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Design of a Multi-Spectrum CANDU-Based Reactor, MSCR, With 37-Element Fuel Bundles Using SERPENT Code

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Abstract

The burning of highly-enriched uranium and plutonium from dismantled nuclear warhead material in the new design nuclear power plants represents an important step towards non-proliferation. The blending of these highly enriched uranium and plutonium with with uranium dioxide from the spent fuel of CANDU reactors, or mixing it with depleted uranium would need a very long time to dispose of this material. Consequently, considering that more efficient transmutation of actinides occurs in fast neutron reactors, a novel Multi-Spectrum CANDU Reactor, has been designed on the basis of the CANDU6 reactor with two concentric regions. The simulations of the MSCR were carried out using the SERPENT code. The inner or fast neutron spectrum core is fuelledbydifferent levels of enriched uranium oxides. The helium is used as a coolant in the fast neutron core. The outer or the thermal neutron spectrum core is fuelled with natural uranium with heavy water as both moderator and coolant. Both cores use 37-element fuel bundles. The size of the two cores and the percentage level of enrichment of the fresh fuel in the fast core were optimized according to the criticality safety of the whole reactor. The excess reactivity, the regeneration factor, radial and axial flux shapes of the MSCR reactor werecalculated at different of the concentration of fissile isotope ^{235}U of uranium fuel at the fast neutron spectrum core. The effect of variation of the concentration of the fissile isotope on the fluxes in both cores at each energy bin has been studied.

Keywords: burning actinides, MCNP, multi-spectrum CANDU reactor, MSCR, coupled reactor, CANDU reactor, and average burnup.

1. Introduction

The possibility of coupling a fast-thermal nuclear reactor system for burning actinides is presentedin as a Multi-Spectrum CANDU Reactor (MSCR). The SERPENT three-dimensional continuous-energy code [1] has become a common tool for calculations of the criticality factor and burnup calculation. It can also be used for the calculation of core physical parameters such as the power distribution, neutron flux, and kinetic parameters in addition to fissile material depletion. The code capabilityis verifiedfor a full core calculation and mutispectrum reactor system[2][3]. The burnup validation of the SERPENT code has been done with MCNP6 and SERPENT with the design of a full-core CANDU reactor [4]. The SERPENTcodewas chosenbecause it has a general modelling and continuous energy cross section capabilities. The continuous energy cross section is substantial because this eliminates the need for collapsing multi-group cross sections. The advantage of the SERPENT code is that it provides an excellent computational time in comparison to MCNP6 for the same tally calculation and burnup analysis.

A design model of the MSCR is created. The coolant and materials of the pressure tube, pressure vessel, and the filling of the fast neutron spectrum core were designated according to the function of the fast core. The thicknesses of the pressure tube, pressure vessels are optimized and calculated according to the pressure difference across them. The reactivity and the regeneration factors were calculated at different concentrations of the fissile isotope ^{235}U in the fuel of the fast core to examine the criticality safety and the non-breeding condition. The behaviour of the radial and axial flux distributions in the full core reactor are shown and investigated at different enrichment concentrations of ^{235}U in the fast core. These behaviours were investigated at each energy bin, thermal, epithermal and fast neutron spectrum, in both the internal fast spectrum core and external thermal spectrum core.

The multi-spectrum reactor calculations have been done using parallel computer clusters in the high performance computer virtual lab, HPCVL [5]. For the optimization of the number of histories, the number of nodes used was 12 nodes and 64Gb shared memory. While, in the flux calculations the number of paralel node are

24 node with 120 Gb shared memory tha time of calculations were between about 40 hours depending.

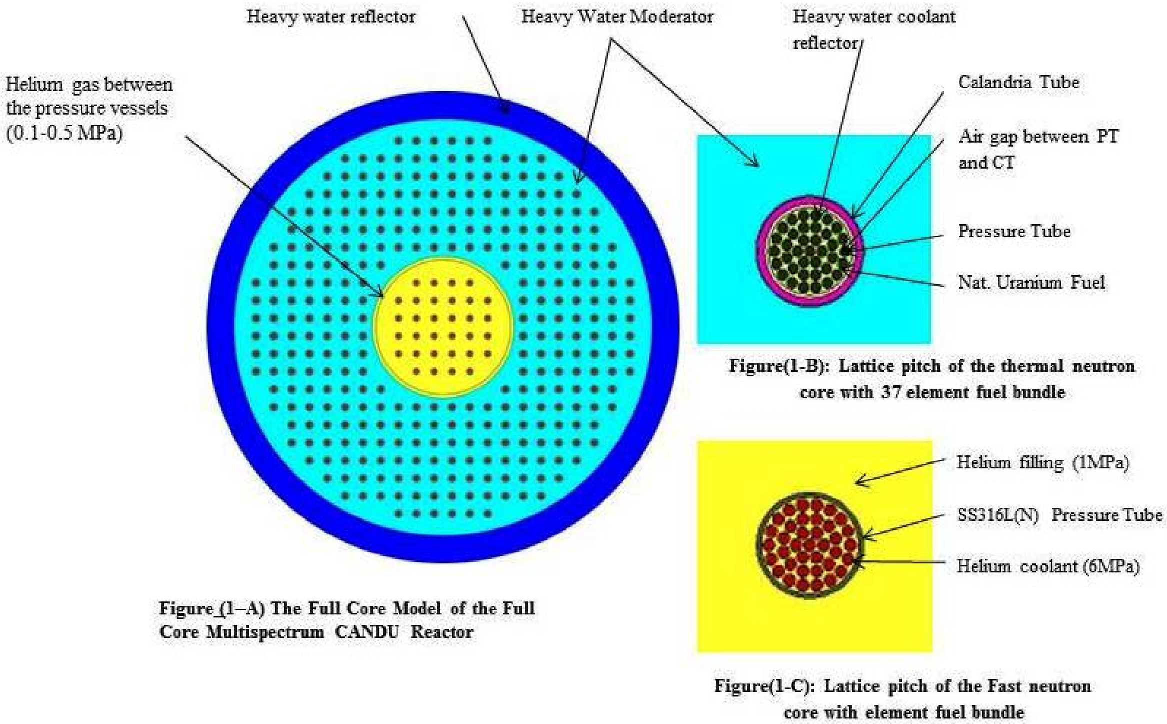
2. The design of Multi-spectrum CANDU reactor, MSCR.

2.1.The layout description of the MSCR

The multi-spectrum CANDU base reactor is a two neutron spectrum reactor that can be used for burning actinides and for the production of electricity. The multi-spectrum CANDU reactor consists of two concentric cores as shown in Figure(1-A). The internal fission region or the inner core is constructed to be dedicated for the fast neutron spectrum. Inbrief, the fast core refers tofast neutron spectrum region or internal core. The lattice pitch of the fast core is shown in the Figure (1-C).

The external fission region or the outer core has the same traditional CANDU structure, so the externalcoreis dedicated to thermal neutron spectrum as a predominant flux. The lattice pitch of the thermal core is shown in the Figure (1-B).

Inbrief, the thermal core refers toa thermal neutron spectrum, region or external core. The two cores, fast and thermal, are separated by stainless steel pressure vessels. The pressure vessels are separated from the low-pressure helium gas to minimize the diffusion of neutrons from the fast core to the moderator of the thermal core that may affect the moderation quality in the thermal core. The structure, materials, and fuel of each core are as presented in Figures(1-A, B, and C) are explained in the following sections.



Figure(1-B): Lattice pitch of the thermal neutron core with 37 element fuel bundle

Figure (1–A) The Full Core Model of the Full Core Multispectrum CANDU Reactor

Figure(1-C): Lattice pitch of the Fast neutron core with element fuel bundle

Figure 1 The design model of the multispectrim CANDU reactor

2.2.The fast neutron spectrum core

2.2.1. Coolant and filling material of the fast core

A) Fast core coolant

The choice of coolant in the fast reactor is dictated by the desire to that introduce a small or negligible amount of absorption and moderation while still being able to remove reliably the heat from this high power density configuration of the fast core. In general, common coolants used in a fast nuclear reactor are: (1) Liquid metal such as sodiumand Pb-Bi Lead(44.5%)-Bismuth(55.5%) or Eutectic alloy where sodium is the most common use as coolant; and (2) a gas coolant. The most common choices for gas-cooled fast reactors are helium, supercritical CO_2, and steam [6].

The use of Pb-Bi and Sodium as coolants in the MSCR was avoided because both Sodium and Bismuth (at high-temperature 350^o C) have substantial chemical reactivity with water which is used as a coolant and moderator in the external thermal core [7]. So, to avoid any inconvenient chemical reaction in case of an accident and save on the requirement for a high pumping power as needed for sodium or Pb-Bi liquid metals coolant, a liquid metal coolant is avoided.Even though most of the gas coolants are composed of light isotopes, the amount of moderation is limited because of the small number density of gas coolants,where most of the gas coolant isotopes have low neutron scattering and capture cross sections. At high temperatures, steam could be reactive with the stainless steel of the bundle sheath, pressure tube, and pressure vessels. The CO_2 suffers from radiolytic dissociation producing free oxygen and carbon monoxide. Oxygen at high temperature will react with stainless steel structure. Consequently, steam and CO_2 are avoided as a possible coolant in the fast neutron spectrum core of the MSCR. Therefore, helium represents the best choice as the coolant for the fast core of the MSCR.

The advantages of choosing helium as a gas coolant for the fast core are[6]: (1) Chemical compatibility with heavy water in case of accidents and leaks between the fast and thermal cores, obviating the need of an intermediate coolant loop, and generally good chemical compatibility with structural materials; and (2) Negligible activation of helium which has only two isotopes: ${}^{4}_{2}\mathrm{He}(99.999863\ \mathrm{a}/0)$ (which has a negligible capture cross section) and

${}^{3}_{2}\mathrm{He}(0.000137\ \mathrm{a}/0)$ (which has reasonable high (n,p) reaction but due to the very small atomic percentage of this isotope this could be inconsiderable); (3) Optically transparent, simplifying fuel shuffling operations and inspection; (4) Helium gas coolants cannot change phase in the core, reducing the potential of reactivity swings under accidental conditions; (5) Helium gas coolants generally allow a harder neutron spectrum, which increases the burnup potential in the fast core reactor; and (6) Helium gas coolants have a low number density which can allow a larger coolant fraction in the core without an unacceptable increase in parasitic capture. The latter property provides the opportunity to increase the helium coolant pressure, consequently increasing the efficiency or heat removal from the core.

B) Filling material of the fast core and the gap between the pressure vessels.

To obtain the highest burnup potential in any reactor, parasitic absorption should be minimized. This translates into the choice of a very tightly packed core in which the volume fractions of structural materials and coolant are kept to a minimum. For reasons of economics and fuel cycle material per unit time characteristics, it is generally desirable to have the highest possible burnup required. Thus, the reactor core is designed to have a very high fast neutron flux level. This high fast neutron flux level generally translates into a very high fast fission rate and power density. It should be noted that the high power density in a fast reactor is a result of design choices rather than an innate feature of this type of reactor.

To make the fast neutron the predominant spectrum in the fast core, the filling materials should have a very small moderation power and transparent to thermal, epithermal and fast neutrons. The perfect choice to match these characteristics is the helium gas. Helium is therefore used as a filling material in the fast cure at a pressure of 1MPa. The use of helium as a filling material leads to the major fission reaction with the fissile materials in the fast core taking place in the fast flux region where the fast core can be used to burn actinides. To avoid heat transfer betweenthe fast core and the thermal core the two stainless steel pressure vessels are separated with a low pressure (0.1MPa) helium gas. Besides the advantages of a low pressure gas as a thermal insulator, helium is used as a coolant, filling material and insulation as facilitated with three separate pumps to keep the pressure of each zone at its proper designated level.

C) The material of fast core bundle, pressure tube and pressure vessels.

The recommended material for the bundle, pressure tube and pressure vessel of the fast core is Stainless Steel 316L(N) [8] rather than Zircaloy. Since the fuel used in the fast core has different enrichment levels, the temperature of the sheath may reach 900 K. Zircaloy is not workable at a high-temperature fast reactor. Stainless Steel 316 SS grade is very applicable for a fast reactor operating at high temperatures.The Type 316 SS alloys are more resistant to general corrosion and pitting/crevice corrosion than the conventional chromium-nickel austenitic stainless steels such as Type 304 SS. Type 316 SS also offer higher creep, stress-rupture and tensile strength at elevated temperatures. [9][10].

The choice of 316L(N) grade of stainless steels is decided by several important factors such as high temperature (about 900^{o}K), mechanical properties like creep, low cycle fatigue and creep-fatigue interaction, compatibility with helium coolant, weldability, fabricability and cost. Type 316L(N) austenitic stainless steel containing 0.02-0.03 wt.% carbon and 0.06–0.08 wt.% nitrogen is the current choice of material for high-temperature sodium-cooled fast reactors (SFRs)[11]. Increasing the nitrogen content from 0.06-0.08 wt % to levels of 0.12-0.14 wt% has been found to increase the creep rupture life of 316L(N) SS by an order of magnitude[12]. Nitrogen in the 316L(N) alloy adds additional resistance to sensitization in some circumstances. The nitrogen content of 316L(N) stainless steel also provides some solid solution hardening, raising its minimum specified yield strength compared to 316L stainless steel[9]. The element contents of the Stainless Steel 316L (N) are shown in Table (1).

Table 1:Elemental composition of the Stainless Steel 316L(N)[9]

Reactor components	Material	Element	Mass ratio wt.%
Fuel bundle, Pressure tube, Internal pressure vessel External pressure vessel ,	Stainless steel 316L (N) Density= 8 g/cm^3	Fe	65.441
		N	0.14
		C	0.025
		Mn	1.74
		Cr	17.57
		Mo	2.53
		Ni	12.15
		Si	0.2
		S	0.004
		P	0.2

i. Bundle used in the fast core

The bundle used in the current fastFast core of MSCR is a traditional 37 fuel element bundle. The geometrical properties are the same as one used in the CANDU reactor but the material used is stainless steel 316L(N)[8] rather than Zircaloy-4.

ii. Pressure tube

One of the advantages of choosing the stainless steel 316L(N) is a high value of the maximum

allowable stress of this type of alloy at high temperature. According to Equation(1), the thickness of the pressure tube or pressure vessels is inversely proportional to the maximum allowable design stress. The minimization of the pressure tube and pressure vessels thickness is important to decrease the effect on the neutron flux.
The pressure tube material is stainless steel 316L(N) because it has a higher maximum allowable pressure at high temperature. It also has excellent material properties for fast reactors. The thickness of the pressure tube is calculated according to the pressure difference across it. The pressure of the helium gas coolant inside most traditional gas-cooled fast reactors is around 5 to 7 MPa [6]. In the current MSCR reactor model, the helium coolant pressure is set to 6 MPa, which can affect the stainless steel pressure tube. The tube thickness can be calculated according to Equation (1)[10].

$$t = \frac{\Delta P \times R_{in}}{(S \times E) - (0.6 \times P)} \quad (1)$$

Where,
t: is the thickness of the pressure tube or pressure vessels (inch)
ΔP: pressure difference between inside and outside the pressure tube or pressure vessel (psi)
R_{in}: inside radius of the tube (inch)
S: Maximum Allowable design stress (psi),
E: Joint coefficient, (supposed to be 1)
The pressure inside the pressure tube is set at 6 MPa and outside it is 1MPa therefor ΔP=5MPa. The temperature of the pressure tube is around 650K. The corresponding maximum allowable design stress is 17050 psi which can be found from tables [13],

iii. Pressure vessels

The external thermal neutron core and the internal fast neutron core are separated by two stainless steel pressure vessels. These two pressure vessels are separated with 5 cm thickness of low-pressure helium gas (0.1 Mpa to 0.5). This low-pressure gap avoids heat transfer from the fast core to the moderator of the thermal core and thus the amount of neutron moderation or thermalization efficiency in the thermal neutron core. The thickness of each pressure vessel is calculated using Equation (1), where ΔP is the pressure differential. The thermal and fast core specifications and materials are shown in Table 2 and Table 3.

iv. The fast neutron spectrum core fuel

The fuel of the fast core is uranium with different fissile material concentrations from 0.7% to 19.9%. Fuelis canned in a 37 fuel element bundle of stainless steel 316L(N). There is no need for the calandria tube because there is no moderator in the fast core. In the current paper the fuel used is uranium oxide, UO_2 of different enrichments of 0.7, 5.0%, 10%, 15.0% and 19.9%. The model avoids a 20% enrichment because it represents the non-proliferation threshold.

2.3. The thermal neutron spectrum core

In the external thermal neutron flux core, the major fission reaction is taking place with thermal neutrons where the thermal neutron flux is predominant in the outercore. The externalcore is the thermal core fuelled by natural uranium using 37 elements bundle Zr-4 with heavy water coolant and moderator and works as the main source of power generation. The external thermal corelooks exactly like a traditional CANDU 6 reactor structure but with lower numbers of fuel channels depending on the radius of the fast core. The lattice pitch of the fast core is set at 28.575 cm. In the current research, the radius of the fast core is chosen where the volume of the fast core is about a tenth of the whole reactor volume. The fast core also contains a tenth of the number of fuels channels in the thermal core. In other words, the radius of the fast core or the radius of the pressure vessels that separate the fast core and thermal core are chosen to minimize the unfuelled fuel channels. The geometric specifications of the MSCR are shown in Table 2.

Table 2: The Geometry and specifications of the fast and thermal cores

	Thermal core	Fast core
Radius	335 cm (379.73 cm -include reflector)	107.72
Number of fuel channels	320	32
Type of fuel bundle	37 elements	37 elements
Type of fuel	UO_2 ,natural uranium	UO_2, 0.7%, 5%,10%,15%,19.9%

The number of fuel channels in the fast core depends on the radius of the fast core and the lattice pitch of this core. The lattice pitch of the fast core is set at 28.575 cm as the same as the lattice pitch of the thermal core. The characteristics of fast core's material are presented in Table 3. In the current research, the MSCR model the lattice pitch will be kept constant.

Table 3: Materials of the fast neutron spectrum

Fast core components	Material	Density (g/cm^3)	Temperature K	Pressure (MPa)
Fuel	Nat. UO_2	10.6	1400	------
Cladding	SS 316L(N)	8	900	------
Coolant	He	6.618E-3	800	6.0
Pressure Tube	SS316L(N)	8	650	------
Filling fast core material	He	1.99E-3		1.0
Internal Pressure Vessel	SS316L(N)	8	340	------
Filling the gap between the pressure vessels	He	1.36E-4		0.1 to 0.5
External Pressure Vessel	SS316L(N)	8	330	-----

At least the fast core or both cores are independently subcritical but the diffusion of neutrons from one core to another will drive the whole reactor to criticality. The simulation of the multi-spectrum CANDU reactor design is done using the SERPENT burnup code. The details of both the fast and thermal cores are described in the next subsections.

3. Optimizing the Number of Histories.

As is common in Monte Carlo calculations, the relative error in the calculation is proportional to the inverse of the root of the number of histories. Therefore, as the number of histories increase the results yield will have a greater confidence. For a converged solution for the uniform flux Optimization of the mutiplication factor k_{eff} with the number of histories using Serpent Code

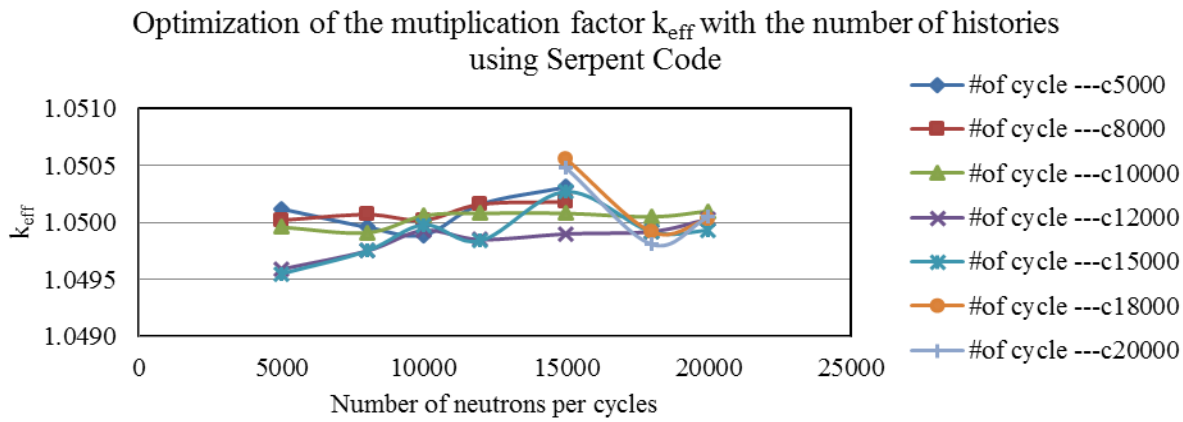


Figure 2 Optimization of the multiplication factor k_{eff} with number of histories using Serpent code

The convergence number of histories has been chosen according to the flux convergence value.The convergence value of the k_{eff} and the average total flux in the core are (1.051 ±0. 00003) and (3.055E+14 ±0. 00001 n/cm^2) at 15000n/15000 cycles respectively.The k_{eff} and the average total flux start to converge at a low number of histories at 10000n/cycle for 12000 cycle k_{eff} (1.051±0.0009) and the average total flux (3.056 ±0.0001). To save computational time, the chosen number of histories for the MCNP6 simulations were 12000n/cycle for 10000 cycles, and the number of theinactivecyclesis optimized at 200 cycles.

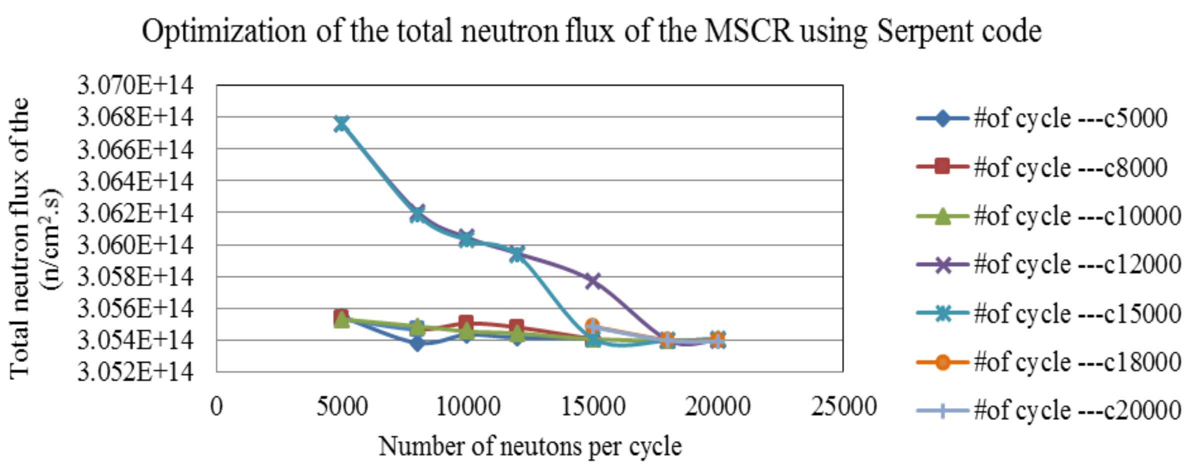


Figure 3 Optimizing of the average total flux in the core with the number of histories using Serpent code.

4.1. Calculation excess reactivity (ρ) and the Regeneration factor (η)

The MSCR consists of an internal fast core spectrum where the reactor is fuelled with different concentrations of fissile isotopes ^{235}U from 0.7% up to 19.9%. There are two reactor physics safety parameters that should be considered: 1) The criticality parameters or excess reactivity ρ, 2) The regeneration factor η, the number of nutron produced per nutron absorbed, to minimize the criteria for breeding.

4.1.1. Calculation of the Calculation Excess reactivity

The maximum excess reactivity of the traditional CANDU 6 reactor without the liquid zone controllers or any control devices is between 80 mk and 120 mk [14]. To model a safe design of the MSCR the excess reactivity should be less than or in between these two values. Figure (4) presents the changing of excess reactivity of the whole MSCR system with the concentration of the fissile isotope of the fuel in the fast core.

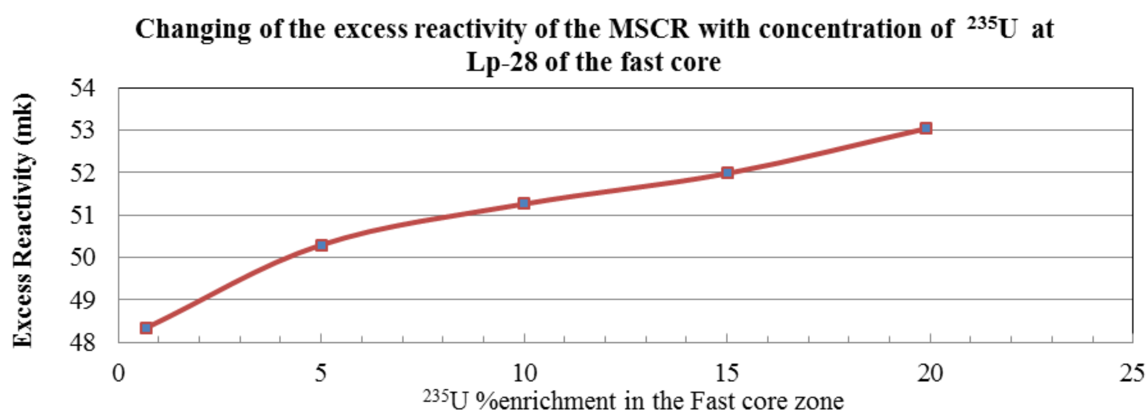


Figure 4 The variation of the excess reactivity of the MSCR with enrichment level of the fuel in the fast core zone.

From Figure (4) one can notice that increasing the concentration of the fissile isotope in the fast core is producing an increase in excess reactivity gradually from ~48.3 mk at natural uranium fuel up to 53.03 mk at 19.9%. Therefore, the MSCR is critically safe to be run because traditional CANDU reactor control devices could control these small values of excess reactivity.

4.1.2. The Regeneration factor (η)

The purpose of the MSCR design is to burn actinides. Consequently, the breeding of new fissile material is contrary to this aim in addition to any non-proliferation concerns. The design of the MSCR should avoid breeding but used to deplete actinide isotopes, so it is useful to apply this criterion to minimize breeding. The minimum criterion for breeding is a regeneration factor less than 2 [15]. This represents a conceptual guide but for a more3 complete analysis the breeding ratio should be calculated after the burnup calculation, which is outside the scope of this work.

The regeneration factor is calculated using the SERPENT code at a different enrichment concentration of the fuel in the fast core. The variation of the regeneration factor (η) with the different enrichment concentrationsis shown in Figure (5). One can notice that the calculated value of the regeneration factor (η) at a maximum enrichment in the fast core of 19.9% is 1.323, as required bycondition η<2. Thereforefor the current design, the MSCR can't be used for breeding.

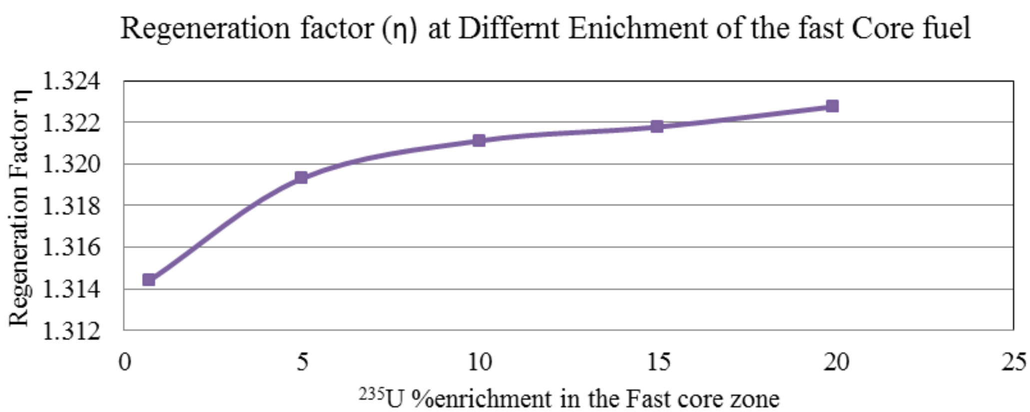


Figure 5 The variation of the Regeneration factor (η) of the MSCR with ^{235}U concentration of the fuel in the fast core zone.

4.2.Neutron flux distributions in the MSCR

The knowledge of the spatial dependent radial and axial neutron flux distribution Φ(r) and Φ(z) is of high importance in any reactor facility, e.g. in research reactors as entry value for many experiments (neutron source strength in irradiations) or in power reactors as a heat source distribution. In the MSCR, the radial and axial neutron flux distributions is important in order to know the power distribution and burnup in both the thermal and fast cores. The flux shape at each energy bin, thermal, epithermal and fast flux have been calculated in both the radial and axial directions. These fluxes are normalized for a constant power of 1280 MWth. The data library used for these calculations for the both reactor materials and fuel is ENFB-VII. In the next subsections, the radial and axial neutron flux distribution for the three energy bins are presented.

The average radial, axial and total fluxes were calculated for different concentrations of fissile isotope ^{235}U in the fast core. Figures (6, 8 and10 present the average radial fluxes at each radial fuel channel of the thermal and fast cores. Figures 7, 9 and 11 present the average axial fluxes at the central fuel channel of the fast core. Figures 12 and 13 present the total average of the radial and axial fluxes.

The average radial and axial fluxes were calculated for three energy bins: thermal neutron energy bin (1e-5 eV to 0.625 eV), epithermal neutron energy bin (0.625 eV to 0.1 MeV) and fast neutron energy bin (0.1 MeV up to 14 MeV). The cross section library used is END-F-VI.

In The Figures (6, 8 and10). 1)The region (A) outside of the two vertical blue dashed lines represent the thermal external core; 2) The region (B) between the two vertical red and blue dashed lines represents the region between the fast and the thermal core which is an non-fuelled area and contain the two pressure vessels and low pressure helium gas; and finally region (C) between the two vertical dashed red lines represent the fueling region of the fast core.

i) ***Thermal Neutron Flux***

Figure 6presents the variation of the radial thermal flux in both the external thermal core and internal fast core with enrichment concentrations of ^{235}U in the fast core. Figure 6shows the variations of the axial neutron flux within the central channels of the fast core.

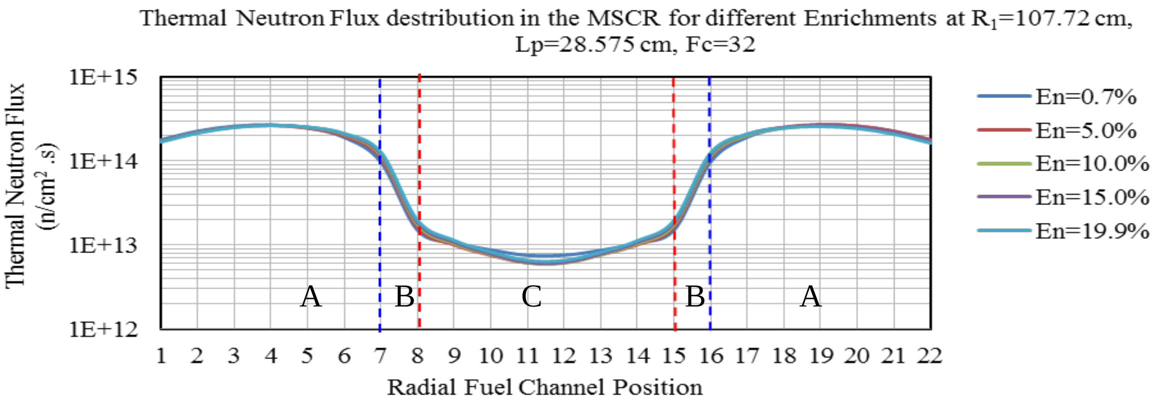


Figure 6 Changes of the thermal fluxes of MSCR with the radial positions at different enrichment concentrations of the fuel in the fast core.

FromFigure 6 one can notice that:

1) The thermal fluxes at a different enrichment of the fast core have the same shape, and then there isa very minor difference between them.
2) The thermal flux in thethermal core, region (A), core is a bit higher in the middle (2.65E+14) while at the edge of the thermalcore it is 1.68E+14 n/cm^2/s and 1.27E+14 n/cm^2/s because of the higher ratio of fuel to moderator.
3) In the region (B), all the thermal fluxes at different enrichments concentrations are dramatically dropping down because there is no fuel in this region. The only materials in this region are stainless steel and helium. The thermal flux behavior is almost the same for each enrichment concentration of the fast core.
4) In region (C) since there is fast core fuel with no moderation the thermal flux is still at a smaller value in comparison to the thermal core. At the center of the fast core, the lowestvalue of the thermal flux of 5.5E12 n/cm^2/s is reached for a 19.9% concentration of ^{235}U and 8.0E12 n/cm^2/s for the surrounding 0.7% concentrations of ^{235}U.

5) The thermal flux in both the thermal core and fast core are semi-uniform. Although there is a very small difference between the thermal fluxes at different enrichments, the differences in the fast core is higher in the center of the fast core, which decreases moving out to the internal edge of the thermal core due to the diffusion of some fast neutrons from the fast core to the thermal one.This diffused neutrons are formalized and absorbed close to the edge (blue dashed line) of the thermal core and explain why there is no difference as one proceeds towards the middle or the outer edge of the thermal core.

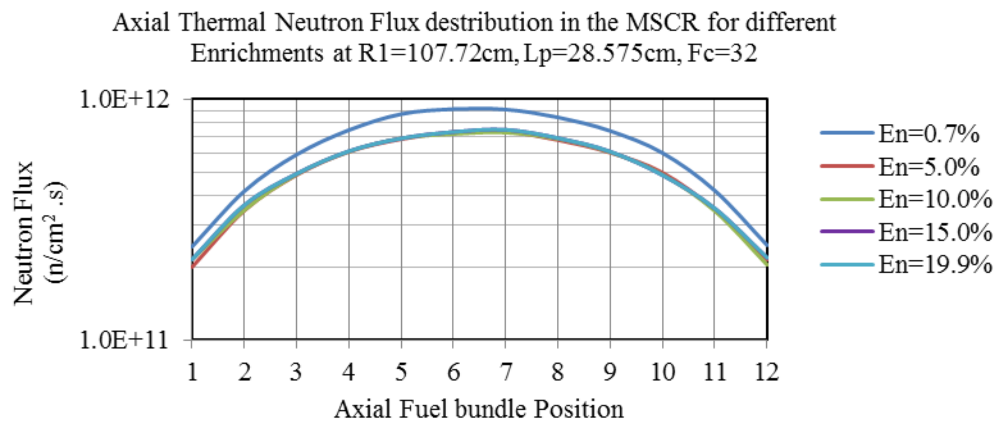


Figure 7 Changes of the thermal fluxes of the MSCR with the axial positions at different enrichment concentrations of the fuel in the fast core.

From Figure 7one notices that:

1) At different enrichment concentrations of the fast core, the flux shapes have the same trend
2) At natural uranium or 0.7% of ^{235}U the axial thermal flux is a bit higher than the other enrichment concentrations. As the concentration of the fissile isotope increases the axial fluxes decrease because of the higher absorption coefficients of the ^{235}U.
3) The sum of the all-axial fluxes at certain^{235}U concentrations is equal to the midpoint of the corresponding concentration in Figure 6.

ii) The Epi-Thermal Flux

From Figure 8 and Figure 9 present the radial and axial variation of the epithermal fluxes at different concentrations of the fissile isotope in the fast core of the MSCR.

Figure 8 one can notice that:

1) All the epithermal fluxes at various^{235}U concentrations in the thermal core region (A) are identical and have the same shape. The radial epidermal neutron fluxes come together in the middle of the thermal core. For different concentrations of fissile isotope, the epithermal fluxes

remain constant at most positions in the thermal core.

2) At the fuel channel positions number 6 and 17 the differences between the epithermal spectrum are increasing as they go toward the edge of the fast core. This increase in the

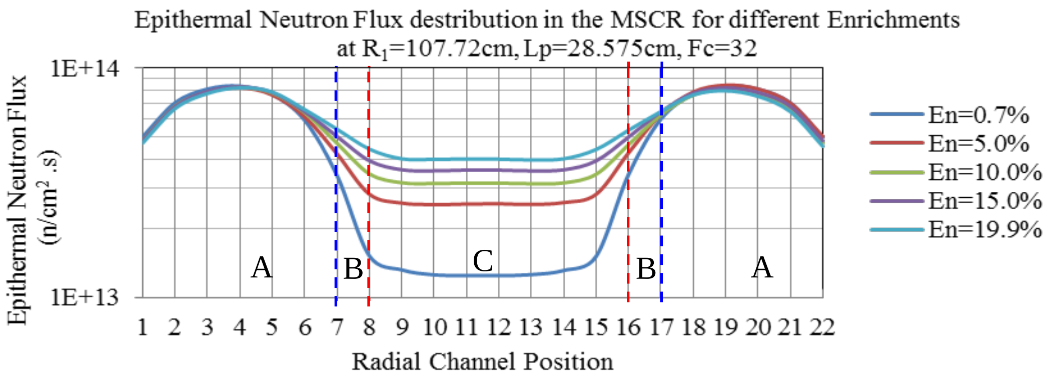


Figure 8 Changes of the epithermal fluxes of MSCR with the radial positions at different enrichment concentrations of the fuel in the fast core.

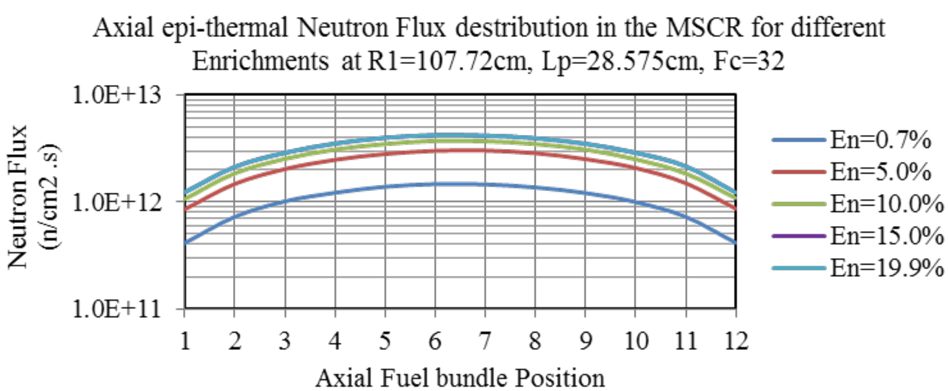


Figure 9 Changes of the Epi-thermal fluxes of the MSCR with the axial positions at different enrichment concentrations of the fuel in the fast core.

differences is because of the diffusion of the fast neutrons from the fast core to the thermal core. This diffusion increases with the increasing fissile isotope concentration.

3) At region (B), there is no fuel in this region and the difference between the different epithermal fluxes is increasing.

4) In the fast core region (C), the epithermal flux increases with fissile concentration due to an increasing fission rate. The epithermal fluxes sources are from the epithermalregion of the fission neutrons and the elastic collision of the fast neutrons with the structural material in the fast core.

5) Overall the average epithermal flux in the fast core is lessthan that in the thermal core.

6) At a given fissile isotope concentrations, the epithermal flux is flat in the fast core. This is very important because it will drive to the uniform burnup of the ^{235}U by fission in the resonance

region, but the ^{235}U (n,ɤ) reaction is also probable in this region which produces other actinides such as (^{239}Pu).

7) On the other hand, the epithermal neutron flux would be interacting with the ^{238}U through (n, ɣ) reactions to breed other fissile materials ^{239}Pu that will be burned. [14]

8) Future work will focus on decreasing the epithermal flux with a decrease in pitch between the fuel channels in the fast core.

From**Figure 9**, one can notice that:

1) The axial fluxes have been calculated as the central fuel channel number 11.
2) The axial epithermal neutron fluxes in the fast core are small when the ^{235}U concentration in the fast fuel is 0.7% (Nat. U).
3) As the concentration of the fissile isotope increases in the fast core the axial epithermal flux increases. At each concentration, the behaviour axial flux is identical.

iii) The Fast Flux

Figure 10presents the variation of the radial thermal flux in both the external thermal core and internal fast core with enrichment concentrations of ^{235}U in the fast core. Figure 10shows the variations of the axial neutron flux within the central channels of the fast core.

FromFigure 10, one can notice that:

1) In the thermal core region (A), all the fast fluxes at different ^{235}U concentrationsare identical and have the same shape. The radial fast neutron fluxes have a small separation in the middle of the thermal core where the minimum fast fluxes occur at the edges with a value of 2.0E13 n /cm^2.s to 3.0E13 n/cm^2.s in the middle of the thermal core.At the different fissile concentrations, the fast fluxes are almost identical at most positions in the thermal core.
2) At the fuel channel positions number 6 and 17, the differences between the fast spectrums are not significant because the diffused fast neutrons from the fast core to the thermal core is thermalized and mostly contribute to the epithermal flux.This explains the small difference in the fast flux right on the edge of the thermal core. This diffusion of fast neutrons increases with increasing of fissile concentrations.
3) At region (B), there is no fuel in this region and the only materials helium, and stainless steel have low fast neutron cross sections.
4) In the fast core region (C), as the fissile isotope concentration increases,there is an increasein the fission rate. The fast neutron fluxes sources are mostly from fast fission in^{235}U.
5) For the fissile isotope concentrations 0.7% and 5% of ^{235}U, the fast flux in the fast core is less than that in the thermal core.
6) For a fissile isotope concentration of 10% of ^{235}U, the fast neutron flux in the fast cure is the same as that in the thermal core.

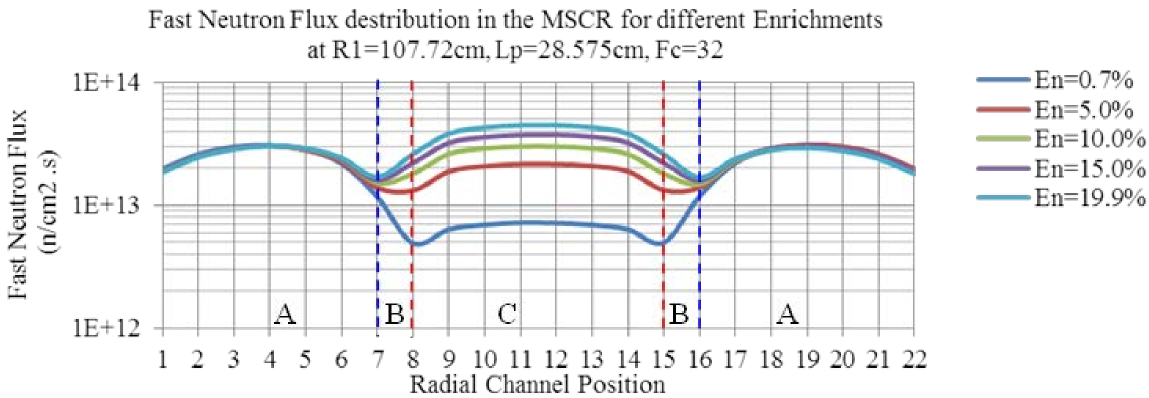


Figure 10 Changes of the fast fluxes of MSCR with the radial positions at different enrichment concentrations of the fuel in the fast core.

7) For the fissile isotope concentrations, 15% and 19.9% of ^{235}U, the fast flux in the fast core is higher than that in the thermal core. For actinide burning the fast neutron flux in the fast core needs to be higher than that in the thermal core, i.e., to increase the fast flux in the thermal core the number of fuel channel should be increased by decreasing the lattice pitch of the fast core to include more enriched all channels.

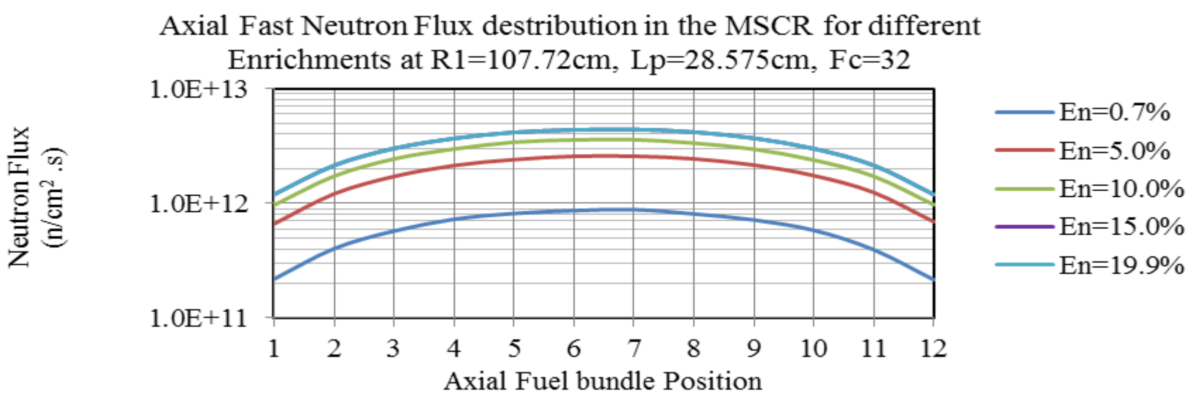


Figure 11 Changes of the fast fluxes of the MSCR with the axial positions at different enrichment concentrations of the fuel in the fast core.

From Figure 11, one can notice that:

1) The axial fluxes have been calculated as the central fuel channel number 11.
2) The axial fast neutron fluxes in the fast core are small at 0.7% (Nat. U).
3) As the fissile isotope concentrations increase so does the fast neutrons flux.
4) The axial flux shape has the same behaviour at different fissile isotope concentrations.

iv) The total Flux

Figure 12 and Figure 13 present the total radial and axial fluxes of the MSCR.

From Figure 12 one can notice that,

1) In the thermal core or region (A), the average total radial fluxes have the same shape and the values are identical andcome together in the middle of the thermal core.
2) In the fast core or region (C), the average total radial fluxes are proportional to the concentration of fissile isotope. The total flux is almost flat in the fast core for each concentration.
3) For each concentration of fissile isotope, the average total radial flux is less than that the average total flux in the thermal core.
4) The significant contributions to the total flux behaviour in the fast core are from the epithermal and fast neutron spectrum while in the thermal core the major contribution is from the thermal neutron spectrum.

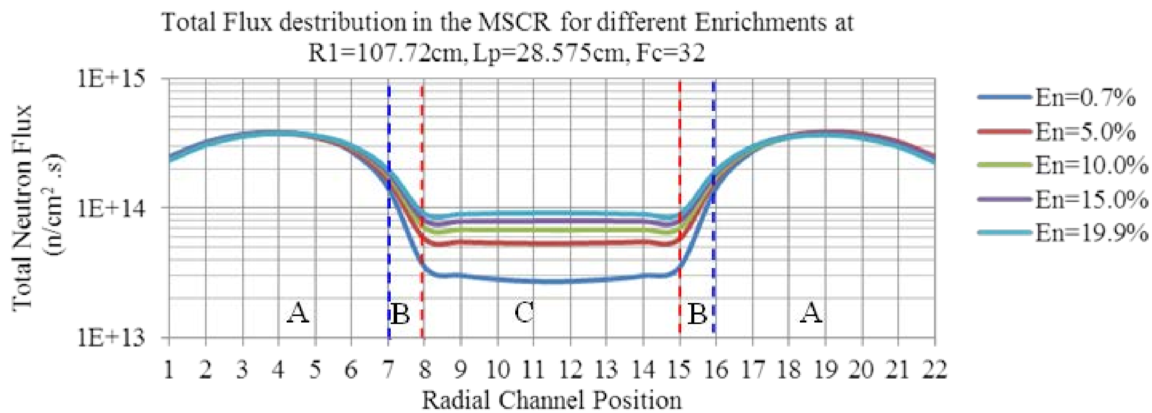


Figure 12 Changes of the Total fluxes of MSCR with the radial positions at different enrichment concentrations of the fuel in the fast core.

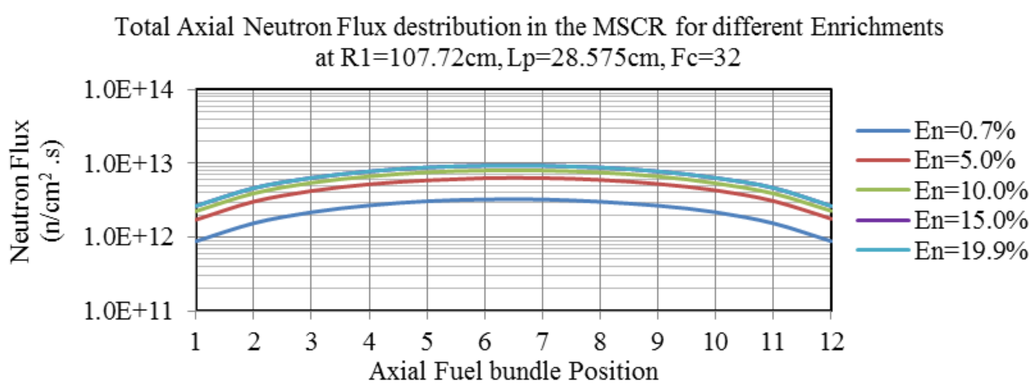


Figure 13 Changes of the Total fluxes of the MSCR with the axial positions at different enrichment concentrations of the fuel in the fast core.

From Figure 13 one can notice that:

1) The average total axial fluxes have been calculated per bundle where the mesh length is

equal to the bundle length and the mesh cross section equal to the fuel channel cross section fuel channel number 11.

2) The average total axial neutrons in the fast core are small at 0.7% (Nat. U) but increases with the fissile isotope concentration. The average total axial neutrons have the same shape at different fissile concentrations.

5. Conclusion

The design model of the multi-spectrum CANDU based reactor has been implemented using the SERPENT code. Helium gas is chosen as the coolant and filling material in the fast neutron core. Stainless steel has been selected as the structural material of the fast core. The two cores are separated by two pressure stainless steel vessels. The two pressure vessels are isolated with low-pressure helium gas.

The calculated reactivity was in the criticality safety margin. To prove the non-breeding activity of the reactor, a regeneration factor is calculated which is smaller than the threshold of breeding.

Finally, the radial and axial fluxes of the MSCR have been calculated at different fissile isotope concentrations in the fast core. There is no effect on the thermal neutron flux in both the thermal core and fast core with an increase in the isotope fiassile concentration. In the fast neutron core, both the epithermal and fast neutron fluxes are significantly increased with an increase in fissile isotope concentration, but the epithermal and fast neutron fluxes are almost identical in the thermal core.

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