THE MODE OF OPERATION OF CANDU POWER REACTOR IN THORIUM SELF-SUFFICIENT FUEL CYCLE

by

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This paper presents the results of calculations for CANDU reactor operation in the thorium fuel cycle. The calculations were performed to estimate feasibility of operation of a heavy-water thermal neutron power reactor in the self-sufficient thorium cycle. The parameters of the active core and the scheme of fuel reloading were considered to be the same as for the standard operation in the uranium cycle.

Two modes of operation are discussed in the paper: the mode of preliminary accumulation of ^{233}U and the mode of operation in the self-sufficient cycle. For calculations for the mode of accumulation of ^{233}U, it was assumed that plutonium was used as the additional fissile material to provide neutrons for ^{233}U production. Plutonium was placed in fuel channels, while ^{232}Th was located in target channels. The maximum content of ^{233}U in the target channels was about 13 kg/t of ThO$_2$. This was achieved by six year irradiation. The start of reactor operation in the self-sufficient mode requires content of ^{233}U not less than 12 kg/t. For the mode of operation in the self-sufficient cycle, it was assumed that all the channels were loaded with the identical fuel assemblies containing ThO$_2$ and a certain amount of ^{233}U. It was shown that the non-uniform distribution of ^{233}U in a fuel assembly is preferable.

Key words: thorium fuel cycle, CANDU reactor, reactor operation

INTRODUCTION

The absence of isotope ^{233}U in nature does not exclude the possibility of its use as a nuclear fuel. ^{233}U is a product of radioactive decay of ^{233}Pa, which is formed by means of neutron capture by ^{232}Th with the subsequent β-decay. Thereby, thorium, existing in nature in a single stable isotope ^{232}Th, can be transformed, using nuclear reactions, into ^{233}U, *i. e.* into the fuel for power reactors. The world reserves of thorium exceed those of uranium by several times. The cost of thorium mining is much lower than that of uranium.

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This is due to approximately 100 times lower radiation danger in the process of mining thorium in comparison with mining uranium [1]. The possibilities of commercial production of ^{233}U in nuclear reactors have been considered many times in publications in connection with thorium-uranium fuel cycle. It should be noted that even partial substitution of uranium-plutonium fuel cycle by thorium-uranium cycle is connected with significant difficulties. One of the difficulties is related to the necessity of changing the process. However, there are also physical problems: half-life of the decay of ^{233}Pa in ^{233}U, which is 27 days, greatly exceeds the analogous half-life of the decay of ^{239}Np in ^{239}Pu in uranium-plutonium fuel cycle, and the cross-section of capture of thermal neutrons in ^{232}Th is 2.8 times higher than the analogous cross-section of ^{238}U. These disadvantages of thorium fuel cycle were seemingly a reason that in the publications of recent years thorium has been considered only as a raw material for feed of nuclear reactors operating in uranium-plutonium fuel cycle [2]. Such mixed mode allows more or less saving natural uranium during electric power production. The possibility of operation of power reactor in thorium-uranium fuel cycle without the feed of fissile ma-

terials has been considered in preceding publications, for instance in [2, 3]. However, this possibility has only been declared without any demonstration of concrete ways of its achieving.

It should be noted that the self-sufficient mode is related with rather big effort in the extraction of isotopes of uranium from unloaded nuclear fuel. However, because of the need for accumulation of the required amount of ^{233}U, this disadvantage is inherent not only to the self-sufficient mode, but also to thorium-uranium fuel cycle in any of its modifications. Accumulation of ^{232}U along with daughter nuclides affecting a radiation environment should also be mentioned as a drawback of thorium fuel cycle.

Nevertheless, thorium-uranium fuel cycle also has certain advantages. In particular, the number of secondary neutrons in fission of ^{233}U by thermal neutrons is higher than for any other isotopes of uranium. This fact gives hope for the possibility of the operation of reactor in self-sufficient mode.

In this paper, the results of calculations for the self-sufficient mode for a heavy-water power reactor are presented. This reactor is hereinafter referred to as "T" reactor. The parameters of the active core and the scheme of refueling of current CANDU heavy-water power reactor (HWPR) with the heat power of 2776 MW [4] were used for calculations.

Calculations were made using the code complex MCCOOR, developed on the basis of codes MCNP, COUPLE, and ORIGEN-S.

MODE OF ACCUMULATION OF ^{233}U

For the reactor start-up, it is necessary to have a certain amount of fissile material, in our case ^{233}U. In this paper, it is assumed that ^{233}U for downstream operation is produced in "T" reactor during the first period of operation. Although it is not improbable that the accumulation of ^{233}U in the special reactor or in other power reactors, in particular in blanket of fast breeder reactor [5], may turn out to be more efficient.

In the mode of accumulation of ^{233}U, the tetragonal heavy-water lattice of "T" reactor is composed of channels of two types, evenly distributed over the active core. Half of the channels are referred to as fuel channels, another half are referred to as target channels. A fuel channel contains a fuel assembly composed of 37 fuel elements with power plutonium or enriched uranium. A target channel contains a target assembly composed of 37 target elements. Each target element is made in the form of zirconium tube filled with pellets of ThO$_2$. Figure 1 demonstrates a part of a channel lattice with 16 channels and an elementary cell accepted for calculations.

During the operation of "T" reactor, power and neutrons are released in the fuel channels due to fission of ^{239}Pu and ^{241}Pu (^{235}U). In the target channels in the
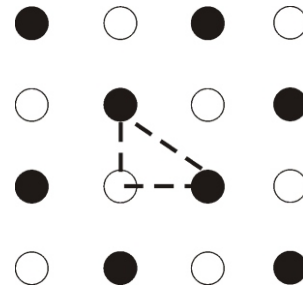

Figure 1. Lattice of channels in active core in the mode of accumulation of ^{233}U and elementary cell accepted for calculations

beginning of operation the excess neutrons are captured with formation of ^{233}Pa. As the ^{233}Pa decays in ^{233}U, power releases in the target channels too. It is assumed that the power of the fuel channels with plutonium or enriched uranium is maintained to keep constant power density per mass unit of heavy nuclei equal to that of the current CANDU type reactor. This condition determines neutron flux density in targets.

The results of calculation of the elementary cell of active core for the variant using fuel containing 10% of power plutonium are presented in figs. 2 and 3.

According to the data of fig. 2, the maximum (equilibrium) content of ^{233}U in targets is ~13 kg/t (these data are given in kilograms of ^{233}U per one ton of ThO$_2$ with the density of 10 g/cm^3). This result is close to that obtained in [6] for the thorium-uranium lattice and neutron spectrum in the fuel of CANDU reactor. The time of irradiation till maximum content of ^{233}U in targets is reached is 5-6 years, the time for achieving maximum multiplication factor K_∞ is about 2 years. During this time, the content of ^{233}U in targets reaches about 8 kg/t.

For the start-up of "T" reactor in the self-sufficient mode, it is necessary to load ThO$_2$ with the content of ^{233}U not less than 12 kg/t in fuel elements of all assemblies. For the accumulation of necessary amount of ^{233}U in the mode of accumulation, it will take, for
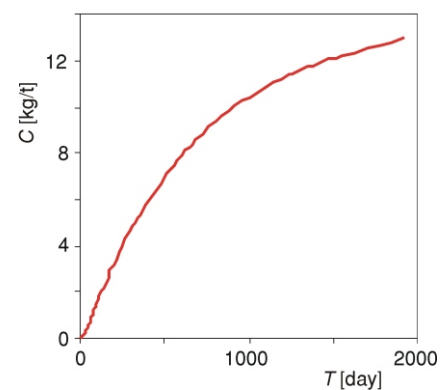

Figure 2. Variation of concentration of ^{233}U in ThO$_2$ in the mode of accumulation of ^{233}U
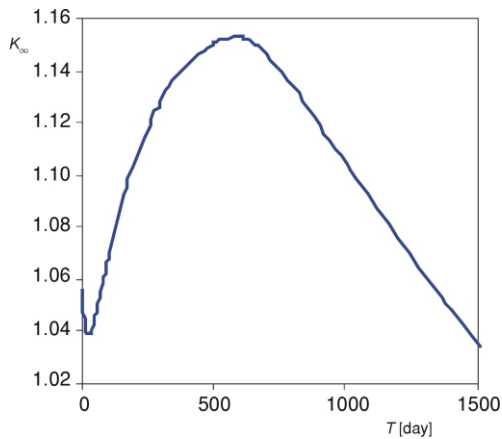
Figure 3. Variation of multiplication factor for elementary cell in the mode of accumulation of ^{233}U

instance, three cycles of irradiation of ThO$_2$ targets during 2 years. Therefore, in the mode of accumulation of ^{233}U, "T" reactor should work approximately 6 years. After each cycle of irradiation, targets should be unloaded from the reactor and replaced by fresh targets without ^{233}U. In order to avoid "T" reactor becoming subcritical during the target reloading, the replacement of irradiated targets by fresh targets without ^{233}U should not be simultaneous. As ^{233}U is accumulated in targets, the total power of the reactor increases in accordance with the given mode of operation, while the power of the fuel channels remains constant.

It is apparent that the considered parameters of operation of "T" reactor in the mode of accumulation of ^{233}U can be highly improved. It is possible to reduce the time of accumulation of ^{233}U and increase the power of a reactor by following measures:

– increase of heat load on a fuel element, in particular at the expense of use of fuel elements with the diameter less than that in the current CANDU reactor,
– compensations of reactivity increase (see fig. 3) due to replacement of fuel channels by target channels,
– optimization of number and locations of target channels and fuel channels in the active core,
– optimization of scheme of reloading target channels,
– increase of number of targets in the target assembly [2], and
– reducing plutonium (uranium) load in fuel elements that would result in the increase of thermal neutron flux in targets at the same power of "T" reactor.

SELF-SUFFICIENT MODE

In the self-sufficient mode, it is assumed that the fuel elements containing ThO$_2$ with the given content of ^{233}U are loaded in all channels of the reactor. The

considered mode could be realized in "T" reactor if two conditions are met (that is a non-trivial task at the strict restriction of total amount of ^{233}U).

Condition 1. The amount of ^{233}U and its layout in channels must ensure overcriticality of "T" reactor in the initial state. Overcriticality of the current CANDU reactor in the cold condition with fresh fuel (natural uranium), calculated by means of the above mentioned code complex, is ΔK 0.1. It is obvious that "T" reactor should have overcriticality not lower than this value.

Condition 2. In "T" reactor, under the acceptable burnup of ^{233}U and other fissile isotopes, the reproduction of ^{233}U must ensure at least the equality of content of fuel in the active core before the beginning and after the end of fuel lifetime. This condition together with the reactivity margin determines the maximum burnup of fuel and the fuel lifetime length. The breeding ratio and its variations during the fuel lifetime depend on the amount of fuel formed in the mode of accumulation of ^{233}U and of its position in the active core of the reactor.

Fuel burnup and fuel lifetime in "T" reactor was estimated by a method which is suitable for CANDU reactors with continuous bidirectional refueling. In this case, fuel assemblies with different burnups in the range from zero to the maximum burnup are present in the active core during the whole lifetime. Such condition can be described by the average multiplication factor determined by the leakage of neutrons from the active core, absorption of neutrons by structure materials of the active core and by control rods. For CANDU reactor with control rods in the active core this value is 1.045, and for the extracted control rods – 1.035 [1]. Fuel burnup W_0 in "T" reactor was estimated in accordance with the parameter α

$$\alpha \quad \int_0^{W_0} K_\infty \mathrm{d}W/W \tag{1}$$

where W is fuel burnup. The dependence $K_\infty(W)$ was calculated in the interval between the initial value $W = 0$ corresponding to the fresh fuel, and the final value W_0 corresponding to the fuel unloaded from the reactor. In the process of calculation of $K_\infty(W)$, neutronic parameters were calculated over again at different values of W with the step of 1 MWd/kg in order to take into account the change of the neutron spectrum in the fuel.

Calculations were made for different burnups and for three variants of initial layout of ^{233}U in the active core of "T" reactor.

In the first variant, the accumulated ^{233}U was distributed in equal parts over all fuel assemblies of the reactor with the content $C = 12$ kg/t in each fuel element.

In the second variant, the accumulated ^{233}U was distributed in equal parts over all fuel assemblies of the reactor. However, in each fuel assembly, ^{233}U was

placed only in 18 external fuel elements with the content $C = 24$ kg/t while 19 internal fuel elements were filled only with ThO_2 without ^{233}U.

In the third variant, the accumulated ^{233}U was distributed in equal parts over all fuel assemblies of the reactor. However, in each fuel assembly, ^{233}U was placed in 18 external fuel elements with the content $C = 16$ kg/t, and in 19 internal fuel elements with the content $C = 10$ kg/t. The balance of the uranium was ensured due to the changing of density of $^{233}U + ThO_2$.

The results of calculations of $K_\infty(W)$ for the considered variants of ^{233}U layout in fuel assemblies are presented in fig. 4. Curves 1-4 were used for the calculation of fuel burnup in "T" reactor for different values of the parameter α (1). The curve (4) corresponds to the common-type CANDU reactor with natural uranium fuel.
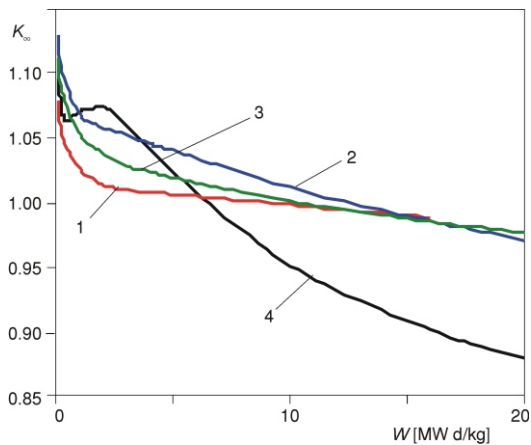

Figure 4. Multiplication factor for elementary cell in the self-sufficient mode; 1, 2, 3 – 1st, 2nd, and 3rd variant of ^{233}U layout in fuel assemblies of "T" reactor, 4 – CANDU reactor with natural uranium

The results of calculations of fuel pin respective power P_{int}/P_{ext} in the fuel assembly are presented in fig. 5 for 2nd and 3rd variants of ^{233}U layout. P_{int} is the total power of 19 internal fuel pins with lower content of ^{233}U, and P_{ext} is the total power of 18 external fuel pins with higher content of ^{233}U in the fuel assembly. These curves were used for the estimation of intervals of heat power variation of "T" reactor. The results of calculation of fuel burnup for three variants of ^{233}U layout in fuel assemblies are given in tab. 1.

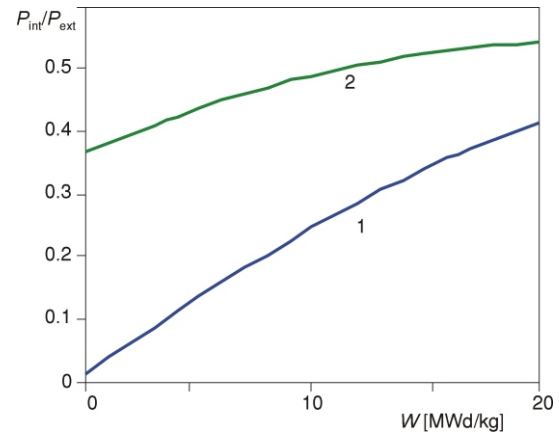The calculated fuel burnup of CANDU reactor with natural uranium is in good accordance with the

Figure 5. Fuel pin respective power P_{int}/P_{ext} in fuel assembly; P_{int} – total power of 19 internal fuel pins, P_{ext} – total power of 18 external fuel pins in fuel assembly; (1), (2) – 2nd and 3rd variant of ^{233}U layout in fuel assemblies of "T" reactor

operating value of 8.3 MWd/kg [4]. So, there is a good reason to think that fuel burnups calculated for three variants of fuel layout in fuel assemblies of "T" reactor using the same method with the same neutronic constants are close to the real values. The data in tab. 1 demonstrate that under the even distribution of ^{233}U over all fuel elements of "T" reactor (first variant) fuel burnup is unacceptably low, while the initial overcriticality of the active core is less than 10% (curve 1 in fig. 4). Fuel burnup can be increased to 15 MWd/kg due to the location of the same amount of ^{233}U in the external fuel elements of the fuel assembly according to the second variant. In this case, the shielding of the internal fuel elements allows increasing burnup by approximately 10 times. At the non-equal load of ^{233}U into external and internal fuel elements in each assembly according to the third variant, in which the shielding of the internal fuel elements is less than that in the second variant, burnup decreases by 1.5 times with respect to the second variant. As one would expect, the heat power of "T" reactor for the third variant is much higher than for the second variant.

It is obvious that the possibility of practical realization of the self-sufficient mode in "T" reactor depends mainly on the amounts of fissile isotopes in the fuel unloaded from the active core in the end of the next fuel lifetime. The curves in fig. 6 describe the changes of total amount of ^{233}U, ^{235}U, and ^{233}Pa during

Table 1. Fuel burnup and heat power of the reactor for three variant of fuel layout

Reactor	Fuel burnup [MWd/kg]			Heat power [MW] for $\alpha = 1.035$
	$\alpha = 1.025$	$\alpha = 1.035$	$\alpha = 1.045$	
CANDU	9.4	8.1	6.7	1750 [4]
"T", variant 1	3.4	1.5	0.7	2760 [4]
"T", variant 2	17.5	14	11	1400-1900
"T", variant 3	10.1	6.7	4	1900-2000
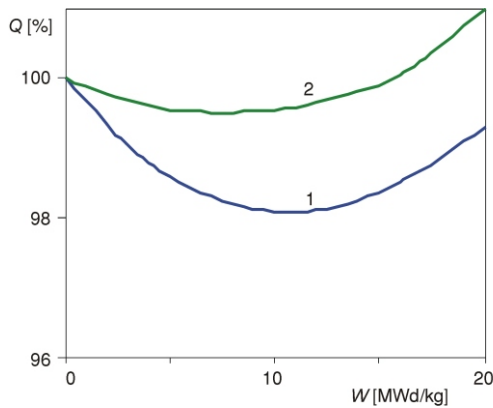
Figure 6. Variation of Q **– total mass of** ^{233}**U,** ^{235}**U, and** ^{233}**Pa during the fuel lifetime; (1) – 2**nd **variant of** ^{233}**U arrangement in fuel assemblies of "T" reactor; (2) – 3**rd **variant of** ^{233}**U arrangement in fuel assemblies of "T" reactor**

the fuel lifetime for the second and the third variants of fuel layout (initial value is accepted as 100%).

For obtaining additional data, the calculations of multiplication factors $K_\infty(0)$ and $K_\infty(W_0)$ were made for the third variant of fuel layout in cold "T" reactor (tab. 2). The fuel burnup of 8 MWd/kg was accepted in these calculations, which corresponded to the fuel lifetime length of 430 days. Nuclides ^{233}U, ^{235}U, and ^{233}Pa extracted from the unloaded fuel of each preceding fuel cycle were used as fuel for the following fuel cycle.

Table 2. Multiplication factor in the beginning and in the end of fuel cycles

Number of fuel cycle	Beginning	End
1	1.120	1.005
2	1.103	1.006
5	1.103	1.010
10	1.102	1.009

In tab. 3, the composition of uranium isotopes extracted from the unloaded fuel after the 1st, 5th, and 10th fuel cycles is presented. The total mass of all isotopes of uranium at the beginning of the first fuel cycle is accepted as 100%.

Table 3. Isotopic composition of the uranium

Number of fuel cycle	^{232}U	^{233}U	^{234}U	^{235}U	^{236}U
1	0.002	99.7	6.5	0.4	0.02
5	0.060	100.3	23.1	3.6	0.9
10	0.085	100.5	34.7	6.0	3.1

CONCLUSIONS

The data in tabs. 2 and 3 are in fact the calculating substantiation of feasibility of the self-sufficient mode for a heavy-water power reactor of CANDU

type, operating in thorium-uranium fuel cycle. Thus, the goal is achieved: to demonstrate the practical feasibility of the self-sufficient mode in a heavy-water power reactor of CANDU type on the basis of proven technology.

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НАЧИН РУКОВАЊА КАНДУ ЕНЕРГЕТСКИМ РЕАКТОРИМА СА САМОДОВОЉНИМ ТОРИЈУМОВИМ ГОРИВНИМ ЦИКЛУСОМ

У раду су приказани резултати прорачуна руковања КАНДУ реакторима са торијумовим горивним циклусом. Прорачуни су спроведени ради опште процене изводљивости руковања тешководним енергетским реакторима на термичке неутроне посредством овог циклуса. Сматрано је да су параметри активног језгра и начини измене горива исти као при стандардном руковању реактором са уранијумовим циклусом.

Разматрана су два начина руковања: поступак са претходном акумулацијом ^{233}U и начин руковања самодовољним циклусом. За прорачуне руковања са претходном акумулацијом ^{233}U претпостављено је да се плутонијум користи као додатни фисиони материјал који обезбеђује неутроне за производњу ^{233}U. Плутонијум је смештен у горивним каналима док се ^{233}Th размешта у канале за мете. Максимални садржај ^{233}U у каналима за мете износио је 13 kg по тони ThO$_2$. То је достигнуто после шест година озрачивања. Започињање рада реактора у самодовољном моду захтева садржај ^{233}U не мањи од 12 kg по тони. У овом начину руковања, претпостављено је да су сви канали попуњени истоветним горивним ансамблима који садрже ThO$_2$ и извесну количину ^{233}U. Показало се да су у горивним ансамблима пожељније неравномерне расподеле ^{233}U.

Кључне речи: торијумов горивни циклус, КАНДУ реактор, управљање реактором