Improving the useful life of a 37-element fuel bundle

Corey French[1], Dr. P.K. Chan[2], Lt(N) Stéphane Paquette[2], Dr. Jordan Morelli[3], Madison Sellers[2]
[1] Fourth-year Student, 8cf7@queensu.ca
[1,3]Department of Physics, Engineering Physics & Astronomy, Queen's University
[2]Department of Chemistry & Chemical Engineering, Royal Military College of Canada

Summary

Preliminary results indicate that CANDU burnup using 37-element fuel bundle with a slight enrichment can improve the useful life in the core. A slight enrichment in this study is increasing U-235 from 0.72 to 0.9 mass percent. A parametric study on criticality using Atomic Energy of Canada Limited's WIMS_AECL 3.1 and the Monte Carlo code, MCNP 5, developed by Los Alamos National Laboratory, is presented in this paper.

1. Introduction

Established in the 1950's, CANDU has progressed and is known world-wide today. It emerged from Canada's strict security requirements and offers a unique reactor type and operation. This heavy-water reactor uses deuterium in its moderator and natural uranium (NU) in its fuel. The fuel, in pellet form, is inserted into 37 elements which are welded shut at both ends to form a fuel bundle. Once inserted into a 6.5MW_{th} channel, the NU bundle experiences an average discharge burnup for approximately 180 full power days (FPD[1]).

2. 37-Element Fuel Bundle

This fuel element is comprised of UO_2 ceramic pellets, Zircaloy-4 sheathing, graphite coatings (CANLUB) on the inner sheath surface and Zircaloy-4 end caps [1]. The natural uranium (NU) oxide ceramic fuel pellets are sintered to ~98% theoretical density. The isotopic composition of NU is 0.715% (mass percent) Uranium-235 (U-235) and 99.285% Uranium-238 (U-238) [2].

The objectives of this paper are to improve the *useful life*[2] of a 37-element fuel bundle by adding a small amount of fissile U-235 and to demonstrate this concept by performing computational neutron analysis. The mass percentage of U-235 is minutely augmented (i.e. 0.715% to 0.90% in this study) in the fuel pellet for the purpose of increasing the fissile content throughout the core. This range of enrichment is conceived as practical, having small impact on manufacturing and storage and support reactor operation if on-power refueling is not available for a short period of time.

The impact of this enrichment on criticality is also illustrated in this paper.

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[1] Full Power Day(s): A fuel bundle subjected to full power flux for one 24-hour period.
[2] The *useful life* of the fuel bundle, measured in FPD, is the time that a fresh (free of fission products) 37-element fuel bundle spends in a reactor from startup until it becomes subcritical.

3. Minute Enrichment

The isotopic composition of the fuel changes significantly during its life in the reactor. It is affected by the burn-up of fissile U-235 and the conversion of fertile U-238 into fissile plutonium-239 (Pu-239). U235 has a large fission cross section (σ_f =580.2b)[3]. The fission process releases Xenon and Samarium, which are high yield fission products (F.P.) of U-235. They have negative effects on reactivity and criticality of the reactor. Before startup with a fresh bundle the reactor is free of any neutron absorbing fission products. Once started up, the production of the F.P. will be relative to the fission of U-235 present in the fuel and will decrease the excess reactivity. It is useful to follow the concentration N^i of isotope i in the fuel throughout burnup. The amount of fissionable U-235 in the fuel has the following relationship:

$$\frac{\partial}{\partial t} N^{235} = -N^{235}\sigma_f^{235}\phi - N^{235}\sigma_c^{235}\phi - \lambda^{235}N^{235} \tag{1}$$

Similarly for U-238,

$$\frac{\partial}{\partial t} N^{238} = -N^{238}\sigma_c^{238}\phi - \lambda^{238}N^{238} \tag{2}$$

where ϕ is the thermal neutron flux, λ^i the decay constant, and σ_c^i is the thermal neutron capture cross section. The difference in U-238 composition the equations arise since the fission cross section, σ_f , of U-238 is effectively zero[3]. These equations can be used to calculate concentration of any isotope at any time present in the reactor core.

The intent of this study is to observe the effect of minutely augmenting U-235 on CANDU reactivity and burnup over time. Production and depletion of U-235 and Pu-239 fissile contents are illustrated in Figure 1. A fractional increase reveals that the extra fissile material present in the fuel will have its major effects within the first 100 FPD of burnup. It is important to keep the enrichment small to minimize the impact on fuel manufacturing and safety analysis. The design approach is to ensure fission gas release is within acceptable limits, minimize the pellet volumetric changes during fuel in-reactor life, while maximizing the amount of fissile material present in the fuel element.

The depletion and production for U-235 and Pu-239 in a CANDU reactor core can be graphically represented as follows:

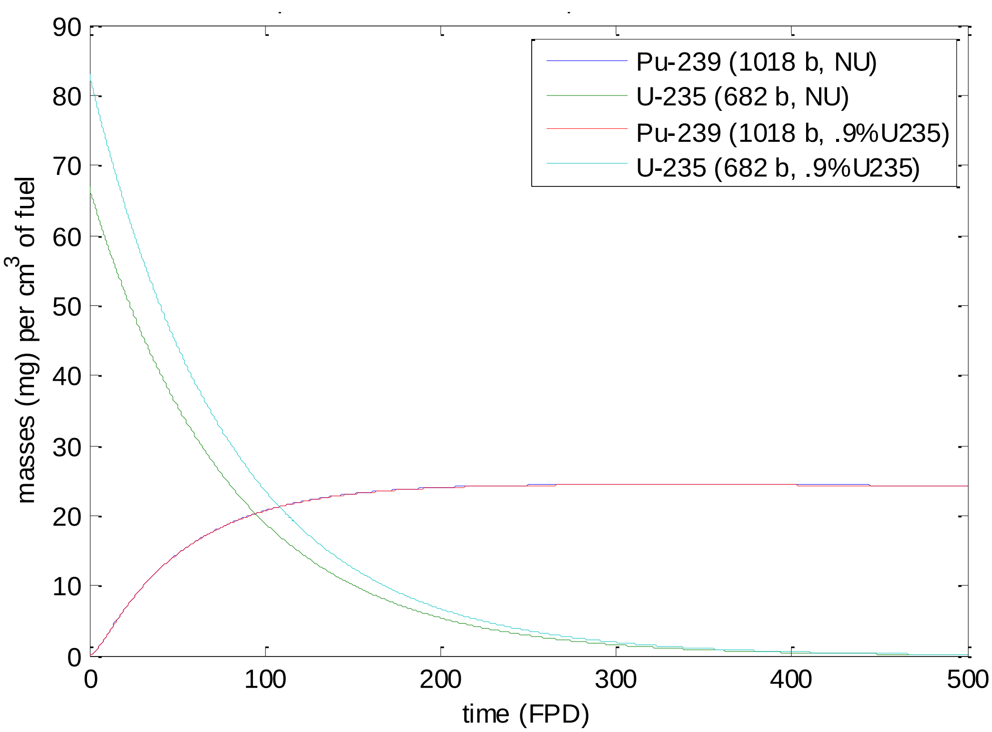

Figure 1: Pu-239 Production & U-235 Depletion in a CANDU Reactor

There is no [visible] difference in the plutonium production curves (blue curve superimposed by red curve) as the fuel is enriched. This is a direct result of the insignificance of U-238 absorption cross section $\left(\sigma_a^{238} = 2.71b\right)$.

4. CANDU Reactivity

The reactivity is formally known as the effective multiplication factor, k_{eff}. Not to be confused with the infinite-lattice multiplication factor, k_∞, the effective factor has an additional term that considers physical properties of the reactor, and can be mathematically described as:

$$k_{eff} = \frac{\nu\Sigma_f}{DB^2 + \Sigma_a} \qquad (3)[4]$$

The numerator describes the production of neutrons in a generation. It is the product of ν, the neutrons produced per fission, and Σ_f the macroscopic fission cross section of the fuel (in this case primarily U-235). The denominator physically describes any loss of neutrons from the reactor. The macroscopic absorption cross section, Σ_a, describes the neutron loss resulting from a capture or transmutation reaction, and DB^2 describes the loss due to leakage in the reactor. B^2 is known as the geometric buckling, defining the reactor's extrapolated boundary that brings the neutron flux to zero.

The physical implication of this equation is the interest of this study; i.e. what will happen to the reactivity of a CANDU reactor core as fissile material is added to a fuel bundle. Buckling will have no effect on the change in reactivity in the reactor. It is anticipated that the change will arise from the gain in fissile material (U-235) from slight fuel enrichment. It is important when considering reactivity in a reactor core that its criticality be profiled with time to record behaviour.

5. Simulation results and discussions

Simulations of reactivity and burnup calculations are carried out with WIMS_AECL 3.1. By physically and parametrically defining a reactor channel and its material properties, WIMS_AECL is capable of simultaneously solving concentration and neutron transport equations to map reactivity and burnup of fuel. In each criticality evaluation, WIMS_AECL assumes a freshly fuelled CANDU reactor and burns at full power until it becomes subcritical ($k_{eff} < 1$). Fuel channel power is held constant in each trial.

Behaviour of the burnup shown in Figure 2 indicates that the useful life of a bundle can be improved significantly with little enrichment. The results demonstrate that the useful life of the fuel bundle, as well as burnup could be significantly increased with minute U-235 enrichment. As a result this could ease the demand on fuelling machine. It is also observed that initial reactor excess reactivity significantly increases with fuel enrichment. This translates into an additional demand for the reactor reactivity control system. This increase in reactivity may also have a significant impact on safety and operating margins. It can also be noticed that the plutonium peaking phenomenon present building up at around 50 FPD becomes less significant with increasing enrichment. As the fuel becomes slightly enriched, its burnup resembles that of a LWR. This seems intuitive since these types of reactors do regularly use enriched uranium around 3% [2].

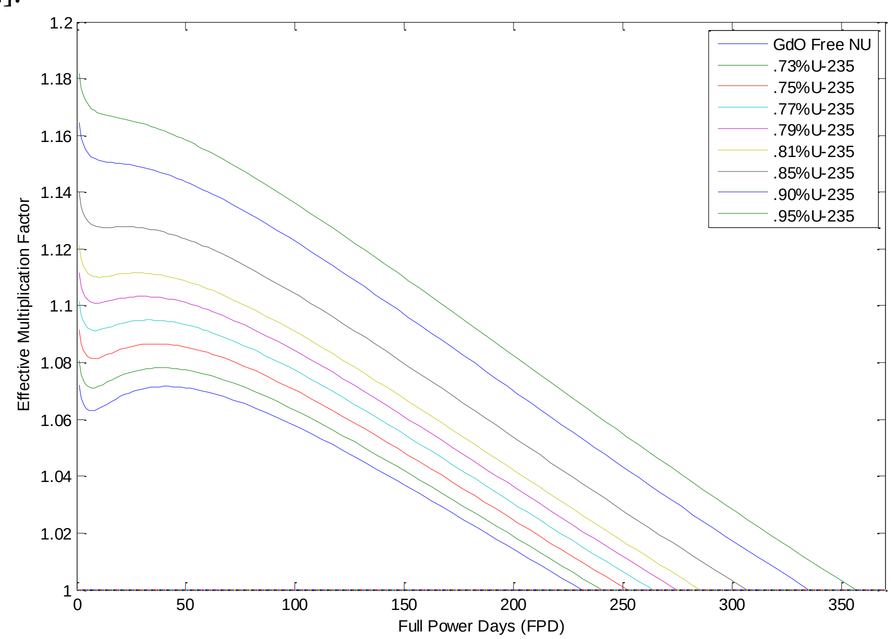

Figure 2: Fuel Bundle Cycle

The impact of enrichment on criticality is demonstrated using the scholastic particle transport code MCNP5. MCNP-5 is a continuous-energy, time-dependent neutron transport code capable of computing k_{eff}. The geometry used here is simply stacking up 60 bundles (approximated as a single rod volume), each having a U-mass of 18.5 kg arranged in a hexagonal array as shown in the following illustration Figure 3.

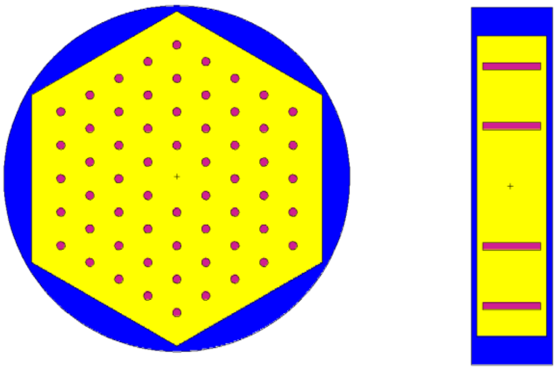

Figure 3: Hexagonal geometry (top and side view) created in MCNP used for criticality calculations

The setup is similar to the dry spent fuel storage of certain CANDU schemes [5]. The purpose is to examine the criticality when this geometry is placed in heavy water, light water and air. This is done to compare both NU and slightly enriched fuel at 0.85% U-235. The following table presents the results obtained from MCNP. In all three cases, the results indicate that the increase in k_{eff} is small for the slightly enriched fuel. Also, k_{eff} remains subcritical.

Table 1: MCNP5 results. k_{eff} for NU and 0.85% U-235 in air, light and heavy water.

Type of surrounding	k_{eff} for NU	k_{eff} for 0.85% U-235
D_2O	0.58350	0.62325
H_2O	0.56522	0.60570
In air	0.10353	0.10437

6. **Closing remarks**

The neutron analysis results indicate that 0.85 mass percent U-235 per fuel bundle meet the objectives of the present work. As illustrated in Figure 2, the life cycle of the fuel bundle that contains 0.85% U-235 is prolonged by about 70 FPD. The impact on criticality with an enrichment of 0.85% is small and remains subcritical in D_2O. From a safety viewpoint, this would mean an increase in demand from zone reactivity control devices. From a fuel management perspective, the fuelling scheme could be improved.

To support fuelling, the initial transient due to Xenon-free effects could be removed with Gd_2O addition into each bundle [6]. A paper to illustrate this conceptual design is also presented in this Conference.

7. References

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