Analysis of CANDU Reactor Performance Using Thorium Fuel: Comparison with Natural UO₂ Case

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Abstract: The purpose of the paper is to study the performance of the CANDU (Canada Deuterium Uranium) reactor when the reactor core is loaded with thorium fuel mixed with plutonium isotopes with ratio 3 and 5%. A three dimensional model is designed for the core of CANDU reactor. The computer code MCNPX (Monte Carlo N–Particle Transport) is used to calculate the processes in its core. The results are compared with natural UO₂ case which is the typical fuel of the reactor. The results show that the multiplication factor of the reactor is higher even in the case of thorium fuel mixed with 3% plutonium isotopes, which indicates longer neutron life cycle length and more economic utilization of the reactor.

Key words: CANDU reactor, MCNPX code, reactor burn up, natural uranium, thorium fuel.

1. Introduction

The Canada Deuterium Uranium (CANDU) is a Canadian pressurized heavy-water reactor design used to generate electric power. The acronym refers to its deuterium oxide (heavy water) moderator and its use of (originally, natural) uranium fuel. The reactor is cooled by pressurized heavy water reactor [1-5].

CANDU can sustain a chain reaction with a lower concentration of fissile atoms than light-water reactors, allowing it to use some alternative fuels; for example, “recovered uranium” (RU) from used light water reactor (LWR) fuel. CANDU was designed for natural uranium with only 0.7% ²³⁵U, so RU with 0.9% ²³⁵U is a rich fuel. This extracts further 30-40% energy from the uranium.

The direct use of spent pressurized water reactor (PWR) fuel in CANDU (DUPIC) process under development can be recycled it even without reprocessing. The fuel is sintered in air (oxidized), then in hydrogen (reduced) to break it into a powder, which is then formed into CANDU fuel pellets [6]. It is the most efficient neutron utilization of any power reactor in the world.

CANDU can also be fueled with thorium mixed with plutonium isotopes or slightly enriched uranium. The plutonium could be obtained either from dismantled nuclear weapons or reprocessed reactor fuel [7].

MCNPX computer code package is used to model half core of CANDU reactor due to its symmetric construction, the model is used to study the performance of the reactor under different fuel types. Natural uranium and thorium fed with different plutonium isotope concentrations are considered and a comparison is made with each other. In the following, Section II gives the details of the reactor, Section III has the description of the reactor model, Section IV shows the model validation, Section V presents the results and discussions and Section VI gives the conclusion and the references are stated at the end of the paper.

2. The Reactor Description

The calculations over the whole core are extensive
computationally, but due to symmetry of the core (geometrical and physical) one can suggest to carry out simulation on half of the core with no much loss in the overall facts of burn up.

The stylized half core consists of 190 horizontal fuel channels, each comprised of 12 fuel bundles. The channels are refueled on-line from both ends of the reactor. The stylized core is assumed to be symmetric about the horizontal mid-plane, so that only half of the core can be modeled. This imposed symmetry is an approximation to an operating CANDU reactor. A heavy water reflector completely surrounds the fuel lattice in the exterior radial direction. The reflector extends to at least two lattice pitches radially beyond the outer-most fuel channels [8].

The model contains only one type of adjuster rod. These rods are located interstitially perpendicular to the fuel channels, arranged axially in three rows by seven at \( y = 217.49, 297.18, 377.18 \text{ cm} \) where \( y = 0 \) is the exterior boundary of the first axial plane. These rods are parallel with the vertical direction and extend 171.45 cm downward from the plane of symmetry (a distance of six lattice pitches). Each rod consists of a solid stainless steel cylinder (shim) that is centered in a stainless steel tube. The shim and tube are composed of two vertical segments with slightly different diameters [8, 9].

The adjuster rod assembly (shim and tube) is centered within a zirconium guide tube. Heavy water fills the gaps between the steel shim and the inner steel tube and between the inner steel tube and outer guide tube [9]. The horizontal and vertical cross sections of the stylized core model are shown in Figs. 1a and 1b.

3. The Reactor Model

MCNPX computer code system is used to design a 3-D heterogeneous model for the half core [10]. Two million neutrons histories are used to simulate the stylized core model and accumulate the tallies. The geometry of the half core model is reflected from the top side of the core to consider the effect of second half. In this model the bundles geometry contains 37 fuel rods, there is an arrangement in the hexagonal shape, a center rod surrounded by three rings of fuel rods, the inner ring contains 6 rods, the intermediate ring contains 12 rods and the outer ring contains 18 rods. The fuel rods are surrounded by clad from zirconium element and heavy water coolant and pressure tube made of zirconium and it is surrounded by moderator (D\textsubscript{2}O) heavy water [9]. The horizontal and vertical heterogeneous cross sections of the computer model for the reactor core used in the program MCNPX are shown in Figs. 1a and 1b. Three models were designed according to the type of the fuel:

Case A: the reactor core is fueled with U\textsubscript{235} with natural enrichment (the practical fuel of CANDU reactors).

Case B: thorium oxide fuel mixed with mixed oxides (MOX) of plutonium by the ratio of Th\textsubscript{232}/Pu\textsubscript{239} [95/5]. The plutonium isotopes are Pu-238, Pu-239, Pu-240, Pu-241, Pu-242 with their ratios 1/62/24/8/5 by weight respectively.

Case C: thorium oxide fuel mixed with MOX with ratio of Th\textsubscript{232}/Pu\textsubscript{239} [97/3] by weight with the same plutonium isotope ratios as Case B. Detailed composition of thorium fuel can be found in Ref. [11].

4. The Model Validation

The present MCNPX model described in the previous section is validated by comparing the results of the model with both homogeneous and heterogeneous cases with the reference and the results are published previously [9].

5. Results and Discussions

Fig. 2 illustrates the multiplication factor for the reactor core versus operation time (days) for the three different cases; Case A: natural uranium fuel, Case B: thorium fuel with 5% Pu isotopes and Case C: thorium fuel with 3% Pu isotopes. The results show that higher multiplication factor for the cases B and C of thorium
fuel with 5 and 3% Pu isotopes all over the operation time, and as the concentration of the Pu isotopes increases in the mixture the multiplication factor increases significantly especially at early operation times. The higher multiplication factor implies longer neutron life cycle in the core and better utilization of the fuel and cost effective economics for the reactor.

Fig. 3 illustrates the $^{235}$U concentrations (atom/barn.cm) versus operation time (days). The results indicate that $^{235}$U concentration decreases from $1.613 \times 10^{-4}$ at fresh fuel to $1.688 \times 10^{-5}$ at 540 days which indicates that 89% of the $^{235}$U is consumed during this period (540 days).

Fig. 4 illustrates the plutonium isotopes concentrations (atom/barn.cm) versus operation time (days) compared with case A of natural uranium case. The results indicate that all Pu isotopes increase with time monotonically over the period of study, but for $^{239}$Pu approaches asymptotic saturation behavior after 250 days of operation with concentration $6.67 \times 10^{-5}$ (atom/barn.cm) through its balanced consumption in the fission process and its production from decay processes of uranium.

Fig. 5 illustrates the evolution of Pu isotopes with operation time during operation of the reactor core fueled with thorium and 5% Pu isotopes. The results
indicate that the concentrations of $^{238}$Pu, $^{239}$Pu and $^{241}$Pu decrease with operation time, while $^{240}$Pu and $^{242}$Pu increase. The most effective $^{239}$Pu isotope which is mainly responsible for fission in the reactor decreases from $6.892 \times 10^{-4}$ (atom/barn.cm) at 0 burn up to $3.488 \times 10^{-4}$ which means that almost 50% of $^{239}$Pu is consumed and burned up, while $^{241}$Pu decreases slightly from $8.819 \times 10^{-5}$ (atom/barn.cm) to $8.326 \times 10^{-5}$ which is about 5% reduction only, this is due to the production and consumption of $^{241}$Pu during the operation time while for $^{240,242}$Pu are slightly increasing monotonically with operation time and $^{238,241}$Pu are almost constants i.e. production and consumption for both are equal.

**Fig. 2** $K_{\text{eff}}$ versus operation time (day) for three types of fuel.

**Fig. 3** U-235 concentration (atom/barn.cm) versus the operation time.
Fig. 4  Plutonium isotopes concentrations (atom/barn.cm) versus the operation time in case in natural uranium fuel only.

Fig. 5  Mixture of thorium-plutonium 5% ratio and different plutonium isotopes concentrations (atom/barn.cm) versus the operation time.

Fig. 6  Illustrates production of $^{233}$U isotope during the operation time of the reactor core fueled with thorium and 5% Pu isotopes, uranium-233 is produced by the neutron irradiation of thorium-232 to form thorium-233 followed by successive decay series which leads to $^{233}$U. It starts from 0 at fresh fuel and monotonically increasing to $1.016 \times 10^4$ (atom/barn.cm) for irradiation time of 540 days, the rise of the uranium-233 concentration in the reactor core is due to its less participation in the fission process since it
Fig. 6 The mixture of thorium-plutonium 5% ratio, the uranium-233 concentration (atom/barn.cm) versus the operation time.

Table 1 Thermal neutrons cross sections for U, Pu isotopes.

| Isotopes | Fission cross section (barns) | Cross section for $(n, \gamma)$ (barns) | Total cross section (barns) |
|----------|-------------------------------|----------------------------------------|-----------------------------|
| $^{233}$U | 529                           | 46.0                                   | 588                         |
| $^{234}$U | 0.456                         | 103                                    | 116                         |
| $^{235}$U | 587                           | 99.3                                   | 700                         |
| $^{236}$U | 47 mb                         | 5.14                                   | 14.1                        |
| $^{238}$U | 11.8 µb                       | 2.73                                   | 12.2                        |
| $^{239}$Pu | 749                           | 271                                    | 1,028                       |
| $^{240}$Pu | 64 mb                         | 289                                    | 290                         |
| $^{241}$Pu | 1,015                         | 363                                    | 1,389                       |
| $^{242}$Pu | 1.0 mb                        | 19.3                                   | 27.0                        |

has relatively lower thermal neutron cross section for fission which is smaller than uranium-235 and plutonium-239 (Table 1), but sometimes retains the neutron to become uranium-234.

Fig. 7 illustrates the concentration of $^{237}$Np versus operation time (days) for thorium fuel with 5% Pu isotopes. The results indicate that $^{237}$Np increases with time due to several processes, one process is that $^{236}$U captures neutron (37.56 b) to produce $^{237}$U which decays to $^{237}$Np. Another process is that energetic neutron knocks out two neutrons from $^{238}$U (45 mb) which is the major constituent of the reactor fuel, then $^{238}$U decays by β emission ($t_{1/2} = 8.75$ d) to get $^{237}$Np. The losses due to capturing neutrons to form heavier short lived isotopes and decay fast to U isotopes. $^{237}$Np half-life is 2.144000 years, which explains the sharp rise of the concentration curve with time and it starts from 0 up to $6.946 \times 10^{-9}$ (atom/barn.cm) after 540 days.

Fig. 8 illustrates the concentration of americium isotopes versus operation time (days) for thorium fuel with 5% Pu isotopes. Americium can be formed by neutron capture of $^{239}$Pu and $^{240}$Pu, forming $^{241}$Pu which then decays by beta to $^{241}$Am. It has half-life of 432.2 years for his alpha decay to produce $^{237}$Np. With the intense neutron flux, $^{241}$Am is the dominant
nuclide though there are small but significant quantities of $^{242}\text{Am}$ and $^{243}\text{Am}$ which are obtained from $^{241}\text{Am}$ by neutron capture. The fission process after neutron capture causes the reduction in $^{241}\text{Am}$ concentration. The concentrations of $^{241}\text{Am}$, $^{242}\text{Am}$ and $^{243}\text{Am}$ after 540 days are $4.667\times 10^{-6}$, $2.226\times 10^{-8}$ and $4.37\times 10^{-6}$ (atom/barn.cm) respectively. The curves have common features starting from small values to saturation which means that the rate of production via neutron capture by Pu equals to the decay rate by alpha.

Fig. 9 illustrates the neutron flux energy distribution (reference values) for thorium fuel with 5% Pu compared with natural uranium. The results indicate that the majority of neutrons are thermal flux in the

Fig. 7  The mixture of thorium-plutonium 5% ratio for neptunium-237 concentration (atom/barn.cm) versus the operation time.

Fig. 8  The mixture of thorium-plutonium 5% ratio, americium isotopes concentrations (atom/barn.cm) versus the operation time.
two cases in the reactor as required for the fission process to continue, governed by the design, structure and materials. The similarity means that the flux in two cases comes to same distribution and flux of mixed fuel (Th and Pu) slightly increased at small thermal neutron energy.

6. Conclusions

(1) MCNPX computer model is set to simulate the design and operation of CANDU reactors in a typical operation conditions.

(2) Three types of fuel are tested: natural uranium, thorium fuel oxides with 5% and 3% Pu isotopes.

(3) Multiplication factor for thorium fuel oxides with 5% and 3% Pu isotopes is higher than natural uranium over the entire cycle of 540 days which ensure long operation core cycle and more economic utilization of thorium fuel mixed with small ratio of plutonium isotopes. And clearly also the multiplication factor as the Pu mixing ratio 5% is always greater than 3% consistently as expected.

(4) Natural uranium fuel builds Pu isotopes while thorium fuel mixed with small ratio of Pu isotopes depletes and consumes Pu isotopes but minor actinides (Np and Am) build and increase with time.

Acknowledgement

I would like to present my thanks and appreciation to the Institute of International Education (IIE) for granting the Rescue Scholar Fund Program Award (IIE-SRF fellowship) that supports, arranges and funds fellowship for threatened and displaced scholars at partnering higher education institutions worldwide.

The authors wish to thank Prof. Mustafa Aziz, professor of nuclear engineering, research fields: reactor core calculations; for his essential help in different aspects during this work.

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