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COMPARATIVE STUDY OF THE WWR-K REACTOR USING U-MO LOW ENRICHED FUEL PIN- AND TUBE-TYPE ASSEMBLIES

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ABSTRACT

Several options of the WWR-K reactor initial and stationary core configurations are considered with the tube/pin-type fuel assemblies on a base of the high-density low-enriched U-Mo fuel. Outcomes of the comparative study of the neutron characteristics and the WWR-K reactor cycle periods are presented.

1. Introduction

Forthcoming conversion of the VVR-K reactor core implies conservation of the reactor vessel, support grid and other devices located inside the vessel. Thus, to a certain extent, chose of the new fuel assembly is determined by the grid step (68.3мм).

In the frame of Russian RERTR the fuel compositions are developed on a base of uranium dioxide of the uranium mass density up to 3 g/cm^3 and on a base of uranium-molybdenum alloy of the uranium mass density up to 5.6 g/cm^3.

We have studied an opportunity to use low-enriched uranium-dioxide fuel with the uranium density 2.8 g/cm^3, and it has been shown that at present the thin-walled tube fuel assemblies (FA) with well-developed heat removal surface are the best candidates for the WWR-K reactor low-enriched core. Similar fuel assemblies with the meat/clad/inter-element gap 0.7/0.45/2.1 mm thick are fabricated at the Novosibirsk Chemical Concentrate Plant (NCCP).

In the closest future high-density uranium-molybdenum fuel will be produced, making it possible to implement the uranium density up to 8 g/cm^3.

This work is devoted to the reactor core with uranium-molybdenum fuel having the uranium density 5 g/cm^3. Enrichment in U-235 is 19.7%.

The fuel assemblies with traditional tube fuel elements (FE) and with pin FE have been considered. Advantage of the latter is enhanced uranium content.

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2. FA characteristics

Two types of LEU FA have been used in this study: (a) thin-walled FA composed of tubular fuel elements (FE) and (b) FA composed of pin FEs. Both types have FA-1 and FA-2. The CPS control rods are placed inside FA-2. If required, FA-2 can also be used to locate small-diameter irradiation channels.

Tubular FA-1 consists of eight FEs, and FA-2 consists of four FEs. The FE and FA characteristics are given in Tables 1 and 2. For comparison, figure 2 shows the views of the present HEU FAs.

A. FA with tube fuel elements

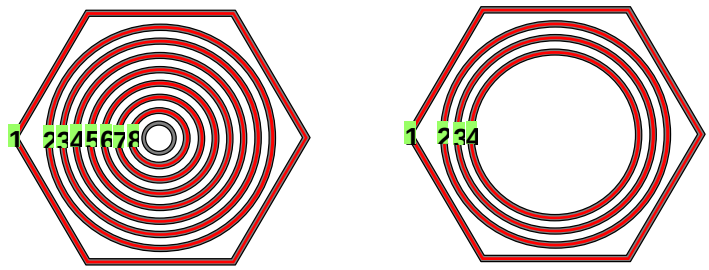

Fig.1. The eight-tube (FA-1) and four-tube (FA-2) fuel assemblies

Table 1 – Geometrical parameters of the eight/four-tube FA

FA-1				FA-2		
Structural element	Radius, mm	Thick, mm	U-235 mass, g per FE	Structural element	Radius, mm	Thick, mm
FE-1	66.1*	1.4	66.2	FE-1	66.1*	1.4
FE-2	59.1	1.4	53.6	FE-2	59.1	1.4
FE-3	52.1	1.4	47.1	FE-3	52.1	1.4
FE-4	45.1	1.4	40.6	FE-4	45.1	1.4
FE-5	38.1	1.4	34.1	rod channel	38.1-35.6	—
FE-6	31.1	1.4	27.6	rod	33.6-31.6	—
FE-7	24.1	1.4	21.1			
FE-8	8.6	1.4	14.6			
Struct. tube	9.3	1.0	-			

* face-to-face size

Table 2 - Some quantitative characteristics of the eight/four tube FA

Parameter	FA-1	FA-2
U-235 mass, g	304.7	207.4
$S_{heat\ removal}$, cm^2	12375	8360
FE / meat /clad, mm	1.4 /0.5 / 0.45	1.4 /0.5 / 0.45
Inter-FE gap, mm	2.1	2.1

B. Fuel assembly with pin elements

The FE geometrical characteristics are presented in table 3, and its design is shown in figure 2.

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Table 3 - Pin fuel element geometrical characteristics

$d_{оп}$, mm	Δ, mm	h, mm	A, mm	δ, mm	S_{FE}, mm ^{2}	P_{FE}, mm	S_{meat}, mm^2	H, mm
4.85	0.4±0.1	0.6	2.86	0.4	9.0	15.59	4.68	320

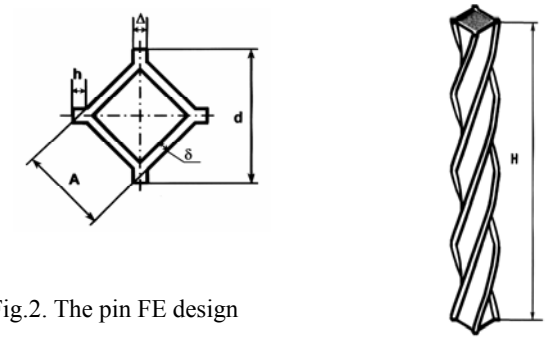
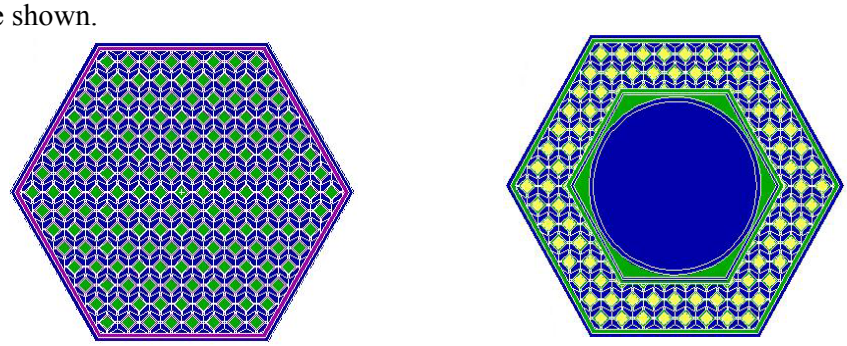


Fig.2. The pin FE design

Here S_{FE} is the FE area, h is the ребра height, P_{FE} is the FE perimeter, A is the square side, S_{meat} is the meat area; δ is the clad thickness, $d_{оп}$ –is the описанный diameter, H is шаг закрутки ребра; Δ is the ребра width. The FE active section height is 600 mm.

In Fig.3 sectional views of the FA-1 and FA-2 models simulated with the computer code MCNP are shown.



FA-1 **FA-2**

Fig.3. FA computational model

In table 4 some quantitative characteristics of the pin FA are given.

Table 4 – Some quantitative characteristics of the pin FA

Parameter	FA-1	FA-2
Number of pins in FA	169	78
The uranium-235 mass, g	507	234
$S_{heat\ removal}$, m^2	1.58	1.01
Cover outer face-to-face size, mm	66.5	66.5

3. Reactor Core

The higher uranium density, compared to the VVR-TS-type FA, which is used in the reactor at present, gives a chance to make the reactor core more compact. Calculations have shown that absence of the aluminum followers, which are used in the CPS control rods now, doesn't result in formation of "hot" points in the core. So a CPS control rod can be placed in any cell of the core. Besides, three central irradiation channels can be arranged, where the neutron flux density values will be high and, practically, the same.

In our calculations initially the core is loaded with twelve FA-1 and twelve FA-2 (fresh core, Fig. 4A), for both tube and pin fuel assemblies. Ten of twelve FA-2 are used to arrange CPS control rods, and two FA-2 (in the cells 5-4 and 8-6 in Fig 4) can be used as small-diameter irradiation channels. The rest cells of the core are occupied by water displacers, which form the side water reflector. As the fuel burns out, the water reflector is replaced by the beryllium one; and stationary core contains 24 beryllium pieces (Fig. 4B).

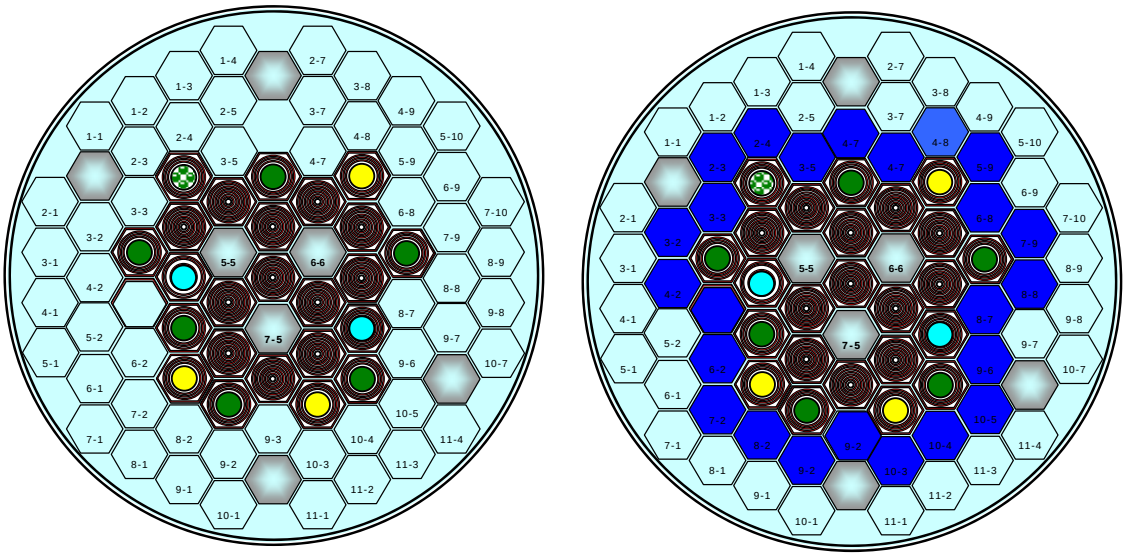


Fig.4. (A) Initial (fresh) core and (S) stationary (burnt) core with 24 beryllium pieces

4. Calculation Results

The following neutron characteristics have been calculated: the effective neutron multiplication factor (k_{eff}), the thermal (E_n<0.4 eV) and fast (E_n>1.15 MeV) neutron flux density values in irradiation channels, control rod efficiencies, the power generated in the core, the reactor operation cycle and the fuel burnup level. The computer codes MCNP-4B [1] and MCU-REA [2] have been used (The first one – only for calculations with fresh core, but in purely heterogeneous model of FE, whereas the second code was used for calculation of burnup and operation cycle periods but in homogeneous representation. This is the reason of differences in results obtained with two codes, and that's why the calculated operation cycle period and burnup values are overestimated). The calculated core parameters are given in Table 5.

(For tubular FA stationary xenon poisoning is (3.2 ÷ 3.4)% Δk/k. Reactivity loss per day, on average, comprises (0.030 ÷0.035)% Δk/k).

Table 5 – The calculated core parameters (MCNP-4B/MCU-REA)

Parameter	Tube FE (version 1)		Pin FE (version 2)	
	A	B	A	B
k_{eff}	1.0977	1.1518	1.096	1.1398
ρ_{BOC}, ρ_{EOC} , %Δk/k	8.9 0.31	13.2 0.31	9.0 0.8	12.3 1.5
ΣKO+AP, %Δ k/k	10.4/10.3	12.7/12.6	9.5	11.3
Σ AZ, %Δ k/k	4.0/3.7	4.3/4.3	3.6	4.0
Operation cycle, day	100	250	225	350
Max. burnup, %	17	48	16	42

Calculated values of the thermal/fast neutron flux density in the central (cells 5-5, 6-6 and 7-7), peripheral (cell 10-2) irradiation channels and inside FA-2 (in the cells 5-4 and 8-6) for fresh and burn (with Be) cores are given in Table 6.

Table 6 - Neutron flux density values in the core irradiation channels

Version	Tube FE (version 1)		Pin FE (version 2)	
	A	B	A	B
Cell #	Thermal neutron flux density, cm^{-2} s^{-1}			
5-5	(2.2±0.1)E+14	(2.2±0.1)E+14	(2.2±0.1)E+14	(2.2±0.1)E+14
6-6	(2.2±0.1)E+14	(2.2±0.1)E+14	(2.3±0.1)E+14	(2.3±0.1)E+14
7-5	(2.3±0.1)E+14	(2.2±0.1)E+14	(2.4±0.1)E+14	(2.3±0.1)E+14
5-4	(1.2±0.1)E+14	(1.3±0.1)E+14	(1.2±0.1)E+14	(1.2±0.1)E+14
8-6	(1.1±0.2)E+14	(1.7±0.1)E+14	(1.0±0.1)E+14	(1.2±0.1)E+14
10-2	(5.0±0.6)E+13	(8.1±0.2E)E+13	(5.8±0.2E)+13	(7.3±0.2E)+13
Cell #	Fast neutron flux density, cm^{-2} s^{-1}			
5-5	(4.2±0.1)E+13	**(3.9±0.1)E+13**	(4.4±0.1)E+13	**(4.5±0.1)E+13**
6-5	(4.4±0.1)E+13	(4.0±0.1)E+13	(4.4±0.1)E+13	(4.7±0.1)E+13
7-5	(4.3±0.1)E+13	(4.5±0.4)E+13	(4.4±0.1)E+13	(4.7±0.4)E+13
5-4	(4.5±0.1)E+12	(4.6±0.4)E+13	(4.5±0.1)E+12	(4.8±0.4)E+13
8-6	(4.2±0.4)E+13	(4.4±0.4)E+13	(4.3±0.4)E+13	(5.0±0.4)E+13
10-2	(3.8±0.5)E+12	(8.1±0.4)E+12	(4.2±0.5)E+12	(4.8±0.4)E+12

For the core with fresh tube fuel assemblies thermal and hydraulic calculation has been performed (code ACTRA [3]) for the 6-MW reactor power. Preliminary, the values of the power and power density generated in fuel assemblies of the tubular-FA core have been calculated with the control rods inserted to critical (see Table 7) by means of the MCNP code. For comparison, the appropriate data referring to the same core with pin FAs are shown too.

Table 7 demonstrates that the central FA-1 (cell 6-5) and FA-2 in the cells 5-4 and 8-6 are the most “hot”. As an input for thermophysical calculation, distributions of the generated power in fuel elements of the hottest fuel assembly (central cell 6-5) and over the height of the hottest (outer) fuel element in the face adjacent to the “wet” channel (in the cell 7-5) have been calculated; the face is divided into 12 sections, each 5-cm high; the control rods KO and AP are inserted to critical (see figures 5 and 6).

Table 7 – Power generated in the core with tube/pin FA

Core cell	P, kW	P, kW/cm^3	Core cell	P, kW	P, kW/cm^3
	Tube/pin FA	Tube/pin FA		Tube/pin FA	Tube/pin FA
3-4 (AP)	144.9/125.2	688.8/618.5	6-7	244.6/262.8	**799.8/598.9**
4-3 (1KO)	152.3/133.7	723.6/660.2	7-3 (1AZ)	170.0/160.3	807.6/791.7
4-4	222.6/226.9	727.9/517.1	7-4	246.2/248.0	805.0/565.3
4-5	239.6/242.0	783.5/551.6	7-6	356.2/380.9	**1164.6**/868.3
4-6 (1KO)	181.5/156.9	862.5/775.1	7-7	280.3/298.5	916.6/680.5
5-4 (FA-2)	244.8/224.3	**1163.2/1107.6**	7-8 (2KO)	167.2/152.5	794.6/753.1
5-6	321.5/337.9	1051.2/770.2	8-3 (3KO)	157.0/135.3	746.0/668.2
5-7	259.7/275.2	849.3/627.3	8-4	261.7/272.8	855.7/621.7
5-8 (3AZ)	196.9/208.9	935.8/1031.6	8-5	266.7/279.7	872.0/637.7
6-3 (2KO)	176.5/149.5	838.9/738.1	8-6 (FA-2)	243.6/**231.9**	**1157.4/1145.2**
6-4	335.6/351.7	1097.3/801.7	**9-4 (2AZ)**	196.0/195.2	931.3/964.2
6-5	**412.8/443.5**	**1349.7/1010.9**	9-5 (3KO)	162.0/146.5	769.6/723.3

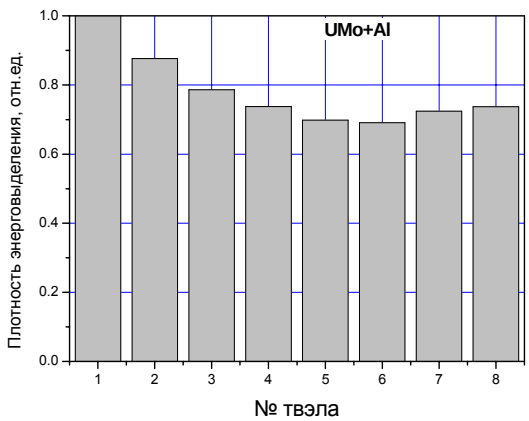


Fig. 5a. Distribution of the generated power in FE

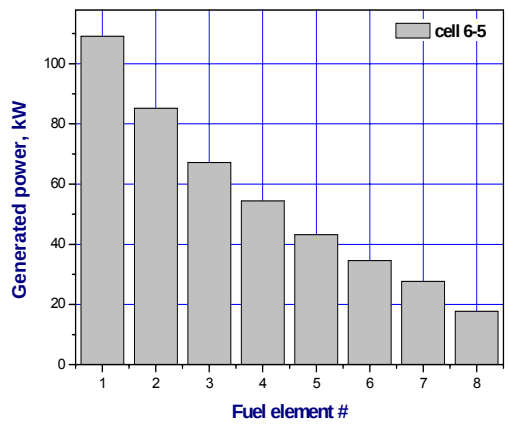


Fig. 5b. Distribution of the power density in FE

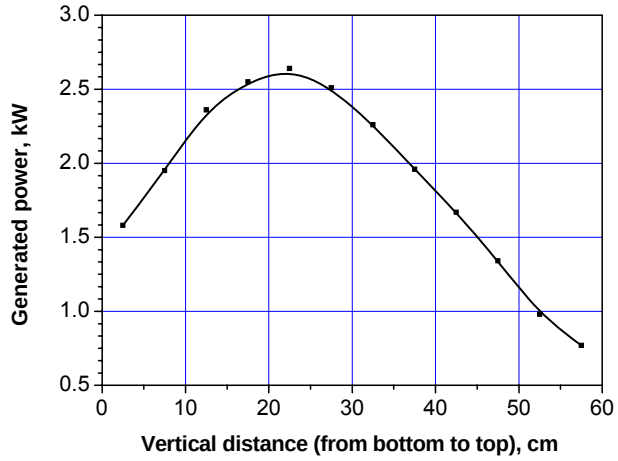


Fig.6 Generated Power Versus FA Height

The maximum value of the generated power density in the outer fuel element meat, along the radius directed from cell 6-5 to cell 7-5, equals $2.66 \cdot 10^6$ kW/m^3 (at the 22.5-cm height).

As the input data, the following parameters are taken:

- Water temperature in the FA inlet: 35·˚C.
- Water pressure in the FA inlet: 1.35 bar.
- Pressure drop in the core: 0.15 bar (~1.5 m of water column) and 0.20 bar (~2.0 m of water column), with the 20% of this value – drop at the inlet, 60% – drop in the FA gaps, and 20% – drop at the outlet.
- Water velocity in gaps for two considered values of the pressure drop in the core: 1.88 and 2.17 m/s respectively.
- The reactor power: 6MW.

Results of thermal calculation for two values of the pressure drop in the core are given in Table 8.

Table 8 – Results of thermal calculation for the "hottest" tubular FA-1

Parameter	Pressure drop, bar	
	0.15	0.20
The maximum density of the generated power in the FE meat, MW/m^3	2660	2660
Water temperature in the core inlet, °C	35	35
The maximum thermal flux (from the FE outer/inner surface), kW/m^2	708/618	705/622
The FA surface maximum temperature, °C	98	92
The ONB temperature on the FE surface (by Forster-Greif), °C	125	125
Minimum ONB margin	1.43	1.58

Table 8 demonstrates that the ONB margin on the FE surface is the minimum accessible value for the 0.15-bar pressure drop, i.e., pressure drop in the core should be not less than 0.15 bar (1.5 m of water column).

The calculated variation in the FE wall temperature is shown in Fig. 7. It's clear that the main requirement – absence of surface boiling – is met.

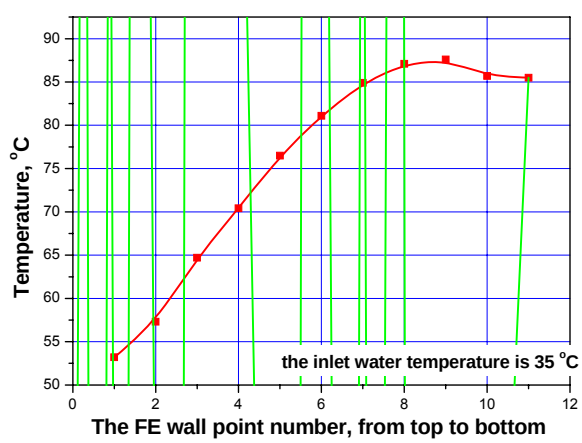


Fig. 7. The FE wall temperature versus height

For the core with pin FA heat-hydraulic calculation has been carried out for fresh core “18FA-1+10FA-2” and the same pin FA but with uranium-dioxide fuel. For both cases data of preliminary calculation are available, such as the generated power density distribution for the hottest fuel assembly. in cell 6-5 (see table 8) and hottest fuel element (fig. 7) in the case when control rods are inserted to critical, and they are, practically, coincide. [Calculation has been made for fresh core]. In Table 8 distribution of the generated power density in 169 fuel elements of the hottest fuel assembly located in cell 6-5 is shown.

Table 8 – Distribution of the generated power density (kW/cm^3)
in 169 fuel elements of the hottest fuel assembly located in cell 6-5

	1	2	3	4	5	6	7	8	9	10	11	12	13	14	15
1	1.240	1.283	**1.314**	**1.329**	**1.333**	**1.314**	1.294	1.252							
2	1.013	1.025	1.036	1.048	1.059	1.055	1.048	1.040	1.029						
3	0.901	0.882	0.882	0.890	0.894	0.894	0.898	0.898	0.894	0.917					
4	0.847	0.817	0.801	0.790	0.797	0.794	0.794	0.797	0.809	0.828	0.867				
5	0.844	0.790	0.763	0.740	0.736	0.736	0.732	0.740	0.751	0.770	0.801	0.863			
6	0.886	0.805	0.755	0.728	0.709	0.697	0.697	0.701	0.716	0.740	0.770	0.824	0.917		
7	0.998	0.871	0.790	0.743	0.705	0.690	0.670	0.678	0.690	0.713	0.751	0.809	0.894	1.021	
8	1.213	1.005	0.867	0.782	0.732	0.693	0.674	0.670	0.674	0.697	0.740	0.797	0.890	1.025	1.248
9	1.256	1.017	0.871	0.778	0.724	0.686	0.670	0.674	0.693	0.732	0.794	0.886	1.040	1.283	
10	1.279	1.025	0.874	0.778	0.716	0.693	0.686	0.693	0.732	0.794	0.890	1.048	**1.310**		
11	**1.302**	1.032	0.874	0.778	0.728	0.701	0.709	0.732	0.797	0.890	1.048	**1.325**			
12	1.290	1.021	0.874	0.778	0.736	0.728	0.743	0.790	0.882	1.052	**1.321**				
13	1.271	1.013	0.863	0.794	0.759	0.759	0.797	0.878	1.032	**1.306**					
14	1.244	1.002	0.878	0.805	0.794	0.813	0.882	1.017	1.279						
15	1.210	0.998	0.890	0.844	0.844	0.905	1.005	1.225							

Red lines show to the faces adjacent to irradiation channels (cells 5-5, 6-6, 7-5), and blue lines show to fuel assemblies (cells 5-6, 6-4, 7-6).

Table 8 demonstrates that the highest values of the generated power density correspond to the FA faces adjacent to ‘wet” channels.

In fig.8 the height distribution of the generated power density in the hottest FE (element (1, 5) from Table 8) of the FE located in the central cell 6-5 of the core.

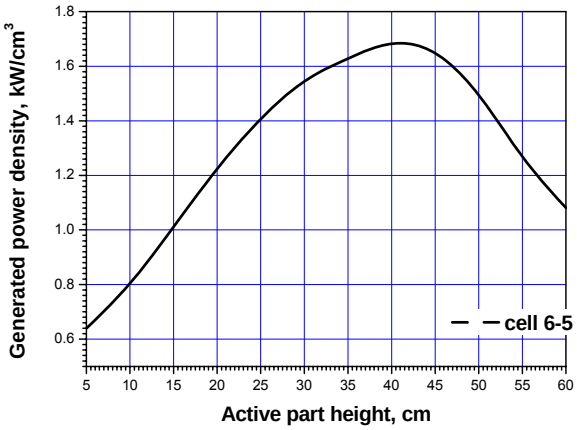


Fig. 8. The generated power density versus height in the hottest pin fuel element

The calculated peak power density value equals $1.7 \cdot 10^6$ kW/m^3 at the 20-cm height both for U-Mo fuel and for UO_2 (compare with $2.66 \cdot 10^6$ kW/m^3 for tube FA).

These data along with results of similar steady-state thermal-hydraulics analysis make it possible to hope that the analyzed core meets the safety criterion for margin to ONB. Nevertheless, corresponding calculation is needed.

Conclusions

- Usage of uranium-molybdenum fuel of the density 5g/cm^3 in the VVR-K reactor core makes it very compact.

- Analysis of the calculation results shows that the thermal flux density in irradiation channels is insignificantly slower with UMo-Al composition, compared to UO_2-Al. The reduction is explained, firstly, by an increase in the U-238 content in UMo alloy and, secondly, by peculiarities in behavior of the cross-section of the Mo-96 isotope in the resonance energy range.

- Usage of beryllium reflector leads to increases in the fluxes in peripheral channels, in reactor operation cycle and in fuel burnup level, making it possible to save fuel.

- An opportunity to arrange small-diameter irradiation channels (inside FA-2) directly in the core occurs – to organize fast-flux high-density irradiation, as well as a large number of large-diameter channels (52 mm) in the reflector.

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