Super-Deep-Burn with the FCM (Fully Ceramic Microencapsulated) TRU Fuel in CANDU

Yonghee Kim[1], Donny Hartanto[1] and Fransesco Venneri[2]
[1] Department of Nuclear & Quantum Engineering, KAIST, Daejeon, Korea
[2] Logos Nuclear Systems, US

Abstract

For sustainable development of the nuclear energy, the spent fuel should be appropriately dealt with and one of the potential approaches is to transmute the TRUs (transuranics) in nuclear reactors. In this paper, transmutation of the PWR TRUs in the CANDU reactor has been studied from the reactor physics point of view. The extremely high neutron economy of the CANDU reactor results in a very high burnup of TRU fuels in CANDU and integrity of the highly burned fuel element is an open problem. In this work, a special nuclear fuel called FCM (Fully Ceramic Microencapsulated) is introduced to achieve a super-deep-burn of the TRUs. In the FCM fuel, the conventional TRISO particle fuels are randomly dispersed in a SiC matrix and it is expected that it can accommodate an extremely high burnup. The core performances and characteristics were analyzed with the fuel lattice analysis. The fuel depletion calculations were conducted by using the Monte Carlo method and the fuel discharge burnup was determined based on the non-linear reactivity theory. We evaluated the major safety parameters of the FCM-loaded core, which includes the fuel temperature coefficient, void reactivity, coolant temperature coefficient.

1. Introduction

In recent years, there has been an interest to utilize transuranic (TRU) nuclides from the Light Water Reactors (LWRs) spent fuel as a nuclear fuel. The TRU nuclides generated in the spent fuel are a big challenge in nuclear waste management due to the potential diversion of the material and their high radiological toxicity and long lifetime. The deep-burn concept [1,2] is one of the proposed concepts to deal with the TRU isotopes while waiting for the fast reactor availability.

In the original deep-burn concept, the transuranic nuclides are put inside the ceramic-coated (TRISO) fuel particles and burned in the high-temperature gas-cooled graphite-moderated reactor or HTGR. In the TRU vector from LWR spent fuel, the Pu fissile (Pu-239 and Pu-241) is usually over 50% and it can be utilized very effectively in thermal spectrums. Together with the low radiation damage by thermal neutrons and the high mechanical strength of the TRISO fuels, an extremely high burnup (~60%) is expected in the deep-burn of TRUs in the HTGR.

In this study, the CANDU reactor is used for the transmutation of the TRUs contained in the PWR spent fuel. CANDU reactor has a high neutron economy and it is highly expected that a super deep-burning of TRUs will be achieved in CANDU. The other features of the CANDU reactor also make it uniquely adaptable to an actinide-burning role: the small, simple fuel bundle simplifies the fabrication and handling of active fuels; and online refueling allows precise management of core reactivity. Previously, the TRU and Pu burning in CANDU were studied many times [3,4,5]. In particular, a deep-burning of TRUs was considered in Ref. 5 by using a SiC inert matrix fuel and it is claimed that a super deep-burn of over 70% burnup of TRUs can be achieved in CANDU.

A new type of TRU fuel is introduced for the transmutation of the TRUs in CANDU in this work. The so-called fully ceramic micro-encapsulated (FCM) fuel is used in this study. This fuel is originally proposed as the new type of nuclear fuel for LWRs, which serves for the transuranic nuclides destruction [6]. In FCM fuel, the TRISO fuel particles are randomly dispersed within a SiC matrix. The TRISO containing SiC pellets are clad with the conventional Zr alloy cladding. One of the potential advantages of the FCM fuel is that an extremely high burnup can be allowable in the CANDU core. Unlike the PWR fuel rod, the CANDU fuel pins do not have gas plenum and the clad is much thinner and a special fuel pellet design may be required to do a super deep-burning of TRUs in the CANDU core.

In this study, the calculation is performed by the Monte Carlo code McCARD [7]. The McCARD code was developed at Seoul National University, Korea. It can directly handle the double-heterogeneous fuel used in VHTRs. In particular, the randomness of the TRISO fuel particles can also be taken into consideration: locations of TRISO fuels are randomly determined. The McCARD code can be run on parallel computers. It also has a built-in depletion routine, thus it can be used in a stand-alone mode for the core depletion analysis. In a Monte Carlo depletion calculation, it is important to consider as many fission products as possible. In the current McCARD depletion calculation, all actinides and over 160 fission products nuclides are considered and the fission product poisoning can be accounted for almost completely.

2. Analysis methodologies and models

In this study, the calculation is done only for the 2-D lattice because the core characteristics can be well determined through lattice analyses in CANDU. The lattice calculation is performed for a 43-element CANFLEX fuel bundle, as depicted in Figure 1. The dimension and calculation conditions of the CANFLEX fuel bundle are listed in Table 1.

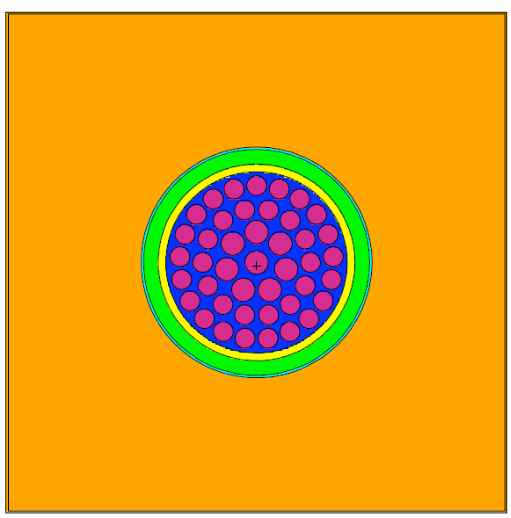

Figure 1. Schematic configuration of the CANFLEX fuel bundle.

The TRISO particles in the form of FCM fuel are located inside the fuel pin. The kernel diameter of the TRISO fuel is assumed to be 350 microns. The TRISO particle dimensions are listed in Table 2. Two values of TRISO packing fractions (PF) are considered in this work, 35% and 40%, respectively.

Table 1. Dimensions and calculational conditions of CANFLEX fuel bundle.

Inner-ring (center and first-ring) - Fuel pin (radius / temperature) - Cladding (radius / temperature)	 0.6350 cm / 960.16 K 0.6710 cm / 561.16 K
Outer-ring (second- and third-ring) - Fuel pin (radius / temperature) - Cladding (radius / temperature)	 0.5350 cm / 960.16 K 0.5680 cm / 561.16 K
Coolant area (radius / temperature)	5.17915 cm / 561.16 K
Pressure tube (radius / temperature)	5.61266 cm / 561.16 K
Gas annulus (radius / temperature)	6.44988 cm / 342.16 K
Calandria tube (radius / temperature)	6.58954 cm / 342.16 K
Pitch	28.575 cm

Table 2. TRISO particle dimension.

Kernel diameter	350 μm
Buffer layer (thickness/density)	100 μm / 1.05 $gr.cm^{-3}$
IPyC layer (thickness/density)	35 μm / 1.9 $gr.cm^{-3}$
SiC layer (thickness/density)	35 μm / 3.18 $gr.cm^{-3}$
OPyC layer (thickness/density)	40 μm / 1.9 $gr.cm^{-3}$

The TRU fuel vector given in Table 3 is taken from the spent nuclear fuel of LWRs that had been cooled for 30 years [5]. The TRU is in the form of $TRUO_2$ and the density is assumed to be 10.36 gr/cm^3. The SiC matrix density is assumed to be the same with the density of the SiC layer of the TRISO particle.

Table 3. TRU fuel vector.

TRU Composition, %			
Np-237	4.69	Pu-242	3.08
Pu-238	1.27	Am-241	9.99
Pu-239	56.27	Am-243	0.77
Pu-240	20.11	Cm-244	0.06
Pu-241	3.04	Total	100

3. Analysis results and discussion

The infinite multiplication factor (k-inf) was calculated as a function of burnup by using the ENDF/B-7 neutron library. The number of the neutron histories is 25,000 in each Monte Carlo cycle and the total number of cycles was 200 including 50 inactive ones. Figure 2 shows the evolution of the k-inf values over the fuel burnup. It is observed that the k-inf is above criticality up to around 435 GWd/MTHM with the two PF values.

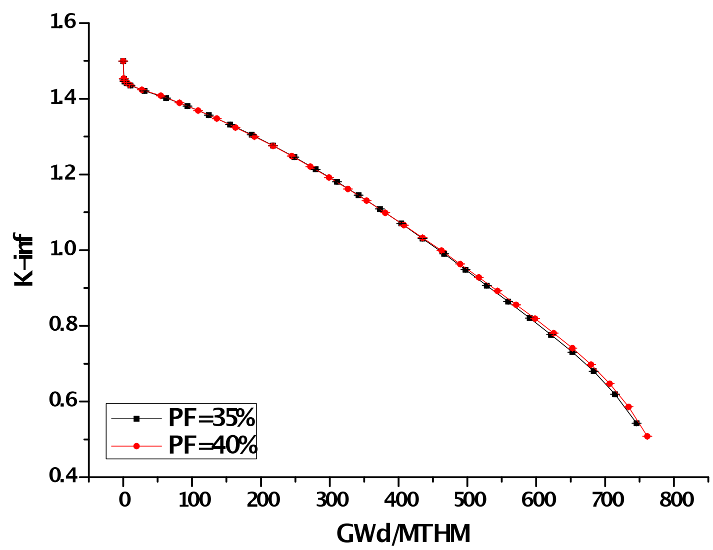

Figure 2. Infinite multiplication factor for regular kernel size.

In the CANDU lattice analysis, the discharge burnup is determined by the non-linear reactivity model [8,9]:

$$\int_0^{B_d} \left(\rho_{(B)} - \rho_{loss}\right) dB = 0 \qquad (1)$$

It is assumed that the reactivity loss is 4.3%, the same value as in CANFLEX with recovered uranium lattices [8]. The get the discharge burnup, the reactivity along burnup is plotted and the equation of $\rho_{(B)}$ can be found from the polynomial fitting. In this work, we have used sixth order polynomial fitting. By solving the equation (1) numerically, the discharge burnup is found to be about 708.53 GWd/MTHM with PF of 35% and around 714.26 GWd/MTHM with PF of 40%. Assuming a linear power of 12.94 kW/cm [5], the fuel residence time is equal to 684 days and 788 days for PF of 35% and 40%, respectively. In the terms of the TRU mass, the discharge burnup corresponds to ~70.8% for PF=35% and ~71.5% for PF=40%. Thus, it is clear that a super deep-burn of the TRUs is feasible in the CANDU reactor. With the same TRU fuel composition, the achieved fuel burnup in Ref. 5 was about 71%, which is quite similar to the burnup with PF=40% in this work. In Ref. 5, the fuel burnup was achieved by assuming a 3% neutron leakage or 3% reactivity loss. With the smaller reactivity loss of 3%, the available fuel burnup with the FCM fuel should be increased to ~72.2% for PF=35% and ~72.8% for PF=40%.

In Fig. 3, the composition variation with burnup is provided for several transuranic elements and Fig. 4 shows the change of the Pu vector as a function of burnup for PF=40%. In the case of Pf=35%, the results are very similar to those for PF=40%. As it is well known, the Pu content decreases rather quickly with burnup. Also, one can see that the total amount of Am and Np decreases slowly with burnup. However, the total mass of Cm accumulates. Figure 4 shows that destruction of Pu-239 is almost complete at discharge. However, Pu-242 increases noticeably due to the neutron capture by Pu-241.

The transuranic nuclides consumptions near the discharge burnup for the two cases are shown in Figure 5. As shown in the figure, the destruction rate is higher in the outer ring of the fuel bundle. This is because the thermal neutron flux is highest in the outer-most ring, which is closest to the moderator region. In the case of PF=40%, the peak discharge burnup is as high as ~75% in the outer-most fuel ring.

Figure 6 shows the normalized fission power distributions at the fresh condition, mid-burnup, and discharge burnup with PF=40%. Because the fissile isotopes in the outer ring deplete faster than in the other ring, the normalized fission power is higher in the fresh fuel and it should slowly decreases with burnup. It is observed that the peak pin power still occurs at the outer ring at mid-burnup and the peak power takes place at the central fuel pin at discharge.

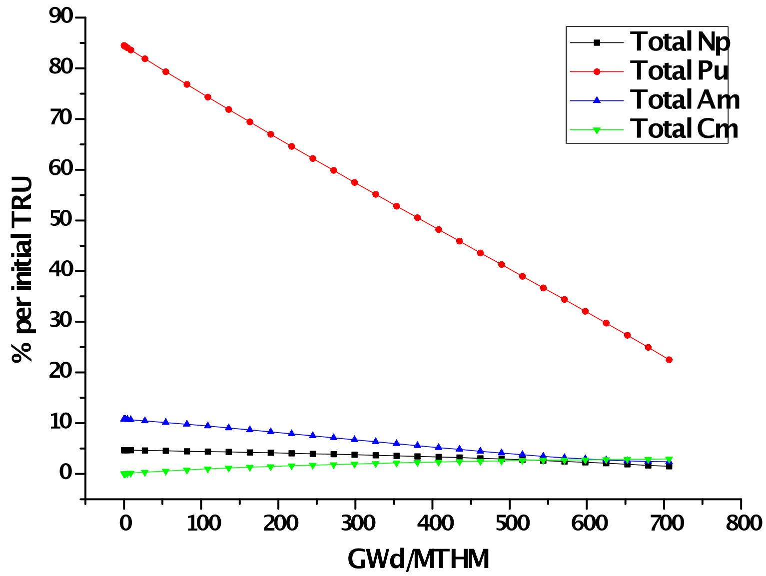

Figure 3. Transuranic composition vs. burnup for regular kernel size with PF=40%.

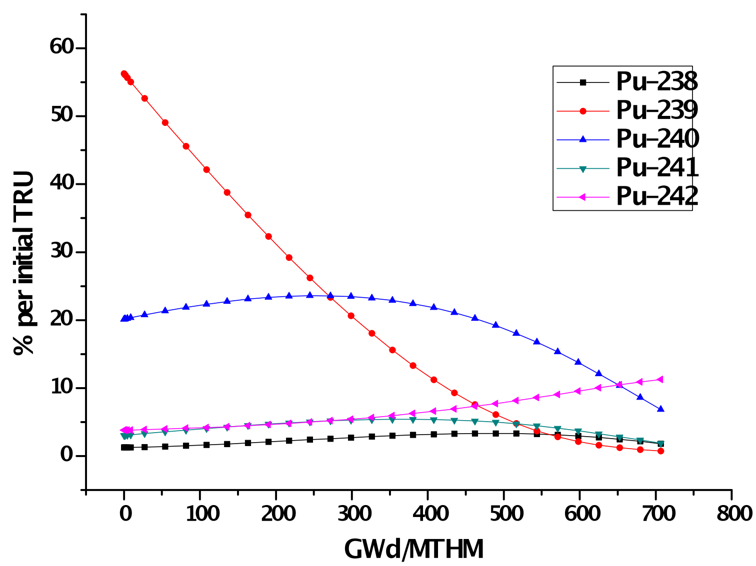

Figure 4. Plutonium composition vs. burnup for regular kernel size with PF=40%.

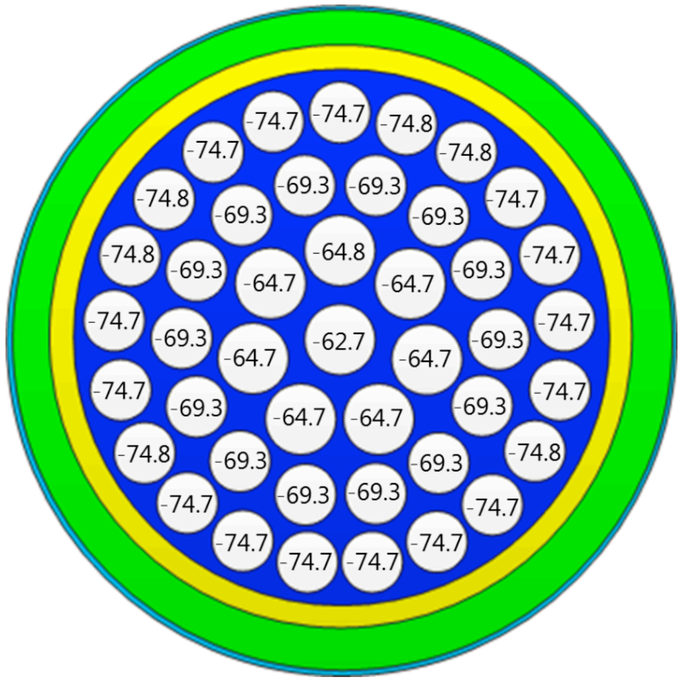

Figure 5. TRU discharge burnup distribution for regular kernel size with PF=40%.

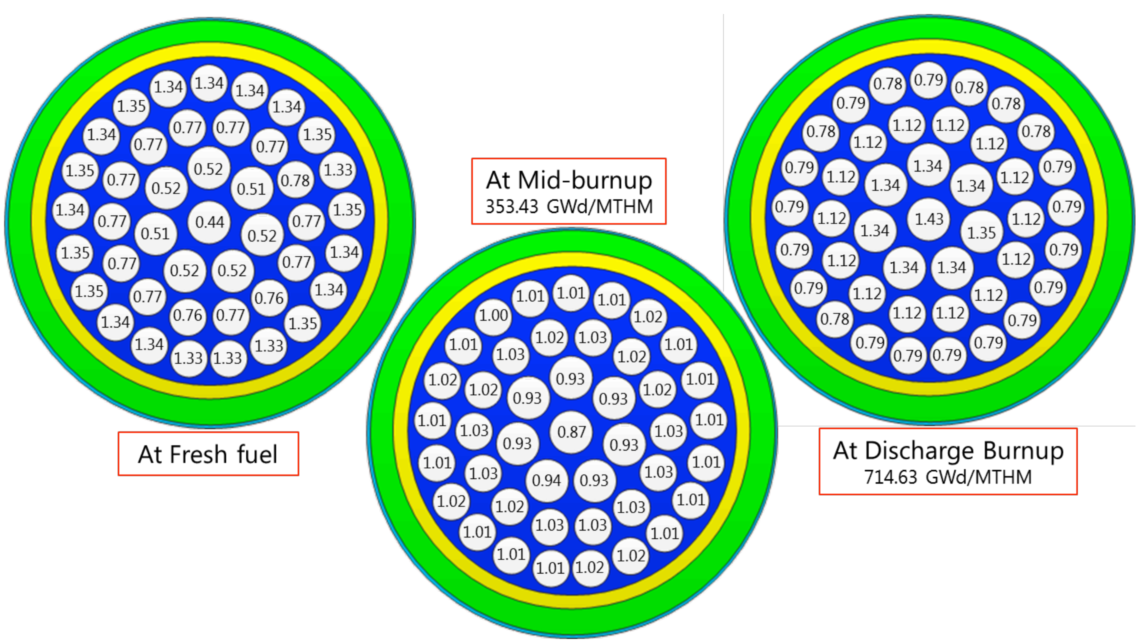

Figure 6. Normalized fission power distribution for regular kernel size PF=40%.

Three safety coefficients have been calculated for the FCM fuel. The reactivity coefficients were calculated at the mid-point of discharge burnup or equilibrium core condition. For the one with 35% packing fraction, the effective delayed neutron fraction β_{eff} is 3.30983E-03 and the prompt neutron generation life time ℓ is 6.76591E-04 second. For the one with 40%, β_{eff} is 3.38227E-03 and ℓ is 6.45911E-04 second. It is clear that the current TRU-loaded cores have a smaller delayed neutron fraction and a shorter neutron lifetime due to the Pu fuel.

The fuel temperature coefficient (FTC) was also evaluated at the mid-burnup condition for the temperature range of 300 K to 1200 K. The FTC is calculated to be negative as shown in Table 4. It should be noted that the FTC of the FCM fuel is clearly negative due to the strong resonant absorber such as Pu-240. In the FTC calculation, we have only changed the fuel kernel temperature, i.e., heavy metal only. In the actual FCM fuel, the SiC matrix temperature should also vary when the fuel kernel temperature changes. However, the temperature change of the SiC matrix should be smaller than that of the fuel kernel temperature. The thermal conductivity of the SiC matrix is rather high and the fuel temperature should be relatively low in the FCM fuel. However, the temperature profile within the fuel pellet is not known at this moment. In this work, we just investigated the impact of the SiC matrix temperature on the lattice reactivity by change the whole temperature of the fuel pellet and the results are provided in Fig. 7. One can note that the reactivity increases with the pellet temperature. As shown in Table 4, the temperature coefficient of the kernel itself is negative and it can be said that the temperature coefficient of the SiC matrix is strictly positive. The volume of the SiC matrix decreases with the fuel packing fraction and the FTC is negative. Therefore, the fuel packing fraction needs to be maximized in order to improve the reactivity coefficient of the fuel pellet.

Table 4. FTC for regular kernel size.

Temperature [K]	FTC (pcm/K) PF=35%	FTC (pcm/K) PF=40%
800 – 900	-0.16769 ± 0.040	-0.21103 ± 0.040
900 – 1000	-0.16013 ± 0.040	-0.19548 ± 0.040
1000 – 1100	-0.18308 ± 0.040	-0.14080 ± 0.040
1100 – 1200	-0.12210 ± 0.040	-0.18781 ± 0.051

Table 5. CTC for regular kernel size.

Temperature [K]	CTC (pcm/K) PF=35%	CTC (pcm/K) PF=40%
500 – 600	3.47116 ± 0.040	3.70145 ± 0.051

The coolant temperature coefficient (CTC) was also calculated for the fuel and the results are provided in Table 5. In the evaluation of CTC, the coolant density was also varied, determined by assuming saturated conditions for corresponding coolant temperature. The CTC is calculated to be positive, but it is substantially smaller than in the conventional CANDU core. The coolant void reactivity (CVR) was also calculated and it is plotted in Fig. 8. The maximum CVR is 7.73 mk with PF= 35% and 9.65 mk with PF=40%. This value is smaller than CVR is much smaller than in the conventional CANDU core.

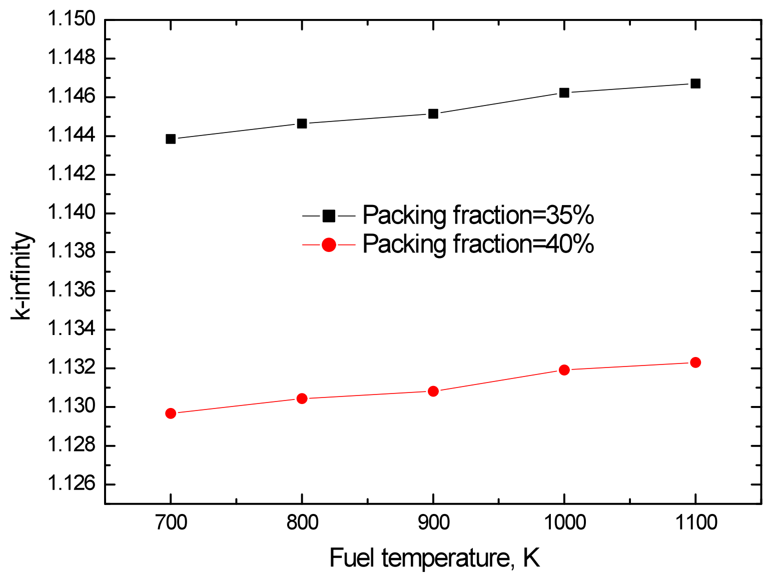

Figure 7. Impact of the SiC matrix temperature on the reactivity.

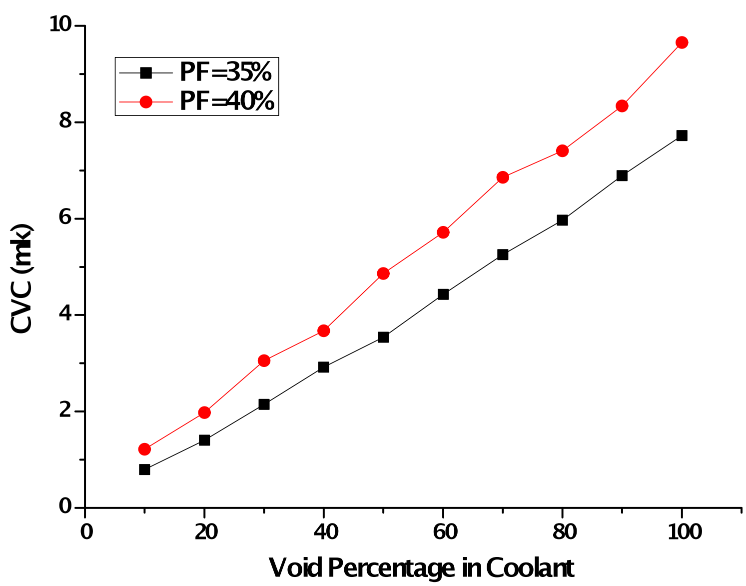

Figure 8. The coolant void reactivity for regular kernel size.

4. Conclusion

A super deep-burn of TRU concept in CANDU reactor has been investigated by using the FCM type of fuel. We have shown that the average discharge burn-up can be as high as ~715 GWd/MTHM with a kernel diameter of 350 microns. In the case of the regular kernel size, the higher packing fraction gives higher discharge burnup due to less amount of silicon in the core. Therefore, it is desirable that the packing fraction of the fuel should be maximized.

Three safety parameters namely the fuel temperature reactivity coefficient (FTC), the coolant temperature reactivity coefficient (CTC), and the coolant void reactivity (CVR), have also been calculated for the fuel. The inherent safety parameters using TRU fuel is much better than using natural uranium. The FTC is negative, the CTC and the CVR are less positive than in the conventional CANDU loaded with natural uranium. Nevertheless, it should be mentioned that uncertainties associated with the reactivity coefficients are not small and accurate evaluation of the parameters should be performed in the future. In particular, the impact of the SiC matrix on the reactivity feedback coefficients should be evaluated in an accurate way.

An optimization to reduce the excess reactivity at BOL and to reduce the positivity of the coolant temperature and coolant void reactivity coefficient using the burnable poison will be considered in a future study.

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