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Chapter 7

Thermal Reactors with High Reproduction of Fission Materials

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Additional information is available at the end of the chapter

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1. Introduction

Efficiency of power reactors is determined by the expenses of raw material for its work and by efficiency of heat to mechanical energy transformation.

In general modern nuclear power plants (NPP) use thermal reactors with enriched by isotope ^{235}U uranium (comparing to raw uranium) in the beginning of campaign. This fuel allows having sufficient reactivity margin for obtaining burn-up more than 30-50 MW*day/kg. This reactor type is developed by itself during short period of nuclear power plants development. Raw uranium cannot supply required burn-up even in reactors with best heavy-water moderator. An effective technology of isotope separation was made for military purposes.

Known shortage of thermal reactors is small usage of raw uranium during its work (0.5 – 1.0 %). Stocks of cheap ore for these reactors are enough for 40-50 years at power level of 4000 GW [1].

The next step in nuclear power plants development is suggested usage of fast neutron reactors. This transition is connected with fuel enrichment increase and it supplies possibility of fission reactions on fast neutrons, which produce more secondary neutrons. By solving technical problems at this direction, nuclear power plants supplied with cheap enough fuel for many centuries can be built. But this development direction has one shortage. It is extremely expensive.

When it`s advocates say that high price is because of modern technical solutions shortages, they are half right. Insuperable high price is because of high raw uranium requirement for its start. For using nuclear power plants with total power of 4000 GW, which are supplied for 3000 years, all cheap uranium stocks must be processed in 40-50 years [2]. It is connected with ecological problems and some complexity in non-proliferation of nuclear fission materials.

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The best thermal reactors related to high reproduction of fission materials are heavy water moderated reactors. Today there are several types of such reactors. But its potential still is not fully discovered. In CANDU reactors and like-CANDU reactors in the best cases are used fuel on the base of natural uranium as advantage. But achieved burn-up in these reactors is significantly lower than in light water reactors. Besides it, neutron moderation energy and heat leak from channels energy is lost in heavy water. These factors and compactness of light water vessel reactors have caused its leadership in modern nuclear power plants. Now this is shadowing potential performance of heavy water reactors.

There are designs of heavy water reactors, which allow improvement of technical and economical performance of it. In general it is related with use of thorium in fuel.

In fifties gaseous coolant in heavy water reactors have been tested, which allowed to use different values of pressure in reactor and maximal pressure in Rankin cycle. With use of fuel rods, which are much the same design as used in majority of modern reactors, coolant temperature up to 500 ^{o}C (EL-4, France) and even a little more (KKN, Germany) was achieved.

Efficiency at temperature, which is similar to achieved, at thermal power plants is more 40%. In the mentioned reactors efficiency is close to 30 % only. Possibly, that this experience served as a reason of transition to high temperature gas cooled reactors with graphite moderator and Briton cycle. Using gas cooled heavy water reactor is not in favor.

If heavy water channels reactors allow better characteristics than existing WWER, PWR, BWR, then it is necessary to know technical solutions, which are needed for this transition.

The purpose of the work is demonstration of thermal reactors development possibility in direction of fission materials reproduction increase, which is sufficient for obtaining burn-up comparable with burn-up of the best modern reactors. This development direction shows that these reactors have high raw uranium usage and can supply high durability of nuclear power plants work at high power with modest requirements in uranium mining. Small amount of fission materials in spent fuel reprocessing is significant advantage. At the same control level it allows less possibility of fission material proliferation. The possibility of reaching the high efficiency coefficient of nuclear plants with the proposed reactors is shown.

2. About usage problems of raw uranium and thorium

In fission reactors can be used uranium and thorium. But only in raw uranium there is fission isotope - ^{235}U. Fission material in thorium is absent, but can be obtained by neutron irradiation (^{233}U). The basic raw uranium isotope - ^{238}U at neutron irradiation becomes fission nuclide - ^{239}Pu.

One of reactor characteristics is raw uranium usage, obtained in its fuel cycle. If core has no conditions for producing ^{239}Pu from ^{238}U, then raw uranium usage is less than isotope ^{235}U portion in it.

In open cycles fuel is once used. A fuel with initial contents of fission and raw isotopes is loaded to a core. At the end of campaign the fission materials contents is decreased and the fuel not used any more. If enriched by ^{235}U uranium is used as an initial fuel, then for raw uranium usage calculations is necessary to calculate raw uranium mass, required for a core loading with enriched uranium [3]. Raw uranium usage I_{sp} is calculated as:

$$I_{sp} = M_{nat} - M_o + M_k; \quad (1)$$

Where:

M_{nat} – raw uranium mass, required for initial fuel producing;
M_o – mass of initial fuel loading;
M_c – fuel mass at the end of reactor campaign.

Portion of raw uranium usage Q_u, as relative quantity is calculated as:

$$Q_u = I_{sp} / M_{nat}; \quad (2)$$

In closed cycles after the end of campaign fuel is reprocessed for extraction of fission material rests. Different situations are possible.

In the first, the most undesirable situation, not all fission nuclides can be separated from raw nuclides. For example, ^{235}U remains in raw ^{238}U in small amounts so this mix cannot be loaded to a core. Produced ^{239}Pu and ^{241}Pu can be extracted from this spent fuel by chemical methods. Extracted fission materials must be diluted in portion of remained ^{238}U to produce new fuel. Portion of raw uranium usage in this case is:

$$Q_u = \left(I_{sp} + M_{Pu} / C_{dv}\right) / M_{nat} \quad (3)$$

Where:

M_{Pu} – mass of isotopes ^{239}Pu and ^{241}Pu, extracted from spent fuel and used in new fuel production;
C_{dv} – fission materials contents in initial fuel.

This formula is not taking into account difference between properties of initial and final fission materials and following history of fuel usage. It is estimation. This formula is more precise for condition $M_{Pu} / C_{dv} < 0.5 * M_{nat}$, that characterize modern thermal reactors.

Account of following fuel usage history can be conducted by formula:

$$D_{is} = Y * \left(1 + \psi + \psi^2 + + \psi^{n-1}\right); \quad (4)$$

Where:

Y – fuel nuclides burn-up during campaign;
ψ – ratio of extracted fuel material mass at the end of campaign to its initial mass;
n – campaign number of this fuel cycle.

The shortage, connected with impossibility of cheap extraction of fission isotope ^{235}U from raw isotope ^{238}U during spent fuel reprocessing, can be overcome by two ways. The first way is in increasing of initial ^{235}U burn-up, so at the end of campaign its contents is negligibly small. Another way is in using fuel with different nucleus charge – using fission isotope of uranium in thorium, fission isotopes of plutonium in raw ^{238}U. Usage of both ways simultaneously is possible.

Because of fission isotopes absence in raw thorium there is no problem with raw thorium usage.

Known stocks of thorium are bigger than known stocks of uranium. It will be ideal if nuclear power industry use both elements in its work. Basic raw material in the present time is uranium. Spending cheap stocks of uranium will lead to necessity of depleted uranium reprocessing from dumps of enrichment plants and spent fuel and thorium usage.

One of the problems with thorium fuel is ^{232}U production, which is source of high energy gamma-ray quanta [4]. One way of this problem solution is usage of automatics at thorium spent fuel reprocessing.

3. Characteristics of hypothetical thermal reactors with high reproduction of fission materials

3.1. Fuel ^{238}U + ^{235}U

High fission nuclides reproduction in thermal reactors is possible if amount of capture acts in fission nuclides is close to amount of absorption acts in raw nuclides.

Dependence of multiplication factor and fission nuclides reproduction coefficient for reactors with different fuel types and no neutron loss is shown on figure 1.

Three types of fuel are shown. Blue lines show data for mix of ^{238}U and ^{239}Pu, red lines for mix of ^{238}U and ^{235}U, brown lines for mix of ^{232}Th and ^{233}U. Continuous lines show multiplication factor, dotted lines – reproduction coefficient. Left lines of each type show data of neatly thermal reactor without epithermal neutron absorption. Right lines show data for reactor with 10% epithermal neutron absorption from total absorptions.

Multiplication factor for such reactor with mix ^{238}U and ^{235}U is calculated as:

$$K = n * \sigma_{5f} * \nu / ((1-n) * \sigma_{8a} + n(\sigma_{5f+}\sigma_{5a})); \quad (5)$$

Where:

n – portion of isotope ^{235}U in fuel;
σ_{5f} – fission cross-section of isotope ^{235}U;
σ_{5a} – absorption cross-section of isotope ^{235}U;
σ_{8a} – absorption cross-section of isotope ^{238}U;
ν – neutron amount at fission of ^{235}U.

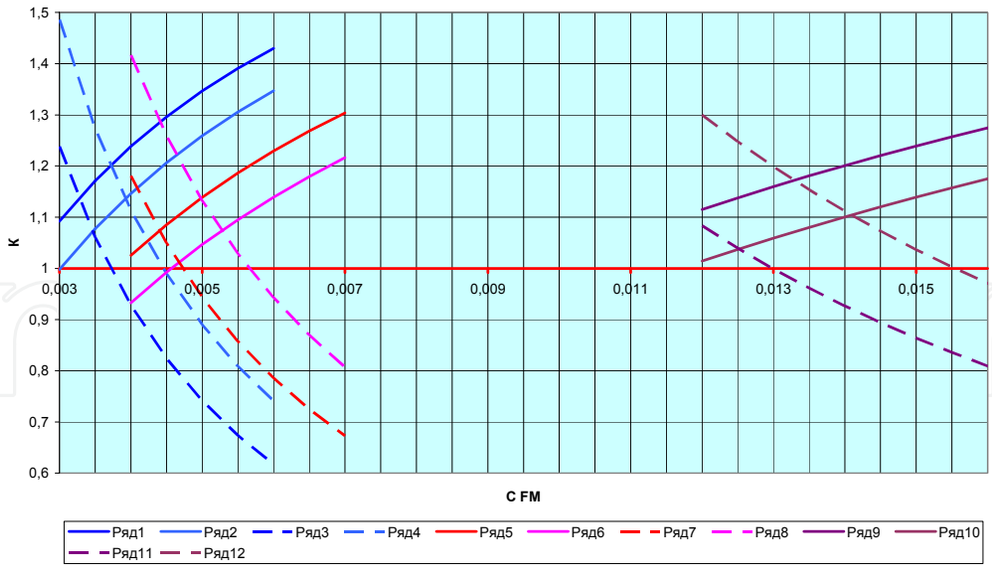


Figure 1. Dependence of multiplication factor and fission nuclides reproduction coefficient for reactors with different fuel types and no neutron loss.

Data for cross-sections and number of secondary neutrons, used in calculations, are taken from [5].

Fission materials reproduction coefficient in initial fuel is calculated:

$$KB = \left(1 - n\right) * \sigma_{8a} / \; n * \sigma_{5f}; \tag{6}$$

From data, which shown at figure 1 for hypothetical reactor, it can be seen that there is diapason of uranium enrichment (from 0.46 up to 0.66 %) in which reproduction coefficient and multiplication factor are more than unity simultaneously. For each of shown variants at reproduction coefficient equal unity multiplication factor is close to 1.1.

To estimate possibility of real reactor work at this diapason of enrichment is necessary to take into account two factors – presence of additional neutron losses in construction materials and neutron leakage, and production influence of secondary fission materials, actinides and fission products, which are sufficient neutron absorbers.

Estimation of these factors in point model of reactor is possible by introduction of neutron loss in construction materials and leakage term in system of equations, describing accumulation and neutron absorption in initial fuel nuclides and additional nuclides produced during reactor work.

Sufficiently precise estimation of reactor campaign characteristics with different fuel types can be made taking into account following nuclides:

- Raw nuclides in fuel – ^{235}U, ^{238}U, ^{232}Th;
- Secondary fission nuclides – ^{233}U, ^{235}U, ^{239}Pu, ^{241}Pu;
- Nuclides of fuel chains – ^{233}Pa, ^{234}U, ^{236}U, ^{237}U, ^{239}Np, ^{240}Pu, ^{242}Pu;
- Actinides – ^{241}Am, ^{242}Am;

- Fission products with high absorption cross-section and atomic mass:
- 99 (Nb, Mo, Tc); 103 (Ru); 131 (Sb, Te, J, Xe); 133 (Te, J, Xe, Cs); 135 (J, Xe); 143 (La, Ce, Pr, Nd); 145 (Ce, Pr, Nd); 149 (Nd, Pm, Sm); 151 (Nd, Pm, Sm, Eu); 152 (Sm); 153 (Sm, Eu); 155 (Sm, Eu, Cd); 157 (Eu, Cd).

In general, for each of nuclides the equation is solved:

$$\frac{dNz}{dt} = \lambda_{z-1} \cdot N_{z-1} - \lambda_z \cdot N_z - \sigma_z \cdot N_z \cdot \Phi; \quad (7)$$

where:

Nz – number of nuclei with charge z in fuel;
λ – decay constant;
σ – neutron cross-section absorption;
Φ – neutron flux in fuel

Calculations of campaign of point reactor model with these conditions are made with the program [6]. The program takes into account fission possibility for ^{238}U, ^{232}Th, ^{233}Pa, ^{234}U, ^{236}U, ^{237}U, ^{239}Np, ^{240}Pu, ^{242}Pu [5].

Results of reactor campaign calculation with 0.47 % initial contents of ^{235}U in fuel and absence of epithermal neutrons absorption in ^{238}U are shown at figure 2. Here (and everywhere in analogous cases) contents of fission nuclides (^{235}U и ^{239}Pu) is normalized on initial contents of ^{235}U in fuel, and fuel power – on its initial value.

For representative comparison of campaigns is made Table 1. It includes:

C FM - the content of the base fission materials at the campaign beginning, %;
F M – Fission materials in fuel;
R M – raw fuel nuclides;
Mode – campaign conducting features:
  Line – the simplest flow of the campaign (without fuel replacements);
  Sup Poz – campaign with the joint work of fuel with different burn-up (look at i.5);
  Sup Poz +^{233}U Gen – campaign with generation of ^{233}U (look at i. 6);
AKM – the absorption of neutrons in construction materials and leakage, %;
Φ – neutron flux, $sm^{-2}s^{-1}$;
T – campaign duration, hours;
Y – fuel burn-up, %.
Rmin – minimum operational reactivity during the campaign, %;
R SZ – integral reactivity at the end of campaign, %;
U3 – portion of ^{233}U fissions from total fissions;
U53 – portion of ^{235}U fissions, which formed from ^{233}U, from total fissions;
U55 – portion of ^{235}U fissions, which is from natural uranium, from total fissions;
Pu9 – portion of ^{239}Pu fissions from total fissions;
Pu1 – portion of ^{241}Pu fissions from total fissions;
Qu op –portion of raw uranium usage in open fuel cycle;
Qu sh – portion of raw uranium usage in closed fuel cycle;

№	C FM	F M	R M	Mode	AKM	Φ	T	Y	R_{min}	R SZ	U3	U53	U55	Pu9	Pu1	Qu op	Qu sh
1	0,47	U5	U8	Line	0.0	9+13	13	1.42	0.002	0.054			29.5	63.6	5.43	2.15	no
2	0,37	Pu9					13	1.63	0.008	0.021				76.6	23.40	-	100
3	1,65	U3	Th				39	8.78	0.019	0.028	90.9	9.10					
4	0,35	Pu 9, 1	U8				13	1.56	0.002	0.03				73.5	26.50		
5	1,65	U3, 5	Th				16	4.09	0.001	0.02							
6						5+13	38	4.89	0.028	0.046	90.4	9.60					
7	0,68	U + Pu	U8+Th			9+13	12	1.80	0.009	0.034	37.26	4.08		44.26	14.40		
8						5+13	25	2.15	0.008	0.030	40.54	3.98		42.33	13.14		
9	0,47	U5	U8	Super-position	1.7	9+13	25	2.52	0.013	0.018			17.52	68.37	14.10	3.81	-
10	0,712						34	3.92	0.010	0.027			17.07	69.05	13.87	3.92	
11	0,35	Pu 9, 1					15	1.79	0.0003	0.009				73.4	26.60	-	100
12	0,712	U5					29	3.38	0.006	0.024			19.78	67.31	12.90	3.38	-
13					2.8		24	2.84	0.008	0.029			23.52	64.91	11.55	2.84	
14					5.2		19	2.41	0.002	0.026			27.72	62.22	10.05	2.41	
15				Super-Position + U3 gene-ration	1.7		44	5.27	0.001	0.017	5.05	0.16	12.70	67.69	14.40	5.27	
16	0,47				0.0		34	3.47	0.012	0.027	5.27	0.13	12.70	64.30	17.60	5.26	
17	0,35	Pu 9, 1			1.7		27	3.13	0.001	0.012	4.10	0.10		72.83	22.99	-	100
18	,17+,35	U5+Pu 9, 1					29	3.51	0.001	0.017	1.89	0.04	4.69	71.22	22.16		14.28
19	,35+,35						34	4.22	0.005	0.024	2.12	0.06	7.78	69.14	20.90		8.58
20	,2+,41		U8(1.2)				24	3.49	0.004	0.014	2.28	0.04	5.52	69.62	22.54		12.12
21	,31+,39		U8(1.1)	Super-position			24	3.34	0.005	0.025			8.35	69.24	22.41		7.61
22	,35+,35		U8				24	3.08	0.016	0.037			10.69	67.46	21.84		6.27
23	0,712	U5	U8	Line	5.2	9+13	10	1.26	0.001	0.085			45.13	50.91	3.95	1.26	-
24				S P G			34	4.12	0.001	0.01	4.79	0.13	16.23	65.66	13.19	4.12	
25	1.65	U3, 5	Th		5.0	6+13	24	4.35	0.015	0.044	92.58	7.42				-	100

Table 1. Campaign characteristics

Reactor power with constant neutron flux increases during campaign. Power has peak of 28 % and at the end of campaign is 20 % greater than initial value. It is caused by ^{239}Pu accumulation, which has bigger fission cross-section than ^{235}U, and ^{239}Pu contents stabilization with decreasing portion of ^{235}U.

Contents of ^{239}Pu becomes stable during campaign but its value is less than initial contents of ^{235}U in fuel. It shows necessity of taking into account characteristics difference between fission nuclides at calculation of reproduction coefficient. With use of formula (2) and ^{235}U fission characteristics reproduction coefficient equal to unity is calculated. At the same conditions reproduction coefficient for ^{239}Pu is less unity (0.786).

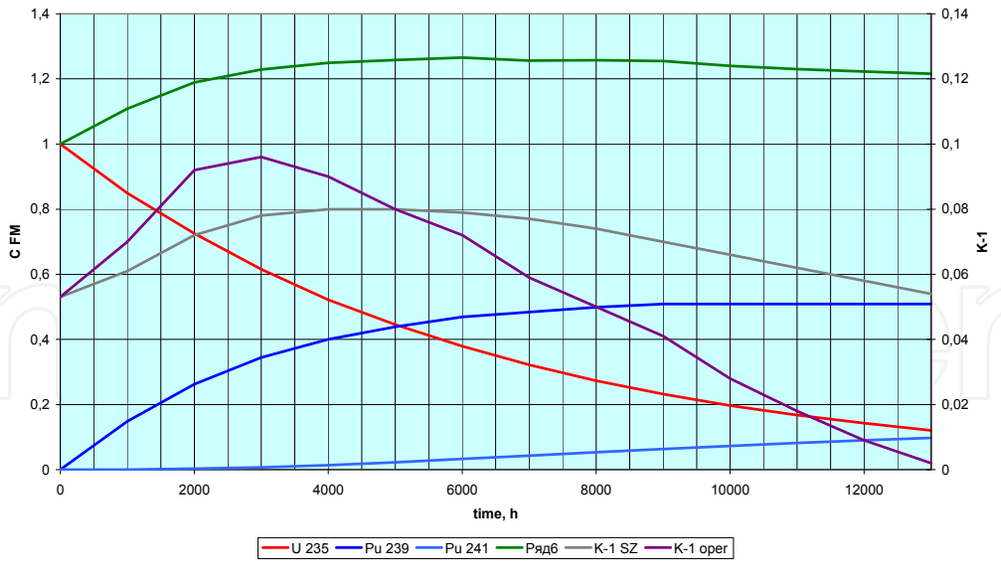


Figure 2. Reactor campaign characteristics with initial contents ^{235}U in fuel 0.47 % and absence of absorption in ^{238}U on epithermal neutrons. (string 1 of Attachment Table 1)

3.2. Fuel 238U + 239Pu

Region with reproduction coefficient equal unity for fuel on the base of mix ^{238}U + ^{239}Pu relocates to less value of fission materials contents comparing to fuel on the base of ^{238}U and ^{235}U.

Multiplication factors for RC=1 is increased.

Reactor campaign characteristics with initial fuel containing ^{238}U and ^{239}Pu are shown at figure 3.

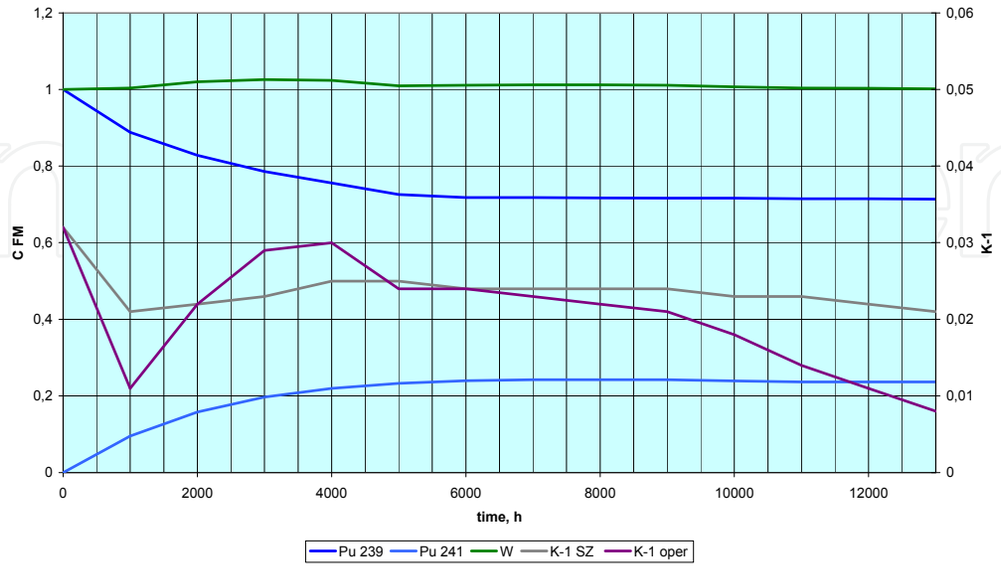


Figure 3. Reactor campaign characteristics with initial contents of ^{239}Pu in uranium-plutonium fuel 0.37 % and absence of absorption in ^{238}U of epithermal neutrons (string 2 of application's table 1).

Portion of ^{239}Pu is 0.37 % from total mass of these nuclides and in ^{238}U there is no absorptions in epithermal region.

Contents of ^{239}Pu during campaign is stable enough, but not equal to initial one and decreasing for ~25 %. Stabilization of ^{239}Pu is reached earlier than in the previous campaign. The role of ^{241}Pu increases. Its amount increases to ~ 1/3 of ^{239}Pu at stationary level. Value of reactivity margin during this campaign is less, but its fluctuation is also less. Despite less contents of fission material in initial fuel slightly higher burn-up is reached after the same work time.

Decreasing of ^{239}Pu amount and reactivity margin in the first hours of campaign is caused by delay of transformation of ^{239}U into ^{239}Pu, and significant neutron losses in the chain of ^{241}Pu. It is important to make comparison with fuel characteristics on the base of mix ^{233}U + ^{232}Th.

3.3. Fuel ^{232}Th + ^{233}U

For the fuel on the base of mix ^{232}Th + ^{233}U the region with reproduction coefficient equal to unity relocates to higher contents of fission materials. Multiplication factor in this region also shows increase comparing to variants with fuel on the base of ^{238}U and ^{235}U.

Reactor campaign characteristics with ^{233}U + ^{232}Th in initial fuel is shown at figure 4. Contents change of ^{233}U during campaign is not big. Reactor power change is also not big. But power is decreasing at the campaign beginning and after that returns to its initial value. Power decrease at the campaign beginning is caused by ^{233}U contents decrease, and return is caused by ^{235}U accumulation. Comparatively small accumulation of ^{235}U is well explained by small neutron absorption cross-section of ^{233}U, from which produces ^{235}U.

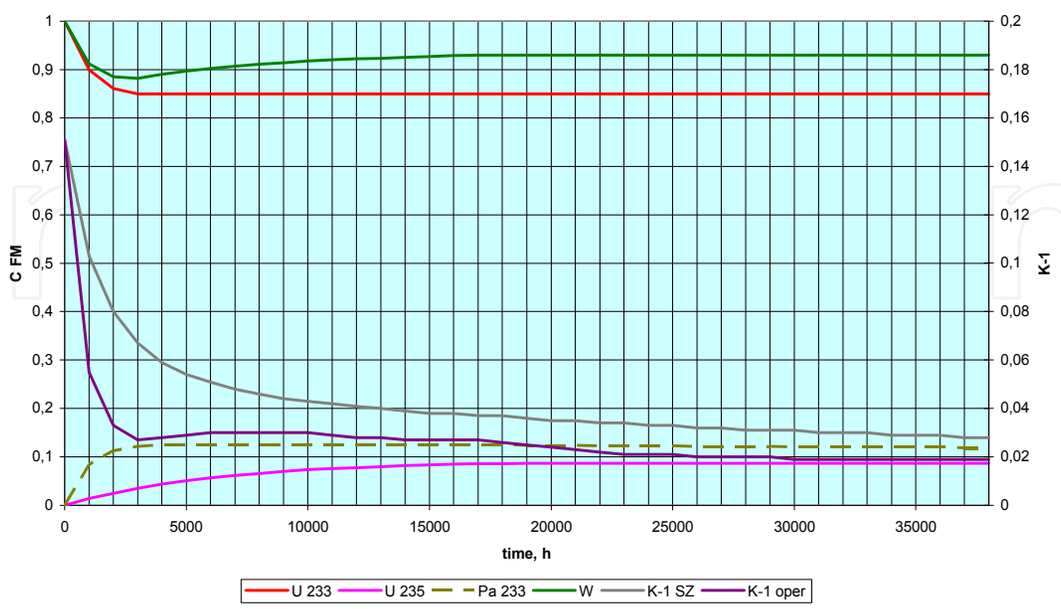


Figure 4. Reactor campaign characteristics with initial contents of ^{233}U in uranium-thorium fuel equal to 1.55 % (string 3 of application's table 1).

Positive reactivity margin in this campaign is decreasing at its beginning because of ^{233}Pa contents increase and its comparatively long half-life. After 2000 work hours reactivity fluctuation is small because all fuel nuclides has stabilized. The reached duration of this campaign is considerably higher than it of the previous campaigns.

4. Analysis of hypothetic reactors campaign

4.1. Variants of campaigns for figures 2, 3, 4

Let us compare some campaign characteristics of three shown variants of reactors with different fuel type, which are significant for its working out for applying in practice. This comparison is needed because the choice of initial conditions for campaign characteristics calculation is, in principle, not equivalent. Comparisons can delete this shortage.

Portion of raw uranium usage is usage rationality of this fuel kind (at this campaign). It is important because uranium is unique natural material containing fission isotope ^{235}U.

Only the first of three considered campaigns can be used in the open cycle, i.e. without nuclides, which were gotten from the spent fuel.

Closed fuel cycles with raw thorium or raw uranium can work without ^{235}U usage. Thus, the complete use of natural uranium, which used at the initial stages of these campaigns, is reached.

At the base of ^{238}U+^{235}U fuel simplest campaign cannot reach high raw uranium usage. Reproduction of fission materials is too small, neutron losses in reactor control elements is too big. Reached burn-up is minimal among all suggested variants.

4.2. Variants with equilibrium fuel

Characteristics of reactors campaign with initial contents of ^{239}Pu and ^{241}Pu sum in the uranium-plutonium fuel, which is equal to equilibrium, which can be formed from the campaign by picture 3, are presented at string 4 of application's table 1.

Fission reproduction coefficient in this campaign is equal to unity practically always during campaign. Neutron absorption in control rods increases up to 3.3 %. Reactor power is practically constant. Its decrease is not more than 1 %.

Optimization of campaign with fuel on the base ^{233}U + ^{232}Th by the same means is also interesting. These data are shown at string 5 of application's table 1.

Equilibrium of fission materials in the campaign with equilibrium contents of ^{233}U, ^{235}U and thorium is achieved after 2000 work hours. Neutron absorption in control rods is 5.8 %. Reactor power decreases to 93.2 % from initial value at 2000 work hours and remains stable. Power decrease is caused by ^{233}U formation and role of ^{233}Pa with long half-time.

Large reactivity margin with increase possibility of neutron losses in construction materials and for leakage is significant difference between this and previous campaign.

4.3. Change of resonant absorption in raw materials

As it can be seen from charts of figures 1 equal reproduction coefficient is achievable with different fission materials contents in fuel. It is done by changing of resonant absorption in raw nuclides. Described results are made for cases with no resonant absorption. Theoretically these characteristics remain constant at the same durability and neutron flux with increase of resonant absorption and achieving the same reproduction coefficient:

- portion of absorption in control rods;
- final contents of fission products;
- final value of reactivity.

Burn-up of fuel nuclides is changed. Reactor work with high burn-up is desirable. For campaign search with increased resonant absorption in raw nuclides data from figures 1 is not sufficient, because it is based on two nuclide campaign when its real number is six in campaign with uranium-plutonium fuel with limitation of ^{240}Pu.

Described cases are not common in reactors with low reproduction coefficient, where resonant absorption in raw nuclides leads to multiplication factor decrease in the campaign beginning and decreases campaign durability.

5. About possibility of practical campaign realization with high reproduction

The basic difference of real reactors is presence of neutron absorption in construction materials and neutron leakage from reactor. These factors can be researched in the described models without reference to the reactor design by additional term insertion for these neutron losses.

Neutron flux values used in previous calculations are not always applicable. Possibility of neutron flux change and its influence on campaign characteristics must be examined.

Arrangements for reactor campaign improvements and its effects should be also considered. Without reference to a reactor design following arrangements can be considered:

- combined use of several fuel kinds (from described above);
- fuel dynamic loading use, which suggests using of several equal fuel portion, each of them works in reactor during specified time, after that it is replaced with next fuel portion. In the cycle each portion is used once;
- zone superposition regime of campaigns with different start moments;
- insertion of excess of fission materials to initial fuel;
- fission materials production with use of control rods with raw nuclides addition (^{232}Th and ^{238}U).

5.1. Neutron flux influence on campaign characteristics

Campaign characteristics calculations, which were presented above, use ~$9*10^{13}$ $sm^{-2}sec^{-1}$ neutron flux. Flux increase allows improving economical issues – decrease of fuel

requirement at specified power output. It is especially valuable for the case when during campaign ratio of neutron flux to power is almost constant.

Campaign characteristics calculations for above mentioned fuel types with neutron flux in diapason from $2.5*10^{13}$ to $2.0*10^{14}$ $sm^{-2}s^{-1}$ and campaign durability from 5000 to 40000 hour were carried out. At flux from $2.5*10^{13}$ to 10^{14} $sm^{-2}s^{-1}$ change in final mass of fission materials, control rods absorption and reactivity at the end of campaign are slight for all fuel kinds. Change of its parameters for fuel with raw ^{238}U at neutron flux $2*10^{14}$ $sm^{-2}s^{-1}$ is slight.

Final reactivity for thorium containing fuel under the neutron fluxes more than 10^{14} $sm^{-2}sec^{-1}$ is fast becoming less than zero [7]. Flux rising influence is could be watched at string 5 and 6 of application's table 1, where the one fuel type is used, but it has the different neutrons fluxes in $9*10^{13}$ $sm^{-2}sec^{-1}$ and $5*10^{13}$ $sm^{-2}sec^{-1}$. At larger flux reactivity is less than zero after 16500 hours, and at smaller flux it is near the 0.03 after 39000 hours. By estimation it becomes zero after 50000 hours of work reactor.

These effects are explained by neutron absorption in ^{233}Pa, which has comparatively high half-time period (27.4 days) and large absorption cross-section (66 barn). However, not everything is so simple. The reactivity in campaign with non-equilibrium fuel (figure 4) and $9*10^{13}$ $sm^{-2}sec^{-1}$ flux remains high during 39000 hours.

5.2. Combined work of several fuel types

Reactor campaign characteristics with 0.64 % initial contents of mix ^{233}U, ^{235}U, ^{239}Pu and ^{241}Pu in uranium-thorium fuel with ^{238}U – 75 % and $9*10^{13}$ $sm^{-2}s^{-1}$ neutron flux are shown at string 7 of application's table 1.

Contents of fission materials is corresponding to its equilibrium contents. Reproduction of fission nuclides is close to unity. Portions of energy release from fission components are:

$$^{233}U - 39.76\ \%;\ ^{235}U - 4.26\ \%;\ ^{239}Pu - 42.21\ \%;\ ^{241}Pu - 13.76\ \%.$$

That means in fission components from raw ^{238}U occurs ~56 % fissions, and in fission components from raw ^{232}Th ~44 % fissions.

Positive reactivity at the campaign end in this variant is less than with equilibrium fuel on the base of ^{232}Th and ^{233}U. Campaign with the same fuel, but $5*10^{13}$ $sm^{-2}s^{-1}$ flux is presented at string 8 of application's table 1. Reached burn-ups in the both cases are close to each other and it is says about decreasing of negative role of thorium nuclides component fuel.

5.3. Dynamic loading regime use

Most part of absorptions in fission products is in ^{135}Xe. Absorption in ^{135}Xe in high neutron flux (more than $5*10^{13}$ $sm^{-2}s^{-1}$) is close to portion of ^{135}I formation from fission products. Portion of ^{135}I formation is ~6 %, and total absorption in all fission products is slightly above 10 %.

Peculiarity of ^{135}Xe is its formation from ^{135}I with half-life 6 hours and decay with 9 hours half-life.

In dynamic loading regime [7] fuel works in reactor during time close to ^{135}I half-life, formation of ^{135}Xe and its neutron absorption is minimal. During fuel exposition out of core ^{135}Xe is decaying.

Dependence of neutron absorption portion in ^{135}Xe, which formed during fuel work on work duration and neutron flux and portion of remaining ^{135}Xe after fuel exposition out of core is shown on figure 5.

It can be seen that even in maximal neutron flux (10^{14} $sm^{-2}s^{-1}$) neutron absorption in ^{135}Xe is 30 % at work time about 8-10 hours.

We can note that portion of remaining ^{135}Xe after fuel exposition practically does not depend on loading characteristics of fuel in reactor.

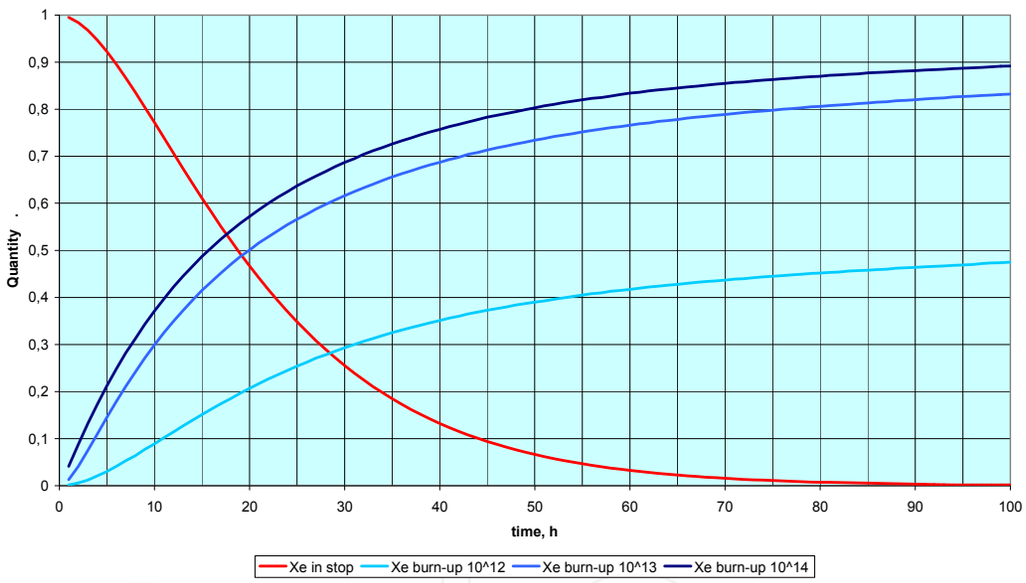


Figure 5. Dependence of ratio neutron absorption in ^{135}Xe to ^{135}I formation on fuel work time and neutron flux.

Data from charts on figure 5 can be used for estimation of ^{135}Xe neutron absorption at dynamic loading regime. For example, in neutron flux $5*10^{13}sm^{-2}s^{-1}$ with 5 work hours and exposition 45 hours neutron absorption in ^{135}Xe is less than 30% of fission product ^{135}I formation and is about ~1.9 %. Saved 4 % of neutrons can be used for construction material absorption and leakage.

For effective usage of dynamic loading duration of working regime can be in the region of 5 to 10 hours and exposition time – 35-50 hours. Increasing exposition time more than 60 hours is unreasonable. So fuel mass can be 3.5 – 5.0 times larger.

Dynamic loading regime for traditional fuel types with rigid placed fuel assemblies in a core is sufficiently complicated. In general it can be made in reactors, which have fuel assemblies' replacements without reactor shut down, such as CANDU or RBMK-1000. But large amount of replacements, needed for this regime realization, decreases durability of these fuel assemblies and replacement mechanism.

Dynamic loading regime is possible in molted salt reactors [8] and in reactors with spherical fuel circulation in heat-exchange loop [9]. Work [10] shows that this regime can significantly simplify a molted salt reactor technology of fuel purification from fission products and actinides.

6. Joint work of zones with different campaign moments in the same core

6.1. Ideal reactor with zero loss of neutrons

Modern power thermal neutron reactors widely use campaigns with multiple fuel reloading [11, 12]. During each reloading fuel with maximal burn-up is moved out core, fuel with different burn-up is rearranged and fresh fuel is loaded. Necessity of reloading is caused by fission materials concentration decrease during irradiation.

It is shown in presented above data that high concentration of fission materials is stabilized in high fission materials reproduction reactors during campaign burn-up. Burn-up increase in campaign of such reactors is purpose of multiple fuel replacements with entering regime of negative reactivity in some part of core with maximal burn-up.

Terms of detailed campaign and compact campaign are introduced here for better understanding. Detailed campaign is theoretically calculated campaign with continuous reactor work with use of negative reactivity. Compact campaign contains the portions of various fuels with different burn-up level. Durability of compact campaign is equal durability of detailed campaign divided on number of fuel zones with different campaign burn-up. For work regime with fuel replacements is used term "zone superposition".

It can be noted that work in compact regime is possible only if neutron absorption in control elements in detailed campaign is more than zero.

Fuel with nuclides ^{235}U and ^{238}U in zone superposition regime is considered. Figure 6 shows detailed campaign characteristics of initial fuel ^{235}U (0.47 %) + ^{238}U in neutron flux $9*10^{13}$ $sm^{-2}s^{-1}$ and multiplication factor of compact campaign with four fuel replacements. In superposition regime fuel works in reactor for 25000 hours. Raw uranium usage in campaign increases up to 3.81%. Multiplication coefficient at the end of detailed campaign becomes lower than unity by 0.04.

Characteristics of campaign with the same fuel type and different initial contents of fission materials in the beginning of campaign are presented at the string 10 of Attachment Table 1. Here it is corresponds to natural uranium fission materials contents. Larger burn-up and raw uranium usage in open fuel cycle is obtained in this variant.

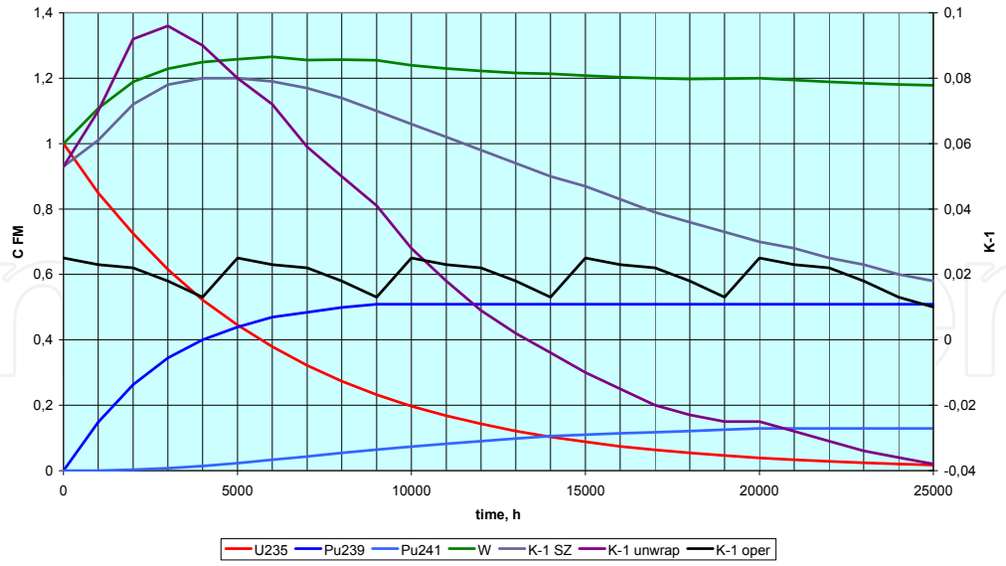


Figure 6. Detailed campaign characteristics with initial fuel $^{235}U + ^{238}U$ and multiplication factor of compact campaign with four fuel replacements (string 9 of Attachment Table 1).

6.2. Reactors with non-zero loss of neutrons

The string 11 of Attachment Table 1 shows characteristics of detailed campaign with fuel $^{239}Pu + ^{239}Pu + ^{238}U$ and multiplication factor of compact campaign with eight fuel replacements. Neutron loss in construction materials and leakage is 1.7% in this campaign.

This campaign is identical to the campaign, which is presented at the string 2 of Attachment Table 1 and figure 3, by the fuel type and composition. In spite of neutron loss presence larger campaign duration and burn-up is obtained in the campaign.

Comparison of campaign characteristics with natural uranium and different neutrons loss in the construction materials and leakage is presented at the strings 10, 12, 13, and 14 of Attachment Table 1. Selected data from this table is presented in Table 2.

1	Neutron loss, %	0	1.7	2.8	5.2
2	The duration of the campaign, hour	34 000	29 000	24 000	19 000
3	Burn-up, %	3,92	3,38	2,84	2,41

Table 2. Comparison of campaign characteristics with natural uranium and different neutron loss in zone superposition regime.

Characteristics of reactor campaign fueled with natural uranium and neutron loss 5.2% is presented at the string 14 of Attachment Table 1, and at the string 12 of Attachment Table 1– campaign with the same fuel and neutron loss 1.7%.

^{241}Pu production in reactor campaign with 5.2 % neutron loss is only reaching stationary level. Stationary level of ^{241}Pu production in the campaign with 1.7 % neutron loss is reached.

For comparison, characteristics of reactor campaign with the same fuel and 5.2 % neutron loss but without zone superposition using are presented at the string 23 of Attachment Table 1. Such campaign has the worst characteristics among others by campaign duration, fuel burn-up and raw uranium usage, and Rsz value – average neutron loss in control elements during campaign.

In reality campaign at string 23 of Attachment Table 1should be finished at least at 1000 hours earlier on condition of sufficient positive reactivity. Characteristics of this campaign are quite close to characteristics of CANDU reactor campaign.

It should be noted, that calculations shown in the present work are made for point model of reactor. In cases, when fission materials contents change at campaigns is minimal, real reactor campaign characteristics matches this calculation in great extent.

7. Fission materials production in control elements

7.1. Task formulation

Previous chapter material analysis shows that campaign characteristics improving, including the reproduction coefficient, with the same fuel type is obtained by decrease of neutron absorption in control elements. Improvement not always can be made by neutron flux change or superposition regime implementation. It is interesting to replace useless neutron absorption in control elements (cadmium, boron, gadolinium etc.) by absorption in raw fuel materials (^{238}U and ^{232}Th). First activation products of raw nuclides must be deleted from the core. This technology should not permit fission materials formation in chain of product transformation in the core. Activation chains of ^{238}U and ^{232}Th are:

$$^{238}\mathrm{U} + \mathrm{n} \rightarrow ^{239}\mathrm{U}\left(1412\ \mathrm{s}\right) \rightarrow ^{239}\mathrm{Np}\left(2.33\ \mathrm{d}\right) \rightarrow ^{239}\mathrm{Pu}; \tag{8}$$

$$^{232}\mathrm{Th} + \mathrm{n} \rightarrow ^{233}\mathrm{Th}\left(1325\ \mathrm{s}\right) \rightarrow ^{233}\mathrm{Pa}\left(27.4\ \mathrm{d}\right) \rightarrow ^{233}\mathrm{U}; \tag{9}$$

Uranium chain has the minor time reserve, but it is big enough.

Any technology always supplies some part of final fission product into the absorber placing area. It is simple to estimate the negative effect of such penetration. It is defined by purification time outside core from primary activation products (^{239}Np and ^{233}Pa) and degree of fission materials penetration into core loop.

It can be noticed that equal effect from absorption in raw and fission nuclides of uranium chain is obtained at raw to fission nuclides contents ratio 530, and for thorium chain it is equal to 182. Comparison of fission nuclide half-life periods, ratios of raw and fission components with same influence to reactivity, and fission characteristics (^{233}U has larger part of fissions from total neutron absorptions) shows, that thorium is preferable primary candidate for control element development with fission material production. Estimation effectiveness can be made by division of precursor half-time to ratio of raw and fission

components with same influence to reactivity. For thorium this parameter is 3.6 hours, and for uranium – 6.33 minutes. Difference is 36 times.

This suggestion realization does not mean that all control elements must produce fission components. There is no need to have it in safety system with total neutron absorption at zero level in any case of campaign.

At initial development stage it is possible to divide two functions of control system: traditional, for fast regulation of reactor power with efficiency of 1 β and slow regulation system with fission materials production and efficiency of maximal reactivity in campaign.

Part of neutrons involved in fission materials production is linked with produced fission material amount and energy emitted in core by formula:

$$n = Q * g * \xi; \quad (10)$$

Where:

n – amount of produced fission materials, nuclei;
Q – fission energy emitted in core, J;
g – link coefficient between emitted power and fissions number ($3.1*10^{10}$), J^{-1};
ξ – part of neutrons, which are involved into fission materials production.

With core power 1000 MW and 1 % of involved in fission materials production neutrons amount of produced fission material is equal to $1.2*10^{-4}$ g per second. For year mass is ~3800 g.

It is possible to use value of fuel burn-up and number of fuel reloading during the campaign. Insertion of produced fission materials is made with each new fuel portion. In this case the number of inserted nuclei of produced fission materials is described by formula:

$$n1 = m * \chi / j; \quad (11)$$

Where:

n1 – amount of nuclei of produced fission materials, which are inserted with each fuel reload;
m – amount of fuel nuclei, including raw and fission components;
χ – fuel burn-up during campaign;
j – number of fuel reloads during campaign.

Calculation of campaign characteristics with fission materials production by use of redundant reactivity compensation is made by iterative method. Adsorption of redundant neutrons at this process leads to additional reactivity insertion. At the good realization of campaign this additional reactivity is added to areas of detailed campaign with negative reactivity, which supplies prolongation of initial fuel work duration. Iterations in carried out calculations are not always optimal. Measures of optimal iteration are values of R_{min} and R SZ, which are shown in the Attachment Table 1.

7.2. Fuel is natural uranium

Let's examine how neutron loss in control elements used for generation of ^{235}U in reactor with initial fuel ^{235}U + ^{238}U influence on campaign characteristics (figures 10, 12 - 14). Characteristics of detailed campaign with 10^{14} $sm^{-2}sec^{-1}$ neutron flux, ^{233}U production and 1,7 % neutron loss are shown at the string 15 of Attachment Table 1.

Maximum fuel burn-up (5.27%) is reached in this campaign in comparison with previous campaigns; duration of detailed campaign is 34000 hours. Reactivity change during work is close to optimum results (K-1 from 1 % up to 3.2 %). Reactor power is stable. Reactor power change in constant neutron flux is not exceeding 2.5% from the average. ^{233}U generation increase in the campaign with this fuel type is possible at the expense of operating reactivity pike lowering.

Enough good indexes by this technology can be received into the reactor with 5.2% neutron loss. Description of such campaign is presented in string 24 of Attachment Table 1. Indexes of such campaign are high enough.

8. Characteristics of reactors with high fission materials reproduction

8.1. Possibility of neutron loss decrease

Obtaining of high fission materials reproduction is possible only in reactors, which have the minimal neutron loss in construction materials and leakage. This loss must be about 1-2% by the preliminary analysis of previous materials. It is not possible to realize on practice all of the discussed fuel cycles.

Let us consider in this chapter could be achieved good results with bigger neutron loss level and what must be considered as good results.

CANDU (and its many versions) with heavy water as coolant and moderator, with zirconium shells and channel walls is the nearest reactor to high fission materials reproduction reactors. Let's look at the characteristics of this reactor and its possible modifications, which are directed to neutron loss decrease. Fuel assembly of CANDU reactor with 37 fuel element (on left) and modification of this assembly with replacement of 7 central fuel element by beryllium block are presented in the figure 7.

Characteristics of initial reactor and its 6 modifications are presented at the table 3:

1. Reactor with fuel assembly at the figure 19 (on the left) with standard reflector, standard fuel on the basis of natural uranium dioxide;
2. Reactor by the i.1, with enriched by isotopes ^{90}Zr and ^{120}Sn in fuel rod shells and fuel assembly casing zirconium and tin.
3. Reactor by the i.2, with bigger thickness of heavy-water reflector.
4. Reactor by the i.2, with addition of graphite reflector to heavy-water reflector.
5. Reactor by the i.4, with fuel rods made of metallic uranium (80% of volume) and bismuth (20% of volume).

6. Reactor by the i.5, with fuel assembly shown at the figure 19 (on the right) with beryllium insertion.
7. Reactor by the i.6, which differs by using of heavy water instead of graphite layer in reflector.

№	Parameter\variant	1	2	3	4	5	6	7
1	Leakage	1.7904	1.7789	1.1121	1.1354	0.915	1.041	1.182
2	D	0,163	0.172	0.391	0.221	0.147	0.213	0.254
3	Be						- 0,513	- 0,531
4	C				0.356	0,285	0,327	
5	O	0,443	0,441	0,489	0,456	0,396	0,399	0,410
6	Zr	2,54						
7	90Zr		0.281	0.283	0.281	0.236	0.240	0.239
8	Sn	0,125						
9	120Sn		0.0298	0.0287	0.0319	0.023	0.024	0.0254
10	Bi					0.079	0.078	0.0772
11	∑	5,061	2,70	2,30	2,481	2,081	1,809	1,660
12	K eff	1,1171	1.1473	1.15489	1.15093	1.1265	1.13233	1.13428
13	RC	0,808	0,804	0,80	0.802	0,871	0,855	0,853
	Reflector	D2O 21		D2O 62	D2O+C			D2O 62
	Cell	37 fuel rods					30 fuel rods + Be	
	Fuel	UO2				0.8*U + 0.2*Bi		

Table 3. Neutron loss (string 1-11, %), multiplication factor and reproduction coefficient (string 12, 13, relative units) and fuel types and reflector types in variants of CANDU reactor models.

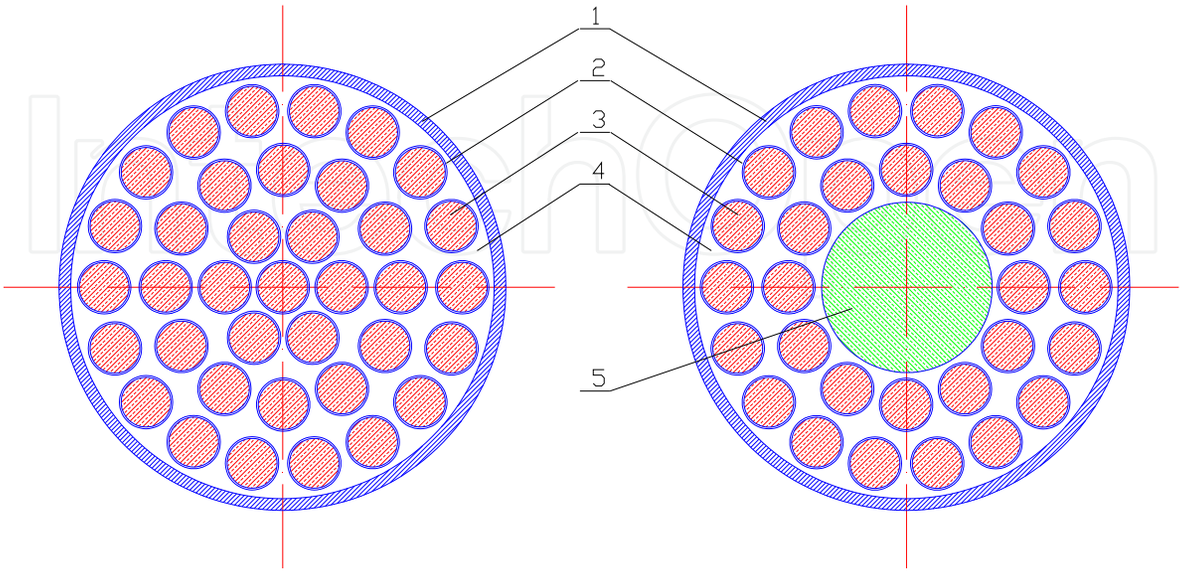


Figure 7. Fuel assembly of CANDU reactor (on the left) and its modification with beryllium insertion (on the right). 1 – fuel assembly casing; 2 – fuel element shell; 3 – fuel rods; 4 – coolant; 5 – beryllium insertion.

Analysis of calculation results allows saying:

1. Initial variant of reactor has the worst characteristics;
2. Major change of properties is reached by the change of natural zirconium and tin to its isotopes ^{90}Zr и ^{120}Sn. These isotopes have the best properties among the rest of isotopes of these elements and have the considerable contents in natural elements. Neutrons loss of this changing has decrease till 5.06% to 2.70%.
3. Decreasing of leakage at the expense of reflector thickness increasing at the i.3 and 4 also leads to appreciable lowering of neutrons loss.
4. Important improvement of characteristics is obtained by replacement of oxide fuel by metallic, including the reproduction of fission materials.
5. The best parameters among the presented variants have the reactor, with fuel assembly made from 30 fuel rods and beryllium insertion. Differences between 6 and 7 variants, which is differ by reflector construction, are minor.

1.7 %, 2.8 % and 5.2 % neutron loss are used in Table 1 as results of CANDU reactor calculation.

8.2. Construction of fuel assembly with composite core

Good neutron-physical characteristics of metallic fuel are well-known [13, 14]. Such fuel is used on reactors by the first stages of atomic energetic progress. Significant swelling under the enough small burn-up is the lack of it. Maximum allowed burn-up of "KC-150" [15] reactor is settled by the level in 15 MW*day/kg. It is indicated that this burn-up equal to 4 MW*day/kg the maximal extension of fuel element is laying down for 5-7% under the temperature to 350^0C. Rod-shaped fuel element of traditional construction with such characteristics of swelling cannot be perspective for reactors with burn-up of 40 MW*day/kg and higher.

Situation can be changed, if we will use the fuel element with composite metal core. 2 variants of such fuel element construction are presented at the figure 8. Core of such fuel element is containing fuel elements 2 and 3, liquid-metal filler 4, which are placed under the protection cover.

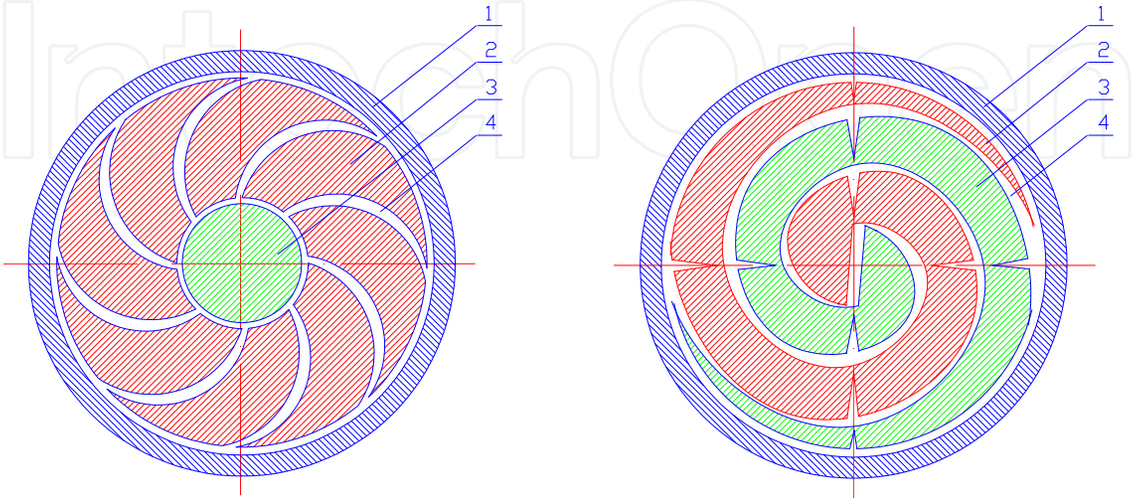
In variant, which is placed on the left, the core contains 8 leaf fuel cells and one cylindrical fuel element. These elements at swelling occupy the space, which was filled by liquid metal (with fuel working temperature). At the second variant only 2 fuel elements are used. These elements are identical and inserted into each other. Cuttings, which bring down the pressure during swelling in radial direction, are made on elements surfaces.

Bismuth, lead [16] and tin can be filler material. If cover is made from zirconium, then lead, by preliminary estimations, leaves the list of candidates because it actively interacts with zirconium and breaks its initial structure. It is possible, that bismuth in clean type will actively interact with zirconium with such result.

Tin at the time of interaction with zirconium forms the solid compound – zirconium stannide Zr_3Sn_2 on its surface, which is melted at temperature 1985 ^{0}C [17]. Tin is included into the composition of zircaloy, which is used in fuel element covers. Shortage of tin (its

enough high absorption cross-section) is possible to delete by using tin enriched by ^{120}Sn isotope. ^{120}Sn has the best properties among tin isotopes with even atomic weight.

It is possible, that alloy of tin with bismuth and lead also will form the tin stannide on the surface of zirconium, which prevents the interaction of zirconium with bismuth and lead. Rigorous research in this direction is necessary.



1 – fuel element cover; 2 and 3 – fuel elements; 4 – filler.

Figure 8. Variants of fuel element construction with composite metallic core.

We should specify requests to whole of possible varieties of core composite elements, which must have high workability of fuel element with maximal level of burn-up.

In this case the statement is follows: space between particles must create stable conditions for separate particles location under fuel element cover when increasing volume in specified limits without appearing of additional effort between particles;

This claim dissects for some independent claims:

- Allowed reduction of areas gage, which is occupied by composite core, is settled;
- In the initial condition surface of particles is must be distant from each other by the all surface of particles;
- Crush elements might be placed between particles without changing the distance between them for realization of increasing of particle sizes;
- In the working progress, under the decreasing of crushed elements dimensions it is desirable, that the distance between particles centers is staying invariable or changing negligible.

It should be marked that important factor in such construction of fuel element, which decrease the metallic fuel elements swelling, is their small thickness. The small thickness increases the migration of fission gaseous splinters over the external surface with decreasing of inner mechanical stresses.

By the estimations, forms changing of metallic fuel rods under the execution of given claims and initial content of liquid-metal filler in 20% by the volume is not leading to trespassing of

fuel element cover under the burn-up till 40-59 MW*day/kg. Such fuel has being used in calculations of reactor models, which was presented in the table 1.

8.3. Open fuel cycle

Open fuel cycle can be realized by the initial fuel, which consists of ^{235}U and ^{238}U. This fuel can contain natural uranium, as in CANDU reactors, enriched uranium, as in the majority of modern reactors, and, as it presented on the calculation above, from the ^{235}U impoverished. Fission materials in all of these cases are made from natural uranium passing the chains, which is bounded with recycling of spent fuel. Let's assume that technology of ^{233}U generation is not bounded with recycling of spent fuel and might take the part in opened fuel cycle.

The final product of open fuel cycle can be the perfect base or composite part of closed fuel cycle. Especially, if fission isotopes of plutonium come to equilibrium.

Open cycles, which are presented in strings 1, 9, 10, 12 -15, 17, 23 and 24 of Attachment Table 1, are described above. Maximum portion of natural uranium usage on it is reached under the conduction of superposition mode areas with ^{233}U generation. Quite high part of using is reached in the superposition mode under the decreased neutrons losses in construction materials and for leakage. In these cases fission isotopes of plutonium are in equilibrium at campaign end. Maximal reached part of natural uranium using in presented cycles forms 5.27%. This amount in cycles with generation of ^{233}U can be increased at the expense of generation mode optimization. Optimization is not appeared by paramount task at this work.

Thereby, there are different ways for reaching of high natural uranium usage in open fuel cycles and reception of closed fuel cycles fission components on its.

8.4. Closed fuel cycles.

In this work it is considered that all initial fuel products use technology of spent fuel recycling. In closed cycles it is possible to use raw uranium based fuel or raw thorium based fuel.

8.4.1. Uranium-plutonium fuel

The best closed fuel cycle characteristics are achieved with use of equilibrium contents of fission materials. But the ideal equilibrium uranium-plutonium cycle has very small contents of fission materials so the neutron leakage plays more important role than in CANDU with raw uranium.

Fission materials contents in equilibrium uranium-plutonium fuel equals 0.34 % in condition of resonant neutrons absorption absence in ^{238}U (in CANDU its contents is more than twice larger – 0.712 %).

Solution of the task is possible in two ways – 1) by increasing of initial contents of ^{235}U by adding it, 2) increasing of fission materials contents by increasing resonant absorption in ^{238}U.

The string 19 of Attachment Table 1shows variant with addition of 0.35 % ^{235}U from mass of ^{238}U, that equals equilibrium contents of ^{239}Pu and ^{241}Pu. This fuel can be made by plutonium isotopes, which are extracted from spent fuel, addition to a mix of raw uranium (half of ^{238}U mass) and ^{238}U from spent fuel.

The string 20 of Attachment Table 1 shows campaign characteristics with initial fuel containing equilibrium contents of ^{239}Pu and ^{241}Pu and addition of ^{235}U, which contents is much less than in previous fuel. Resonant neutron absorption in ^{238}U is 1.2 times higher. Production of ^{233}U is lead. Equilibrium contents of ^{239}Pu and ^{241}Pu is also 1.2 times greater.

Burn-up of 3.49 % and raw uranium usage 12.12 % is achieved.

String 18 of Table 1 of Attachment shows campaign with this technology. Strings 21 and 22 show campaign with no ^{233}U production. Maximal burn-up 3.51 % and maximal raw uranium usage 14.28 % is achieved.

8.4.2. Uranium-thorium fuel

Equilibrium cycle with uranium-thorium fuel even without resonant neutron absorption in thorium has high fission materials contents, which mean possibility of high burn-up achieving in detailed campaign. Features of uranium-thorium campaign are high neutron absorption in ^{233}Pa and its long half-life, which are used in technology of ^{233}U production by reactivity margin decreasing.

Detailed campaign with neutron flux about 10^{14} $sm^{-2}s^{-1}$ becomes very short even in case of low neutron losses which equals 1.7 %. The campaign with twice lower neutron flux is possible at neutron losses of 5.0 %.

Figure 9 shows detailed campaign characteristics with equilibrium uranium-thorium fuel with ^{233}U production at neutron flux $6*10^{14}$ $sm^{-2}s^{-1}$ and neutron losses 5.0 %.

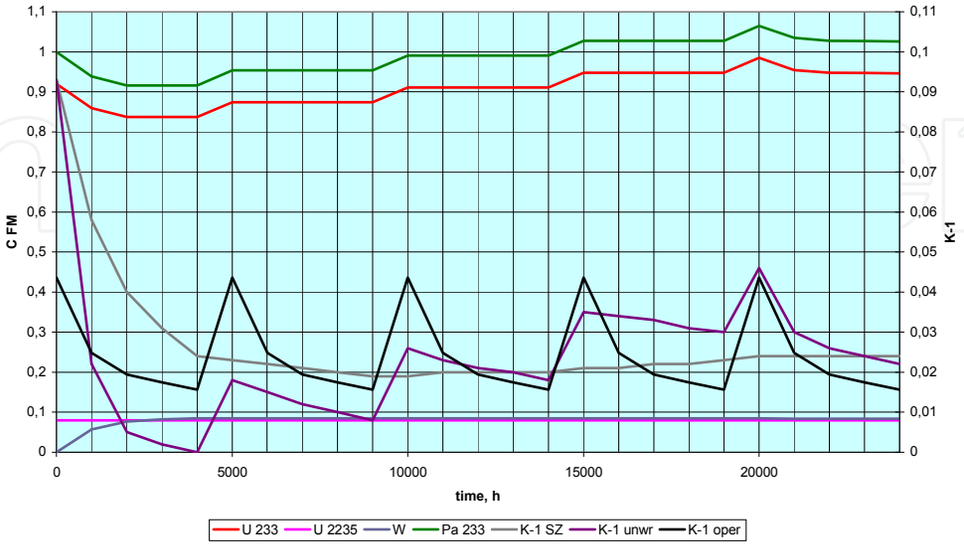


Figure 9. Detailed campaign characteristics with equilibrium uranium-thorium fuel and ^{233}U production. Neutron flux is $6*10^{14}$ $sm^{-2}s^{-1}$, neutron loss – 5.0 % (string 25 of Table 1 of Attachment).

High stability of reactor power is obtained in compact campaign. Reactivity margin decrease in detailed campaign is observed in first 5000 hours. During all time of detailed campaign reactivity stays positive. Reactor work prolongation over 24000 hours is possible. At the shown campaign burn-up of 4.35% is reached.

The difference of the campaign from uranium-plutonium campaign is possibility of significant amount increase of fission materials to the end of the campaign. Reproduction coefficient of the campaign, which is shown at figure 20, is more than unity by 11 %.

8.4.3. Mixed fuel

Mixed fuel (with uranium-thorium raw materials) is rational to use in closed cycles with equilibrium between fission and raw materials.

This fuel is less dependent from neutron flux than pure thorium fuel. Together with it the fuel is more sensitive to the neutron losses. At ^{238}U contents 75 % in campaign beginning, fission materials contents 1.16 % neutron losses 2.8% neutron flux less than $8*10^{13}$ $sm^{-2}s^{-1}$ is possible effective work. The duration of the campaign can be 34000 hours.

Campaign with mixed fuel can supply simultaneous use of thorium and uranium fuel with burn-up corresponding to its natural contents with high and even full use of raw uranium [18].

Closed fuel cycles with uranium fuel after several cycles can have increased contents of ^{242}Pu, which is feeble burning out and being absorber and filler decreasing of campaign effectiveness. The possible way of solving is plutonium isotope separation. In mixed fuel ^{242}Pu accumulation will be less and its critical contents will be reached much later. The losses for it are also less.

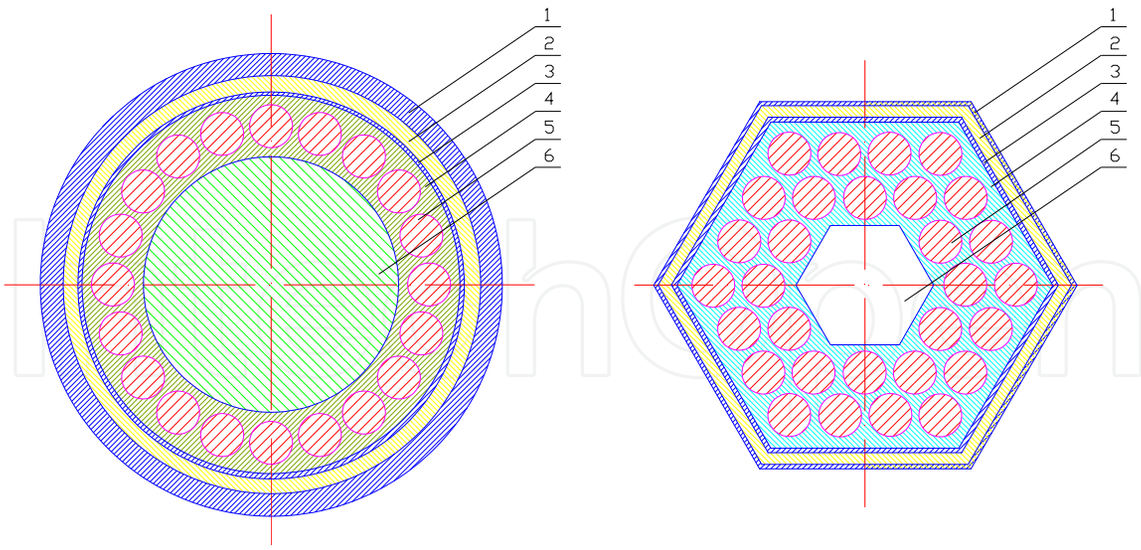
8.5. About neutron balance in the campaign

In previous materials model of abstract reactor with parameter "neutron losses in constructive materials and leakage" is used. Dependence of reactor characteristics and its campaign is more complex in real life.

More detailed presentation of these relations can be obtained from neutron balance during reactor work. Neutron creation is present not only in fission nuclides but in raw fuel components and such constructive materials as heavy water and beryllium.

Figure 11 show neutron balance in reactors with two types of fuel assemblies with different coolants. Fuel assembly for heavy water coolant is shown at figure 10-a, and fuel assembly for liquid metal – at figure 10- b. Measures for neutron energy loss prevention at fuel location are used in fuel assembly for liquid metal coolant.

Reactor calculations with use of program [19] are conducted, which are the base for reactor campaign calculations with use of program [20].



1 – case of fuel assembly, 2 – gaseous gap, 3 – screen, 4 – coolant, 5 – fuel rod, 6 – beryllium insert for a) and gaseous cavity for b).

Figure 10. Fuel assemblies with water (a) and liquid metal (b) coolant.

Neutron absorptions (black columns) neutron fissions (blue columns) in different fuel nuclides are shown on diagrams. Two types are used ^{235}U – natural raw uranium signed as U235N, and formed during transformations of thorium chain (^{232}Th-^{233}Pa-^{233}U-^{234}U-^{235}U), signed as U235S.

Fission nuclides have columns for difference between secondary neutrons and total neutron absorptions (red columns). Difference for raw ^{232}Th and ^{238}U between number of secondary neutrons and number of total absorptions with fission are indicated in yellow columns. Columns 1 and 2 indicate neutron absorptions in construction materials and fission products correspondingly.

In left part of each column lay data for reactor with liquid metal coolant, in right part – data for reactor with water coolant/

Total height of red and yellow columns must be equal to total height of black columns for nuclides ^{232}Th, ^{238}U, ^{234}U, ^{240}Pu and columns for neutron absorptions in construction materials and fission products.

If neutron loss in construction materials and leakage are less, than more neutrons can be absorbed in fission products, which number increases during campaign.

For the compared variants fission activity in raw nuclides and ratio of absorptions and fissions in ^{241}Pu are better. In result total neutron absorption in fission products (8 %) for heavy water coolant reactor is less than total neutron absorption in fission products (13 %) for liquid metal coolant reactor at the same neutron losses for construction materials and leakage. So reactor with liquid metal coolant has campaign with burn-up 25.5 MW*day/kg when heavy water coolant reactor has campaign with burn-up 11,3 MW*day/kg. Liquid metal cooled reactor has higher raw uranium usage in open cycle campaign.

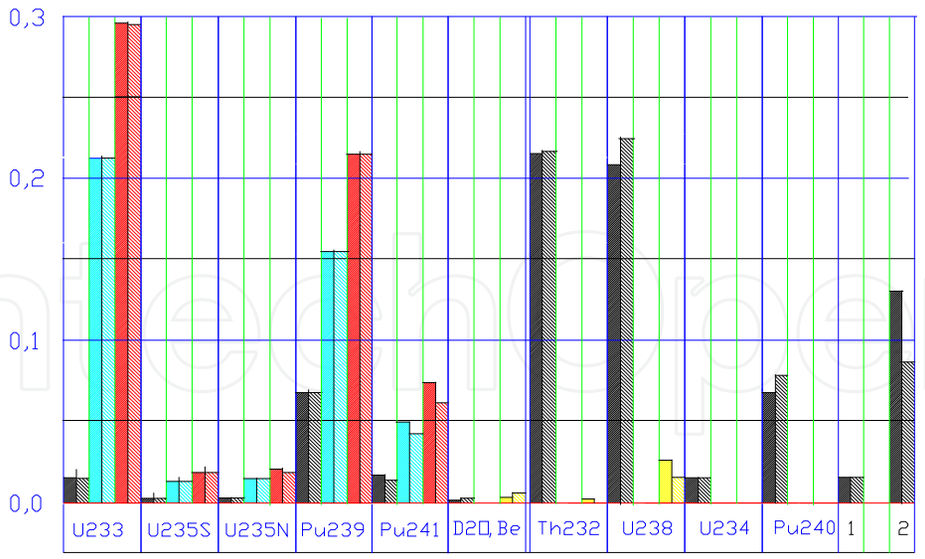


Figure 11. Liquid metal and heavy water cooled reactors neutron balance in campaign with equilibrium by fuel nuclides regime.

8.6. External neutron sources for reproduction coefficient increase in thermal reactor

External neutron sources can be included in neutron balance if it is linked somehow with reactor work. There are schemes, where neutrons, which are formed in reactions between accelerated protons and nuclei of heavy metals, are emitted to the reactor's core. This scheme has significant lack of reactor work reliability because of use less reliable source – electric nuclear installation. When autonomic electric nuclear installation is used only for ^{233}U formation by neutron absorptions in thorium, reactor work almost is not depending on work reliability of electric nuclear installation. Work [21] demonstrates that raw uranium usage can be increased up to 100 % with use of comparatively small energy expenses (less than 2,5 % of produced energy).

Note, that modern thermal reactors, which use less than 1% or raw uranium, about 1% of produced energy spent for fuel enrichment.

Possible characteristics of joint work of fission reactor and neutron source based on fusion reactor are shown [22].

9. About development directions of nuclear reactors and devices for energy transformation

Achieving of good results of raw uranium usage with small uranium mass used cannot be the sole claim for highly effective reactor. Claim for effective energy transformation is also significant. High value of efficiency allows not only decrease uranium usage for unit of power production but decrease of negative effect to the nature, because small efficiency means more thermal energy release.

Complexity of nuclear reactor technology, effective work with energy transformation installations requests contradictions make us to look for different technical solutions, which can provide desired result. During development of atomic power plants many solutions, different coolant types, construction materials, schemes of thermal to electrical power transformation were tested.

In field of thermal reactors the base niche is occupied by water-water reactors. Between two types of these reactors (pressurized and boiling) there is a competitive contest. Now in heavy water moderated reactors there are both transfer directions of water coolant thermal energy to turbine. Maximal efficiency of this scheme is limited by ~34 %, which is depending on acceptable water pressure in a core and its temperature correspondingly.

High efficiency can be obtained in fast liquid metal cooled reactors with coolant temperature about 500 ^{o}C, which is typical for combustible fuel powered electro stations. Feature of this scheme is decoupling of coolant load conditions and Rankin cycle actuating medium. Pressure of coolant in reactor vessel can be close to atmospheric.

Decoupling of coolant load conditions and actuating medium also exists in schemes with gaseous coolant [23]. Energy transfer direction with use of gaseous coolant in 50-60-ties of XX century has transformed by type of energy receiver from Rankin cycle to Briton cycle [24].

Potential some efficiency increase (up to ~48 %) in this scheme is possible, but has some negative effects. Main effects are listed below.

The first. There is a need to steep increase of coolant temperature, up to 1000 ^{o}C. It causes increase of expenses for reactor and energy transformation installation.

The second is fuel rod design complication caused by higher coolant temperature. Besides expenses increase it causes tendency of fission products release increase.

The third is power increase on the Briton shafting four times higher than Rankin steam turbine with the same power output and cost rise, correspondingly.

All these factors are negative for NPP economy, time for solution preparation to practical realization.

The paper [25] shows usage possibility in Briton cycle instead of turbines piston machines, which supply high efficiency obtaining possibility at lower coolant temperature. It is obtained because of absence energy transfer chain from high speed gas to blades of turbine (and vice versa in compressor), which makes basic energy loss in Briton cycle. Shortages of this solution are absence of practical schemes of needed piston machines and less power of a unit.

9.1. Heavy water reactor with gaseous coolant in Rankin cycle

9.1.1. Technology problems

In channel heavy water reactors portion of energy, which is released in moderator, is lost for energy production because the moderator has low temperature. In channel released energy

is used for energy transformation. Together with heat flow to moderator from channel thermal energy loss is about 10 % from total energy released in the core. Maybe it is a reason of HTGR design with high temperature graphite, which transfers energy of neutron moderation to fuel assemblies.

Rankin cycle with two steam overheating, one at maximal pressure in cycle (at the entrance to turbine) and additional – after steam expansion to specified pressure, is used in modern reactors. At small maximal pressure levels steam at last stage and turbine exit is wet with high specific humidity. It leads to exit blades corrosion, necessity of valuable alloys usage. Blades of last stage have maximal size and determine total cost of turbine.

9.1.2. Solution description

Scheme of joint work of heavy water reactor with gaseous coolant and Rankin cycle steam turbine, which is based on full use of fission energy (including neutron moderation energy) and three stepped steam entering to turbine, is suggested in the paper [26].

First feature of the solution is based on heat emission to the steam in Rankin cycle, which is made not in a single process, but several with different temperature intervals.

The second feature allows avoiding presence of steam with high specific humidity on exit stage blades of turbine. Good parameters are achieved at maximal steam temperature 500 °C and maximal pressure 20.0 MPa.

Scheme of coolant ducts and steam loop of heavy water channel reactor with gaseous coolant is shown at figure 12. Differences of this scheme from other solutions are usage of neutron moderation energy in Rankin cycle for water of steam loop heating, separation of coolant duct on four ducts, which supply with energy re-heater and water evaporator, separate steam overheaters. Coolant ducts for heating and water evaporation can be water ducts.

Neutron moderation energy transfer to water in Rankin cycle is realized with use of vessel of reactor design with greater than atmospheric pressure, but less than maximal pressure in coolant duct. It allows having moderator with temperature greater than water heated in Rankin cycle and decrease thickness of duct walls.

Separation of overheaters ducts allows decreasing of danger of hermetization loss of steam loop with high pressure. Actions for sequence avoiding of hermetization loss is made to duct with small portion of reactor power (~20 %). Reactor scheme at figure 12 gaseous coolant duct, which is linked to high pressure steam duct, has vessel, which supplies decrease of maximal pressure at accidental steam leakage to coolant (volume of steam is limited) and increase time of pressure growth in coolant duct at such leakage (work condition of automatics becomes better).

Besides this separation optimizes energy expenses for coolant pumping by use of temperature differential in corresponding external heat exchangers.

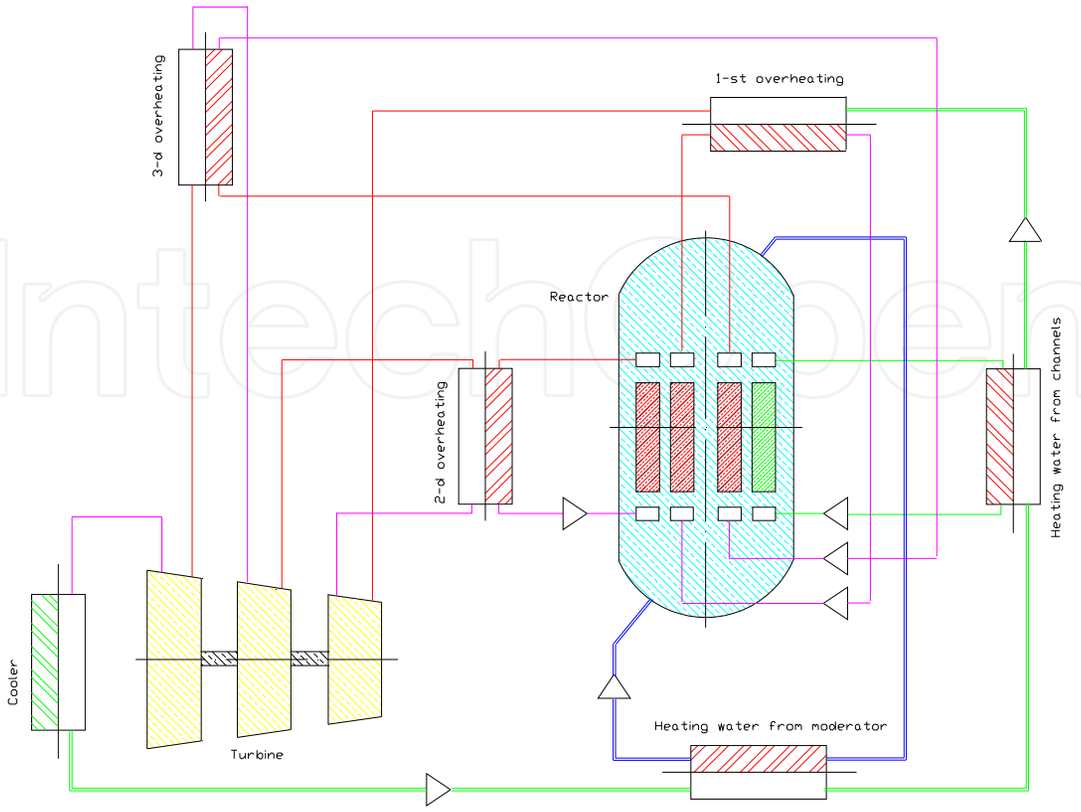


Figure 12. Scheme of coolant ducts and steam loop of heavy water channel reactor with gaseous coolant.

9.2. Description of reactor with thermal power 80 MW

Calculation of neutron and thermal characteristics of the reactor are conducted with use of programs [16, 27], campaign characteristics with use of program [19], Rankin cycle and turbine characteristics with use of programs [28, 29].

Sketches of reactor design with thermal power 80 MW with gaseous cooled fuel assembly and with water cooled fuel assembly are worked out. Water cooled fuel assemblies are located in central part of the core.

The reactor has 85 fuel assemblies, which are located in nodes of triangular grid with step 18 sm. Core is surrounded by two-layer reflector. Inner layer is heavy water, outer layer is graphite.

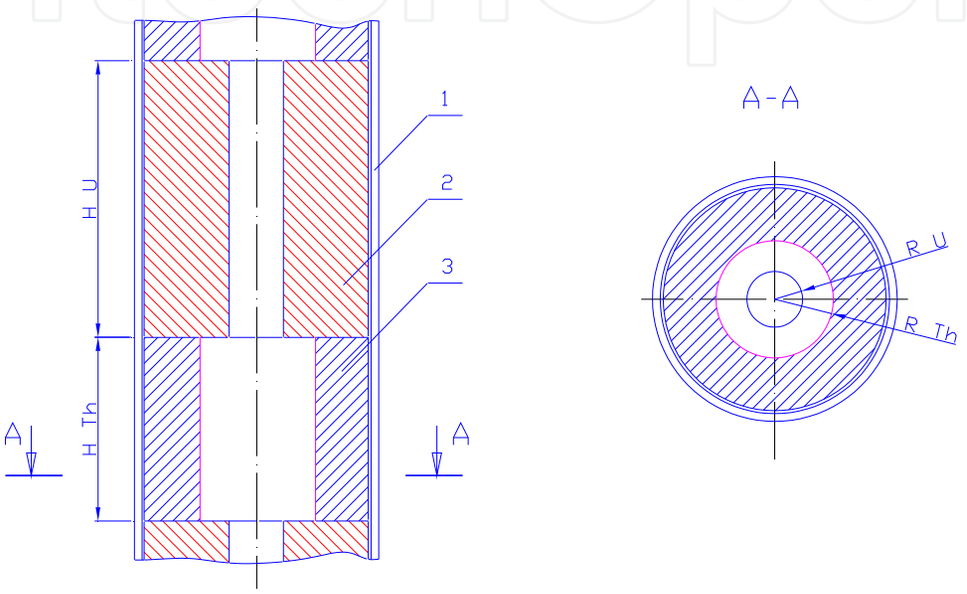
9.2.1. Description of fuel assembly

Each fuel assembly has 59 fuel rods with outer diameter 6 mm.

Coolant cross-section is limited by a screen made of zirconium alloy thin shell (thickness 1 mm), and a casing made of the same alloy shell (thickness 3 mm) at distance of 2 mm from the screen. The gap between the screen and the casing is gas filled. The gas pressure equals to average by channel height pressure of actuating medium.

Gaseous coolant in fuel assembly can be hydrogen or helium. Fuel on base of metallic uranium and thorium is chosen for the work variant (figure 13). Each fuel rod has uranium fuel elements alternating by height with thorium fuel elements [30]. Initial contents of ^{235}U in uranium elements is 0.5 %. Initial contents of ^{239}Pu and ^{241}Pu in uranium elements, and initial contents of ^{233}U in thorium elements equal to equilibrium contents of these nuclides during durable campaign.

This design supplies constant energy release distribution by height of fuel rod independent from fuel type under the shell – uranium or thorium.



1 – rod shell, 2 – uranium fuel element, 3 – thorium fuel element.

Figure 13. Disposition of uranium and thorium fuel elements by fuel rod height.

Quality of fuel rod height energy release distribution becomes better with increase of portion of equilibrium nuclides and, correspondingly, raw uranium usage portion increase.

Difference of fuel elements form with uranium and thorium has significant role at reprocessing of spent fuel, when there is need to separate uranium and thorium fuel.

With use of equivalent fuel elements with mix of uranium and thorium, where also can be obtained uniform energy release, we will have problem of separation fission ^{233}U and ^{235}U from raw ^{238}U. Without this separation raw uranium usage portion has steep decrease.

9.2.2. Campaign characteristics

Change of reactivity margin during two variants of the campaign without fuel replacement and with one fuel replacement.

At constant neutron flux in core ratio of reactor power at campaign end to power in campaign beginning in variant without fuel replacement equals 37,6 %, and with one replacement – 17,6%. Fuel burn-up is 5,1 % or 48,7 MW*day/kg.

Uranium tablets of first campaign are made from raw uranium with adding of 0,357 % ^{239}Pu and 0,12 % ^{241}Pu. It is possible to have the same mass of nuclides with different mixture contents. In thorium tablets of the first campaign is added 2.3 % ^{235}U of high enrichment.

Uranium tablets are divided on two parts at spent fuel reprocessing. Fission products and plutonium is extracted from 70% of spent fuel uranium mass of the first campaign. This fuel part is replaced with raw uranium, in which extracted plutonium is added.

Fission products are separated from the second fuel part. Add mass of raw uranium which equals mass of burned ^{238}U in uranium fuel.

Fission products are separated from the thorium fuel part. Add mass of thorium which equals mass of extracted fission products.

Usage of raw uranium in closed cycle is 5%.

In reactor core there is 2312 kg of fuel. Spent fuel reprocessing requirement is 680 kg per annum. Raw uranium requirement is 510 kg per annum. Required mass of thorium is 14 kg per annum.

For comparison, reactor WWER-1000 uranium requirement is 10 times larger per unit power and relative amount of dissipated power is more 1.4 times.

9.2.3. Thermal characteristics of fuel assemblies

Thermal characteristics of fuel assemblies with hydrogen coolant are presented in table 4. Characteristics of fuel assemblies with helium coolant are slightly different. Hydrogen is cheaper than helium so it is preferable.

Coolant	**H_2**			
Gas in the gap of fuel assembly	**CO_2**			
Function of fuel assembly	3 overheating	2 overheating	1 overheating	Water heating
Temperature at fuel assembly entrance, ^0C	**285.9**	**285.3**	**376**	**141**
Temperature at fuel assembly exit, ^0C	**510**	**510**	**510**	**375**
Fuel assembly number in reactor	10	12	17	46
Coolant flow in fuel assembly, kg/s	0.286	0.285	0.478	0.274
Coolant velocity at fuel assembly entrance, m/s	28.8	28.6	55.36	20
Coolant velocity at fuel assembly exit, m/s	40.1	40	67.35	31.5
Pressure difference in fuel assembly, Pa	3892	3882	9617	2923
Coolant pump power in fuel assembly, kW	5.24	6.25	36.8	17.34
Total coolant pump power, kW / % from total	65.63/0,2			

Tmax at fuel rod shell, ^{0}C	805		758	813
Tmax fuel rod, ^{0}C	1125		979	1035
Tmax of gas gap of fuel assembly, ^{0}C	337	336	388	203
Tmax of fuel assembly screen, ^{0}C	364		420	223
Tmax of fuel assembly casing, ^{0}C	182	181	202	168

Table 4. Thermal characteristics of fuel assemblies with gaseous coolant.

Basic features of these assemblies are listed below.

- Work of all fuel assembly elements is conducted in acceptable temperature region.
- Hydraulic loss for pumping is small.

All this features obtained with simple fuel assembly design and its base elements – fuel rods. Used fuel rods are well fine-tuned at practice of large number of reactors. Small hydraulic losses lead to small required power of coolant pumps. It characterizes costs level for reactor creation and losses level during exploitation.

9.2.4. Rankin cycle characteristics

Calculations of cycle variants, which are different by maximal steam pressure, number of interim stages of re-heating and steam expansion characteristics on it, are conducted. The most economic variant of possible by complexity manufacturing is shown at table 5 and figure 14.

This cycle use two interim overheating and minimal total steam expansion. Zero water content in steam at turbine exit is obtained. This means small turbine HWD, usage possibility in turbine design cheaper materials for the blades and correspondingly smaller cost.

Cooler of this variant also has small HWD and comparatively high temperature of fluidized steam supplies possibility of its use as heat source for home and business needs.

The second variant has steam quality on the exit of turbine 0.93, which differs much from steam quality in VVER – 0.75. Total steam expansion is 3.7 times bigger than in VVER, and 42.4% efficiency is obtained.

Problem of material for turbine last stage blades here is not so strong, as in nuclear power plants with VVER or ABWR reactors [31].

For electro stations, which work in autonomic regime, for example in distant areas from common electrical supply network, the first variant is preferable by two reasons:

1. In this case obvious, that there is need in heat of low potential;
2. Electro station cost is significantly cheaper with less HWD of turbine and less cost of blades. Turbine cost is significant part of NPP cost [31].

№ re-heater	T, °C	P, gauge atmospheres	Enthalpy	Portion of Q	Entropy	Sp. volume	Vapor quality
1 Cooler	69,6	0,304	2625,4	-0,583	7,764	5,17	1
	69,3		290,1		0,946	~0,001	0
2 Water from moderator		10		0,08			
	131		610,5		1,645		
3 Water from fuel assembly				0,2998	1,626		
	365		1811,5		3,99		
4 Water-steam		200		0,1527			
	366		2423,1		4,95	0,0074	1
5 The 1-st overheating				0,2042			
	500		3241,2		6,1446	0,0148	
6 The 2-nd overheating	275,3	45	2865,3	0,1436		0,047	
	500		3440,4		7,0323	0,076	
7 The 3-rd overheating	275,9	10	2999,9	0,1196		0,246	
	500		3479,1		7,764	0,354	

Table 5. Steam and water parameters in cycle with 3 re-heating at Pmax=200 gauge atmospheres and Tmax= 500 °C. Theoretical efficiency is 41.7 %, with account of losses on turbine 38.8 %

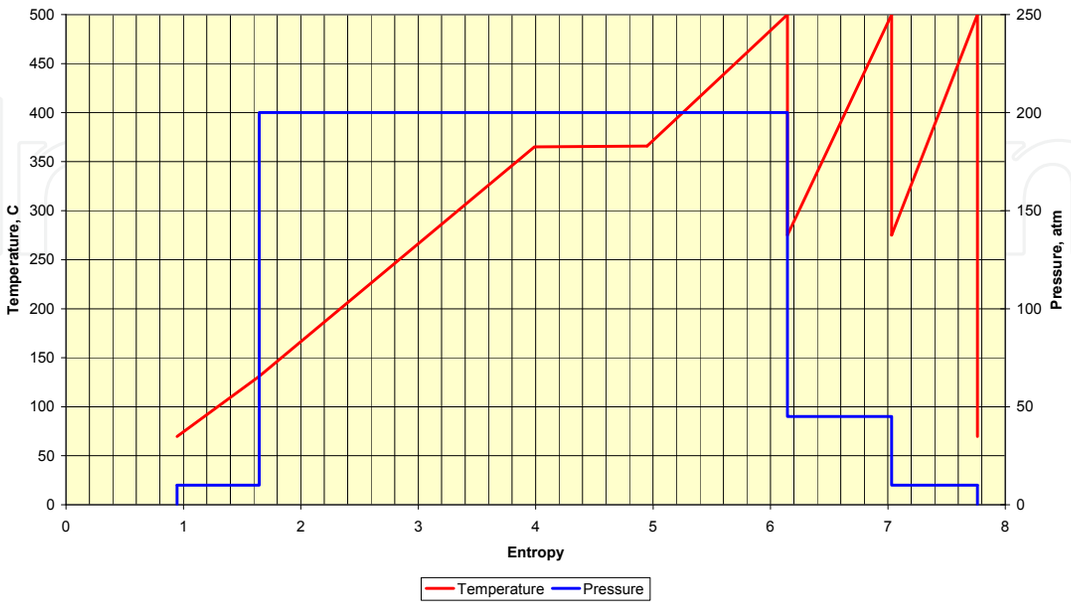


Figure 14. Pressure and temperature dependencies from entropy at final steam quality 1.0.

At variant with common electrical supply network and absence of heat need (for example in the tropics) it is required to take into account turbine cost difference of first and second variants, works financing at NPP building.

In all case it should be noted, that NPP with 39% efficiency is more attractive than other small power NPP variants with efficiency up to 33%. Especially if we take into account many times lower raw uranium requirement and absence of uranium enrichment works for fuel preparation.

On the base of described solutions heavy water gas cooled high power nuclear reactor can be built. Increase of core HWD leads to neutron leakage decrease, which is base neutron loss for nuclear reactor with 80 MW thermal power.

9.2.5. About nuclear safety

Figure 15 shows coolant and moderator ducts scheme in heavy water gas cooled reactor, which prevents power increase at reactivity accident [32].

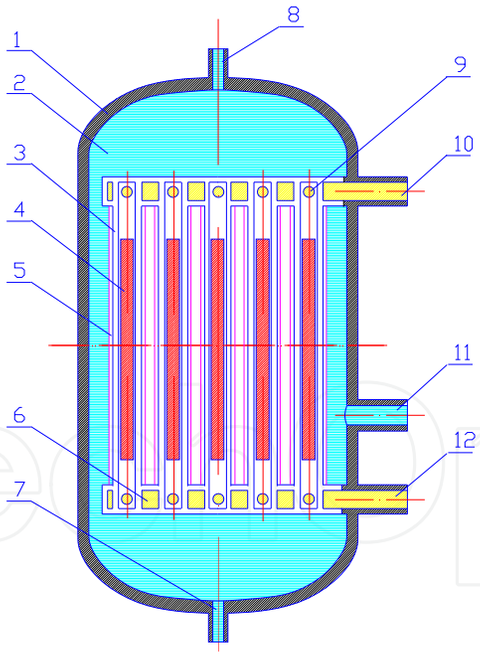


Figure 15. Coolant and moderator ducts scheme in heavy water gas cooled reactor. 1 – reactor vessel, 2 – moderator, 3 – channel casing, 4 – fuel assembly, 5 – channel thermal isolation, 6 – integral coolant collector, 7, 8 – inlet and outlet of moderator, 9 – opening, which connects channel with collector, 10, 12 inlet and outlet of coolant, 11 – nozzle of accident drainage of moderator.

Coolant temperature, which supplies possibility of effective usage of neutron moderation energy, plays here positive role. Moderator temperature increase requires pressure rise in reactor vessel that allows making channel walls thinner and decrease neutron loss in it. Let us take that moderator inlet temperature is 210 °C, outlet temperature 225 °C, flow - 80 kg/s, moderator pressure - 25 atmospheres.

Moderator boiling at unplanned power increase with work pressure and temperature leads to 12 time volume increase of evaporated mass and pressure increase at moderator volume. Core bottom water moves out through nozzle of accident drainage of moderator. Liquid moderator of core upper part is replaced with steam. Preliminary calculations have shown that heavy water deletion from core upper quarter decreases reactivity margin by 2.5 β, that supplies damping of earlier added reactivity.

Damping effect of reactivity accident in this reactor may be not less than in graphite HTGR. Positive feature of this scheme is possibility of work parameters adaptation of safety system by way of specifying work temperature and pressure of moderator, when it starts boiling.

10. About possible scale of nuclear power engineering development

Described actions sufficient using for fission materials reproduction in thermal reactor supplies besides high portion of raw uranium usage (up to 25% in contrast to ~1 % for thermal reactors with enriched fuel) small fuel requirement for core loading. Sum of these effects allows creation of world big nuclear power production industry.

Requirement in raw uranium and thorium of these reactors can be determined by formulae:

$$mU = n * t * \left(m_s * Yu * + m_g * K_i * Ku / 2\right); \quad (12)$$

$$mTh = n * t * \left(m_s * \left(1 - Yu\right) * + m_g * \left(1 - Ku\right) / 2\right); \quad (13)$$

where:

n – number of this type built reactor per annum;
t – time of nuclear power engineering development, years;
m_s – mass of raw materials, needed for core loading;
Yu – raw uranium portion in fuel;
m_g – fuel mass needed for year feeding of reactor;
K_i – portion of raw uranium usage in fuel cycle;
Ku – portion of raw uranium in feeding fuel.

Figure 16 shows development variant of power production with even power grow to the level of 8000 GW during 80 years with subsequent power level stabilization.

Work duration of thermal reactors with cheap uranium stocks at this power level and full raw uranium usage is ~2500 years. It is understandable that at such small raw uranium requirements is rational use of other more expensive deposits, where uranium stocks are much more than in cheap deposits.

So, there is significant reserve in world nuclear power production industry.

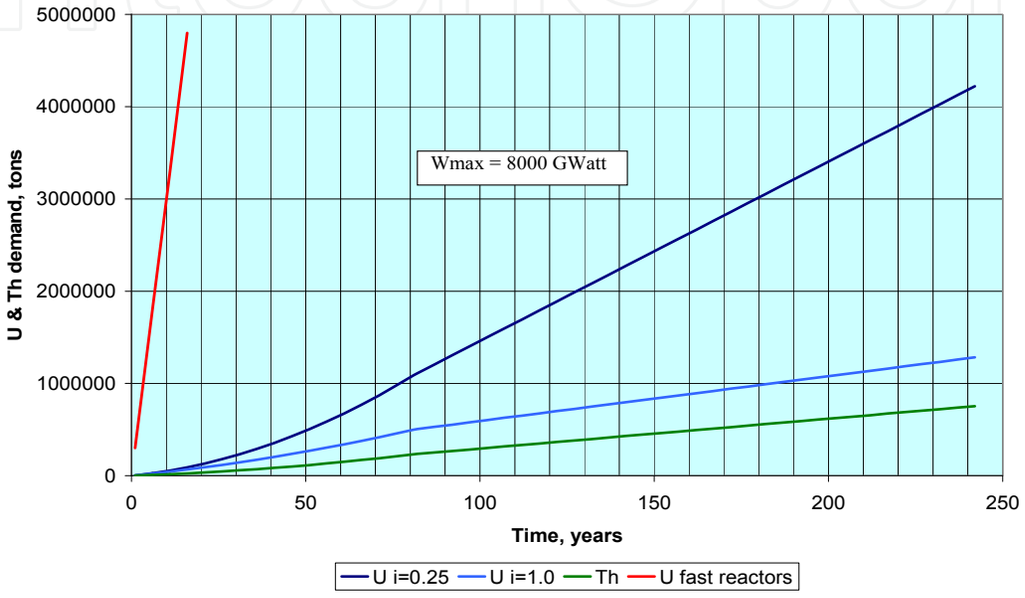


Figure 16. Dependence of uranium and thorium requirements for different reactor types with zero initial power and its even increase up to level of 8000 GW with subsequent power level stabilization.

11. Comparison of possibilities of unauthorized fission materials proliferation at different technologies of fuel cycles

Nonproliferation regime in common case is supplied by IAEA control. With control the possibility of proliferation is limited by value less than detection error. The base of this error is inaccuracy of mass detection at chemical reprocessing of irradiated in core material. Typical value of the error can be in our case ~0.1 % from reprocessed mass of ^{233}U.

For thermal reactor of 1000 MW requirement in ^{233}U is about 10 kg per annum. Accordingly, in ideal case of reprocessing conducting, possible error in ^{233}U is less than 10 g per annum. If inaccuracy of the error is 10 %, than proliferation is not much than 1 g per annum. For minimal warhead in this case is needed more than 1000 years. This term exceeds possible duration not only a terrorist organization but even a terrorist state.

12. Conclusion

Fission materials reproduction possibility in different fuel types in ideal core without neutron losses in construction materials and leakage is shown. Equilibrium concentrations of fission materials in different fuel types are determined.

Features of detailed regime campaign conduction (with fixed fuel location during all campaign) and compact regime (with staged spent fuel replacement with fresh fuel).

Characteristics of loss and reproduction in case of CANDU and its possible modernization variants are examined.

Replacement in zirconium containing materials natural zirconium by isotope ^{90}Zr and natural tin by isotope ^{120}Sn, replacement of 7 fuel rods in fuel assembly with 37 fuel rods by beryllium insert for extra neutron production, change of fuel rods with oxide fuel by metallic fuel is considered as possible modernization directions, which supply high fission materials reproduction.

Compound metallic fuel rod construction, placed in liquid metal heat transferring medium is suggested. Shape and small size of fuel rod ensure decrease of negative effect of swelling.

Possibility of neutron loss decrease in CANDU from 5% to 2.8% in case of isotope modified zirconium and tin and to 1.7% in case of metallic fuel and beryllium insert.

It is suggested that excess neutrons of detailed campaign beginning are used for fission materials reproduction. By the set of characteristics ^{233}U is the best candidate for reproduction.

Portion of raw uranium use increase in open fuel cycle up to 5.3% and full raw uranium usage in closed uranium and thorium fuel cycles is shown.

Conditions of high efficiency obtaining in Rankin cycle with heavy water gas cooled reactor are shown. Scheme of coolant ducts and steam loop of heavy water channel reactor with gaseous coolant, which ensures full use of neutron moderation energy, decrease energy loss for coolant pumping and obtaining high steam quality on the turbine exit. These actions allow to decrease cost for NPP creation, to have efficiency of 43 % taking into account possible losses in core and steam cycle.

World nuclear power production industry creation with power of 8000GW to the end of XXI century on the base of suggested thermal reactors with high fission materials reproduction is shown.

Suggested technologies usage allows increasing world nuclear power industry to the end of XXI century with requirement decrease of natural uranium mining, proliferation danger decrease comparing to fast reactors technology.

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