ANALYSIS OF REUSABILITY OF ThO_2 AND SPENT UO_2 FUELS ENRICHED WITH ADS IN A CANDU REACTOR

by

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The study presents the analysis of the reusability of ThO_2 and spent UO_2 fuels enriched in two different ADS reactors fuelled with Minor Actinide. The spent UO_2 fuels are taken out from pressurized water reactor and CANDU spent fuels. For this analysis, the CANDU-37 reactor having a total fission thermal power of 2156 MW is considered and 14 different cases of enriched fuels taken from the previous enrichment processes are analysed by burning in this reactor. The 3-D and time-dependent critical burnup calculations are carried out by using the MCNP 2.7 code. To determine the effective burn time of each case, these calculations are performed until the values of kinf decrease to about the criticality threshold of 1.05 for all investigated cases. The percentages of the ^{239}Pu and ^{233}U fissile isotopes appear to be below weapons-grade plutonium and uranium, respectively, in all enriched fuel cases. At the end of effective burn times, the burnup values can reach the values varying in the range of 26.770 and 33.540 GWd/MTU which are a mean of 3.5-4.5 times the burnup value of the CANDU-37 reactor fed with the NatUO_2 fuel. The results of this study bring out that in terms of energy production, the CANDU-37 reactor fuelled with the ThO_2 and spent UO_2 fuels enriched in ADS designs demonstrates higher neutronic performance than the conventional CANDU-37 reactor.

Key words: CANDU reactor, thermal reactor, accelerator-driven system, enriched spent fuel utilization, thorium utilization

INTRODUCTION

The management of spent fuels is one of the most significant issues complicating nuclear development. Recently, the *wait and see* method has been preferred by a lot of countries for nuclear waste management. On the other hand, some researchers study reusing spent fuel in various reactors as fuel.

Tak *et al*. [1] work on the optimization of the Ultra-Long Cycle Fast Reactor (UCFR) design. They use a pressurized water reactor (PWR) fuelled with spent fuel in their designs that can be operated at full power for 60 years without refuelling. Hartanto and Kim [2] utilize spent fuel in Traveling Wave Reactor. Monte Carlo code McCARD is used in their calculations. some parameters like fission power, reactivity coefficients, and delayed neutron fraction are analysed. Sugawara [3] perform a study on transuranium transmutation in a molten salt accelerator-driven system (ADS) having 400 MW thermal power. They bring out that 100-120 kg plutonium transmutation can be achieved in one year period. Meng *et al.* [4] aim to increase the transmutation of minor actinide (MA) efficiency in an ADS having a power of 800 MW and fuelled with a mixture of mononitride and plutonium as fuel. They demonstrated that the minor actinides can be transmuted efficiently in the ADS.

On the other hand, a lot of researchers investigate the performance of Canada Deuterium Uranium (CANDU) reactors fuelled with various spent fuels and their mixtures. Sahin *et al*. [5] investigate a CANDU reactor fuelled with a mixture of ThO_2 and nuclear waste. They use several compositions to obtain a uniform power distribution. They bring out that the best fuel mixture is thorium and 14 % minor actinide dioxide (MAO_2) in terms of long-term reactivity and uniform power distribution. Sahin *et al*. [6] perform a power flattening operation in CANDU fuel rods in their study. They investigate three fuel cases, the natural UO_2 case, the light water reactor (LWR) spent fuel case and a mixture of LWR spent fuel and ThO_2 fuel, for the CANDU reactor. They bring out that the LWR spent fuel has a high burnup. The other studies of Sahin *et al*. [7] work on weapon-grade plutonium in TRISO form in

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CANDU reactors. Three different mixtures of thorium with plutonium are considered. They achieve high burnup values and transmutation of nuclear waste. In another, Sahin *et al*. [8] work on spent fuel and thorium utilization in a CANDU reactor. They show that a good amount of spent fuel utilization occurs.

Jeong *et al*. [9] investigate homogeneous ThO_2UO_2 and heterogeneous ThO_2UO_2-DUPIC fuels in CANDU by using dynamic analysis. The results bring out that the spent fuel amount can be decreased by 13 % and 18 % for the homogeneous and heterogeneous fuel cycles, respectively. Gholamzadeh *et al*. [10] investigate the usage of transuranic fuel with thorium in a CANDU 6 reactor by using the Monte Carlo N-particle transport code. They obtain negative temperature reactivity coefficients and also show that better transmutation is obtained in a 1.0 % TRU-containing fuel matrix case than 0.5 % TRU-containing fuel matrix. Ozdemir *et al*. [11] study two spent fuel utilization in a PWR-CANDU system. The calculations are performed with the MONTEBURNS code. Their results show that their system decreases the natural uranium requirement since considerable burnup is obtained. Ellithi and Al-Khawlani [12] investigate the performance of the CANDU reactor by loading a mixture of a certain ratio (3-5 %) of plutonium isotopes and thorium fuel into the reactor core. The neutronic calculations of reactions occurring in the CANDU reactor core are performed by the MCNPX code. The results show that a mixture of plutonium isotopes and thorium fuel oxides has a higher value than natural uranium in terms of the multiplication factor. Bergelson *et al*. [13] use simulations to anticipate how the CANDU reactor would operate with a self-sufficient thorium cycle. The calculations bring out that the self-sufficient mode is suitable for a CANDU heavy water-cooled power reactor using the thorium-uranium fuel cycle. Mirvakili *et al*. [14] perform neutronic calculations in the CANDU6 thermal reactor's fuel rods loaded with ThO_2 and UO_2 fuels. They have demonstrated that experimental and computational studies can be done for future alternatives in terms of the neutron economics of thorium-based fuels as a result of their findings. Yang *et al*. [15] investigate to come up with a fuel cycle strategy for recycling uranium-thorium production in CANDU reactors. They analyse in-core fuel management, neutronic performance, spent fuel characteristics, and resource usage. The findings of the design and evaluation show that it is technically feasible to recycle uranium-thorium production in existing CANDU reactors. Saldideh *et al*. [16] investigate the possibility of using CANDU reactors to tap into the world's thorium supplies. They use the MCNP code to calculate how long it took the reactor to reach criticality in their analysis. As a result, they bring out that mixing PuO_2 and ThO_2 enabled a CANDU reactor to function for longer periods without requiring refuelling.

In recent years, our research group carries out many studies [17-22] on spent fuel rejuvenation, nuclear waste transmutation, and fuel enrichment in various subcritical reactors. Arslan *et al*. [17] investigate the transmutation of fission products in an accelerator-drive system using spent fuels taken out from CANDU and PWR thermal reactors. For neutronic calculations, they employ the computer programs MCNPX 2.7 and CINDER 90. This study shows that a significant quantity of energy is released in the accelerator-driven system as well as effectively the transmutation of fission products. Bakir *et al*. [18] attempt to establish refuelling periods for spent fuels in CANDU-37 reactors in a D-T fusion reclamation reactor supplied with a mixture of $^{Nat}UO_2$ and ThO_2. The results show that the designed fusion breeder reactor has a high performance in terms of energy production and fuel rejuvenation. Bakir *et al*. [19] investigate the breeding of fissile fuel and the transmutation potential of transuranic isotopes in a gas-cooled ADS. It is observed that the designed ADS has a high neutronic performance in terms of the conversion of nuclear wastes and the production of effective energy. Bakir *et al*. [20] study a helium gas-cooled ADS loaded with ThO_2, UO_2, PuO_2, and CmO_2 TRISO fuel particles to obtain a flattened power profile, and they achieve obtaining a quasi-flattened power profile in the subcritical zone of ADS. Bakir *et al*. [21] examine the neutronic performances of the system's fissile fuel production and energy generation using LBE + natural UO_2 and LBE + 15 % enriched UO_2 as target materials in an accelerator-driven system. In terms of fissile fuel breeding and energy gain, the developed system by this study has been shown to have a good neutronic performance.

In our previous study [22], the UO_2 fuels taken out from the PWR and CANDU spent fuels and the ThO_2 fuels are enriched in two different ADS reactors (DESIGN A and DESIGN B) fuelled with the MAO_2 fuels extracted from PWR-MOX spent fuels. This study shows that as well as an energy gain of 10.8-25.1, spent fuels can be enriched up to 2.49-4.23 %. The numerical results demonstrate that both ADS designs have a good neutronic performance and significant energy generation potential as well as fuel enrichment capability. As this study is a continuation of the previous work [22], it is analysed that these fuels enriched at different periods, tab. 1, in the ADS designs A and B are reused in a CANDU-37 reactor as fuel.

MONTEBURNS code [23] also can be used for time-dependent calculations, but our previous study [18] brings out that the results of the calculations performed with both codes are compatible with each other, so in this study, the calculation with the use of MONTEBURNS code is not considered necessary.

Table 1. Summary table for the most important neutronic data obtained from the numerical calculations in DESIGN A and B [22]

FEZ	MAZ	BOC EOC [d]	k_{eff}	Cumulative fuel enrichment, CFFE [%]			^{233}U [kg]	MATF* [%]	G	Burnup [GWd/MTU]
				MAZ	FEZ1	FEZ2				
DESIGN A										
ThO_2	MOX12	0	0.904	91.21	0	–	0	–	0	0
		1250	0.985	86.12	3.07	–	104.783	9.21	10.8	25
	MOX22	0	0.910	88.59	0	–	0	–	0	0
		991	0.986	84.47	2.85	–	97.829	7.97	11.2	21
CANSF	MOX12	0	0.916	91.21	0.17	–	–	–	0	0
		920	0.985	86.68	2.76	–	–	7.98	18.4	29
	MOX22	0	0.927	88.59	0	–	–	–	0	0
		630	0.985	85.44	2.49	–	–	5.91	17.9	20
PWRSF	MOX12	0	0.943	91.21	0.91	–	–		0	0
		530	0.985	87.94	3.38	–	–	5.70	21.0	20
	MOX22	0	0.952	88.59	0.91	–	–	–	0	0
		300	0.985	86.37	2.95	–	–	4.07	25.1	14
	**MOX22B	300	0.939	87.36	0.91	–	–	–	0	0
		600	0.985	85.40	2.79	–	–	3.04	20.8	11
DESIGN B										
CANSF	MOX12	0	0.907	91.21	0.17	0	0	–	0	0
		2050	0.985	83.10	3.63	3.85	160.444	15.17	15.6	37
	MOX22	0	0.915	88.59	0.17	0	0	–	0	0
		1700	0.986	81.94	3.33	3.58	150.853	13.11	15.5	30
PWRSF	MOX12	0	0.913	91.21	0.91	0	0	–	0	0
		1950	0.985	83.41	4.23	3.84	160.673	14.56	15.4	35
	MOX22	0	0.922	88.59	0.91	0	0	–	0	0
		1600	0.985	82.09	3.99	3.58	151.183	12.97	16.0	30

* TF is transmutation fraction. The initial masses of MOX12 and MOX22 compositions are 5541.39 and 5550.08 kg, respectively.
**At the second cycle (see the sub-sections "Cumulative Fissile Fuel Enrichment" and "Sample additional analysis")
Note: MOX12 and MOX22 abbreviations are transferreed from [24] and the other abbreviations are transferred from [22]

THE CANDU REACTOR DESCRIPTION

It appears that the CFFE values of the spent UO_2 and the ThO_2 fuels enriched in the two different ADS reactors (DESIGN A and DESIGN B) fuelled with the MAO_2 fuels in our previous study [22] can be sufficient to reuse them in thermal reactors. In light of these data, in this study, to effectively reburn the enriched fuels taken out from the ADS reactors, the CANDU-37 thermal reactor is considered. The CANDU reactor, developed in Canada, is a type of thermal nuclear reactor fuelled with natural uranium ($^{Nat}UO_2$) fuel composed of 0.005 % ^{234}U, 0.711 % ^{235}U, and 99.284 % ^{238}U [24]. The most important feature that differentiates it from other conventional thermal reactors is the use of natural uranium as fuel, as well as the use of heavy water (D_2O) as a coolant and moderator. The CANDU-37 reactor is a typical CANDU reactor having a total fission thermal power of 2156 MW_{th} and contains 380 fuel channels consisting of 37 fuel elements [24]. The fuel elements consist of 12 fuel rods with a length of 49.53 cm fitted longitudinally end-to-end. Thus, the total length of fuel channels is 595 cm. The radial cross-section view of one-quarter of a heterogeneous CANDU-37 fuel channel is plotted in fig. 1. As can be seen from this figure, the fuel rods are surrounded by the pressure tube, CO_2 gap, and calandria tube from the inside to the outside in the channel filled with D_2O. The fuel channels are placed inside the cylindrical reactor vessel as a square lattice with a pitch of 28.575 cm.

The performances of the CANDU-37 reactor fuelled with 14 different fuels enriched in the two different ADS reactors (DESIGN A and DESIGN B) fuelled with the MAO_2 fuels in [22] are individually analysed for each fuel case. To compare the neutronic performances of these fuels with the $^{Nat}UO_2$ fuel, the CANDU reactor fuelled with the $^{Nat}UO_2$ fuel is also analysed. The definitions of the fuel cases are given in tabs. 2(a) and 2(b) along with their atomic densities. The fuel enrichment processes are in detail explained in [22]. Furthermore, the non-fuel materials used in the CANDU-37 reactor are given in tab. 3.

The statement ADS Designs A and B and the abbreviations MAZ (minor actinide zone), FEZ1 and FEZ2 (fuel enrichment zones 1 and 2) are in detail described in CONCEPTUAL ADS DESIGNS of ref. [22]. Furthermore, they are shown in detail in fig. 1 of [22].

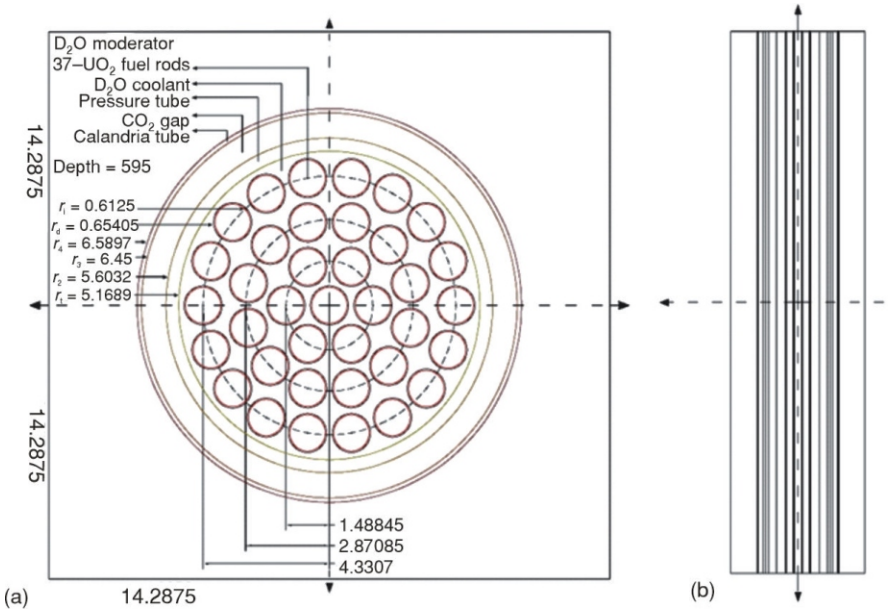


Figure 1. Cross-section views of a heterogeneous CANDU -37 fuel channel; (a) radial and (b) axial (dimensions are in cm and except the height and width of the fuel channel are in scale) [24]

Table 2(a). Atomic densities* of spent fuels enriched in ADS reactor (DESIGN A) in [22] in atoms/(barn-cm), and definitions of fuel cases

	DESIGN A					
	MOX12			MOX22		
ISOTOPE	CASE A11 ThO_2 FEZ1	CASE A12 CANSF FEZ1	CASE A13 PWRSF FEZ1	CASE A21 ThO_2 FEZ1	CASE A22 CANSF FEZ1	CASE A23 PWRSF FEZ1
^{16}O	$9.02572\ 10^{-3}$	$9.80267\ 10^{-3}$	$9.80354\ 10^{-3}$	$9.02580\ 10^{-3}$	$9.80285\ 10^{-3}$	$9.80368\ 10^{-3}$
^{232}Th	$4.27236\ 10^{-3}$	–	–	$4.30191\ 10^{-3}$	–	–
^{233}Pa	$7.16056\ 10^{-6}$	–	–	$7.94531\ 10^{-6}$	–	–
^{233}U	$1.34690\ 10^{-4}$	–	–	$1.25750\ 10^{-4}$	–	–
^{234}U	$1.03008\ 10^{-5}$	–	–	$8.60766\ 10^{-6}$	–	–
^{235}U	$1.35552\ 10^{-6}$	$2.01827\ 10^{-6}$	$2.13062\ 10^{-5}$	$1.01135\ 10^{-6}$	$2.91953\ 10^{-6}$	$2.58787\ 10^{-5}$
^{236}U	–	$4.57658\ 10^{-6}$	$2.13785\ 10^{-5}$	–	$4.54451\ 10^{-6}$	$2.10154\ 10^{-5}$
^{238}U	–	$4.54197\ 10^{-3}$	$4.57313\ 10^{-3}$	–	$4.63049\ 10^{-3}$	$4.64447\ 10^{-3}$
^{237}Np	–	–	$1.56229\ 10^{-6}$	–		$1.15654\ 10^{-6}$
^{239}Pu	–	$1.08405\ 10^{-4}$	$1.26667\ 10^{-4}$	–	$1.03380\ 10^{-4}$	$1.09015\ 10^{-4}$
^{240}Pu	–	$3.17692\ 10^{-5}$	$2.32187\ 10^{-5}$	–	$2.52733\ 10^{-5}$	$1.61100\ 10^{-5}$
^{241}Pu	–	$1.79431\ 10^{-5}$	$1.19702\ 10^{-5}$	–	$1.17995\ 10^{-5}$	$6.37685\ 10^{-6}$
^{242}Pu	–	$5.41972\ 10^{-6}$	$1.64907\ 10^{-6}$	–	$2.39649\ 10^{-6}$	
TOTAL	$1.34516\ 10^{-2}$	$1.45148\ 10^{-2}$	$1.45844\ 10^{-2}$	$1.34710\ 10^{-2}$	$1.45837\ 10^{-2}$	$1.46277\ 10^{-2}$
CFFE [%]	3.07	2.76	3.38	2.85	2.49	2.95

*Atomic densities taken after burned in the range of 11-29 GWd/MTU in the previous study [22], (for the details see tab. 1)

Minor actinide compositions, MOX12 and MOX22 are used in ADS in [22] and are in detail explained in the Sub-critical Minor Actinide zone section of [22]. They do not use in this study.

Some expressions in tabs. 2(a) and 2(b): In CASE, the initial capital, the first number (1 or 2), and the second number (1 or 2 or 3) represent the Design types (A or B), the minor actinide compositions (MOX12 or MOX22) and row numbers (1 or 2 or 3), respectively. CANSF and PWRSF represent CANDU and PWR spent fuels enriched in the ADS designs in [22], respectively. These enrichment processes are in detail explained in [22].

CALCULATION TOOLS

In this study, the neutronic calculations are made by considering an assembly instead of the whole CANDU 37 reactor and only the neutronic performances of the enriched fuels in ADS DESIGN A and B are investigated.

The MCNPX version 2.7 computer code [25] written by Los Alamos National Laboratory is used for neutronic calculations. This code can be defined generally as a general-purpose Monte Carlo N-Particle code that can be used for neutron, photon, electron, or coupled neutron-photon-electron transport [25]. The MCNPX 2.7 can calculate a lot of tallies (such as sur-

Table 2(b). Atomic densities* of spent fuels enriched in ADS reactor (DESIGN B) in [22] in atoms/(barn-cm), definitions of fuel cases

DESIGN B								
	MOX12				MOX22			
ISOTOPE	CASE B11 CANSF FEZ1	CASE B12 CANSF FEZ2	CASE B13 PWRSF FEZ1	CASE B14 PWRSF FEZ2	CASE B21 CANSF FEZ1	CASE B22 CANSF FEZ2	CASE B23 PWRSF FEZ1	CASE B24 PWRSF FEZ2
^{16}O	$9.80212\ 10^{-3}$	$9.02592\ 10^{-3}$	$9.80288\ 10^{-3}$	$9.02593\ 10^{-3}$	$9.80229\ 10^{-3}$	$9.02598\ 10^{-3}$	$9.80301\ 10^{-3}$	$9.02595\ 10^{-3}$
^{232}Th	–	$4.17222\ 10^{-3}$	–	$4.17976\ 10^{-3}$	–	$4.21814\ 10^{-3}$	–	$4.21908\ 10^{-3}$
^{233}Pa	–	$6.14672\ 10^{-6}$	–	$6.32101\ 10^{-6}$	–	$6.44725\ 10^{-6}$	–	$6.82329\ 10^{-6}$
^{233}U	–	$1.64990\ 10^{-4}$	–	$1.65225\ 10^{-4}$	–	$1.55127\ 10^{-4}$	–	$1.55466\ 10^{-4}$
^{234}U	–	$1.63935\ 10^{-5}$	–	$1.58835\ 10^{-5}$	–	$1.36073\ 10^{-5}$	–	$1.36273\ 10^{-5}$
^{235}U	$4.61482\ 10^{-6}$	$2.85101\ 10^{-6}$	$2.86033\ 10^{-5}$	$2.71107\ 10^{-6}$	$4.98445\ 10^{-6}$	$2.06008\ 10^{-6}$	$3.01334\ 10^{-5}$	$2.04780\ 10^{-6}$
^{236}U	$4.31961\ 10^{-6}$	–	$2.04030\ 10^{-5}$	–	$4.29979\ 10^{-6}$	–	$2.02707\ 10^{-5}$	–
^{238}U	$4.59750\ 10^{-3}$	–	$4.56683\ 10^{-3}$	–	$4.63706\ 10^{-3}$	–	$4.59707\ 10^{-3}$	–
^{237}Np		–	$2.06578\ 10^{-6}$	–	–	–	$1.88133\ 10^{-6}$	–
^{239}Pu	$1.55396\ 10^{-4}$	–	$1.62742\ 10^{-4}$	–	$1.44760\ 10^{-4}$	–	$1.53130\ 10^{-4}$	–
^{240}Pu	$2.14711\ 10^{-5}$	–	$1.76129\ 10^{-5}$	–	$1.75085\ 10^{-5}$	–	$1.46932\ 10^{-5}$	–
^{241}Pu	$1.22560\ 10^{-5}$	–	$9.35112\ 10^{-6}$	–	$9.36904\ 10^{-6}$	–	$7.07194\ 10^{-6}$	–
^{242}Pu	$1.20163\ 10^{-6}$	–	–	–	–	–	–	–
TOTAL	$1.45989\ 10^{-2}$	$1.33885\ 10^{-2}$	$1.46105\ 10^{-2}$	$1.33958\ 10^{-2}$	$1.46203\ 10^{-2}$	$1.34214\ 10^{-2}$	$1.46273\ 10^{-2}$	$1.34230\ 10^{-2}$
CFFE [%]	3.63	3.85	4.23	3.84	3.33	3.58	3.99	3.58

*Atomic densities taken after burned in the range of 30-37 GWd/MTU in the previous study [22], (for the details see tab. 1)

face current, surface and volume fluxes, point and ring detectors, particle and fission heatings, *etc*.) as 3-D. All neutronic reactions specified in a particular cross-section evaluation (such as ENDF/B-VI), are accounted. Furthermore, thermal neutrons can be described by both the free gas and $S(\alpha,\beta)$ models. The criticality calculation is performed with the KCODE card and time-dependent burnup reactions can be easily simulated with the BURN TIME option of MCNPX 2.7code. (BURN TIME = 30 9R MAT = 1 2 3 4 MAT = = 1 2 3 4BOPT = 1.0, 24.0, 1.0).

The neutronic calculations are performed for one fuel channel model with a total fuel channel length of 595 1. The four sides have a reflective boundary condition, and the top and bottom surfaces have a white boundary condition. Three dimension nuclear processes can be successfully simulated by this code. Nonetheless, the MCNPX2.7 version of MCNPN is required for the time-dependent neutronic calculations.

In addition to this code, to correctly evaluate quite large and complicated outputs of MCNPX 2.7, XBURN [26] interface computer code is used. The time-dependent critical burnup/depletion calculations are performed until the values of kinf decrease to about the criticality threshold of 1.05 for all investigated cases and thus the effective burn time of each case is determined.

NUMERICAL RESULTS

Infinite neutron multiplication factor

The ratio of the neutrons produced with fission in one generation (N_{pf}) to the number of neutrons lost with absorption reactions in the preceding generation (N_{la}) is defined as the infinite neutron multiplication factor (k_{inf}) and this factor can be formulated mathematically as follows

$$k_{inf} \quad \frac{N_{pf}}{N_{ln}} \tag{1}$$

Table 3. Non-fuel materials used in CANDU-37 reactor [24]

Materials	Elements	Fraction [%]
Clad (Zirc-4) (6.56258) gcm^{-3}	Cr	$9.99101\ 10^{-4}$
	Fe	$2.09811\ 10^{-3}$
	Zr	$9.81417\ 10^{-1}$
	Sn	$1.44870\ 10^{-2}$
	Hf	$9.99101\ 10^{-4}$
	Total	100
Coolant (D_2O) ($8.45610\ 10^{-1}$ gcm^{-3})	D	$6.66667\ 10^{-1}$
	O	$3.33333\ 10^{-1}$
	Total	100
Pressure tube (Zirc2.5Nb) (6.54332 gcm^{-3})	Zr	$9.75000\ 10^{-1}$
	Nb	$2.50000\ 10^{-2}$
	Total	100
CO_2 (GAP) ($1.58089\ 10^{-3}$ gcm^{-3})	C	$3.33333\ 10^{-1}$
	O	$6.66667\ 10^{-1}$
	Total	100
Calandria tube (Zirc-2) (6.56135 gcm^{-3})	Cr	$9.99101\ 10^{-4}$
	Fe	$1.34879\ 10^{-3}$
	Ni	$5.49505\ 10^{-4}$
	Zr	$9.81617\ 10^{-1}$
	Sn	$1.44870\ 10^{-2}$
	Hf	$9.99101\ 10^{-4}$
	Total	100
Moderator (D_2O) (1.08522 gcm^{-3})	D	$6.66667\ 10^{-1}$
	O	$3.33333\ 10^{-1}$
	Total	100

Table 4. The most important neutronic data obtained from the numerical calculations in CANDU-37 reactor and the effective burn times

	Fuel type	Fuel case	BOC EOC* [d]	k_{inf}	CFFE [%]	^{233}U [%]	^{239}Pu [%]	Burnup [GWd/MTU]
	$^{Nat}UO_2$	$^{Nat}UO_2$	0	1.119	0.711	–	98.43	0
			180	1.053	0.591	–	76.33	7.674
ADS DESIGN A	ThO_2 FEZ1	CASE A11	0	1.461	3.07	91.95		0
			189	1.050	1.75	77.26		24.270
		CASE A21	0	1.424	2.85	92.83		0
			168	1.050	1.72	77.51		21.470
	CANSF FEZ1	CASE A12	0	1.572	2.76	–	66.05	0
			173	1.049	0.87	–	26.77	20.320
		CASE A22	0	1.564	2.49	–	72.20	0
			156	1.051	0.81	–	30.64	18.060
	PWRSF FEZ1	CASE A13	0	1.614	3.38	–	77.26	0
			215	1.049	0.62	–	28.05	24.90
		CASE A23	0	1.600	2.95	–	82.36	0
			186	1.051	0.91	–	31.80	21.340
ADS DESIGN B	CANSF FEZ1	CASE B11	0	1.634	3.63	–	81.45	0
			232	1.049	0.95	–	29.56	26.770
		CASE B21	0	1.628	3.33	–	83.80	0
			212	1.051	0.89	–	31.55	24.350
	CANSF FEZ2	CASE B12	0	1.557	3.85	89.40	–	0
			257	0.977	1.85	70.90	–	33.460
		CASE B22	0	1.528	3.58	90.71	–	0
			236	1.049	1.81	73.31	–	30.500
	PWRSF FEZ1	CASE B13	0	1.653	4.23	–	85.27	0
			271	1.049	1.08	–	27.95	31.200
		CASE B23	0	1.652	3.99	–	87.15	0
			255	1.049	1.05	–	29.46	29.260
	PWRSF FEZ2	CASE B14	0	1.557	3.84	89.74	–	0
			258	1.085	1.85	71.19	–	33.540
		CASE B24	0	1.528	3.58	90.73	–	0
			236	1.104	1.82	73.41	–	30.490

* In this study, EOC also means effective burn time

The k_{inf} value being less or equal to or greater than 1 indicates that the reactor is in a subcritical or critical or supercritical mode, respectively. This factor, therefore, is one of the most important parameters to be considered primarily in reactor calculations.

In each fuel case, the effective burn times during which the reactor can be effectively operated are determined by performing the time-dependent critical burnup/depletion calculation until the value of k_{inf} decreases to the criticality threshold of 1.05 as shown in tab. 4 for all fuel cases. One can see easily from these results that the CANDU-37 reactor fuelled with the enriched fuels in the ADS DESIGN B operates for a longer time than that fuelled with the enriched fuels in the ADS DESIGN A and also the $^{Nat}UO_2$ fuel.

Figures 2(a) and 2(b) show the decreases in k_{inf} values in all enriched fuel cases taken out from ADS DESIGN A and B, respectively, during the operation time. Figure 2(a) shows also the decrease in value k_{inf} in the $^{Nat}UO_2$ fuel case during the operation time. Furthermore, the k_{inf} values at the beginning and end of the cycle (at the BOC and EOC, respectively) in all investigated fuel cases are given in tab. 4. It can be observed in these figures that as the k_{inf} profile in the $^{Nat}UO_2$ fuel case decreases gradually from about 1.118 to 1.055 after 180 days, those in the other fuel cases decrease more rapidly from 1.4-1.6 to 1.05. The k_{inf} profile in the CANDU 37 reactor fuelled with the $^{Nat}UO_2$ fuel is almost the same as the k_{inf} profiles in the other CANDU reactor studies [6, 12, 13] in the literature. In another word, it is in good agreement with them. Furthermore, its effective burn time (180 days) is shorter than that of fuels taken from ADS DESIGN B. This means that the CANDU 37 reactor fuelled with the enriched spent fuels in ADS DESIGN B can be longer operated than that fuelled with $^{Nat}UO_2$ fuel.

Since the starting CFFE percentages of the fuels taken from ADS in [22] are different (they vary in the range of 2.49 %-4.23 %), it is a normal situation that the burn times in the CANDU-37 reactor are also different.

Cumulative fissile fuel enrichment

In nuclear fission reactors, the ratio of the sum of the atomic density of all fissile isotopes to the sum of

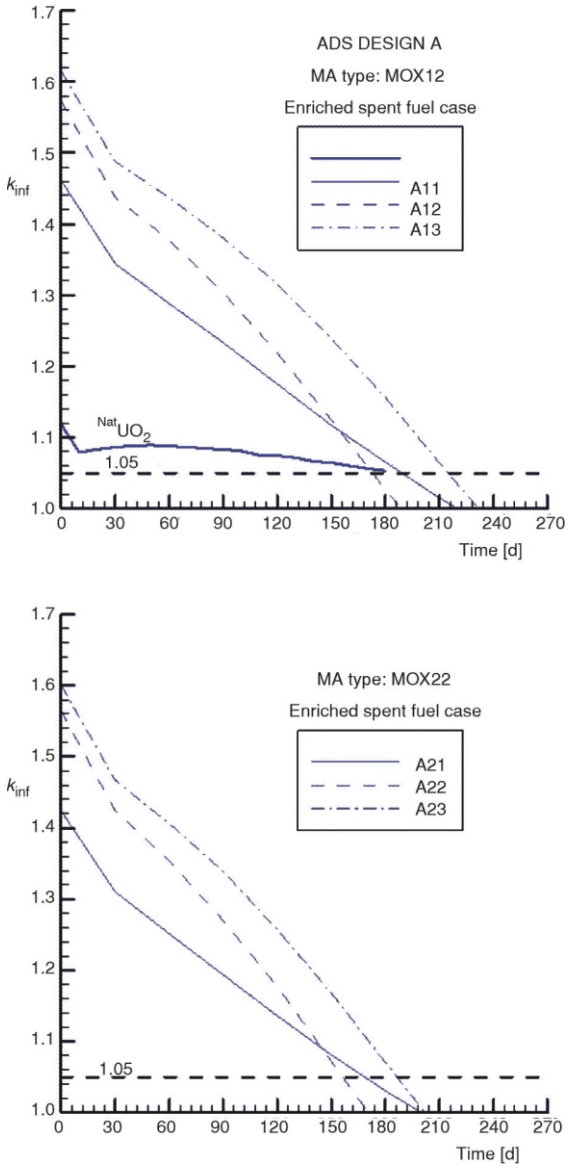


Figure 2(a). During operation time, decreases in effective neutron multiplication factor values in all enriched fuels taken out from ADS DESIGN A ($^{Nat}UO_2$ case that it is in good agreement with [6, 12, 13])

the atomic densities of all isotopes with atomic numbers greater than or equal to 90 is expressed as the Cumulative fissile fuel enrichment (CFFE). It indicates the quality of nuclear fuel and can be calculated in percentages as follows

$$CFFE \quad \frac{N_{\text{fissile fuel}}}{N_{\text{nuclear fuel}}} \; 100\,[\%] \qquad (2)$$

where N denotes the atomic density of isotopes in atoms, cm^{-3}.

In our cases, the fissile fuels are ^{233}U, ^{235}U, ^{237}Np, ^{235}Np, ^{239}Pu, ^{241}Pu, ^{244}Pu, ^{241}Am, ^{243}Am, ^{245}Cm, and ^{247}Cm. Choosing these fissile fuels is based on microscopic fission reaction cross-section on thermal neutron region.

Figures 3(a) and 3(b) depict the decreases in CFFE values of all enriched fuel cases taken out from ADS DESIGN A and B, respectively, during the operation time. In addition to these figures the CFFE values at the BOC and EOC in all investigated fuel cases are given in tab. 4. Comparing this table with tab. 1 and tab. 4 in the previous study [22], it will be seen that the initial CFFE values of the fuels in this table are equal to the CFFE values of the same fuels at the EOC in tab. 1 and tab. 4 in the previous study [22]. As apparent from these figures the CFFE profiles are curvilinear and relatively more rapid decline. At the end of effective burn times (at the EOC), these decreases vary in the range of 0. 0.62 % and 1.85 % according to the fuel cases (see tab. 4). Later, if desired, these CFFE values can be increased by enriching in the ADS DESIGN in the same way. In the $^{Nat}UO_2$ case, CFFE values decrease from 0.711 % to 0.591 % after 180 days (its effective burn time). As apparent from tab. 4, these CFFE values are lower than those of enriched fuels in ADS DESIGN.

Fissile isotope fractions

During operation time, the decreases in ^{239}Pu and ^{241}Pu fractions in plutonium in all enriched fuel cases taken out from ADS DESIGN A and B are plotted in figs. 4(a) and 4(b), respectively. Furthermore, the decreases in ^{233}U and ^{235}U fractions in uranium in the same enriched fuel cases are also plotted in fig. 5. In addition to these figures the percentages of ^{233}U and ^{239}Pu in uranium and plutonium are given in tab. 4, respectively. As apparent from figs. 4(a) and 4(b), the percentages of total of ^{239}Pu and ^{241}Pu in plutonium vary in the range of 77 % (in the case of A12) to 91 % (in the case of B23) in all fuel cases at the BOC, and they decrease gradually to around 39 % during the operation time. Figure 5 shows that although the percentages of total ^{233}U and ^{235}U in uranium are around 94 % in all fuel cases at the BOC, they gradually decreased to around 80 % after approximately 60 days. These values demonstrate that the percentages of a total of ^{239}Pu and ^{241}Pu in all fuel cases are below the weapons-grade plutonium (about 93 %). At BOC, the percentages of a total of ^{233}U and ^{235}U (above 90 %) can be denatured by mixing with natural uranium. Nonetheless, these fuels by burning in the CANDU-37 reactor fissile percentage can be decreased in a short time (about 60 days).

Fuel burnup

In nuclear reactors, fuel burnup known as fuel utilization is one of the most important parameters and is defined as a measure of how much energy is obtained from a primary nuclear fuel source. The value of burnup can be calculated in GWd/MTU or MWd/MTU as follows

$$\text{Burnup}(t \quad t) \quad \text{Burnup}(t) \quad \frac{\text{Fission power}}{MTU} \quad t \qquad (3)$$

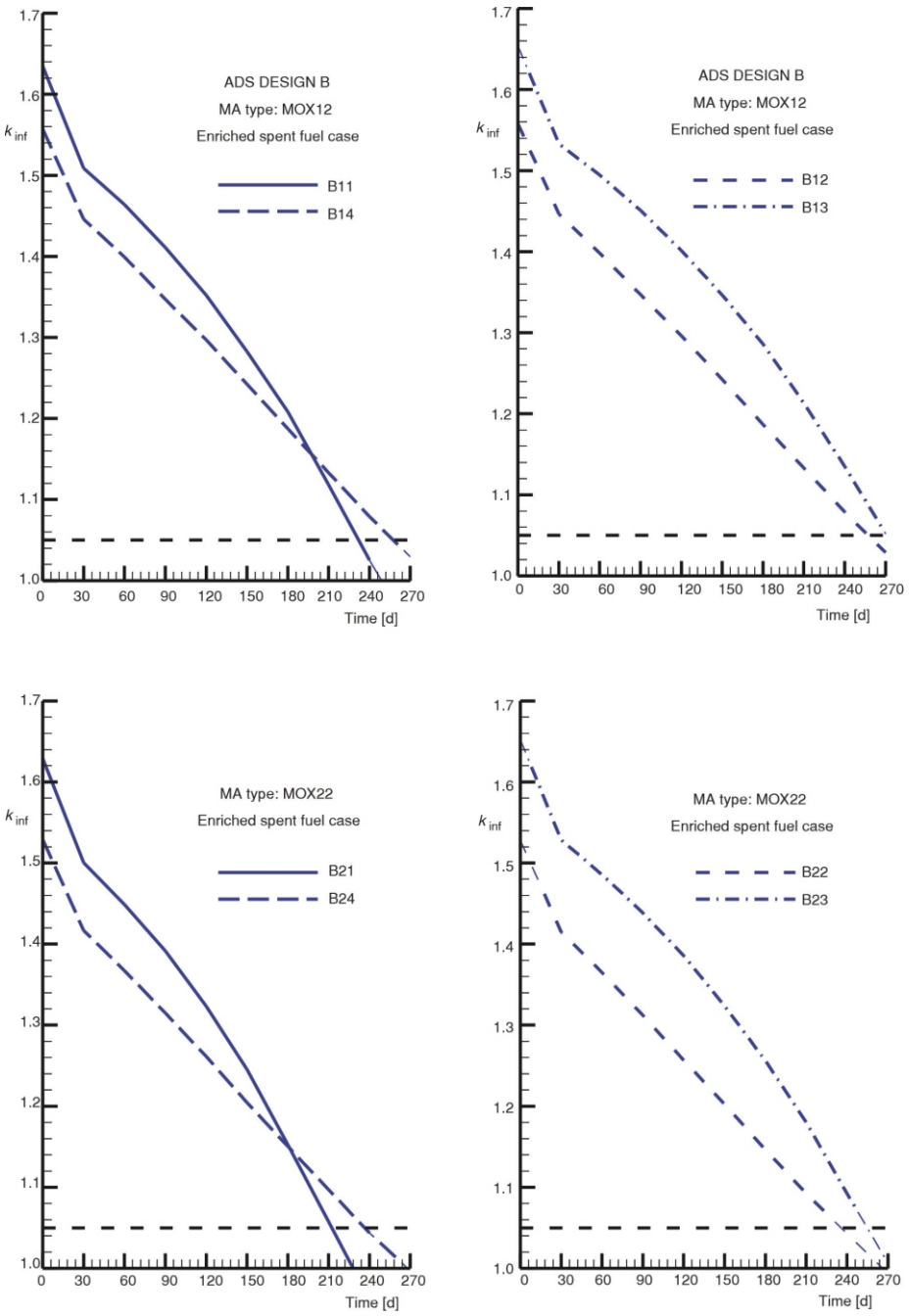
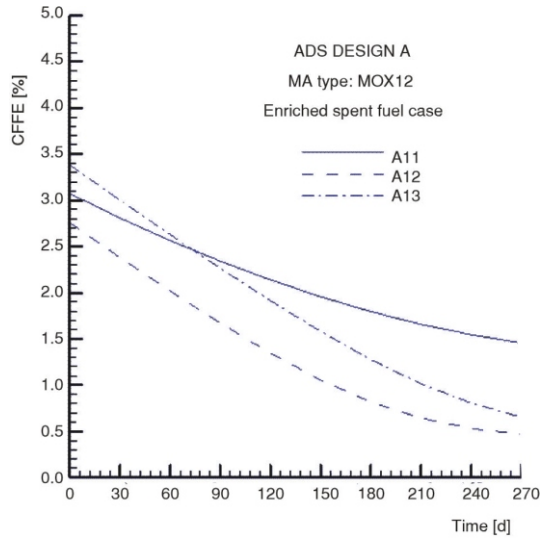


Figure 2(b). During operation time, decreases in effective neutron multiplication factor values in all enriched fuels taken out from ADS DESIGN B

where t is the operation time and MTU is the abbreviation of metric tons of uranium.

In our previous study [22], in the ADS design A, burnup values increased to 11 and 29 GWd/MTU according to the fuel cases. These increases are between 30 and 37 GWd/MTU in the ADS design B, according to the fuel cases, (see tab. 4 in [22]).

Figures 6(a) and 6(b) depict the accumulations of burnup values in all enriched fuel cases taken out from ADS DESIGN A and B, respectively, during the operation time. As apparent from eq. (3) and these figs., the burnup value increases by adding on during the operation time. One can see from figs. 6(a) and 6(b) that the burnup profiles of cases of A12 and A13 and A22 and A23 are separately very close to each other. Similar closeness also exists separately between the profiles of cases of B11 and B21, B12 and B22, B13 and B23, and B14 and B24. In addition to figs. 6(a) and 6(b), the burnup values at the EOC are given in tab. 4 for all enriched fuel cases. As apparent from this tables generally, at the EOC, the burnup values of enriched fuels taken out from ADS DESIGN B, (between 26.770 and 33.540 GWd/MTU) are a higher mean of 30 % than those from ADS DESIGN A, (between 18.060 and 24.270 GWd/MTU). In the $^{Nat}UO_2$ case, this value increases to 7.674 GWd/MTU) at EOC



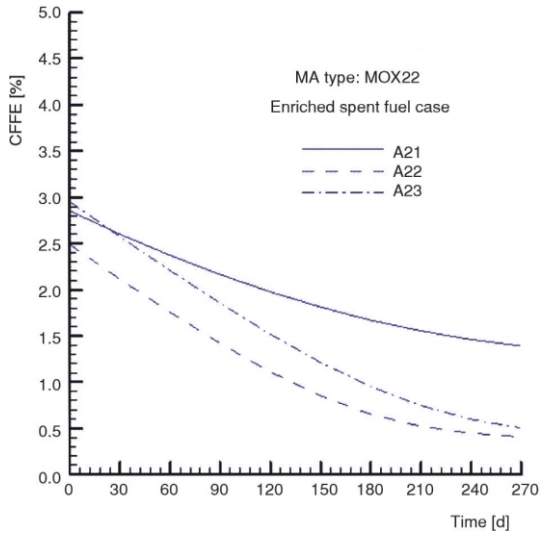
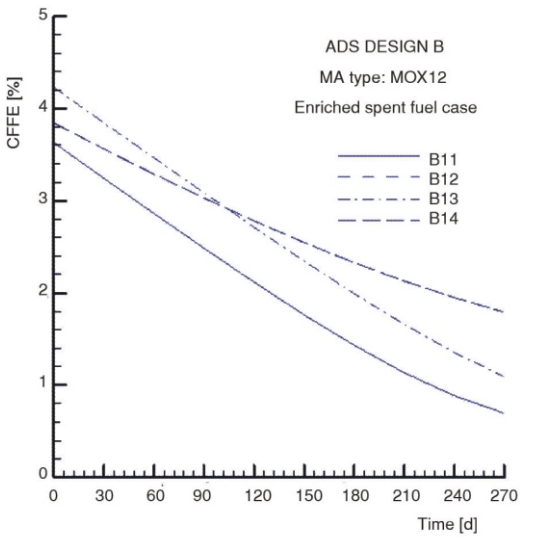


Figure 3(a). During operation time, decreases in CFFE values of all enriched fuels taken out from ADS DESIGN A



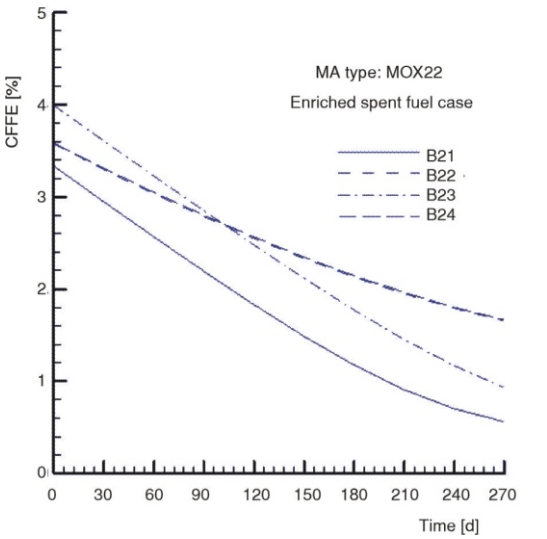


Figure 3(b). During operation time, decreases in CFFE values of all enriched fuels taken out from ADS DESIGN B

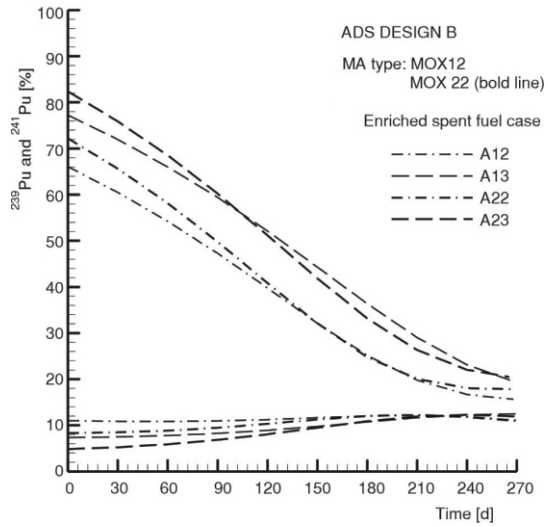


Figure 4(a). During operation time, decreases of ^{239}Pu and ^{241}Pu fractions in plutonium in all enriched fuel cases taken out from ADS DESIGN A

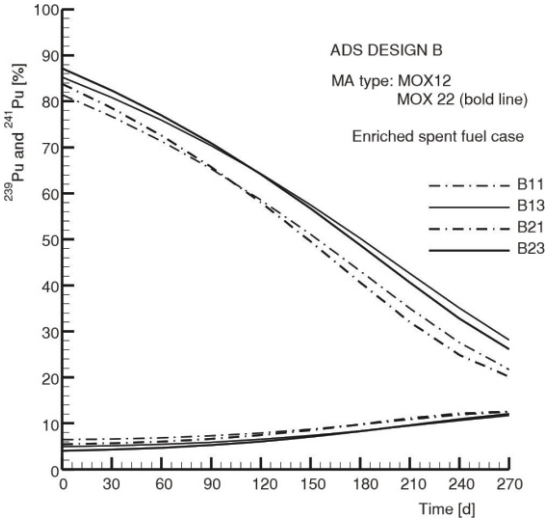


Figure 4(b). During operation time, decreases of ^{239}Pu and ^{241}Pu fractions in plutonium in all enriched fuel cases taken out from ADS DESIGN B

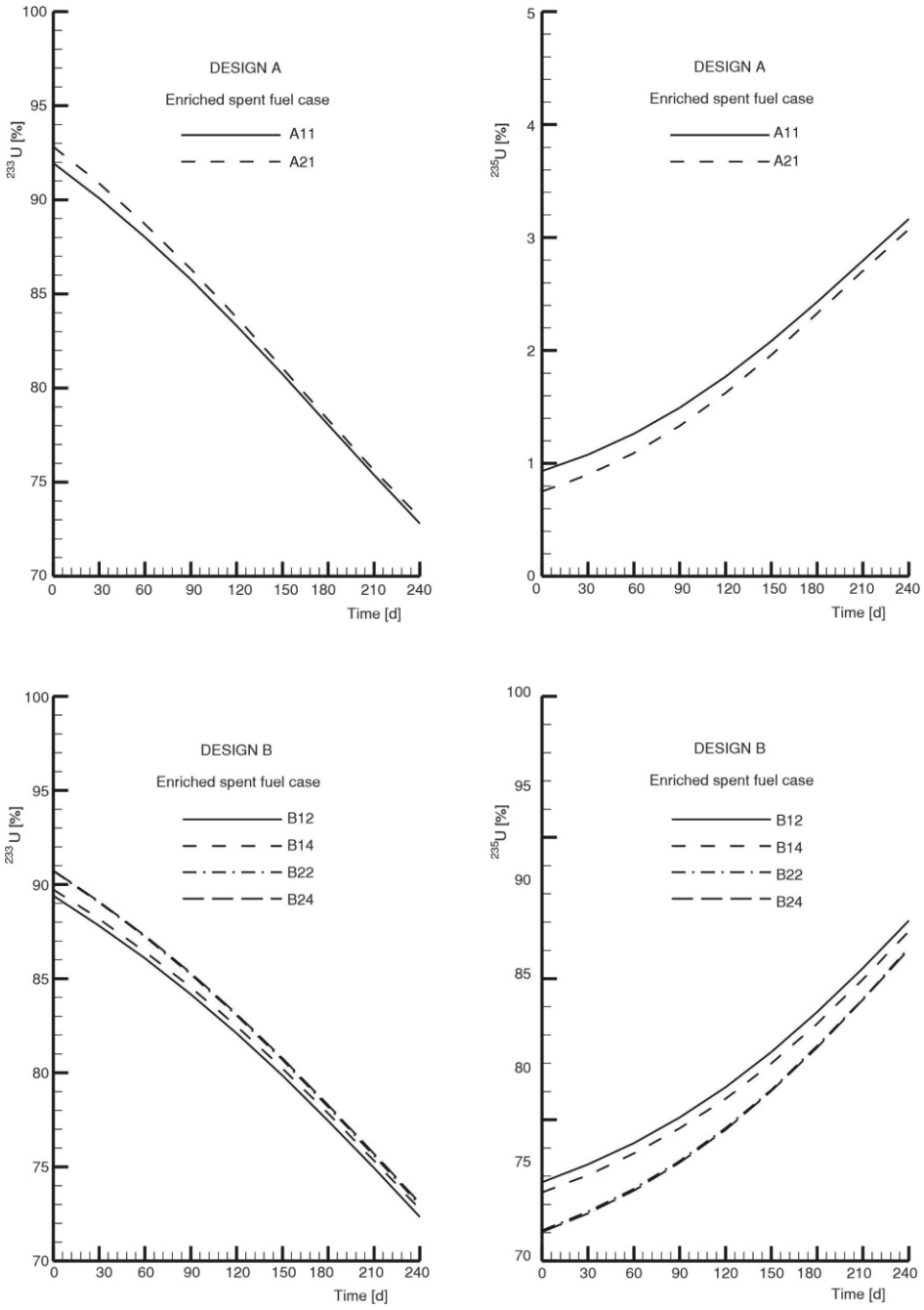


Figure 5. During operation time, decreases of ^{233}U and ^{235}U fractions in uranium in all enriched fuel cases taken out from ADS DESIGN A and B

(180 days) 180 days. Sahin *et al*. [6] report that the burnup value increases to about 7.700 GWd/MTU in the CANDU reactor fuelled with the $^{Nat}UO_2$ at the end of a burn time of 170 days and is in good agreement with the burnup value in [24]. This means that the burnup values of enriched fuels taken out from ADS DESIGN B are the mean 3.5-4.5 times the burnup value of the CANDU-37 reactor fed with the $^{Nat}UO_2$ fuel. These results bring out that in terms of energy generation, the CANDU 37 reactor was fuelled with the enriched spent fuels taken from [22] has a higher performance than the $^{Nat}UO_2$ fuel.

CONCLUSIONS AND RECOMMENDATIONS

In this study, the neutronic calculations are made by considering an assembly instead of the whole CANDU 37 reactor and only the neutronic performances of the enriched fuels in ADS DESIGN A and B are investigated.

Conclusions: To analyse the reusability performance of the enriched ThO_2, PWR, and CANDU spent fuels (14 different enriched fuel cases) taken out from two different ADS reactors fuelled with minor actinide, the CANDU-37 reactor is considered and the obtained results are briefly presented as follows:

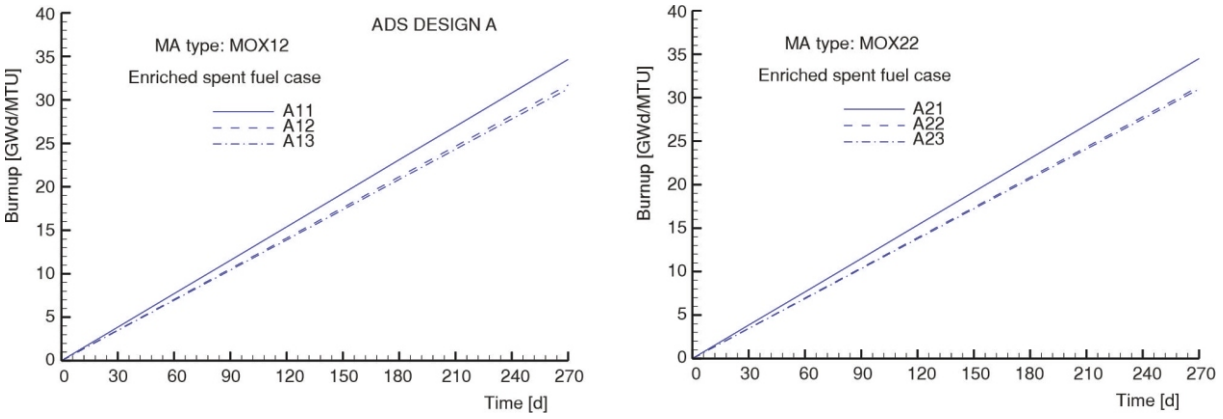


Figure 6(a). During operation time, accumulation of burnup values in all enriched fuel cases taken out from ADS DESIGN A

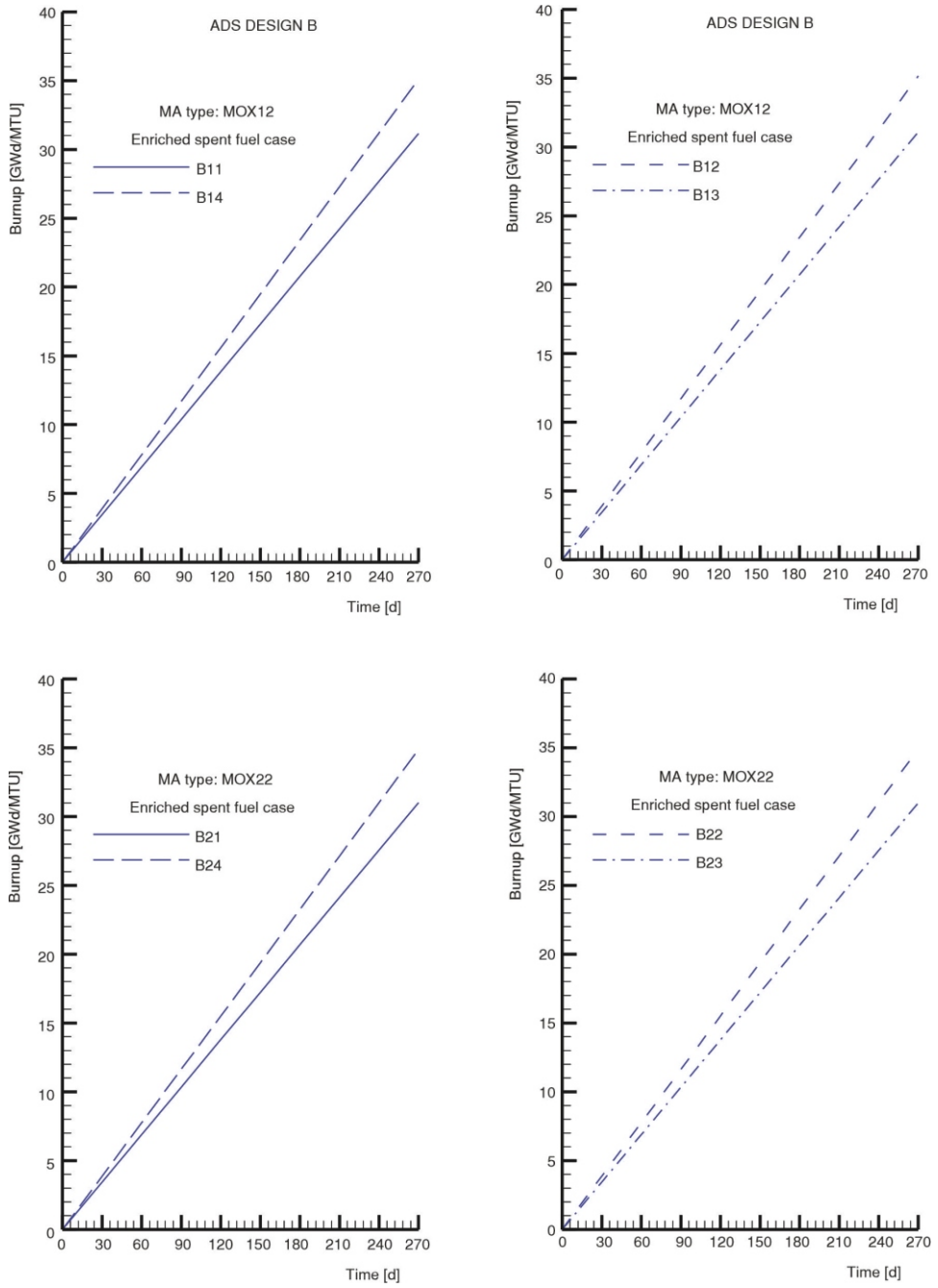


Figure 6(b). During operation time, accumulation of burnup values in all enriched fuel cases taken out from ADS DESIGN B

- The CANDU-37 reactor fuelled with the enriched fuels in the ADS DESIGN B operates for a longer time than that fuelled with the enriched fuels in the ADS DESIGN A and also the $^{Nat}UO_2$ fuel.
- At the end of effective burn times, the CFFE decreases to values varying from 0.62 % to 1.85 % according to the enriched fuel cases. In the $^{Nat}UO_2$ fuel case, this decrease is from 0.711 % to 0.591 %.
- The percentages of a total of ^{239}Pu and ^{241}Pu in all fuel cases are below weapons-grade plutonium (about 93 %). Although, at BOC, the percentages of a total of ^{233}U and ^{235}U (above 90 %) this percentage decreased to below 90 % in a short time (about 60 days).
- At the end of effective burn times, the burnup value reaches the values varying in the range of 26.770 and 33.540 GWd/MTU in the cases of enriched fuels taken out from ADS DESIGN B. These values are mean 30 % higher than the burnup values of enriched fuels taken out from the ADS DESIGN A and are a mean 3.5-4.5 times the burnup value of the CANDU-37 reactor fed with the $^{Nat}UO_2$ fuel.
- The results bring out that the ThO_2 and PWR and CANDU spent fuels enriched in minor actinide fuelled ADS reactors instead of $^{Nat}UO_2$ can be reused with high efficiency in the CANDU-37 reactor.

Consequently, in terms of energy production, the CANDU-37 reactor fuelled with the ThO_2 and the UO_2 extracted from PWR and CANDU spent fuels, which are enriched in ADS designs, has higher neutronic performance than the CANDU 37 reactor fuelled with the $^{Nat}UO_2$ fuel.

Recommendations: The decreasing CFFE values in the CANDU-37 reactor during the effective burn time can be increased by repeating the enrichment in ADS DESIGN and these enriched fuels can be used as fuel in the CANDU-37 reactor again. Among the 14 different enriched fuel cases, those with sufficient CFFE values can be used with the same processes in also the PWR reactor.

AUTHORS' CONTRIBUTIONS

Theoretical and numerical analyses were carried out by B. Durmaz, A. B. Arslan, G. Bakir, and H. Yapici analyzed and discussed the results. The manuscript was written and the figs. prepared by all authors.

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АНАЛИЗА ПОНОВНЕ УПОТРЕБЕ ThO_2 И ИСТРОШЕНИХ UO_2 ГОРИВА ОБОГАЋЕНИХ СИСТЕМОМ ПОКРЕТАНИМ АКЦЕЛЕРАТОРОМ У CANDU РЕАКТОРУ

Представљена је анализа поновне употребе ThO_2 и истрошених UO_2 горива обогаћених у два различита реактора покретана акцелератором са горивом од минорних актинида. Истрошена UO_2 горива извлаче се из истрошених горива PWR и CANDU реактора. За ову анализу разматран је реактор CANDU-37 укупне термичке снаге фисије од 2156 MW и анализирано је 14 различитих случајева обогаћених горива узетих из претходних процеса обогаћивања сагоревањем у овом реактору. Тродимензионални временски зависни прорачуни критичног сагоревања изводе се коришћењем MCNP 2.7 кода. Да би се одредило ефективно време сагоревања ови прорачуни спроводе се све док се вредности k_{inf} не смање приближно до прага критичности од 1.05 за све истрашене случајеве. Испоставља се да су у свим касетама обогаћеног горива проценти фисибилних изотопа ^{239}Pu и ^{233}U испод нивоа плутонијума и уранијума за оружје. На крају ефективног времена сагоревања, изгарање може достићи вредности које варирају у опсегу од 26.770 GWd до 33.540 GWd по метричкој тони уранијума, што је у просеку 3.5-4.5 пута више од вредности изгарања реактора CANDU-37 који се напаја природним UO_2 горивом. Резултати показују да у смислу производње енергије, реактор CANDU-37 са горивом ThO_2 и истрошеним UO_2 горивима обогаћеним системима покретаним акцелератором, испољавају боља неутронска својства од уобичајеног CANDU-37 реактора.

Кључне речи: CANDU реактор, термички реактор, систем покретан акцелератором, коришћење обогаћеног истрошеног горива, искоришћење торијума