REPORT

ON CONTRACTUAL SERVICE AGREEMENT

"Concept of a lifetime-core particle-bedded 300Mwe boiling water reactor (BWR-PB)"

Contract: 302-12.50.01-RUS-13094 B5-RUS-32948

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Moscow, 2005

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Introduction

The work is carried out on IAEA contract 302-I2.50.01-RUS-13093 B5-RUS-32970

In this paper the results of modification of VKR-MT vessel boiling water reactor with micro fuel elements are presented [1].

The targets of this work are the follows:

- optimization of core parameters for fuel cycle characteristics improvement and core lifetime extension;

- micro fuel elements under operation conditions reliability rising;

-optimization of reactor design for core lifetime prolongation without refueling within 25-30 years;

-development of refueling (reload) technique without reactor lid opening which is proposed after core operation during 25-30 years with special refueling (reload) tank using.

In general noted actions promote to proliferation resistance of nuclear materials. They are provided with the following engineering solutions:

- Internal reservoir is intended for fresh micro fuel elements;
- Internal repository is intended for spent micro fuel elements;
- Transport equipment of micro fuel elements inside the reactor vessel without power reduction and vessel lid opening is intended for realization of gradual core refueling (reloads) in accordance with the process of uranium depletion;
- Container and equipment for micro fuel elements reloading after reactor operation during 25-30 years without vessel lid opening.

VKR-MT reactor design [1], fuel assembly and micro fuel elements design, equipment for gradual core refueling (reloads) without power reduction and vessel lid opening have been modified for realization of the noted actions.

1. Reactor design description

A design scheme of the VKR-MT power unit is shown in Fig 1.1. The distinctive features of reactor design are:

- internal reservoir for fresh micro fuel elements in upper part of reactor vessel under the vessel lid;

- internal repository for spent micro fuel elements mounted upon the vessel bottom;

- equipment for micro fuel elements transition inside the reactor vessel;

- ball transport pipelines for loading of fresh and discharge (outloading) of spent micro fuel elements.

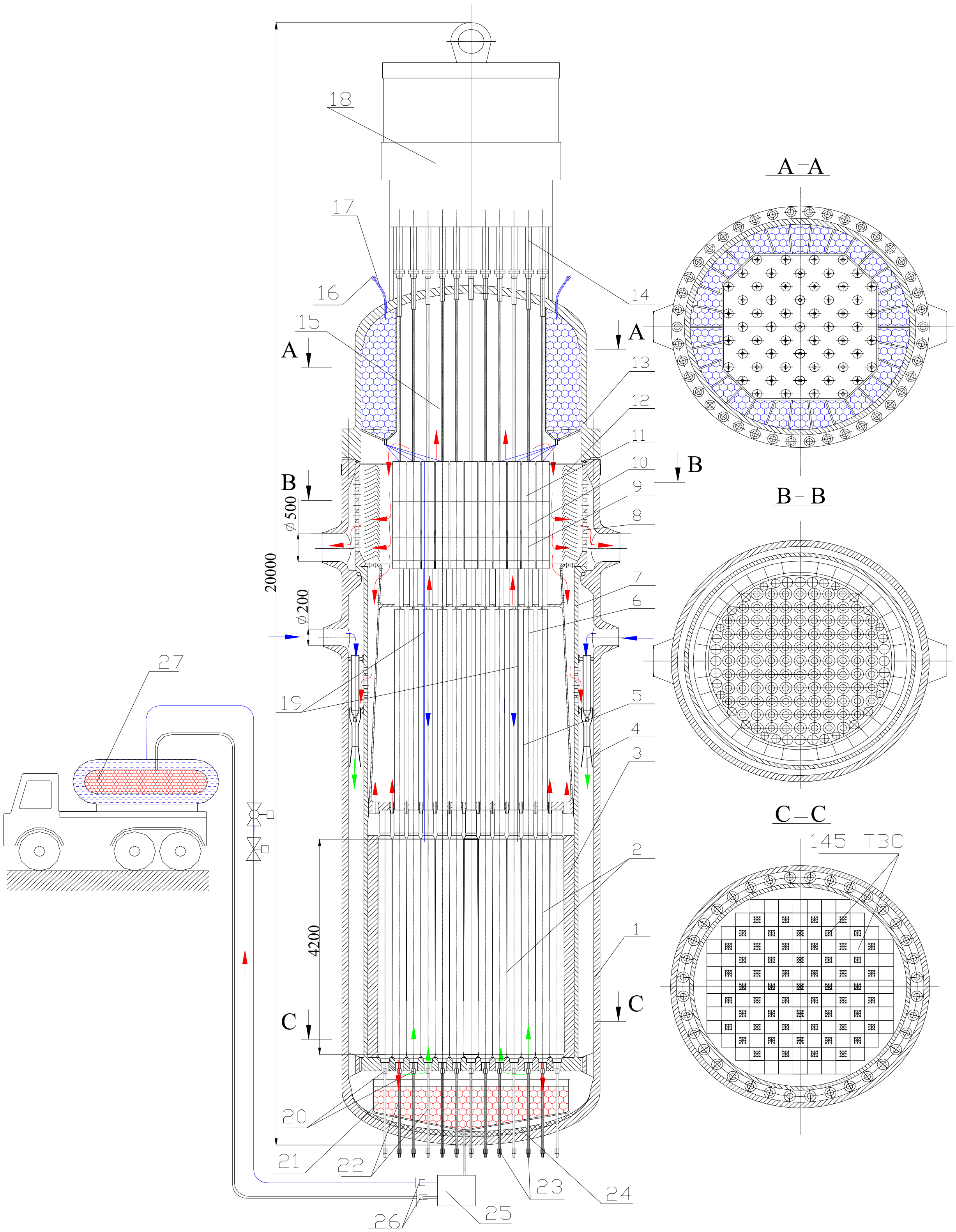

Fig.1.1. Reactor design scheme

1-reactor vessel, 2-fuel assembly, 3-enclosure, 4- jet pump, 5- guiding tube of a cluster, 6- protective tube of the bar of a control rod drive, 7- internal metallic shaft, 8- block of protective tubes, 9Anti-holdup device, 10- first-stage separator, 11- second-stage separator, 12-re-hydrator, 13- vessel lid, 14- control rod drive, 15- internal reservoir for fresh micro fuel, 16-weld plug, 17- branch pipe for loading of fresh micro fuel elements, 18 – upper block, 19- Pipelines for ball transport within the block of protective pipes, 20- ball-stop armature, 21- Internal repository for spent micro fuel elements, 22- drive rod of ball-stop armature, 23- electric drive of rod 24- biological shielding, 25- device for spent micro fuel elements discharge, 26- weld plugs, 27- transport equipment.

Reactor design presented in Fig.1.1 includes the reactor vessel 1, reactor core 2 composed of fuel assemblies, enclosure 3, jet pumps 4, guiding tubes 5 of a cluster of control rods, protective tubes 6 of the bar of a control rods drive, internal metallic shaft 7, block of protective tubes 8, first- and second-stage separator 10 and 11, re-hydrator 12, vessel lid 13, electromagnetic drives 14 of control rods, internal reservoir 15 for fresh micro fuel elements, weld plug 16, branch pipes 17 for loading of fresh micro fuel elements, upper block 18, pipeline for ball transport 19, ball-stop armature 20, internal repository 21 for spent micro fuel elements, drive rod 22 of ball-stop armature.

145 electromagnetic drives of ball-stop armature for every fuel assembly are located upon the reactor vessel bottom. The vent aligned with the discharge ball transport pipeline of 20 mm diameter is located in the reactor vessel bottom. It is intended for spent micro fuel elements from the internal repository of the reactor vessel discharge.

Internal reservoir for fresh micro fuel elements is divided into 30 sections. Each section is attached to branch pipe of 20 mm outer diameter for loading of fresh micro fuel elements. From below each section is attached to the pipe collector, which distributes micro fuel elements among groups of 48-49 fuel assemblies. To provide that, pipelines lay within the block of protective tubes and separators. The upper ends of these pipelines are aligned with the pipe collectors, while the bottom ends are fixed to the inlet ball transport pipeline in the fuel assemblies.

All of joints are detachable. When it is necessary, the unloading all of internal devices and fuel assemblies for a maintenance and re-equipment could be carried out. Ball-stop armature in the discharge pipeline inside the reactor vessel is produced in the knee form, where the movement of micro fuel elements is possible only by using of hydraulic transport. The ball transport pipeline of internal repository of spent micro fuel elements and all of sections of internal reservoir of fresh micro fuel elements are welded at the reactor operation. They are decapsulated at the core refueling after reactor operation during 25-30 years, shutdown, cooling and depressurization. The return valve is located in the ball transport pipeline of every section of the internal reservoir, which eliminates the possibility of a suck of fresh micro fuel elements from the reactor vessel.

2. Reactor core

Reactor core consists of 145 square fuel assemblies. Map of reactor core is shown in Fig. 1.1. The choice of square cross section of fuel assembly is caused with the requirement of uniform accommodation and alternation of inlet and outlet coolant collectors in the form of truncated cones.

Research of emergencies and beyond design basis accident shows a safety and reliability of VKR-MT reactor concept [2]. For example, maximum of the core temperature does not exceed a value of 700 ^{0}C for the accident process caused by the rupture of the reactor vessel bottom. This temperature regime practically excludes the zirconium-steam reaction and weakly influences on the strength of structural material. Therefore it is possible to use the zirconium alloys as a structural material in fuel assembly of VKR-MT reactor. In turn, the application of zirconium alloys essentially improves the fuel cycle characteristics, reduces the initial enrichment and non-uniformity of fuel burn-up in the reactor core.

For the core based on micro fuel elements and directly cooled by water coolant-moderator, optimal hydrogen-to-heavy metal ratio is essentially higher than the ratio for a reactor core based on the fuel rods. It is caused by more neutron absorption in the resonance region due to more homogenization of neutron-physical properties of VKR-MT core. Increasing of hydrogen-to-heavy metal ratio goes down the probability of neutron absorption by uranium-238 at the resonance region, goes up the multiplication factor and reduces an initial enrichment for targeted value of average fuel burnup.

Increasing of hydrogen-to-heavy metal ratio is possible in ways as follows: by the application of additional tubes filled with moderator in fuel assembly and due to reduction of diameter of uranium dioxide kernel at fixed diameter of micro fuel elements. The last way is more profitable because the operation reliability of micro fuel elements under deep burnup condition increases of the cumulative thickness of a multi-layer coating. Decreasing of fuel loading and core lifetime without refueling (reloading) are compensated by the increasing of pebble bed height.

Therefore a number of tubes filled with water are chosen just taking into account control rods efficiency. Geometrical sizes of coolant collector in fuel assembly are performed such that pressure loss in coolant circuit is minimal.

The measures for decreasing of burn-up non-uniformity along fuel assembly are accepted. Particularly, on the base of development of once-through reactor with steam overheating [3] it is known that diameter of tubes filled with water should be preferably equal to 12 mm (radial burnup peaking factor in the fuel assembly is equal 1.06) and should not exceed 15 mm (radial burn-up peaking factor in fuel assembly is equal 1.2).

Two operation regimes are provided for the VKR-MT reactor. During the first 13 years of reactor operating, the fuel transport inside the reactor vessel is not produced. This operational regime is characterized with the high non-uniformity of fuel burnup along the radius and the height of the core and low economic parameters of fuel cycle. Reactivity margin for fuel burnup in the first operational regime is compensated by the using of burnable poisons and control rods.

The second operational regime is realized after reactivity margin exhaustion during approximately 13 years, which is characterized with the permanent refueling in every fuel assembly by small batches of micro fuel elements. Permanent refueling is carried out without the reactor vessel lid opening and power reduction. The electromagnetic drives mounted (located) upon the reactor vessel bottom open the ball stop valves in the tail part of every fuel assemblies.

At that the discharge of spent micro fuel elements from the bottom of fuel assembly is performing in an hourglass mode. Accordingly, the fresh micro fuel elements using the similar mode are loading from the internal reservoir into the upper of fuel assembly. Permanent refueling is performed by means of the special program, which will be developed later. Refueling rate is inversely proportional to the power of fuel assembly. Therefore in the second operational regime, minimal volume burn-up peaking factor and high quality of fuel cycle characteristics are reached. Burnable poison is absent in the reactor core at the second operational stage. Specific consumption of micro fuel element per unit of produced power is reduced by one half in comparison with one of the first operational regime. Lifetime of the second operational regime is also equal to 13 years.

After fresh micro fuel elements reserve depletion in the internal reservoir (via 26 years) the reactor shuts down and the loading of fresh and discharge of spent micro fuel elements is carried out. These measures are performed without the reactor vessel lid opening as follows. The reactor is cooled down and coolant pressure is gone down to 1 bar. The container with fresh micro fuel elements and the hydraulic transport are supplied to the NPP. Branch pipes for fresh micro fuel elements loading into the internal reservoir are decapsulated and connected to the supplied container. After that the internal reservoir with fresh micro fuel elements loading is carried out. Further the empty container is connected to the branch pipe for spent micro fuel elements discharge from the internal repository. The spent micro fuel elements are transferred to the container by hydraulic transport. After that the branch pipes are welded. The reactor is prepared for full power and the operation during 13 years. At that the reactor core is characterized practically equilibrium operational regime of permanent refueling with high parameters of fuel cycle.

In boiling water reactors non-uniformity of neutron flux increasing and correspondingly non-uniformity of fuel burnup is observed because of high non-uniformity of coolant-moderator density in the axial direction.

At the permanent refueling (reloading), the fresh micro fuel elements having maximum of the multiplicating properties and uranium-235 contents are driven by gravity from the upper of fuel assembly and caused by the spent micro fuel elements discharge from the bottom. Therefore the decreasing of neutron flux in the upper part of the core is advisable and it realizes for example by means of coolant-moderator density decreasing. This property can be reached in the fuel assembly including several axial stages, in which of outlet collectors of the first stage are connected with the inlet collectors of the second axial stage and so on. Such a technical decision presented in the paper [3] provides step-type coolant-moderator density decreasing from the bottom to up of the reactor core. This decision provides an appropriate axial power distribution under permanent refueling condition. Besides, coolant mixing at the cocurrent flow in the collectors makes it possible to decrease a non-uniformity of coolant heating. As it has been presented in paper [3] that three axial stages are quiet sufficient for achievement of requested purposes.

During the first 13 years the movement of micro fuel elements inside the reactor vessel is absent and distribution of neutron flux is managed by means of non-uniform accommodation of burnable poison along fuel assembly height. Ball absorber elements having boron carbide kernel and outer diameter of 1.8 mm are used as a burnable poison. They are required during the first 13 years for

operational regime without the permanent refueling and after that ball absorber elements became unnecessary under permanent refueling conditions.
reactor core main characteristics are presented in table 2.1.

Table 2.1. reactor core main characteristics

Characteristics	first core life-time	First regime of refueling	Second regime of refueling
1. Number of fuel assemblies	145	145	145
2. Outer diameter of micro fuel elements, mm	1,8	1.8	1,8
3. Diameter of UO_2 kernel, mm	1,3	**1,3 (1/2) и 1,4 (1/2)**	1.4
3. Pebble bed height, mm	4000	4000	4000
4. Volume of pebble bed in fuel assembly, $м^2$	0,200	0,200	0,200
5. Porosity of a pebble bed of micro fuel elements	0,4	0,4	0,4
6. UO_2 load, t	70	(70+87,70)/2=78,8	87,7
7. Average/maximum fuel burn-up, % of fissile materials	6/10		6/7
8.Initial fuel enrichment, %	9	9 и 6	6
9. Core lifetime, years	13	another 14*	another 16
10. Volume of internal reservoir of fresh micro fuel elements, (% of reactor core volume	50	50	
11. Diameter of UO_2 kernel of micro fuel elements being in storage of “fresh” fuel, mm	-	1,4	

In condition of fresh coated particles storage loading at half core loading.

3. Micro fuel element

The design of a micro fuel element are shown in Fig. 3.1 and Fig. 3.2. The main difference of presented design variants from previous one is that the decreasing of uranium dioxide kernel diameter to increase its reliability. Loss of uranium content is compensated with the core height rising.

Diameter of uranium dioxide kernel is equal to 1.3 mm for the first core. Its life-time is characterized by an absence of gradual refueling regime. This action allows to provide with fuel elements reliability for maximum fuel burnup as high as 10% heavy nucleus. At that condition the average burnup is equal 6% heavy nucleus during the first core life-time. Design scheme of micro fuel element for the first core life-time is presented in Fig. 3.1.

Diameter of uranium dioxide kernel for the following life-times with gradual refueling regime is planed to select at 1.4 mm. At that maximum burn-up is equal to 7% and the average burn-up is equal to 6%. Design scheme for this operation regime is presented in Fig. 3.2.

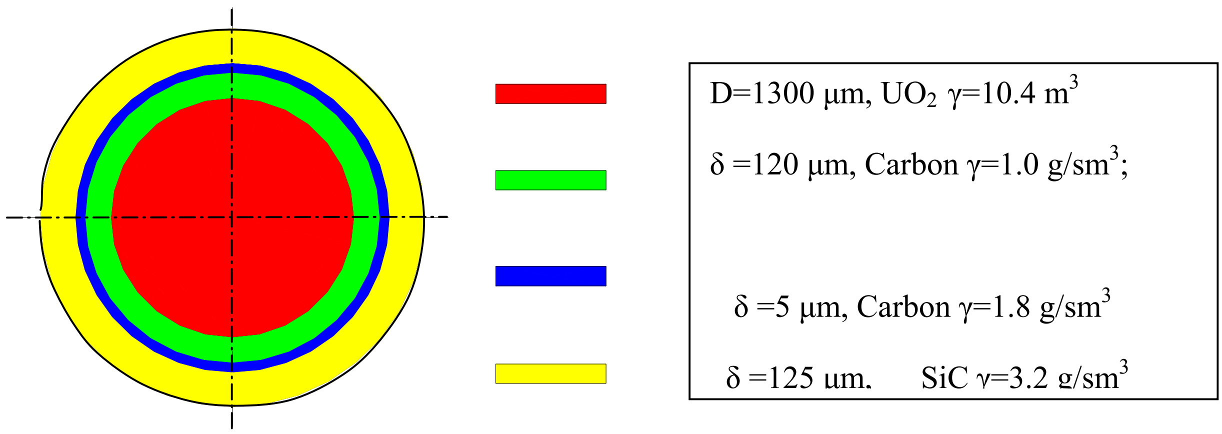

Fig.3.1. Design scheme of micro fuel element for the first core life-time

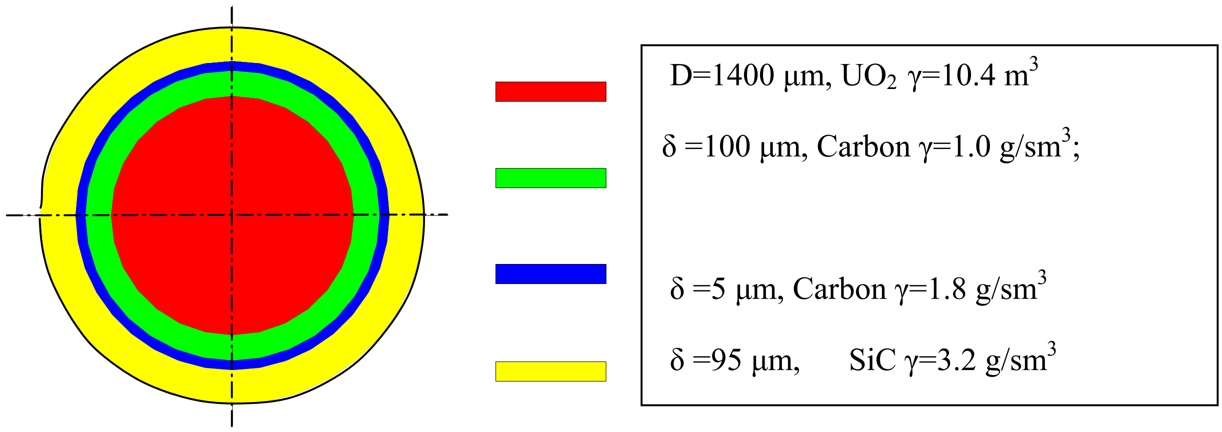

Fig.3.2. Design scheme of micro fuel elements are in-pile storage of “fresh” fuel

4. Fuel assembly

Three-stage- and many-collector fuel assembly has been accepted on the base of development of once-through reactor with steam overheating [3]. Three-stage fuel assembly is well accorded to regime of permanent reloading in the boiling reactor. In this case fresh micro fuel elements loading from the reactor core top burn in the region with the least coolant density and relatively low neutron flux. Many-collector design of fuel assembly provides for minimal micro fuel elements burnup non-uniformity. Based on the experience in the development of once-through reactor it is known that inner diameter of a collector and guiding tube should not exceed 14-12 mm in order to go down a non-uniformity coefficient of burn-up of micro fuel elements within a value of 5-20% respectively. Additional accommodation of moderator in the pebble bed should be highly uniform.

Each stage of fuel assembly consists of 18 inlet and 18 outlet coolant collectors. Inlet coolant collectors of lower stage in the form of truncated cone are take turns inverted outer collectors. Three stages provide for satisfactory mixing of reactor coolant. At that it is provided for three-stage decreasing of coolant density along core height. Square form of fuel assembly has been accepted due to design reason. It is easier and more uniform to perform the alternation of inlet and outlet coolant collectors in the square fuel assembly. 25 guiding tubes of control rods are uniformly accommodated between coolant collectors. The gap between fuel assemblies is equal to 2 mm. Quantity of coolant in the gap is nearly equal to one in peripheral row of guiding tubes that it is provided for high uniformity of moderator distribution in lateral section of fuel assembly. The gap between fuel assemblies plays a role of non-ideal outer collector due to perforated duct assembly.

Used geometrical characteristics are provided for hydrogen-to-heavy metal ratio is equal to 4.5 at the input of lower stage. This value is 20% greater than appropriate one for VVER-1000 core and it allows to increase the fuel burnup at the reactor operation in regime of permanent reloading.

Fig. 4.1 presents a lateral section of the group of fuel assemblies. Fig. 4.2 presents a longitudinal section of the group of fuel assemblies. In Fig. 4.3 design scheme of fuel assembly with ball stop device is shown. The main characteristics of fuel assembly are presented in Table 4.1.

Table 4.1. Major geometry and thermal-hydraulic characteristics of fuel assembly

Characteristic	1 axial stage	2 axial stage	3 axial stage
1. Number of fuel assemblies in reactor	145	145	145
2. Stage height, mm	1400	1200	1400
3. Distance between fuel assembly centers, mm	250	250	250
4. Fuel assembly size, mm	248	248	248
2. Thermal power, MW	2,05	2,05	2,05
3. Enthalpy increment, kJ/kg	82	81	81
4. Coolant flow rate through the fuel assembly, kg/s	25,4	25,4	25,4
5. Steam fraction at stage inlet, %	0	0	6.25
6. Steam fraction at stage outlet, %	0	6,25	12,5
7. Temperature at core inlet, ^{0}C	280	295	295
8. Number of cone collectors	36	36	36
9. Maximum diameter of collectors, mm	17x0.5	17x0.5	17x0.5
10 Minimum diameter of collectors, mm	12x0,5	12x0,5	12x0,5
8. Number of guiding tubes for control rods	25	25	25
9. Diameter of the guiding tubes, mm	15x0.5	15x0.5	15x0.5
10 Thickness of duct wall, mm	1		
11. Structural material of the guiding tubes, the assembly duct, and the collectors.	ZrNb1%		
12. Perforation density in collectors and duct wall of fuel assembly, %	0.05		
13. Average density of coolant in pebble bed, kg/m^3	746	637	380
14. Coolant density in inlet collectors, kg/m^3	750	743	531
15. Coolant density in outlet collectors, kg/m^3	743	531	227
16. Maximum temperature of micro fuel elements, ^{0}C	300	300	300

Вариант 3

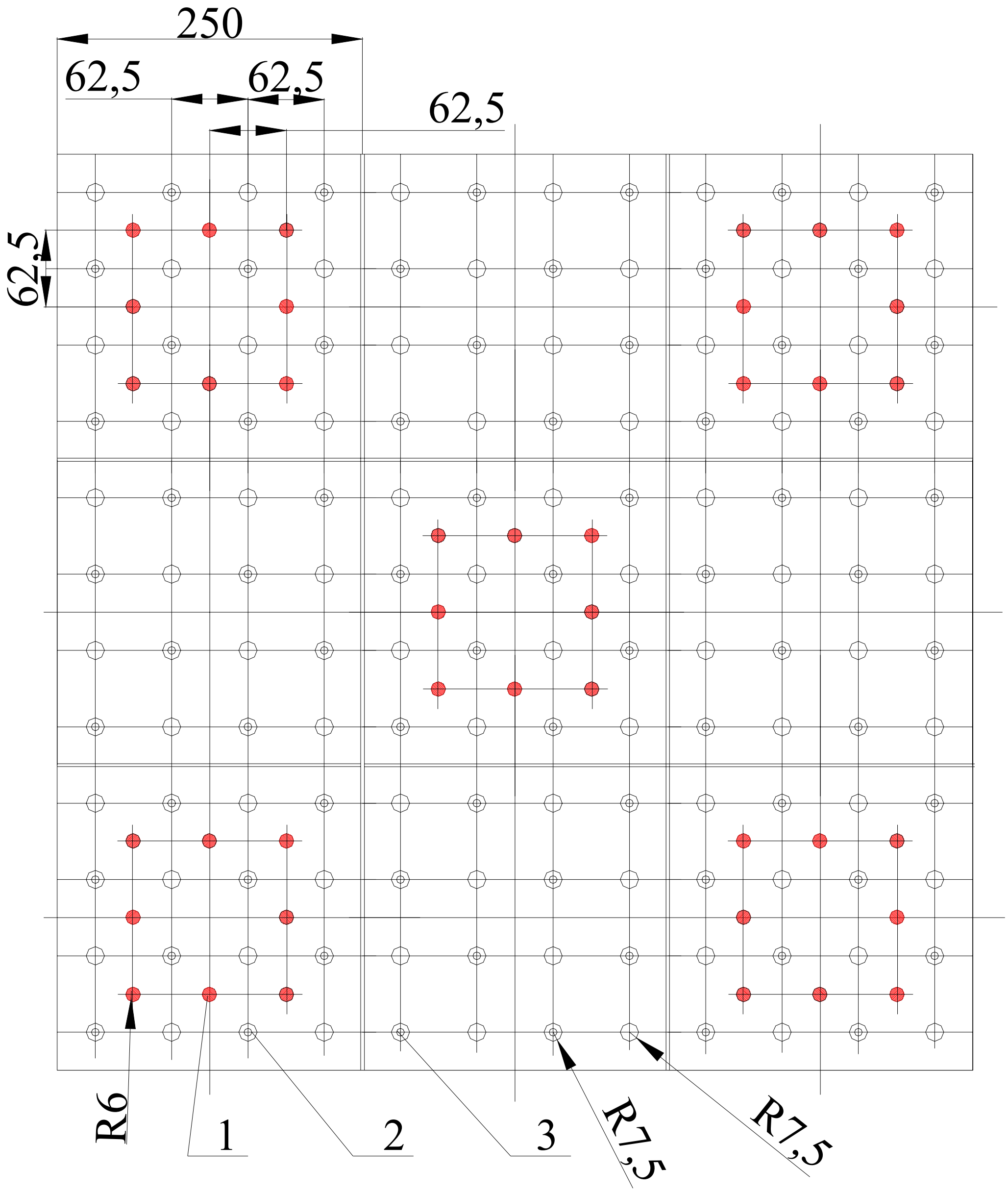

Fig.4.1. Lateral section of a group of fuel assemblies
1-guiding tubes of control rod, 2-inlet collector, 3-outlet collector

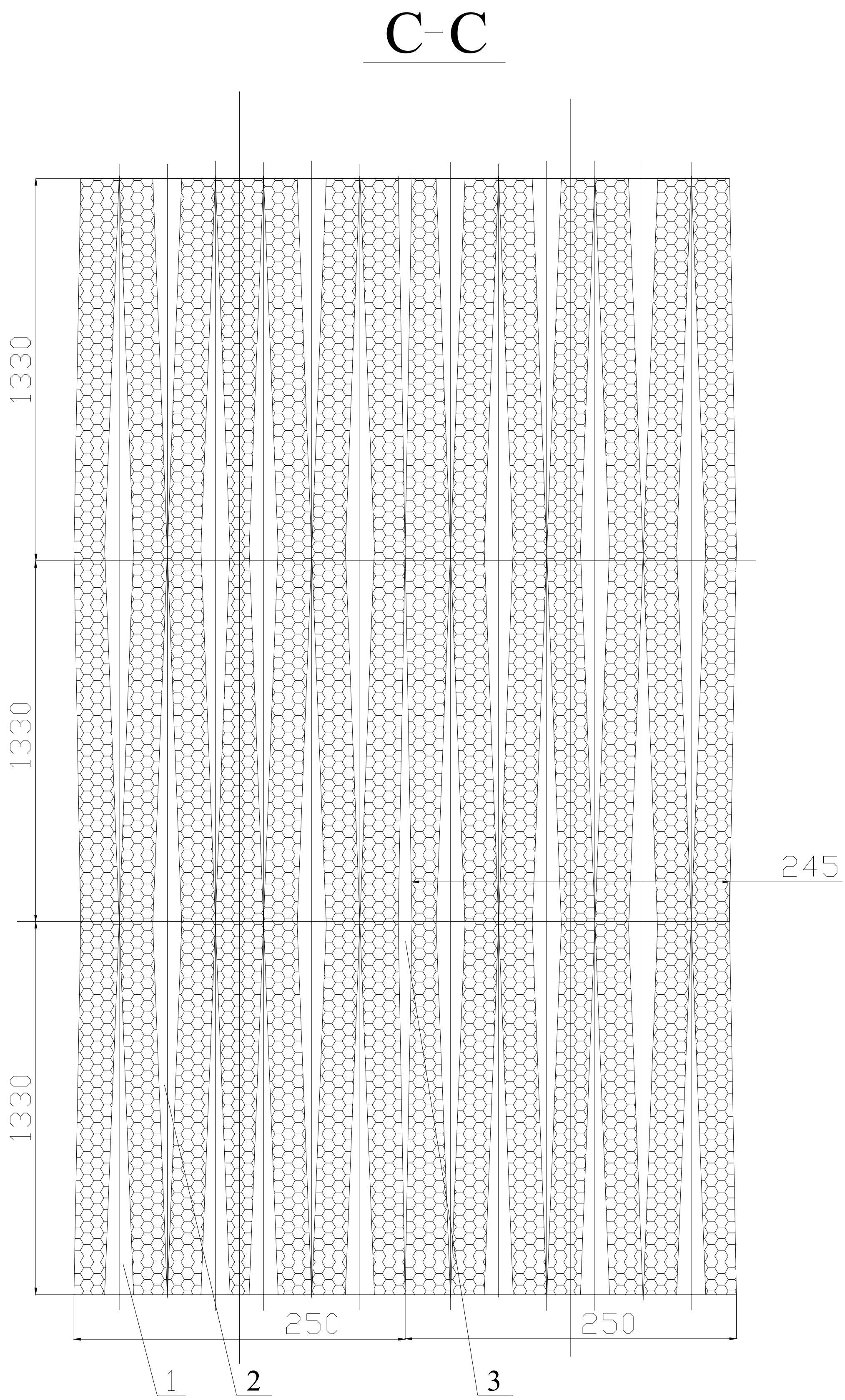

Fig.4.2. Longitudinal section of a group of fuel assemblies
1-inlet collector, 2-outlet collector of first stage of fuel assembly

5. Neutron-physical characteristics

5.1 Method of calculation of neutron-physical characteristics

Neutron-physical research has been performed by the use of Structure-UNK complex code. Structure [4] is intended for calculation of the neutron-physical processes in critical facilities and nuclear reactor core with triangular and square lattices. The action of Structure code is based on the PS_n-method of the solution of group neutron transport equation in two- and three-dimensional geometry.

The PS_n method [5] is a specific synthesis of the discrete ordinates and collision probability methods. By analogy with the discrete ordinates method, flux angular distribution at the cell boundary is presented in discrete form (N angular directions). The flux is assumed to be constant in each *n*-th angular direction. The coefficients responsible for the neutron transport from one cell boundary to others, and from the inner volume of a cell to its boundaries in each angular direction are derived by balance equations of collision probability method.

Let computational volume consists of a set of elementary cells. Using the approach of isotropic scattering, the average flux in an elementary cell for every energy group is derived from the set of equations (1)-(2):

$$J_{jn}^{-} = \sum_{i}\sum_{m} \alpha_{i\to j}^{m\to n} J_{im}^{+} + \beta_j^n \left(\Sigma_s \Phi + Q\right), \tag{1}$$

$$\overline{\Phi} = \frac{Q(V - \sum_j \sum_n \beta_j^n S_j) + \sum_j \sum_n J_{jn}^{+}(S_j - \sum_i \sum_l \alpha_{j\to i}^{n\to l} S_i)}{\Sigma_t V - \Sigma_S (V - \sum_j \sum_n \beta_j^n S_j)} \tag{2}$$

$$J_{jn}^{+} = \int_{\vec\Omega_n}^{\vec\Omega_{n+1}} d\vec\Omega \left|\vec\Omega, \vec n\right| \varphi_j\left(\vec\Omega\right) \qquad \left(\vec\Omega, \vec n_j\right) < 0,$$

$$J_{jn}^{-} = \int_{\vec\Omega_n}^{\vec\Omega_{n+1}} d\vec\Omega \left(\vec\Omega, \vec n\right) \varphi_j\left(\vec\Omega\right) \qquad \left(\vec\Omega, \vec n_j\right) > 0,$$

$$\overline{\Phi} = \frac{1}{V}\int_V d\vec r \int_{4\pi} \varphi\left(\vec r, \vec\Omega\right) d\vec\Omega, \qquad Q = \frac{1}{V}\int_V d\vec r \int_{4\pi} q\left(\vec r, \vec\Omega\right) d\vec\Omega$$

Where S - area of the cell boundary, V- cell volume, *i, j*- indices of the cell boundary, J_{jn}^{-}, J_{jn}^{+} - outlet and inlet neutron cell currents in *n*-th angular direction $\left(\vec\Omega_n, \vec\Omega_{n+1}\right)$, *n=1,...,* *N*, *N*- number of directions dividing angular space on the cell boundary, $\overline{\Phi}$ - an average neutron flux in the cell, Q - an average neutron source in the cell.

Coefficient $\alpha_{i\to j}^{m\to n}$ is the probability for a neutron entering through *i*-th boundary in *m*-th angular direction to escape through *j*-th boundary in *n*-th angular direction without any collision. Coefficient β_j^n is the probability for a neutron born in a cell to reach the j-th boundary without any collision and escape in n-th angular direction.

The set of equations (1)-(2) is solved by a method of consequent approaches at simultaneous correction of inner part of neutron source Q. For every energy group, an external neutron source and inner neutron scattering are considered to be the full neutron source. After calculation of one-side neutron currents at all cell boundaries, an average neutron flux in the elementary cells is computed. The average neutron flux is used to calculate a source of inner group's scattering in the following iterations.

Neutron cross-sections and parameters for modeling of fissile material depletion process are provided by UNK code system [6]. UNK has own library of nuclear data generated from files of nuclear data ENDF/B, JEF-2.2 and JENDL. Basic library contains 89-groups cross-sections of isotopes – 24 groups in slowing down region and 65 ones in thermal region. In addition, resonance region (2.15 eV – 2.15 KeV) is presented in microgroup form – about 7000 groups. In so doing, fine irregular energy grid is thickened in vicinity of resonances of different isotopes and wider one in interval between resonances is used. Spatial distribution of microgroup fluxes in the reactor cells (including micro fuel elements with appropriative fraction of surrounding moderator, guiding tube cells, duct wall cells, etc.) is performed by the use of collision probability method or characteristic method.

Computed microgroup fluxes in materials of reactor cells are used for spatial homogenization of representative reactor cells and condensation of the microgroup (up to 7000 groups) cross-sections of materials to the few-group ones. Energy boundaries of the few-group cross-sections for calculations of spatial neutron flux distribution in VKR-MT core are presented in Table 5.1.

Table 5.1. 9-group division

Group number	Upper boundary, eV
1	10000000
2	800000
3	1000
4	2.15
5	1.4839
6	1.0305
7	0.6967
8	0.4890
9	0.3440

5.2. Computational results of neutron-physical characteristics

They are presented in Fig. 5.1-5.8 and Table 5.1-5.5.

5.2.1. Features of reactor core

Reactor core of VKR-MT can be divided on three axial zones with essentially different neutron-physical properties (see Table 4.1). At the lower zone, coolant density has a maximum value, hydrogen-to-heavy metal ratio is equal to 4.2 and neutron energy spectrum is the most thermalized. At the middle zone, hydrogen-to-heavy metal ratio is equal to 3.6 due to decreasing of coolant density, which hardens the neutron spectrum. At the upper zone of reactor, hydrogen-to-heavy metal ratio is equal to 2.4 and neutron spectrum is nearly epithermal.

Multiplication factor of the infinite lattice of VKR-MT fresh fuel assemblies (without control rods) is equal to 1.4250.

5.2.2. Power distribution and neutron leakage in radial direction at hot condition

Table 5.2. Radial power peaking factor in the upper zone of reactor core. Radial neutron leakage (K_{inf}/K_{eff}-1) = 2.2%

1.724	1.667	1.537	1.332	1.065	0.753	0.479
1.667	1.610	1.484	1.284	1.023	0.721	0.457
1.537	1.484	1.364	1.173	0.927	0.644	0.401
1.332	1.284	1.173	0.998	0.771	0.518	0.343
1.065	1.023	0.927	0.771	0.570	0.446	
0.753	0.721	0.644	0.518	0.446		
0.479	0.457	0.401	0.343			

Table 5.3. Radial power peaking factor in the middle zone of reactor core. Radial neutron leakage (K_{inf}/K_{eff}-1) = 1.4%

1.809	1.746	1.604	1.379	1.088	0.750	0.420
1.746	1.684	1.546	1.327	1.043	0.716	0.400
1.604	1.546	1.414	1.206	0.938	0.633	0.346
1.379	1.327	1.206	1.014	0.768	0.493	0.270
1.088	1.043	0.938	0.768	0.547	0.358	
0.750	0.716	0.633	0.493	0.358		
0.420	0.400	0.346	0.270			

Table 5.4. Radial power peaking factor in the lower zone of reactor core. Radial neutron leakage (K_{inf}/K_{eff}-1) = 1.2%

1.827	1.764	1.619	1.391	1.094	0.749	0.405
1.764	1.702	1.561	1.338	1.049	0.715	0.385
1.619	1.561	1.427	1.214	0.941	0.631	0.333
1.391	1.338	1.214	1.019	0.768	0.487	0.253
1.094	1.049	0.941	0.768	0.541	0.337	
0.749	0.715	0.631	0.487	0.337		
0.405	0.385	0.333	0.253			

As seen in Table 5.2-5.4, maximal neutron leakage is observed in the upper zone of reactor core, where neutron spectrum is nearly epithermal.

Because maximum of power distribution lies in middle and lower zone of reactor core that radial neutron leakage along all height of reactor core is accepted to 1.5% at beginning of work and 2.5% at the end of core lifetime.

5.2.3. Reactivity effects and worth of the reactivity control systems.

Temperature effect is based on the reactivity inserted in reactor core at decreasing of material's temperature from nominal values down to 300 K and by the increasing of coolant density from nominal parameters up to 1.0 g/cm^3 with taking into account the decreasing of neutron leakage. Temperature effect is equal to ~ 7%Δk/k.

Xenon's poisoning reactivity effect is equal to 2%Δk/k and it is calculated by the variation of multiplication factor by small irradiation [about 0.1 MWtdays/kg U] of fresh reactor core. Total reactivity effect including temperature effect and xenon's poisoning effect is approximately equal to 9%Δk/k.

Worth of the mechanical system of reactivity control is determined by the substitution of water coolant in guiding tubes with control rods, containing boron carbide. Weight content of ^{10}B isotope in control rod is equal to 1.554E-02 (barn*cm)$^{-1}$.

Computational analysis has been performed for cold (temperature of reactor core is equal to 20^{0}C), unpoisoned (with ^{135}Xe) conditions.

Inner radius of guiding tubes for control rods is equal to 7 mm. In Table 5.4 the worth of the reactivity control system is presented for two control rods of 4.1 mm and 6.1 mm outer radius. First control rod of 4.1 mm outer radius seems to control rod of WWER-1000 type reactor. Second control rod of 6.1 mm outer radius is practically highest possible because water gap between inner surface of guiding tube and outer surface of control rod is less than 1 mm.

As seen from Table 5.5, worth of the reactivity control system based on the control rods with outer radius of 4.1 mm is less than total reactivity effect, which consists of temperature reactivity effect and the reactivity effect of reactor poisoning by accumulation of ^{135}Xe. Increasing of outer diameter of control rod up to 1.22 cm goes up the worth of the reactivity control system and allows to compensate of total reactivity effect.

Table 5.5. Worth of the reactivity control system

Outer diameter of control rod, cm	0.82*	1.22
Cover thickness of control rod, cm	0.06	0.06
. Worth of the reactivity, %Δk/k	7.4	10.7

*- Being correspond to the radial sizes of control rod of VVER-1000 reactor type

5.2.4. Core lifetime

In addition to compensation of the overall reactivity effect, control rods should compensate the burn-up reactivity margin, which is essential taking into account the longitudinal core lifetime (see Table 2.1).

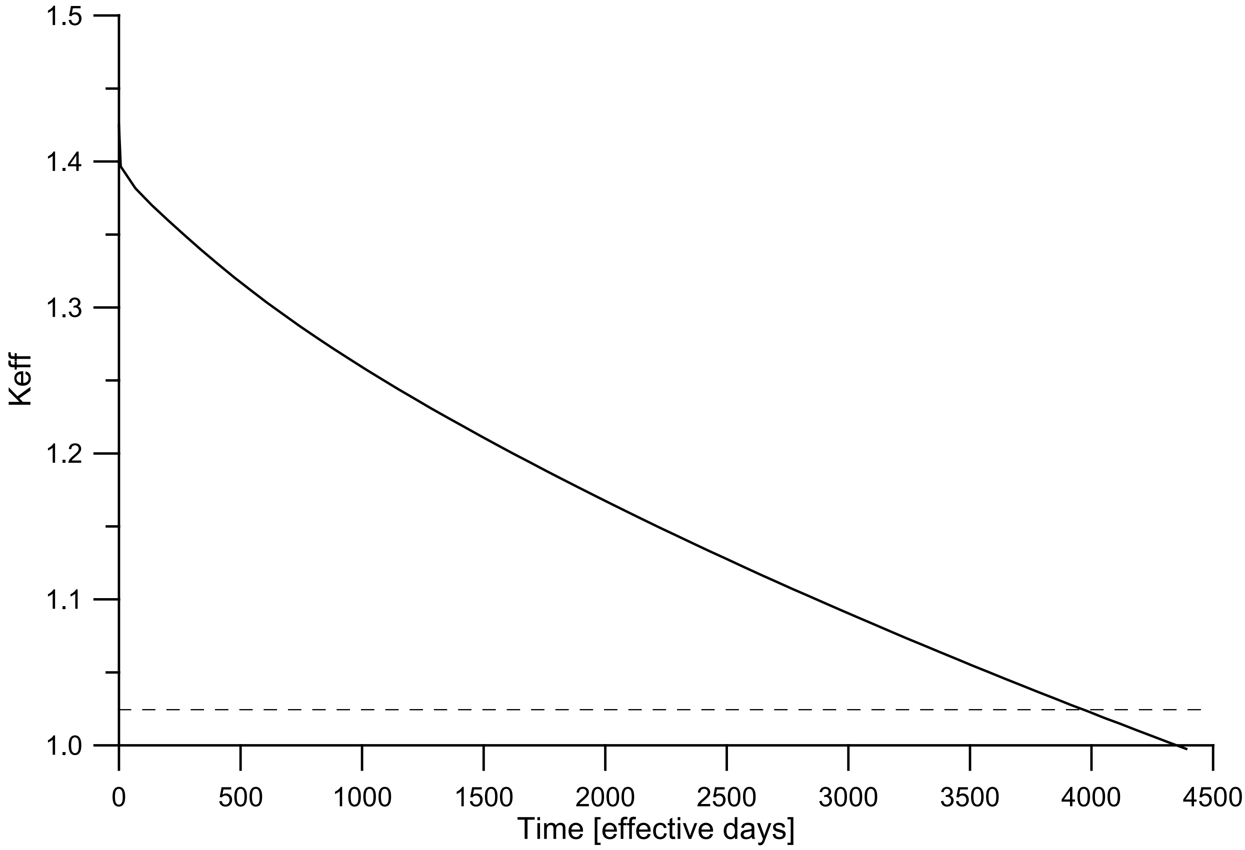

Fig. 5.2 K_{eff} changes with fuel burnup in VKR-MT core

Figure 5.2 presents the reactivity change over fuel burn-up cycle in a VKR-MT core without burnable poisons, calculated in the assumption that control rods are not inserted. It can be seen that the maximum burn-up reactivity to be compensated by control rods is around 40 % $\Delta k/k$.

Calculation of uranium depletion process is carried out for the representative assembly in the infinite lattice, which is characterized an average thermal power and nominal temperatures of fuel and moderator. Core lifetime is determined by the use of time point of disappearance of the reactivity margin (taking into account radial neutron leakage at the end of reactor operation, see Fig. 5.2).

As seen from Fig. 5.1, core lifetime is equal to 3900 effective full power days [EFPD] (or approximately 13 years) at the 2.5% of radial neutron leakage.

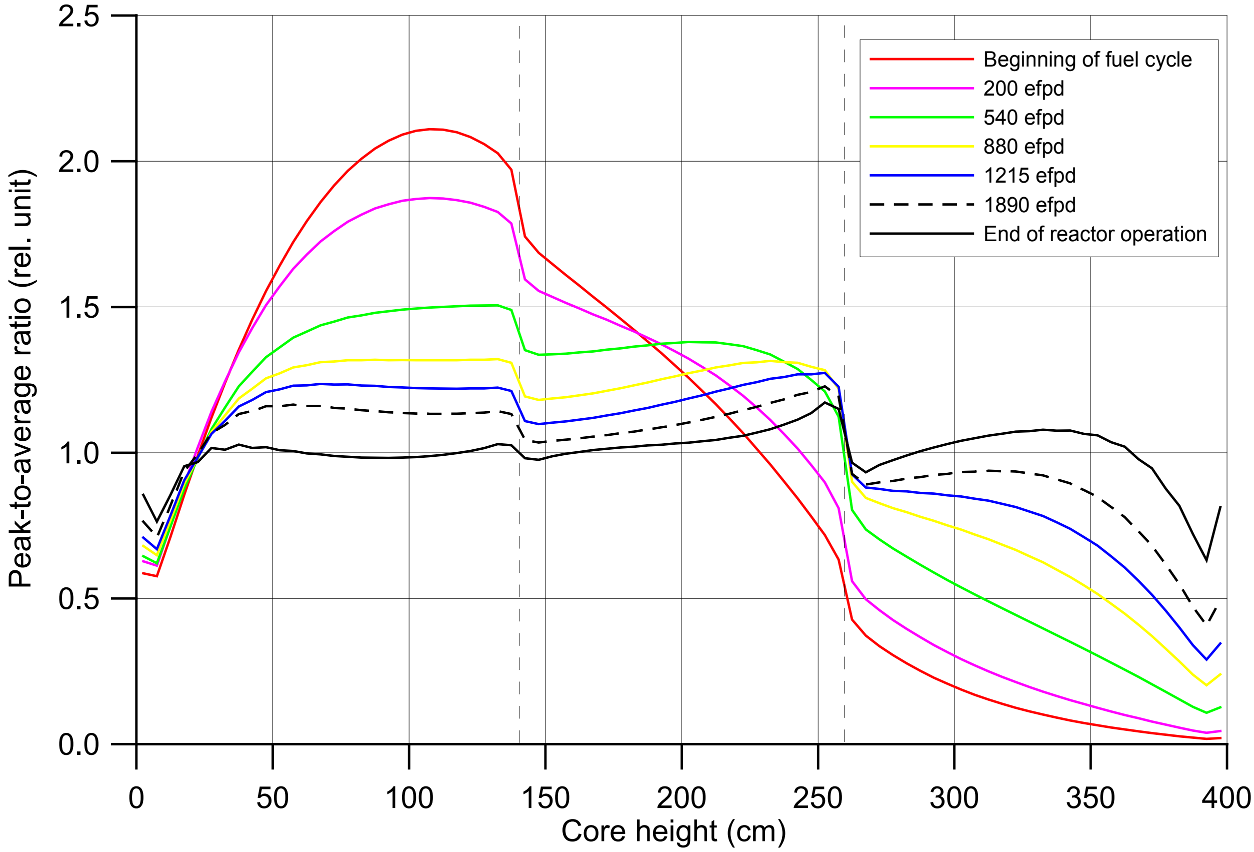

Рис. 5.2 Axial power distribution at the different operational time

As seen from Fig. 5.2, maximal value of axial power peaking factor is equal to 2.15 at the beginning of reactor operation and the most heated point is located in the lower zone at the level of 110 cm height.

After that, power distribution flattens due to non-uniformity of fuel burn-up in the axial direction. At 600 effective full power days, axial power peaking factor goes down to approximately 1.5 and the most heated point is shifted up to the boundary between lower and middle zones (135 cm, see Fig.5.2).

In further the most heated point is moved near to the boundary between middle and upper reactor zone (250 cm). At the end of reactor operation axial power peaking factor is equal to 1.2.

In additional to essential radial and axial non-uniformity of power distribution (see also Table 5.2-5.4) there is power peaking inside the fuel assembly due to mainly impact of guiding tubes filled with coolant-moderator (in the assumption that control rods are not inserted).

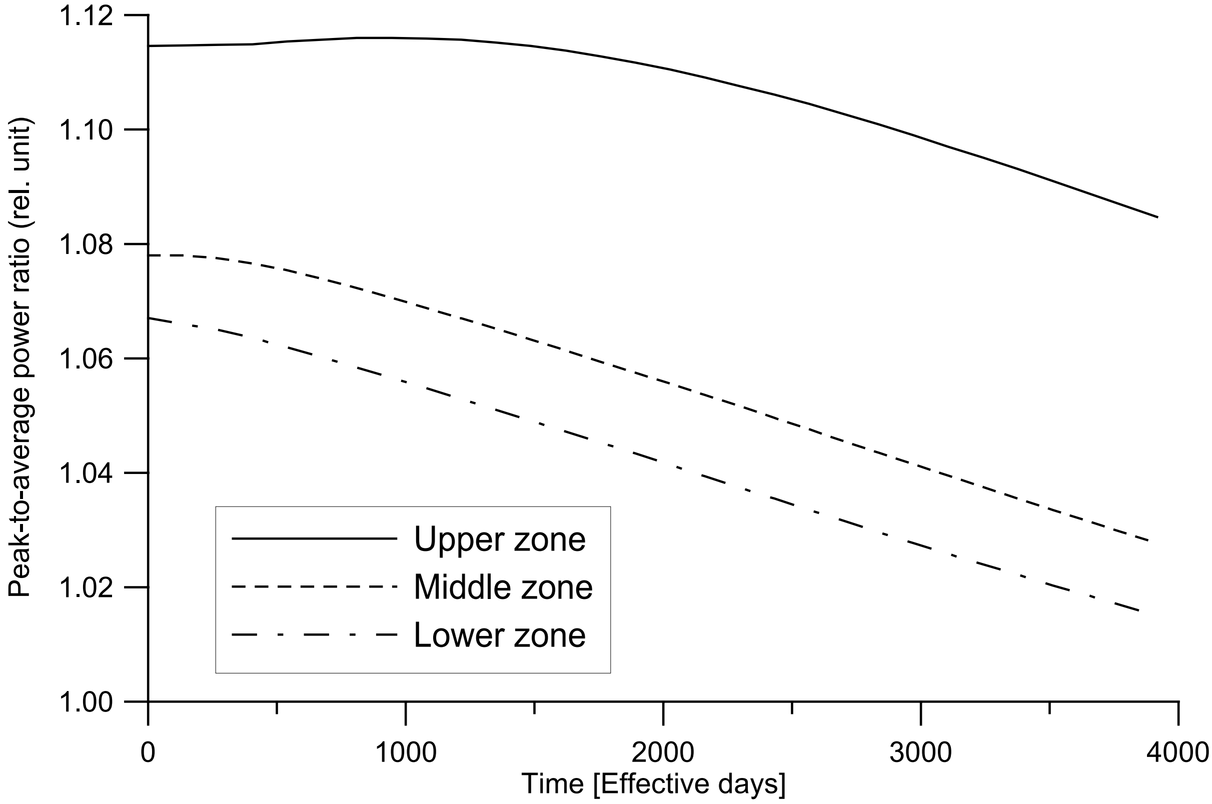

Рис. 5.3. Radial power peaking factor in representative assembly versus fuel burnup

Fig. 5.3 presents the variation of radial power peaking factor in upper, middle and lower zone. Power non-uniformity is caused by over-thermalized neutron spectrum near to guiding tubes filled with coolant-moderator. Value of non-uniformity directly depends on the measure of this over-thermalization in comparison with an average neutron-physical property of current axial position of fuel assembly. During reactor operation power peaking factor is decreased due to higher rate of burn-up of micro fuel elements near to guiding tubes in comparison with other ones.

5.2.5. Distribution of fuel burnup in fuel assembly.

On the base of thermal power (890 MW), fuel load with heavy metal (67833 kg) and core lifetime (3900 effective days) it is possible to determine the average fuel burnup in reactor core, this value is equal to 51.2 MWdays/kg U.

Axial and radial power peaking produces non-uniformity of burn-up fuel within fuel assembly. As seen from Fig.5.2, maximal value of fuel burn-up is reached in the lower zone of reactor core. Fig. 5.3 presents that maximal fuel burn-up in plane of fuel assembly is achieved near to guiding tubes filled with coolant-moderator.

Axial distributions of average and maximal values of fuel burnup in plane of fuel assembly are presented in fig.5.4.

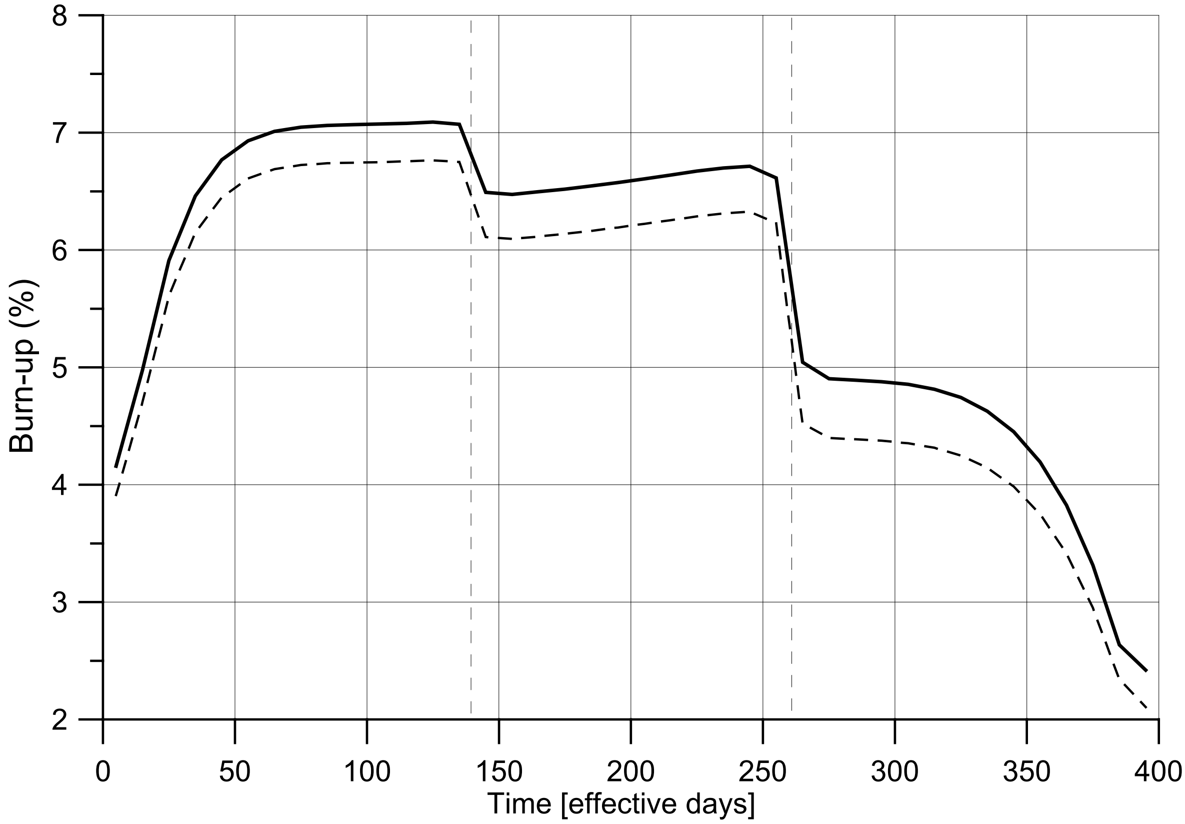

Fig. 5.4 Distribution of average and maximal values of fuel burn-up (in plane of fuel assembly) versus reactor core height

As seen in Fig.5.4, maximum value of fuel burnup in the representative fuel assembly with average power is equal to 7%. The most burned point is located on the boundary between lower and middle zone and nearly to guiding tube with coolant-moderator.

It should be noted that power non-uniformity in the reactor core also influences on the maximal value of fuel burn-up. At the beginning of reactor operation, radial power peaking factor of fuel assemblies is about 1.8 (see Table 5.1-5.3). Hence at the end of core lifetime, fuel burnup in the central assembly exceeds the average value on 25-30%.

Thus, fuel burn-up of the most irradiated point in the central assembly of reactor core is equal to 10%.

5.2.6. Insertion of burnable poison

As seen from Fig. 5.1-5.2, accommodation of burnable poison in reactor core is carried out with following purposes: firstly, to go down the sufficient burn-up reactivity margin and, secondly, to decrease the axial power peaking factor at the beginning of fuel cycle.

Independent research is requested for the solution of this problem.

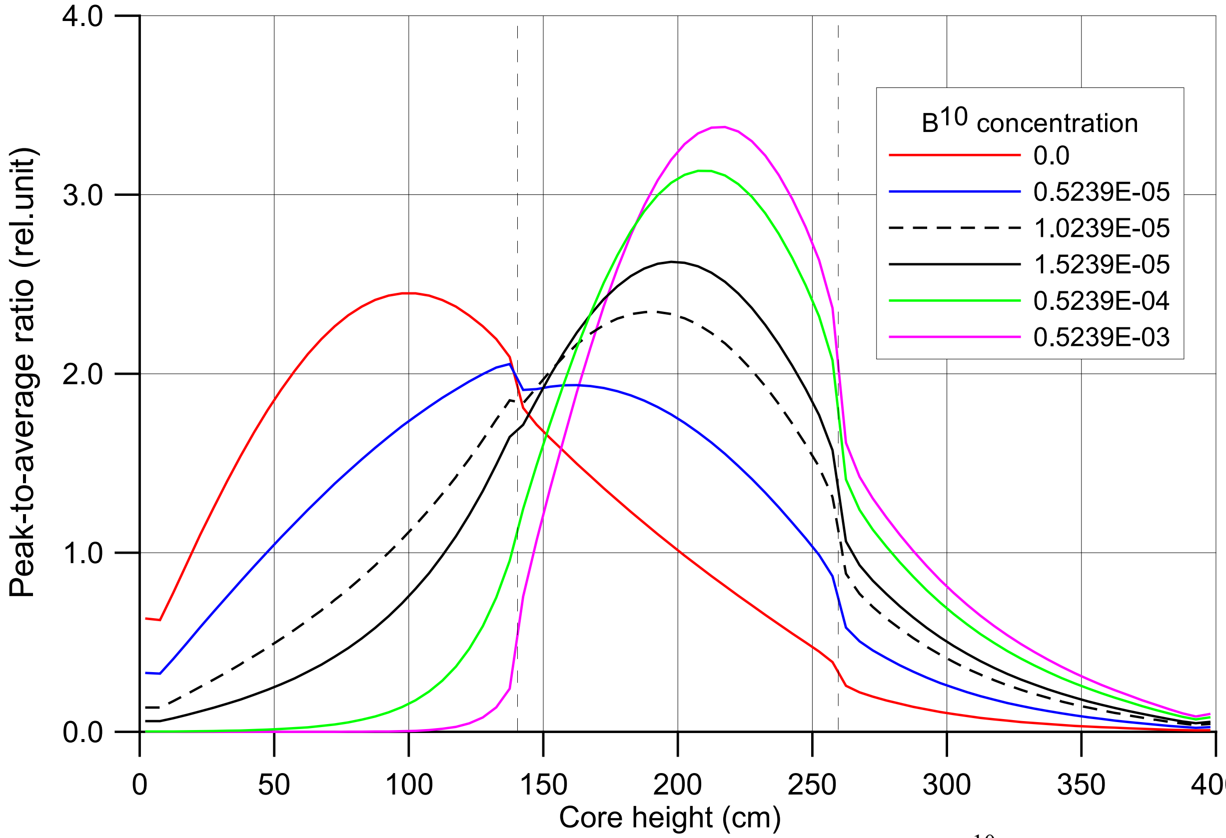

Fig. 5.5 Axial power peaking factor versus concentration of B^{10} isotope in the lower zone of reactor core

Fig. 5.5 and Table 5.5 presents the problem of insertion of burnable poison only in the lower zone with the most thermalized neutron spectrum. At the increasing of poison content, the maximum of neutron flux is migrated to middle zone and reactivity efficiency of burnable poison is decreased.

Table 5.6 Multiplication factor versus isotope content of B^{10} in the lower zone of reactor core

Concentration of B^{10} $(barn*cm)^{-1}$	K_{eff}
0.0	1.42496
0.5239E-05	1.40572
1.0239E-05	1.39746
1.5239E-05	1.39384
2.5239E-05	1.39053
0.5239E-04	1.38738
0.5239E-03	1.38326

Thus, development of detail map of insertion of burnable poison distributed over both radial and axial directions is required.

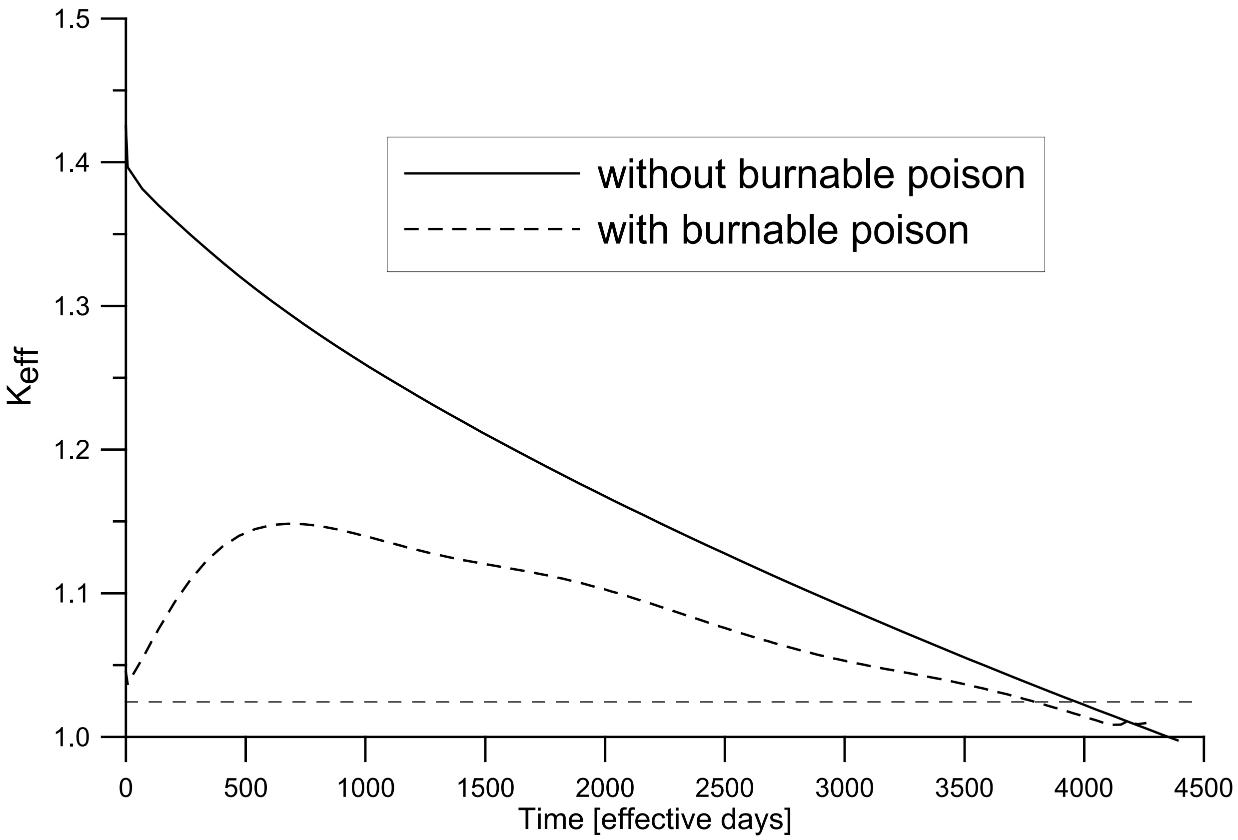

Fig. 5.6 K_{eff} changes versus fuel burnup in VKR-MT core both with burnable poison and without one

Figure 5.6 presents the reactivity change over fuel burn-up cycle in a VKR-MT core with burnable poisons, calculated in the assumption that control rods are not inserted. It can be seen that the maximum burn-up reactivity to be compensated by control rods is around 15 % Δk/k. At that initial concentration of ^{10}B is equal to 9.0E-05 $(barn*cm)^{-1}$ along all height of reactor core.

Thus, application of burnable poison is efficient for the decreasing of burn-up reactivity margin, which to be compensated by the mechanical system of reactivity control.

5.3. Main results of neutron-physical analysis

Main results of neutron-physical analysis of VKR-MT reactor core can be presented as follows:

- Total reactivity effect including temperature effect and xenon's poisoning effect is equal to 9%Δk/k;
- Accepted accommodation and geometrical sizes of guiding tubes allows to compensate total reactivity effect only at the maximum possible outer diameter of control rod (1.22 cm), but worth of the reactivity control systems (10.4%Δk/k) is not sufficient for compensation of burnup reactivity margin;
- Radial power peaking factor of fuel assemblies in reactor core is equal to 1.83, the most heated fuel assembly is central FA at the beginning of the work;
- Axial power peaking factor is equal to 2.2;
- Radial power peaking factor inside the fuel assembly does not exceed 12%;
- Core lifetime is equal to 3900 effective full power days [EFPD] (or approximately 13 years), at that average fuel burn-up is equal to 51.2 MWtday/kg U;
- Maximum fuel burnup is equal to ~10%, the most burned point is located near to guiding tubes of central fuel assembly on the boundary between lower and middle axial zones;
- Addition of burnable poison only in lower zone of reactor core does not strongly affect on the decreasing of burn-up reactivity margin due to movement of the neutron flux from lower to middle axial zone;
- Burnup reactivity margin (without burnable poison) is equal to ~40%, insertion of burnable poison in core allows to decrease the burnup reactivity margin down to ~14%;
- On the base of the results of neutron-physical analysis it is recommended to increase a number of guiding tubes of control rods in every fuel assembly at the least in 2-2.5 times more and also to find the technological solutions of the problem of insertion of boron carbide or gadolinium oxide into reactor core.

6. Proliferation nuclear materials resistance

The major technical means provide for resistance of proliferation nuclear materials are essential long-duration core lifetime and elimination of on-site refueling. The design of VKR-MT reactor core proposed in the paper increases the core lifetime without refueling over than 25 years (13 years without permanent refueling and ~13 years of permanent refueling regime). Application of internal repository for fresh and spent micro fuel elements and permanent refueling in reactor core without the opening of reactor vessel lid and power reduction essentially improves the fuel cycle characteristics. In particular, it decreases the initial fuel enrichment and fuel burn-up peaking factor.

Vulnerable feature of nuclear reactors without on-site refueling from the point of view of non-proliferation problem is a transportation of reactor with fresh fuel to the site of NPP.

Besides, reactor transportation with burned core is technically possible only in case of small reactor sizes, i.e. very small power capacity. If reactor power is more than 100 MW(e) that such procedure is practically impossible. It is hard to imagine the transportation of reactor vessel of 4.5 m diameter and 500 tons mass in a container with biological shielding! Technical problem is also demounting of irradiated reactor and mounting of fresh reactor in radioactive concrete shaft with following licensing!

In core with micro fuel elements in a pebble bed form for every fuel assemblies there is a technical possibility to perform the refueling reactor without opening of the vessel lid, using of different methods of ball movement in an hourglass mode or by hydraulic transport.

Fresh micro fuel elements are delivered to the NPP in the special container. After that the loading of fresh micro fuel elements in internal reservoir is carried out. Further the empty container is connected to the branch pipe for spent micro fuel elements discharge from the internal repository. The spent micro fuel elements are transferred to the container by hydraulic transport. Hydraulic transport is delivered in the NPP before refueling and removed immediately after refueling. This procedure is repeated every 25-30 years under IAEA inspection.

Long-duration core lifetime, realization of core refueling without the opening of vessel lid, absence of internal equipment for hydraulic transport, presence of return valve on the ball transport pipeline inside the reactor vessel are intended for the effective application of non-proliferation safeguards. From the point of view of non-proliferation safeguards proposed refueling technique is equivalence to method of reload of whole nuclear reactor.

On the other hand, such technical decision is considerably more realistic and efficient in comparison with reload of whole reactor because it is not performed the welding, millwright works and licensing of fresh reactor after mounting.

7. Conclusion

7.1. The improved design of VKR-MT reactor has been presented in this report. The reactor design is characterized by the maximal use of possibilities and profits of core with micro fuel elements in the form of pebble bed. For example, internal repository for fresh and spent micro fuel elements inside the reactor vessel and permanent internal refueling of core without power reduction and opening of the reactor vessel lid are envisaged.

7.2. Developed technical decisions provide for operational period of reactor core within 26 years.

7.3. Developed reactor design allows to elongate the reactor operation after 25 years of reactor lifetime by additional load of new portion of fresh micro fuel elements and discharge of spent micro fuel elements without opening of the vessel lid by the use of special container with equipment for hydraulic transport (every 13 years).

7.4. Application of internal permanent refueling and substitution of steel with zirconium in structural materials of fuel assembly essentially improve the characteristics of fuel cycle. In the regime of permanent internal refueling the ratio of average fuel burn-up to initial fuel enrichment is not worse than the one for fuel cycle of VVER-type reactors.

7.5. Developed reactor design has a very high resistance of proliferation of nuclear materials.

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