# An optimization scheme for selecting alternative fuels in CANDU-6 reactor

**E. St-Aubin[1] and G. Marleau[1]**
[1] École Polytechnique de Montréal, Québec, Canada

## Abstract

We use DRAGON and DONJON codes and standard CANDU-6 models to develop a fuel management calculations sequence based on both time-average and instantaneous approaches that discriminates fuels along the modelling refinement process. This low computer time cost methodology provided various fuel management important quantities, such as average exit burnup and refueling frequency, and criterion for selecting best alternative fuels. The method is applied to natural uranium reference fuel cycle and to an advanced thorium-DUPIC fuel cycle for which we quantify fertile conversion. Results show that viable and exploitable thorium-DUPIC fuel cycle could be an interesting option in CANDU-6 reactor.

## 1. Introduction

Fuel management strategies in currently operating nuclear reactors have been limited to relatively simple solutions without taking benefits of the large number of nuclides that could be used as alternative nuclear fuel. With uranium resources dwindling in the last fifty years and the increasing price of nuclear fuel, interest for advanced fuel cycles in CANDU reactors including thorium and the direct use of spent PWR fuel (DUPIC) cycles has been revived.

Up until now, CANDU reactors have used natural uranium fuel. However, because of their unique characteristics (on-line refueling, low temperature moderator, etc.) these reactors have already been identified by many authors [1,2] as a privileged environment to develop advanced fuel cycles. One of the ideas pursued in this study is to promote DUPIC as non-traditional driver fuel for thorium instead of using recycled plutonium (RP) or slightly enriched uranium (SEU). The fuel management challenge to introduce such alternative fuels in Canadian reactors could be resumed as: *How can we used existing capacity of CANDU-6 to insure safety and exploitability of advanced fuel cycles while maximising average exit burnup and thorium conversion efficiency?*

The International Atomic Energy Agency (IAEA) has estimated that thorium, which contains Th232 (a fertile isotope) has an abundance 3 to 4 times greater than that of natural uranium in the Earth crust. However, this resource is still not exploited today [2]. The main nuclear characteristic of Th232 that could make it an attractive fuel is its conversion into fissile uranium-233 (U233) after irradiation in a neutrons flux:

$$ {}^{1}_{0}n + {}^{232}_{90}\mathrm{Th} \xrightarrow{\text{capture}} {}^{233}_{90}\mathrm{Th} \xrightarrow{\beta-} {}^{233}_{91}\mathrm{Pa} \xrightarrow{\beta-} {}^{233}_{92}\mathrm{U} \qquad (1) $$

U233 is one of the most efficient nuclear fuels since the average number of neutrons emitted by fission $\nu$ is greater than that for U235 and Pu239, for incident neutron under 1eV [2]. However since this isotope is not present in nature, it has to be produced first in a nuclear reactor. This is where DUPIC fuel can play an important role.

The use of DUPIC fuels in CANDU reactor has been studied for the last 20 years [3]. It is considered an original way to recycle the large quantity of "wastes" produced by PWRs. For PWR fuel initially enriched at 3,5%m. and burned to 35 GWd/$T_{hn}$, DUPIC fuels still contain about 0,9%m. of fissile isotopes including U235, Pu239, Pu241 and Am242m. Accordingly CANDU reactor cores fueled with a combination of DUPIC and thorium should be possible and viable.

The aim here is to propose an advanced thorium/DUPIC cycle. Our analysis will be based on well-known CANDU-6 modelling techniques using the deterministic transport and diffusion codes DRAGON and DONJON. Accordingly, we will first briefly describe in Section 2 the standard DRAGON and DONJON models before introducing a discrete refueling scheme leading to a time-average (TAVG) fuel management approach. The TAVG calculation procedure developed to find the optimal refueling strategy will then be explained in details and then be compared with instantaneous (INST) fuel management component. Section 3 presented an advanced fuel cycle selection procedure based on safety, exploitability and resources utilization criterion evaluated at different modelling steps. In Section 4 we will validate our method by comparing two cases: a reference natural uranium cycle and an advanced thorium-DUPIC fuel cycle. The results for the advanced fuel cycle will also be discussed in details. Finally, concluding remarks and perspectives will be provided in Section 5.

## 2. Fuel management in CANDU-6

The DRAGON and DONJON lattice and finite core codes have been developed at École Polytechnique de Montréal [4,5]. DRAGON is used here to generate the databases for fuels, reflector and reactivity devices that will be used in the DONJON code. Reflector and fuel properties (burnup dependent) are generated using an infinite 2-D lattice calculation on a standard CANDU-6 cell with white reflective boundary condition and a homogeneous $B_1$ leakage model [6]. For reactivity devices, we selected a simplified DRAGON 3-D supercell model [7]. A mesh-centered finite difference solution of the diffusion equation is considered in DONJON for a coarse mesh core discretization associating 1 region per bundle [8]. DONJON can also deal with various fuel management strategies that we will now discuss.

### 2.1 In-line refueling and time-average model

CANDU unique in-line refueling capacity permits nuclear environment modifications at any time $t$, for any bundle $jk$ in the core (bundle $k$ in channel $j$) since every channel can be reloaded with a varying number of fresh bundles $n_j$. Starting with an initial instantaneous burnup distribution $B_{jk}^{\text{initial}}$ in the core, the average exit burnup $B^{\text{exit}}$ increases as the refueling process (or time) goes on since at any position $jk$, the heavy nuclides of mass $m_{\text{hn}}$ produce

fission energy at the rate $P_{jk}$. When $n_j < 12$, $B_{jk}^{\text{initial}}$ depends on the flux existing before the last channel refueling: the final burnup $B_{j(k-n_j)}^{\text{final}}$ is equal to $B_{jk}^{\text{initial}}$, except for the fresh bundles that have been inserted in the core. Equation (2) shows a recursive expression for $B_{jk}^{\text{final}}$ in term of bundle burnup increment $\Delta B_{jk}$ at power $P_{jk}$ during a refueling period $T_j$.

$$B_{jk}^{\text{final}} = \Delta B_{jk} + B_{jk}^{\text{initial}} = \frac{P_{jk} T_j}{m_{\text{hn}}} + \begin{cases} 0 \text{ , if } k \le n_j \\ B_{j(k-n_j)}^{\text{final}} \text{ , if } k > n_j \end{cases} \tag{2}$$

As the fuel burns, the reactivity $\rho$ decreases and must be compensated by refueling fresh fuel to keep the core critical at all time. In the same way, as $B^{\text{exit}}$ increases, $T_j$ decreases since the fresh bundles reloaded have a higher reactivity effect $\Delta\rho$. Accordingly, if the operating parameters (refueling strategy, thermohydraulic behaviour, total power) and reactivity devices positions remain the same, the refueling period $T_j$ and exit burnup $B^{\text{exit}}$, after a few months reach some constant values $\overline{T}_j$ and $\overline{B}^{\text{exit}}$ corresponding to *equilibrium*.

The equilibrium concept is introduced practically as a consequence of refueling strategy. It can be defined mathematically by taking the average over burnup (or time) of the macroscopic cross sections (say $\overline{\Sigma}_{jk}^{x}$ for reaction $x$) and adding the incremental cross sections $\Delta\Sigma_{jk}^{x}$ of the reactivity devices (position $\tilde{y}_{dev}$):

$$\overline{\Sigma}_{jk}^{x} = \frac{1}{n_j \overline{B}_j^{\text{exit}} \overline{\psi}_{jk}} \int_{B_{jk}^{\text{initial}}}^{B_{jk}^{\text{final}}} \Sigma_{jk}^{x}(B) dB + \Delta\Sigma_{jk}^{x}(\tilde{y}_{dev}) \tag{3}$$

Here the channel average exit burnup $\overline{B}_j^{\text{exit}}$ can be evaluated using equation (2) and $\psi_{jk}$ is the axial power shape. Since $\overline{\psi}_{jk}$ depends on the time-average flux $\overline{\phi}$, the diffusion equation together with (3) define a non-linear problem which can be solved by an iterative process known as the *time-average model*. In this context, equilibrium could be physically seen as an invariant (or static) diffusion solution $\overline{\phi}$ that verifies equation (3). The TAVG model is used by core designer to define the average behaviour of a particular fuel cycle. Many useful TAVG quantities can be define, but we concentrate here on channel TAVG exit burnup $\overline{B}_j^{\text{exit}}$, TAVG refueling frequency $\overline{F}$ and fissile inventory ratio ($\overline{FIR}^{\text{exit}}$), define in equations (4), (5) and (6), respectively.

$$\overline{B}_j^{\text{exit}} = \frac{1}{n_j} \sum_k \Delta\overline{B}_{jk} = \frac{\overline{T}_j}{m_{\text{hn}} n_j} \sum_k \overline{P}_{jk} = \frac{\overline{P}_j \overline{T}_j}{m_{\text{hn}} n_j} \tag{4}$$

$$\overline{F} = \sum_j \frac{1}{\overline{T}_j} = \frac{1}{m_{\text{hn}} n_j} \sum_j \frac{\overline{P}_j}{\overline{B}_j^{\text{exit}}} \tag{5}$$

$$\overline{FIR}^{\text{exit}} = m_{\text{fissile}}(\overline{B}^{\text{exit}}) / m_{\text{fissile}}(0) \tag{6}$$

## 2.2 Refueling strategy optimization

The CANDU *radial refueling strategy* is defined as the ratio $\overline{T}_j / \overline{T}_1$ at equilibrium, i.e. when $\overline{T}_j$ is constant with channel $j = 1$ usually taken in the middle of the core. According to equation (4), we can also define a radial refueling strategy $\vec{R}$ as the vector $\vec{B}^{\text{exit}}$, containing the ordered ratio of $\overline{B}_j^{\text{exit}}$ over $\overline{B}_1^{\text{exit}}$ showed in equation (7). The radial refueling strategy $\vec{R}$ and the *axial mode* $n_j$ (and fuel isotopic composition, say $\vec{X}$) describe totally the refueling process imposed to the core: for now on, the *refueling strategy* will be denoted $(\vec{R}, n_j)$.

$$\vec{B}^{\text{exit}} = \overline{B}_1^{\text{exit}} \vec{R} \tag{7}$$

The choice of a good refueling strategy for a particular fuel $\vec{X}$ is very difficult to make since there is a large number of constraints depending on $(\vec{R}, n_j)$ that must be respected at all time, in particular for the equilibrium state, to ensure core integrity and exploitability while maximising $\overline{B}^{\text{exit}}$ and $\overline{FIR}^{\text{exit}}$. To quantify the "constraints respect level" for a particular refueling strategy, it is useful to define an *objective function* $\Xi$ that depends on all the parameters that influence the core behaviour in the TAVG model. We define $\Xi$ in equation (8) as a sum of three terms: $\mathrm{E}$, $\mathrm{P}$ and $\mathrm{B}$ that represent TAVG convergence at equilibrium, TAVG thermohydraulic limits and TAVG exit burnup.

$$\Xi = \mathrm{E}(\varepsilon_{ax}) + \mathrm{P}(\overline{P}_{jk}^{\max}, \overline{P}_j^{\max}) + \mathrm{B}(\overline{B}^{\text{exit}}) \tag{8}$$

$\mathrm{E}$ is a function that depends on the relative convergence $\varepsilon_{ax}$ of the axial flux shape and quantify departure from equilibrium. Acceptable values of $\varepsilon_{ax}$ are taken between 0,1% and 5%, with an increasing value for larger $\varepsilon_{ax}$. $\mathrm{P}$ is a 4-steps function that depends on the maximal channel ($\overline{P}_j^{\max}$) and bundle ($\overline{P}_{jk}^{\max}$) powers. Every step is associated with thermohydraulic limits: $\overline{P}_j^{\lim}$ and/or $\overline{P}_{jk}^{\lim}$. Note that $\mathrm{P,E} \le 0$. Finally, $\mathrm{B}$ is simply defined as the ratio of $\overline{B}^{\text{exit}}$ to $\overline{B}_{ref}^{\text{exit}} = 7{,}5$ GWd/T$_{\text{hn}}$.

The objective function $\Xi$ will be useful in the optimization strategy for the refueling scheme since it provides a mean to compare simply and globally the quality of a particular strategy $(\vec{R}, n_j)$ for an alternative fuel $\vec{X}$.

Next, we need to find the *optimal radial refueling strategy with an imposed axial mode* $\vec{R}_{opt}(n_j)$ by evaluating $\Xi$ over space $S = \left\{(\vec{R}_{j\neq 1}, n_j)\right\}$, in the vicinity of a *unit radial refueling*

*strategy* $\vec{1}$ ($R_j = 1$, $\forall j$), for $n_j = 4,8$ bundles-shift (BS) since these are the only options that preserve a 3-zones axial flux shape symmetry. This information will be used as an input for reactivity devices optimization in a whole core design process.

We will now describe the refueling optimization scheme presented in Figure 1 and based on a standard DONJON CANDU-6 core model. The main idea here is to maximize $\Xi$ by performing an intelligent equilibrium search using a sequence of calculations in the vicinity of $\vec{1}$. This simple optimization method is based on a "numerically verified" assumption: $\Xi$ is a concave function over $S$ (a multi dimensional Cartesian plane), i.e. it has a single global maximum.

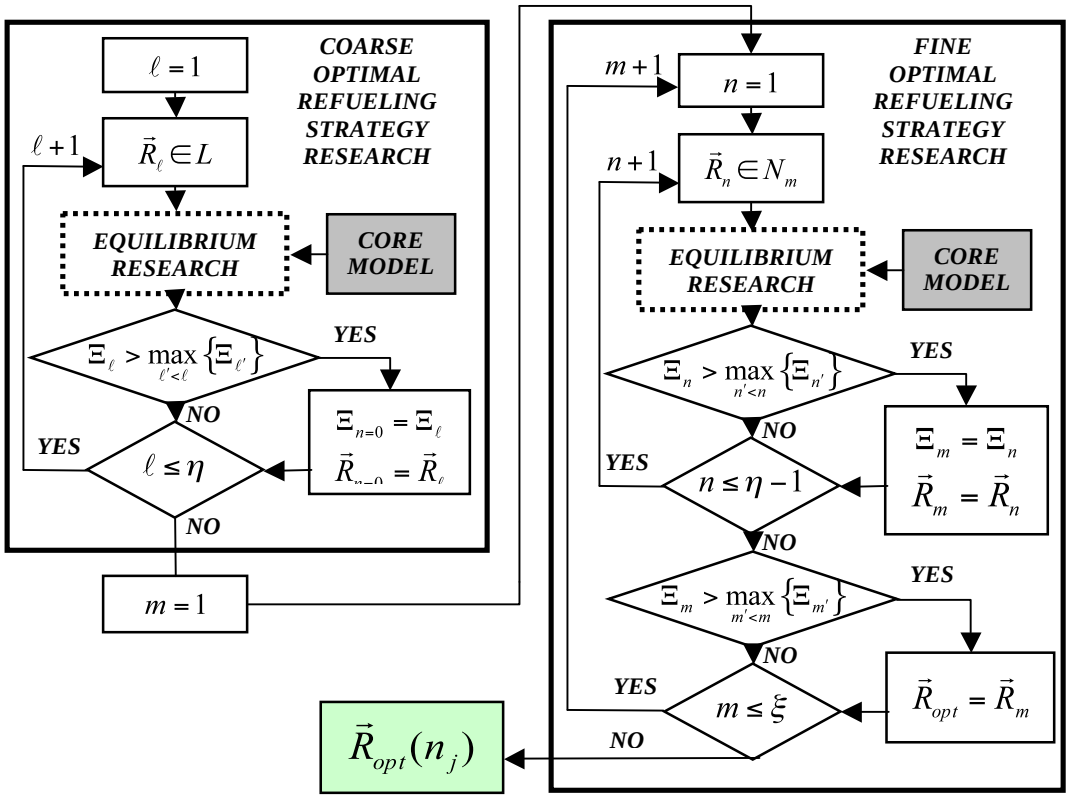


Figure 1 Optimal refueling strategy $\vec{R}_{opt}(n_j)$ search algorithm.

The calculation scheme is separated in 2 parts: coarse (CORSR) and fine (FORSR) optimal refueling strategy researches. The CORSR procedure consists in selecting $\eta = 3^{\dim(S)}$ equilibrium searches around $\vec{1}$, i.e. a dummy span of Cartesian $S$ in any direction. For example, for a 2-channels core, we have $\eta = 3^1$ and calculations are made for $R_2 = 1, 1 \pm \Delta R_L$, where the increment $\Delta R_L$ is linked to the maximal uncertainty on $\vec{R}_{opt}(n_j)$: $e_R^{\max} = \Delta R_L / 2^\xi$. Here, $\xi$ is the number of FORSR procedure performed to find $\vec{R}_{opt}(n_j)$. Note that FORSR follows the same algorithm as CORSR, but it is performed around a better approximation of

$\vec{R}_{opt}(n_j)$ than that initially is provided to CORSR. Both procedures find a refueling strategy that maximizes $\Xi$, the only difference it that FORSR use pre-computed (result of CORSR) initial values.

Our search procedure has the difficult mission of finding a critical TAVG diffusion solution $\bar{\phi}$ for any refueling strategy $(\vec{R}, n_j)$ and for any type fuel $\vec{X}$. This is achieved by using the Brent's algorithm to find the roots of $\delta k_{eff}(\bar{B}_1^{\text{exit}})$ computed using the TAVG model that must itself converges on the axial flux shape and criticality.

When equilibrium is found with refueling strategy $(\vec{R}_\ell \in L, n_j)$, we then evaluate the objective function $\Xi_\ell$ and compare it with that obtained at the previous iteration. At the end of the CORSR process, we have selected the $\vec{R}_{n=0}$ corresponding to the best refueling strategy, i.e. the one associated with $\Xi_{n=0}$, the $\Xi$ maximal value in $L$ sequence. Then, FORSR use the CORSR solution as a starting point for a new search sequence using smaller increment $(\Delta R_L / 2^m)$ around $\vec{R}_{n=0}$. Using the fact that at the point, $\Xi_{n=0}$ is already known from a previous calculation, we only have to perform $\eta - 1$ calculations to span this reduced space to find a new maximal value for $\Xi_m$. FORSR is repeated until $m = \xi$, where $\xi$ is used to define the maximal absolute uncertainty $e_R^{\max}$ on $\vec{R}_{opt}(n_j)$. This general procedure always works if the assumption that $\Xi$ is concave for the fuel $\vec{X}$ selected.

### 2.3 Instantaneous fuel management approach

The TAVG fuel management approach provides reference neutronic behaviour at equilibrium. However, in general during reactor operation, neutron flux oscillates around TAVG and it is very important to quantify such instantaneous states and the size of these oscillations relative to the steady state conditions. This evaluation is based here on the *zonal power peaking factors* (ZPPF) that can be defined using equation (9) for any physically meaningful space region.

$$ZPPF_{\text{inst}} = \max_Z \left( \frac{P_Z^{\text{inst}}}{\bar{P}_Z} \right) \tag{9}$$

Three main approaches have been used to qualify these ZPPF. The *statistical method* that consist in simulating many uniform random age distributions without taking any credit for operator's experience and to analyse the variance of the results. This method is privileged in safety context since it gives conservative results because it doesn't reflect any particular physical process. The *explicit method* simulates every refueling explicitly by a diffusion calculation. However, it needs a strong in-line algorithm to choose the channel to refuel. Both techniques are very expensive in computer time. The third alternative is to use a predetermined fueling scheme that respects the guidelines enounced by Rouben [9] and to refuel every channel one time after the equilibrium is obtained. This *deterministic method*

requires only one diffusion calculation and can be used to validate our equilibrium research algorithm. Rozon has proposed a simple method to generate such a scheme [10].

The method retained here is to simulate only two instantaneous states to generate meaningful information for advanced fuel cycle management. The first is the fresh core (INITIAL), characterised by a homogeneous zero burnup map, which can be used to estimate initial poisoning (boron, depleted uranium) necessary in early core for advanced cycles. The second state (REFUELED) used a deterministic method, based on [9,10], with unity radial refueling strategy $\vec{1}$ to provide a "credible" core after a long-term normal operation.

## 3. Advanced fuel cycles selection

In Section 2, we presented the simulation methods used to quantify advanced fuel cycles. We now focus on how to analyse the information thus generated and to discriminate non viable advanced fuel cycles while models refining. As presented earlier, DRAGON and DONJON CANDU-6 modelling goes through a few steps: infinite lattice model (unnecessary), leaking cell model, reactivity devices supercell model, TAVG and INST finite core calculations. While calculations progressed, many fuel management parameters can be derived and taken as inputs to an advanced fuel cycles selection scheme. Section 3 presents a 3-steps alternative fuels selection scheme based on infinite lattice criterion and on fundamental TAVG and INST fuel management constraints. In each case, selection criterion will be defined and justified based on different physical arguments.

### 3.1 Preliminary alternative fuels selection

CANDU-6 infinite lattice models consist in typical bundle behaviour evolving under constant power density. Even if this model is relatively simple compared to the static finite core diffusion model used in DONJON where control devices are presents, many preliminary conclusions in the potential of new fuels in CANDU-6 can be achieved by analysing the results.

Infinite lattice model realism is limited by 3 main factors: no neutrons leaking, absence of reactivity devices and no poisons in the moderator. The impacts of these factors can be approximated by appropriate "effective criticalities" using some realistic static terms $\delta k$ to compensate their absence in the model. Guillemin [11] estimated $\delta k_{\text{leaks}}$ at various burnup steps using natural uranium CANDU-6 model without devices by comparing $k_{\infty}$ to $k_{eff}$ and showed that it is almost constant. Normally inserted devices, namely stainless steel adjuster rods (100%) and liquid zone controllers (50%), possess well-known reactivity worth [9] that are almost constant during irradiation and are used to estimate $\delta k_{\text{devices}}$. Finally, the maximal poisoning needed $\delta k_{\text{poisons}}$ to compensate xenon saturation and flux flattening can be estimated by subtracting $\delta k_{\text{leaks}}$ and $\delta k_{\text{devices}}$ from $k_{\infty}(0)$, the initial transport eigenvalue.

For selecting advanced fuels that can be operated in a way similar to the reference cycle, it seems natural to impose a maximal value on $k_{\infty}(0)$ (or poisoning). Our first criteria

expressed in equation (10) necessitates only one cell calculation and limits the poisoning needs by imposing maximal value on $k_\infty(0)$ (or $\delta k_{\text{poisons}}$).

$$k_\infty(0) < 1 + \delta k_{\text{leaks}} + \delta k_{\text{devices}} + \delta k_{\text{poisons}} \tag{10}$$

During core start up, poison decreases as the fuel burns. Nuttin [12] approximates the time when core reloading became necessary with $t_{\text{static}}$, defines in equation (11). At this moment, $\delta k_{\text{poisons}}$ become null. In the same way, we define $t_{\text{cycle}}$ in equation (12) as the moment when the time-average eigenvalue $\bar{k}_\infty$ is equal to "effective criticality" level without poison.

$$k_\infty(t_{\text{static}}) = 1 + \delta k_{\text{leaks}} + \delta k_{\text{devices}} \tag{11}$$

$$\bar{k}_\infty(t_{\text{cycle}}) = \frac{1}{t_{\text{cycle}}}\int_0^{t_{\text{cycle}}} k_\infty(t)dt = 1 + \delta k_{\text{leaks}} + \delta k_{\text{devices}} \tag{12}$$

Since $t_{\text{static}}$ measure the no-refueling early core period and $t_{\text{cycle}}$ evaluates bundle in-core period, alternative fuels should have greater $t_{\text{static}}$ and $t_{\text{cycle}}$ values than natural uranium, as expressed in equation (13). Imposing these criterion insures, in infinite lattice approximation, that alternatives fuels exploitability and average exit burnup are equal or better than reference.

$$t_{\text{static}} \geq t_{\text{static}}^{ref} \quad , \quad t_{\text{cycle}} \geq t_{\text{cycle}}^{ref} \tag{13}$$

### 3.2 Optimized advanced fuel cycles selection

For fuels that respected preliminaries criterion (10) and (13), finite core TAVG and INST calculations are justified and should be performed assess our preliminaries results and to provide TAVG optimized refueling strategy and INST states core behaviour, as presented in section 2. Information thus generated have to be analysing on a few aspects before concluding that the optimized advanced fuel cycles are good enough to be pushed forward in accidents, thermohydraulic and kinetic simulations which could follow the present study. We now present optimization constraints and selection criterion based on fundamental neutronic, thermohydraulic and exploitability constraints and how they are implemented in our selection scheme. We will also justify the assumptions leading to these criterion.

Table 1 presents links between physical fundamental constraints, optimization constraints and selection criterion that will be detailed in this section.

| **Fundamental constraints** | Criticality | Exit burnup | Pellets melting | Cladding dry out | Refueling capacity | Conversion |
|---|---|---|---|---|---|---|

| **Optimization constraints** | $\mathrm{E}(\varepsilon_{ax})$ | $\mathrm{B}(\overline{B}^{\mathrm{exit}})$ | $\mathrm{P}(\overline{P}_{jk}^{\max}, \overline{P}_{j}^{\max})$ | | - | - |
|---|---|---|---|---|---|---|
| **Selection criterion** | $\Xi \geq 1$ | $\Xi \geq 1$ | $P_{jk,inst}^{\max} \leq P_{jk}^{\lim}$ | $P_{j,inst}^{\max} \leq P_{j}^{\lim}$ | $\overline{F} \leq \overline{F}_{ref}$ | $\overline{FIR}_{\max}^{\mathrm{exit}}$ |

Table 1 Fuel management constraints and associates criterion for selecting alternative fuels.

Firstly, neutronic fundamental constraints are expressed by TAVG criticality and by maximizing TAVG exit burnup. These constraints are implicitly respected since they are implemented in the equilibrium search (respectively represented by $\mathrm{E}$ and $\mathrm{B}$ terms in the objective function $\Xi$ defines in (8)). We assume here that cladding radiation resistance is good enough to prevent important degradation during advanced cycle operation, i.e. that bundles could be burned up to the maximal age achieved in both TAVG and INST burnup maps.

Secondly, core damaging is also controlled by the cooling capacity provided by thermohydraulic circuitry that limits maximal power density that could be extracted from the core. Nuclear fuel pellets melting is prevented by imposing a maximal value of conductivity integral along bundle length, i.e. by imposing TAVG ( $\overline{P}_{jk}^{\lim}$ ) and INST ( $P_{jk}^{\lim}$ ) maximal bundle power. To avoided critical heat flux, channels are reloaded in the same direction as coolant flow to minimize water enthalpy where fresh bundles are inserted, but we must also put some TAVG and INST limits on channel maximal power ( $\overline{P}_{j}^{\lim}$ , $P_{j}^{\lim}$ ).

Since INST power maps are generally more peaked than TAVG, $P_{j,jk}^{\lim}$ are greater than $\overline{P}_{j,jk}^{\lim}$ by 10-20%. Finally, according to the $\mathrm{P}$ term in $\Xi$, we already implicitly take into account TAVG thermohydraulic limits in equilibrium research algorithm, so only INST component should be explicitly expressed by criterion based on $P_{jk,inst}^{\max}$ and $P_{j,inst}^{\max}$, or simply on $ZPPF_{inst}$. Here, we made the assumption that power limits, based on fuel pellet thermal conductivity, are equal to those of the reference cycle.

Finite core TAVG simulations also provide global results relatives to exploitability and fertile conversion, i.e. refueling frequency $\overline{F}$ and $\overline{FIR}_{\max}^{\mathrm{exit}}$, defined in (5) and (6) respectively. Even if the TAVG and INST criterion are satisfied, the core won't be exploitable the reactor cannot be reloaded at rate $\overline{F}$. Knowing that $\overline{F}_{ref}$ is easily achieved in operating stations, we choose cycles with $\overline{F} \leq \overline{F}_{ref}$. Finally, the best cycle, that respects all the criterion mentioned above, can be analyze for fertile conversion ratio, which we represented here simply by $\overline{FIR}_{\max}^{\mathrm{exit}}$.

## 4. Results and analysis: natural uranium and advanced thorium/ DUPIC fuel cycles

The methodology presented in sections 2 and 3 provides a general way to answer the fundamental question asked in the introduction. In section 4, we focus on applying an easily implementable version of the method for an advanced cycle that takes profits of CANDU-6 existing capacities. For comparison with reference cycle (UNAT), we consider a

homogenous mix of thorium (20%vol.) and DUPIC (80%vol.) dioxides operated in 4BS axial mode, that will be identified as 2T8D. Infinite lattice, TAVG refueling strategy optimization and INST results will now be briefly compared and discussed.

### 4.1 Infinite lattice

The preliminary DRAGON infinite lattice results presented in Table 2 have been generated with the IAEA 69 group cross sections library, a simplified cell discretization and low-order quadrature parameters.

| **CYCLES** | $k_\infty(0)$ | $t_{\text{static}}$ [d] | $t_{\text{cycle}}$ [d] |
|---|---|---|---|
| **UNAT** | 1,12067 | 120 | 220 |
| **2T8D** | 1,22772 | 209 | 454 |

Table 2 Infinite lattice results for UNAT and 2T8D fuels.

Table 2 shows that 2T8D initial poisoning needs are 2,5 times greater than for UNAT, but that the early no refueling period is almost 2 times longer. It also indicates that the fuel management period is longer. Comparing $t_{\text{cycle}}$ gives hopes for a burnup more than 2 times greater in 2T8D than in UNAT, which is a strong argument to study further thus cycle.

### 4.2 Equilibrium and refueling strategy

An equilibrium search ($\delta k_{eff}(\overline{B}_1^{\text{exit}}) < 0{,}1$ mk) for an axially converged solution ($\varepsilon_{ax} < 10^{-3}$) and optimal refueling strategy $\vec{R}_{opt}(n_j)$ have been determined using a 3-zones CANDU-6 burnup model used by Chambon [13] that allowed us to write $\vec{R}_{opt}(n_j)$ simply as a 3-components vector. We perform here $\xi = 2$ FORSR sequences. Main TAVG results are presented in Table 3.

| **CYCLES** | $n_j$ | $\vec{R}_{opt}(n_j)$ | $\varepsilon_{ax}$ | $\overline{P}_{jk}^{\max}$ [kW] | $\overline{P}_j^{\max}$ [MW] | $\overline{B}^{\text{exit}}$ [GWd/$T_{hn}$] | $\overline{F}$ [$d^{-1}$] | $\overline{FIR}^{\text{exit}}$ |
|---|---|---|---|---|---|---|---|---|
| **UNAT** | 8 | (1;1;15/16) | 1,172E-4 | 840 | 6,668 | 7,499 | 1,725 | 0,693 |
| **2T8D** | 4 | (1;1;15/16) | 2,285E-4 | 806 | 6,673 | 15,655 | 1,673 | 0,605 |

Table 3 Equilibrium and optimal refueling strategy results for 2T8D and UNAT cycles.

Globally, TAVG results show that 2T8D average exploitation parameters are very similar to UNAT cycle. In fact, refueling strategy and frequency, and TAVG power map peaking are the same for both cycles (or better for 2T8D). However, $\overline{FIR}^{\text{exit}}$ is 13% lower for 2T8D, but for an average exit burnup 108% greater than UNAT, as expected by preliminary $t_{\text{cycle}}$ analysis. Th232 conversion in U233 is unable to compensate totally minor actinides burnup in DUPIC,

but is sufficient to double extractable energy from fuel. Higher thorium fuel contents should be considered to achieve better conversion efficiency, mainly because of the large amount of U238 in DUPIC.

### 4.3 Instantaneous cores

INITIAL state results resumed in Table 4 show larger core poisoning needs for 2T8D than UNAT (factor of 3 that is consistent with $k_\infty(0)$ analysis). ZPPFs define on bundle (BPPF) and on channel (CPPF) regions are almost the same for both cycles in INITIAL state since flux is here mainly governed by axial and radial burnup map symmetries, present in both cycles.

| CYCLES | INITIAL | | | REFUELED | | |
|---|---|---|---|---|---|---|
| | $k_{eff}$ | *BPPF* | *CPPF* | $k_{eff}$ | *BPPF* | *CPPF* |
| **UNAT** | 1,0669 | 1,354 | 1,067 | 1,0005 | 1,133 | 1,095 |
| **2T8D** | 1,1813 | 1,365 | 1,065 | 1,0003 | 1,354 | 1,146 |

Table 4 2T8D and UNAT cycles in INITIAL and REFUELED instantaneous states.

REFUELED instantaneous state $k_{eff}$ results in Table 4 demonstrate that TAVG equilibrium search was successful, i.e. that it is invariant under a given refueling strategy. On top of that, its show that both cycles are insensitive to small refueling strategy variation in the vicinity of $\vec{R}_{opt}(n_j)$, since REFUELED is generated with unity strategy $\vec{1}$ after equilibrium. We note an important BPPF for 2T8D cycle, but $P^{\max}_{jk,\text{REFUELED}}$ appears on an axial-end bundle due to 4BS axial mode, where TAVG power is low compared to the limit, increasing BPPF artificially.

## 5. Conclusion

We have presented a methodology for selecting alternatives fuel in CANDU-6 reactor based on infinite lattice preliminary simulations, an optimal in-line refueling strategy search algorithm using the time-average model and a deterministic simplified instantaneous fuel management method. These models permit us to evaluate poison needs, static safety margins, refueling strategy and frequency, average exit burnup and thorium conversion efficiency for 2T8D and UNAT cycles. Results have showed that the 2T8D cycle is well adapted for the CANDU-6 and that we can expect to double extracted energy from the fuel compared to current UNAT option, without any disadvantages.

The methodology developed here is sufficiently general that it can be used to assess the effect of multiple parameters variations. It has the strong advantages to use in parallel DRAGON and DONJON and to provide results at low computer cost on many fuel management quantities usually used as input for further studies like reactivity devices adjustment for new fuels.

## 6. Acknowledgments

This work was support in part by a grant from the Natural Science and Engineering Council of Canada (NSERC) and a scholarship from Hydro-Québec.

## 7. References

[1] S. Şahin and al., "CANDU reactor as minor actinide/thorium burner with uniform power density in the fuel bundle", *Annals of Nuclear Energy*, 35, 690-703 (2008).

[2] International Atomic Energy Agency, "Thorium fuel cycle – Potential benefits and challenges", IAEA, Nucelar fuel cycle and materials section, Technical report IAEA-TECDOC-1450, May (2005).

[3] H. Choi, B.W. Rhee and H. Park, "Physics study on direct use of spent pressurized water reactor fuel in CANDU (DUPIC)", *Nuclear Science and Engineering*, 126, 80-93 (1997).

[4] G. Marleau, R. Roy, A. Hébert, "A user guide for DRAGON version 3.05", École Polytechnique de Montréal, Institut de Génie Nucléaire, report IGE-174 Rev. 6, 2006.

[5] E. Varin, A. Hébert, R. Roy and J. Koclas, "A user guide for DONJON version 3.01", École Polytechnique de Montréal, Institut de Génie Nucléaire, report IGE-208 Rev. 4, 2005.

[6] A. Hébert, G. Marleau and R. Roy, "Application of the lattice code DRAGON to CANDU analysis", *Transaction of American Nuclear Society*, 72, 335-337 (1995).

[7] R. Roy, G. Marleau, G. Tajmouati and D. Rozon, "Modelling of CANDU reactivity control devices with lattice code DRAGON", *Annals of Nuclear Energy*, 21, 115-132 (1994).

[8] D. Rozon, R. Roy and E. Varin, "DRAGON/OPTEX predictions of channel power peaking factors and average exit burnup in CANDU-6", *Transaction of American Nuclear Society*, 72, 334-335 (1995).

[9] B. Rouben, "Le CANDU – Étude du cœur et gestion du combustible", Physic Department, Atomic Energy of Canada Limited, Mississauga, Canada, technical report AECL-8333(F), 1984.

[10] D. Rozon, "Gestion du combustible – Chapitre 4 : La gestion du combustible dans le CANDU", *ENE6109 course notes*, École Polytechnique de Montréal, 2000.

[11] P. Guillemin (2009). "Recherche de la haute conversion en cycle thorium dans les réacteurs CANDU et REP. Développement des méthodes de simulation associées et étude de scénarios symbiotiques", Ph.D. thesis, INPG, Grenoble, France, 2009.

[12] A. Nuttin and al., "Study of CANDU thorium-based fuel cycles by deterministic and Monte Carlo methods", Proceedings of PHYSOR 2006, ANS Topical Meeting on Reactor Physics, Vancouver, British-Columbia, Canada, September 10-14 (2006).

[13] R. Chambon (2006). “Optimisation de la gestion du combustible dans les réacteurs CANDU refroidis à l’eau légère”, Ph.D. thesis, École Polytechnique de Montréal, (2006).