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CANDU FUEL:
Evolution towards Optimal Performance

Neutronic Analysis of an In-core Co-6 Production Method in CANDU6 Reacto

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ABSTRACT –In this study, an innovative way of producing Co-60 in CANDU6 reactors is introduced and the neutronic analysis is performed. Unlike the conventional Co-60 production method, the central fuel element in the standard fuel bundle is replaced by a Co-59 target and Co-60 is obtained after the fuel bundle is discharged. Various lattice based studies are performed to determine the possible amount of Co-59 loading and to derive an optimal Co target configuration. Through the lattice depletion analysis, the achievable fuel discharge burnup as well as the possible Co-60 production capacity have been calculated with several neutronic assumptions. Major safety parameters including fuel temperature coefficient, void reactivity, and coolant temperature coefficient are also evaluated and compared with those of the standard CANDU6 core. For a more reliable investigation, the Serpent2-COREDAX-2 code system is applied to simulate the 3-D equilibrium core for a Co-loaded CANDU6 core. To compensate for the negative reactivity caused by the Co loading, some of the adjuster rods are removed to achieve the typical fuel burnup. Derived equilibrium core performances are evaluated in terms of fuel burnup and power profile. The power coefficient of reactivity is also evaluated for the 3-D equilibrium core. Also, in this work, Co-60 production in the 3-D time-average core is evaluated and compared with those obtained with simple lattice-based analyses.

Introduction

Co-60 is a synthetic radioactive isotope of cobaltwith a half-life of 5.27 years by emitting a beta particle with two energetic gamma rays. The high-energy gamma rays have fundamental applications in medical radiotherapy, industrial radiography, sterilization 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 by using a research reactor at the AECL Chalk River Laboratory. In their experiments, a Co-59 target was introduced into a research reactor where the Co-59 atom absorbed one neutron and then became Co-60. The same technology was adapted and applied in a CANDU6 power reactor later. The stainless steel adjuster rods are replaced with some neutronically-equivalent Co-59 adjuster rods in order to produce Co-60.With high neutron flux and optimized fuel burnup, CANDU6 reactors have a high Co-60 production rate in a relatively short time.

In this work, an innovative way of producing Co-60 in CANDU6 reactors, originally proposed by Dr. Park Y. W., is introduced and analyzed[1]. The fuel in central element is replaced by a Co-59 targettogether with other materials, and Co-60 can be obtained after the fuel bundle is discharged. To increase the production of Co-60, different target models are considered and compared with the conventional Co-60 production method. The core performance and characteristics are analyzed from the neutronic perspective through the fuel lattice analysis. The fuel depletion calculation is conducted with the continuous-energy

Monte Carlo code Serpent2[2]. Major safety parameters such as fuel temperature coefficient (FTC), coolant void reactivity (CVR) and coolant temperature coefficient (CTC) are also evaluated.

For a more reliable investigation, 3-D whole-core analyses are performed with a time-average core model for a CANDU6 reactor loaded with Co-59 using the coupled Serpent2-COREDAX-2 code system [3].The two-group homogenized cross-sections are calculated using the ~~Monte Carlo~~Serpent2 code. A stand-alone time-average module is developed to determine the time-average burnup distribution in the core for theso called 8-bundle-shift scheme fuel managementstrategy. In the whole-core analysis, the bundle power distribution is analyzed and the whole-core Co-60 production is also consistently evaluated. Besides, the power coefficient of reactivity (PCR) value is also analyzed, which is one of the most important safety parameters in the CANDU6 core.

1. Analysis Methods

1.1. Lattice analysis methods

In order to derive a favorable target configuration, four kinds of target modelsare considered, which are shown in Table 1-1, and two kinds of Co-59 target shapeareadopted, either rod-type or thin annular-type.

Table 1-1: Co-59 target models

Fuel type	ModelNo.	Center pin	Volume fraction of Co-59
Standard	0	Fuel (UO_2)	No Co-59
	1	Co-59 surrounded by graphite	2.70%
	2	Graphitesurrounded by Co-59	2.70%
CANFLEX	3	Co-59 surrounded bygraphite	2.70%

Note that Model 0 is just a conventional fuel design without any Co-59 loaded in the central element, as shown inFigure 1-1.

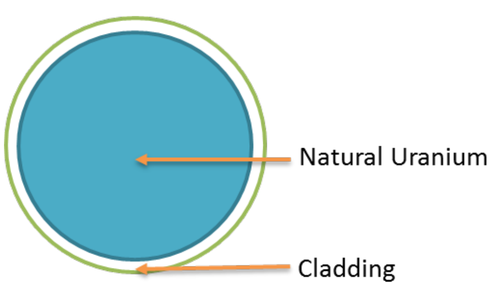

Figure 1-1:Structure of the centralfuel pin in Model0 (the standard CANDU fuel pin)

In Model 1, the UO_2 fuel in the central pin is replaced with a rod-type Co-59 target surrounded by graphite. The graphite is selected for the reason that it has a small neutron absorption cross section. As shown in

Figure 1-2 (left), in this model, the radius of Co-59 cylinder is 0.1 cm, which means that the Co-59 target occupies around 2.7% of the total natural uranium fuel volume in Model 0.

For comparison, as shown in Figure 1-2 (right), a thin annular Co-59 target is considered in Model 2, in which Co-59 surrounds the central graphite and the volume fraction of the annular Co-59 target is also restricted to be 2.7% as in Model 1. It is expected that in this model the annular Co-59 target will absorb more neutrons since its self-shielding effect is smaller than that in the compact cylindrical target of Model 1.

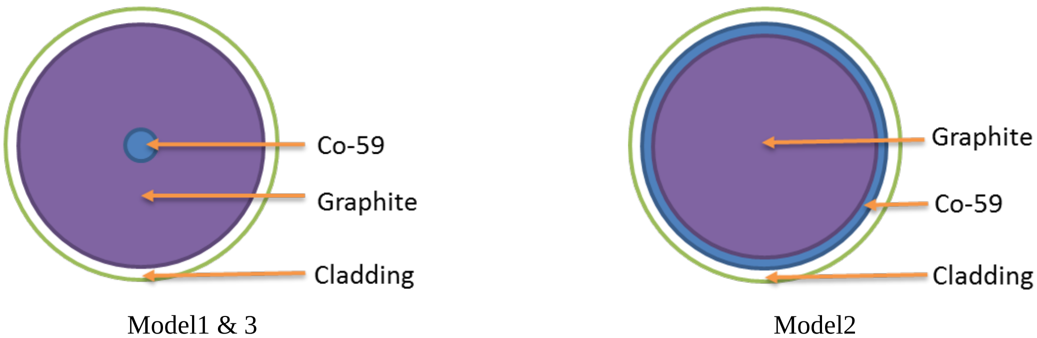

Figure 1-2Structure of the central pin in Model1 & 3 (left) and Model 2 (right)

The CANFLEX fuel design is also considered in this study for comparison since this design can improve the operating and safety margins as well as the economics of CANDU6 reactors. The central pin structure is similar to that of Model 1, except for the radius of Co-59 and graphite. In order toget the same volume fraction of Co-59 as Model 1, the radius of Co-59 target is adjusted to be 0.1045 cm, which is a little larger than that of Model 1 since the radius of the pins in the inner two rings of the CANFLEX fuel design is larger than that of the standard CANDU6 fuel bundle.

In this study, the 3-D single lattice depletion calculation and preliminary evaluation of safety parameters are performed using ~~the Monte Carlo~~Serpent2~~code~~with evaluated nuclear data from ENDF/B-VII.1 [4] and JEFF 3.2. In the lattice analysis, the fuel and coolant temperature profiles are derived for 100% power conditions [5], which are based on the whole core power and temperature distribution calculated by coupling 3-D RFSP [6] and NUCIRC codes [7].

To calculate the fuel temperature coefficient, the effective multiplication factor values, $k_{eff}$is calculated at six fuel temperatures ~~for the FTC evaluation~~ and then they are fitted into an empirical formula [8]. The coolant temperature coefficientisalso calculated at a set of discrete temperatures based on the definition.~~For the evaluation of the CTC of the CANDU6 lattice, only the temperature and density of coolant are changed with all other parameters fixed at the nominal values.~~The CVR is calculated at hot full power condition by changing coolant density from nominal value to the voided value.~~ as follows:~~

$$\cancel{\alpha_{void}(mk) = \left[\frac{1}{k_{nominal}} - \frac{1}{k_{voided}}\right] \times 1000 \qquad (1)}$$

1.2. Whole core analysis methods

Because of the non-line refueling of CANDU6 reactors during the operation, the time-average model is used to obtain an equilibrium core which neglects daily variations due tothe loading and discharge of fuel. In this study, the time-average model is applied to COREDAX-2 code, which is a three-dimensional rectangular (x-y-z) geometry diffusion nodal code based on the Analytic Function Expansion Nodal (AFEN) [9] method and developed at Korea Advanced Institute of Science and Technology (KAIST).

1.2.1 CANDU core modeling for COREDAX-2

To generate the irradiation-dependent two-group homogenized cross section parameters for COREDAX-2, a three-dimensional fuel latticeis depleted using~~the~~Serpent2~~code,~~ which is also used to calculate the cross sections for all kinds of reactivity devices using the detailed 3-D fuel bundle model~~s~~. The incremental cross sections due to reactivity devices are calculated from the below definition:

$$\Delta\Sigma_{T,x} = \Sigma_{T,x} - \cancel{\Delta}\Sigma_{A,x}, \tag{1}$$

where$\Delta\Sigma_{T,x}$ is an incremental cross section of reaction *x*, $\Sigma_{T,x}$ is a lattice cross section with reacitivity device of reaction *x*, and $\Sigma_{A,x}$ is a lattice cross section without reacitivity device of reaction *x*. In the rectangular geometry of COREDAX-2, 26 * 26 * 14 nodes are needed to describe 3-D CANDU reactors. As shown inFigure 1-3, this modeling includes 380 fuel channels, 2 layers of radial reflectors, and 1 layers of axial reflector at the top and bottom. It should be noted that the top two and botton two channels have vault water (light water) as the outer layer of radial reflector. Considering very high nertron absorption cross sections of the light water compared to heavy water, the impact on the core is not very small at all. Also, axial reflectors are homogenized including the very complex on-power fuel reloading mechanism.

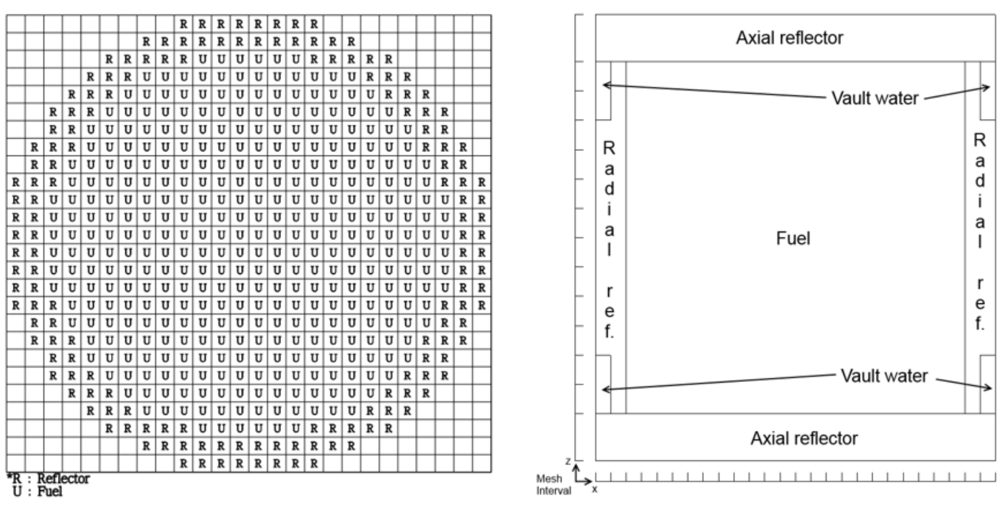

Figure 1-3 Radial (left) and axial (right) configuration of the CANDU core modeling in COREDAX-2

1.2.2 Time-average module

Since the fuel loading changes everyday in CANDU6 reactors, an 'average'or 'quilibrium' core model is developed for a representative reactor physics analysis. In the so-called 'time-average' core for CANDU6 reactors, lattice cross sections are averaged over residence time of the fuel bundles. In this study, the time-average module is developed for COREDAX-2 code based on the standard method used in the RFSP-IST code. The resulting time-averaged lattice cross-section of reaction *i* at "channel number *j* and axial position *k* within that channel" is defined as follows [10]:

$$\Sigma_{i,jk}(\mathrm{t.av.}) = \frac{1}{\omega_{out,jk} - \omega_{in,jk}} \int_{\omega_{in,jk}}^{\omega_{out,jk}} \Sigma_{i,jk}(\omega) d\omega, \quad (2)$$

where $\Sigma_i(\omega)$ is the irradiation-dependent cross section *i*, $\omega_{in,jk}$ and $\omega_{out,jk}$ are the incoming and outgoing values of irradiation at location *jk*.

The final solution of Eq. (2) is obtained as a result of the iterative calculation between Eq. (2) and the whole-core diffusion equation. The flux distribution from COREDAX-2 code is used to calculate the incoming and outgoing irradiations in Eq. (2) and the lattice-wise time-average cross sectionsare used in the COREDAX-2 code. COREDAX-2 and the time-aver module are coupled using a Windows batch script.

2. Reactivity compensation by removing adjuster rods

In the multiplication factor calculation, the total number of cycles is 600, which includes 100 inactive ones. The number of the neutron histories is 500,000 in each cycle. As result, the standard deviation of the effective multiplication factor during the depletion is less than 4 pcm. Figure 2-1 shows the evaluation of the reactivity over the fuel burnup for the four models. It indicates that replacing the fuel in the center pin with a Co-59 target does reduce the reactivity quite noticeably. This is simply because a non-fuel Co-59 absorber replaces the central fuel and the absorption cross section of Co-59 is much higher than that of natural uranium in the thermal and resonance energy range.

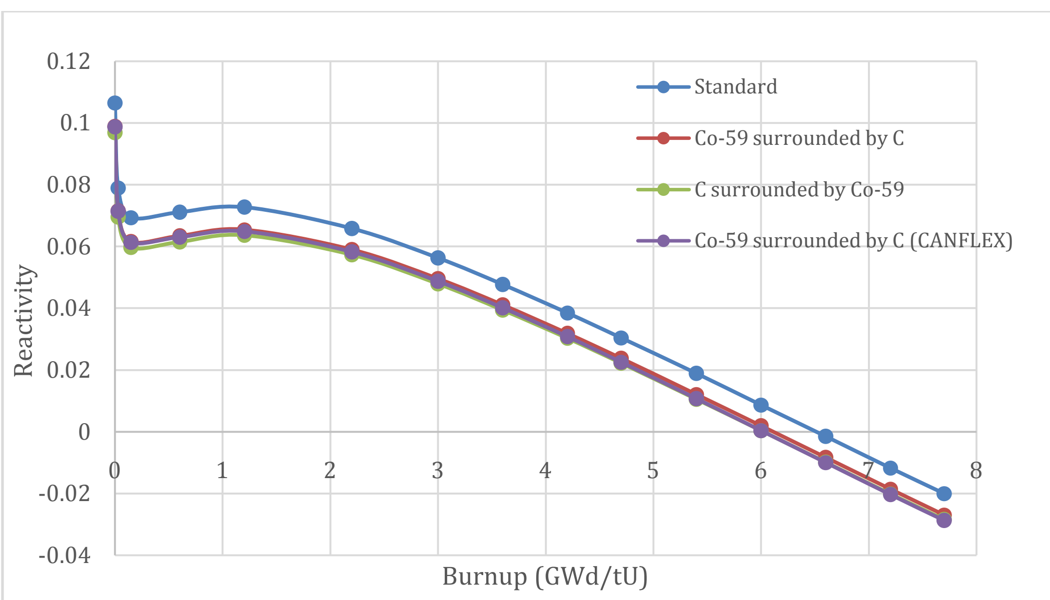

Figure 2-1: Reactivity value as a function of burnup

In ~~the~~ CANDU6 lattice analysis, the approximate discharge burnup is determined by the non-linear reactivity model:

$$\int_0^{B_d}(\rho_{(B)} - \rho_{loss})dB = 0, \tag{3}$$

where$\rho_{(B)}$ is the lattice reactivity as a function of burnup; ρ_{loss} is the reactivity loss due to the leakage and parasitic neutron absorption in the core; B_d is the fuel discharge burnup [11].

Due to Co-59 loaded in the central pin, the achievable discharge burnup of the natural uranium fuel would be lower than 7.2 GWd/tU, which is the conventional discharge burnup of standard CANDU6cores. Without increasing the initial enrichment for the fresh fuel, a unique methodology is implemented in this work to maintain the typical discharge burnup even in the Co-loaded fuel. In this study, removal of part of the adjuster (ADJ) rods from the CANDU6 reactors is proposed to pay for the reactivity reduction by the cobalt loading.

In onestandard CANDU6 reactor, there are 21 ADJ rods and all of them are basically stainless steel neutron absorbers, which are usually fully inserted into the core during the normal operation of CANDU6 reactors [12]. The total reactivity worth of the adjuster rods is about 16 mk and this positive reactivity is required to compensate for a large negative reactivity due to xenon buildup ~30 minutes after a reactor trip at full power equilibrium state. However, this use of adjuster rods has become not practical nowadays since nuclear reactors cannot be quickly restarted after a sudden reactor trip due to more stringent regulations. In some Canadian CANDU reactors, some of the ADJ rods werealready removed from the core to maximize the fuel utilization [13]. Taking into account the current experiences with the ADJ rods, only few adjuster rods are kept in the core for adjusting the power shape.

Table 2-1 shows the discharge burnup for several values of the reactivity loss. It is obvious that, with 3,100 pcm reactivity loss, which means most adjustor rods are removed from the core, the discharge burnup is still larger than 7.2 GWd/tU. It indicates that limited amount of Co-59 can be loaded into the central pin without compromising the fuel discharge burnup.

Table 2-1:Discharge burnup with different reactivity loss

Bundle Type	ModelNo.	ρ_{loss} (pcm)		
		4,300	3,700	3,100
		B_d (GWd/tU)		
Standard	0	7.17	7.96	8.74
	1	6.32	7.12	7.90
	2	5.97	6.8	7.60
CANFLEX	3	6.07	6.88	7.67

3. Results and Discussions

3.1. Three-Dimension Single Lattice Analysis

3.1.1. Co-60 production capacity

Nowadays, in the standard CANDU6 core ~~loaded with natural uranium~~, 8-bundle-shift scheme fuel management is applied and around 1.9 channels are reloaded every day. Therefore, the Co-60 production capacity can be easily calculated with the Co-60 mass at the discharge burnup of 7.2 GWD/tU. Table 3-1 shows the annual production of Co-60 in gram withits activity in curie.

Table 3-1: Productionof Co-60

Fuel type	ModelNo.	Per bundle	Per year	Radio-activity produced per year
		Gram	Gram	Curie
Standard	0	0	0	0
	1	0.5408	3000.4	3.30E+06
	2	0.6755	3747.6	4.12E+06
CANFLEX	3	0.6014	3336.3	3.67E+06

From Table 3-1, it is clear that the maximum amount of Co-60 can be achieved in Model 2, in which the annular Co-59 target surrounds a graphite cylinder. This is mainly because the thermal neutron flux is higher at the edge of center pin than that in the central region of the pin due to smaller self-shielding effect.In all four models, the radio-activity produced per year is over 3 million curie every year.

By increasing the radius of Co-59 rod in the central pin, the discharge burnup is reduced due to more neutron capture. Meanwhile, the production of Co-60 increases. To achieve the targeted discharge burnup (7.2 GWd/tU), the radius is calculated to be 0.135 cm and the Co-60 production is 0.97 gram per bundle. So the radio-activity produced is around 5.92 million curie per year, which is almost twice of the production in Qinshan Nuclear Power Plant [14]. It means that this new method has a high potential and can bevery competitive.

3.1.2. Safety parameters calculation

The fuel temperature coefficient (FTC) is calculated for all~~the~~ Co-59 targetmodels by using the Serpent2 code and is comparedwith that of the standard fuel bundle without Co-59. In order to make the standard deviation smaller than 1 pcm, 1.2 million histories for 150 inactive cycles and 1000 active cycles are used in the Serpent2 calculation.The FTC as a function of fuel temperature is shown inFigure 3-1. It is clearly observed that there is no significant changes in the FTC withthe Co loading in the lattice.

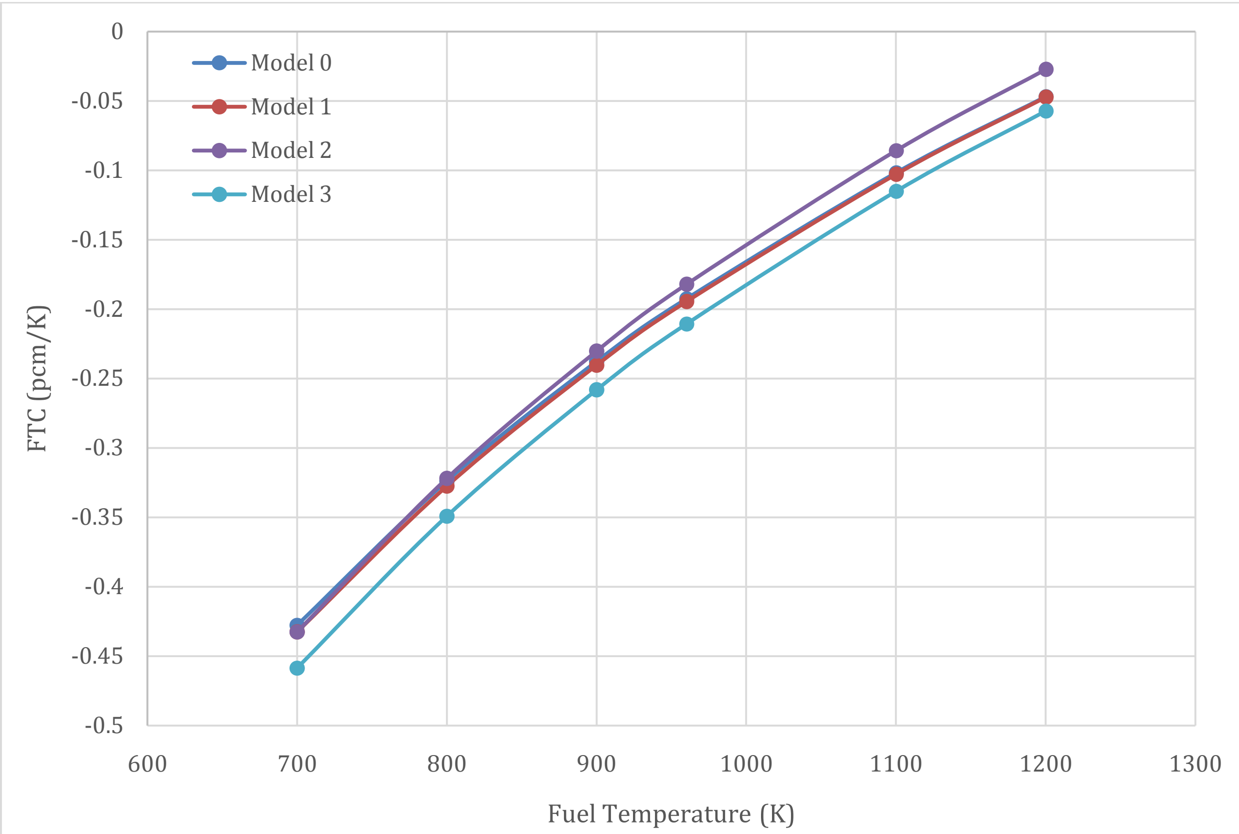

Figure 3-1: Fuel temperature coefficient with temperature

Unlike the FTC calculation, the CTC is evaluated for a set of discrete temperatures based on the definition. For the evaluation of the CTC of the CANDU6 lattice, only the temperature and density of coolant are changed with all other parameters fixed at the nominal values. To calculate the CTC, 3-D lattice-average coolant temperature is used ~~for calculation~~ with 1.2 million histories in 150 inactive and 1000 active cycles to makethe standard deviation of multiplication factor is around 1 pcm. The CTC calculation results of the four models are shown in Figure 3-2. It also shows that cobalt loading does not significantly affect the CTC.

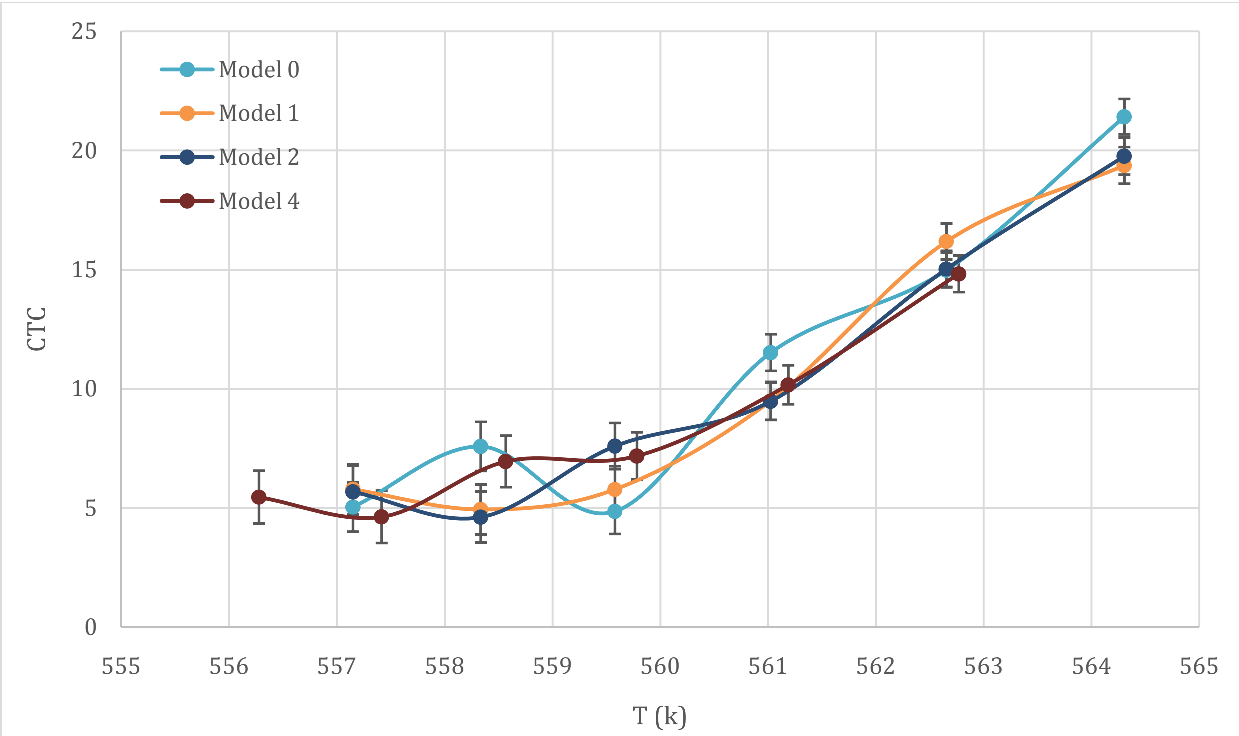

Figure 3-2: Coolant temperature coefficient with temperature

To see the impacts of the Co-59 loading in the fuel bundle on the coolant void reactivity (CVR), the CVR values are ~~also~~ evaluated for both the fresh and mid-burnup conditions of the fuel bundle. It should be noted that the mid-burnup condition lattice well represents the actual equilibrium core of the CANDU6 reactor. The CVR evaluation results are given in Table 3-2.

Table 3-2Impacts of Co-59 loading on the CVR

Fuel type	ModelNo.	CVR (Fresh)	CVR (Mid-burnup)
Standard	0	16.982mk	14.995mk
	1	16.415mk	13.889mk
	2	16.172mk	12.925mk
CANFLEX	3	17.442mk	14.130mk

FromTable 3-2, it is obviousthat the Co-59 loading noticeably reduces CVR both at zero burnup and mid-burnup. In the case of coolant loss, 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 due to weakerspecial self-shielding, and consequently more thermal neutrons are absorbed by Co-59.

3.2. Three-Dimension Time-average Core Analysis

3.2.1. 3-D Equilibrium core

The time-average core analysis is done using the COREDAX-2 code for Model 1, in which the central pin is filled with Co-59 surrounded by graphite. The time-average iteration is assumed to be converged when the maximum relative flux difference between twoiteration steps is smaller than 0.01%. As mentioned in Section 2, some adjustor rods should be removed to make the reactor critical. In this whole-core analysis, 17 adjustor rods areremoved and only fourcentraladjuster rods are utilizedfor adjusting the power shape.The resulting k-eff is around 0.999, which is very close to the critical state. However, removing adjuster rods results ina much higher peak channel power in some positions, so the optimization of targeted exit irradiation is also performed to flatten the channel power distribution. Figure 3-3shows the channel power distribution after the optimizing the exit irradiation pattern. It shows that all channel power isbelow6,700kW, which is much smaller than the allowable peak channel power (7,300kW).

	1	2	3	4	5	6	7	8	9	10	11	12	13	14	15	16	17	18	19	20	21	22
A									3267	3411	3511	3508	3409	3267								
B						2927	3524	4034	4258	4407	4458	4457	4406	4259	4035	3524	2928					
C					3290	3852	4449	4924	5224	5347	5328	5330	5351	5227	4926	4449	3852	3296				
D				3390	4047	4733	5318	5737	5978	6042	5975	5980	6049	5983	5740	5319	4734	4054	3399			
E			3258	4014	4773	5421	5924	6255	6417	6436	6356	6361	6444	6424	6260	5926	5423	4781	4024	3266		
F			3897	4637	5321	5851	6235	6470	6419	6425	6383	6390	6435	6428	6477	6241	5857	5330	4647	3906		
G		3546	4381	5141	5672	6064	6218	6391	6469	6501	6522	6529	6512	6479	6400	6227	6073	5680	5150	4389	3553	
H		3998	4841	5547	5944	6213	6298	6439	6510	6551	6589	6594	6562	6521	6450	6310	6225	5953	5555	4849	4005	
J	3322	4308	5236	5878	6172	6203	6384	6500	6539	6549	6557	6561	6558	6550	6511	6397	6216	6180	5885	5243	4316	3327
K	3542	4574	5525	6143	6389	6353	6486	6557	6554	6510	6447	6448	6515	6562	6568	6498	6365	6398	6151	5534	4582	3548
L	3704	4737	5695	6329	6605	6536	6587	6603	6561	6480	6380	6380	6483	6568	6612	6598	6547	6615	6339	5703	4745	3711
M	3707	4746	5717	6378	6683	6610	6628	6617	6561	6474	6372	6371	6476	6567	6625	6638	6620	6694	6388	5726	4754	3714
N	3550	4597	5580	6267	6607	6562	6592	6592	6554	6496	6428	6427	6498	6559	6599	6601	6572	6617	6277	5589	4604	3556
O	3325	4323	5280	5989	6376	6396	6471	6512	6517	6513	6517	6518	6516	6522	6519	6479	6405	6385	5998	5289	4330	3331
P		3987	4837	5563	5985	6246	6287	6394	6449	6485	6521	6522	6488	6453	6400	6294	6253	5993	5571	4845	3994	
Q		3501	4312	5038	5532	5913	6086	6274	6360	6399	6422	6423	6402	6365	6279	6090	5919	5538	5045	4317	3506	
R			3774	4437	5024	5539	5994	6283	6263	6283	6244	6245	6285	6267	6288	6000	5545	5030	4444	3780		
S			3114	3788	4446	5075	5648	6036	6231	6267	6190	6191	6269	6235	6041	5654	5081	4452	3793	3118		
T				3197	3807	4475	5079	5528	5794	5873	5812	5813	5876	5797	5532	5084	4480	3813	3202			
U					3123	3673	4265	4749	5064	5201	5192	5193	5203	5067	4753	4269	3677	3127				
V						2800	3387	3897	4133	4297	4368	4368	4299	4135	3900	3390	2803					
W									3176	3334	3449	3450	3335	3178								

Figure 3-3 Channel power distribution (kW)

3.2.2. Whole-core cobalt-60 production

Based on the irradiation for all bundles which are provided by COREDAX-2 code, the whole-core Co-60 production for one year can be calculated. COREDAX-2 code provides the burnup of all 12 bundles in each fuel channel. With 8-bundle-shift scheme, 8 bundles with higher burnup in one channel are replaced when discharging. Through calculation, the cobalt-60 production in acore per year is approximate 3,058 grams, which is around 2% larger than the previous lattice-based estimation.

3.2.3. <u>Power coefficient of reactivity</u>

In this work, to evaluate the power coefficient of reactivity (PCR), the effective multiplication factors are calculated at 7 power levels (60%, 70%, 80%, 90%, 100%, 110% and 120%).

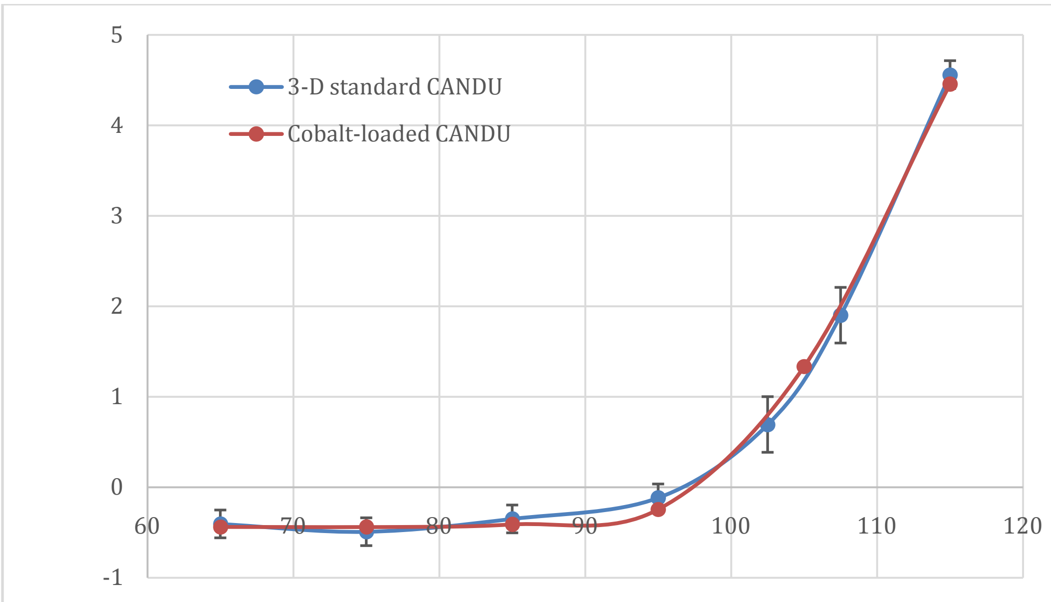

Figure 3-4 PCR evaluation results

The PCR evaluation results of standard CANDU reactor [15] and Co-loaded CANDU reactor are shown inFigure 3-4. As well known, the PCR is a combined effect of the FTC and CTC. As shown in Section 3.1.2, there is no significant effect due to cobalt loading in the FTC and CTC values in lattice analysis, therefore the difference of the PCR values of the two situations is also very small.

4. Conclusions

In this work, bothlattice and whole-core analysis are conducted to assess the feasibility of an innovative Co-60 production method in CANDU6 cores.

Through calculations and analyses, it has been shown that the reactivity reduction due to cobalt-loading can be appropriately compensated by removing some of the adjuster rods in the core. As a result, conventional discharge burnup (7.2 GWd/tU) can be obtained even in the cobalt-loaded CANDU cores. Several cobalt-loading models are analyzed and compared.~~And~~In theory the cobalt-60 production capability of all models are larger than that of the conventional method. This study also shows that the cobalt-loading ~~helps to~~ improves CVR a little ~~of the CANDU core~~and it has negligible impact on the FTC and CTC. Besides, zone-wise targeted irradiation should be adjusted to make the bundle power distribution acceptable in the Co-loaded CANDU6 core since some of the adjuster rods are removed. The Co-60 production in the time-average 3-D coreagrees well with that fromlattice analysis.

5. Future Studies

In this study, the Co-60 production capacity is calculated in terms of reactor physics. However, during the actual production process, in order to calculate the production capacity more accurately, more factors should be taken into consideration, e.g., the loss in process or transport.The Co-60 conversion rate of the new method is smaller than the value of the conventional method. Further work is necessary to improve the conversion rate. What's more, detailed technologies that are used to extract Co-60 from the bundle should also be developed in future study.

6. Reference

[1] Y. W. Park, "Manufacturing Method of Isotopes using CANDU6 Type Reactor" Korean Intellectual property office,10-1502414, 2015.
[2] Leppanen, J. Serpent2—A Continuous-energy Monte Carlo Reactor Physics Burnup Calculation Code User's Manual. VTT Technical Research Centre of Finland, 2013.
[3] Mohammad Abdul Motalab, Bumhee Cho, Woosong Kim, Nam Zin Cho, Yonghee Kim. Serpent2-COREDAX-2 Analysis of CANDU-6 Time-Average Model. 7th International Conference on Modelling and Simulation in Nuclear Science and Engineering (7ICMSNSE), Ottawa Marriott Hotel, Ottawa, Ontario, Canada, October 18-21, 2015.
[4] Chadwick, M. B., et al. "ENDF/B-VII. 1 nuclear data for science and technology: cross sections, covariances, fission product yields and decay data." Nuclear Data Sheets 112.12 (2011): 2887-2996.
[5] W. Kim, M. A. Motalab, J. H. Bae, Y. Kim, "Improvement of CANDU safety parameters by using the CANFLEX fuel", Vol. 94, P.789-801, 2016.
[6] W, Shen, and P. Schwanke, "Evolution of RFSP 3.5 for CANDU Analysis", Proc. Of the 33rd Canadian Nuclear Society (CNS) Annual Conference, Saskatoon, Saskatchewan, Canada, June 10-13, 2012.
[7] M. F. Lightstone, "NUCIRC-MOD1.505 User's Manual," Atomic Energy of CANADA Limited, 1993.
[8] R. Dagan, B. Becker and D. Roubtsov, "Evaluation of the CANDU6 Neutron Characteristics in view of Application of the Resonance Dependent Scattering Kernel in MCNP(X)," Progress in Nuclear Science and Technology, vol. 2, pp. 782-787, 2011.
[9] Noh, Jae Man, and Nam Zin Cho. "A new approach of analytic basis function expansion to neutron diffusion nodal calculation." Nuclear science and engineering 116.3 (1994): 165-180.
[10] W. Shen, "RFSP-IST Version REL_3-04: Theory Manual", SQA DOCUMENT, SQAD-06-5058, December, 2006.
[11] Driscoll, Michael J., Thomas Joseph Downar, and Edward E. Pilat. The linear reactivity model for nuclear fuel management. American Nuclear Society, 1990.
[12] "Design Manual: CANDU6 Generating Station Physics Design Manual, Wolsong NPP 1," 59RF-03310-DM-000, Revision 0, Korea Power Engineering Company, 2009.
[13] D. Gho, et al, "Feasibility Study on Optimization of Adjuster System for CANDU6," KHNP report, A07NJ23, 2007.
[14] Tang Jiongran, Independence Research Succeed in Co-60 Producing for CANDU6 Reactor, Process Report on China Science & Technology, 2009.
[15] Mohammad Abdul Motalab, Woosong Kim, and Yonghee Kim. "Evaluation of CANDU6 PCR (power coefficient of reactivity) with a 3-D whole-core Monte Carlo Analysis." Nuclear Engineering and Design 295 (2015): 127-137.