IMPACT OF THE HEU/LEU CONVERSION ON EXPERIMENTAL FACILITIES

M. Marek, J. Kysela, J. Ernest, S. Flibor, V. Broz
Reactor Services Division
Nuclear Research Institute Rez, plc., Husinec 130, CZ-25068, Czech Republic

ABSTRACT

The LVR-15 reactor is a multipurpose research facility used for basic research on horizontal channels, material and corrosion studies in loops and irradiation rigs, and for the isotope production. A conversion from HEU (IRT-2M 36%, so far used) to LEU (IRT-3M 19.5%, IRT-4M 19.5%) is planned till 2010. The influence of the new type of fuel on the performance of the experimental facilities operated at the reactor has been studied. The comparison of the calculated neutron fluence rates and spectra using NODER operational code (3D nodal diffusion) and MCNP code for both the fresh and depleted cores was performed. Results of the analyses and future plans are presented in the article.

1. Introduction

Reactor LVR-15 is operated with the IRT-2M 36% fuel and has sufficient fuel reserve till 2010, nevertheless, first studies about the HEU to LEU conversion had started as was already reported [1]. There exist two prospective types of fuel assembly designs that could be potentially used - IRT-3M with U_9Mo-Al fuel meat and IRT-4M with UO_2-Al fuel meat.

According to the calculations the performance of the two FA designs have their positive as well as negative features. Generally both FA types will require a more challenging licensing of mixed cores during gradual conversion for the LVR-15 reactor environment as ^{235}U content and the geometry of FA currently loaded and LEU are substantially different.

Because of possible higher amount of ^{235}U in LEU IRT-3M FA the consumption of FAs per year at the nominal power of 10 MW would be correspondingly lower compared with the current IRT-2M (36%). Basic advantage of the IRT-3M design is that the FA could be used at 15 MW. Unfortunately, the IRT-3M FA with the U_9Mo-Al fuel meat has not been fully tested yet and will not be available in the course of next couple years.

The IRT-4M FA have been tested in Uzbekistan and used for the conversion of the Vrabec VR-1 reactor at Prague Technical University. The oxide fuel design is commercially available.

It seems that a perspective representative of advance LEU fuel witch is compatible with the LVR-15 core lattice and control rods system is a IRT-4M type (19.75%). Therefore, the paper is devoted to the study of the fuel performance with respect to the amount of ^{235}U in the fuel meat and influence on the experimental facilities operated at the reactor as well.

2. Reactor description.

The LVR-15 is a light water moderated and cooled tank reactor with combined water - beryllium reflector. It has a square core lattice (pitch 7.15cm) of 8x10 positions in the separation basket and downwards forced-water cooling. The generated heat is taken away to the water of Vltava river via 3 circuits. The number of the IRT-2M (36%) FAs in the core varies usually from 28 to 34. The system of control absorbers consists of 12 rods of an identical construction but a different function when the reactor is running. Three of them are predetermined as safety, the next one is the rod of the automatic control and the remainder is the shape rods. The absorbing rod consists of stainless steel tube filled with powdery B_4C. An aluminum follower is connected with at its lower end to shape the neutronic and thermo hydraulic conditions in the middle of FA.

The reactor provides now a wide range of possibilities to cover needs in the field of nuclear science and technology. It serves mainly as a radiation source for:

- Material testing at water loops and rigs
- Radio-pharmaceutical production
- Silicon doping
- Activation analyses with the pneumatic rabbit system
- Experiments in nuclear and applied physics via beam tubes
- Experiments for medical purposes (BNCT).

The reactor works at the power level 8-9MW continuously in the cycles of 18 to 25 days long with the several days' interruptions for technology maintenance.

3. Description of the potential LEU fuel

A perspective representative of advance available LEU fuel, which is compatible with the LVR-15 core lattice and control rods system, seems to be a IRT-4M fuel type (19.75%). This type of nuclear fuel for research reactors of Russia fabrication has been derived from known IRT-3M(36%) type.

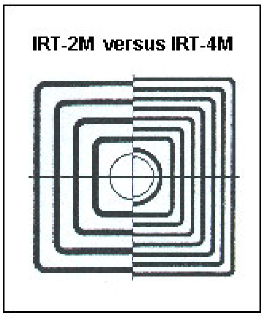

Fig.1. Cross section of the fuel part of IRT-2M and IRT-4M type

The essential differences between both IRT-3M(36%) and IRT-4M(19.75%) types consist in some internal dimensions because they are both based on UO_2-Al dispersion and have the same geometry and the number of fuel elements. The IRT-4M(19.75%) has bigger both the outside tube dimension and the fuel meat layer thickness (1.6 against 1.4 mm and 0.7 against 0.5mm). The differences result in the smaller water channel thickness at IRT-4M type (1.85 against 2.05 mm). A horizontal cross-section of the active part of both IRT-4M and the compared IRT-2M assemblies of reference together are shown in Fig.1. The dimension characteristics of the proposed IRT-4M type are essential for this study because the amount of uranium was taken as a variable parameter and all other neutronic properties are derived from this fact.

Weight characteristics calculated from geometry and total uranium content of all the fuel types to be compared are shown in the Table 1.

Table 1. Basic characteristics of the IRT-2M and IRT-4M FAs

Fuel Type	Enrichment [%w ^{235}U]	Weight of ^{235}U [g]		Weight of Uranium [g]		Weight of ^{238}U [g] (*)		Vol. of fuel meat[cm^3]		Uranium density in fuel meat [g/cm^3]	H/^{235}U (**)
		Reference	Control	Reference	Control	Reference	Control	Reference	Control		
IRT - 2M	36.13	230	198.2	636.6	548.4	406.6	350.3	259.4	223.5	2.45	247
IRT - 4M											
mod. 1	19.75	275	242.0	1392.4	1225.1	1117.4	983.1	552.1	485.7	2.52	164
mod. 2	19.75	300	263.9	1519.0	1336.4	1219.0	1072.5	552.1	485.7	2.75	150
mod. 3	19.75	325	285.9	1645.6	1447.8	1320.6	1161.8	552.1	485.7	2.98	139
mod. 4	19.75	350	307.9	1772.2	1559.2	1422.2	1251.2	552.1	485.7	3.21	129
mod. 5	19.75	375	329.9	1898.7	1670.5	1523.7	1340.6	552.1	485.7	3.44	120

(*) Active length of all FAs was taken 59cm
(**) A ratio of hydrogen and ^{235}U atoms in the elementary cell

4. Calculation methods

Calculations of the MEU and LEU reactor models were performed with NODER nodal-diffusion code [3] in 3D. A four-group burnup dependent cross-section set for the components taken place in the model was generated with WIMSD4m code using ENDF/B-V based data [5].

Energy grouped Monte Carlo criticality code OMEGA (Thermos, BNAB78 data) [1] was utilized to validate the calculation methodology and to spectrum calculation in fresh cores.

MCNP-C code [4] with ENDF/B-VI based data library was used for the calculation of the neutron spectra in models of irradiation facilities operated at the reactor.

5. Reactivity excess and control rods worth in the fresh core

Both the Monte Carlo OMEGA code and the diffusion NODER code were used for the calculations of the fresh cores loaded with only one type of fuel. The total reactivity excess and the total control rods worth refers to cold state of the core with 28 FAs reflected with water. The values of k_{eff} for the extreme positions of all absorber rods together and their resulting worth are shown in the Table 2. They show the well-known fact that effectiveness of absorber system generally descends with decreasing enrichment and increasing amount of ^{235}U in the core as a result of spectrum hardening.

It is evident the diffusion approach overestimates the reactivity up to 1.6% $\delta k/k$ compared to Monte Carlo neutron transport simulation. The differences, for the greater part, could have their origin in energy group cross section data prepared with the WIMSD4m code as indicated in overview of k_{inf} parameter of cylinderized elementary cells.

Table 2. k_{eff} for extreme positions of all absorber rods (up, down) and their worth

Fuel Type (modification)	k-eff [OMEGA *)] UP [rods]	k-eff [OMEGA *)] DOWN	k-eff (NODER) UP [rods]	k-eff (NODER) DOWN	Summary Rods Worth OMEGA	Summary Rods Worth NODER	δ(%) [NODER/OMEGA] UP [k-eff]	δ(%) [NODER/OMEGA] DOWN	δ(%) [NODER/OMEGA] WORTH
IRT - 2M	1.153	0.9650	1.168	0.975	16.87	16.92	1.30	1.05	0.30
IRT - 4M									
mod. 1	1.106	0.935	1.124	0.948	16.52	16.57	1.61	1.30	0.30
mod. 2	1.120	0.950	1.131	0.956	16.01	16.18	0.97	0.66	1.06
mod. 3	1.129	0.962	1.145	0.973	15.36	15.40	1.37	1.22	0.26
mod. 4	1.138	0.974	1.149	0.979	14.77	15.14	1.01	0.50	2.51

*) Standard deviation 0.001

The reactivity excess and control rods worth to reactivity excess ratio are summarized in the Table 3.

Table 3. Reactivity excess and control rods worth to reactivity excess ratio of examined FAs

Fuel type (Enrichment, content of ^{235}U in FA)	**Reactivity Excess in Fresh Core** REF [% Δk/k]	**Total Absorber Worth** SAW [% Δk/k]	**SAW/REF**
IRT-2M (36%, 230g)	13.25	16.87	**1.27**
IRT-4M (20%, 275g)	9.62	16.52	**1.72**
IRT-4M (20%, 300g)	10.74	16.01	**1.49**
IRT-4M (20%, 325g)	11.45	15.36	**1.34**
IRT-4M (20%, 350g)	12.11	14.77	**1.22**
IRT-4M (20%, 375g)	12.88	14.61	**1.13**

The values of reactivity excess and the worth indicate how the operational conditions and custom practice differ from those relating to the current FAs. Very low values warn that effectiveness of the absorber system of the real operational core configuration would not have to comply with facility operation safety criteria. From those points of view the ^{235}U content of 300 – 350 g/FA in the fuel meat is the suitable fuel modification usable at the LVR-15 reactor.

6. Fuel depletion

To compare the studied fuel types from point of view of their impact on cycle length and fuel effective exploitation a specific model of fuel depletion history was proposed [1]. It starts with fresh core reflected with water only. After the initial reactivity excess is spent the beryllium blocks are successively added to reflector positions in several steps to ensure the needed burnup reactivity excess in the next cycle. After the effect of Be-reflector is completed and reactivity excess drop to zero level again so called Transient Macrocycle (TMC) is finished. The first discharge of several FAs must be made and so called Main Macrocycle no.1 (MMC) starts. After several successive changes of FAs according to the optimized reshuffling schema the last group of FAs from the origin (starting) set is discharged and the last cycle in framework of 1.MMC can start. After the reactivity excess in this cycle is spent the 1.MMC is finished and the second one could be started.

The suppressing of local power extremes and the limit of maximum reactivity excess are taken into account at the reshuffling schema design so they cannot be general for all fuel types. The zero level of reactivity excess was taken at 2.5% δk/k. The value covers all reactivity losses given by uncertainty and simplifications in the fuel depletion model. In addition, the model does not take rods moving into consideration. The absorbers are always fully withdrawn so only the followers are inside the core. Some of the results presented in tables below are expressed in term of standard operational cycle length (SOCL), which means 21 full power days at 10MW.

Table 4. Fuel consumption and cycle length in equilibrium core for the compared fuel types.

Fuel type (Enrichment , content of ^{235}U in FA)	**Length of 1.MMC** [MWd] (**)	**Number of SOCL in 1.MMC** (***)	**Reached Burnup** **Average** [% wt. ^{235}U]	**Max.**	**Average consumption of FAs per 1 SOCL**
IRT-2M (36%,230g	2 577	12.3	56.3	60.7	2.28
IRT-4M (20%,275g)	3 063	14.6	53.4	58.4	1.92
IRT-4M (20%,300g)	3 419	16.3	54.6	59.5	1.72
IRT-4M (20%,325g)	3 920	18.7	57.5	62.6	1.51
IRT-4M (20%,350g	4 267	20.3	58.1	63.4	1.38

**) Main MacroCycle (MMC) is a summary of operation cycles between two complete changes of the fuel.
***) Standard Operational Cycle Length (SOCL) is represented by 210 MWd of irradiated power.

7. Spectrum change assessment

The LVR-15 reactor is used as a multipurpose facility for irradiation of different types. Material samples prevalently require fast neutrons with low and controlled portion of thermal neutrons. Also the epithermal neutron beam of the BNCT facility installed in a horizontal channel needs a source of epithermal and fast neutrons. On the other hand the radiopharmaceutics production, silicon doping, and the other horizontal channels use thermal neutrons.

To get the essential information about the spectrum changes depending on what fuel is loaded, a detailed calculation with MCNP were performed. Only one representative LEU fuel (IRT-4M with 325g of ^{235}U) was chosen for comparison with the current FA type, it means that the effect of different content of Uranium of the same enrichment was not the subject of the study.

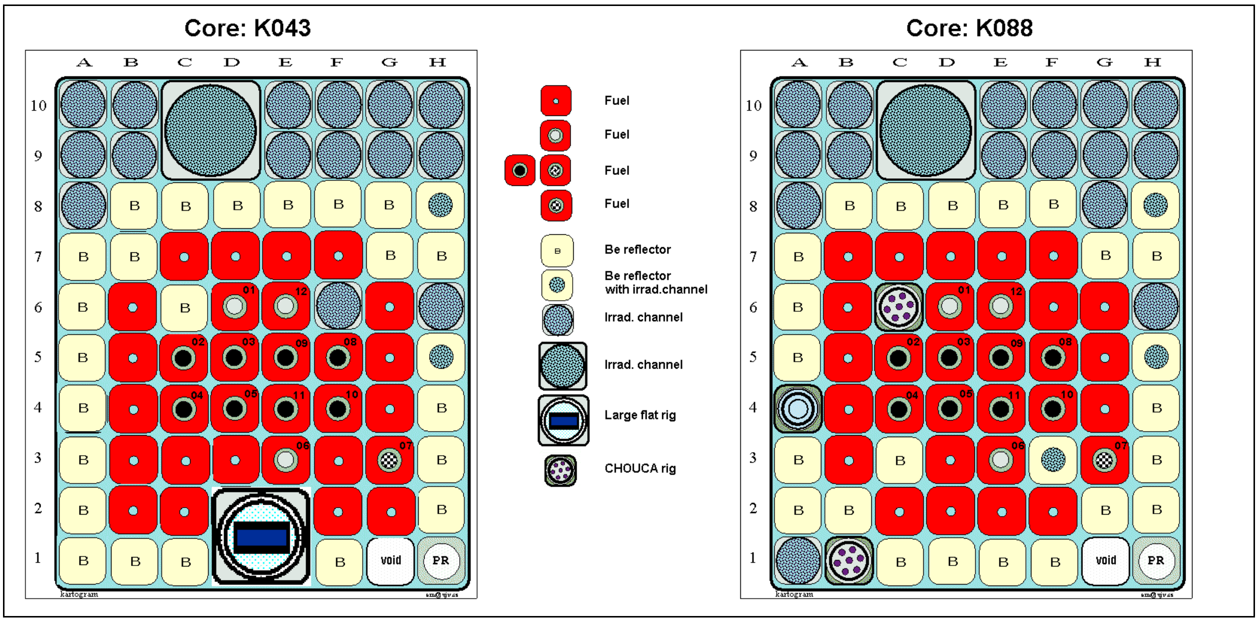

Fig.2. Cross section of the reactor core describing the positions of irradiation facilities in two core configurations – K043 and K088 – used for the evaluation of the neutron spectra. Large flat rig in C1-D2 of K043, CHOUCA rig in C6 of K088, and a capsule for Ir irradiation in F3 of K088.

The calculations were performed with the MCNP code in the source mode with the 3D source term determined by the NODER code. Neutron spectra were evaluated in the BUGLE 47 group structure [6]. Two different core configurations (K043 and K088) with IRT-2M fuel that were already operated during irradiation experiments were analyzed (Fig.2). Both the cores consisted of 30 FAs – eighteen 4-tube assemblies plus twelve 3-tube assemblies – surrounded by the beryllium reflector. In the case when the calculation with LEU core was performed in the above-described configurations, each 4-tube IRT-2M FA was replaced by an 8-tube IRT-4M FA and 3-tube IRT-2M by a 6-tube IRT-4M FA.

Neutron spectra were calculated for the fresh core and 34% burned fuel that characterize an average realistic state of the core. The fresh core calculation took into account only steady state Xe and Sm. The burnt core fuel composition was determined with the NODER code.

Three models of irradiation facilities positioned in different positions with respect to the reactor core were subjected to the analyses (Fig. 2) :

- Core configuration K043 with a model of a large flat rig in four cells C1-C2-D1-D2 at the reactor core edge. The rig was used for the irradiation of stainless steel material samples.
- Core configuration K088 with a model of the CHOUCA cylindrical rig in one cell. The rig was also filled with material samples that required fast neutron flux as high as possible. Therefore, the rig was in the C6 cell in the core.
- Core configuration K088 with a model of an irradiation capsule for the production of Ir. The capsule was in the F3 cell in the core.

Resulted neutron spectra per one starting neutron for the chosen positions are presented in the Fig.3-5. Estimated STD of total neutron flux was less then 1% in each case, the error of the group fluxes were less then 10% except for the neutron energies above 10MeV.

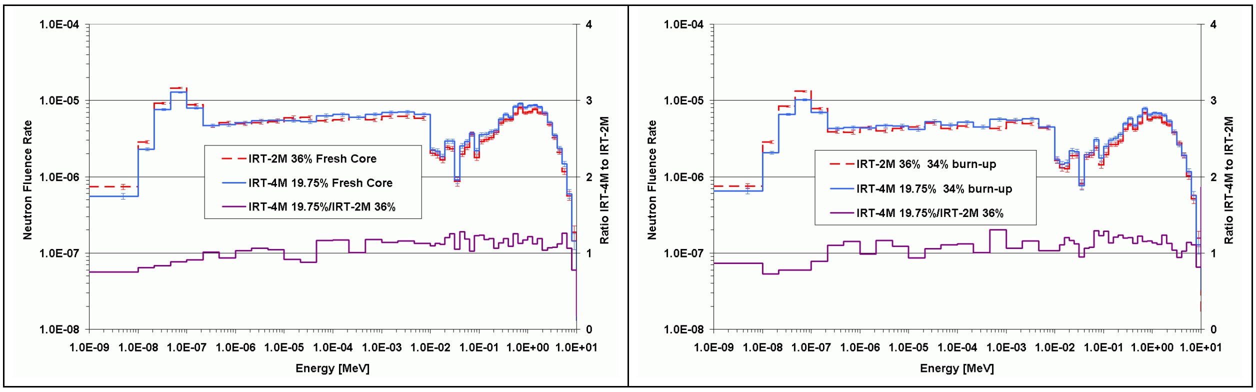

Fig.3. Neutron spectra averaged over the stainless steel samples irradiated in the large flat rig positioned in the C1-D2 cells

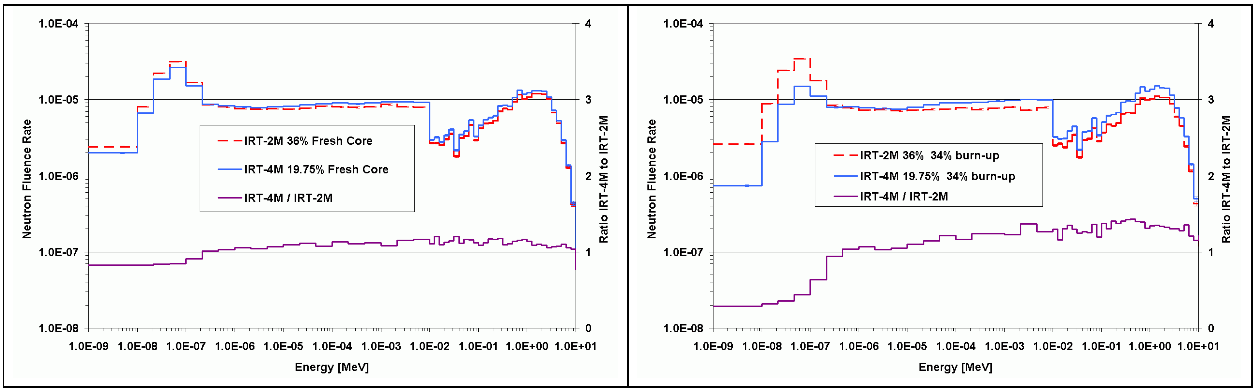

Fig.4. Neutron spectra averaged over the stainless steel samples irradiated in the CHOUCA rig in the C6 cell

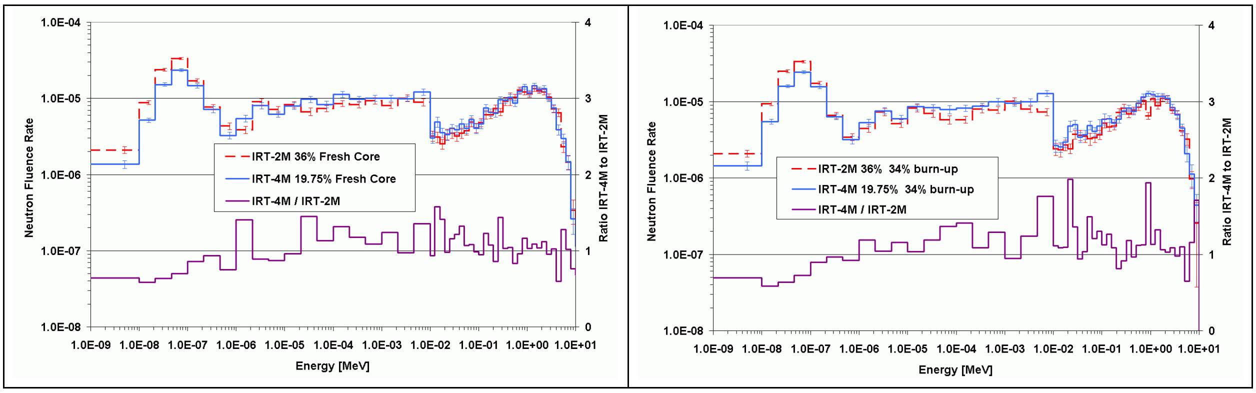

Fig.5. Neutron spectra averaged over Ir samples irradiated in a graphite capsule in the F3 cell

The calculation confirmed the known fact that only in-core positions give remarkable shift in spectrum (group flux to total flux) towards to higher energies when enrichment decreases. In the reactor core (position C6) the increase in the fast neutron fluxes above 0.1, 0.5, and 1 MeV varies from 9% to 11 % for the fresh core and from 32% to 35% for the burned fuel. The thermal neutron flux decreases for the same conditions from –13% in the fresh core case to –52% as for the burned core. The same is effective also for the core periphery (large flat rig in C1-D2 cells) but the differences are not so distant and large – increase in fast fluxes ~ -14%, decrease in thermal fluxes ~15%.

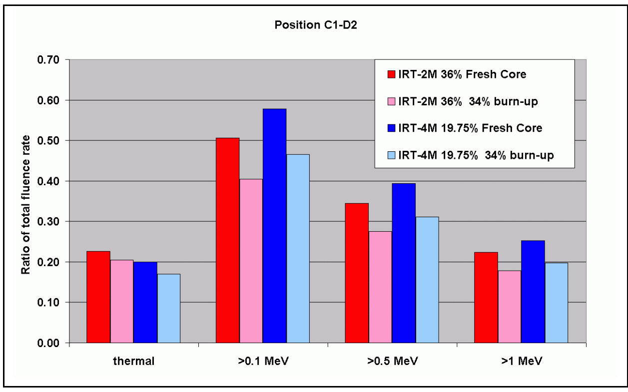

Fig.6. Neutron groups averaged over the stainless steel samples irradiated in the large flat rig positioned in the C1-D2 cells

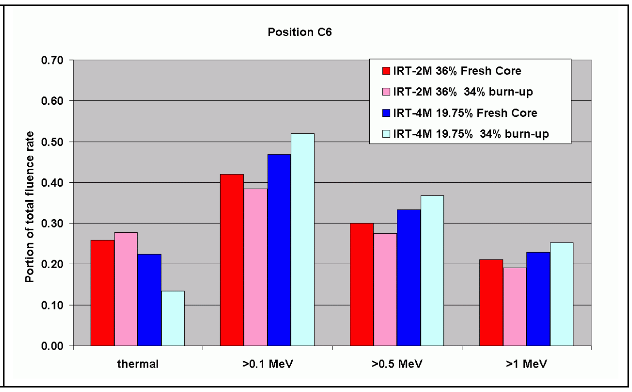

Fig.7. Neutron groups averaged over the stainless steel samples irradiated in the CHOUCA rig in the C6 cell

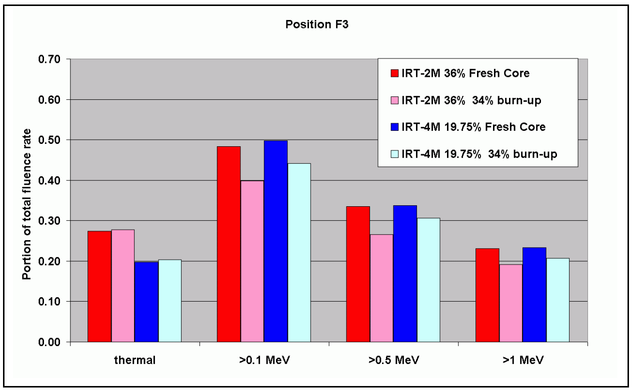

Fig.8. Neutron groups averaged over Ir samples irradiated in a graphite capsule in the F3 cell

On the other hand, the LEU will make irradiation conditions worse as far as the production of medical and industrial sources as Ir, Co, Mo is concerned. The analyzed graphite capsule with Ir samples in the position F3, which is usually used for this purpose, shows 28% less thermal neutron flux then for the current MEU core under comparable conditions.

8. Conclusions

One of the safety criteria of the LVR-15 operation binds the summary rods worth with total reactivity excess. The value of their ratio is a key parameter deciding if the given core configuration can be utilized. Based on the performed calculations and operational experience this criterion could be fulfilled only up to ^{235}Ucontent of about 325g in IRT-4M (8tb) fuel assembly. That refers to partial uranium density in fuel meat of about 3.0g/cm3.

On the other hand, the exploitation of the proposed LEU fuel could reduce the annual consumption of FAs number more than 30% comparing with the IRT-2M (36%) currently used.

Regarding the prevalence of commercially important irradiations based on material studies that are realized in the core region and its close periphery, the increase of the fast neutron fluxes is a beneficial consequence of the conversion. At the same time, the decrease of the thermal fluxes at in-core positions up to 10% is not substantial.

A complex conversion study of the LVR-15 reactor, which could cover all needed neutronic and thermo-hydraulic aspects with example of the loading patterns of transition from MEU to LEU core via mixed cores should be provided as a next step.

The impact on the effectiveness of reactor operation is, however, influenced also by other aspects as guaranteed burnup, residual amount of ^{235}U in the used FAs, price of ^{235}U in view of the price of the whole FA and others. Decision what type of fuel is the most appropriate for the LVR-15 will be based on a complex analysis of physical and economical aspects of the conversion that will be performed in the near future.

9. Acknowledgments

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10. References

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