Transactions of the Korean Nuclear Society Spring Meeting
Jeju, Korea, May 12-13, 2016

A New In-core Production Method of Co-60 in CANDU Reactors

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1. Introduction

In addition to electricity generation, CANDU6 reactors are also used to produce Co-60, which is an important radioactive nuclide in medical, industry and many other fields.

The basic technology for Co-60 producing was developed by MDS Nordion and Atomic Energy of Canada Limited (AECL) in 1946 and the same technology was adapted and applied in CANDU6 power reactors. The standard CANDU6 reactor has 21 adjuster rods which are fully inserted into the core during normal operation. The stainless steel adjuster rods are replaced with neutronically-equivalent Co-59 adjusters to produce Co-60 [1].

Nowadays, the roles of the adjuster rods are rather vague since nuclear reactors cannot be quickly restarted after a sudden reactor trip due to more stringent regulations. In some Canadian CANDU6 reactors, some or all the adjuster rods are removed from the core to maximize the uranium utilization [2].

Dr. Y. W. Park proposed an idea of Co-60 production by using the CANDU6 fuel bundle, instead of the adjuster rods [3]. In this study, an innovative way of producing Co-60 in CANDU6 reactors is introduced. The fuel in central element is replaced by a Co-59 target, and Co-60 is obtained after the fuel bundle is discharged. To get higher production of Co-60, different target models are considered and compared with the conventional Co-60 production method. The necessary reactivity loss is calculated to get the same discharge burnup as conventional CANDU6 reactors. Coolant void reactivity and lattice power distribution of different models are also studied. All the neutronic analyses are performed with the continuous-energy Monte Carlo code Serpent in this work [4].

2. Analysis Models and Methods

2.1 Analysis Models

As shown in Table 2-1, two types of fuel bundle are analyzed in this study: the standard and CANFLEX fuel bundle. Two kinds of Co-59 targets are also considered, either cylindrical or thin annulus shape.

In order to derive a favorable target configuration, 5 kinds of models are considered. The conventional fuel pin structure which is filled with natural UO_2 is applied in Case 1. In Cases 2, 3 and 5, the UO_2 fuel in the center pin is replaced with a cylindrical Co-59 target surrounded by graphite and SiC, as shown in Fig. 2-1. The radius of Co-59 cylindrical is 0.1 cm in Case 2 and 3, which means Co-59 occupies 2.7% of the total volume of the original natural uranium fuel in Case 1. In Case 5, in order to get the same volume fraction of Co-59, the radius of Co-59 target is adjusted to 0.1045 cm, which is a litter larger since the radius of the pins in the inner two rings is also larger than that of the standard CANDU6 fuel bundle. For comparison, an annular Co-59 target is considered in Case 4, in which Co-59 surrounds the center graphite and the volume fraction of the annular Co-59 target is also 2.7%, as shown in Fig. 2-2.

Table 2-1: Co-59 target models

Fuel type	Case No.	Center pin	Volume fraction of Co-59
Standard	1	Fuel (UO_2)	Not available
	2	Co surrounded by C	2.70%
	3	Co surrounded by SiC	2.70%
	4	C surrounded by Co	2.70%
CANLFEX	5	Co surrounded by C	2.70%

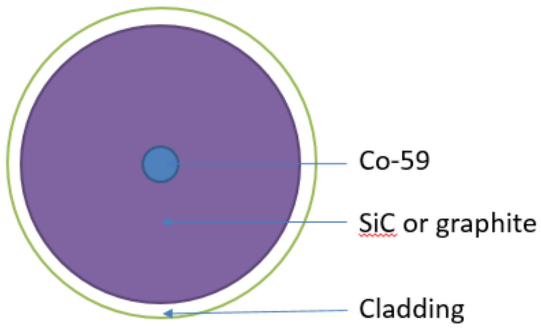

Fig. 2-1. Structure of the center pin in Case 2, 3 & 5

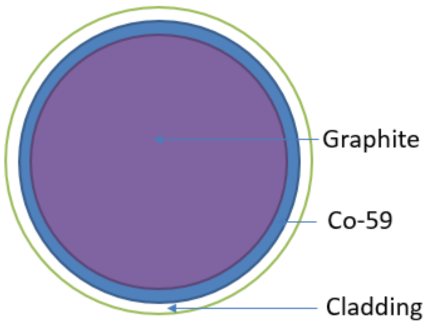

Fig. 2-2. Structure of the center pin in Case 4

2.2 Analysis Methods

In this study, the analysis is done for a 3-D single lattice. Infinite multiplication factor is calculated by the Serpent code as a function of burnup with the ENDF/B-VII.1 nuclear data library. In the Monte Carlo calculations, the number of the neutron histories is 50,000 in each cycle and the total number of cycles is 400 including 100 inactive ones. The standard deviation of the infinite multiplication factor during the depletion ranges from 22 to 40 pcm. In CANDU6 lattice analysis, the approximate discharge burnup is determined by the non-linear reactivity model [5]. In typical CANDU6 core, the reactivity loss is calculated to be about 4.3% or 4,300 pcm.

3. Results and Discussions

3.1 Multiplication Factor and Discharge Burnup

Figure 3-1 shows the infinite multiplication factor values over the fuel burnup for five cases in Table 2-1. It indicates that replacing the fuel in the center pin with a Co-59 target does reduce the multiplication factor quite noticeably. This is because that a non-fuel absorber replaces the uranium fuel. The absorption cross section of Co-59 is much higher than that of natural uranium in thermal and resonance energy range.

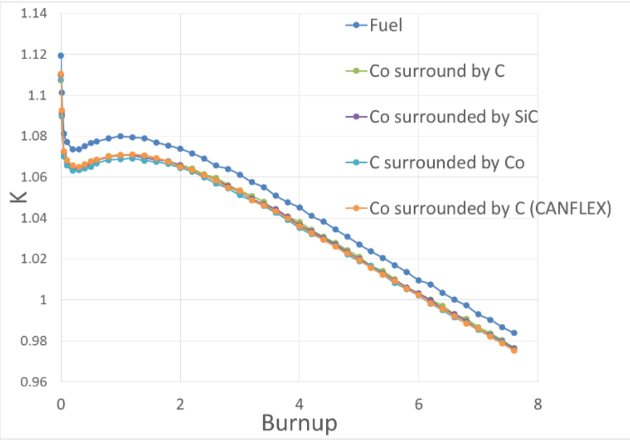

Fig. 3-1. Kinf value as a function of burnup

Table 3-1: Discharge burnup with different reactivity loss

Bundle Type	No.	ρ_{loss} (pcm)		
		4300	3700	3100
		B_d (GWd/MTU)		
Standard	1	7.19	8.00	8.80
	2	6.24	7.08	7.89
	3	6.16	7.00	7.81
	4	6.01	6.86	7.67
CANLFEX	5	6.13	6.96	7.77

Due to the reduced reactivity over the burnup period, the achievable discharge burnup is also smaller than that of standard CANDU6 fuel bundle. The discharge burnup of the fuel are summarized Table 3-1 for several values of the reactivity loss. In practice, the reactivity loss in CANDU6 core can be reduced by removing some of the adjuster rods. The total reactivity worth of the 21 adjuster rods in a typical CANDU6 core is assumed to be around 1,200 pcm. In Table 3-2, the required reactivity loss to achieve the conventional fuel burnup of 7200 MWD/tU is provided and corresponding number of adjuster rods to be removed is also given.

Table 3-2: Necessary reactivity loss for same discharge burnup

Fuel bundle	No.	ρ_{loss} (pcm)	No. of removed adjustor rods
Standard	1	4293	0
	2	3611	12
	3	3552	13
	4	3448	15
CANLFEX	5	3522	14

According to Table 3-2, desired discharge burnup can be obtained in the Co-loaded cases by removing some adjuster rods. It is important to note that limited amount of Co-59 can be loaded into the central pin without compromising the fuel discharge burnup.

3.2 Co-60 Production Capacity

Nowadays, in the standard CANDU6 core, 8-bundle fuel shifting is used for the daily fuel management and about 1.9 channels are reloaded every day. So the Co-60 production capacity can be calculated at the discharge burnup of 7,200 MWD/MTU, as shown in Table 3-3.

Table 3-3: Production of Co-60

Fuel type	No.	Per bundle	Per year	Radioactivity produced per year
		gram	gram	curies
Standard	1	0	0	0
	2	0.542	3164.5	3.5E+06
	3	0.543	3171.9	3.5E+06
	4	0.677	3954.0	4.3E+06
CANLFEX	5	0.602	3518.5	3.9E+06

From Table 3-3, it is clear that the maximum amount of Co-60 can be achieved in Case 4, in which annular Co-59 target surrounds a graphite cylinder. This is mainly because the thermal neutron flux can be higher at the edge of center pin. In addition, it is also evident that Co target surrounding SiC cylinder produces slightly smaller amount of Co-60 since the neutron capture by Si is noticeably higher than that of C.

In this work, analytic calculation of Co-60 mass is also conducted to make sure that the Serpent prediction is acceptable. The change of atomic number of Co-59 is as below:

$$\frac{dN_{Co-59}(t)}{dt} = -N_{Co-59}(t) * \overline{\sigma_{Co59_c}} * \overline{\Phi} \quad (1)$$

In Eq. (1), the production of Co-59 is negligible and the loss of Co-59 is assumed only due to neutron absorption.

For Co-60, as shown in Eq. (2), the production term is due to neutron absorption of Co-59 and the loss term comes from beta and gamma decay.

$$\frac{dN_{Co-60}(t)}{dt} = N_{Co-59}(t) * \overline{\sigma_{Co59_c}} * \overline{\Phi} - \lambda * N_{Co-60}(t) \quad (2)$$

The one-group neutron flux and microscopic cross section can be calculated through Serpent, which are shown in Table 3-4. For simplicity, the whole residence time (around 215 days) is divided into six intervals and neutron flux, cross section are assumed to be constant in each interval. The analytic calculation of Co-60 mass in Case 2 is 0.530 gram, which is 2.4 % smaller than the value from Serpent depletion calculation.

Table 3-4: Six-interval data of analytic calculation

Burnup	Time	Neutron flux	$\sigma_{c,Co-59}$	Atomic Number of Co-59
MWd/kgU	day	#/cm^2-sec	barn	#
0	0	4.435E+14	8.182	9.0944E+22
1.2	35.83	4.548E+14	7.564	9.0307E+22
2.4	71.65	4.544E+14	7.363	8.9697E+22
3.6	107.48	4.644E+14	7.282	8.9099E+22
4.8	143.30	4.736E+14	7.267	8.8504E+22
6.0	179.13	4.854E+14	7.200	8.7901E+22
7.2	214.95	4.996E+14	7.106	8.7290E+22

As shown in Table 3-3, the Co-60 production in terms of activity per year is over 3.5 million curies every year, which is noticeably higher than that produced in a CANDU6 reactor core in Qinshan Nuclear Power Plant, around 3 million curies per year [6].

3.3 Coolant Void Reactivity

Due to the unique fuel channel design, CANDU reactor has a strong positive coolant void reactivity (CVR), which is an outstanding safety concern of CANDU6 reactors. Fortunately, CVR can be reduced substantially by loading a strong neutron absorber in the central region of the fuel bundle, like the Co-59 target in this study. To see the impacts of the Co-59 loading in the fuel bundle, the CVR values are evaluated for both the fresh and mid-burnup conditions of the fuel bundle, as shown in Table 3-5.

From Table 3-4, it is obvious that the Co-59 loading does reduce CVR both at both zero and mid-burnup. The reason is described below. Coolant voiding causes a slight increase of thermal flux in the center of the lattice, simply because more thermal neutrons, produced in the moderator region, are able to reach the central region of the fuel bundle in the case of coolant loss so that more thermal neutrons are absorbed by Co-59.

Table 3-5: Impacts of Co loading on the CVR

Fuel type	No.	CVR (Fresh)	CVR (Mid-burnup)
Standard	1	16.982	14.995
	2	16.415	13.889
	3	15.725	12.374
	4	16.172	12.925
CANLFEX	5	17.442	14.130

3.4 Power Distribution in Fuel Bundle

Normalized pin power distribution in the Co-loaded fuel bundle in all cases is analyzed at zero burnup, mid-burnup, and discharge burnup conditions. In all cases, the peak power decreases with depletion. It is because the fissile isotopes in outer rings deplete faster than those in inner rings. The spatial self-shielding of the fuel bundle becomes weaker with depletion, and the normalized power in the inner rings tends to increase with burnup.

Table 3-6: Burnup-dependent power peaks

Fuel type	No.	BOL	MOL	EOL
Standard	1	1.141	1.136	1.124
	2	1.163	1.158	1.146
	3	1.163	1.157	1.145
	4	1.166	1.160	1.146
CANLFEX	5	1.151	1.148	1.133

Table 3-6 shows the power peak at BOL (beginning of life), MOL (mid of life) and EOL (end of life). It is clear that replacing the fuel with Co-59 target slightly increases the peak power in the fuel bundle. One can also note that the peak power is noticeably lower using the CANFLEX design. It is mainly because that CANFLEX fuel bundle has more fuel pins and it provides a smaller linear power than the standard one. Additionally, the bundle power distributions are flattened in CANFLEX by introducing two types of fuel elements.

4. Conclusions and Future Works

This study introduces an innovative method for Co-60 production in the CANDU6 core. In this new scheme, the central fuel element is replaced by a Co-59 target and Co-60 is obtained after the fuel bundle is discharged. It has been shown that the new method can produce significantly higher amount of Co-60 than the conventional Co production method in CANDU6 reactors without compromising the fuel burnup by removing some (<50%) of the adjuster rods in the whole core. The coolant void reactivity is noticeably reduced when a Co-59 target is loaded into the central pin of the fuel bundle. Meanwhile, the peak power in a fuel bundle is just a little higher due to the central Co-59 target than in conventional CANDU6 fuel design.

In future study, the Co-59 target design will be optimized to maximize the efficiency of Co-60 production in CANDU6 and more detailed analyses will be performed to characterize the new Co-60 production scheme.

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