PRAMANA — journal of physics

Vol. 68, No. 2
February 2007
pp. 143–150

Operation of CANDU power reactor in thorium self-sufficient fuel cycle

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Abstract. This paper presents the results of calculations for CANDU reactor operation in thorium fuel cycle. Calculations are performed to estimate the feasibility of operation of heavy-water thermal neutron power reactor in self-sufficient thorium cycle. Parameters of active core and scheme of fuel reloading were considered to be the same as for standard operation in uranium cycle. Two modes of operations are discussed in the paper: mode of preliminary accumulation of ^{233}U and mode of operation in self-sufficient cycle. For the mode of accumulation of ^{233}U it was assumed for calculations that plutonium can be used as additional fissile material to provide neutrons for ^{233}U production. Plutonium was placed in fuel channels, while ^{232}Th was located in target channels. Maximum content of ^{233}U in target channels was estimated to be $\sim$13 kg/t of ThO_2. This was achieved by irradiation for six years. The start of the reactor operation in the self-sufficient mode requires ^{233}U content to be not less than 12 kg/t. For the mode of operation in self-sufficient cycle, it was assumed that all channels were loaded with identical fuel assemblies containing ThO_2 and certain amount of ^{233}U. It is shown that nonuniform distribution of ^{233}U in fuel assembly is preferable.

Keywords. Thorium cycle; CANDU; self-sufficient mode.

PACS Nos 28.41.Ak; 28.50.Hw

1. Introduction

Absence of ^{233}U isotope 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 following the β-decay. Thus, thorium, existing in nature as a single stable isotope, can be transformed using nuclear reactions into ^{233}U, which is used as fuel for power reactors. Reserves of thorium in the world exceed that of uranium by several times. Cost of thorium mining is much less than that of uranium because the radiation danger in the process of thorium mining is $\sim$100 times less than in the uranium mining. Possibilities of commercial production of ^{233}U in nuclear reactors were discussed many times in publications in connection with thorium–uranium fuel cycle. However, even partial substitution of uranium–plutonium fuel cycle by thorium–uranium cycle is faced with significant difficulties.

These disadvantages of thorium fuel cycle were seemingly the reasons why that in publications of last years, thorium was considered only as a raw material in nuclear reactors operating in uranium–plutonium fuel cycle mode (see, Boczar *et al* [1]). Such mixed mode allows more or less to save natural uranium during electric power production. Possibility of operating power reactors in thorium–uranium fuel cycle without feed by fissile materials was considered in preceding publications (see Boczar *et al* [1], Dastur *et al* [2]). However, the possibility was only discussed without demonstration of concrete ways of achievement.

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

In this paper, results of the calculations for self-sufficient mode for heavywater power reactor are presented. This reactor is hereafter referred to as 'T' reactor. Parameters of active core and scheme of re-fuelling of current heavy-water power reactor HWPR (CANDU) with heat power of 2776 MW [3] were used for calculations.

Calculations were made using code complex MCCOOR, developed on the basis of codes MCNP, COUPLE, ORIGEN-S. Nuclear data libraries ENDF-B/VI and JENDL 3.2 were used in MCNP code. Calculations of fuel burning and isotope transformations were performed step by step by the code ORIGEN-S with corrections of reaction rates at every step on the basis of calculations with MCNP code.

2. Mode of accumulation of ^{233}U

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

In the mode of accumulation of ^{233}U, tetragonal heavy-water lattice of reactor 'T' is composed of two types of channels, evenly distributed over the active core. Half of the channels are referred to as fuel channels, the other half are referred to as target channels. Fuel channel contains fuel assembly composed of 37 fuel elements with plutonium or enriched uranium. Target channel contains target assembly composed of 37 target elements. Each target element is in the form of zirconium tube filled with pellets of ThO_2. Figure 1 demonstrates part of the channel lattice with 16 channels and elementary cell accepted for calculations.

During the operation of the reactor 'T', power and neutrons are released in fuel channels due to the fission of ^{239}Pu and ^{241}Pu (^{235}U). In target channels, in the beginning of operation, excess neutrons are captured with the formation of ^{233}Pa. As ^{233}Pa decays to ^{233}U, power releases in target channels too. It is assumed that power of fuel channels with plutonium or enriched uranium is maintained to keep constant power density per mass unit of heavy nuclei, equal to that of current CANDU-type reactor. This condition determines neutron flux density in targets.

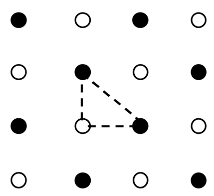

Figure 1. Lattice of channels in active core in the mode of accumulation of ^{233}U and elementary cell accepted for calculations.

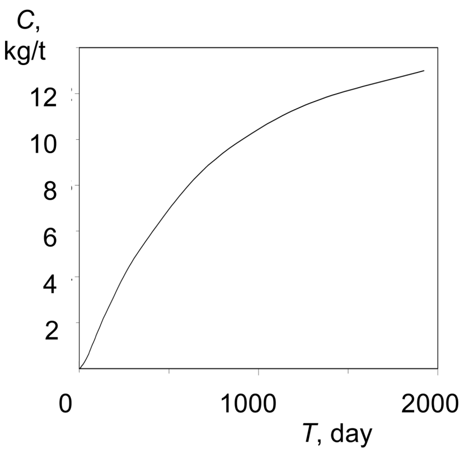

Figure 2. Variation of concentration of ^{233}U in ThO_2 in the mode of accumulation of ^{233}U.

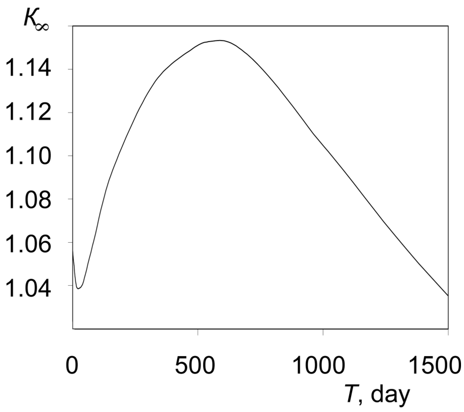

Figure 3. Variation of multiplication factor for elementary cell in the mode of accumulation of ^{233}U.

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

According to the data of figure 2, 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 a density of 10 g/cm^3). This result is close to that obtained by Bergelson *et al* [4] for thorium–uranium lattice and neutron spectrum in the fuel of CANDU reactor. Time of irradiation till maximum ^{233}U content in targets is 5–6 years, time for achievement of maximum multiplication factor K_∞ is about 2 years. During this time, ^{233}U content in targets reaches about 8 kg/t.

For start-up of reactor 'T' in the self-sufficient mode, it is necessary to load ThO_2 with ^{233}U content 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 instance, three cycles of irradiation of ThO_2 targets during 2 years. Therefore, in the mode of accumulation of ^{233}U, reactor 'T' should work approximately for 6 years. After each cycle of irradiation, targets should be unloaded from the reactor and replaced by fresh targets without ^{233}U. In order that the reactor 'T' should not become subcritical during target reloading, replacement of irradiated targets by fresh targets without ^{233}U should not be simultaneous. As ^{233}U is accumulated in targets, total power of the reactor increases in accordance with the given mode of operation, while power of fuel channels remains constant. It is apparent that the considered parameters of operation of reactor 'T' 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 the reactor by the following measures:

- compensations of reactivity increase (see figure 3) due to replacement of fuel channels by target channels;
- optimization of the number and locations of target channels and fuel channels in the active core;
- optimization of the scheme of reloading of target channels;
- increase the number of targets in the target assembly [1], reducing plutonium (uranium) load in fuel elements, that would result in increase of thermal neutron flux in targets at the same power of reactor 'T'.

3. Self-sufficient mode

In the self-sufficient mode, it is assumed that fuel elements containing ThO_2 with a certain content of ^{233}U are loaded in all channels of the reactor. Considered mode could be realized in the reactor 'T', if the following two conditions are met (that is nontrivial task at the strict restriction of total amount of ^{233}U).

Condition 1. Amount of ^{233}U and its layout in channels must ensure overcriticality of the reactor 'T' 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 $\Delta K \geq 0.1$. It is obvious that the reactor 'T' should have overcriticality not less than this value.

Condition 2. In the reactor 'T' under the acceptable burnup of ^{233}U and other fissile isotopes, reproduction of ^{233}U must ensure at least equality of content of fuel in the active core before and after the fuel cycle. This condition together with the reactivity margin determine the fuel cycle. Breeding ratio and its variations during the fuel cycle 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 cycle in the reactor 'T' was estimated by the method, which is suitable for CANDU reactors with continuous bidirectional re-fuelling. In this case, fuel assemblies with different burnups in the range from zero to maximum burnup are present in active core during the whole fuel cycle. Such condition can be described by average multiplication factor determined by the leakage of neutrons from the active core, absorption of neutrons by structure materials of active core and by control rods. For the CANDU reactor with control rods in active core, this value is 1.045, and for extracted control rods it is 1.035 [1]. Fuel burnup W_0 in the reactor 'T' was estimated in accordance with the parameter α as

$$\alpha = \int_0^{W_0} K_\infty \mathrm{d}W/W_0,$$

where W is the fuel burnup. Dependence $K_\infty(W)$ was calculated in the interval between initial value $W = 0$ corresponding to the fresh fuel, and final value W_0 corresponding to the fuel unloaded from the reactor. In the process of calculating

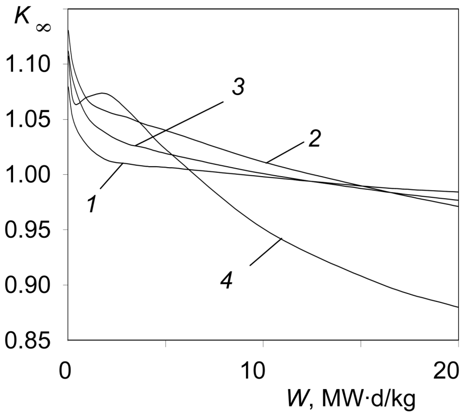

Figure 4. Multiplication factor for elementary cell in the self-sufficient mode: 1, 2, 3 – 1st, 2nd, and 3rd variant of ^{233}U layout in fuel assemblies of the 'T' reactor, 4 – CANDU reactor with natural uranium.

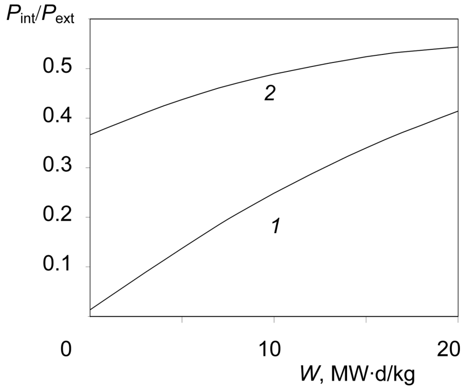

Figure 5. Ratio of the total power of 19 internal fuel elements to total power of 18 external fuel elements in fuel assembly: 1, 2 – 2nd and 3rd variant of ^{233}U layout in fuel assemblies of the 'T' reactor.

of $K_\infty(W)$, neutronic parameters were calculated at different values of W with the step of 1 MW·d/kg in order to take into account change of neutron spectrum in fuel.

Calculations were made for different burnups and for three variants of initial layout of ^{233}U in the active core of the reactor 'T'.

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

In the second variant, 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 Th without ^{233}U.

In the third variant, 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. Balance on the uranium was ensured due to change of density of ^{233}U+ThO_2 (figures 4 and 5, table 1).

Calculated fuel burnup of the CANDU reactor with natural uranium is in good agreement with operating value of 8.3 MW·d/kg [3]. So, there is good reason to think that fuel burnups calculated for three variants of fuel layout in fuel assemblies of the reactor 'T' using the same method with same neutronic constants, are close to real values. Data of table 1 demonstrate that under the even distribution of ^{233}U over all fuel elements of the reactor 'T' (first variant), fuel burnup is unacceptably low, while initial overcriticality of active core is less than 10% (curve 1 in figure 4). Fuel burnup can be increased to 15 MW·d/kg due to 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 shielding of internal fuel elements allows to increase burnup approximately by 10 times. At nonequal load of ^{233}U into external and internal

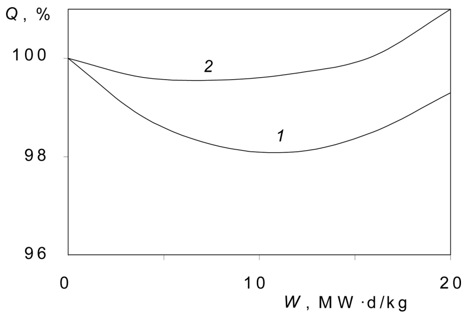

Figure 6. Variation of the total mass of ^{233}U, ^{235}U, ^{233}Pa during the fuel cycle: 1, 2 – 2nd and 3rd variant of ^{233}U arrangement in fuel assemblies of the 'T' reactor.

fuel elements in each assembly according to the third variant, in which shielding of internal fuel elements is less than in the second variant, burnup decreases by 1.5 times with respect to the second variant. As one would expect, heat power of the reactor 'T' for the third variant is much higher than for the second variant.

It is obvious that possibility of practical realization of self-sufficient mode in the reactor 'T' depends mainly on the amount of fissile isotopes in the fuel unloading from the active core in the end of the next fuel cycle. The curves in figure 6 describe changes of total amount of ^{233}U, ^{235}U and ^{233}Pa during the fuel cycle for the second and the third variants of fuel layout (initial value is accepted as 100%).

These changes do not exceed 1–2% of the initial amount, i.e. they are within the error interval of our calculations. So, these results can be considered as preliminary data, demonstrating the possibility of operation of the reactor 'T' in self-sufficient mode.

For obtaining additional data, calculations of multiplication factors $K_\infty(0)$ and $K_\infty(W_0)$ were made for the series of 10 fuel cycles for the third variant of fuel layout in cold reactor 'T'. $K_\infty(0)$ and $K_\infty(W_0)$ were values of K_∞ in the beginning and end of each fuel cycle in the series. Fuel burnup of 8 MW·d/kg was accepted in these calculations, that corresponded to the fuel cycle of 430 days (table 2). Nuclides ^{233}U, ^{235}U and ^{233}Pa extracted from unloaded fuel of each preceding fuel cycle were used as fuel for the following fuel cycle. In table 3, composition of

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

	Fuel burnup (MW·d/kg)			Heat power for $\alpha = 1.035$
Reactor	$\alpha = 1.025$	$\alpha = 1.035$	$\alpha = 1.045$	(MW)
CANDU	9.4	8.1	6.7	1750 (Directory, 1994)
'T', variant 1	3.4	1.5	0.7	2760 (Directory, 1994)
'T', variant 2	17.5	14	11	1400–1900
'T', variant 3	10.1	6.7	4	1900– 2000

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

Fuel cycle number	Beginning	End
1	1.120	1.005
2	1.103	1.006
5	1.103	1.010
10	1.102	1.009

Table 3. Isotopic composition of uranium.

Fuel cycle number	^{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

uranium isotopes extracted from unloaded fuel after the 1st, 5th and 10th fuel cycle is presented. Total mass of all isotopes of uranium at the beginning of the first fuel cycle is accepted as 100%.

Comparison of the data of tables 2 and 3 demonstrates that accumulation of uranium isotopes, mainly ^{234}U, does not result in the reduction of the multiplication factor. This fact can be explained by accumulation of fissile ^{235}U, which compensates neutron capture in ^{234}U and ^{236}U.

Table 3 demonstrates that the amount of ^{232}U in self-sufficient mode increases to a level, for which hard gamma radiation of its daughter nuclides exceed significantly prescribed radiation standards. Therefore, practical development of thorium self-sufficient mode will require application of remote handling technologies in reprocessing the irradiated fuel and manufacturing new fuel. It is apparent that generating the atomic power in the 21st century is possible only with the application of closed nuclear fuel cycle. This means that independent of the type of fuel cycle (U–Pu or Th–U), development and application of remote handling technologies will be required. It is also necessary to remember that the presence of ^{232}U is favourable for non-proliferation

We have not taken into account the possible losses of ^{233}U in reprocessing and fabrication of new fuel and also uncertainties in nuclear data in our calculations. However, these effects can lead only to a little change in fuel burnup as can be seen from figure 4.

4. Conclusions

The power and fuel burnup obtained for the reactor 'T' give only estimation of the real values. As with the mode of accumulation of ^{233}U, use of the parameters of current CANDU reactor for the determination of conditions, in which self-sufficient mode is possible, does not allow to obtain solutions close to optimum. However, it should be emphasized that authors only had a limited goal: to demonstrate the practical feasibility of self-sufficient mode in heavy-water power reactor of CANDU type on the basis of proven technology. Therefore, optimization calculations by including economic considerations were not performed at this stage. At the same time, experience of calculation study of such systems allows to make the following conclusion. The development of advanced heavy-water reactor with specific requirements of mode of ^{233}U accumulation and self-sufficient mode will allow to improve highly such most important parameters as the time of fuel accumulation, burnup, and power. Herewith problems connected with the safety should also be solved.

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 of accumulation of required amount of ^{233}U, this disadvantage is inherent not only to the discussed modes, but also to thorium–uranium fuel cycle in any of its modifications.

References

[1] P Boczar, G Dyck, P Chan and D Buss, Recent advances in thorium fuel cycles for CANDU reactors, *Proc. of Three IAEA Meetings on Thorium Fuel Utilization: Options and Trends*, IAEA-TECDOC-319, 2002, p. 104, 120

[2] A Dastur, D Menely and D Bass, Thorium cycle options in CANDU reactors, *Proc. of the Internat. Conf. on Evaluation of Emerging Nuclear Fuel Cycle Systems*, Global'95, September 11–14, 1995, Versailles, France, **2**, 1908 (1995)

[3] Directory of Nuclear Power Plants in the World (1994); Japan Nuclear Energy Information Centr. Co., Ltd, Tokyo, Japan

[4] B R Bergelson and G V Tikhomirov, *Breeding of* ^{233}*U in heavy-water blanket of ADS*, Atomic Energy, 2001, **91**, 91 (2001) (in Russian)