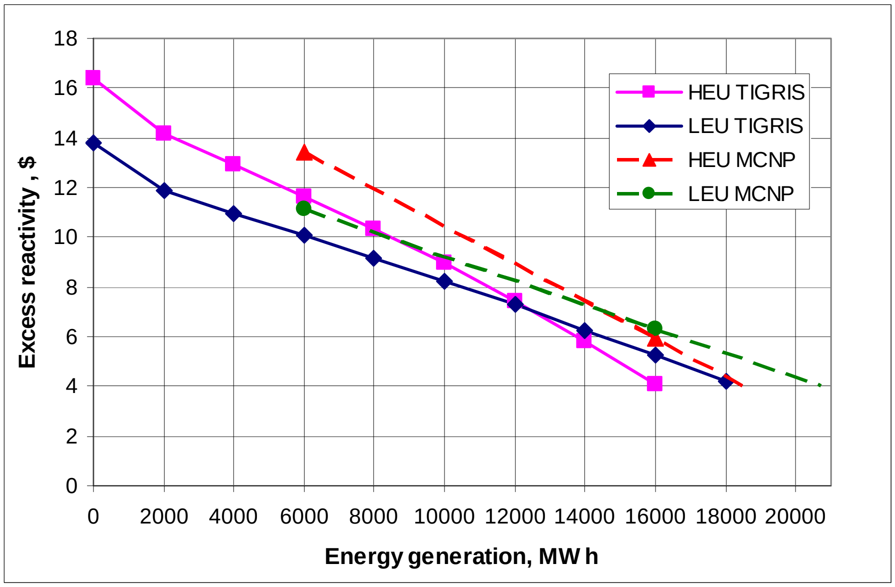


Figure 3 - Excess reactivity (without Xe) versus energy generation for the first cycle

The first cycle for LEU core is longer by12.5% than that for HEU core. Reactivity loss due to burnup is 4.8$ per 10000 MWh (6000÷16000 MWh) for LEU core and 7.5$ for HEU core.
Using the results of the first cycle calculation total mass of burnt-up ^{235}U can be estimated. Total mass of burnt-up ^{235}U is 830 g for LEU core and 846 g for HEU core. That is for the first cycle the saving of ^{235}U due to plutonium production is 2% for LEU fuel in comparison with HEU fuel. So the saving of reactivity (35%) is more significant than the saving of ^{235}U. It should be noticed that estimations mentioned above were made for the core with average burnup 0÷20%. For burnup >20% such estimation is presented further.
Excess reactivity (without Xe) versus energy generation for the cycles 2 - 18 is shown in figure 4.

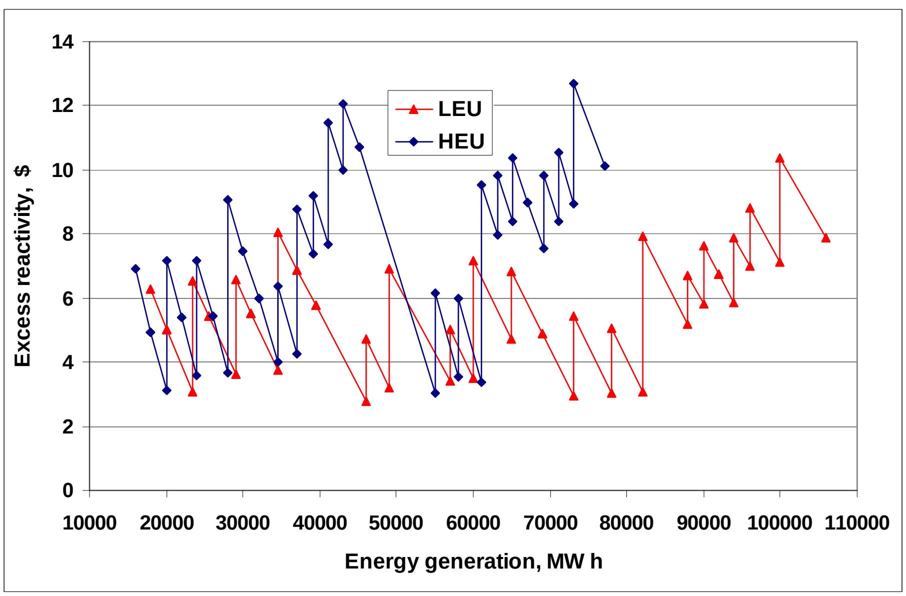


Figure 4 - Excess reactivity (without Xe) versus energy generation for the cycles 2 – 18 for LEU and HEU core (TIGRIS)

FA consumption was estimated. The number of used FA during the burnup process for LEU and HEU core is presented in figure 5.

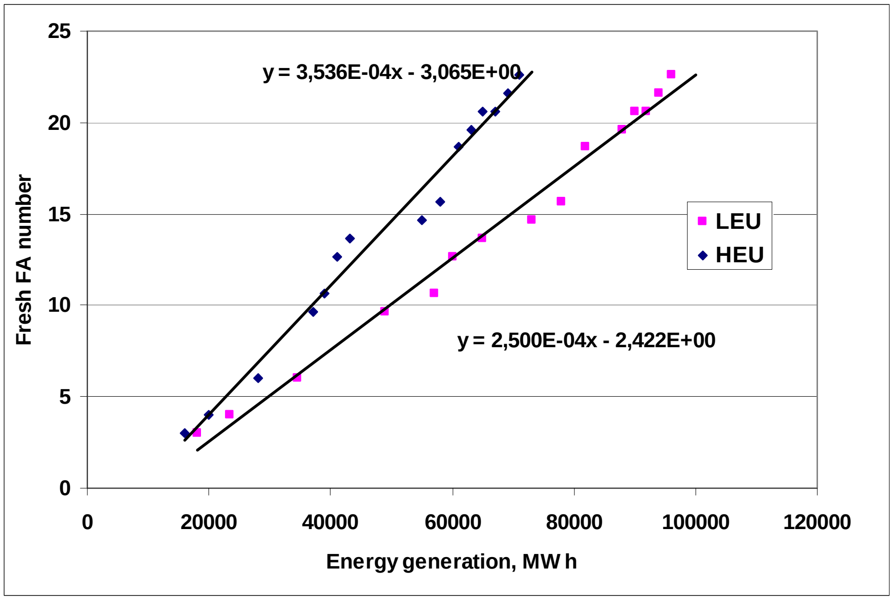


Figure 5 - The number of used FA during the burnup process

Using lines gradient from figure 5 (2.5e-4 and 3.5e-4) the fuel consumption can be estimated. If energy generation per year is 4000 MW·h then fuel consumption is 1.0 FA/year for LEU core and 1.4 FA/year for HEU core. That is if fuel consumption is measured in FA number, then for LEU core fuel consumption is less by 40% than for HEU core.

Fuel consumption in FA/year takes into account the impact of FA fabrication and transportation costs on the reactor operation costs. But from physical point of view it is also useful to compare fuel consumption in mass of burnt-up ^{235}U. Table 2 presents some parameters of fuel consumption in HEU and LEU core which were calculated from the results of calculation of 18 cycles. The mass of burnt-up ^{235}U in discharged FA (for cycles 1-18), the mass of burnt-up ^{235}U in FA's of the last cycle (18) and the sum of these values. The mass of burnt-up ^{235}U was calculated taking into account the difference in initial ^{235}U in 8-tube and 6-tube FA.

Table 2 Fuel consumption of HEU and LEU cores comparison

Parameter	HEU	LEU
Total energy generation, MWh	77100	106000
Mass of burnt-up ^{235}U in discharged FA, g	2586.6	3357.2
Mass of burnt-up ^{235}U in FA of the last cycle, g	1199.8	1456.0
Total mass of burnt-up ^{235}U, g	3786.4	4813.2
^{235}U burnup rate, g^{5}U/ MWh	0.04911	0.045408
Mass of burnt-up ^{235}U per 4000 MWh, g	196.4	181.6
Energy generation per mass of burnt-up ^{235}U =196.4 g, MWh	4000	4326

^{235}U burnup rate (Total mass of burnt-up 235U/ Total energy generation) for LEU fuel is less by 8% than that for HEU fuel. The mass of burnt-up ^{235}U per year (4000 MWh) for LEU fuel is also less by 8% than that for HEU fuel. The difference 8% is caused by larger plutonium production in LEU case. It should be noticed that the advantage of LEU in fuel consumption measured in the mass of burnt-up ^{235}U is not as large as in fuel consumption measured in FA number. That is the saving of ^{235}U due to plutonium production is not large but created plutonium leads to positive reactivity effect and reactivity loss due to burnup is less for LEU core than for HEU core. The average rate of reactivity loss due to burnup for LEU fuel is less by 40% than that for HEU fuel. The increase of excess reactivity after reloading for LEU fuel is less by 10-20%. The behavior of excess reactivity for LEU fuel is preferable since it is smoother.

7. Calculation of key experiment performance indicators

Parameters at the outlet of the irradiation beams HEC-4 and HEC-1 used for BNCT investigations were calculated by MCNP code. Two core configurations are considered: the initial core with 12 FA and operational core with 16 FA.
The scheme of HEC-1 facility used in the calculations is presented in figure 6.

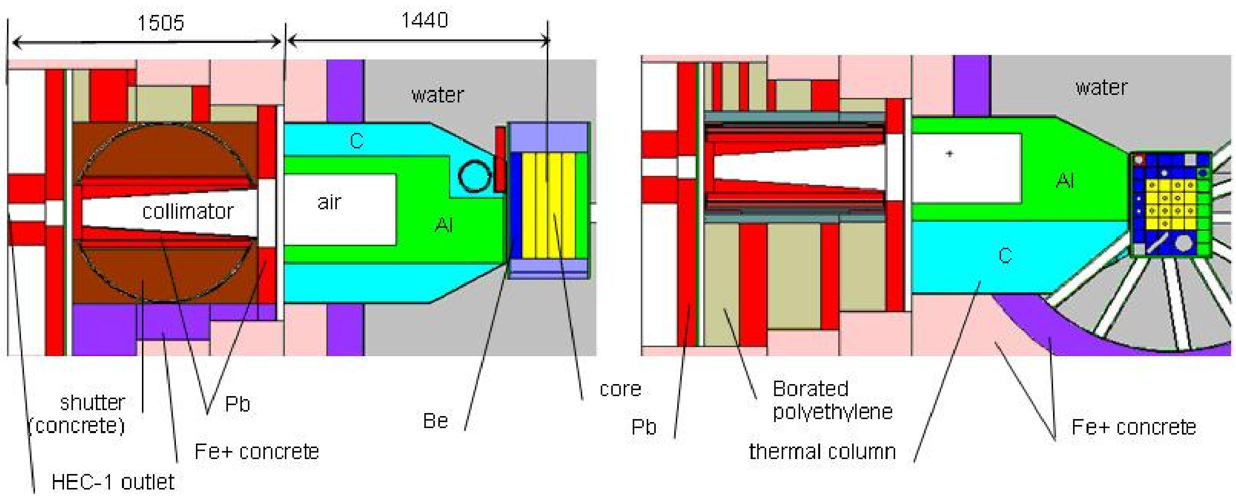


Figure 6 -HEC-1 facility (2009)

The real geometry of the core is described, but fuel assemblies are homogeneous with the exception of channels for control rods. The neutron source spatial distribution was set in accordance with the power distribution obtained as a result of TIGRIS diffusion calculation (six axial layers for each separate fuel assembly). Fuel composition for the operational core is set in accordance with the isotope concentrations for 27% (HEU) and 24% (LEU) ^{235}U burnup.
Calculated thermal neutron flux (E<0.5eV), epithermal neutron flux (0.5eV<E<10keV), fast neutron flux (E>10keV), fast neutrons kerma and photons kerma at the outlet of the irradiation beams HEC-4 and HEC-1 (in air) are presented in table 3, 4. Relative kerma of the fast neutrons and photons (per thermal or epithermal neutron) is also presented in tables 3, 4.

Table 3 Horizontal beam HEC-4 parameters for LEU and HEU core

Parameter	Fresh core		Operational core	
	HEU	LEU	HEU	LEU
Thermal neutron flux, n/cm^{2}s	8.42E+08 (±0.014)	7.61E+08 (±0.015)	6.77E+08 (±0.019)	6.56E+08 (±0.015)
Epithermal neutron flux, n/cm^{2}s	4.53E+08 (±0.02)	4.53E+08 (±0.028)	3.61E+08 (±0.018)	3.80E+08 (±0.033)
Fast neutron flux, n/cm^{2}s	1.92E+08 (±0.039)	1.71E+08 (±0.017)	1.50E+08 (±0.012)	1.52E+08 (±0.04)
Fast neutron Kerma, Gy/s	2.15E-03	1.97E-03	1.69E-03	1.71E-03
Photon Kerma, Gy/s	5.30E-04	4.66E-04	4.32E-04	4.46E-04
Fast neutron kerma /Thermal flux, Gy cm$^{2/}$n	2.56E-12	2.59E-12	2.49E-12	2.61E-12
Fast neutron kerma /Epithermal flux, Gy cm^2/n	4.76E-12	4.34E-12	4.67E-12	4.50E-12
Photon kerma / Thermal flux, Gy cm^2/n	6.29E-13	6.12E-13	6.37E-13	6.79E-13
Photon kerma / Epithermal flux, Gy cm^2/n	1.17E-12	1.03E-12	1.20E-12	1.17E-12

Table 4 Horizontal beam HEC-1 parameters for LEU and HEU core

Parameter	Fresh core		Operational core	
	HEU	LEU	HEU	LEU
Thermal neutron flux, n/cm^{2}s	7.56E+08 (±0.016)	6.92E+08 (±0.011)	6.00E+08 (±0.012)	5.62E+08 (±0.012)
Epithermal neutron flux, n/cm^{2}s	1.17E+09 (±0.020)	1.11E+09 (±0.014)	9.04E+08 (±0.015)	8.84E+08 (±0.015)
Fast neutron flux, n/cm^{2}s	4.37E+08 (±0.033)	4.14E+08 (±0.021)	3.4E+08 (±0.028)	3.24E+08 (±0.026)
Fast neutron kerma, Gy/s	2.16E-03	2.14E-03	1.74E-03	1.57E-03
Photon kerma, Gy/s	1.57E-03	1.50E-03	1.40E-03	1.19E-03
Fast neutron kerma /Thermal flux, Gy cm$^{2/}$n	2.85E-12	3.10E-12	2.91E-12	2.79E-12
Fast neutron kerma /Epithermal flux, Gy cm^2/n	1.84E-12	1.92E-12	1.93E-12	1.77E-12
Photon kerma / Thermal flux, Gy cm^2/n	2.08E-12	2.17E-12	2.34E-12	2.12E-12
Photon kerma / Epithermal flux, Gy cm^2/n	1.34E-12	1.35E-12	1.55E-12	1.35E-12

Table 5 presents the ratio of the parameters of HEC-1,4 for LEU fresh (operational) core and HEU fresh core to the parameters of HEC-1,4 for HEU operational core. HEU operational core was chosen as a sample variant because it corresponds to the current status.

Table 5 Comparison of the horizontal beam parameters for LEU and HEU core

Parameter	HEC-1				HEC-4			
	Fresh		Operational		Fresh		Operational	
	HEU	LEU	HEU	LEU	HEU	LEU	HEU	LEU
Thermal neutron flux	1.26	1.15	1	0.94	1.24	1.12	1	0.97
Epithermal neutron flux	1.30	1.23	1	0.98	1.26	1.26	1	1.05
Fast neutron flux	1.28	1.22	1	0.95	1.28	1.14	1	1.01
Fast neutron kerma	1.24	1.23	1	0.90	1.28	1.17	1	1.01
Photon kerma	1.12	1.07	1	0.85	1.23	1.08	1	1.03
Fast neutron kerma /Thermal flux	0.98	1.06	1	0.96	1.03	1.04	1	1.05
Fast neutron kerma/ Epithermal flux	0.95	1.00	1	0.92	1.02	0.93	1	0.96
Photon kerma/ Thermal flux	0.89	0.93	1	0.91	0.99	0.96	1	1.07
Photon kerma/ Epithermal flux	0.86	0.87	1	0.87	0.98	0.86	1	0.98

The increase of thermal and epithermal neutron fluxes is considered as a positive effect, the increase of the other parameters listed in the tables 3, 4 is considered as a negative effect. The ratios of fast neutron kerma or photon kerma to the thermal neutron flux or to the epithermal neutron flux define so-called “beam quality”. It is desirable to minimize these ratios.

Fresh LEU core would have more neutron fluxes of all energies at the outlet of HEC-1 and HEC-4 than the operational HEU core. The flux increase is not connected with the fuel type (there is such increase for the HEU fresh core in comparison with HEU operational core). The flux increase is caused by the fact that the core with less FA number at the same power level would have higher neutron flux in peripheral FA faced the thermal column. For HEC-1 the increase of epithermal and thermal neutron flux is 15÷20%.

Operational LEU core would have approximately the same neutron fluxes at the outlet of HEC-1 and HEC-4 as the operational HEU core (the difference is approximately ±5%).

It should be noticed that the ratio of photon kerma to the thermal neutron flux or to the epithermal neutron flux is less for LEU fuel case (except of HEC-1 case for operational core).

Presented results enable to state that the negative changing of any considered parameters of HEC-1,4 is below 10%. There are some positive changes especially for fresh LEU core. The amount of positive change is ~10÷20%. The positive changes will not only be for the fresh LEU core but also for (at least) the equivalent of the first five years of operation with LEU (because the LEU core will have only 12 FA).

So the conversion to LEU fuel does not lead to the decrease of neutron fluxes at horizontal beam outlet. It can be explained by the fact that only fast neutrons in the core volume give a contribution to the neutron fluxes at horizontal beam outlet. The neutrons which were moderated in the core volume can not reach the point of the measurement. The conversion to LEU fuel leads to the reduction of the ratio of thermal and fast neutrons but the number of fast neutrons is approximately the same for LEU and HEU fuel (at the same power level).

Also it should be noticed that the changing of HEC-1,4 parameters which is less than 10% can not be considered as a significant changing. Such changing is not larger than error of experimental and calculation estimation (including calculation statistics) of neutron fluxes at the beam outlet and is not larger than power measurement error.

8. The summary of the results

Within the framework of feasibility studies of IRT MEPhI conversion the comparison of neutronic performance of the reactor and its experiments using LEU (19.7%) tube-type U-Mo FA IRT-3M with the performance of the current HEU FA and core was carried out.

As a result of presented investigations the parameters of operational performance, safety parameters and experimental performance parameters for the initial core (12 fresh FA) and operational core (16 FA) of IRT MEPhI with LEU and HEU fuel IRT-3M were defined. The summary of the results is presented in table 6.

Table 6 Parameters of fresh and operational IRT MEPhI cores with HEU and LEU fuel

Parameter		Fresh core		Operational core	
		HEU	LEU	HEU	LEU
Operational performance					
Excess reactivity, β_{ef}		18.2	14.8	11.9	9.2
AR, KC worth, β_{ef}		26.9	24.6	21.1	19.3
AZ worth, β_{ef}		20.8	18.2	16.7	14.3
FA/4000MW·h				1.4	1.0
g ^{235}U/4000 MW·h				196.4	181.6
Reactivity loss due to burnup, β_{ef} / MW·h				-7.59e-04	-4.615e-04
Safety parameters					
Shutdown margin, β_{ef}		-5.4	-7.0	-9.9	-10.3
Maximum FA power, kW		280.0	274.5	241	218.9
Peak power density, kW/cc		1.677	1.541	1.605	1.232
Experiment performance					
HEC-1	Thermal neutron flux (E<0.5eV)	1.26	1.15	1	0.94
	Epithermal neutron flux (0.5eV<E<10keV)	1.30	1.23	1	0.98
	Fast neutron flux (E>10keV)	1.28	1.22	1	0.95
	Fast neutron kerma	1.24	1.23	1	0.90
	Photon kerma	1.12	1.07	1	0.85
HEC-4	Thermal neutron flux (E<0.5eV)	1.24	1.12	1	0.97
	Epithermal neutron flux (0.5eV<E<10keV)	1.26	1.26	1	1.05
	Fast neutron flux (E>10keV)	1.28	1.14	1	1.01
	Fast neutron kerma	1.28	1.17	1	1.01
	Photon kerma	1.23	1.08	1	1.03

In table 6 the excess reactivity is defined as a result of calculation with all rods withdrawn, shutdown margin is defined as a result of calculation with the AR and all KC rods fully inserted, AZ rods fully withdrawn. AZ worth is defined as a difference between the results of calculation with AR, KC rods inserted, AZ rods withdrawn and calculation with all rods inserted. AR, KC worth is defined as a sum of AR, KC-1, KC-2, KC-3 rods worth's. Each AR, KC group worth is calculated as difference between the results of calculation of the state with this group of rods fully withdrawn and fully inserted, other groups AR, KC rods are inserted to some medium position, AZ rods are withdrawn.

Conclusions

1. Selected LEU FA ensures practically the same operation mode as the current HEU FA (on the condition that LEU FA will be able to reach average burnup 55%). FA number in the core, the number of discharged and charged FA after reloading can be preserved. The cycle length can vary but IRT MEPhI has not a fixed cycle length.

2. The excess reactivity of the fresh LEU core is less by 3$ than excess reactivity of the fresh HEU core. The excess reactivity at BOC of the most of cycles is less by approximately 1÷3$ for LEU core than that for HEU core. But the reactivity loss rate due to burnup for LEU fuel is less by 40% and it compensates the difference in excess reactivity at BOC. The change of excess reactivity during the burnup process and after reloadings is smoother in LEU case and it is a positive effect.

3. CR worth for LEU core is less by ~10% but it has no impact on the requirements to shutdown margin. With current CR number the fresh and the operational cores with LEU and HEU fuel meet and exceed the shutdown margin criterion.

4. Fuel consumption in FA number per year (energy generation =4000 MWh) is less by 40% for LEU core. ^{235}U consumption (in gram of ^{235}U per year) is less by 8% for LEU core. So LEU fuel has the advantage in fuel consumption (in FA number per year), but the saving of ^{235}U is not significant.

5. Peak power density for fresh LEU core is less by 8% than for fresh HEU core. Peak power density for operational LEU core is less by ~20% than for operational HEU core. It is a positive effect but for IRT MEPhI it has no meaning because of low power and large margin to critical heat flux.

6. The parameters at the outlet of the irradiation beams HEC-4 and HEC-1 used for BNCT investigations were considered as main experiment performance indicators. Namely thermal and epithermal neutron fluxes at the beam outlet and "beam quality" were compared for LEU and HEU core. Fresh LEU core would have more neutron fluxes of all energies at the outlet of HEC-1 and HEC-4 than the operational HEU core. Operational LEU core would have approximately the same neutron fluxes at the outlet of HEC-1 and HEC-4 as the operational HEU core (the difference is approximately ±5). The negative changing of any considered parameter of HEC-1,4 is ~10% or less. There are some positive changes especially for fresh LEU core and during the first 5÷7 years of reactor operation with 12 FA in the LEU core. The amount of positive changing is ~10÷20%. So reactor experiment performance with LEU fuel does not decrease when compared with the operational (current) HEU core.

The performance results show that the conversion of the IRT MEPhI to the LEU IRT-3M U-Mo fuel will essentially maintain the performance of the reactor and its experiments.

9. Acknowledgements

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