THORIUM UTILIZATION IN ACR (ADVANCED CANDU REACTOR) AND CANDU-6

ACR (GELİŞMİŞ CANDU REAKTÖRÜ) VE CANDU-6 REAKTÖRLERİNDE TORYUM KULLANIMI

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ABSTRACT

It is the main objective of this study to investigate fuel composition options for CANDU type of reactors that are capable of using a mixture of U-Th as fuel. A homogenous mixture of (U-Th)O_2 was used in all elements of fuel bundles. The cores of CANDU-6 and Advanced CANDU reactors (ACR) were modeled using MCNP5. In equilibrium core, using MONTEBURNS2 code (coupled with MCNP5 and ORIGENS) for once-through uranium and once-through uranium-thorium fuel cycles of CANDU-6 and ACR, discharge burnups and spent fuel compositions were computed. For various enrichments of uranium and different fractions of thorium in a (U-Th) fuel mixture, performing burnup calculations, expressions relating burnup to fuel composition and reactivity to burnup were derived. Conversion ratios, natural U and fuel requirements, nuclear resource utilization factors, and natural uranium savings were calculated, and their changes with burnup were observed. In addition, recycling of U and Pu contents of spent CANDU-6 and spent ACR fuels into CANDU-6 was worked out; discharge burrnups to be reached in CANDU-6 for various compositions of recycle fuels were determined. Appropriate fuel compositions were discussed.

Keywords: Thorium, ACR, CANDU-6, Burnup, Nuclear Resource Utilization, Natural Uranium Savings, Recycling, Conversion Ratio, Spent Fuel

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Mehmet TÜRKMEN

ÖZ

Çalışmanın ana konusu, toryum katkılı uranyum kullanabilen CANDU tipi nükleer reaktörler için yakıt kompozisyonu seçeneklerini araştırmak ve karşılaştırmaktır. Uranyum-toryum yakıtı olarak, yakıt demetinin tüm elemanlarında homojen olarak dağılmış (U-Th)O_2 kullanıldı. ACR ve CANDU-6 reaktörlerinin kor tasarımı MCNP5 kodu kullanılarak yapıldı. Dengedeki sistemde ACR ve CANDU-6 nın tek-geçişli uranyum yakıt çevrimleri ve tek-geçişli uranyum-toryum yakıt çevrimleri için yakıt yanma oranları ve kullanılmış yakıt içerikleri, MONTEBURNS2 (MCNP5 ve OrigenS ile birlikte) kodu kullanılarak hesaplandı. Çeşitli uranyum zenginlikleri ve farklı toryum oranları içeren uranyum-toryum yakıt karışımları için, yanma oranı hesapları yapılarak yanma oranını yakıt kompozisyonuna ve reaktiviteyi yanma oranına bağlayan eşitlikler türetildi; dönüştürme oranı, doğal uranyum ve yakıt gereksinimi, doğal kaynaklardan yararlanma faktörü ve doğal uranyum kazancı hesaplandı, ve yanma oranı ile nasıl değiştikleri gözlendi. Ayrıca, kullanılmış CANDU-6 ve ACR yakıtlarının U and Pu içeriği geri kazanılarak, CANDU-6'da yeniden kullanıldı. Yakıtların CANDU-6 da ulaşılabilecekleri yanma oranları, farklı kompozisyonlar için belirlendi. Uygun yakıt kompozisyonları ve yakıt çevrim seçenekleri elde edilen sonuçlar doğrultusunda irdelendi.

Anahtar Kelimeler: Toryum, ACR, CANDU-6, Yanma Oranı, Nükleer Kaynaklardan Yararlanma, Doğal Uranyum Kazancı, Dönüştürme Oranı, Geri Dönüşüm, Kullanılmış Yakıt

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SYMBOLS AND ABBREVIATIONS

B_d	Discharge Burnup	[MWd/kgU(HM)]
B	Specific Burnup	[MWd/kgU(HM)]
ε	U-235 Enrichment or Fraction	[%] weight percent
θ	Th-232 Fraction	[%] weight percent
ρ	Reactivity	[-]
k	Multiplication Factor	[-]
α	Capture to Fission Ratio	[-]
ω	Recycled Fuel Thorium Fraction	[-] weight percent

AECL	Atomic Energy of Canada Limited
ACR	Advanced CANDU Reactor
BOC	Beginning of Cycle
CANDU	CANadian Deuterium/Uranium reactor
CANFLEX	CANDU FLEXible Fueling
DU	Depleted Uranium
EOC	End of Cycle
FP	Fission Product
HM	Heavy Metal
MCNP	Monte Carlo N-Particle Transport Code
MOX	Mixed Oxide Fuel
NU	Natural Uranium
PWR	Pressurized Water Reactor
SEU	Slightly Enriched Uranium
SF	Spent Fuel

CHAPTER 1. INTRODUCTION

1.1 THORIUM AND ITS UTILIZATION IN NUCLEAR REACTORS

Thorium occurs in several minerals, the most common source being the rare earth-thorium-phosphate mineral, monazite, which contains 6-7% thorium oxide (ThO_2) on the average. Naturally occurring Th is mainly Th-232, with a half-life of 14.05 billion years.

According to the reactions given below, Th-232 can be transformed into U-233, which does not exist in nature. Table 1. 1 shows some important characteristics of U-233 together with the two most important fissile nuclei, U-235 and Pu-239. As seen from the table, U-233 is a valuable fissile isotope; therefore, natural Th (Th-232) is a valuable fertile element.

$$^{232}_{90}\mathrm{Th} + {}^{1}_{0}\mathrm{n} \xrightarrow[7.40\mathrm{b}]{(n,\gamma)} {}^{233}_{90}\mathrm{Th} \xrightarrow[(t_{1/2}:22.2\mathrm{min})]{\beta^-} {}^{233}_{91}\mathrm{Pa} \xrightarrow[(t_{1/2}:27.0\mathrm{days})]{\beta^-} {}^{233}_{92}\mathrm{U}$$

Table 1. 1 Some Important Characteristics of Three Basic Fissile Nuclei (2200 m/s cross sections)

Nucleide	Fission Cross Section (σ_f)	Absorption Cross Section (σ_a)	Neutron Yield ν	Fission Factor η
U-233	531.1	578.8	2.492	2.287
U-235	582.2	680.8	2.418	2.068
Pu-239	742.5	1011.3	2.871	2.108

A nuclear reactor cannot be made critical with Th alone, just as it cannot be made critical with U-238 alone. Th-232 can replace U-238 in current nuclear reactors, and subsequently the fissile U-233 produces from Th-232 partly replaces the fissile Pu-239 produced from U-238. Such a replacement will result in the long-term extension of nuclear fuel resources.

The thermal conductivity of ThO_2 is about 50% higher than that of UO_2 over a large temperature range, and its melting temperature is 340ºC higher than that of UO_2. As a

consequence, all thermally activated processes, such as creep and fission gas diffusion will be reduced. Fission-gas release from ThO_2 fabricated with proper control of microstructure will be lower than that from UO_2 operating under similar ratings. ThO_2 is chemically very stable and does not oxidize easily, which offers advantages for normal operation, postulated accidents and in waste management. Reactor-grade ThO_2 can be blended with UO_2 in required fractions to produce (U-Th)O_2 fuels, which can be irradiated in current/future nuclear reactors.

While the full exploitation of the energy potential of Th requires recycling, which may not be economically justified for many years, the attraction of using Th in CANDU reactors is that benefit can be derived from this fuel today, in existing reactors, at fuel-cycle costs that are comparable with the already low cost of NU fuelling, and with improved U utilization compared to NU fuel. A strategic mine of U-233 can be produced that is safeguarded in the SF, and available for future recovery and recycle when predicated by economic, technical, and strategic considerations. This possibility will be of particular interest in those countries having abundant Th reserves, but lacking U. The bridge between the Th recycle options of the future and current U fuel cycles is the once-through (U-Th) fuel cycle in CANDU reactors, which is to be discussed in Section 1.3.

For burning (U-Th) fuel in CANDU type of reactors, two general approaches have been devised. The first is a “mixed-core” approach, which has been investigated by Boczar ([1], [2]), in which a large number of channels fuelled with “driver” fuel would provide the external source of neutrons for a fewer number of channels fuelled with ThO_2. Theoretically, enrichment levels, burnups, and relative feed rates can be chosen to make this fuel cycle competitive (both in terms of resource utilization and in economics) not only with NU, but also with SEU fuel. On-power refueling enables the ThO_2 fuel to remain in the core much longer than the driver fuel. With the large disparity between the properties of the “driver” fuel and the ThO_2 channels, fuel management would be particularly challenging.

A “mixed-fuel bundle” approach, which has been studied by Milgram [3] and Gupta [4], is an alternative strategy that has recently been devised by AECL, which provides a practical means of utilizing Th in operating CANDU reactors. Although the uranium utilization is not quite as good as in the “mixed-core” approach, this strategy has many benefits: uranium resource utilization is better than that with NU fuel, and fuel-cycle costs are comparable; fuel management is particularly simple; refueling rates (in bundles per day) are a third of those with NU; excellent axial power distributions are obtained, with or without adjuster rods; maximum bundle and channel powers are lower than that with NU fuel; and void reactivity is reduced. The “mixed-fuel bundle” contains ThO_2 in the central 8 elements of a CANFLEX (CANDU FLEXible Fueling) bundle, and SEU in the outer 2 rings of elements. The disadvantage compared to the “mixed-core” approach is that separate dwell times cannot be achieved for the ThO_2 and the driver fuel because they are part of the same bundle. However, even with a modest bundle-averaged burnup of about 22 MWd/kg HM (Heavy Metal), the ThO_2 elements, upon experiencing sufficient irradiation, considerably contribute to the overall U utilization. [5]

1.2 NUCLEAR REACTORS UNDER CONSIDERATION

1.1.1 CANDU-6

The CANDU-6 is a 600-MWe Generation III nuclear power plant with a 40-year design life at a lifetime plant capacity factor of 85%. It is a heavy-water-cooled and heavy-water-moderated pressure-tube reactor. The current CANDU-6 design uses NU as fuel. The CANDU-6 fuel bundle, shown in Figure 1. 1, has 37 fuel elements. Each element carries NU in the form of cylindrical pellets of UO_2 enclosed with a zircaloy-4 sheath. The fuel channel includes a zirconium-niobium alloy pressure tube, a zirconium calandria tube, stainless steel end-fittings at each end and four spacers.

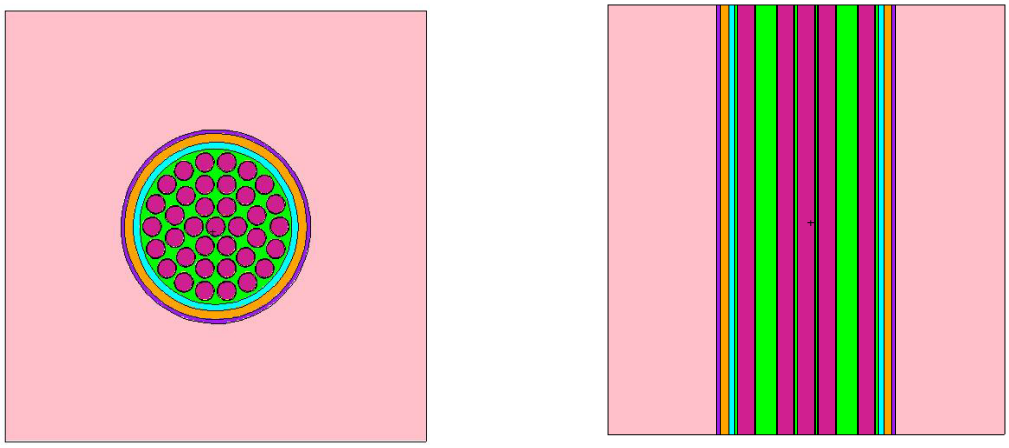


Figure 1. 1 Front and Side View of CANFLEX-CANDU Fuel Bundle

The CANDU fuel bundle is relatively small (0.5 m in length, 10 cm in diameter), and easy to handle (about 20 kg). It consists of only 7 distinct components (pellets, sheath, CANLUB coating inside the sheath, spacer pads, bearing pads, end-plugs and end-plates). The use of NU fuel itself simplifies manufacturing and handling. After 350 reactor-years of operation, the failure rate of CANDU fuel is very low - less than 0.1% bundle failure rate [6]. The ability to detect the fuel defects and to remove the failed fuel during normal on-power refueling operations minimizes coolant system contamination and the economic effect of fuel defects.

After the introduction of CANFLEX bundles containing NU fuel, the next step in the evolution of CANDU fuel cycles would be the introduction of SEU fuel, using the CANFLEX bundle. The 20% lower linear element ratings in CANFLEX reduce the peak operating temperatures and hence, fission-gas release, facilitating the achievement of higher burnups. Moreover, the increased thermal-hydraulic margins obtained with CANFLEX fuel provide a significant performance enhancement in addition to the other benefits of enrichment.

The optimal enrichment that minimizes the fuel cycle cost in CANDU is estimated to be between 0.9% and 1.2 [7]. Enrichments around 0.9% are below the threshold at which criticality considerations result in restrictions and complications in fuel fabrication and fuel handling. Moreover, with this level of enrichment, fuel management is extremely simple: a regular 2- or 4-bundle shift, bi-directional fuelling

scheme results in excellent axial power distributions, with or without the presence of the adjuster rods. At these enrichments, the transition from a NU-fuelled core to an SEU-fuelled core can be achieved in a straightforward fashion, by simply replacing NU fuel with SEU during the normal course of refueling. Operational considerations are easily met with enrichment at this level, with no changes to the reactor. SEU also offers greater flexibility in fuel-bundle design, providing, for example, a means of tailoring reactivity coefficients. [8]

The use of recovered U from spent PWR fuel offers access to a potentially very economical supply of enrichment at the optimal enrichment level. A detailed assessment of the use of recovered U in CANDU reactors is currently underway as part of a collaborative program among AECL (Atomic Energy of Canada Limited), BNFL (British Nuclear Fuels Limited) and KAERI (Korea Atomic Energy Research Institute). [8]

1.1.2 Advanced CANDU Reactor (ACR)

The ACR-700 is a 1000-MWe Generation III+ nuclear power plant with a 60-year design life at a lifetime plant capacity factor of 90%. It is a light-water-cooled, heavy-water-moderated pressure-tube reactor. The current ACR-700 design uses SEU (UO_2) as fuel. The ACR-700 reactor consist of 284 fuel channels and has a total thermal power output 1982 MW(th). The fuel channels are prepared in a compact array with a square lattice pitch of 24 cm. Each fuel channel involves twelve fuel bundles.

Figure 1. 2 shows a CANFLEX-ACR fuel bundle, which contains 43-fuel elements, each 49.53 cm in length, with uniform enrichment. The bundle includes 2 different element sizes. The center and inner ring consist of eight elements with a diameter of 13.5 mm, whereas the outer two rings consist of 35 elements with a smaller diameter of 11.5 mm. The outer three rings of fuel elements contain enriched U pellets with 2.1 % U-235, while the central fuel element contains burnable neutron poison, dysprosium (Dy), in the form of $(U\text{-}Dy)O_2$ pellets with 7.5% Dy in NU. [9]

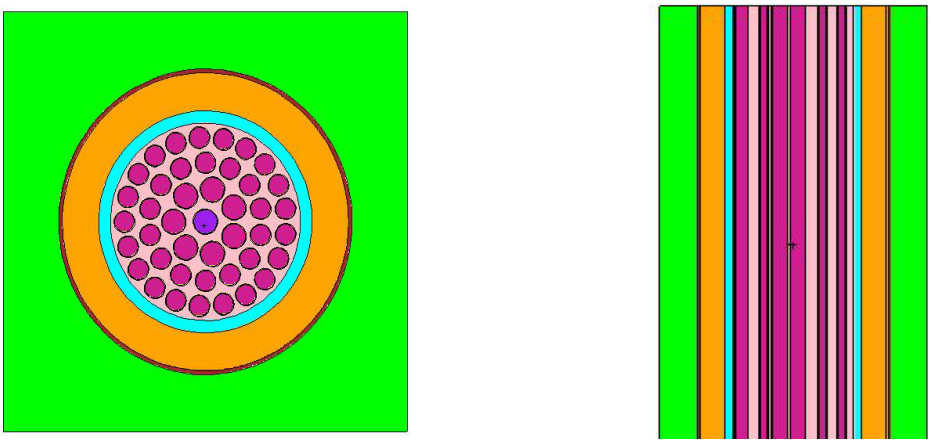


Figure 1. 2 Front and Side View of CANFLEX-ACR Fuel Bundle

Use of light water as coolant reduces the heavy water inventory and results in lower costs and emission reduction. The design of the CANFLEX-ACR fuel bundle allows to burn different fuel types efficiently, to help achieve negative void reactivity, and to accumulate higher burnup.[10]

In order to maintain sufficient positive reactivity, fuel is replaced while the reactor is on power. This feature contributes to high availability factors and improved outage flexibility since refueling outages at fixed cycle times are not required.

The ACR-700 reactor uses zirconium alloys for the core structures, including horizontal fuel channels and fuel cladding that contains the fuel. The fuel sheaths, endcaps, endplates and appendages are made of Zircaloy-4 because of its excellent nuclear characteristics of low neutron absorption, good corrosion resistance and low hydrogen/deuterium pickup. [11]

Improved passive safety systems, improved plant thermal efficiency through the use of higher pressures and temperatures in the coolant and the steam supply systems, enhanced accident resistance and core damage prevention features are important characteristics of ACR, which lead to the more compact core design with improved stability and higher output.

These technical improvements, along with system simplifications and advancements in project engineering, manufacturing, and construction, result in a reduced capital cost and construction schedule, while enhancing the inherent safety and operating performance of the ACR design. [10]

ACR-700 can be adapted more easily to different fuel cycle approaches [e.g., use of mixed oxide fuel (MOX) or Th added fuels] without changing the basic reactor core design.

1.3 SUMMARY OF RELATED FUEL CYCLES

The nuclear fuel cycle consists of all the processes starting with the mining of U and ending with the disposal of high-level waste and/or SF. When SF is directly sent to final disposal (i.e., no reprocessing to recover the valuables in it), the nuclear fuel cycle is named “once-through”. A simplified block diagram of a once-through fuel cycle that uses (U-Th) fuel is presented in Figure 1. 3.

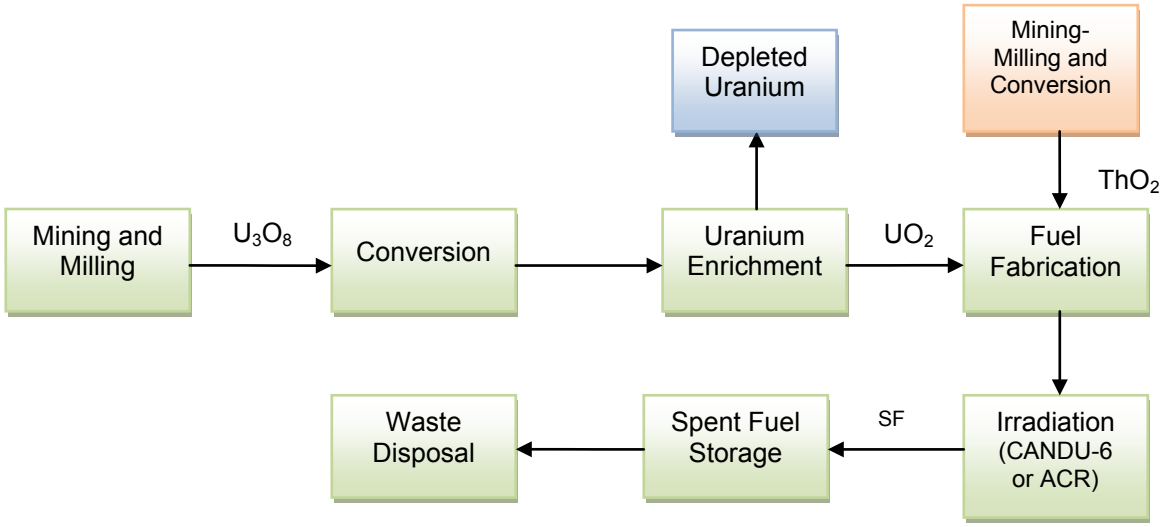


Figure 1. 3 Once-Through (U-Th) Fuel Cycle

In case of a “closed fuel cycle”, following irradiation, SF is sent to a reprocessing plant where all U and Pu contained in SF are recovered and recycled. Recovered U and Pu are blended with an appropriate fraction of fresh Th and sent to fuel fabrication. A closed (U-Th) fuel cycle is shown Figure 1. 4.

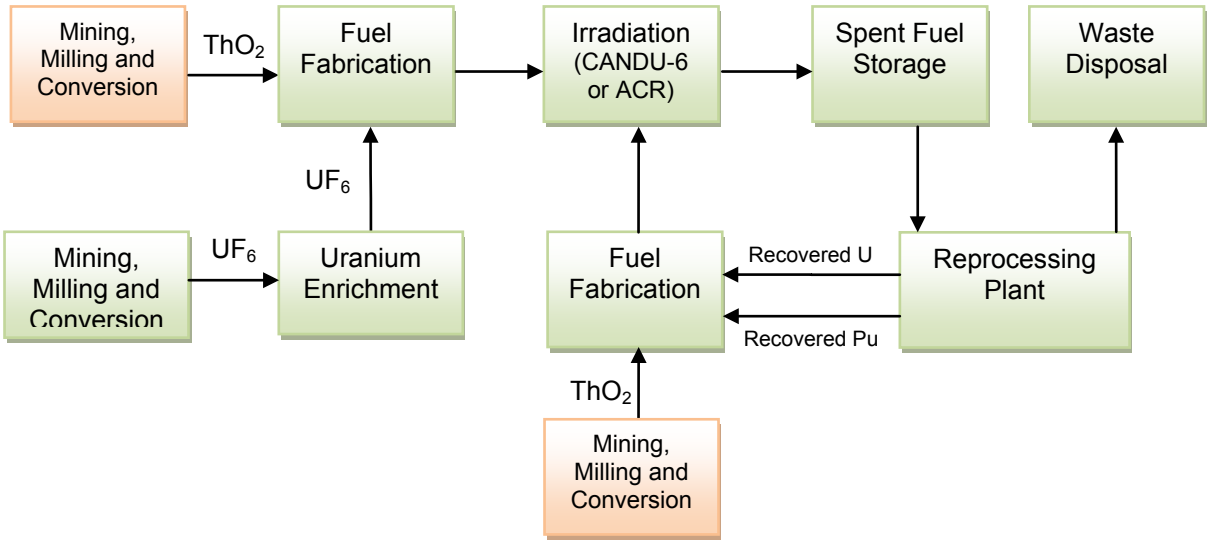


Figure 1. 4 Closed (U-Th) Fuel Cycle

CHAPTER 2. COMPUTATIONAL METHODS, PROGRAMMING AND MODELING

In burnup analysis, a “bundle-based” approach was adopted instead of full core modeling. Calculations were carried out for a single fuel bundle, and then adjusted to full core accordingly. In case of (U-Th) fuels, a homogenous mixture of (U-Th)O_2 was used in all elements of fuel bundles.

2.1 CODES

2.1.1 SCALE-5.1

2.1.1.1 OrigenArp/OrigenS

ORIGEN-ARP [12] is a SCALE depletion analysis sequence used to perform point-depletion calculations with the ORIGEN-S code [13] using problem-dependent cross sections. Problem-dependent cross section libraries are generated using the ARP (**A**utomatic **R**apid **P**rocessing) module using an interpolation algorithm that operates on pre-generated libraries created for a range of fuel properties and operating conditions. Methods are provided in SCALE to generate these libraries using one-, two-, and three-dimensional transport codes. The interpolation of cross sections for U fuels may be performed for the variables burnup, enrichment, and water density. An option is also available to interpolate cross sections for mixed-oxide (MOX) fuels using the variables burnup, plutonium content, plutonium isotopic vector, and water moderator density. Interpolation may also be performed for flux-based calculations using neutron fluence as the interpolation parameter.

ORIGEN-S computes time-dependent concentrations and source terms of a large number of isotopes, which are simultaneously generated or depleted through neutronic transmutation, fission, radioactive decay, input feed rates, and physical or chemical removal rates. The calculations may pertain to fuel irradiation within nuclear reactors, or the storage, management, transportation, or subsequent chemical processing of removed fuel elements.

The primary objective of ORIGEN-S is to make it possible for the calculations to utilize the multi-energy-group cross sections from any currently processed standardized ENDF/B data base. These codes compute flux-weighted cross sections; simulating conditions within any given reactor fuel assembly, and convert the data into a library that can be input to ORIGEN-S. Time-dependent libraries may be produced, reflecting fuel composition variations during irradiation.

2.1.1.2 TRITON/ Cross Section Processing (CSAS)

The TRITON [14] control module was originally developed in tandem with the NEWT functional module of SCALE to support two-dimensional (2-D) transport and depletion calculations.

TRITON can be used to provide automated, problem-dependent cross-section processing followed by calculation of the neutron multiplication factor for a 2-D configuration using NEWT. Additionally, this functionality can be iterated in tandem with ORIGEN-S depletion calculations to predict isotopic concentrations, source terms, and decay heat as a result of time-varying fluxes calculated in a 2-D deterministic fashion or in a 3-D stochastic approach.

The depletion capabilities within TRITON were developed to overcome some of the modeling shortcomings of the SAS2 depletion sequence. SAS2 has been shown to perform amazingly well in the isotopic predictions for fuel burnup, in spite of a simple one-dimensional transport model. However, increasingly complex fuel assembly designs, along with other non-lattice configurations, are beyond the capabilities of SAS2. TRITON has been developed in part to provide improved rigor and transport modeling accuracy to predict the burnup of nuclear materials in configurations that have a strong spatial dependence on the neutron flux and other physics parameters characterizing the system. Furthermore, TRITON provides the ability to independently deplete multiple materials within a given system.

2.1.2 MCNP-5

MCNP5 ([15], [16]) is a general-purpose Monte Carlo N–Particle code that can be used for neutron, photon, electron, or coupled neutron/photon/electron transport, including the capability to calculate Eigen-values for critical systems. The code treats an arbitrary three-dimensional configuration of materials in geometric cells bounded by first- and second-degree surfaces and fourth-degree elliptical tori.

Pointwise cross-section data are used. For neutrons, all reactions given in a particular cross-section evaluation (such as ENDF/B-VI) are accounted for. Thermal neutrons are described by both the free gas and S(α,β) models. For photons, the code accounts for incoherent and coherent scattering, the possibility of fluorescent emission after photoelectric absorption, absorption in pair production with local emission of annihilation radiation, and bremsstrahlung. A continuous-slowing-down model is used for electron transport that includes positrons, k x-rays, and bremsstrahlung, but does not include external or self-induced fields.

Important standard features that make MCNP very versatile and easy to use include a powerful general source, criticality source, and surface source; both geometry and output tally plotters; a rich collection of variance reduction techniques; a flexible tally structure; and an extensive collection of cross-section data.

2.1.3 MONTEBURNS-2

MONTEBURNS2 [17] is a fully automated tool that links the Monte Carlo transport code MCNP with the radioactive decay and burnup code ORIGEN2. MONTEBURNS produces a large number of criticality and burnup results based on various material feed/removal specifications, power(s), and time intervals. The program processes input from the user that specifies the system geometry, initial material compositions, feed/removal specifications, and other code-specific parameters. Various results from MCNP, ORIGEN2, and other calculations are then output successively as the code runs. The principle function of MONTEBURNS is to transfer one-group cross-section and flux values from MCNP to ORIGEN2, and then transfer the resulting material compositions (after irradiation and/or decay) from ORIGEN2 back to MCNP in a

repeated, cyclic fashion. The basic requirement of the code is that the users have a working MCNP input file and other input parameters; all interaction with ORIGEN2 and other calculations are performed by MONTEBURNS.

2.2 MCNP/MONTEBURNS MODELING

MCNP-5 with a set of ENDF/B-VI continuous neutron cross-section libraries has been used to model the fuel bundles. Since libraries are evaluted at different temperatures, appropriate libraries are chosen for different materials of bundle. In addition, neutron cross-section data are adjusted by TMP card and the slow neutron scattering law S(α,β), evaluated at 600 K that is used to account for the molecular binding effects in water.

Number of automatic tally materials is obtained for 72 isotopes. Also, Origen2 library are chosen for NU and SEU. Fractional power are set to 1 assuming constant power at each cycle. Burn days are entered into Origen feed input by considering the effect of poisoner fission products at startup. A sufficient number of cycles and internal-outer burn steps are chosen to minimize variance in multiplication factor, hence in burnup.

Since the cross-section libraries of OrigenS do not cover the burnup ranges to be investigated, it is necessary to determine burnup-dependent multi-group microscopic/macroscopic neutron cross-sections. Libraries are created for all isotopes in TRITON for 0, 2, 4, 6, 8… 100 MWd/kgHM burnup steps. It would also be possible to generate libraries for even shorter burnup intervals but that increases the computational time.

The benchmark specifications used in MCNP5 modeling of CANDU-6 and ACR are listed in Appendix A.

2.3 COMPUTATIONAL FLOW DIAGRAM

First, the cores of ACR and CANDU-6 are modeled using MCNP5. Subsequently, the models are arranged for MONTEBURNS2 in order to investigate neutronic and

material properties (such as burnup, reactivity, spent fuel content, etc...). MONTEBURNS2 joins OrigenS [fuel-depletion code] and MCNP5 [neutron transport code]. MCNP5 provides neutron cross-section sets to MONTEBURNS2. OrigenS calculates fuel compositions with regard to exposure time of fuel. Then, leakage reactivities are determined for the reference reactors using MONTEBURNS2. The reactivity model used to compute the reference leakage reactivities is given in Section 2.4. The leakage reactivities computed are used in the course of discharge burnup calculations. Calculation of SF composition is performed using SCALE5.1/OrigenS-Arp. Although MONTEBURNS2 and SCALE5.1 give the same results, the results from SCALE5.1 are used because it is more sensitive. Finally, burnup dependent cross-section sets are generated by TRITON package of SCALE5.1. A computational flow chart is presented in Figure 2. 1.

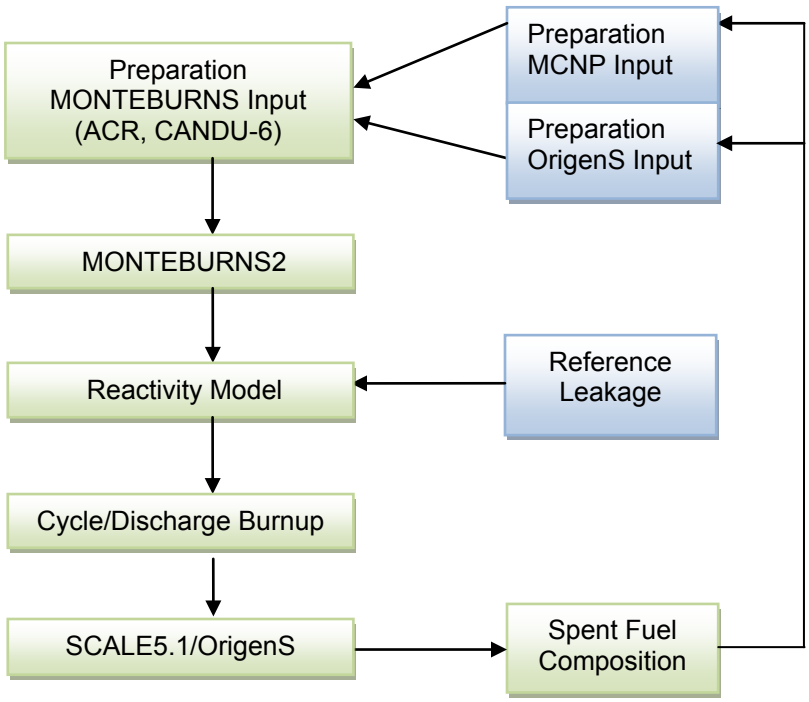


Figure 2. 1 Computational Flow Diagram

Calculations performed are, in turn:

a. Leakage reactivity
b. Discharge burnup [UO_2 and (U-Th)O_2 fuels]
c. Reactivity [UO_2 and (U-Th)O_2 fuels]
d. SF composition [UO_2 and (U-Th)O_2 SFs]
e. Use of CANDU-6 SFs in CANDU-6 [UO_2 and (U-Th)O_2 fuels]
f. Use of ACR SFs in CANDU-6 [UO_2 and (U-Th)O_2 fuels]
g. Core power sharing [UO_2 fuels]
h. Conversion factor [UO_2 and (U-Th)O_2 fuels]
i. NU and fuel requirement [UO_2 and (U-Th)O_2 fuels]
j. Nuclear resource utilization [UO_2 and (U-Th)O_2 fuels]
k. NU Savings [UO_2 and (U-Th)O_2 fuels]

2.4 CALCULATION OF LEAKAGE REACTIVITY

The reactivity model used is given by (Eq. 2.1).

$$\rho(B) = a_0 + a_1 B + a_2 B^2 + a_3 B^3 + ... \qquad \textbf{[Eq. 2.1]}$$

$\rho_c - \rho_{leak} = \rho_s$ where ρ_c is reactivity of core and ρ_s is reactivity of system.

For equillibrium condition; $\rho_s = 0$

$$\rho_c - \rho_{leak} = 0 \Rightarrow \rho_{leak} = \frac{1}{B_d}\int_0^{B_d} \rho_c(B)dB \qquad \textbf{[Eq. 2.2]}$$

Finally, the burnup dependent reactivity is calculated by (Eq.2.3).

$$\rho(B_d) = \rho_o - A_1 B_d - A_2 B_d{}^2 - .... \qquad \textbf{[Eq. 2.3]}$$

Leakage reactivities are calculated on the basis of the reference burnup values.

Reference leakage reactivity of ACR is at 20.500MWd/kgU;

$\rho_{leak} = 0.094467 = \%9.4467$

Reference leakage reactivity of CANDU-6 is at 7.200 MWd/kgU;

$$\rho_{leak} = 0.0479 = \%4.79$$

The reference leakage reactivity computed by MONTEBURNS2 for CANDU-6 is compatible with literature values: between 4.5% and 5.0%, given by [18] and [19].

CHAPTER 3. BURNING U AND (U-Th) IN ACR

3.1 U IN ACR

3.1.1 Discharge Burnup as a Function of Fuel Composition

Fresh UO_2 fuels with various enrichments are burned and results are tabulated in Table 3. 1. For the central fuel pin [(NU-Dy)O2], outer rings (SEU) and the whole bundle, discharge burnups are calculated separately. Bundle-averaged discharge burnup is defined as the total energy produced from a bundle per unit mass of heavy metal.

Table 3. 1 Discharge Burnup Values

	Burnup (MWd/kgU)			
U-235 Enrichment (%)	*SEU*	*(NU-Dy)O2*	*Bundle Discharge Burnup (MWd/kgU)*	*Fuel Exposure Time (Days)*
1.7	12.300	3.600	12.200	585
1.9	16.800	5.000	16.600	795
2.1	20.800	6.000	20.500	980
2.3	24.700	7.000	24.400	1170
2.5	28.300	7.900	27.900	1345
2.9	35.200	9.500	34.700	1670
3.1	38.400	10.200	37.900	1825
3.5	44.130	10.700	43.355	2015
4	50.990	11.190	50.065	2325
4.4	57.150	12.500	56.110	2605
4.8	63.000	13.320	61.845	2870
5.2	68.690	14.115	67.420	3130

SEU achieves 20.8 MWd/kgU while NU fuel at center pin gains the burnup of 6.0 MWd/kgU with 980 irradiation days. Bundle discharge burnup is obtained as 20.5 MWd/kgU. By increasing the enrichment up to 5.2%, bundle-averaged burnup reaches 67.420 MWd/kgU.

As seen in Figure 3. 1 and Figure 3. 2, discharge burnup increases almost linearly as enrichment increases; however, for better accuracy a second order fit (Eq.3.1) is applied in all cases (bundle, center pin and outer pins).

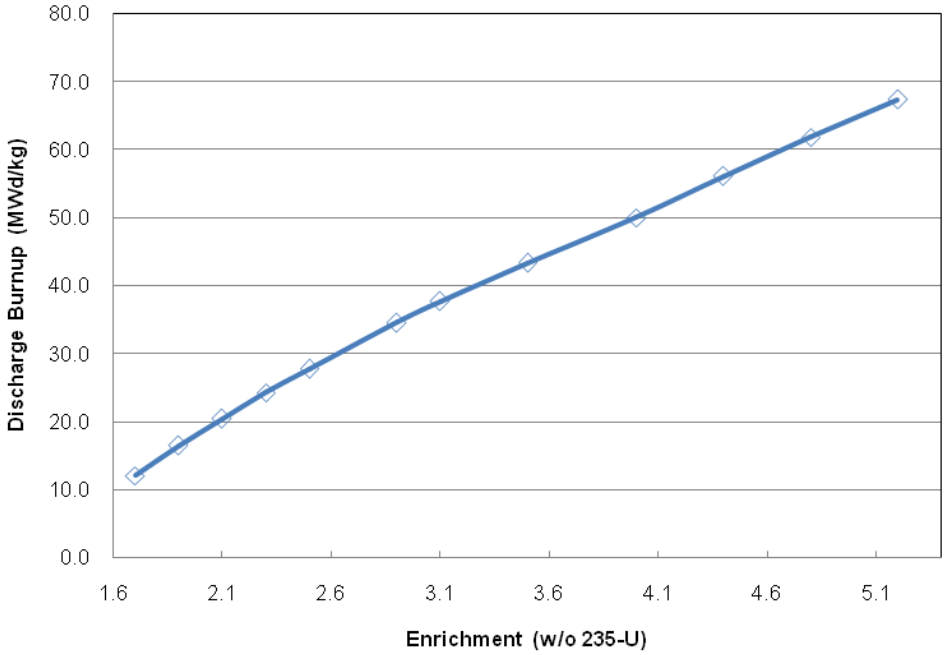


Figure 3. 1 Bundle Discharge Burnup as a Function of Enrichment

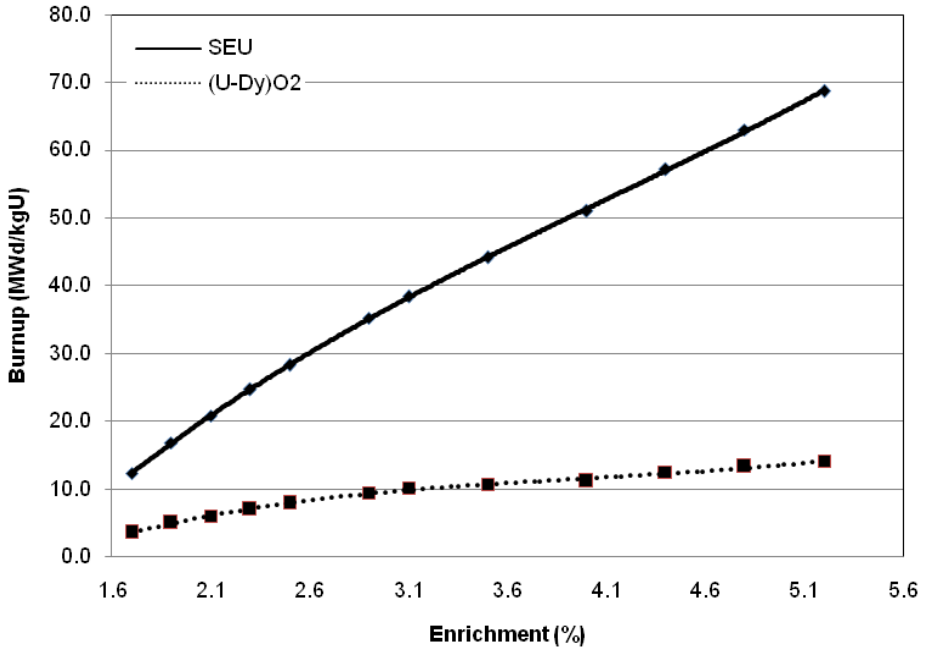


Figure 3. 2 Discharge Burnup as a Function of Enrichment

$$B_d = K_o + K_1\varepsilon + K_2\varepsilon^2 \qquad \textbf{[Eq.3. 1]}$$

where ε is in weight percent; and K_o, K_1 and K_2 are constants with unit of kg/MWd.

Table 3. 2 Burnup Curve Fit Parameters

Coefficients			
Burnup	K_o	K_1	K_2
SEU	-22.34	22.53	-0.990
$(NU+Dy)O_2$	-5.932	6.94	-0.610
Bundle Average	-21.96	22.17	-0.982

3.1.2 Reactivity as a Function of Discharge Burnup

Reactivity change for various enrichments is shown in Figure 3. 3.

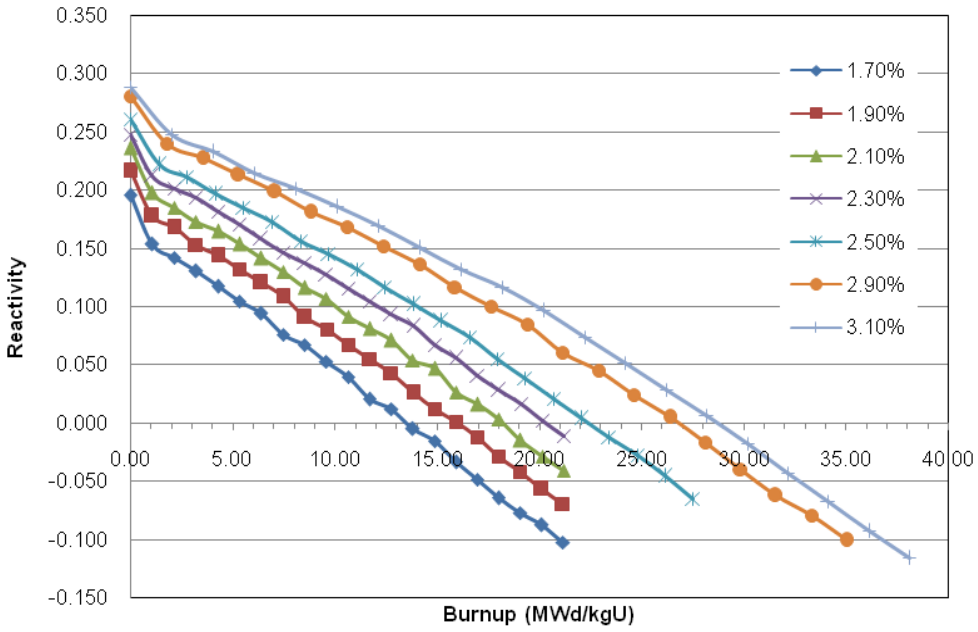


Figure 3. 3 Reactivity for Various Enrichments

At the beginning of the cycle, fuels have got positive reactivity. However, as the fuels burn in core, reactivity decreases sharply in a few days of beginning of life due to the effect of the Xe and Sm, and then goes to a negative value with a downward slope. It takes first 5 to 10 days from the beginning for fission products to reach the equilibrium

concentration. At the end of the cycle, total reactivity becomes zero, and thus, fuel can no longer be burned.

For an enrichment smaller than 3.1%, change of reactivity with burnup can be described by a second order polynomial fit with sufficient accuracy.

$$\rho(B_d) = \rho_o - A_1 B_d - A_2 B_d^2 \qquad \textit{[Eq.3. 2]}$$

where ρ_o, A_1 (in kg/MWd) and A_2 [in $(kg/MWd)^2$] are constants.

Table 3. 3 Reactivity Curve Fit Parameters

Coefficients			
U-235 Enrichment (%)	ρ_o x10^1	A_1 x10^3	A_2 x10^5
1.7	1.675	11.47	6.723
1.9	1.902	10.64	7.980
2.1	2.063	9.434	10.71
2.3	2.208	8.802	10.08
2.5	2.337	8.190	9.682
2.9	2.536	7.185	8.573
3.1	2.591	6.198	9.763

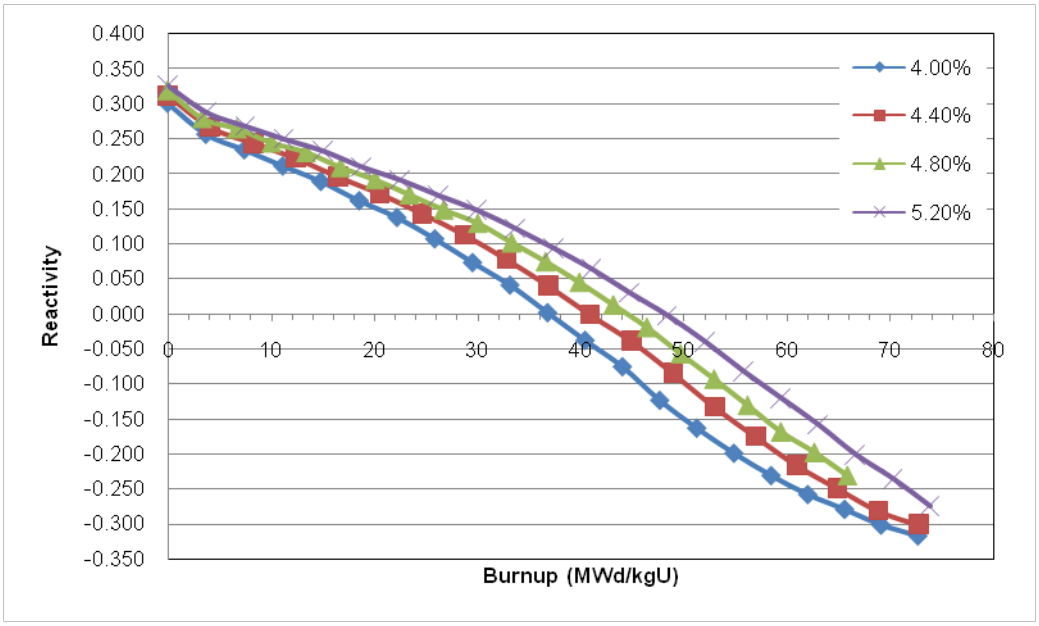


Figure 3. 4 Fuel Reactivity Change for Different Enrichments

For an enrichment greater than 3.1%, change of reactivity with burnup can be described by a third order polynomial fit with sufficient accuracy.

$$\rho(B_d) = \rho_o - A_1 B_d - A_2 B_d^2 - A_3 B_d^3 \qquad \textit{[Eq.3. 3]}$$

where ρ_o, A_1 (in MWd/kg), A_2 [in $(MWd/kg)^2$] and A_3 [in $(MWd/kg)^3$] are constants.

Table 3. 4 Reactivity Curve Fit Parameters

Coefficients				
U-235 Enrichment (%)	ρ_o x10^1	A_1 x10^3	A_2 x10^5	A_3 x10^7
4	2.796	4.825	18.22	-15.27
4.4	2.882	2.574	16.45	-12.12
4.8	3.045	4.825	3.150	3.217
5.2	3.124	4.530	2.920	2.526

3.1.3 Core Power Sharing

Change of power shares from fissioning of U-235 and Pu-239 plus Pu-241 with enrichment of fresh SEU is given in Figure 3. 5. It should be noted that the contribution of U-238 in fast region changes from 3.5% to 5% for SEU and from 13% to 18% for (U-Dy)O_2. More detailed data are listed in Appendix B.

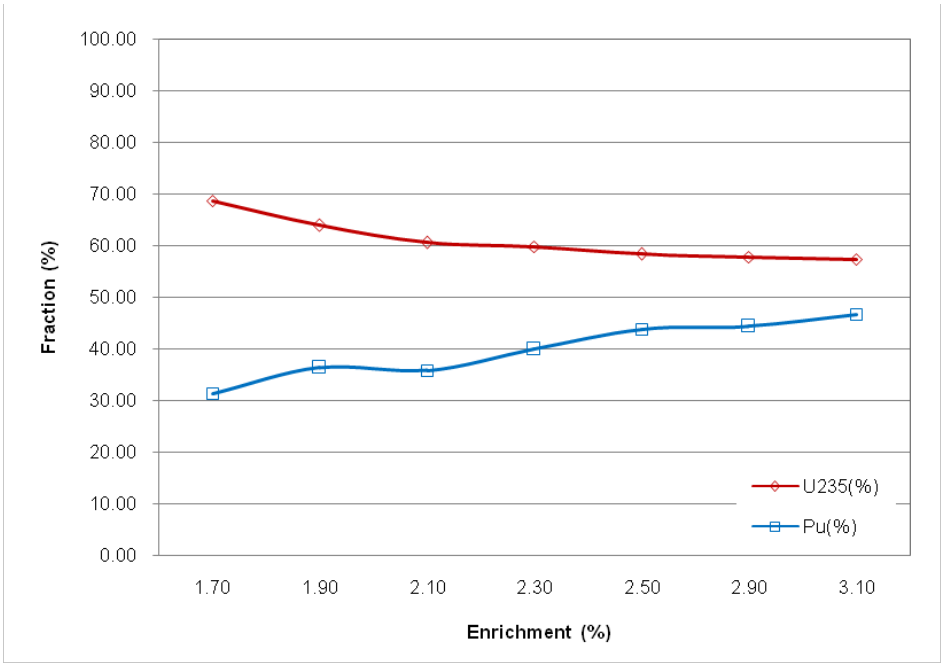


Figure 3. 5 Power Fractions in ACR

3.2 (U-Th) IN ACR

3.2.1 Discharge Burnup as a Function of U-235 and Th Fractions

Fresh (U-Th)O_2 fuels with various fractions of U-235 and Th are burned and results are tabulated in Table 3. 5. Here, U-235 fraction (%) is defined as amount of U-235 divided by total (U-Th) and multiplied by 100.

Table 3. 5 Burnup Values of (U-Th) Fuels

Th Fraction (%)	U-235 Fraction (%)	Burnup (MWd/kgHM)		
		(U-ThO_2)	*(NU-Dy)O_2*	*Bundle-Averaged*
10	**2.1**	16.320	4.295	16.040
	2.5	24.900	6.545	24.475
	2.9	32.485	8.335	31.920
	3.4	41.260	10.330	40.540
	4.0	51.285	12.355	50.380
30	**2.1**	6.975	1.430	6.845
	2.5	18.220	4.090	17.890
	2.9	27.070	6.170	26.580
	3.4	36.990	8.375	36.325
	4.0	47.700	10.705	46.835
	4.3	52.880	11.655	51.920
	4.6	56.013	10.636	54.958
	5.0	62.633	12.018	61.456
	5.4	68.936	12.945	67.634
	5.8	75.184	13.891	73.759
	6.2	80.802	14.691	79.265
50	**2.5**	9.610	1.745	9.425
	2.9	20.900	4.195	20.510
	3.4	32.105	6.530	31.510
	3.7	38.315	7.880	37.605
	4.0	44.105	9.070	43.290
	4.3	49.805	10.270	48.885
	4.6	52.998	9.383	51.984
	5.0	60.111	10.579	58.959
	5.4	66.393	11.388	65.114
	5.8	72.861	12.671	71.461
	6.2	79.083	13.677	77.562

Th Fraction (%)	U-235 Fraction (%)	Burnup (MWd/kgHM)		
		(U-ThO_2)	*(NU-Dy)O_2*	*Bundle-Averaged*
70	**2.9**	13.905	2.355	13.635
	3.4	27.235	5.020	26.720
	3.7	33.945	6.355	33.305
	4.0	39.870	7.655	39.125
	4.3	46.450	8.895	45.580
	4.6	52.240	9.925	51.255
	5.0	57.518	9.413	56.399
	5.4	64.521	10.625	63.268
	5.8	70.891	11.734	69.515
	6.2	77.409	12.991	75.911
90	**2.9**	7.730	1.055	7.575
	3.4	23.590	3.950	23.130
	3.7	31.490	5.540	30.885
	4.0	38.270	6.785	37.540
	4.3	44.555	7.940	43.700
	4.6	50.490	9.170	49.525

Discharge Burnup versus U-235 Fraction for Fixed Th Fractions

For fixed Th fractions, change of discharge burnup with U-235 fraction is given in Figure 3. 6 and can be described by a second order polynomial fit (Eq.3.4) with sufficient accuracy.

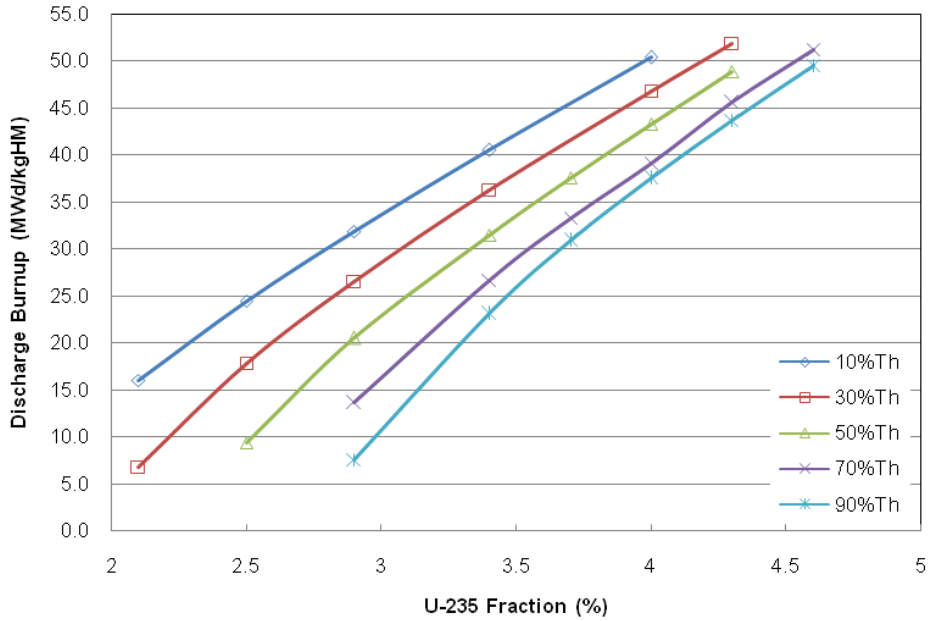


Figure 3. 6 Discharge Burnup as a Function of Enrichment

$$B_d(\varepsilon) = B_o + K_1\varepsilon + K_2\varepsilon^2 \qquad \textbf{[Eq.3. 4]}$$

where B_o, K_1 and K_2 are constants with unit of MWd/kg.

Table 3. 6 Discharge Burnup Curve Fit Parameters

Coefficients			
Th Fraction	B_o	K_1	K_2
10	-34.54	27.38	-1.542
30	-59.93	37.75	-2.747
50	-76.20	41.74	-2.954
70	-83.69	40.99	-2.541
90	-127.30	60.49	-4.805

It can be observed from Figure 3. 6 and Table 3. 5 that as Th fraction increases a higher U-235 fraction is required to accumulate the same discharge burnup that can be obtained from U fuels (with no Th). For a specific Th fraction, discharge burnup increases as the fraction of U-235 increases.

Discharge Burnup versus Th Fraction for Fixed U-235 Fractions

For fixed U-235 fractions, change of discharge burnup with Th fraction is shown in Figure 3. 7 and can be expressed well by a linear fit (Eq.3.5).

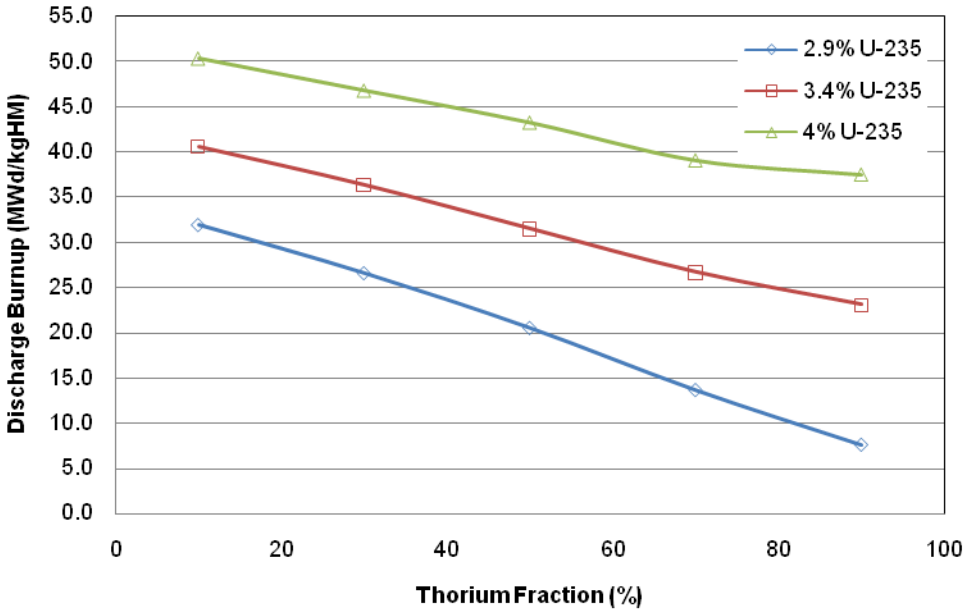


Figure 3. 7 Discharge Burnup as a Function Th Fraction

$$B_d(\theta) = B_o + K\theta \qquad \textbf{[Eq.3. 5]}$$

where θ is in weight percent; and B_o and K are constants with unit of MWd/kg.

Table 3. 7 Discharge Burnup Curve Fit Parameters

Coefficients		
U-235 Fraction (%)	B_o	*K*
2.9	35.45	-0.308
3.4	42.75	-0.222
4.0	51.78	-0.167

The Resulting Relation

Change of discharge burnup with Th and U-235 fractions can be expressed by a two-variable first-order polynomial fit with sufficient accuracy.

$$B_d(\varepsilon,\theta) = (B_{o1} + A_1\theta)\varepsilon + (B_{o2} + A_2\theta) \qquad \textbf{[Eq.3. 6]}$$

where ε and θ are in weight percent; and B_{o1}, B_{o2}, A_1 and A_2 are constants with unit of MWd/kg. (Eq.3.6) is valid for $1.7 < \varepsilon < 4.6$ and $10 < \theta < 90$, with ±5.4% computational error.

Table 3. 8 Discharge Burnup Curve Fit Parameters

Coefficients			
A_1	A_2	B_{o1}	B_{o2}
0.481	-0.074	17.35	-17.50

U-235 Fraction versus Th Fraction for Fixed Discharge Burnup

For fixed discharge burnups, change of U-235 fraction with Th fraction is plotted in Figure 3. 8 and can be expressed by a second-order polynomial fit (Eq.3.7) with adequate accuracy.

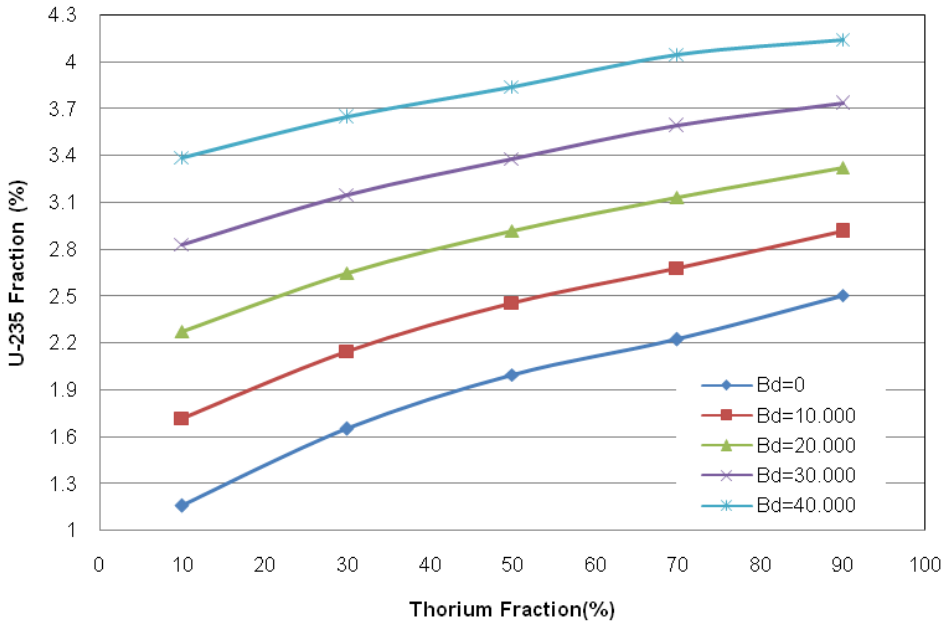


Figure 3. 8 U-235 Fraction as a Function of Th Fraction for Constant Burnups

$$\varepsilon(\theta) = \varepsilon_o + K_1\theta + K_2\theta^2 \qquad \textbf{[Eq.3. 7]}$$

where ε_o, K_1 and K_2 are constants.

Table 3. 9 U-235 Fraction Curve Fit Parameters

Coefficients			
Burnup MWd/kgHM	ε_o	K_1 x10^3	K_2 x10^5
0	0.928	2.595	-9.643
10	1.506	2.328	-8.661
20	2.084	2.054	-7.607
30	2.662	1.786	-6.663
40	3.240	1.514	-5.589

3.2.2 Reactivity as a Function of Discharge Burnup

Reactivity change for various enrichments is shown in Figure 3. 9.

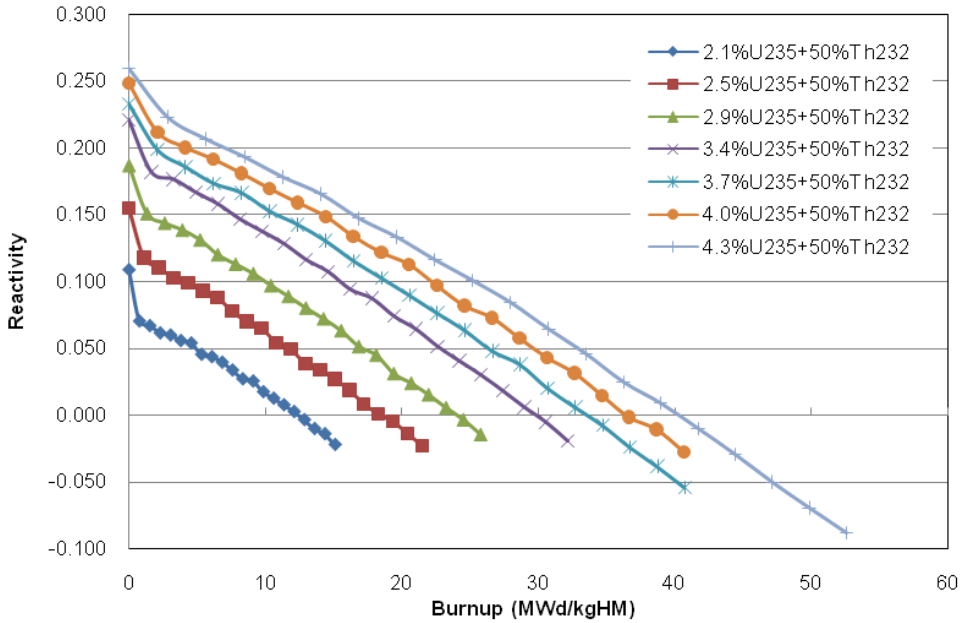


Figure 3. 9 Reactivity change as a Function of Burnup

Change of reactivity with burnup can be described by a second order polynomial fit with sufficient accuracy.

$$\rho(B_d) = \rho_o - A_1 B_d - A_2 B_d^2 \qquad \textbf{[Eq.3. 8]}$$

where ρ_o, A_1 (in kg/MWd) and A_2 [in $(kg/MWd)^2$] are constants.

Table 3. 10 Reactivity Curve Fit Parameters

Coefficients [a]			
U-235 Fraction (%)	ρ_o x10^2	A_1 x10^3	A_2 x10^5
2.1	7.353	4.197	14.0
2.5	12.4	5.754	5.079
2.9	15.9	5.479	4.965
3.4	19.31	5.328	3.889
3.7	20.85	5.049	3.482
4.0	22.17	4.736	3.512
4.3	23.66	4.828	2.579
5.0	24.37	3.668	3.230
5.4	25.65	3.643	2.770
5.8	26.21	3.081	3.131
6.2	26.89	2.715	3.217

CHAPTER 4. BURNING U AND (U-Th) IN CANDU-6

4.1 U IN CANDU-6

4.1.1 Discharge Burnup as a Function of Fuel Composition

Fresh UO_2 fuels with various enrichments are burned and results are tabulated in Table 4. 1.

Table 4. 1 Burnup Values

U-235 Enrichment	*Bundle Discharge Burnup (MWd/kgU)*	*Fuel Exposure Time (Days)*
0.711	7.154	320
0.8	9.900	435
1.0	16.500	715
1.2	22.000	930
1.4	27.000	1170
1.6	31.600	1380
1.8	36.000	1575
2.0	40.200	1760

NU fuel gains the burnup of 7.154MWd/kgU with 320 irradiation days. By increasing the enrichment up to 2.0%, bundle burnup accumulates 40.200MWd/kgU with the fuel exposure days of 1760. As to results seen Figure 4. 1, discharge burnup increases almost linearly as enrichment increases.

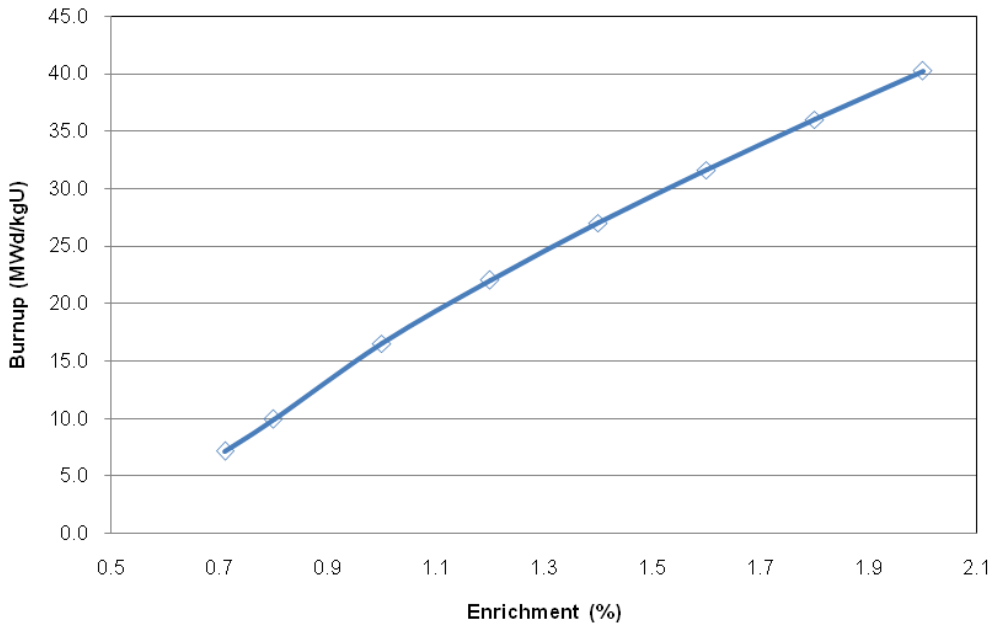


Figure 4. 1 Discharge Burnup as a Function of Enrichment

A linear expression adequately explains the relationship between enrichment and discharge burnup.

$$B_d = 22.62 \times \varepsilon - 9.876 \qquad \textbf{[Eq.4. 1]}$$

Results obtained are fully in agreement with the results given by Combustion Engineering (of a PHWR for U.S. Applications) and AECL. [18]

4.1.2 Reactivity as a Function of Discharge Burnup

Reactivity change for various enrichments is shown in Figure 4. 2.

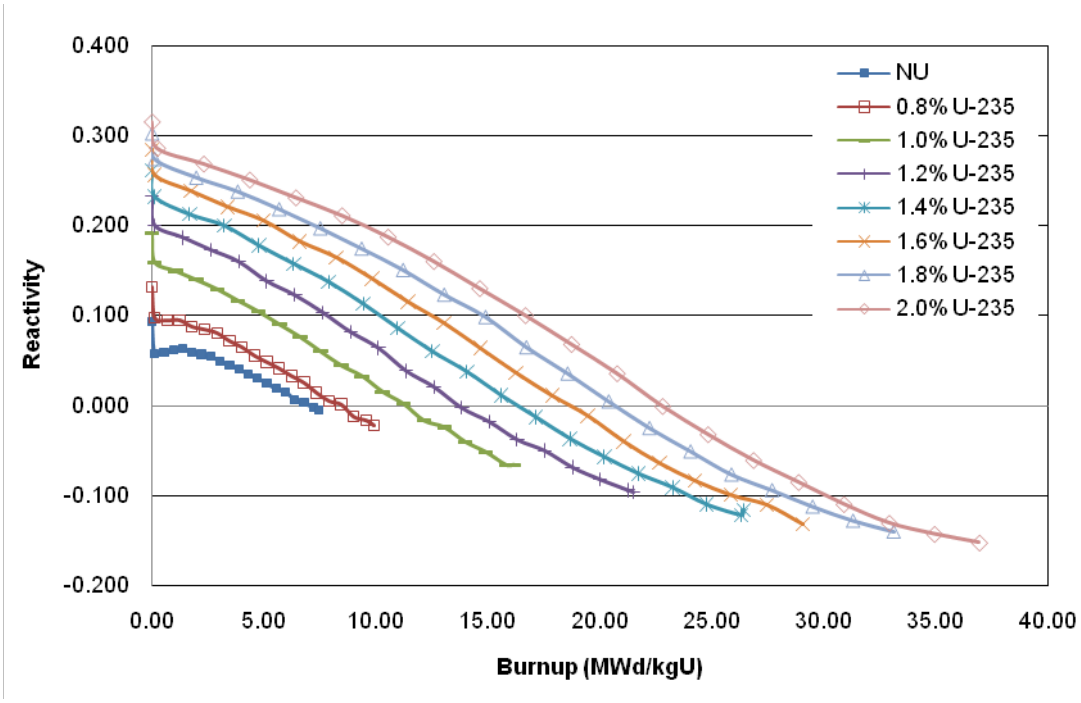


Figure 4. 2 Reactivity as a Function of Burnup

At start-up all fuels are highly reactive; however, the reactivities fall sharply due to fission product poisoning in 5 to 10 days from the beginning. Similar to CANDU-6 U fuels, after equilibrium of poisoning, reactivities decrease with negative slope except NU fuel. Reactivity of NU fuel drops rapidly in a few days of start-up; but, after equilibrium, it keeps on increasing till production of fissile Pus reaches equilibrium. Before decreasing due to consumption of fissile materials, it reaches to a local maximum.

Change of reactivity with burnup can be expressed by a third order polynomial fit with sufficient accuracy.

$$\rho(B_d) = \rho_o - A_1 B_d - A_2 B_d^2 - A_3 B_d^3 \qquad \textbf{[Eq.4. 2]}$$

where ρ_o, A_1 (in kg/MWd), A_2 [in $(kg/MWd)^2$] and A_3 [in $(kg/MWd)^3$] are constants.

Table 3. 11 Reactivity Curve Fit Parameters

Coefficients				
U-235 Enrichment (%)	ρ_o x10	A_1 x10^3	A_2 x10^4	A_3 x10^5
NU	0.5794	-7.671	4.118	-2.621
0.8	0.9843	1.397	2.115	-1.045
1.0	1.608	8.281	9.078	-3.381
1.2	2.015	8.450	7.770	-2.457
1.4	2.317	7.924	6.739	-1.780
1.6	2.537	6.624	6.455	-1.477
1.8	2.692	5.003	6.551	-1.305
2.0	2.816	3.158	6.707	-1.181

Results obtained are fully in agreement with the results given by Combustion Engineering (of a PHWR for U.S. Applications) and AECL. [18]

4.1.3 Core Power Sharing

Change of power shares from fissioning of U-235 and Pu-239 plus Pu-241 with enrichment of fresh SEU is given in Figure 4. 3. CANDU-6 fueled with NU produces approximately 40% of power from U-235 and 55% from Pu. It should be noted that the contribution of U-238 in fast region changes from 3.5% to 5%. More detailed data are listed in Appendix B.

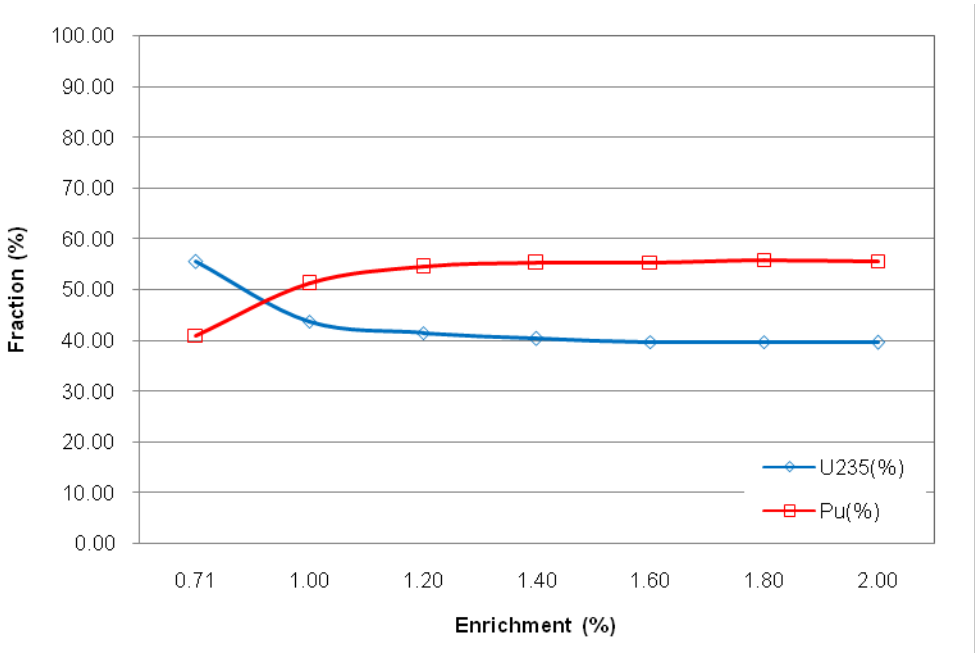


Figure 4. 3 Power Fractions in CANDU-6

4.2 (U-Th) IN CANDU-6

4.2.1 Discharge Burnup as a Function of U-235 and Th Fractions

Fresh (U-Th)O_2 fuels with various fractions of U-235 and Th are burned and results are tabulated in Table 4. 2. Here, U-235 fraction (%) is defined as amount of U-235 divided by total (U-Th) and multiplied by 100.

Table 4. 2 Discharge Burnup of (U-Th) Fuel

U-235 Faction (%)	*Bundle Discharge Burnup (MWd/kgHM) (10 % Th)*	*Bundle Discharge Burnup (MWd/kgHM) (30 % Th)*
1	12.260	
1.2	18.490	9.570
1.4	25.105	18.580
1.6	30.485	25.730
1.8	35.220	31.800
2.1	42.110	39.990
2.3	46.380	44.925
2.6	52.725	51.915
3.1	61.285	62.470
3.5		71.808
3.9		79.566
4.4		88.857

U-235 Fraction (%)	*Bundle Discharge Burnup (MWd/kgHM) (50 % Th)*	*Bundle Discharge Burnup (MWd/kgHM) (70 % Th)*
1.4	5.535	
1.6	17.520	1.245
1.8	25.545	15.515
2.0	32.460	24.865
2.1	35.615	29.230
2.3	41.510	36.465
2.5	47.330	43.140
2.6	49.765	46.185
2.7	52.970	49.060
3.1	61.200	60.000
3.5	71.718	71.289
3.9	80.669	80.581
4.4	90.767	91.613

U-235 Fraction (%)	*Bundle Discharge Burnup (MWd/kgHM) (90 % Th)*
2.0	14.675
2.1	20.365
2.3	29.540
2.5	37.605
2.6	41.395
2.8	47.990
3.0	54.185
3.1	57.140

Discharge Burnup versus U-235 Fraction for Fixed Th Fractions

For fixed Th fractions, change of discharge burnup with U-235 fraction is given in Figure 4. 4 and can be described by a second order polynomial fit (Eq.4.3) with sufficient accuracy.

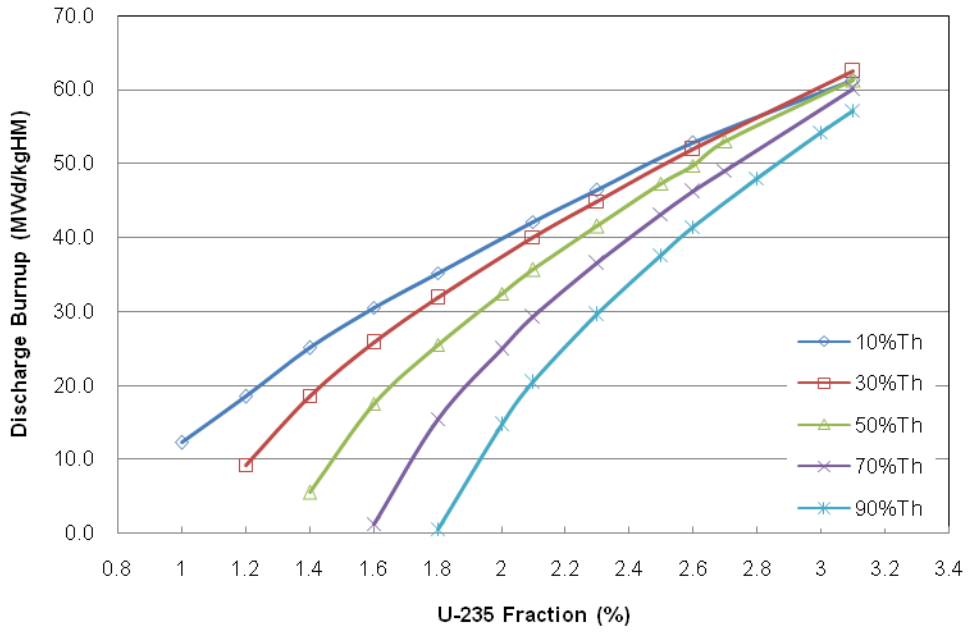


Figure 4. 4 Discharge Burnup as a Function of U-235 fraction

$$B_d(\varepsilon) = B_o + K_1\varepsilon + K_2\varepsilon^2 \qquad \textbf{[Eq.4. 3]}$$

where B_o, K_1 and K_2 are constants with unit of (MWd/kg).

Table 4. 3 Discharge Burnup Curve Fit Parameters

Coefficients [a]			
Th Fraction (%)	B_o	K_1	K_2
10	-22.92	39.48	-3.993
30	-44.49	52.90	-5.988
50	-85.90	82.27	-11.54
70	-122.8	99.94	-13.32
90	-164.8	121.8	-16.28

It can be observed from Table 4. 2 and Figure 4. 4 that as Th fraction increases a higher U-235 fraction is required to accumulate the same discharge burnup that can be obtained from U fuels (with no Th). For a specific Th fraction, discharge burnup increases as the fraction of U-235 increases.

Discharge Burnup versus Th Fraction for Fixed U-235 Fractions

For fixed U-235 fractions, change of discharge burnup with Th fraction is shown in Figure 4. 5 and can be expressed well by a second order polynomial fit (Eq.4.4).

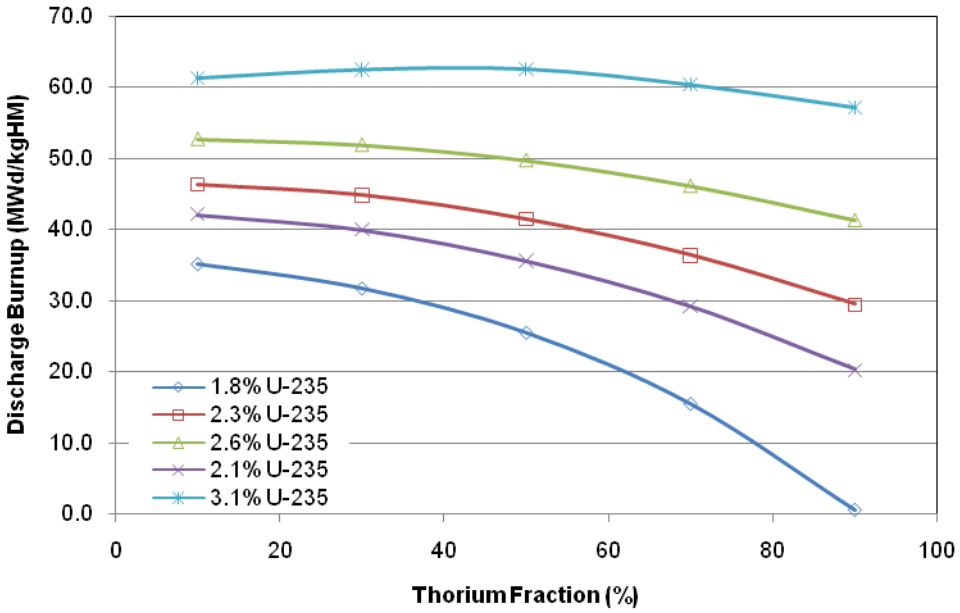


Figure 4. 5 Discharge Burnup as a Function of Th Fraction

$$B_d(\theta) = B_o + K_1\theta + K_2\theta^2 \qquad \textbf{[Eq.4. 4]}$$

where B_o, K_1 and K_2 are constants with unit of (MWd/kg).

Table 4. 4 Discharge Burnup Curve Fit Parameters

Coefficients [a]			
U-235 Fraction (%)	B_o x10^{-1}	K_1 x10^{2}	K_2 x10^{3}
1.8	3.499	5.110	-4.791
2.1	4.232	0.560	-2.768
2.3	4.648	1.377	-2.245
2.6	5.264	2.588	-1.678
3.1	6.000	14.56	-1.977

The Resulting Relation

Change of discharge burnup with Th and U-235 fractions can be expressed by a two-variable second-order polynomial fit with sufficient accuracy.

Change of discharge burnup with Th fractions and U-235 fractions can be expressed by a two-variable second-order polynomial fit with sufficient accuracy.

$$B_d(\varepsilon,\theta) = (B_{o1} + A_1\theta + C_1\theta^2)\varepsilon^2 + (B_{o2} + A_2\theta + C_2\theta^2)\varepsilon + (B_{o3} + A_3\theta + C_3\theta^2) \qquad \textbf{[Eq.4. 5]}$$

where ε and θ are in weight percent; and B_{o1}, B_{o2}, B_{o3}, A_1, A_2 , A_3 , C_1, C_2 and C_3 are constants with unit of MWd/kg. (Eq.4.4) is valid for $1 < \varepsilon < 3.1$ and $10 < \theta < 90$, with ±3.5% computational error.

Table 4. 5 Discharge Burnup Curve Fit Parameters

Coefficients								
A_1	A_2	A_3	B_{o1}	B_{o2}	B_{o3}	C_1	C_2	C_3
-0.192	0.966	-1.161	-1.689	27.93	-8.699	3.289	9.250	-6.491

U-235 Fraction versus Th Fraction for Fixed Discharge Burnup

For fixed discharge burnups, change of U-235 fraction with Th fraction is plotted in Figure 4. 6 and can be expressed well by a linear fit (Eq.4.6).

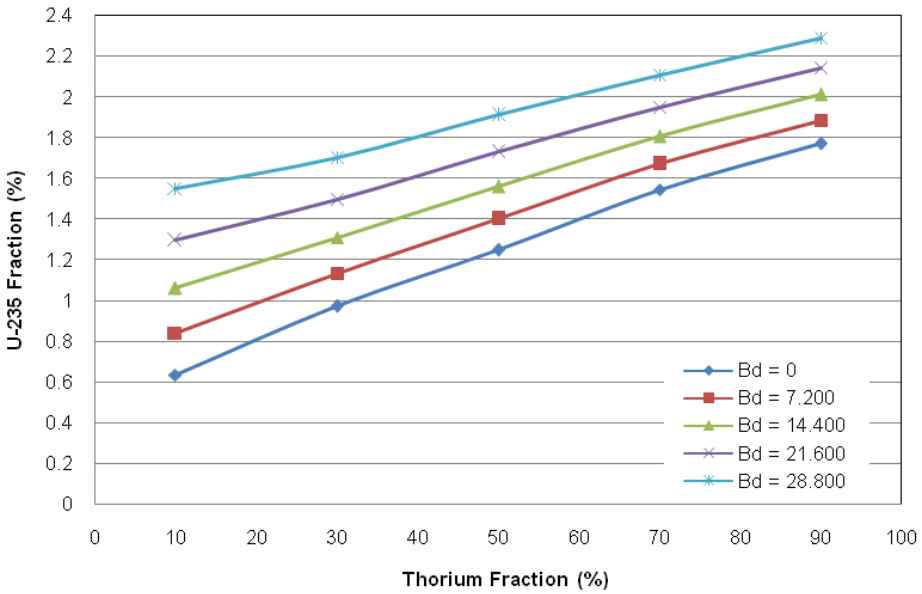


Figure 4. 6 U-235 Fraction vs. Th Fraction for Specific Discharge Burnups

$$\varepsilon(\theta) = \varepsilon_o + K\theta \qquad \textbf{[Eq.4. 6]}$$

where ε_o and K are constants.

Table 4. 6 U-235 Fraction Curve Fit Parameters

Coefficients		
Burnup MWd/kgHM	ε_o	K x10^2
0	0.523	1.425
7.2	0.731	1.314
14.4	0.951	1.199
21.6	1.187	1.075
28.8	1.443	0.945

4.2.2 Reactivity as a Function of Discharge Burnup

Reactivity changes for various U-235 fractions are shown in Figure 4. 7.

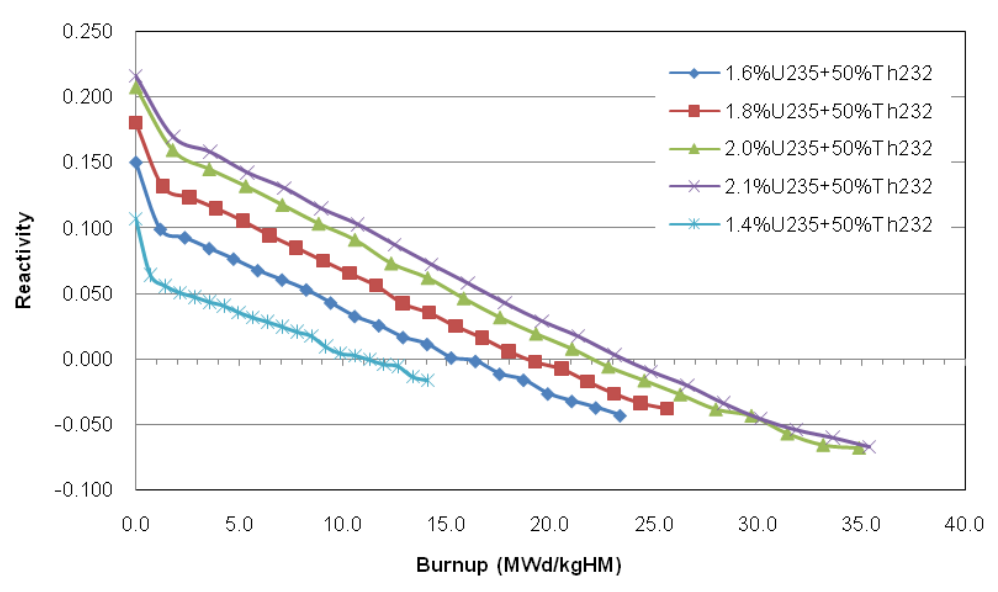


Figure 4. 7 Reactivity as a Function of Burnup for (U-Th) Fuel

Change of reactivity with burnup can be described by a second order polynomial fit with sufficient accuracy.

$$\rho(B_d) = \rho_o - A_1 B_d - A_2 B_d^2 \qquad \textbf{[Eq.4. 7]}$$

where ρ_o, A_1 (in kg/MWd) and A_2 [in $(kg/MWd)^2$] are constants.

Table 4. 7 Reactivity Curve Fit Parameters

Coefficients			
U-235 Fraction (%)	ρ_o x10^1	A_1 x10^3	A_2 x10^5
1.4	0.658	6.167	-2.457
1.6	1.111	7.789	-4.849
1.8	1.463	8.464	-4.511
2.0	1.800	9.521	-6.520
2.1	1.910	9.156	-4.797

CHAPTER 5. CONVERSION RATIO, NU AND FUEL REQUIREMENT, NUCLEAR RESOURCE UTILIZATION, AND NU SAVING

In this chapter, based on the results of burnup calculations presented so far, conversion ratios, NU and fuel requirements, nuclear resource utilization factors, and NU savings are determined and their variation with burnup and/or fuel composition are put forward.

5.1 DEFINITIONS AND/OR DEFINING EXPRESSIONS

Conversion Ratio:

$$CR = \frac{\text{number of fissile nuclei produced from fertile nuclei}}{\text{number of fissile nuclei consumed}} \qquad \textbf{[Eq.5. 1]}$$

NU and Fuel Requirement

Fuel requirement [19] is conventionlly defined as the total amount of fuel load in the reactor required to produce 1*GW(e)*-year, in [TonHM/GW(e)-year].

$$\text{Fuel Requirement} = 365\frac{1}{\eta_{th} B_d} \quad \text{in [TonHM/GW(e)-year]} \qquad \textbf{[Eq.5. 2]}$$

where η_{th} is the thermal efficiency, the unit of B_d is again MW(th)-day/kgHM (multiple of 365 comes from unit adjustment). NU requirement is then equal to the total amount of NU needed to load the reactor, i.e., to produce the fuel load given by (Eq.5. 2).

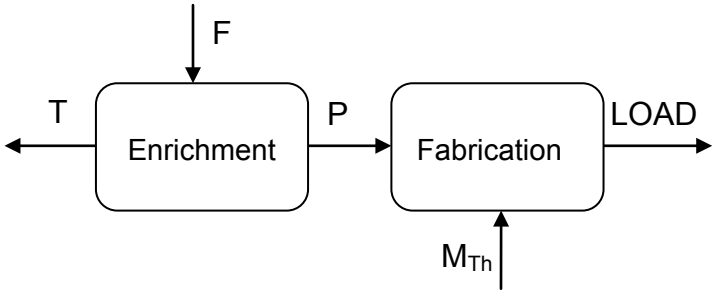
Nuclear Resource Utilization

Nuclear resource utilization, NRU, is defined as the ratio of amount of material fissioned to resource input. [20] Taking into account the fact that fissioning of 1 gram of fissile material yields approximately 1 MW(th)-day, NRU can be related to burnup by (Eq.5.3)

$$\text{NRU} = \frac{B_d \times 10^{-3}}{\text{RES/LOAD}} \qquad \textbf{[Eq.5. 3]}$$

where RES/LOAD is the amount of resources in HM required to produce a unit mass of fuel load in HM and B_d is in MW(th)-day/kgHM.

In case of (U-Th) fuels, fuel load is comprised of SEU and Th, and resources needed to produce LOAD consist of NU and Th. Considering the mass balance for enrichment and fabrication processes, RES/LOAD can be obtained in the following manner.



$LOAD = P + M_{Th}$ and $RES = F + M_{Th}$

where P is the product of enrichment and F is the feed to enrichment.

$$\frac{RES}{LOAD} = \frac{F}{LOAD} + \frac{M_{Th}}{LOAD}$$

Since $\frac{F}{P} = \frac{x_P - x_T}{x_F - x_T}$ from the enrichment mass balance, and $\frac{M_{Th}}{LOAD} = \frac{\theta}{100}$ and $\frac{P}{LOAD} = 1 - \frac{\theta}{100}$ from fabrication mass balance; where θ is the Th fraction in fuel load in weight percent, ignoring all the process losses.

$$\frac{RES}{LOAD} = \frac{\theta}{100} + (1 - \frac{\theta}{100})(\frac{x_P - x_T}{x_F - x_T}) \qquad \textbf{[Eq.5. 4]}$$

Note that when there is no Th in the load, (Eq.5.4) reduces to F/P of the enrichment.

NU Saving

NU saving, NUS, is defined as the reduction in NU requirement for a fuel cycle of interest with comparison to the NU requirement of a reference fuel cycle. The unit of NUS is TonHM/GW(e)-year. NUS can also be expressed in percentage as the ratio of this reduction to the NU requirement of the reference case. The reference cases are selected to be "NU fuel on the once-through cycle" for CANDU-6 and "2.1% SEU fuel on the once-through cycle" for ACR.

5.2 ACR-700

5.2.1 Conversion Ratios

Capture-to-fission ratios, computed using burnup-dependent values averaged over cycle length for a bundle, are given for the fissile isotopes in Table 5. 1.

Table 5. 1 Capture to Fission (α) Ratios for Fissile Isotopes

Fissile Isotope	α Computed*			
	U fuels	30% Th	50% Th	70% Th
^{233}U	0.112	0.116	0.117	0.118
^{235}U	0.195	0.201	0.203	0.207
^{239}Pu	0.513	0.525	0.529	0.536
^{241}Pu	0.361	0.362	0.363	0.364

* MCNP5 results

Conversion ratios calculated using capture-to-fission ratios, listed in Table 5. 1, are given in Table 5. 2 and their variation with burnup are plotted in Figure 5. 1. As can be observed, conversion ratios of (U-Th) fuels are significantly higher than that of U fuels, mainly due to greater absorption cross-section of Th.

Table 5. 2 Conversion Ratios in ACR

Initial Fuel	CR Computed*
SEU (2.1% ^{235}U)	0.515
SEU (2.9% ^{235}U)	0.489
SEU (3.5% ^{235}U)	0.48
3.4 % ^{235}U+30 %^{232}Th	0.535
3.4 % ^{235}U+50 %^{232}Th	0.556
3.4 % ^{235}U+70 %^{232}Th	0.572
4.0 % ^{235}U+70 %^{232}Th	0.559
4.6 % ^{235}U+70 %^{232}Th	0.546

* MCNP5 results

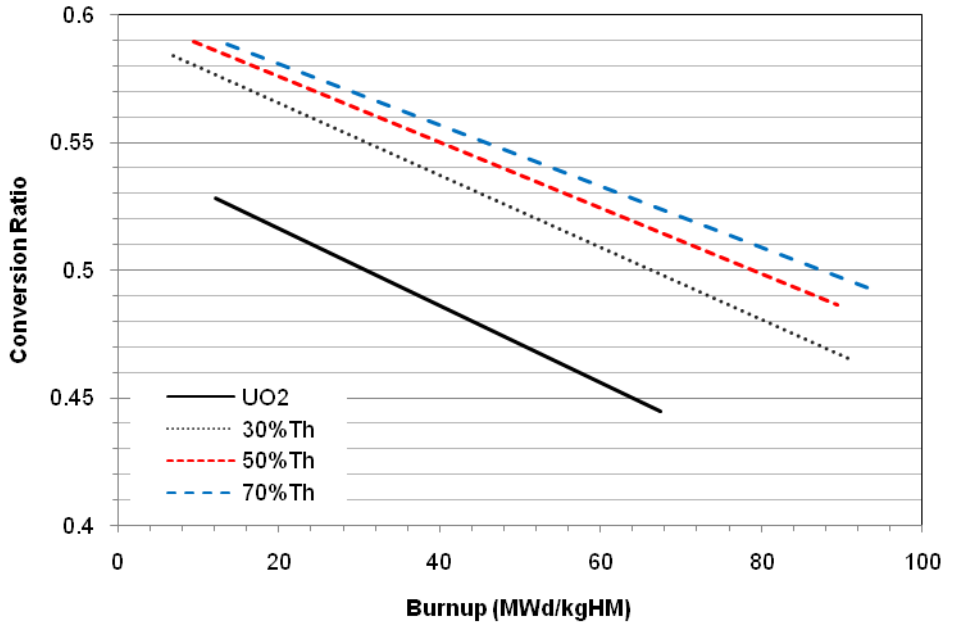


Figure 5. 1 Conversion Ratio as a Function of Burnup

The fertile-to-fissile conversion characteristics depend on the fuel cycle and the neutron energy spectrum. For a thermal neutron spectrum [$E < 1\ eV$], U-233 has the largest value of η. Thus the best possibility for fertile-to-fissile conversion in a thermal spectrum is with a Th-232-U-233 fuel cycle.

Appendix C includes a list of results of conversion ratios for various U and (U-Th) fuels. Two examples of cross-section sets used in computing capture-to-fission ratio are given in Appendix D.

5.2.2 NU Requirement

Change of NU requirement with discharge burnup is given in Figure 5. 2, and values calculated for various fuel compositions are tabulated in Table 5. 3 and Table 5. 4, together with fuel requirements. As seen, NU requirements in all (U-Th) fuels approach a constant value as burnup increases.

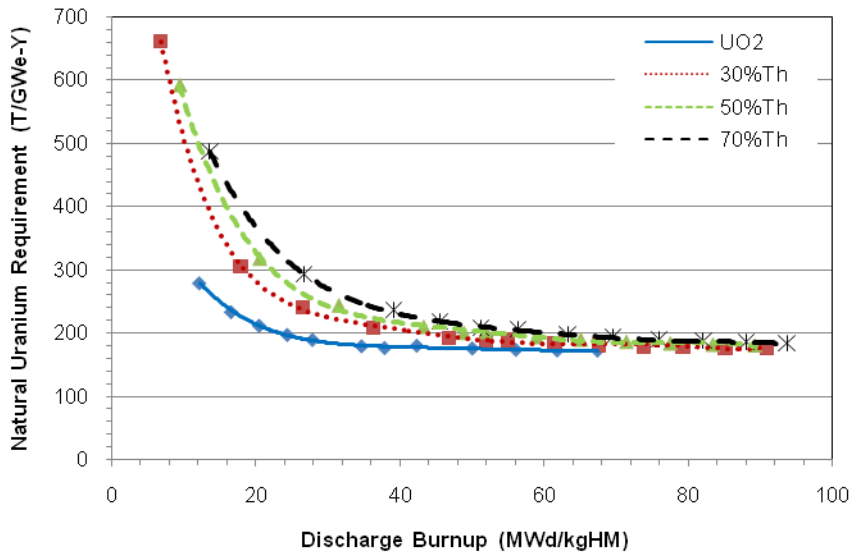


Figure 5. 2 NU Requirement vs. Discharge Burnup

5.2.3 Nuclear Resource Utilization

Resource utilization factors calculated using (Eq.5.3) are shown in Figure 5. 3. Nuclear resource utilization monotonically increases with discharge burnup. Utilization for (U-Th) fuels approach utilization from U fuels as burnup goes up.

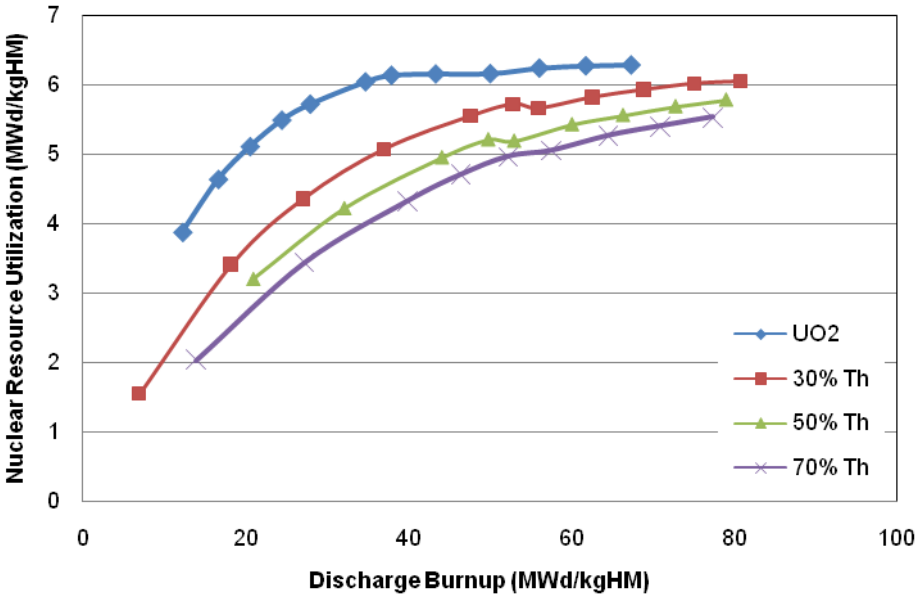


Figure 5. 3 Nuclear Resource Utilization as a Function of Burnup

5.2.4 NU Savings

For U fuels of various enrichments and (U-Th) fuels of various compositions, fuel and NU requirements, and NU savings are given in Table 5. 3 and Table 5. 4.

Table 5. 3 Fuel and NU Requirements, and NU Savings for U Fuels

Fuel	Enrichment (%)	Fuel Requirement (THM/GW(e)-year)	NU Requirement (THM/GW(e)-year)	NU Savings (THM/GW(e)-year)	NUS (%)
UO_2	1.7	88.709	279.02		
	1.9	65.195	233.34		
	2.1	**52.792**	**211.85**	**0**	**0**
	2.3	44.354	197.23	14.62	6.90
	2.5	38.790	189.32	22.53	10.63
	2.9	31.188	179.28	32.57	15.37
	3.1	28.555	176.53	35.32	16.67
	3.5	25.551	180.13	31.72	14.97
	4	21.617	175.84	36.01	16.99
	4.4	19.287	173.62	38.23	18.04
	4.8	17.500	172.72	39.13	18.47
	5.2	16.052	172.36	39.49	18.64

The shaded row represents the reference case.

Table 5. 4 Fuel and NU Requirements, and NU Savings for (U-Th) Fuels

Fuel	U-235 Fraction (%)	Fuel Requirement (THM/GW(e)-year)	NU Requirement (THM/GW(e)-year)	NU Savings (THM/GW(e)-year)	NUS (%)
UO2	**2.1**	**52.792**	**211.85**		
(U-Th) (30%Th)	2.1	158.108	660.21		
	2.5	60.494	305.09		
	2.9	40.716	240.67		
	3.4	29.793	208.42	3.43	1.61
	4	23.107	191.72	20.12	9.50
	4.3	20.844	186.51	25.34	11.96
	4.6	19.692	189.02	22.87	10.77
	5	17.610	184.31	27.53	13.00
	5.4	16.001	181.36	30.49	14.39
	5.8	14.672	179.03	32.82	15.49
	6.2	13.653	178.44	33.41	15.77
	6.6	12.717	177.25	34.60	16.33
	7	11.915	176.40	35.45	16.73

(U-Th) (50%Th)	2.5	114.828	591.57		
	2.9	52.767	317.63		
	3.4	34.346	244		
	4	25	210.14	1.71	0.81
	4.3	22.138	200.49	11.36	5.36
	4.6	20.818	202.09	9.76	4.60
	5	18.356	194.11	17.74	8.37
	5.4	16.620	190.18	21.67	10.22
	5.8	15.144	186.43	25.42	12.00
	6.2	13.953	183.87	27.98	13.20
	6.6	12.965	182.11	29.74	14.04
	7	12.112	180.63	31.22	14.73
(U-Th) (70%Th)	2.9	79.373	486.39		
	3.4	40.503	292.13		
	4	27.661	235.51		
	4.3	23.744	217.61		
	4.6	21.115	207.25	4.60	2.17
	5	19.189	205.00	6.85	3.23
	5.4	17.105	197.58	14.26	6.73
	5.8	15.568	193.34	18.51	8.74
	6.2	14.256	189.42	22.43	10.59
	6.6	13.192	186.73	25.12	11.86
	7	12.293	184.67	27.18	12.83

The shaded row represents the reference case.

Change of NU savings with discharge burnup is shown in Figure 5. 4. For all fuel types, NU savings approach a maximum as the burnup goes up. In (U-Th) fuels, as Th fraction goes up, NU savings decrease.

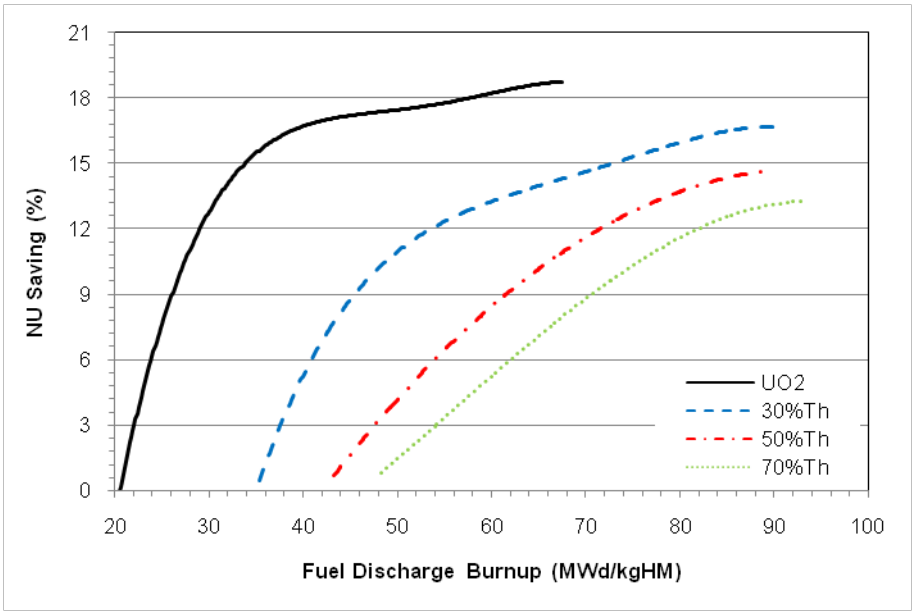


Figure 5. 4 NU Savings as a Function of Burnup

5.3 CANDU-6

5.3.1 Conversion Ratios

Capture-to-fission ratios, computed using burnup-dependent values averaged over cycle length for a bundle, are given for the fissile isotopes in Table 5. 5.

Table 5. 5 Core-Averaged Capture to Fission (α) Ratios for Fissile Isotopes

Fissile Isotope	α Literature [20]	α Computed*
^{233}U	0.0899	0.102
^{235}U	0.169	0.179
^{239}Pu	0.362	0.434
^{241}Pu	0.365	0.355

* MCNP5 results

Conversion ratios calculated using capture-to-fission ratios, listed in Table 5. 5, are given in Table 5. 6 and their variation with burnup is plotted in Figure 5. 5. As expected, like that in the case of ACR, conversion ratios of (U-Th) fuels are significantly higher than that of U fuels.

Table 5. 6 Conversion Ratios in CANDU-6

Initial Fuel	CR Literature [21]	CR Computed*
NU	0.7-0.8	0.771
SEU (1.2% ^{235}U)		0.696
SEU (1.6% ^{235}U)		0.670
SEU (1.8% ^{235}U)		0.656
1.8 % ^{235}U+30 %^{232}Th		0.736
1.8 % ^{235}U+50 %^{232}Th		0.766
1.8 % ^{235}U+70 %^{232}Th		0.785
2.5 % ^{235}U+70 %^{232}Th		0.758
3.1 % ^{235}U+70 %^{232}Th		0.735
4.4 % ^{235}U+70 %^{232}Th		0.689

* MCNP5 results

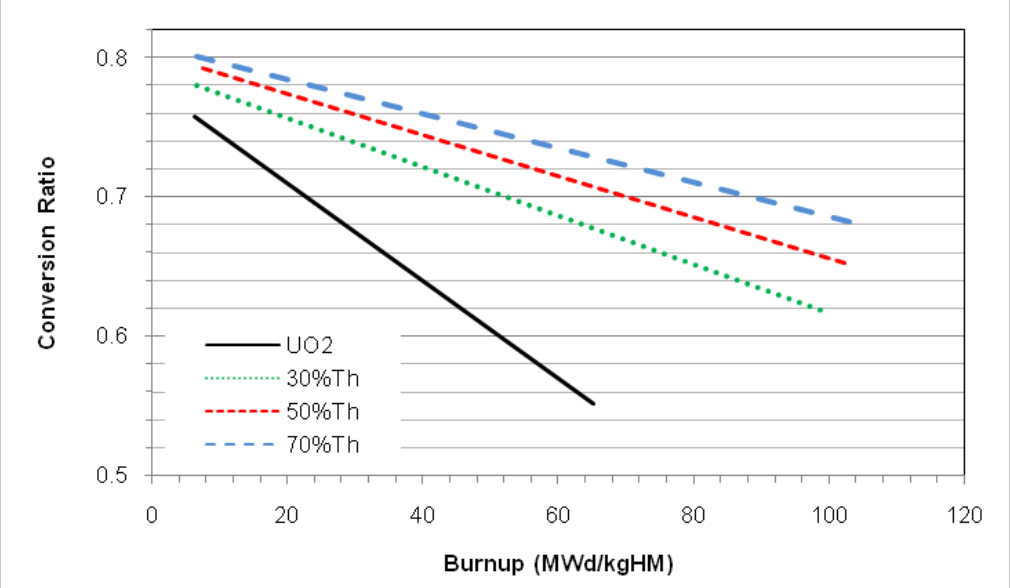


Figure 5. 5 Conversion Ratio as a Function of Burnup

5.3.2 NU Requirement

Change of NU requirement with discharge burnup is given in Figure 5. 6, and values calculated for various fuel compositions are tabulated in Table 5. 7 and Table 5. 8, together with fuel requirements. As seen, NU requirements in all (U-Th) fuels approach a constant value as burnup increases.

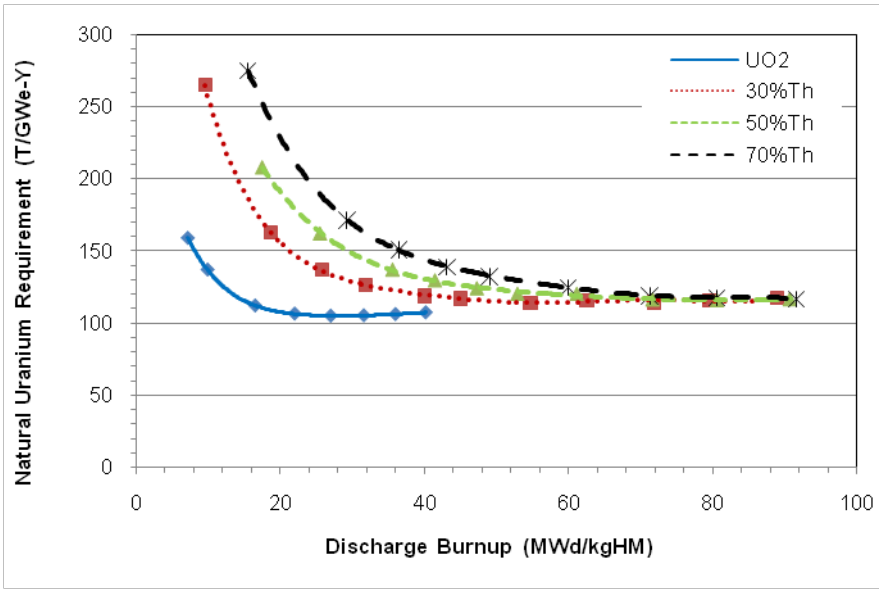


Figure 5. 6 Amount of NU Requirement with Discharge Burnup

5.3.3 Nuclear Resource Utilization

Resource utilization factors calculated using (Eq.5.3) are shown in Figure 5. 7. Nuclear resource utilization monotonically increases with discharge burnup. Utilization for (U-Th) fuels approach utilization from U fuels as burnup goes up. Results for U fuels are in good agreement with published data [18].

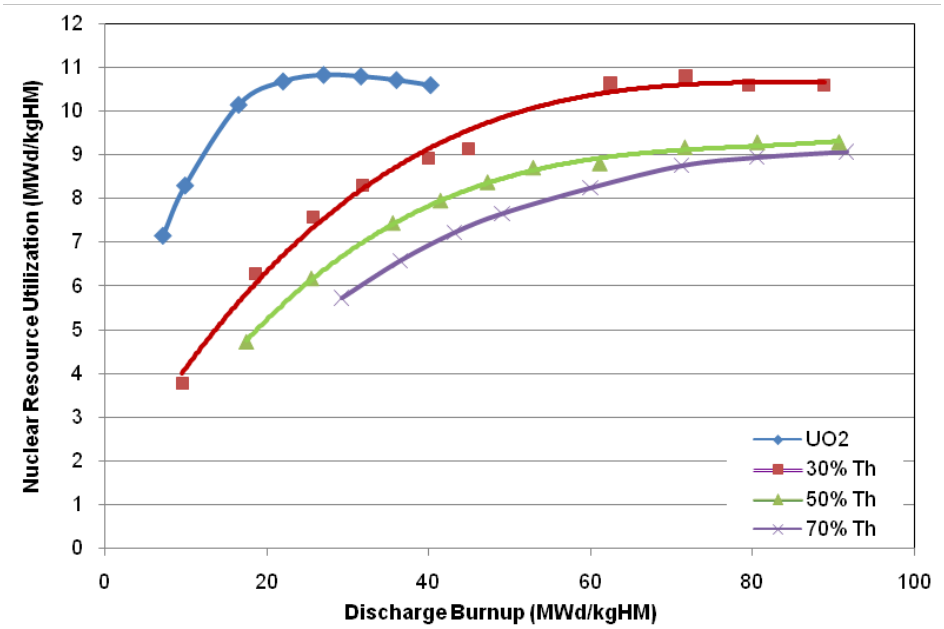


Figure 5. 7 Resource Utilization as a Function of Burnup for (U-Th) Fuels

5.3.4 NU Savings

For U fuels of various enrichments and (U-Th) fuels of various compositions, fuel and NU requirements, and NU savings are given in Table 5. 7 and Table 5. 8.

Table 5. 7 Fuel and NU Requirements, and NU Savings for U Fuels

Fuel	Enrichment (%)	Fuel Requirement (THM/GW(e)-year)	NU Requirement (THM/GW(e)-year)	NU Savings (THM/GW(e)-year)	NUS (%)
UO2	**0.711**	**159.386**	**159.386**	**0**	**0**
	0.8	115.177	137.413	21.973	13.78
	1	69.106	112.429	46.957	29.46
	1.2	51.829	106.807	52.579	32.99
	1.4	42.231	105.350	54.036	33.90
	1.6	36.083	105.669	53.717	33.70
	1.8	31.673	106.495	52.891	33.18
	2	28.364	107.674	51.712	32.44

The shaded row represents the reference case.

Table 5. 8 Fuel and NU Requirements, and NU Savings for (U-Th) Fuels

Fuel	U-235 Fraction (%)	Fuel Requirement (THM/GW(e)-year)	NU Requirement (THM/GW(e)-year)	NU Savings (THM/GW(e)-year)	NUS (%)
UO_2	**0.711**	**159.386**	**159.386**		
(U-Th) (30%Th)	1.2	119.148	264.918	-	-
	1.4	61.369	163.076	-	-
	1.6	44.316	136.985	22.401	14.05
	1.8	35.856	126.394	32.992	20.70
	2.1	28.513	119.064	40.322	25.29
	2.3	25.381	116.998	42.388	26.59
	3.1	20.857	114.238	45.148	28.32
	3.5	18.252	115.812	43.574	27.33
	3.9	15.879	114.53	44.856	28.14
	4.4	14.330	115.797	43.589	27.34
(U-Th) (50%Th)	1.6	65.082	208.237	-	-
	1.8	44.636	162.184	-	-
	2.1	32.016	137.162	22.224	13.94
	2.3	27.469	129.6	29.786	18.68
	2.5	24.091	124.116	35.27	22.12
	2.7	21.526	120.239	39.147	24.56
	3.1	18.631	120.236	39.15	24.56
	3.5	15.899	116.398	42.988	26.97
	3.9	14.134	115.747	43.639	27.37

	4.4	12.562	116.495	42.891	26.91
(U-Th) (70%Th)	1.8	73.493	275.003	-	-
	2.1	39.009	171.355	-	-
	2.3	31.269	150.922	8.464	5.31
	2.5	26.431	139.037	20.349	12.76
	2.7	23.242	132.343	27.043	16.96
	3.1	19.004	124.702	34.684	21.76
	3.5	15.994	118.833	40.553	25.44
	3.9	14.150	117.408	41.978	26.33
	4.4	12.446	116.769	42.617	26.73

The shaded row represents the reference case.

Change of NU savings with discharge burnup is shown in Figure 5. 8. For all fuel types, NU savings approach a maximum as the burnup goes up. SEU with an enrichment between 1.0% and 1.4% results in the highest NU savings. In (U-Th) fuels, as Th fraction goes up, NU savings decrease; however, as discharge burnup increases, the differences diminish, and finally they all reach the same value.

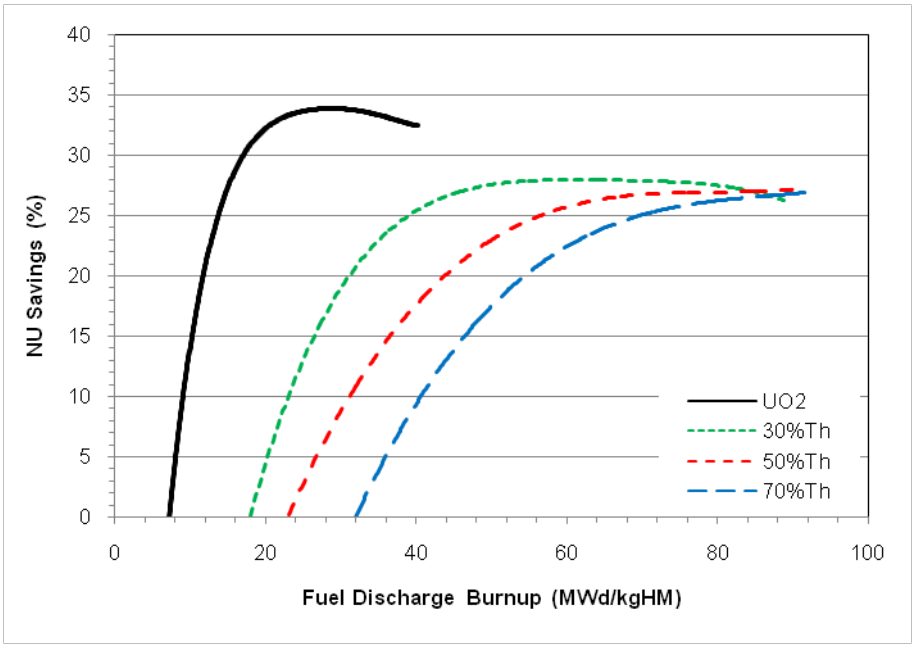


Figure 5. 8 NU Savings as a Function of Burnup

CHAPTER 6. RECYCLING OF U AND Pu CONTENTS OF SPENT CANDU-6 AND SPENT ACR FUELS INTO CANDU-6

Re-use of U and Pu contained in spent CANDU-6 and ACR fuels in CANDU-6 are to be investigated. Technical difficulties to recover U and Pu from SF containing Th are beyond the scope of this study. It is assumed that a reprocessing scheme to recover U and Pu with sufficient purity exists. To prepare recycle (U-Th) fuel from recovered U and Pu, a certain fraction of Th is required to be added. Fresh Th is used for this purpose.

6.1 RECYCLING OF U AND Pu CONTENTS OF SPENT CANDU-6 FUELS INTO CANDU-6

6.1.1 Composition of Spent CANDU-6 Fuel

For a few selected compositions of fresh U and (U-Th) fuels, weight percents of important isotopes in SF are tabulated in Table 6. 1 and Table 6. 2.

Table 6. 1 A Summary Table of Isotopic Compositions in Spent U Fuels

Isotopes	***Fresh Fuel Enrichment (%)***			
	NU	**1.2**	**1.6**	**2.0**
^{235}U	0.214	0.069	0.040	0.025
^{236}U	0.073	0.164	0.227	0.285
^{238}U	99.309	99.105	98.980	98.872
Total U	**99.596**	**99.338**	**99.248**	**99.182**
^{239}Pu	0.287	0.334	0.347	0.350
^{240}Pu	0.091	0.221	0.260	0.285
^{241}Pu	0.019	0.056	0.069	0.075
^{242}Pu	0.005	0.042	0.071	0.099
Total Pu	**0.402**	**0.653**	**0.752**	**0.809**
Fissile Pu (%)	**0.307**	**0.396**	**0.416**	**0.425**
Total Fissile (%)	**0.521**	**0.465**	**0.456**	**0.450**

Table 6. 2 A Summary Table of Isotopic Compositions in Spent (U-Th) Fuels

Isotopes	***30% Th***			***50% Th***			***70% Th***		
U-235 Fraction (%)	**1.6**	**1.8**	**2.3**	**1.6**	**1.8**	**2.3**	**1.8**	**2.3**	**2.6**
^{233}U	0.397	0.416	0.439	0.540	0.631	0.707	0.663	0.946	0.988
^{234}U	0.067	0.082	0.115	0.059	0.093	0.155	0.059	0.174	0.211
^{235}U	0.148	0.115	0.065	0.367	0.230	0.113	0.563	0.182	0.140
^{236}U	0.219	0.253	0.334	0.190	0.241	0.334	0.192	0.327	0.380
^{238}U	68.990	68.907	68.805	48.690	48.583	48.393	28.288	27.991	27.762
Total U	**69.821**	**69.773**	**69.758**	**49.847**	**49.778**	**49.702**	**29.766**	**29.619**	**29.481**
^{239}Pu	0.261	0.265	0.271	0.191	0.200	0.207	0.122	0.136	0.138
^{240}Pu	0.152	0.171	0.202	0.075	0.105	0.139	0.039	0.078	0.085
^{241}Pu	0.043	0.049	0.059	0.022	0.032	0.044	0.012	0.028	0.031
^{242}Pu	0.027	0.038	0.067	0.008	0.017	0.040	0.003	0.021	0.028
Total Pu	**0.487**	**0.528**	**0.607**	**0.297**	**0.357**	**0.437**	**0.177**	**0.268**	**0.290**
Total Th	29.692	29.699	29.635	49.856	49.866	49.861	70.057	70.113	70.229
Fissile U (%)	**0.545**	**0.531**	**0.505**	**0.908**	**0.861**	**0.820**	**1.226**	**1.128**	**1.128**
Fissile Pu (%)	**0.304**	**0.314**	**0.329**	**0.213**	**0.232**	**0.251**	**0.134**	**0.164**	**0.168**
Total Fissile (%)	**0.849**	**0.845**	**0.834**	**1.121**	**1.093**	**1.071**	**1.360**	**1.292**	**1.296**

For two selected fresh fuel compositions (2.6% U-235+50% Th-232 and 2.6% U-235+70% Th-232), changes of fractions of fissile isotopes during irradiation are shown in Figure 6. 1 and Figure 6. 2. Presence of Th in fresh fuel results in production of U-233; however, it has a negative influence on the production of fissile Pu.

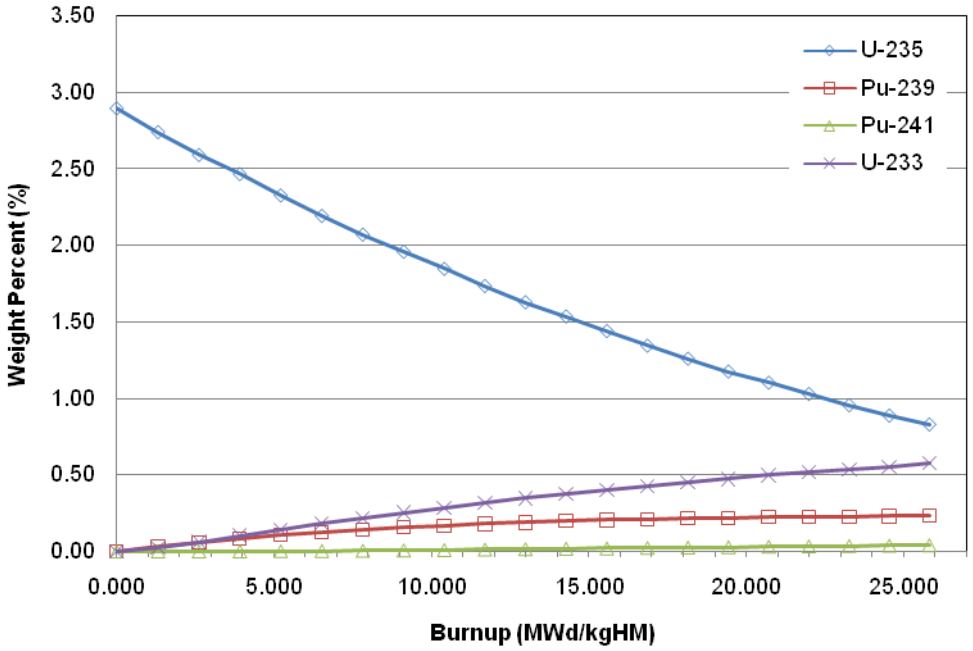


Figure 6. 1 Fissile Isotopes as a Function of Burnup in CANDU-6 (2.6% ^{235}U+50% ^{232}Th)

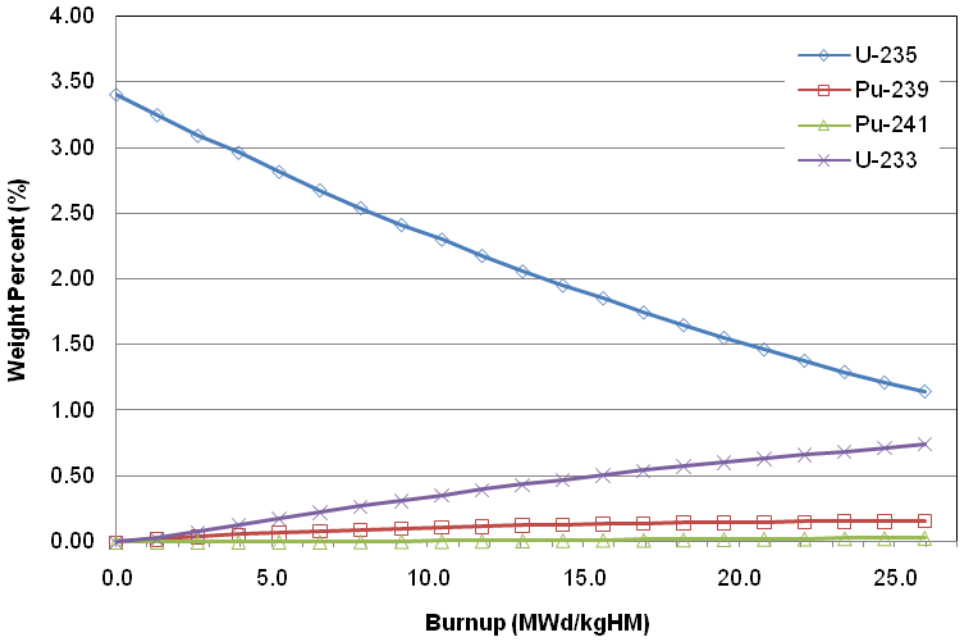


Figure 6. 2 Fissile Isotopes as a Function of Burnup in CANDU-6 (2.6% ^{235}U+70% ^{232}Th)

For fixed U-235 fractions, change of U-233 fraction in total U in SF with Th fraction of fresh fuel is plotted in Figure 6. 3. Fraction of U-233 goes up as high as 14 percent when both Th and U-235 fractions are sufficiently high.

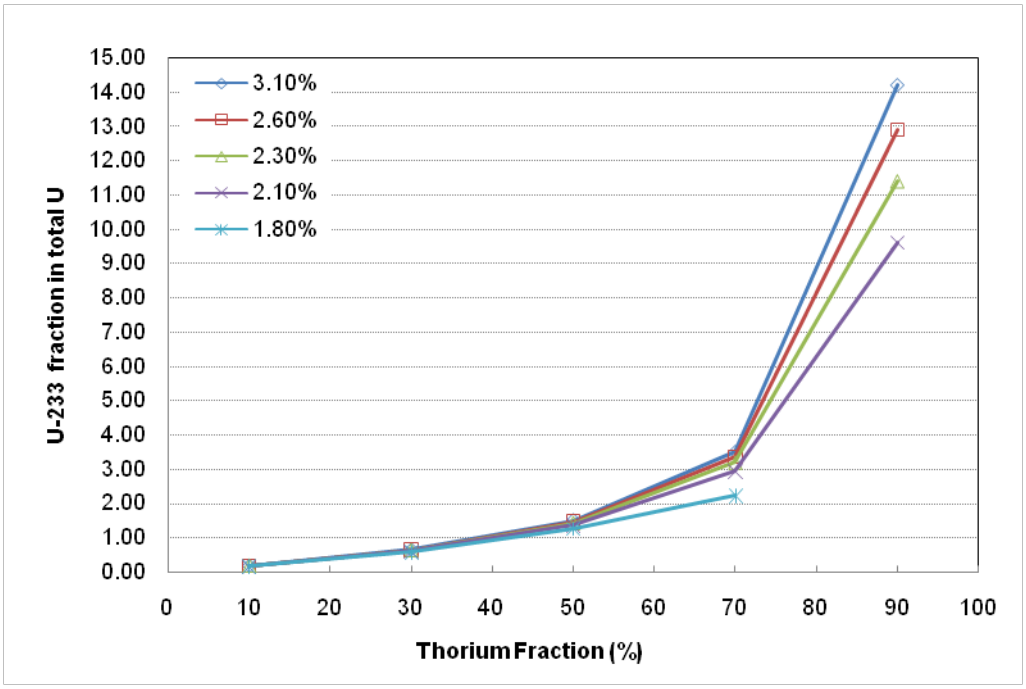


Figure 6. 3 U-233 Fissile Fraction as a Function of Th Fraction in CANDU-6

6.1.2 Burnup Analysis

As can be observed from Figure 6. 1, since the total fissile content of spent U fuels of CANDU-6 is too low, recycling is not taken into account in case of U fuels.

For (U-Th) fuels, fissile contents given in Table 5.2 are at sufficient levels to consider recycling. Consequently, re-use of U and Pu in SF after blending with an appropriate amount of Th is to be investigated.

Discharge Burnup as a Function of Th Fraction in Recycle Fuel

Results obtained for a discharge burnup of 10 MWd/kgHM from fresh fuels are presented in Table 6. 3. The compositions of fresh fuels given in the leftmost column of Table 6. 3 all yield 10 MWd/kgHM. Fissile contents of SFs resulting from these fresh fuels are mixed with various amounts of Th (as specified on the right column of the table), and each Th fraction yields the discharge burnup indicated in the shaded rows.

Table 6. 3 Discharge Burnup of Recycle Fuels

Fresh Fuel	Fuel at first recycle					
10 % Th 0.95 % U-235	Added Th Fraction (%)	0				
	Burnup	9.135				
30 % Th 1.206 % U-235	Added Th Fraction (%)	0	5	10	15	20
	Burnup	25.800	23.030	19.620	15.720	10.930
50 % Th 1.46 % U-235	Added Th Fraction (%)	30	35	40	42.5	45
	Burnup	28.970	23.415	16.815	12.890	8.605
70 % Th 1.71 % U-235	Added Th Fraction (%)	60	62.5	65	66	
	Burnup	28.165	22.100	15.755	12.360	
90 % Th 1.92 % U-235	Added Th Fraction (%)	85	87.5	88.5		
	Burnup	47.470	23.270	13.860		

Discharge burnup of recycle (U-Th) fuel can be correlated to the Th fraction (ω) by the following expressions.

$$\begin{aligned} B_d(\omega) &= 0.741(30-\omega)+4.201 \quad for \quad \theta = 30\% \quad \text{and} \quad 0<\omega<20 \\ B_d(\omega) &= 1.348(50-\omega)+2.633 \quad for \quad \theta = 50\% \quad \text{and} \quad 30<\omega<45 \\ B_d(\omega) &= 2.481(70-\omega)+3.393 \quad for \quad \theta = 70\% \quad \text{and} \quad 60<\omega<66 \end{aligned} \qquad \text{(Eq. 5.1)}$$

where θ is the Th percentage in fresh fuel and ω is in recycle fuel.

Note that there is a certain value of ω (always lower than θ) above which it is impossible to obtain a significant burnup from recycle fuel without adding new fissile material.

The recycling scheme, based on the values shown in Table 6. 3, is demonstrated in Figure 6. 4. In the scheme, as an example case, fresh (U-Th) fuel [1.46%U-235 and 50% Th] is burned to 10 MWd/kgHM in CANDU-6. The SF is reprocessed, and U (containing U-233) and Pu are recovered. Recovered U and Pu are mixed with appropriate amounts of Th to be able to accumulate burnup from recycle-1 fuel. Th fractions and resulting burnups of recycle-1 fuel are shown in the middle column of Figure 6.4.

Similarly, in order to obtain burnup from recycle-2 fuel, a further reduction in Th fraction is required. For a few selected Th fractions in recycle-2 fuels, discharge burnups that can be obtained are given in the rightmost column of Figure 6. 4.

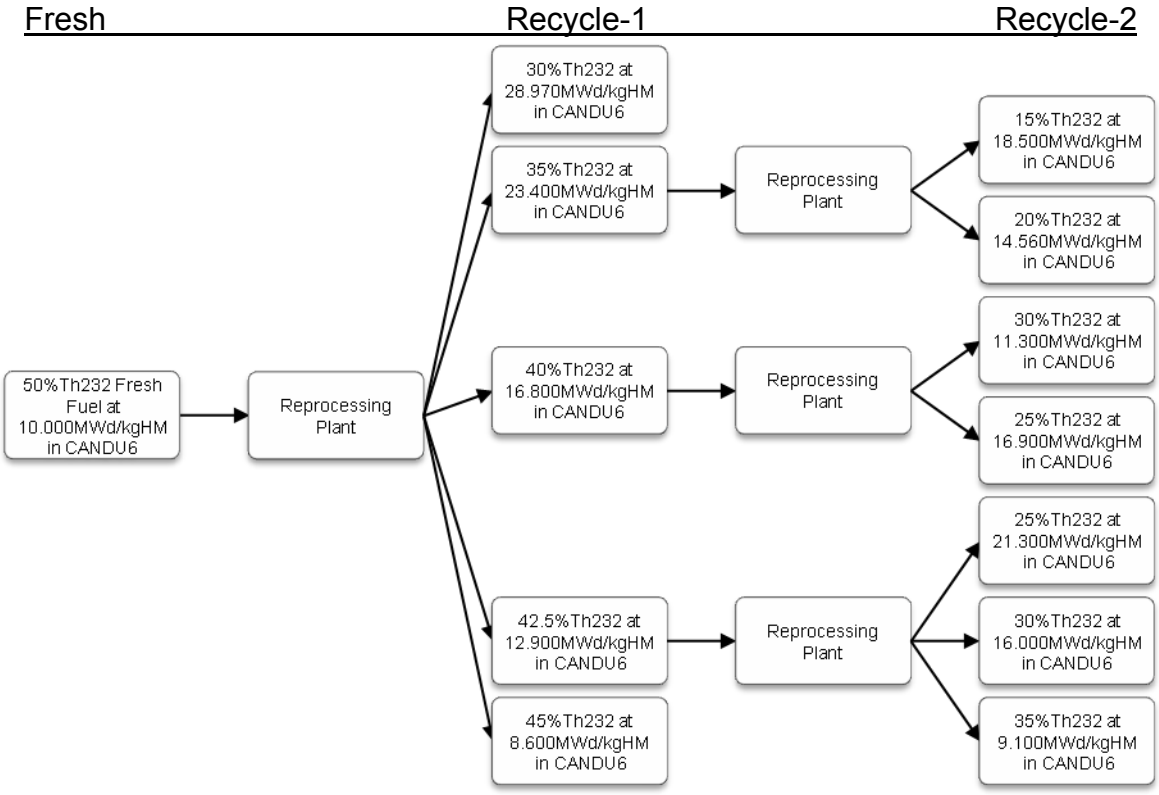


Figure 6. 4 A Schematic Presentation of Recycling

6.1.3 Multiple Recycling

The scheme described in Figure 6. 4 can be extended to multiple recycling. Figure 6. 5 shows multiple recycling for selected initial burnups of 10 MWd/kgHM (with 1.46% U-235+ 50% Th-232), 15 MWd/kgHM (with 1.55% U-235+ 50% Th-232) and 20 MWd/kgHM (with 1.65% U-235+ 50% Th-232). As can be observed from the figure, it is necessary to reduce the Th fraction gradually in order to reach the same burnup value in each recycle step. In the last step of each burnup case, since Th fraction has already been reduced to zero, the same (constant) burnup cannot be reached; instead, whatever is available is burned.

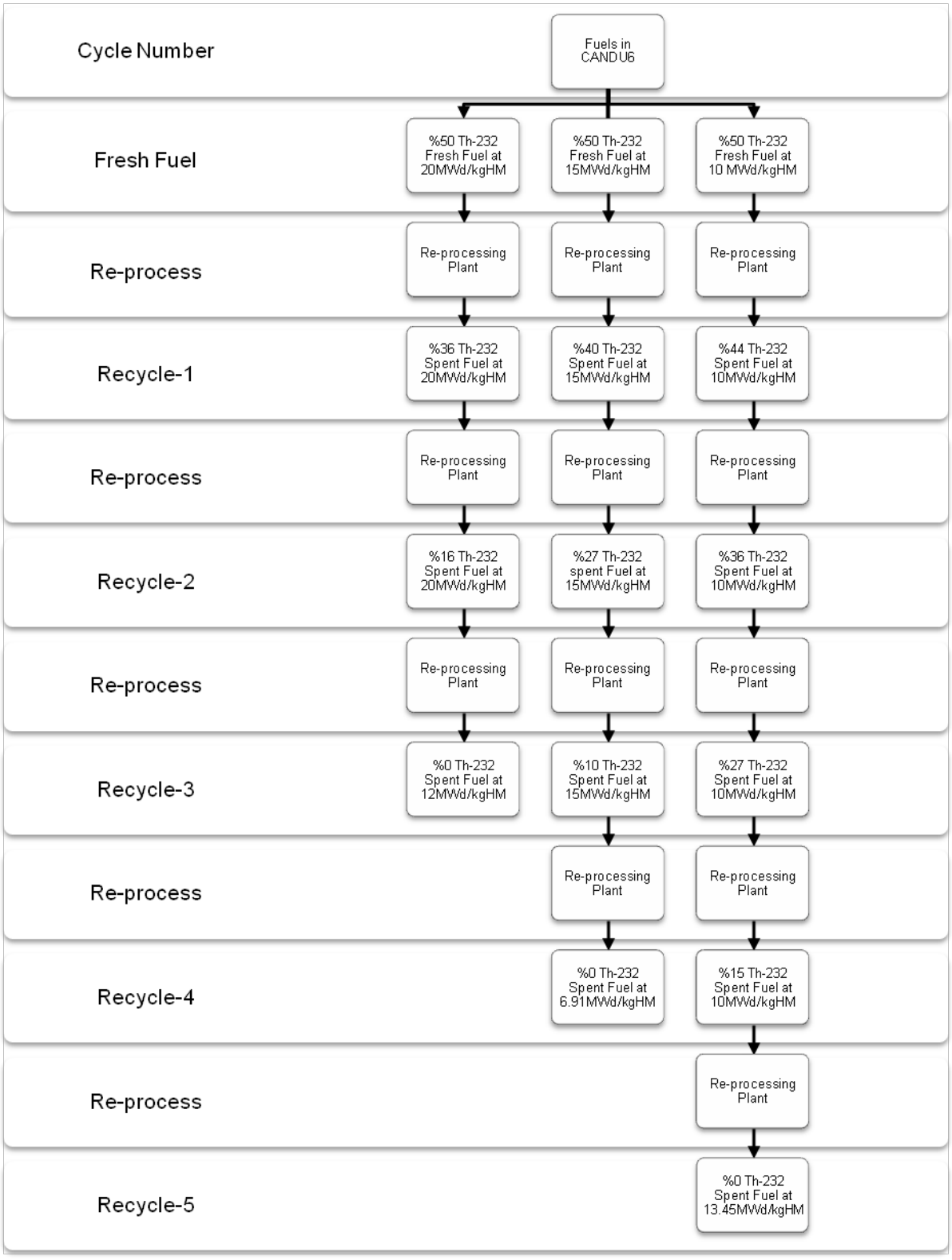


Figure 6. 5 Multiple Recycling for 10, 15 and 20 MWd/kgHM Burnup

Change of Spent Fuel Composition during Multiple Recycling

For selected initial burnups of 10 MWd/kgHM, Figure 6. 6 shows the change of amount of fissile isotopes in gram in SF with recycle number, displayed in Figure 6. 5. U-233 content of SF increases in the first two stages of recycling and then decreases due to Th reduction. For the three selected cases, mentioned in Section 6.1.3, change of total fissile content in SF with recycle number is plotted in Figure 6. 7. In the following figures, the beginning of "recycle number", shown as "0", is the "fresh fuel" demonstrated in Figure 6. 5. In the last recycle of all cases, total fissile content of SF becomes less than that of fresh NU. Appendix E includes a list of the composition of SF for selected burnups.

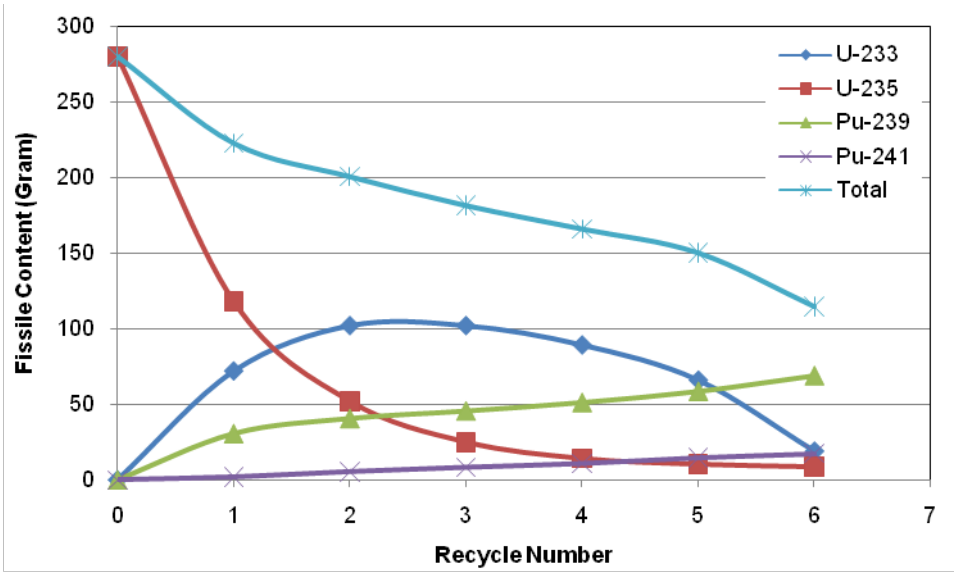


Figure 6. 6 Fissile Content as a Function of Recycle Number

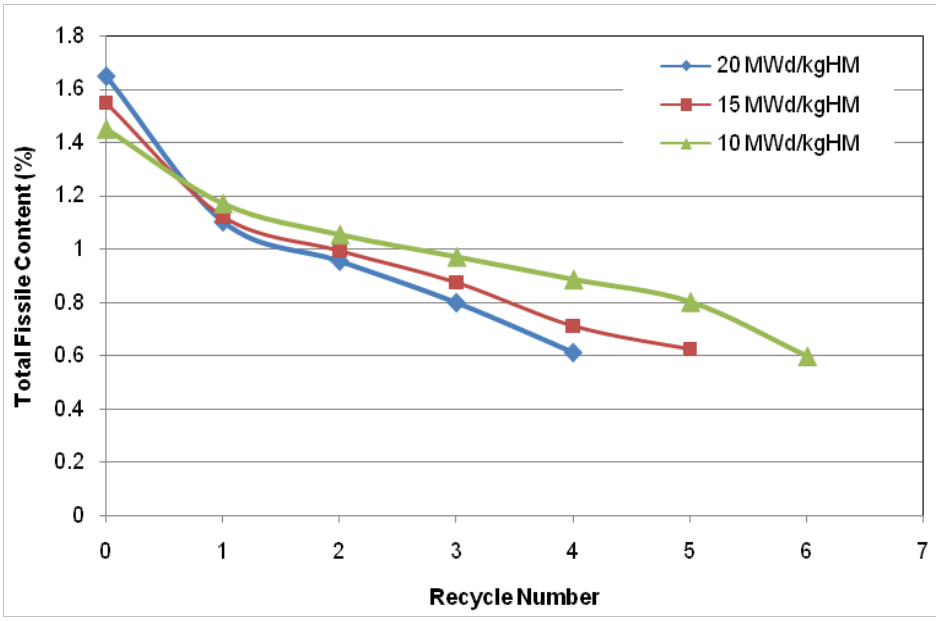


Figure 6. 7 Total Fissile Content as a Function of Recycle Number

6.2 RECYCLING OF U AND Pu CONTENTS OF SPENT ACR FUELS INTO CANDU-6

6.2.1 Composition of Spent ACR Fuel

U Fuel

For some selected enrichments of fresh U fuel, weight percents of important isotopes in SF are tabulated in Table 6. 4.

Table 6. 4 A Summary Table of Isotopic Compositions in Spent U Fuels

Isotopes	***Fresh Fuel Enrichment (%)***						
	1.7	**1.9**	**2.1**	**2.3**	**2.5**	**2.9**	**3.1**
^{235}U	0.693	0.574	0.526	0.487	0.450	0.393	0.356
^{236}U	0.160	0.210	0.249	0.287	0.324	0.395	0.432
^{238}U	98.683	98.653	98.604	98.555	98.512	98.429	98.392
Total U	**99.536**	**99.437**	**99.379**	**99.329**	**99.286**	**99.217**	**99.18**
^{239}Pu	0.324	0.356	0.371	0.384	0.391	0.401	0.404
^{240}Pu	0.100	0.140	0.165	0.185	0.203	0.232	0.246
^{241}Pu	0.032	0.049	0.059	0.068	0.075	0.088	0.093
^{242}Pu	0.007	0.015	0.022	0.030	0.037	0.053	0.062
Total Pu	**0.464**	**0.562**	**0.621**	**0.672**	**0.713**	**0.785**	**0.818**
Fissile Pu (%)	**0.356**	**0.405**	**0.430**	**0.452**	**0.466**	**0.489**	**0.497**
Total Fissile (%)	**1.049**	**0.979**	**0.956**	**0.939**	**0.916**	**0.882**	**0.853**

For 2.1% SEU, changes of fractions of fissile isotopes during irradiation are shown in Figure 6. 8.

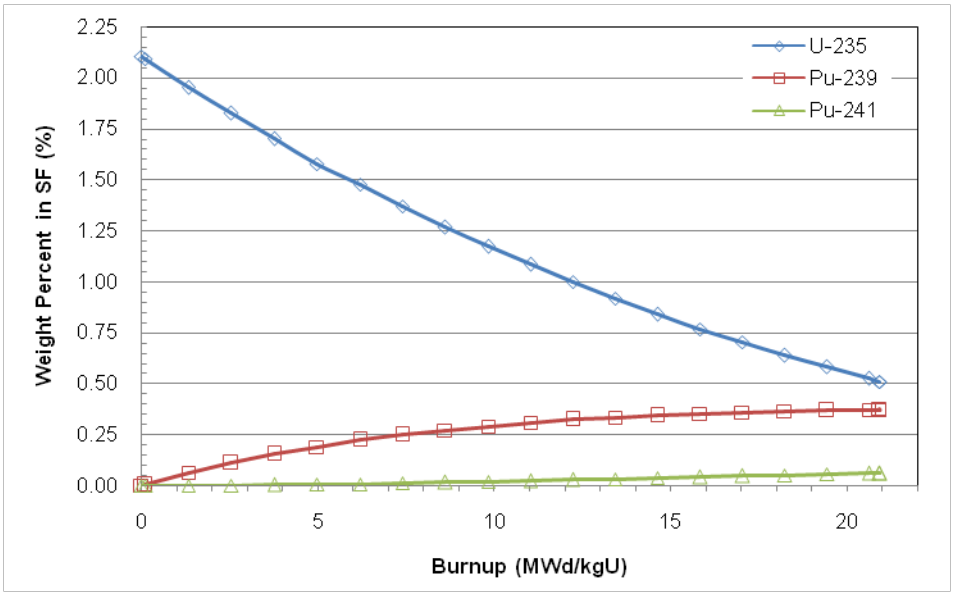


Figure 6. 8 Fissile Materials as a Function of Burnup in ACR

(U-Th) Fuel

For a few selected compositions of fresh (U-Th) fuel, weight percents of important isotopes in SF are tabulated in Table 6. 5.

Table 6. 5 A Summary Table of Isotopic Compositions in Spent (U-Th) Fuels

Isotopes	***30% Th***			***50% Th***			***70% Th***		
U-235 Fraction (%)	**2.9**	**3.4**	**4.0**	**2.9**	**3.4**	**4.6**	**3.4**	**4.6**	**5.4**
^{233}U	0.382	0.426	0.467	0.499	0.614	0.753	0.740	0.979	1.055
^{234}U	0.044	0.059	0.081	0.044	0.072	0.114	0.072	0.163	0.177
^{235}U	0.752	0.660	0.511	1.102	0.870	0.618	1.142	0.655	0.501
^{236}U	0.349	0.446	0.566	0.298	0.418	0.665	0.379	0.658	0.812
^{238}U	67.771	67.661	67.422	47.468	47.166	46.317	26.806	25.811	25.025
Total U	**69.299**	**69.252**	**69.048**	**49.409**	**49.140**	**48.467**	**29.140**	**28.266**	**27.570**
^{239}Pu	0.305	0.319	0.331	0.225	0.244	0.288	0.160	0.170	0.196
^{240}Pu	0.130	0.155	0.184	0.072	0.101	0.153	0.054	0.081	0.099
^{241}Pu	0.056	0.069	0.083	0.033	0.049	0.075	0.030	0.048	0.058
^{242}Pu	0.021	0.033	0.054	0.009	0.020	0.048	0.009	0.031	0.044
Total Pu	**0.517**	**0.587**	**0.670**	**0.341**	**0.421**	**0.567**	**0.257**	**0.349**	**0.400**
Total Th	30.184	30.161	30.282	50.250	50.438	50.966	70.603	71.385	72.029
Fissile U (%)	**1.134**	**1.085**	**0.899**	**1.600**	**1.484**	**1.371**	**1.882**	**1.641**	**1.555**
Fissile Pu (%)	**0.361**	**0.389**	**0.414**	**0.258**	**0.294**	**0.363**	**0.190**	**0.212**	**0.255**
Total Fissile (%)	**1.495**	**1.474**	**1.393**	**1.858**	**1.778**	**1.734**	**2.072**	**1.853**	**1.810**

For two selected fresh fuel compositions (2.9% U-235+50% Th-232 and 3.4% U-235+70% Th-232), changes of fractions of fissile isotopes during irradiation are shown in Figure 6. 9 and Figure 6. 10. Presence of Th in fresh fuel results in production of U-233; but it has a negative influence on the production of fissile Pu.

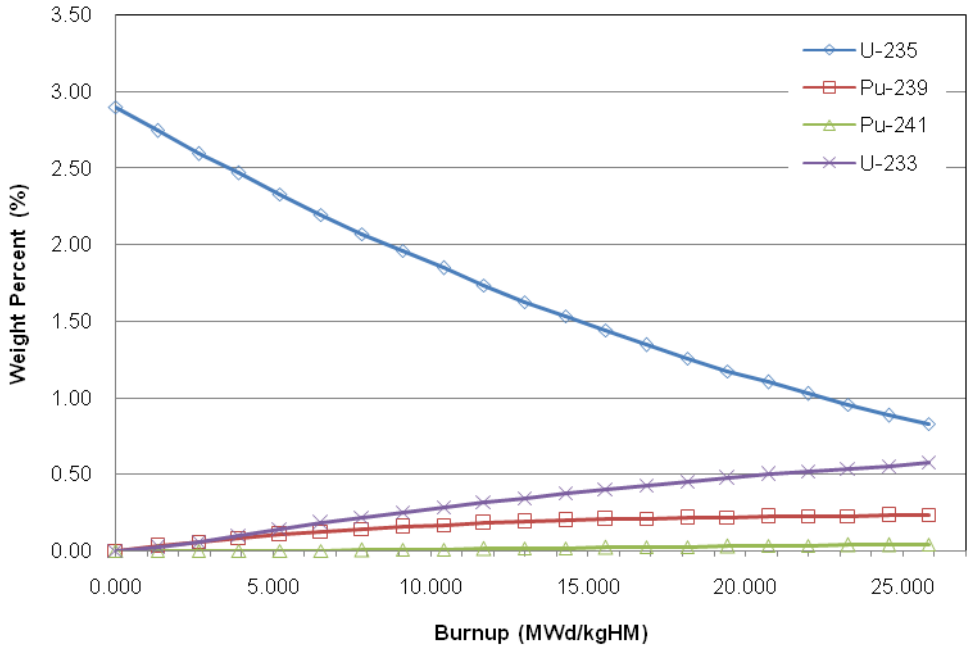


Figure 6. 9 Fissile Isotopes as a Function of Burnup in ACR (2.9% ^{235}U+50% ^{232}Th)

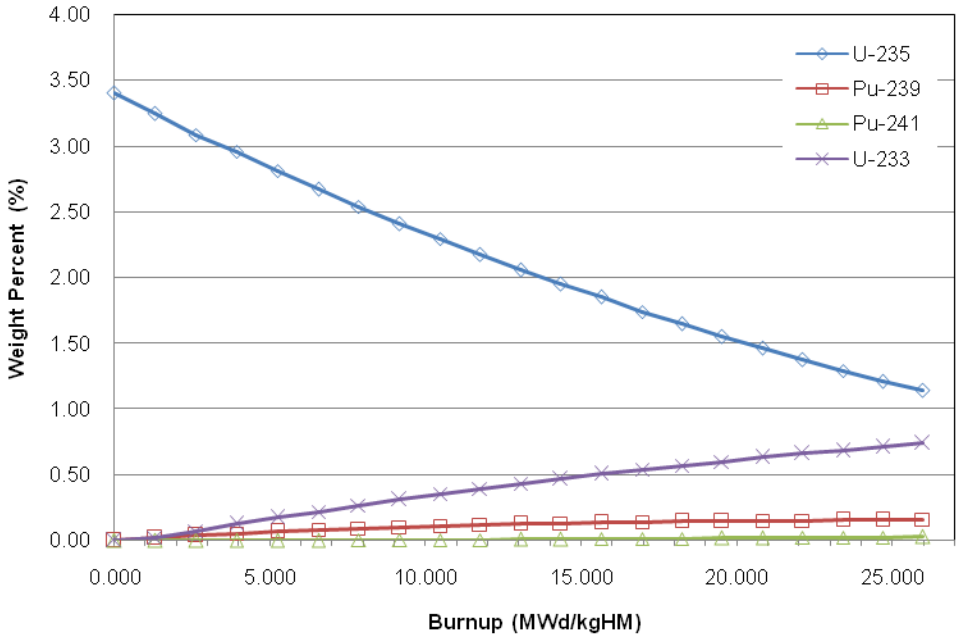


Figure 6. 10 Fissile Isotopes as a Function of Burnup in ACR (3.4% ^{235}U+70% ^{232}Th)

6.2.2 Burnup Analysis

In this section, discharge burnups resulting from recycling of U and Pu contents of spent ACR fuels into CANDU-6 are to be determined.

Discharge Burnups for Fresh 2.1 % SEU of ACR

For the selected case of 2.1 % SEU initially loaded into ACR, discharge burnups obtained from ACR and CANDU-6 are presented in Table 6. 6.

Table 6. 6 Discharge Burnups for 2.1 % SEU Fuel in ACR+CANDU-6

	Fresh ACR Fuel	Recycle CANDU-6 Fuel	Total
Discharge Burnup (MWd/kgU)	20.500	13.640	34.140

Total fissile content in SF of CANDU-6 is 0.5 %, and U-235 fraction is 0.09 %.

Discharge Burnups for (U-Th) Fuels of ACR

For a few selected compositions of fresh (U-Th) fuels in ACR, compositions of recycle fuels are determined and their discharge burnups that can be obtained in CANDU-6 are calculated. Results are summarized in Table 6. 7.

Table 6. 7 Discharge Burnups for (U-Th) Fuels in ACR+CANDU-6 Reactor

Fresh ACR Fuel	Discharge Burnup	Recycle CANDU-6 Fuel	Discharge Burnup	Total Irradiation
2.5%^{235}U+10%^{232}Th	24.475	1.19%total+10%^{232}Th	18.220	42.690
2.9%^{235}U+30%^{232}Th	26.580	1.50%total+30%^{232}Th	22.535	49.120
2.9%^{235}U+50%^{232}Th	20.510	1.85%total+50%^{232}Th	29.045	49.555
3.4%^{235}U+70%^{232}Th	26.720	2.00%total+70%^{232}Th	31.745	58.465
3.4%^{235}U+90%^{232}Th	23.135	2.20%total+90%^{232}Th	31.840	54.970

CHAPTER 7. CONCLUSION

7.1 SUMMARY OF RELATIONS (DISCHARGE BURNUP VS. FUEL COMPOSITION)

For U fuels, the expression relating discharge burnup to enrichment is in the general form:

$$B_d = K_o + K_1\varepsilon + K_2\varepsilon^2 \qquad \textbf{[Eq.7. 1]}$$

where ε is enrichment in weight percent; and K_o, K_1 and K_2 are constants with unit of kg/MWd given in Table 7. 1.

Table 7. 1 Discharge Burnup Curve Fit Parameters

	K_o	K_1	K_2
ACR	-21.96	22.17	-0.982
CANDU-6	-9.876	22.62	0

For (U-Th) fuels, the expression relating discharge burnup to U-235 and Th fractions is in the general form:

$$B_d(\varepsilon,\theta) = (B_{o1} + A_1\theta + C_1\theta^2)\varepsilon^2 + (B_{o2} + A_2\theta + C_2\theta^2)\varepsilon + (B_{o3} + A_3\theta + C_3\theta^2) \qquad \textbf{[Eq.7.5]}$$

where ε is U-235 fraction in weight percent, θ is Th fraction in weight percent; and B_{o1}, B_{o2}, B_{o3}, A_1, A_2 , A_3 , C_1, C_2 and C_3 are constants with unit of MWd/kg, given in Table 7. 2. The relation for ACR is valid for $1.7 < \varepsilon < 4.6$ and $10 < \theta < 90$, with ±5.4% computational error. The relation for CANDU-6 is valid for $1 < \varepsilon < 3.1$ and $10 < \theta < 90$, with ±3.5% computational error.

Table 7. 2 Discharge Burnup Curve Fit Parameters

	A_1	A_2	A_3	B_{o1}	B_{o2}	B_{o3}	C_1	C_2	C_3
ACR	0	-0.074	0.481	0	17.35	-17.50	0	0	0
CANDU-6	-0.192	0.966	-1.161	-1.689	27.93	-8.699	3.289	9.250	-6.491

7.2 CONCLUDING REMARKS

About (U-Th) fuels

- ✓ As Th fraction in (U-Th) fuels increases, a higher U-235 fraction is required to accumulate the same discharge burnup because Th-232 has more than twice the thermal absorption cross-section of U-238.
- ✓ A higher Th fraction in fuel load directly results in a lower SEU requirement; however, despite this lower SEU requirement, NU requirement increases with Th fraction. The higher NU requirement that is caused by the increased enrichment of SEU outweighs the effect of the reduction in SEU need, and the net result becomes an increased NU requirement (refer to Figure 5. 2 and Figure 5. 6). At high burnups, NU requirements for all fuel compositions approach the same level.
- ✓ Similarly, and mainly due to the same reasons, Th fraction has a negative effect on nuclear resource utilization and NU savings; they both decrease as Th fraction increases. In general, high burnups cause the differences to narrow down.
- ✓ Conversion ratio of (U-Th) fuels is significantly greater than that of U fuels. In (U-Th) fuels, as Th fraction goes up, conversion ratio also goes up.

About ACR and CANDU-6

- ✓ CANDU-6 has higher conversion ratios than ACR because ACR uses light water as coolant; in other words, CANDU-6 provides more neutron economy by converting more fertile atoms to fissile ones.
- ✓ For a selected Th fraction of 50 %, change of fuel requirement with discharge burnup is shown in Figure 7. 1. As seen, fuel requirements of ACR and CANDU-6 are almost the same for the same burnup.

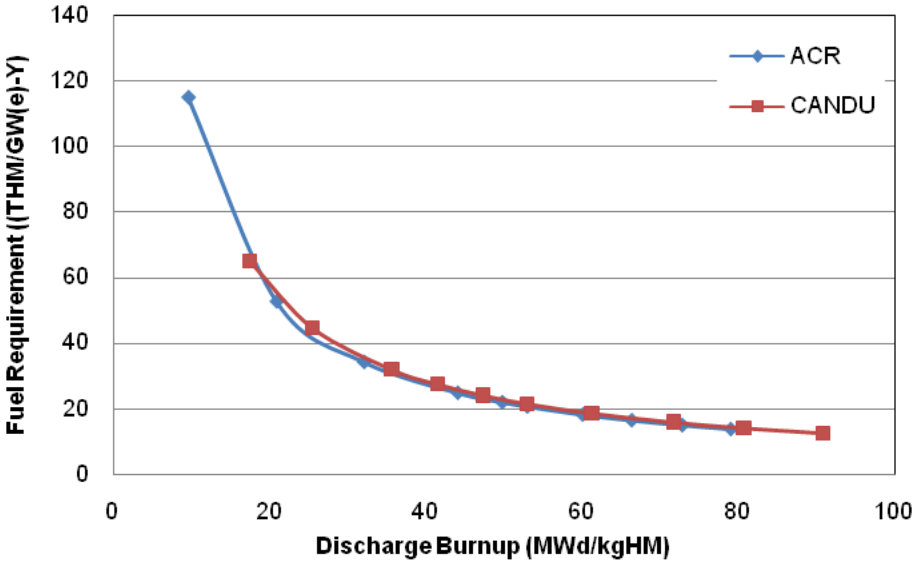


Figure 7. 1 Fuel Requirement as a Function Discharge Burnup [50% Th Fraction]

- ✓ In both ACR and CANDU-6, as discharge burnup increases, NU requirement first decreases sharply, and then levels off (see Figure 5. 2 and Figure 5. 6). Since a higher enrichment is required for a higher discharge burnup, then, use of SEU fuels with higher enrichments in both CANDU-6 and ACR reduces NU requirements significantly down to a point, the level-off point (in Figure 5. 2 and Figure 5. 6); a further increase in enrichment beyond that point would not have a meaningful effect on NU requirement. In CANDU-6, NU requirement for SEU at the level-off point is 33% less than that for NU fuel; and for (U-Th) fuels, it is about 27% less. In ACR, compared to SEU with 2.1% U-235, NU requirement can be reduced by nearly 17% at the level-off point, for both SEU and (U-Th) fuels.
- ✓ For a selected Th fraction of 50 %, change of NU requirement with discharge burnup is shown in Figure 7. 2. As seen, for the same burnup, ACR requires more NU than does CANDU-6.

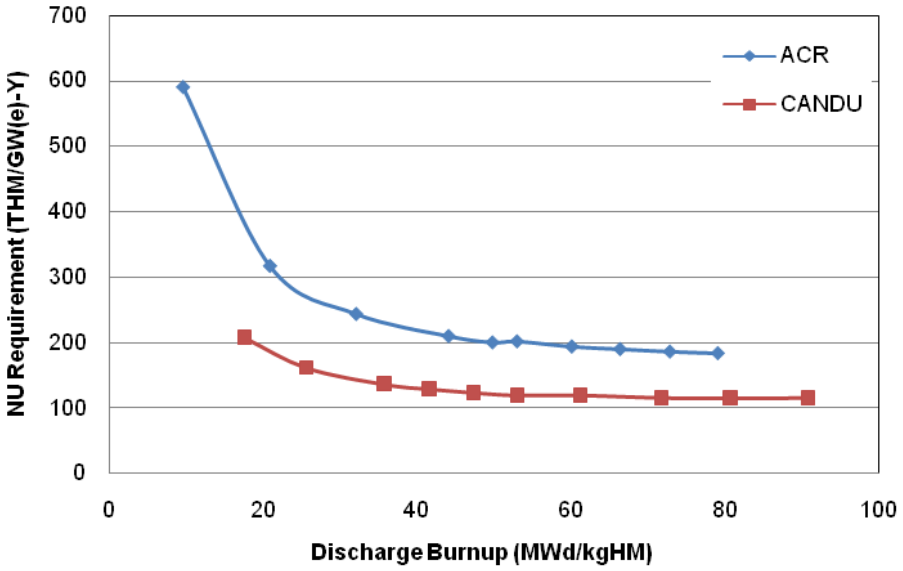


Figure 7. 2 NU Requirement as a Function of Discharge Burnup [50% Th Fraction]

✓ For a selected Th fraction of 50 %, change of nuclear resource utilization with discharge burnup is shown in Figure 7. 3. For the same burnup, utilization for CANDU-6 is significantly greater than that for ACR. The same trend can be observed in all cases studied. NRU values approach about 10 MWd/kgHM for CANDU-6 and 6 MWd/kgHM for ACR as discharge burnup goes up.

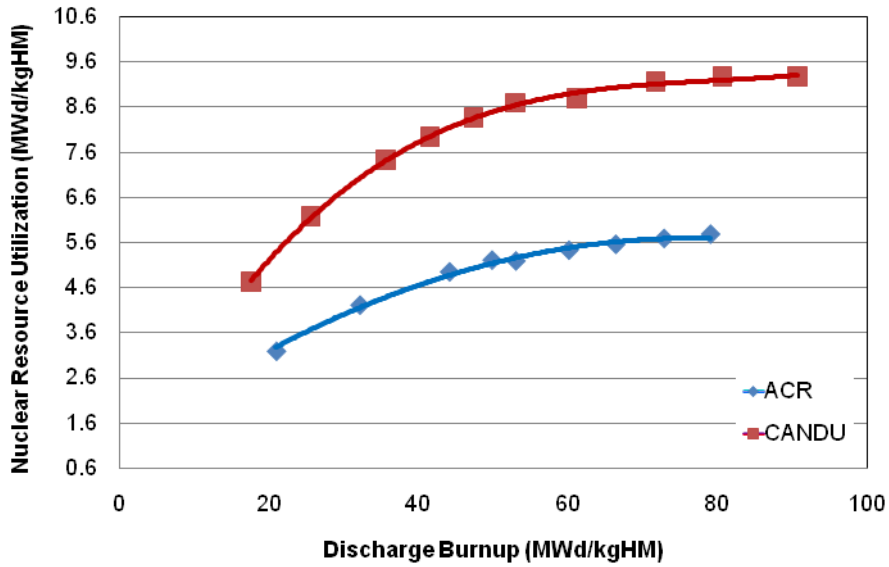


Figure 7. 3 Resource Utilization as a Function of Discharge Burnup [50% Th]

- ✓ CANDU-6 seems to be superior to ACR from respects considered above; however, almost all of those respects become more favorable as discharge burnup increases, and ACR has a greater ability to reach higher burnups. Then, to sum up, CANDU-6 is more suitable lower-burnup U[†] and (U-Th) fuels while ACR[‡] is attractive for higher-burnup SEU[§] and (U-Th) fuels. Also note that ACR is a GEN III+ reactor, and the advancements in its design makes it more readily adaptable to usage of Th.

Re-use of U and Pu in SF

- ✓ Spent U fuels of CANDU-6 do not contain sufficient fissile material to consider recycling.
- ✓ In case of spent (U-Th) fuels of CANDU-6, due to considerably higher U-235 fractions in fresh fuels and U-233 fractions in SF emerging from Th-232, U and Pu in SF contain sufficient amounts of fissile isotopes, and can be re-used in CANDU-6 without adding any new fissile material, provided that the SF is reprocessed to get rid of fission products and parasitic actinides. Such a re-use scheme can lead to multiple recycling, by simply lowering the Th fraction gradually in each recycle fuel (see Figure 6. 4 and Figure 6. 5).
- ✓ By re-burning U and Pu content of spent ACR fuels in CANDU-6, significant additional burnups can also be achieved (see Table 6. 6 and Table 6. 7).

[†] According to results of AECL Chalk River National Laboratory, the highest enrichment for CANDU-6 is about 2% tested in experiments and 1.2% in practice. [18]

[‡] Previous CANFLEX irradiations in the experimental loops of the NRU reactor in Chalk River have demonstrated that the CANFLEX bundle can successfully perform at 1200 kW (≈65,600MWd/kgHM). The time-average peak bundle power of about 851 kW (≈46,500MWd/kgHM) for ACR fuel is well below this performance limit. With power ripple, the peak bundle power in ACR is expected to remain below about 910 kW (≈50,000MWd/kgHM), similar to the license limits for 37 element fuel bundles in other CANDU reactors. [9] The discharge burnup of the fuel is also subject to constraint. Although some developments in the fuel rods and assemblies have been exposed to burnups in excess of 100 MWd/kgU, current experience for CANDU-6 reactor is limited to average discharge burnups of 40 to 50 MWd/kgU, and irradiation of these values implies a higher risk failure. In addition to being a potential constraint in terms of fuel failure probability, increasing the discharge fuel burnup has an impact on power distribution.

[§] Experimental studies carried out are up to 5% for ACR. [22]

- ✓ Problems associated with and conditions for economic justification of any recycling (or multiple recycling), which are not addressed in this study, seem to be a proper subject for a follow-up.

APPENDIX

APPENDIX A. BENCHMARK SPECIFICATIONS OF REACTORS

A.1. ACR

ACR-700 benchmark specifications: [9][10][23]

Table A. 1 Core

Parameters	**Value**
Type	PTR
Thermal Power (MWth)	1982
Reactor Presure (MPa)	12.6
Core Lenght (mm)	5940
Number of bundles per fuel channel	12
Number of Fuel Channels (PTs)	284
Pressure-Tube inner radius (mm)	51.689
Pressure-Tube outer radius (mm)	58.169
Number of fuel elements per channel	43
Calendria Diameter (m)	5.2
Reflector Thickness(mm)	480
Exposure	1094days
Total delayed neutron fraction (β)	0.0056
Prompt neutron life time (ms)	0.33

Table A. 2 Fuel

Lattice Pitch (mm)	220
Fuel	Sintered pellets of slightly enriched UO_2 with 2.1% U-235 in 42 pins & Natural UO_2 in central element with burnable posion $(U,Dy)O_2$ pellet with 7.5 % Dysprosium
Fuel density (g/cm^3)	10.44/10.36
Enrichment Level	Average 2.1 % U-235 in 42 elemnets, central element NU with Dysprosium
Fuel Burn-up	20500-24000 (MWd/teU)
Max. Fuel Element Bunup	26 (MWd/teU)
Fuel pin outer diameter	Central pin and inner ring of seven elements of 13.5 mm & the outer two rings consist of 35 elements with 11.5 mm diameter
Fuel bundle assembly	43 element CANFLEX
Lenght of Fuel Bundle (mm)	495.3
Bundle weight (kg)	22.7 (includes 18 kg U)

Table A. 3 Coolant

Coolant	Pressurized Light Water (H_2O)
Atom Purity	99.75%
Density (g/cm^3)	1.8360
Temperature(^{o}C)	300
Coolant inlet temperature (^{o}C)	279
Coolant outlet temperature (^{o}C)	325

Table A. 4 Moderator

Moderator (and reflector)	Heavy Water (D_2O)
Atom Purity	99.85%
Density (g/cm^3)	1.0829
Temperature(^{o}C)	80

Table A. 5 Material

Annulus Gas	CO_2
Zr-2.5Nb (%)	Zr (97.5%) & Nb (2.5%)
Zircaloy-2 (%)	Zr (98.225%) & Sn (1.5%) & Fe (0.175%) & Cr (0.10%)
Zircaloy-4 (%)	Zr (98.23%) & Sn (1.45%) & Fe (0.21%) & Cr (0.11%) & Hf (0.01%)
Fuel Channels	Horizonral Zr 2.5% Nb Alloy with modified 403 SS end fittings

A.2. CANDU-6

CANDU-600 benchmark specifications: [6][24]

Table A. 6 Core

Parameters	***Value***
Type	PTR
Thermal Power(Mh)	2061.4
Operating Pressure (Mpa)	11.1
Core Lenght (cm)	594.4
Number of bundles per fuel channel	12
Number of Fuel Channels(PTs)	380
Pressure-Tube (2.5-2.5% Nb) inner diameter (cm)	10.3378
Average pressure tube wall thickness (cm)	0.4343
Calandria Tube (Zr-2) Inside diameter (cm)	12.8956
Average calandria tube wall thickness (cm)	0.1397

Calandria Diameter (cm)	760
Reflector Thickness(cm)	65.6
Reactor Core Radius[1]	314.3
Inner Radius of Main Calandria (cm)	379.7
Inner Radius of Subcalandria (cm)	337.8
Lenght of Calandria Notch (cm)	96.5
Exposure Time (days)	320
Extrapolated[2] lenght of fuel channel (cm)	606.0
Extapolated[2] reactor radius (cm)	384.7

Table A. 7 Fuel

Lattice Pitch (cm)	28.6 (square)
Fuel	Sintered Pellets of Natural UO_2
Fuel density (g/cm^3)	10.6
Enrichment Level (Initial weight percent)	NU
^{234}U	0.0054
^{235}U	0.7110
^{238}U	99.2836
Fuel Burn-up (MWd/teU)[3]	7154.1
Fuel pin outer diameter (cm)	1.310
Fuel pellet diameter (cm)	1.220
Fuel bundle assembly	37-element[4]
Lenght of Fuel Bundle (cm)	48.20
UO2 weight per bundle (kg)	21.67
U weight per bundle (kg)	19.10
Zr weight per bundle (kg)	2.206
Cluster Average exit $^{235}U/U$	0.213
Fuel Temperature (K)	1155

Table A. 8 Coolant

Coolant	D_2O
Atom Purity (nominal)	99.10 %
Density (g/cm^3)	1.8360
Coolant inlet temperature (^oC)	266
Coolant outlet temperature (^oC)	310
Temperature(^oC)	288

Table A. 9 Moderator

Moderator (and reflector)	D_2O
Atom Purity	99.85 %
Density (g/cm^3)	1.0829
Temperature(^oC)	69

Table A. 10 Material

Annulus Gas	CO2
Fuel Channels	Horizonral Zr 2.5% Nb Alloy with modified 403 SS end fittings
Zr-2.5Nb (%)	Zr (97.5%) & Nb (2.5%)
Zircaloy-2 (%)	Zr (98.225%) & Sn (1.5%) & Fe (0.175%) & Cr (0.10%)
Zircaloy-4 (%)	Zr (98.23%) & Sn (1.45%) & Fe (0.21%) & Cr (0.11%) & Hf (0.01%)

[1]This is given by π * (core radius)^2 = 380 * (Pitch)^2
[2]Extrapolated boundaries are introduced for the purposes of core diffusion calculations.
[3]Discharge burnup is a strong function of moderator isotopic purity, which is assumed to be 99.85 atom % in this calculation.
[4]Arranged in concentric rings of 1, 6, 12 and 18 elements having pitch circle diameters of 0 cm, 2.9769 cm, 5.7506 cm and 8.6614 cm, respectively.

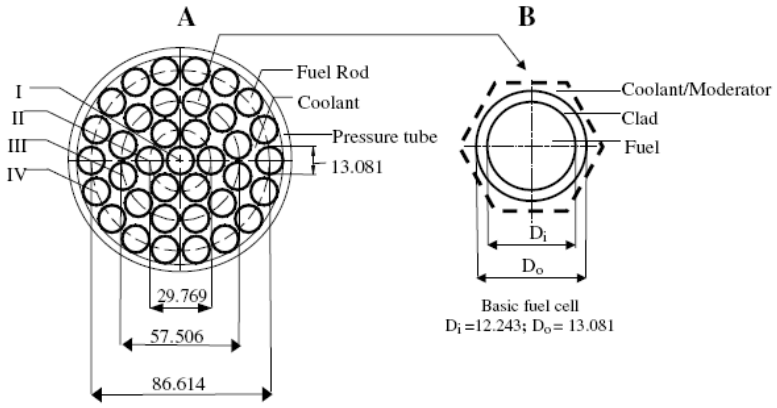


Figure A. 1 Placement of 37 Fuel Rods in the Bundle [4]

(Dimensions are in mm, not in scale)

APPENDIX B. ISOTOPIC DISTIRIBUTION OF CORE POWER

B.1. ACR

Table B. 1 Power Fractions of U and Pu Isotopes in Total Core Power

ACR					
Bundle					
	U235		U238		Pu
Enrichment (%)	**(%)**	**cap to fis**	**(%)**	**cap to fis**	**(%)**
1.70	68.79	0.191	3.21	13.076	28.00
1.90	63.93	0.193	4.72	12.718	31.35
2.10	60.65	0.194	5.69	12.377	33.66
2.30	59.87	0.195	5.09	12.127	35.03
2.50	58.36	0.196	4.53	11.913	37.11
2.90	57.89	0.197	5.01	11.547	37.10
3.10	57.40	0.198	5.73	11.372	36.87

SEU						
	U235		U238		Pu	
Enrichment	**(%)**	**cap to fis**	**(%)**	**cap to fis**	**(%)**	**Total (%)**
1.70	69.01	0.191	3.05	13.131	27.95	98.41
1.90	64.16	0.192	4.57	12.773	31.27	98.38
2.10	60.88	0.193	5.54	12.432	33.58	98.34
2.30	60.17	0.194	4.92	12.182	34.92	98.34
2.50	58.67	0.195	4.33	11.966	37.00	98.33
2.90	58.24	0.197	4.78	11.598	36.97	98.30
3.10	57.77	0.198	5.54	11.422	36.69	98.29
$(U\text{-}Dy)O_2$						
	U235		U238		Pu	
Enrichment	**(%)**	**cap to fis**	**(%)**	**cap to fis**	**(%)**	**Total (%)**
1.70	55.51	0.207	13.12	9.713	31.37	1.59
1.90	49.85	0.209	13.77	9.359	36.38	1.62
2.10	46.76	0.212	17.34	8.850	35.89	1.66
2.30	42.42	0.214	17.41	8.801	39.98	1.66
2.50	40.69	0.215	15.87	8.599	43.85	1.67
2.90	37.65	0.219	17.99	8.365	44.36	1.70
3.10	36.15	0.220	17.05	8.459	46.80	1.71

B.2. CANDU-6

Table B. 2 Power Fractions of U and Pu Isotopes in Total Core Power

CANDU-6					
Bundle					
	^{235}U		^{238}U		Pu
Enrichment (%)	(%)	cap to fis	(%)	cap to fis	(%)
0.71	55.57	0.178	3.42	19.027	41.00
1.00	43.81	0.179	4.88	17.204	51.31
1.20	41.60	0.179	3.83	16.470	54.58
1.40	40.43	0.180	4.27	15.960	55.30
1.60	39.84	0.180	4.70	15.445	55.46
1.80	39.75	0.180	4.25	15.014	56.00
2.00	39.67	0.181	4.66	14.638	55.66

APPENDIX C. CONVERSION RATIO

C.1. ACR

C.1.1. U Fuels

Table C. 1 Conversion Ratios for ACR U Fuels

Reactor System	Initial Fuel	CR Computed
ACR	1.7% ^{235}U	0.534
	1.9% ^{235}U	0.525
	2.1% ^{235}U	0.515
	2.3% ^{235}U	0.508
	2.5% ^{235}U	0.502
	2.9% ^{235}U	0.489
	3.1% ^{235}U	0.482
	3.5% ^{235}U	0.48
	4.0% ^{235}U	0.475
	4.4% ^{235}U	0.466
	4.8% ^{235}U	0.457
	5.2% ^{235}U	0.443

C.1.2. (U-Th) Fuels

Table C. 2 Conversion Ratios for ACR (U-Th) Fuels

Reactor System	Initial Fuel	CR Computed		
Th Fraction		30%	50%	70%
ACR	2.1% ^{235}U	0.595	-	-
	2.5% ^{235}U	0.57	0.597	-
	2.9% ^{235}U	0.554	0.574	0.59
	3.4% ^{235}U	0.535	0.556	0.572
	4.0% ^{235}U	0.522	0.544	0.556
	4.3% ^{235}U	0.515	0.538	0.551
	4.6% ^{235}U	0.514	0.534	0.546
	5.0% ^{235}U	0.506	0.525	0.538
	5.4% ^{235}U	0.496	0.517	0.529
	5.8% ^{235}U	0.493	0.51	0.517
	6.2% ^{235}U	0.482	0.502	0.51

C.2. CANDU-6

C.2.1. U Fuels

Table C. 3 Conversion Ratios for CANDU-6 U Fuels

Reactor System	Initial Fuel	CR Computed*
CANDU-6	NU	0.771
	0.8% ^{235}U	0.74
	1.0% ^{235}U	0.713
	1.2% ^{235}U	0.696
	1.4% ^{235}U	0.68
	1.6% ^{235}U	0.67
	1.8% ^{235}U	0.656
	2.0% ^{235}U	0.646

C.2.2. (U-Th) Fuels

Table C. 4 Conversion Ratios for CANDU-6 (U-Th) Fuels

Reactor System	Initial Fuel	CR Computed*		
Th Fraction		30%	50%	70%
CANDU-6	1.2 % ^{235}U	0.779	-	-
	1.4 % ^{235}U	0.759	-	-
	1.6 % ^{235}U	0.746	0.78	-
	1.8 % ^{235}U	0.736	0.766	0.785
	2.1 % ^{235}U	0.72	0.75	0.774
	2.3 % ^{235}U	0.71	0.742	0.767
	2.5 % ^{235}U	-	0.733	0.758
	2.7 % ^{235}U	0.693	0.726	0.75
	3.1 % ^{235}U	0.675	0.709	0.735
	3.5 % ^{235}U	0.668	0.7	0.726
	3.9 % ^{235}U	0.655	0.685	0.714
	4.4 % ^{235}U	0.639	0.67	0.689

APPENDIX D. NEUTRON CROSS-SECTION VALUES FOR U AND (U-TH) FUELS

D.1. ACR

Table D. 1 One-Group Neutron Cross-Section Data of SEU and (U-Th) Fuels

Enrichment		**2.1% ^{235}U at 20.500**			
		σ_γ	**σ_f**	**Σ_γ**	**Σ_f**
Th-232	BOC	3.61	0.0189	0	0
	EOC	3.69	0.0195	9.64E-12	5.09E-14
U-233	BOC	11.2	99.8	0	0
	EOC	11.2	101	5.33E-11	4.81E-10
U-234	BOC	31.3	0.453	0	0
	EOC	30.7	0.464	3.78E-07	5.71E-09
U-235	BOC	17.2	89.3	0.008449	0.043866
	EOC	17.8	92.4	0.002141	0.011116
U-236	BOC	9.34	0.273	0	0
	EOC	6.33	0.241	0.000359	1.37E-05
U-238	BOC	0.993	0.0776	0.02251	0.001759
	EOC	1.01	0.0799	0.022511	0.001781
Pu-238	BOC	73.8	3.39	0	0
	EOC	77.8	3.55	6.29E-05	2.87E-06
Pu-239	BOC	97.6	186	0	0
	EOC	87.6	174	0.007313	0.014526
Pu-240	BOC	271	0.477	0	0

	EOC	133	0.461	0.004911	1.7E-05
Pu-241	BOC	77.2	212	0	0
	EOC	75.2	209	0.000997	0.002771
Pu-242	BOC	39.4	0.334	0	0
	EOC	36.6	0.341	0.000181	1.69E-06

2.9% ^{235}U+50% ^{232}Th at 20.500

		σ_γ	σ_f	Σ_γ	Σ_f
Th-232	BOC	1.4	0.0205	0.0163873	0.00024
	EOC	1.44	0.0209	0.0165562	0.0002403
U-233	BOC	9.44	80.8	0	0
	EOC	9.39	82.2	0.0010665	0.0093358
U-234	BOC	29.1	0.47	0	0
	EOC	27	0.474	0.0002674	4.695E-06
U-235	BOC	13.7	68.9	0.009172	0.0461279
	EOC	14.4	72.4	0.0035829	0.0180142
U-236	BOC	9.83	0.293	0	0
	EOC	6.72	0.257	0.0004497	1.72E-05
U-238	BOC	1.1	0.0841	0.0118128	0.0009031
	EOC	1.12	0.0854	0.0118574	0.0009041
Pu-238	BOC	55.6	2.93	0	0
	EOC	59.4	3.05	3.483E-05	1.788E-06
Pu-239	BOC	80.5	150	0	0
	EOC	76.5	146	0.0038203	0.0072911
Pu-240	BOC	270	0.514	0	0
	EOC	157	0.497	0.002504	7.927E-06
Pu-241	BOC	61.3	168	0	0
	EOC	61.5	170	0.0004476	0.0012374
Pu-242	BOC	40.4	0.361	0	0
	EOC	36.6	0.364	6.838E-05	6.8E-07

D.2. CANDU-6

Table D. 2 One-Group Neutron Cross-Section Data for NU and (U-Th) Fuels

		NU at 7.200				***Literature at Mid-burnup [25]***	
		σ_v	σ_f	Σ_v	Σ_f	σ_v	σ_f
U-234	BOC	38.3	0.415	0	0	38.432	0.421
	EOC	35.1	0.428	4.593E-08	5.6E-10		
U-235	BOC	30.5	172	0.005211	0.029387	28.64	159.1
	EOC	27.6	155	0.001435	0.008059		
U-236	BOC	5.80	0.1800	0	0	5.659	0.1075
	EOC	5.48	0.1900	9.592E-05	3.326E-06		
U-238	BOC	1.17	0.0546	0.0274062	0.001279	1.165	0.05424
	EOC	1.11	0.0613	0.025866	0.001428		
Pu-238	BOC	155	5.57	0	0	142.5	5.087
	EOC	140	5.2	1.123E-05	4.172E-07		
Pu-239	BOC	121	272	0	0	123.1	267.3
	EOC	102	236	0.006854	0.0158588		
Pu-240	BOC	224	0.355	0	0	144.5	0.333
	EOC	138	0.370	0.002940	7.882E-06		
Pu-241	BOC	123	347	0	0	115.6	339.4
	EOC	108	306	0.000478	0.001353		
Pu-242	BOC	24.6	0.240	0	0	23.81	0.2517
	EOC	25.8	0.265	2.732E-05	2.806E-07		

		1.46 ^{235}U+50% ^{232}Th at 10.000			
		σ_v	σ_f	Σ_v	Σ_f
Th-232	BOC	1.83	0.0163	0.02192	0.00019
	EOC	1.82	0.0171	0.02153	0.00020
U-233	BOC	11.7	114	0	0
	EOC	11.3	112	0.00101	0.0100
U-234	BOC	30.4	0.425	0	0
	EOC	28.1	0.434	0.00018	2.74E-06
U-235	BOC	20.1	111	0.00694	0.03834
	EOC	20.1	110	0.00292	0.01596

U-236	BOC	6.58	0.216	0	0
	EOC	5.03	0.202	0.00015	6.11E-06
U-238	BOC	1.05	0.0669	0.01189	0.00076
	EOC	1.04	0.0699	0.01169	0.00079
Pu-238	BOC	97.4	4.01	0	0
	EOC	97.5	4.03	7.89E-06	3.26E-07
Pu-239	BOC	87.3	187	0	0
	EOC	80.7	178	0.00299	0.00661
Pu-240	BOC	215	0.421	0	0
	EOC	140	0.418	0.00127	3.79E-06
Pu-241	BOC	82.3	231	0	0
	EOC	79.7	225	0.00018	0.00051
Pu-242	BOC	27.7	0.292	0	0
	EOC	26.6	0.3	1.06E-05	1.19E-07

APPENDIX E. SPENT FUEL CONTENTS

E.1. CANDU-6

E.1.1. Multiple Recycling Option

Table E. 1 A Summary Table of Isotopic Compositions in Recycle SF for 20 MWd/kgHM Cycle Burnup

Cycle #	**SF Content (w/o)**								
	^{233}U	^{234}U	^{235}U	^{236}U	^{239}Pu	^{240}Pu	^{241}Pu	^{242}Pu	Total Fissile
Cycle 1	0.568	0.069	0.317	0.204	0.194	0.086	0.025	0.010	1.104
Cycle 2	0.565	0.163	0.089	0.306	0.250	0.177	0.053	0.054	0.956
Cycle 3	0.546	0.121	0.135	0.261	0.244	0.153	0.045	0.034	0.971
Cycle 4	0.113	0.230	0.050	0.481	0.360	0.322	0.089	0.207	0.613

Cycle #	**SF Content (g)**								
	^{233}U	^{234}U	^{235}U	^{236}U	^{239}Pu	^{240}Pu	^{241}Pu	^{242}Pu	Total Fissile
Cycle 1	107.000	13.000	59.600	38.500	36.600	16.100	4.720	1.910	207.920
Cycle 2	104.000	29.900	16.300	56.300	45.900	32.600	9.700	9.910	175.900
Cycle 3	60.600	41.400	9.810	75.300	58.000	49.100	14.000	24.800	142.410
Cycle 4	21.200	43.200	9.440	90.200	67.500	60.400	16.700	38.900	114.840

Table E. 2 A Summary Table of Isotopic Compositions in Recycle SF for 15 MWd/kgHM Cycle Burnup

Cycle #	SF Content (w/o)								
	^{233}U	^{234}U	^{235}U	^{236}U	^{239}Pu	^{240}Pu	^{241}Pu	^{242}Pu	Total Fissile
Cycle 1	0.499	0.049	0.422	0.170	0.184	0.065	0.018	0.005	1.124
Cycle 2	0.577	0.119	0.145	0.255	0.231	0.141	0.042	0.031	0.995
Cycle 3	0.472	0.182	0.066	0.322	0.276	0.211	0.061	0.077	0.876
Cycle 4	0.256	0.216	0.050	0.396	0.328	0.281	0.079	0.147	0.713
Cycle 5	0.133	0.219	0.049	0.438	0.357	0.316	0.087	0.192	0.626

Cycle #	SF Content (g)								
	^{233}U	^{234}U	^{235}U	^{236}U	^{239}Pu	^{240}Pu	^{241}Pu	^{242}Pu	Total Fissile
Cycle 1	94.400	9.220	79.900	32.200	34.900	12.300	3.500	1.040	212.700
Cycle 2	107.000	22.150	26.950	47.350	42.950	26.250	7.820	5.730	184.720
Cycle 3	87.700	33.800	12.300	59.800	51.300	39.100	11.400	14.300	162.700
Cycle 4	47.800	40.300	9.350	74.000	61.300	52.500	14.700	27.500	133.150
Cycle 5	25.800	42.400	9.410	84.700	69.000	61.200	16.900	37.200	121.110

Table E. 3 A Summary Table of Isotopic Compositions in Recycle SF for 15 MWd/kgHM Cycle Burnup

Cycle #	***SF Content (w/o)***								
	^{233}U	^{234}U	^{235}U	^{236}U	^{239}Pu	^{240}Pu	^{241}Pu	^{242}Pu	Total Fissile
Cycle 1	0.379	0.027	0.620	0.130	0.161	0.040	0.010	0.002	1.170
Cycle 2	0.536	0.073	0.274	0.208	0.214	0.099	0.029	0.013	1.054
Cycle 3	0.546	0.121	0.135	0.261	0.244	0.153	0.045	0.034	0.971
Cycle 4	0.477	0.163	0.076	0.306	0.274	0.201	0.059	0.065	0.886
Cycle 5	0.352	0.196	0.056	0.359	0.314	0.281	0.080	0.111	0.802
Cycle 6	0.099	0.199	0.046	0.421	0.361	0.330	0.091	0.192	0.598

Cycle #	***SF Content (g)***								
	^{233}U	^{234}U	^{235}U	^{236}U	^{239}Pu	^{240}Pu	^{241}Pu	^{242}Pu	Total Fissile
Cycle 1	72.000	5.110	118.000	24.700	30.700	7.540	1.880	0.333	222.580
Cycle 2	102.000	13.900	52.200	39.500	40.700	18.900	5.540	2.470	200.440
Cycle 3	102.000	22.600	25.200	48.750	45.650	28.550	8.465	6.410	181.315
Cycle 4	89.200	30.500	14.300	57.300	51.300	37.600	11.000	12.200	165.800
Cycle 5	65.900	36.700	10.500	67.100	58.700	52.600	14.900	20.800	150.000
Cycle 6	19.000	38.100	8.790	80.600	69.100	63.200	17.500	36.800	114.390

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RESUMES

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