ONCE-THROUGH URANIUM THORIUM FUEL CYCLE IN CANDU REACTORS

Özdemir, Sancak*; Çubukçu, Erol**,
*Nuclear Safety Department, Turkish Atomic Energy Authority Ankara/TURKEY
** Hacettepe University, Nuclear Energy Engineering Department

ABSTRACT

In this study, the performance of the once-through uranium-thorium fuel cycle in CANDU reactors is investigated. (Th-U)O_2 is used as fuel in all fuel rod clusters where Th and U are mixed homogeneously. CANDU reactors have the advantage of being capable of employing various fuel cycle options because of its good neutron economy, continuous on line refueling ability and axial fuel replacement possibility. For lattice cell calculations transport code WIMS is used. WIMS cross-section library is modified to achieve precise lattice cell calculations. For various enrichments and Th-U mixtures, criticality, heavy element composition changes, diffusion coefficients and cross-sections are calculated. Reactor core is modeled by using the diffusion code CITATION. We conclude that an overall saving of 22% in natural uranium demand can be achieved with the use of Th cycle. However, slightly enriched U cycle still consumes less natural Uranium and is a lot less complicated.

1. INTRODUCTION

The thorium cycle, as an alternative to the uranium cycle, for both heavy and light water moderated reactors is considered in an effort to extend the nuclear fertile material resources because of the lower sensitivity of electricity generation cost to the speculative uranium prices. While most of the scientific basis required is already available, some engineering demonstrations are needed to provide better economic data for rational decisions. There appears to be no major feasibility problem associated with the use of thorium, although development is needed in the area of fuel testing and fuel management.

In studies of the use of thorium cycles in these types of reactors, it is generally accepted that thorium cycles can be employed with no basic change in reactor design, and that existing reactors could operate on the thorium cycle.

The fact that the supply of low-cost uranium is limited, motivated a number of studies devoted to the search for alternative fuel cycles with improved uranium utilization for the present generation of nuclear power plants. One method is to add the significant thorium resources to the world's nuclear resource base. And also, advanced converters based on the U/Th cycle can give much more energy per pound of uranium than advanced converters based on U/Pu cycles.

The neutronically preferred fuel for thermal spectrum reactors is U233, the fissile material bred from thorium. Thermal spectrum reactors with conversion ratios close to unity can be developed with U233 as the fuel. U233 can be created and stockpiled for possible future use with no imperative that it be used unless future conditions warrant. Thus production of a U233 stockpile can be begun independently of the state of reprocessing technology.

Since U233 has superior nuclear characteristics in the thermal spectrum when compared to plutonium, its creation should be the direct objective of fuel cycle design. The plutonium created from natural uranium in a CANDU-PHW is relatively dilute in spent fuel, whereas U233 created directly in a once-through cycle is about five times more concentrated. Thus by creating U233 directly, ultimate reprocessing costs per gram of fissile material extracted from

spent fuel may be reduced if once-through cycles are ever used to produce this material. Reprocessing and refabrication must be performed by remote control because, spent fuel containing U233 contains small amounts of U232 (10-50ppm of fuel), which is the precursor of a family of γ-active nuclides[1].

Because of these considerations, closed cycles and Pu topping once-through cycles are eliminated and once-through U235 topping Th cycle is chosen.

The utilization of thorium fuel in CANDU reactors has been a long-standing goal, because of the superior fissile properties of U233 bred from Th232 in a thermal spectrum [1]. Fuel bundle simplicity (fabrication and handling) is an extra advantage of CANDU reactors for new fuel cycles. The other advantage in CANDU reactors is the on-line axial fuel management practiced, which introduces a new degree of freedom into design calculations. Canada and India have studied thorium cycles for 20 years and are continuing both theoretical and experimental studies. It is anticipated that essentially the same reactor system can be adapted to employ the thorium cycle.

Another concern about the enrichment of fissile material is the proliferation of nuclear weapon grade materials. One suggestion is that fissile uranium isotopes, occurring at any point in the fuel cycle, be associated with enough U238 that an isotopic separation would be required to obtain weapons usable material. Weapons usable material can be defined as

a) separated plutonium with any isotopic composition, or

b) separated uranium for which

$$\frac{W_3 + xW_5}{W_{TU}} > 0.12$$

where W_3 is the weight of U233, W_5 is the weight of U235, W_{TU} is the total weight of uranium, and x is a weighting factor with a value of 0.6.[2]

For the case (a), the associated production of plutonium is relatively small amount. For (b), limitation is the topping material used which could be less than 20 w/o U235.

With all these considerations assessed, there are two ways to perform the once-through U235 topping thorium fuel cycle in CANDU reactors. The first one is the "heterogeneous case" in which Th and U fuel bundles are burned in separate channels. The other one is the "homogenous case" in which Th and U are mixed homogeneously and burned. In "heterogeneous case", power peaking factors are very high and flattening of power shape is very difficult. In this study, "homogeneous case" is chosen for the analysis.

2. COMPUTER CODES EMPLOYED

For Lattice cell calculations WIMS code was used. WIMS is a general lattice cell program, which uses transport theory to calculate flux as a function of energy and position in the cell. In the interests of economy of machine time this is done in two steps: WIMS first calculates the spectra for few spatial regions in the full number of energy groups of its library, and uses these spectra to condense the basic cross-sections into few groups. A few group calculation is then carried out using a much more detailed spatial representation. The resulting fluxes are then expanded using the spectra of the previous calculation, so that the reaction rates at each spatial point can be calculated in the library group structure [3].

To perform the lattice cell calculations, the WIMS cross-section library was modified. Modifications include the addition of Th decay chain data which has been generated at H.U.

Nuclear Energy Engineering Department, and the modification of Pu decay chain and cross-section data sets according to the results of JAERI in Japan. We demonstrated that the modified library performs extremely well by comparing the results with those presented in the Canadian report [1]. The comparison is shown in Figures 1.

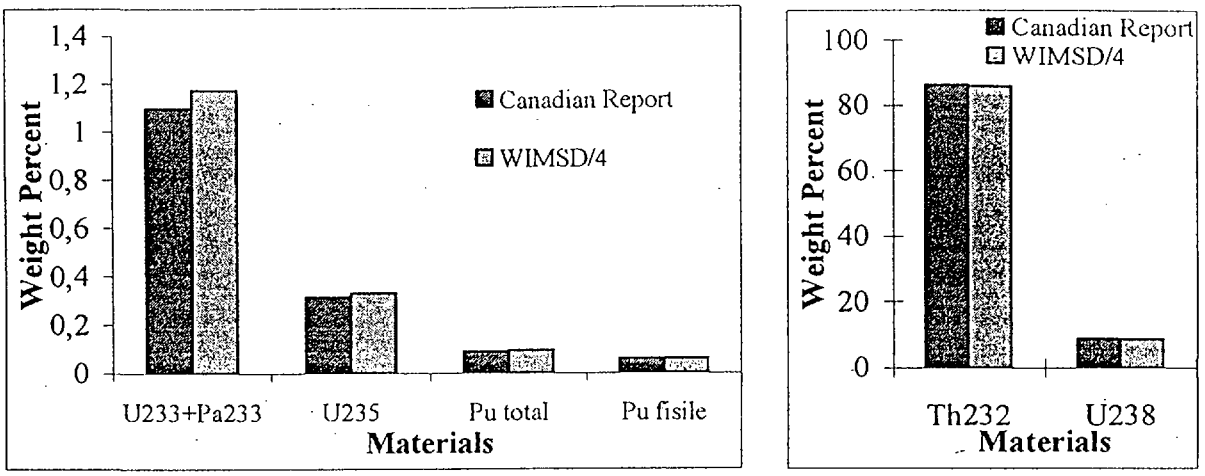

Figure 1. Comparison of Canadian report[1] with WIMSD/4 results

For two dimensional core modeling, diffusion code CITATION was used. CITATION can be used for two or three dimensional modeling. Cross-section library can be prepared in two different formats; first, a set of macroscopic cross-sections obtained from lattice-cell transport codes, second, a set of microscopic cross-sections with number densities. These data can be obtained from output of WIMS code. In the calculations, macroscopic cross-section set was used.

3. CALCULATIONAL METHODS AND MODELING

Reactor Configuration

The physics of the reactor determined by the quantity and distribution of various materials in the calandria. These materials constitute:

- A lattice of fuel channels arranged in a moderator of heavy water.
- Various reactivity devices that are used to alter the rate of neutron multiplication.
- Neutron detector assemblies that provide a measurement of the rate of neutron multiplication at suitable locations.
- A reflector of heavy water that surrounds the above materials and which reduces the rate of neutron leakage from the cylindrical surface of the calandria.

In these calculations, first, lattice cell parameters were calculated and discussed. Consequently, core modeling performed and discussed. From above list, second and third options were not included to the calculations. At the end of the calculations, effects of those options are discussed.

Lattice Cell Parameters

The fuel channels are arranged on a square lattice pitch of 28.575 cm. The fuel consists of 37 elements of uranium-thorium dioxide sheathed in Zircaloy and held together as a bundle by end plates. These fuel bundles are enclosed in zirconium-niobium pressure tubes which form part of the heat transport circuit. The pressure tubes are supported by Zircaloy calandria tubes using

garter springs suitably located along each channel. The pressure and calandria tube dimensions used in the calculation of the lattice parameters are given in Table 1. [4]

Table 1. Lattice cell data [4]

Fuel	37 element UO_2
Element outside diameter	13.081 mm
Average sheath wall thickness	0.419 mm
Pellet outside diameter	12.154 mm
Stack length	480.31 mm
UO_2 density (Natural U)	10.45 g/cm^3
UO_2 area	41.206 cm^2
Coolant area	34.969 cm^2
UO_2 weight per bundle (Natural U)	18.8 kg
Pressure tube (Zr:2.5 percent Nb) inside diameter	103.378 mm
Average pressure tube wall thickness	4.343 mm
Calandria tube (Zr-2) inside diameter	129.0 mm
Average calandria tube wall thickness	1.397 mm

Other data used in calculation of the lattice parameters is given in [4]. The calculation of lattice parameters precedes the calculation of the power distribution. Some data required for lattice parameter calculation is affected by the power distribution. Such data is revised as the design proceeds.

The Lattice parameters are calculated as neutron cross-section. They determine the reaction rates of various processes in the lattice for a given neutron flux. These are neutron absorption, production, diffusion and moderation. The cross-sections depend on neutron energy and the composition and temperature of the lattice components. Changes in composition occur mainly in the fuel. Neutron irradiation leads to the formation of fission products and of the depletion and production of heavy isotopes.

The first step of calculations is calculation of k_{inf} vs. burnup values for the cases below;

Table 2. Material compositions for lattice cell calculations

Case	Composition	Case	Composition
1	2.8% U235 + 87.2% U238 + 10% Th	11	3.3% U235 + 86.7% U238 + 10% Th
2	2.8% U235 + 77.2% U238 + 20% Th	12	3.3% U235 + 76.7% U238 + 20% Th
3	2.8% U235 + 67.2% U238 + 30% Th	13	3.3% U235 + 66.7% U238 + 30% Th
4	2.8% U235 + 57.2% U238 + 40% Th	14	3.3% U235 + 56.7% U238 + 40% Th
5	2.8% U235 + 47.2% U238 + 50% Th	15	3.3% U235 + 46.7% U238 + 50% Th
6	2.8% U235 + 37.2% U238 + 60% Th	16	3.3% U235 + 36.7% U238 + 60% Th
7	2.8% U235 + 27.2% U238 + 70% Th	17	3.3% U235 + 26.7% U238 + 70% Th
8	2.8% U235 + 17.2% U238 + 80% Th	18	3.3% U235 + 16.7% U238 + 80% Th
9	2.8% U235 + 7.2% U238 + 90% Th	19	3.3% U235 + 6.7% U238 + 90% Th
10	2.8% U235 + 0% U238 + 97.2% Th	20	3.3% U235 + 0% U238 + 96.7% Th

The results of these calculations are showed in Figures 2 and 3.

Figure 2 shows that below 28.000 MWD/t k_{inf} of 0%Th mixed fuel is higher than the others. In other words, below the 28.000 MWD/t there is no additional positive reactivity effect with thorium content. Since the Th content increased, the U238 content is decreased and therefore the production of Pu is decreased. Above 28.000 MWD/t, with same thorium content

criticality can be achieved and additional burnup can be provided. This situation has also an advantage of flattening the reactivity over the burnup cycle of a fuel bundle. Additional burnup is enough and difference between the criticality values of 0% Th mixed fuel and the others are significant. So, these cases can be used for core calculations where burnup level exceeds 28000 MWD/t.

Figure 3. shows that below 33.000 MWD/t k_{inf} of 0% Th mixed fuel is higher than the others. In other words, below the 33.000 MWD/t, there is no additional positive reactivity effect with thorium content. Above 33.000 MWD/t and with some thorium content, criticality can be achieved and additional burnup can be provided. This situation has also an advantage of flattening the reactivity over the burnup cycle of a fuel bundle. Additional burnup is enough and the difference between the criticality values of 0% Th mixed fuel and the others are significant. Therefore, these cases can be used for core calculations where burnup level exceeds 33000 MWD/t.

In conclusion of these results, for 2.8% U235 and 3.3% U235 fissile uranium isotope contents, it is observed that Th can be used effectively from neutronic point of view.

Material compositions versus burnup values are also calculated. These values can be seen in Figure 4. Discussion of the figure showed that with 2.8% U235 and 3.3% U235 content and 30%, 50% and 70% Th compositions, reasonable core models can be formed. For these cases, core models were formed, calculated and discussed in the next section.

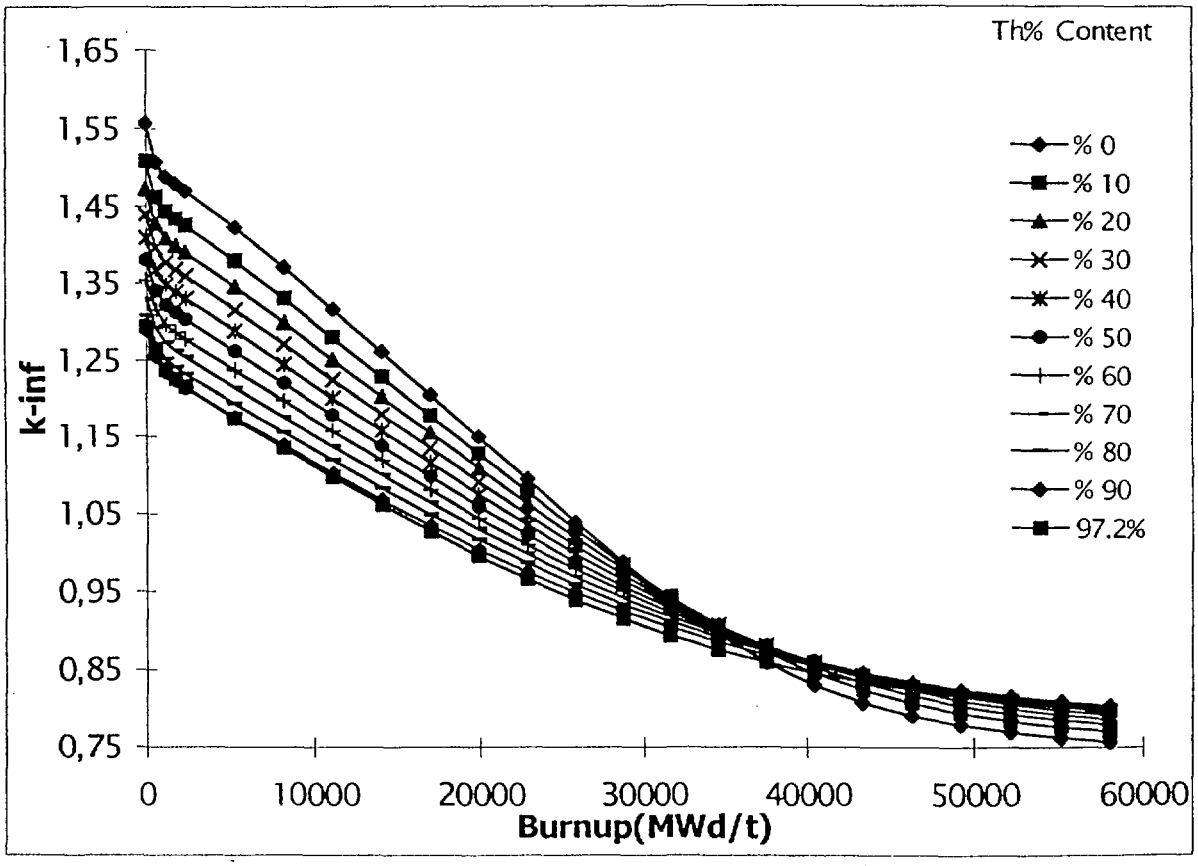

Figure 2. 2.8% U235 + 0% U238 + 97.2% Th to 2.8% U235 +97.2% U238 + 0% Th thoria fuel versus burnup

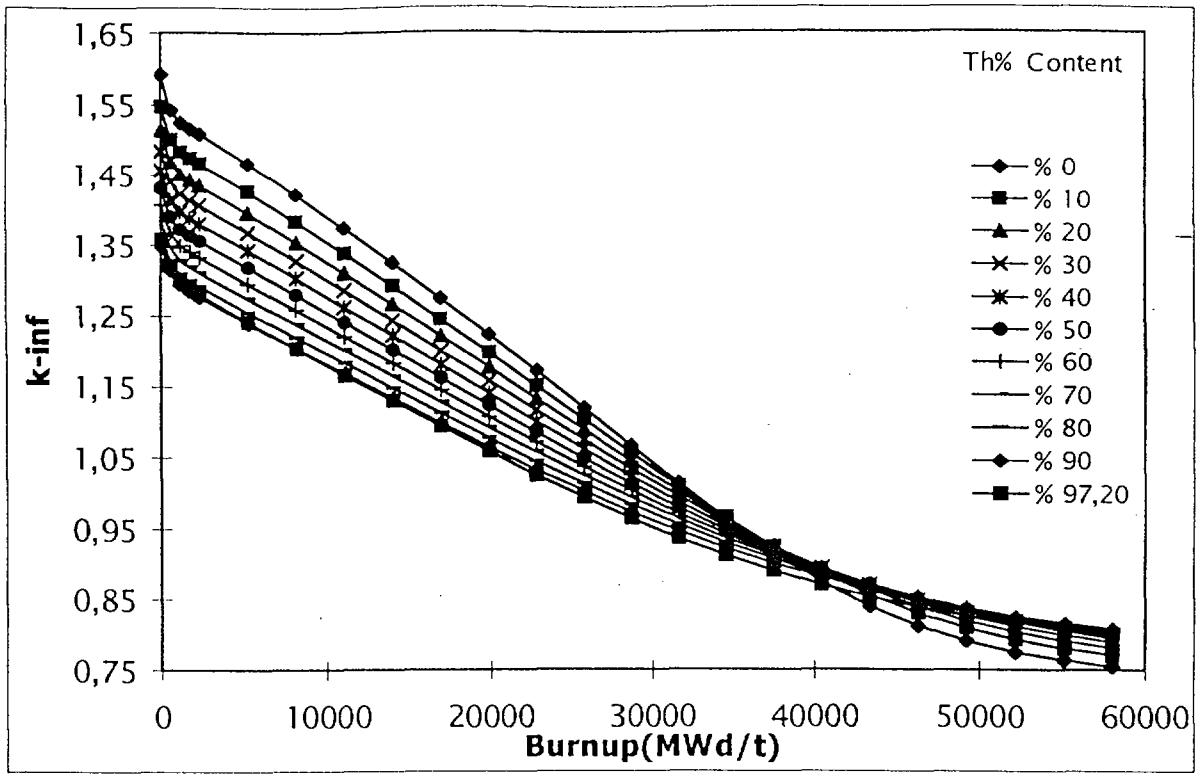

Figure 3. 3.3% U235 + 0% U238 + 96.7% Th to 3.3% U235 + 96.7% U238 + 0% Th thoria fuel versus burnup

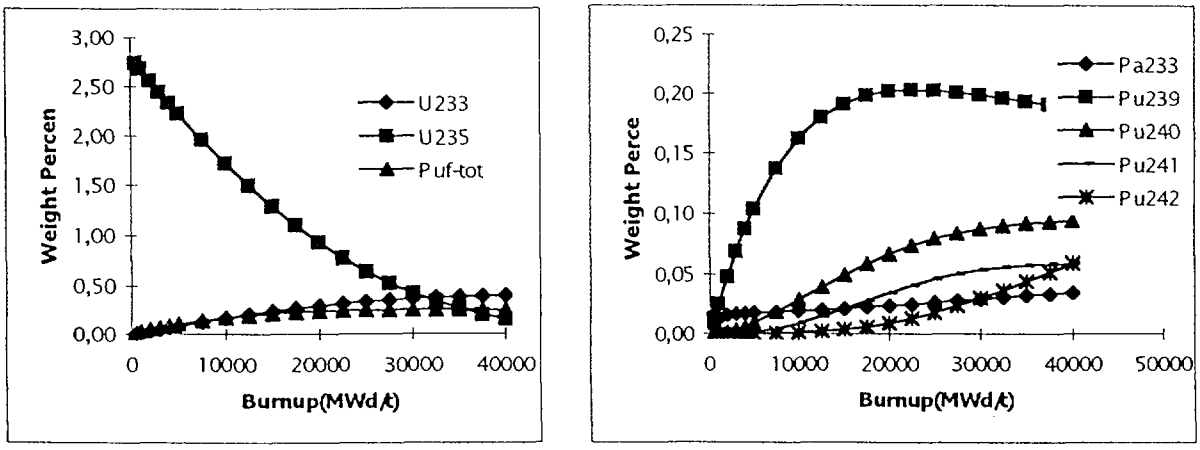

Figure 4. Change Of Fissile Material Weight Percents and Change of Pu isotopes and Pa233 weight percents vs. Burnup For (2.8% U235 + 67.2% U238 + 30%Th) Fuel .

Core Modeling

A quarter of core was modeled as four burnup zones with 25, 23, 23, and 24 fuel channels, respectively. All structural materials were included except guide tubes for the adjusters, zone controllers, shut-off rods, mechanical control absorbers, flux detectors and poison injection nozzles. The lattice properties for reflector and all fuel zones are obtained by executing WIMSD/4 computer code. The core was divided into four burnup regions, 25, 23, 23, and 24 fuel channels, respectively. The temperatures of the moderator, coolant and fuel were taken as

73°C, 290°C and 936 °C, respectively. The moderator and coolant purity were assumed as 99.722 atom percent.

Acceptance criteria for core calculations are:

- Criticality must be greater than one
- Power peaking factor (point power/average power) is low enough
- Reasonable burnup level must be achieved.

From lattice cell calculations, Cases 3, 5, 7, 13, 15, and 17 were selected for core modeling.

With these compositions, various core configurations were modeled. For all cases criticality, power peaking factors and exit burnups were calculated. Results of these calculations showed that maximum achievable exit burnup level can be 50000 MWD/t.

From these results, annual thorium and natural uranium consumption, required separative work for enrichments, inventories and basically annual fuel costs are calculated, compared with reference natural uranium fuel cycle and tabulated in Table 3.

1. RESULTS AND DISCUSSION

The results of this study, as given in Table 4, are compared to the reference natural uranium fuel cycle from the point of view of annual fuel consumption, required separative work for enrichment, inventory, and annual fuel consumption cost. In this table, the maximum burnup level is taken to be 50000 MWD/t. The best core configurations obtained are based on cases 3, 5, 7, 13, 15, and 17.

For listed fuel compositions, annual fuel consumption of different fuel compositions were calculated to be 107, 105, 104, 123, 125, and 127 tons natural uranium per year, respectively. As can be seen from these values, natural uranium consumption can be decreased down to 104 tons/year. For these cases, thorium consumption are 6, 10, 14, 6, 10, and 14 Th tons/year, respectively. The cost of Th consumption is very low when compared to that of uranium, because of the low unit cost of Th (20 Canadian $/kg)

Table 3. Basic Values of Economic Parameters for CANDU PHWR
(All costs are 1978 Canadian dollars) [4]

Parameters	value	unit
U_3O_8 Cost	117	$/kg(U)
ThO_2 Cost	20	$/kg(Th)
Seperative Work Cost	100	$/kgSWU
Natural UO_2 fabrication cost	55	$/kg(U)
Enriched $(U\text{-}Th)O_2$ fab. Cost	75	$/kg(HE)
Conversion U_3O_8-UF_6	5	$/kg(U)

In annual fuel cost calculations, values in Table 3. are used. These values are very speculative, however. Commercial cost of ThO_2 is unknown. $(U\text{-}Th)O_2$ CANDU PHWR fuel is assumed to be 75 Canadian $/kg HE. With these economic parameters, annual fuel cost calculations are performed. For above listed cases, annual fuel costs are 26065, 24493, 23338, 30031, 29353, 28803, 23541 $x10^3$Canadian $, respectively. Among these values, 23338 $x10^3$Canadian $/y represents the lowest achievable annual fuel cost for thorium cycle. This value compares to the natural nranium fueled reference cycle case (0.711% U235 + 99.299% U238). These values are nearly equal. This result shows that there is no advantage of using Once Through Thorium Fuel Cycle versus the reference natural uranium fuel cycle. But Once Through Thorium Fuel Cycle still competes with the natural uranium fueled reference cycle. However, other economic parameters, especially investment costs have a dominant role. In the case of Once Through Thorium Fuel Cycle, to attain the high burnup level of 50.000 MWD/t, some changes are required in the fuel design. Therefore, the investment cost will be higher. The advantage of

Once Through Thorium Fuel Cycle is the significant U233 content in the spent fuel. It can be recovered and used as fuel as well. If U233 can be reprocessed and used, this fuel cycle can be considered as an alternative one.

Table 4. Comparison of Thorium fuel cycles in this study with natural uranium once-through fuel cycle

Fuel Composition	2.8%U235 27.2%U238 70%Th	2.8%U235 47.2%U238 50%Th	2.8%U235 67.2%U238 30%Th	3.3%U235 26.7%U238 70%Th	3.3%U235 46.7%U238 50%Th	3.3%U235 66.7%U238 30%Th	0.7%U235 99.3%U238 0%Th
Burnup (MWd/t)	50000	50000	50000	50000	50000	50000	7500
Equilibrium Feed Fuel	$(U+Th)O_2$	$(U+Th)O_2$	$(U+Th)O_2$	$(U+Th)O_2$	$(U+Th)O_2$	$(U+Th)O_2$	UO_2
Equilibrium Net Feed Rates for 1GWe at 80% Load Factor							
Equivalent Natural	107	105	104	123	125	127	133
Thorium (Mg Th/a)	6	10	14	6	10	14	0
Seperative Work (kg	89,6	87,9	91,0	112,4	114,3	116,1	0
Annual fuel cost ($x10^3$ $/a)	26065	24493	23338	30031	29353	28803	23541
Inventories for 1GWe at 80% Load Factor							
Equivalent Natural	1060	1469	2422	1260	1740	2864	140
Thorium (Mg Th/a)	61	138	315	61	138	315	0
Seperative Work (kg	2806	1541	992	3363	1873	1227	0

CONCLUSION

Upon investigating the use of Once Through Thorium Fuel Cycle in CANDU reactors, we concluded that:

The use of thorium is beneficial at only high burnup values (≥20000 MWD/t). The optimized cycle, that we propose, requires a burnup level of 50000 MWD/t. However, such high burnup values require changes in the fuel design.

A reduction of 22 tonnes/y in natural uranium consumption can be achieved by using the thorium cycle when compared to natural uranium cycle.

The reduction in the consumption of natural uranium is not enough to compete with the slightly enriched uranium fuel cycle proposed for CANDU.

Therefore, unless reprocessing is employed or the purpose of the thorium cycle is to produce U233 in the spent fuel, thorium can not be considered as a fuel for CANDU reactors.

In our study, we model the reactor core in two dimensions and ignoring the presence of any reactivity control mechanism. For future works, we suggest a 3-D modeling of the core and we believe that the effect of control mechanisms (especially when fabricated from fertile Th232) should also be taken into account.

REFERENCES

1. M. S. MILGRAM, "Once-through Thorium Cycles in CANDU Reactors", AECL-7516, *Atomic Energy of Canada Limited.* (1982)
2. "Database for a CANDU-PHWR Operating on the Thorium Cycle"- AECL-6595 *Atomic Energy of Canada Limited.* (1979)
3. SLAVIC, S., B. ZEFRAN, M. RAVNIK, Wimsd/4 for PC-AT386 Computer, version 1.0, *Ljubljana*, 1990
4. "Akkuyu CANDU-1 Nuclear Generating Station Akkuyu, Turkey- Preliminary Safety Report" v.1 *Atomic Energy of Canada Limited,* January 1984