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PRELIMINARY NEUTRONICS CALCULATIONS OF THE FIXED BED NUCLEAR REACTOR – FBNR

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http://www.rccg.urfgs.br/fbnr.htm

Content

1 Introduction

The Fixed Bed Nuclear Reactor (FBNR) is being developed under the IAEA Coordinated Research Project (CRP) on Small Reactors Without O-site Refueling (SRWOR) [IAEA Research Contract No. 12960/ Regular Budget Fund (RBF)].

The Small Reactors without On-Site Refuelling are defined by IAEA "As reactors which have a capability to operate without refuelling and reshuffling of fuel for a reasonably long period consistent with the plant economics and energy security, with no fresh and spent fuel being stored at the site outside the reactor during its service life. They also should ensure difficult unauthorized access to fuel during the whole period of its presence at the site and during transportation, and design provisions to facilitate the implementation of safeguards. In this context, the term "refuelling" is defined as the ´removal and/or replacement of either fresh or spent, single or multiple, bare or inadequately confined nuclear fuel cluster(s) or fuel element(s) contained in the core of a nuclear reactor`. This definition does not include replacement of well-contained fuel cassette(s) in a manner that prohibits clandestine diversion of nuclear fuel material."

2 Reactor description

The Fixed Bed Nuclear Reactor (FBNR) is a small reactor (40 MWe) without the need of on-site refueling. It utilizes the PWR technology but uses the HTGR type fuel elements. It has the characteristics of being simple in design, modular, inherent safety, passive cooling, proliferation resistant, and reduced environmental impact.

The FBNR is modular in design, and each module is assumed to be fuelled in the factory. The fuelled modules in sealed form are then transported to and from the site. The FBNR has a long fuel cycle time and, therefore, there is no need for on-site refuelling. The reactor makes an extensive use of PWR technology.

It is an integrated primary system design. The basic modules, as shown in the schematic figure, have in its upper part the reactor core and a steam generator and in its lower part the fuel chamber. The core consists of two concentric perforated zircaloy tubes of 20 cm and 160 cm in diameters, inside which, during the reactor operation, the spherical fuel elements are held together by the coolant flow in a fixed bed configuration, forming a suspended core. The coolant flows vertically up into the inner perforated tube and then, passing horizontally through the fuel elements and the outer perforated tube, enters the outer shell where it flows up vertically to the steam generator. The reserve fuel chamber is a 40-cm diameter tube made of high neutron absorbing alloy, which is directly connected underneath the core tube. The fuel chamber consists of a helical 25 cm diameter tube flanged to the reserve fuel chamber that is sealed by the international authorities. A grid is provided at the lower part of the tube to hold the fuel elements within it. A steam generator of the shell-and-tube type is integrated in the upper part of the module. The control rods slide inside the core The reactor is provided with a pressurizer system to keep the coolant at a constant pressure. The pump circulates the coolant inside the reactor moving it up through the fuel chamber, the core, and the steam generator. Thereafter, the coolant flows back down to the pump through the concentric annular passage. At a certain pump velocity, the water coolant carries up the 15 mm diameter spherical fuel elements from the fuel chamber into the core. A fixed suspended core is formed in the reactor. In a shut down condition, the suspended core breaks down and the fuel elements leave the core and fall back into the fuel chamber. The fuel elements are made of TRISO type micro spheres used in HTGR.

The control system is based on the inherent safety philosophy that when all the signals from all the detectors are within the design ranges, the pump can operate, thus the normal situation of pump is “off” position.. Therefore, any initiating event will cut-off power to the pump, causing the fuel elements to leave the core and fall back into the fuel chamber, where they remain in a highly sub critical and passively cooled condition. The fuel chamber is cooled by natural convection transferring heat to the water in the tank housing the fuel chamber.

The pump circulates the water coolant in the loop and at the mass flow rate of about 141 kg/sec, corresponding to the terminal velocity of 1.64 m/sec in the reserve fuel chamber, carries the fuel elements into the core and forms a fixed bed. At the operating mass flow rate of 668 kg/sec, the fuel elements are firmly held together by a pressure of 10 bar forming a stable fixed bed. The coolant flows radially in the core and after absorbing heat from the fuel elements enters the integrated heat exchanger of tube and shell type. Thereafter, it circulates back into the pump and the fuel chamber. The long-term reactivity is supplied by fresh fuel addition and a fine control rod that moves in the center of the core controls the short-term reactivity. A piston type core limiter adjusts the core height and controls the amount of fuel elements that are permitted to enter the core from the reserve chamber. The control system is conceived to have the pump in the “not operating” condition and only operates when all the signals coming from the control detectors simultaneously indicate safe operation. Under any possible inadequate functioning of the reactor, the power does not reach the pump and the coolant flow stops causing the fuel elements to fall out of the core by the force of gravity and become stored in the passively cooled fuel chamber. The water flowing from an accumulator that is controlled by a multi redundancy valve system cools the fuel chamber as a measure of emergency core cooling system. The other components of the reactor are essentially the same as in a conventional pressurized water reactor.

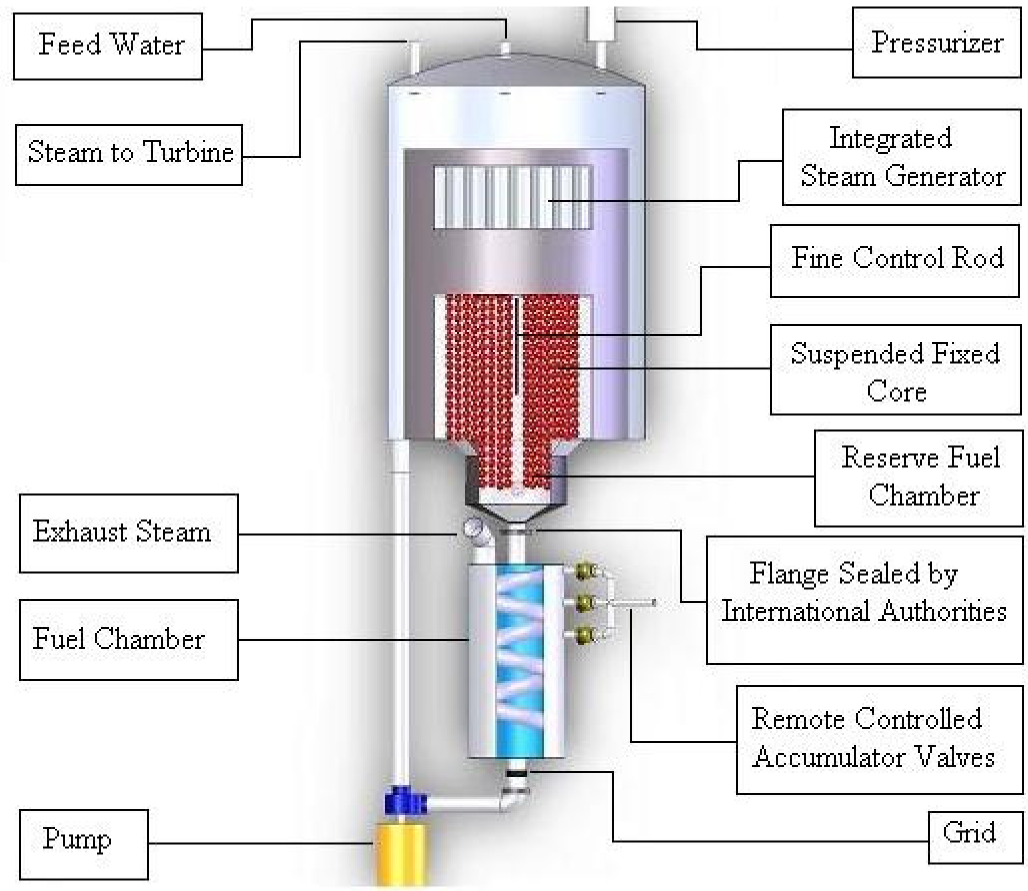

Figure 2-1: Schematic Design of FBNR

Table 2-1: Technical data for the Fixed Bed Nuclear Reactor (FBNR)

Parameter	**Value**
Power:	
Net power generation (MWe)	40
Power generation (MWt)	134
Core power density (KWt/lit)	33.7
Pump power (MWe)	3.4
Hydraulics:	
Coolant volume (m^3)	12
Coolant mass flow (kg/sec)	668
Coolant pressure (bar)	160
Pressure loss in the loop (bar)	100
Pressure loss in the bed (bar)	9.5
Terminal velocity (m/sec)	1.64
Thermal:	
Coolant inlet temperature (ºC)	290
Coolant outlet temperature (°C)	326
Coolant inlet enthalpy (kJ/kg)	1284
Coolant inlet density (kg/m^3)	747
Enthalpy rise in the core (kJ/kg)	1490
Film boiling convective heat transfer coefficient at 300 ºC (W/m^2ºC)	454
Fuel element average thermal conductivity (W/m.ºC)	30.58
Fuel element average specific heat (J/kg.ºC)	802.5
Fuel element average density (gr/cm^3)	4.041
Maximum fuel temperature after a LOCA (ºC)	< 357
Maximum fuel temperature after a LOCA (ºC)	< 357
Coolant temperature rise after a LOFA after 10 days (ºC)	< 1
Water needed to cool during 10 days after LOCA (m^3)	0.45
Module dimensions:	
Core height (cm)	200
Core inner diameter (cm)	20
Core outer diameter (cm)	160
Core volume (m^3)	3.96
Fuel in the core (Ton)	9.6
UO2 in the core (Ton)	4.8
Fuel element	
Fuel element diameter (cm)	1.5
SiC clad thickness (cm)	0.1
Number of microspheres in a fuel element.	165
Number of fuel elements in the core.	1.34×10^6
UO2 in each fuel element (% vol)	19.3
Dense graphite in each fuel element (% vol)	27.8
Porous graphite in each fuel element (% vol)	7.4
SiC in each fuel element (% vol)	45.5
UO2 density (gr/cm^3)	10.5
PYC porous density (gr/cm^3)	1.0
PYC dense density (gr/cm^3)	1.8
SiC density (gr/cm^3)	3.17

3 Fuel element description

Coated particle fuel has been used for more than 30 years in nuclear reactors. These reactors have benefited from this fuel's higher burnup and temperature capabilities and its multiple barriers to fission product release. The use of a particle fuel form in LWRs has the potential to significantly increase burnup , safety margin, and proliferation resistance. In addition, it will reduce the fission product release relative to the present clad UO2 fuels. Using a coated particle fuel form tailored to a water-reactor environment can eliminate the constraints of the present pressurized water reactor (PWR) fuel system. Particle fuel reduces fuel temperatures, lowers stored energy, and has better fission product retention.

One of the significant features of the coated particle fuel form is the vast increase in surface area per fuel volume over the commonly used pellet and clad fuel.

Coated particle nuclear fuel has been irradiated to more than ten times higher than the present LWR range. This allows much greater energy extraction from the same amount of fuel, which results in less fuel throughput per energy produced. The reduction in spent fuel minimizes the burden on both temporary and permanent storage of spent fuel. This increase in burnup can also be used to provide longer fuel cycles, which is a significant benefit in refuelling reactors in remote locations or countries with modest infrastructure.

3.1 SCALE computational codes

SCALE (**S**tandardized **C**omputer **A**nalyses for **L**icensing **E**valuation) is a modular code system that was originally developed by Oak Ridge National Laboratory (ORNL).. The SCALE system utilizes well-established computer codes and methods within standard analysis sequences that:
(1) provide an input format designed for the occasional user and/or novice,
(2) automate the data processing and coupling between modules, and
(3) provide accurate and reliable results.

System development has been directed at problem dependent cross-section processing and analysis of criticality safety, shielding, depletion/decay, and heat transfer problems.

Criticality Safety Analysis Sequence (CSAS) was developed to provide a search capability for three-dimensional (3-D) configurations in the SCALE system. At the center of the **C**riticality **S**afety **A**nalysis **S**equences (CSAS) is the library of subroutines referred to as the **M**aterial **I**nformation **P**rocessor **Lib**rary or MIPLIB. The CSAS control module is the primary criticality safety control module for the calculation of the neutron multiplication factor of a system. Multiple sequences within the CSAS module provide capabilities for a number of analyses, such as modelling a one dimensional (1-D) or a 3-D system, searching on geometry spacing or material concentrations, and processing cross sections.

The 238-group ENDF/B-V library (238GROUPNDF5) is the most complete library in SCALE 5. This library contains data for all ENDF/B-V nuclides and has 148 fast and 90 thermal groups. The 238- and 44-group libraries are the preferred criticality safety analysis libraries in SCALE. The 44-group library is recommended for LWR systems, and the 238-group library is recommended for all other types of systems.

CSAS: control module for enhanced criticality safety analysis sequences has the following inherent limitations:

1. Double heterogeneity such as HTGR or Pebble Bed fuel, where uranium encased in small graphite spheres are used to make larger spheres or rods which are then placed in a regular lattice.
2. Two-dimensional (2-D) effects such as fuel rods in assemblies where some positions are filled with control rod guide tubes, burnable poison rods and/or fuel rods of different enrichments. The cross sections are processed as if the rods are in an infinite lattice of identical rods.

CSAS performs a search for 3-D problems. CSAS25 calculates the *keff* for 3-D problems. KENO V.a is a functional module in the SCALE system. It calculates the *keff (i.e.,* neutron multiplication) of a 3-D problem using the Monte Carlo methodology. . A 238-energy-group neutron cross-section library based on ENDF/B-V2 is the latest cross-section library in SCALE. All the nuclides that are available in ENDF/B-V are in the library. A 44-group library has been collapsed from this 238-group library and validated against numerous critical measurements.

4 Cell calculations

4.1 Cell description:

Before running a k_{eff}, calculation of the whole reactor, one single fuel cell was analyzed to simulate k_{∞}.

One Fuel-element of the FBNR consists of spherical elements surrounded by water. The 15 mm diameter spherical fuel elements are made of compacted coated particles in a graphite matrix. The coated particles are similar to TRISO fuel with outer diameters about 2mm. They consist of 1.58 mm diameter uranium dioxide spheres coated with 3 layers. The inner layer is of 0.09 mm thick porous pyrolitic carbide (PYC) with density of 1 g/cm3 called buffer layer, providing space for gaseous fission products. The second layer is of 0.02 mm thick dense PYC (density of 1.8 g/cm3) and the outer layer is 0.1 mm thick corrosion resistant silicon carbide (SiC, density of 3.17 g/cm3). The fuel element is cladded by 1mm thick SiC.

Table 4-1: Fuel particle (2 mm diameter)

Material	density (g/cm³)	d. inside (cm)	d.outside (cm)	volume (cm³)	mass (gr)
UO2	**10.5**	**0**	**0.158**	**0.002065237**	**0.021684988**
PYC (porous)	**1**	**0.158**	**0.176**	**0.000789306**	**0.000789306**
PYC (dense)	**1.8**	**0.176**	**0.18**	**0.000199085**	**0.000358353**
SiC	**3.17**	**0.18**	**0.2**	**0.001135162**	**0.003598464**
Average for microsphere	**6.3099629**		**0.2**	**0.00418879**	**0.026431111**

Due to the computer code limitation, one fuel-element is divided into two regions: The inner region, consisting coated particles inside a graphite matrix, is simulated as a homogenized mixture of these components. The volume fractions of each material are listed in Table 4-2. The outer region consists the 1mm SiC cladding.

Table 4-2: Mixture of Region 1

Material	Mass (gr)	Volume (cm3)	Density (g/cm3)	Mass fraction	Volume fraction	Thermal conductivity (W/m.°C)	Specific heat (kJ/kg.°C)
UO2	3.578	0.341	10.5	0.501	0.193	5.2	
PYC porous (amorfo) 600K	0.130	0.130	1	0.0182	0.0737	2.19	1406
PYC dense (amorfo) 600K	0.887	0.493	1.8	0.124	0.279	2.19	1406
SiC	2.549	0.804	3.17	0.357	0,455	77.5	1300
fuel element	7.145	1.768	4.041	1	1	30.566	1400

To simulate the reactor as a cylinder, filled by fuel spheres and water, each fuel element is surrounded by a dodecahedronal water region. Arranging several Dodecahedrons on each other allows modelling the reactor core. The radius of one dodecahedron is chosen as such, to get a porosity of 40% (volume-fraction of water to fuel). The composition of these 3 regions (Fuel, SiC, and Water) creates one fuel unit.

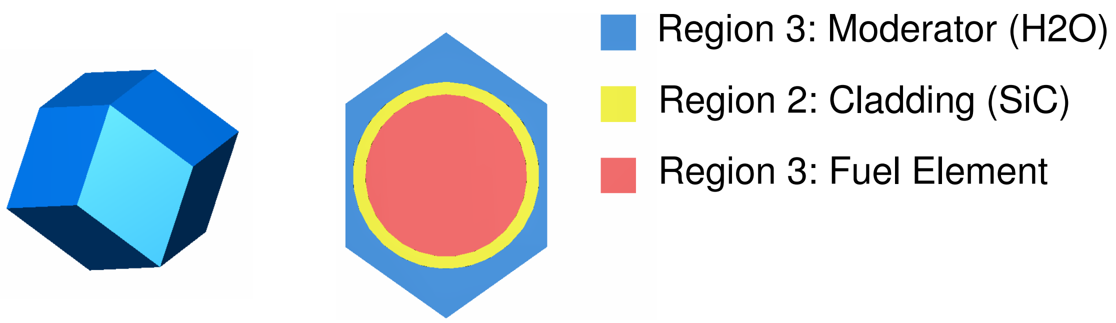

Figure 4-1: Unit cell

The reactivity of a single cell for a porosity of 40% is shown in Table 4-3. The calculations were made for two boundary conditions: Mirror and Vacuum. A mirrored boundary condition will give the best possibility to simulate k_∞ by using a single cell. The following reactivities are obtained for a 5% enriched fuel cell.

Table 4-3: Single cell (5% enrichment) with different Boundary conditions

Boundary condition	k_∞
Mirror	1.40319
Vacuum	0.00353

4.2 Reactivity as a function of enrichment

The reactivity as a function of enrichment for a single sphere was calculated. In case of the single cell, the values approach those of K_∞. The results are shown in Figure 4-2.

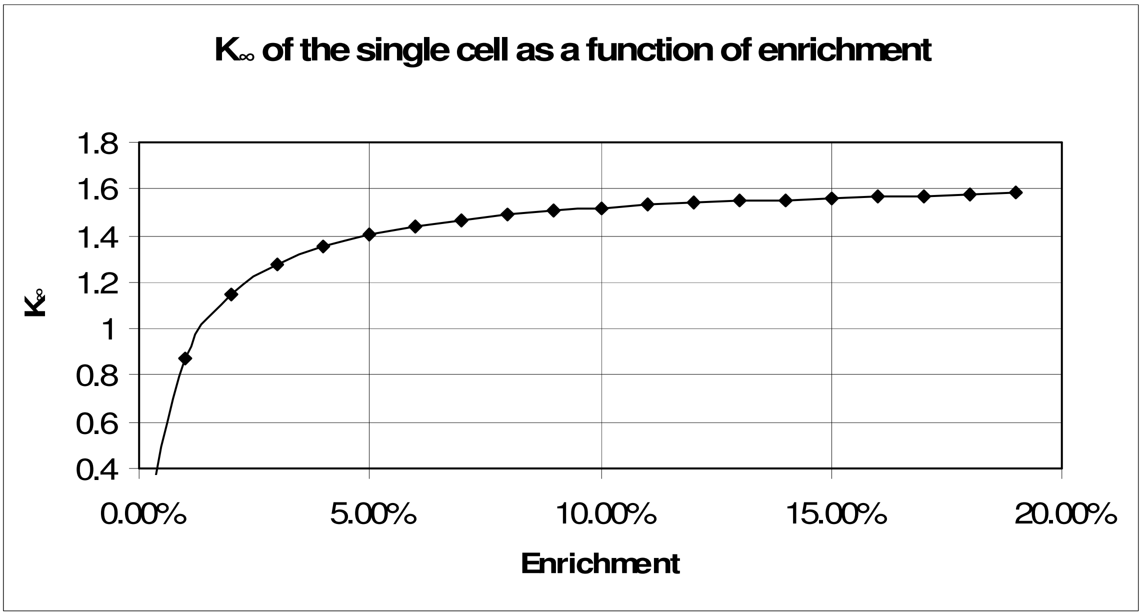

Figure 4-2: K_∞ for a single cell

Up to an enrichment of 5%, k_∞ increases considerably. After that point, k_∞ rises moderately up to the maximum of 1,79 for an enrichment of 99% (Table 4-4). The reactivity for maximum hypothetical enrichment is investigated. k_∞ will be around 1.79 for a water moderated reactor and 1.82 for a graphite moderated reactor.

Table 4-4: k_∞ for 99% enrichment

Moderator	$\mathbf{K_\infty}$
H20	1.79
Graphite	1.82

5 Effect of heterogenity

Because of the computer capacities, it was not possible to compare the reactor core assuming a homogeneous with the case assuming a heterogeneous model (arrays of fuel cells). Also the code SCALE does not allow treatment of double heterogeneity.
For present studies, the objective being the study of the behavior of the reactor, the homogeneous calculations were considered sufficiently adequate. The homogeneous mixture at the fuel region consists of UO_2, H_2O, Graphite and SiC. The volume fractions of each material inside this region are listed in Table 5-1.

Table 5-1: Mixture of homogeneous Reactor

	Mass Fraction	**Density [g/cm³]**
Core material		
UO_2	0.446	10.50
Graphite (porous)	0.016	1.00
Graphite (dense)	0.111	1.80
SiC	0.3177	3.17
H_2O	0.110	0.747
Moderator material		
H_2O	0.110	0.747
Structural material		
Stainless Steel SS-304	1	7.49
Zirkaloy	1	6.56
Absorber material (lower Part)		
Cadmium	8.642	8.642

Figure 5-1 to Figure 5-4 show the homogenous model of the reactor, as it was used for the k_{eff} and burnup calculations.

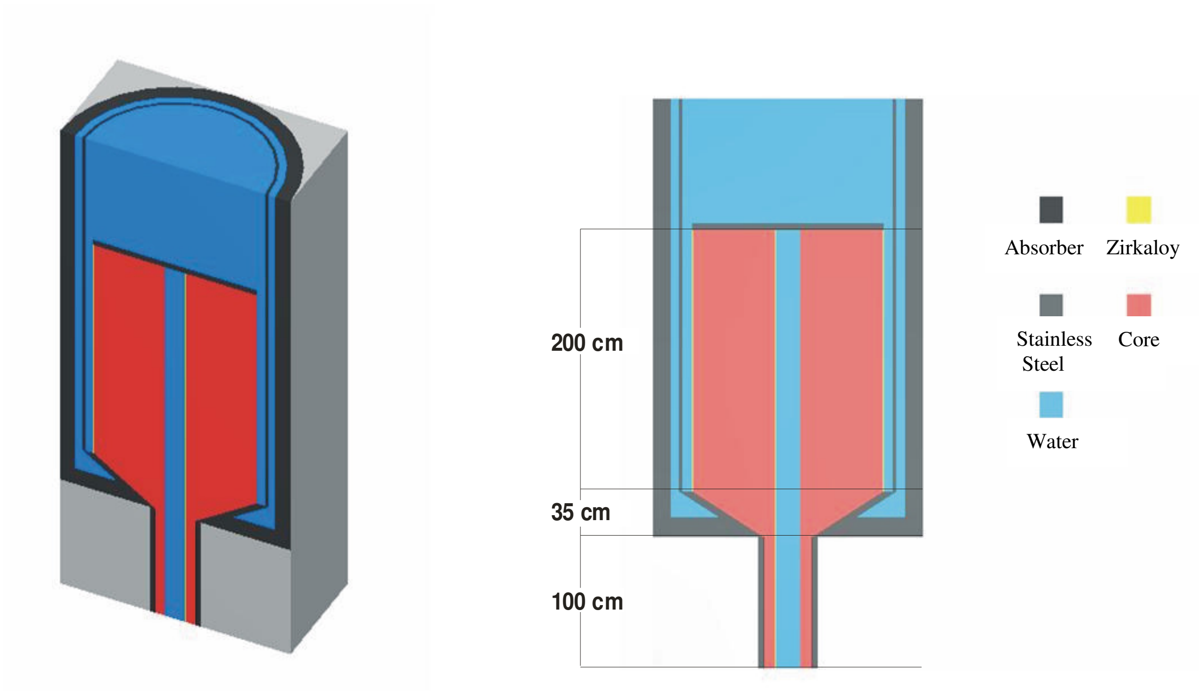

Figure 5-1: Keno VI model of homogenous reactor

Transversal sections of the upper part, middle part and lower part of the reactor are shown below:

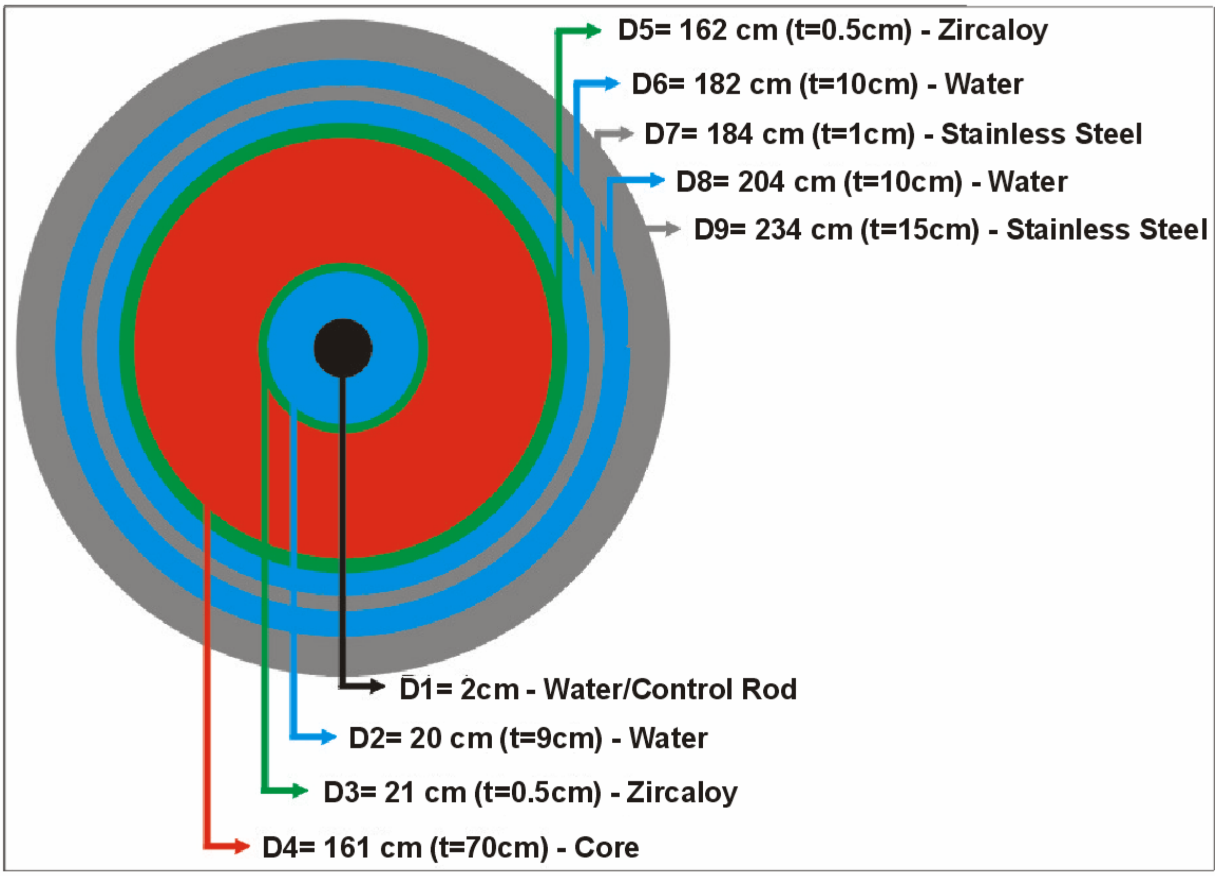

Figure 5-2: Upper Part

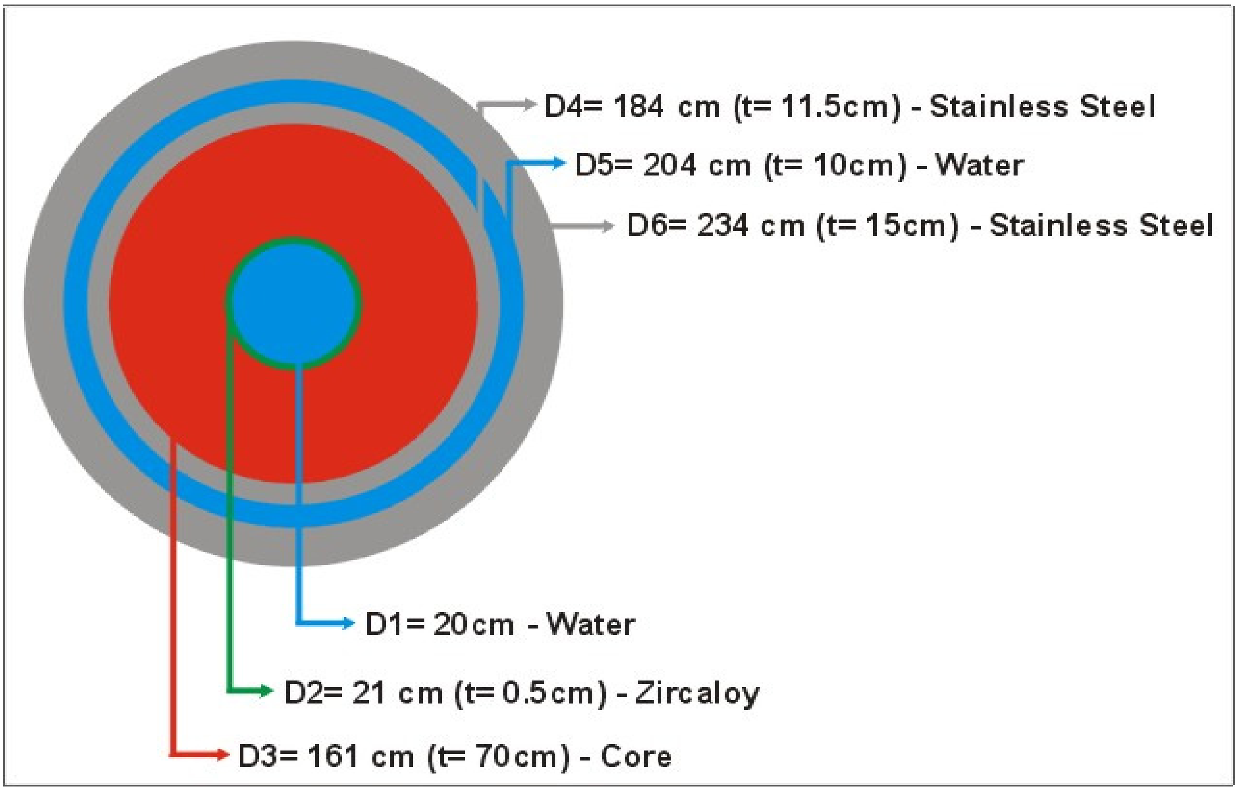

Figure 5-3: Middle Part

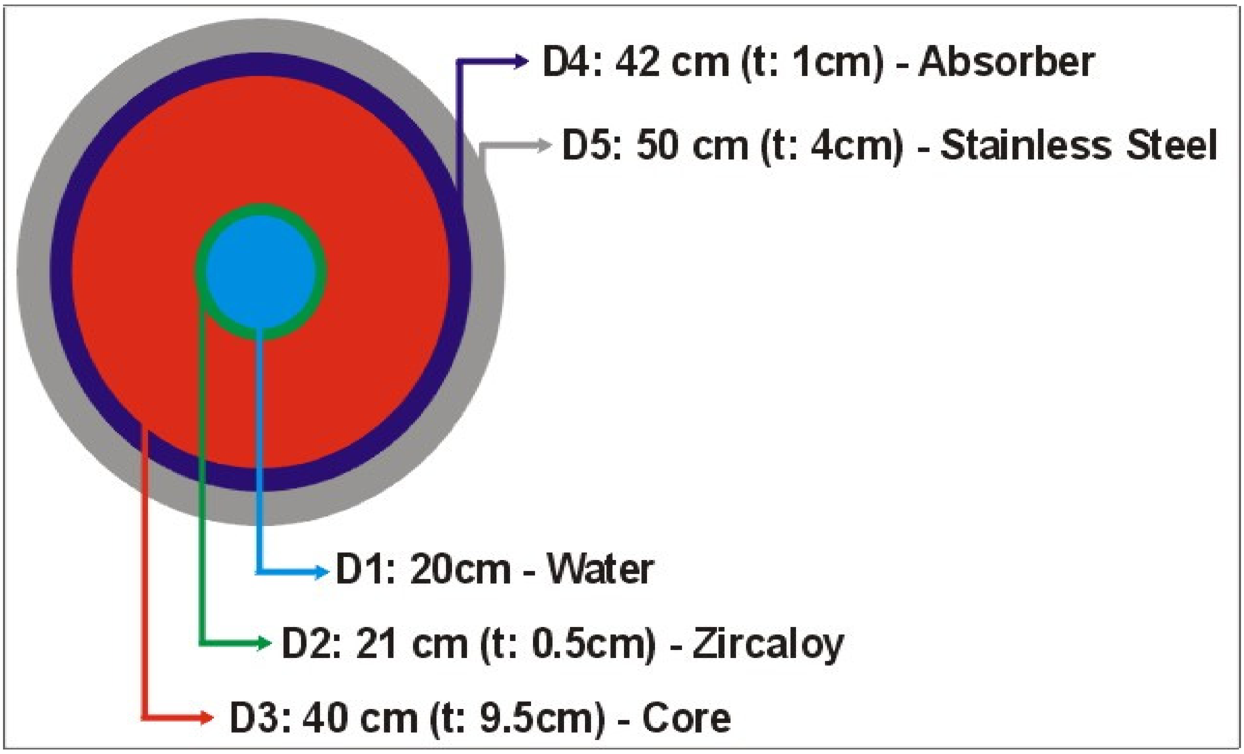

Figure 5-4: Lower Part

5.1 Reactivity of the reactor as a function of core height and enrichment

The global neutron multiplication factors of the reactor as a function of core height for enrichments of 2.2%, 5%, 9% and 19% are shown in Figure 5-5. Up to a value of about 120cm, the core height has a significant influence in reactivity. But as this influence is very low for core heights in the range between 120cm and 250cm, there is a need to use poison to reduce k_{eff} at the beginning of the burnup cycle. This possibility permits the use of higher enriched Uranium for an increase of core lifetime.

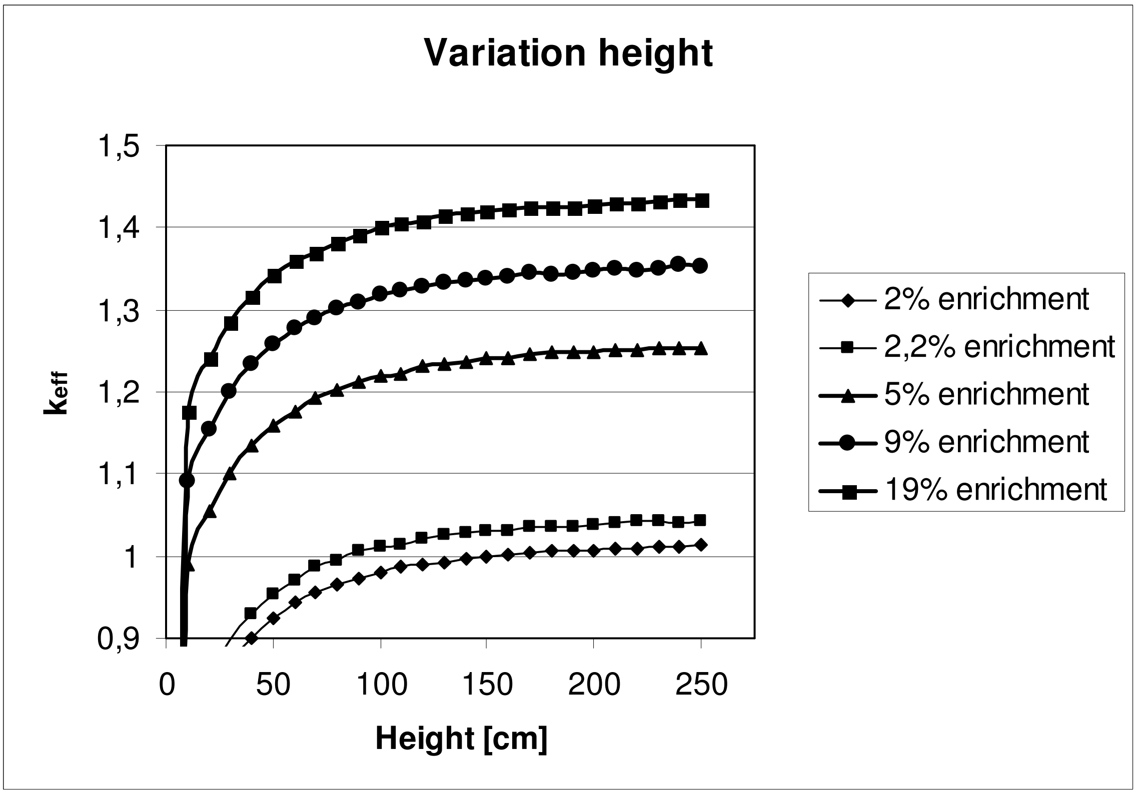

Figure 5-5:Variation of height for different enrichment values

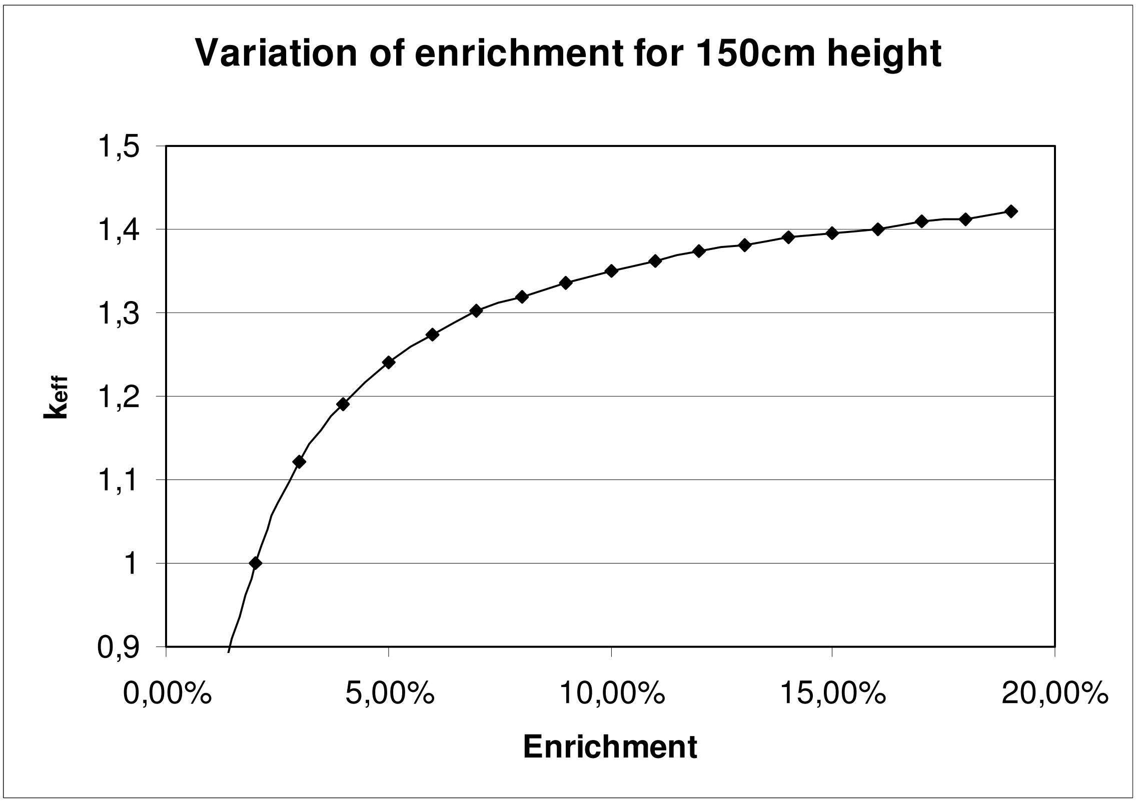

Figure 5-6: Variation of enrichment

The reactivity as a function of enrichment for a homogenous reactor was calculated for a height of 150cm (Figure 5-6). The maximal value was a k_{eff} of 1.421 for an enrichment of 19%.

5.2 Control rod reactivity worth

The FBNR reactor contains of 5 Control Rods. One Rod is centred inside the middle water tube and four rods are arranged inside the Core region.

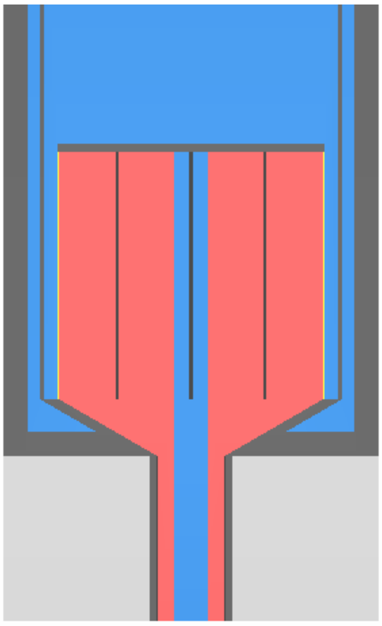

Figure 5-7: Rodded Reactor

Each Control Rods contains of 5% Cadmium, 15% Indium and 80% Silver and has a radius of 1cm. The total worth of the control rods are shown in Table 5-2. The maximal difference of k_{eff} was 1340 pcm. As the rods have a little influence in the reactivity worth, they can only be used for short-term operations. The standard deviation of keff is +/-0.0011.

Table 5-2: k_{eff} for different CR positions

Control Rods	**k_{eff}**	**Δk_{eff}**
Without control rod	1.2413	-
Central CR without in-core CRs	1.2391	-0.0023
Central CR + one in-core CRs	1.2347	-0.0066
Central CR + two in-core CRs	1.2313	-0.0100
Central CR + three in-core CRs	1.2307	-0.0107
Central CR + four in-core CRs	1.2279	-0.0134
One in-core CRs	1.2372	0.0041
Two in-core CRs	1.2351	0.0062
Three in-core CRs	1.2321	0.0092
Four in-core CRs	1.2302	0.0111

6 Burnup

The Burnup calculations were made by using the SCALE5 module STARBUCS. This module allows automatic criticality analyses of spent fuel systems employing burnup credit. As a first step STARBUCS starts the burnup sequence for a depletion analysis calculation, performed using the ORIGEN-ARP module of SCALE5. The spent fuel compositions are then used to generate resonance self-shielded cross sections, which are applied in a three-dimensional criticality safety calculation using the KENO code.

The variables for the burnup-calculations were the burnup-time, the average specific power of the assembly for each cycle (POWER) and the enrichment of the core. Power density [MW/MTU] is thermal power generation per mass of uranium inside the core [t]. The mass of uranium in the middle part of the reactor is 324kg. The mass of the upper part as a function of the core-height is calculated to be 24.32kg/cm (for a density of uranium of 10,5g/cm³).

6.1 Burnup calculations

Figure 6-1 shows keff as a function of burnup for a core height of 200cm and 250cm in order to evaluate the core lifetime. Since STARBUCS will not calculate a burnup for enrichment higher than 5%, all calculations were made for a 5% enriched core.

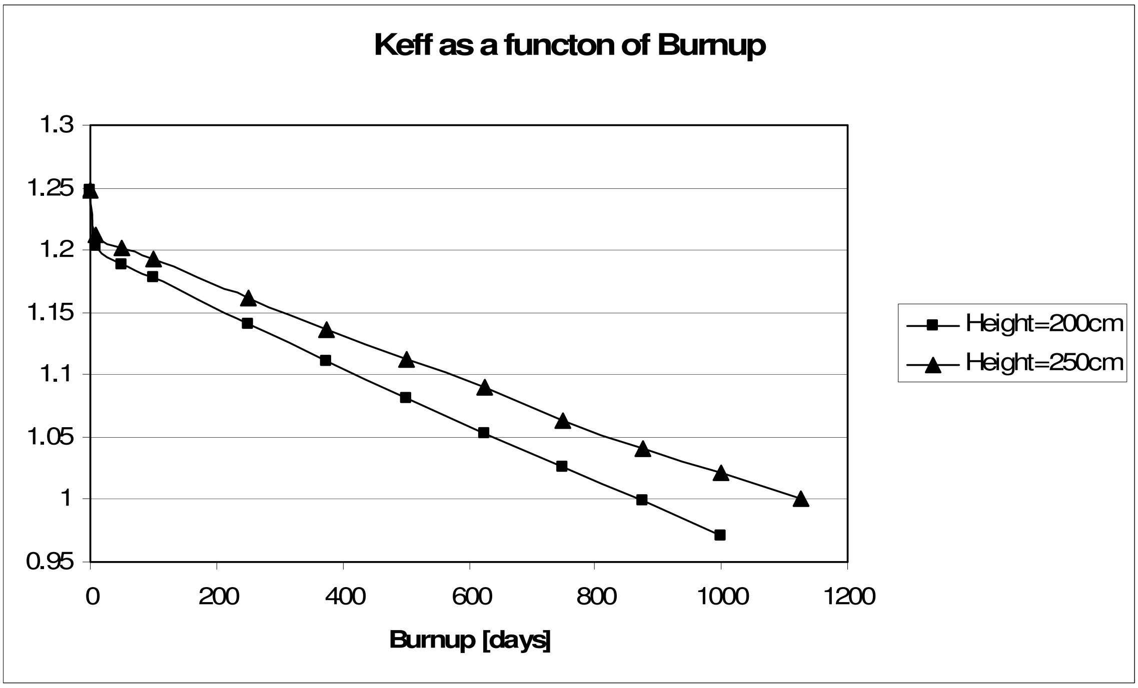

Figure 6-1: Keff as a function of Burnup

The burnup calculation for a core height of 200cm give an estimated core burnup cycle of 875 days, while a core of a height of 250cm achieves a lifetime of 1125 days (for an enrichment of 5%).

The estimated lifetime of fuel with a higher enrichment than 5% can be extrapolated by the results of 5% enrichment, as the annual loss of enrichment by burnup is an almost linear function (see Figure 6-2). In chapter 5.1, the minimal enrichment of 2% was determined to get a keff of 1 (Figure 5-5). As STARBUCS also includes the fission products for its reactivity calculations, the new minimal enrichment has to be set as 2.5%. In conclusion of these results, the annual loss of enrichment will be -0.8% per year for a core of 250cm height and -1.05% per year for a core of 200cm height (caused by a higher burnup rate for smaller core dimensions).

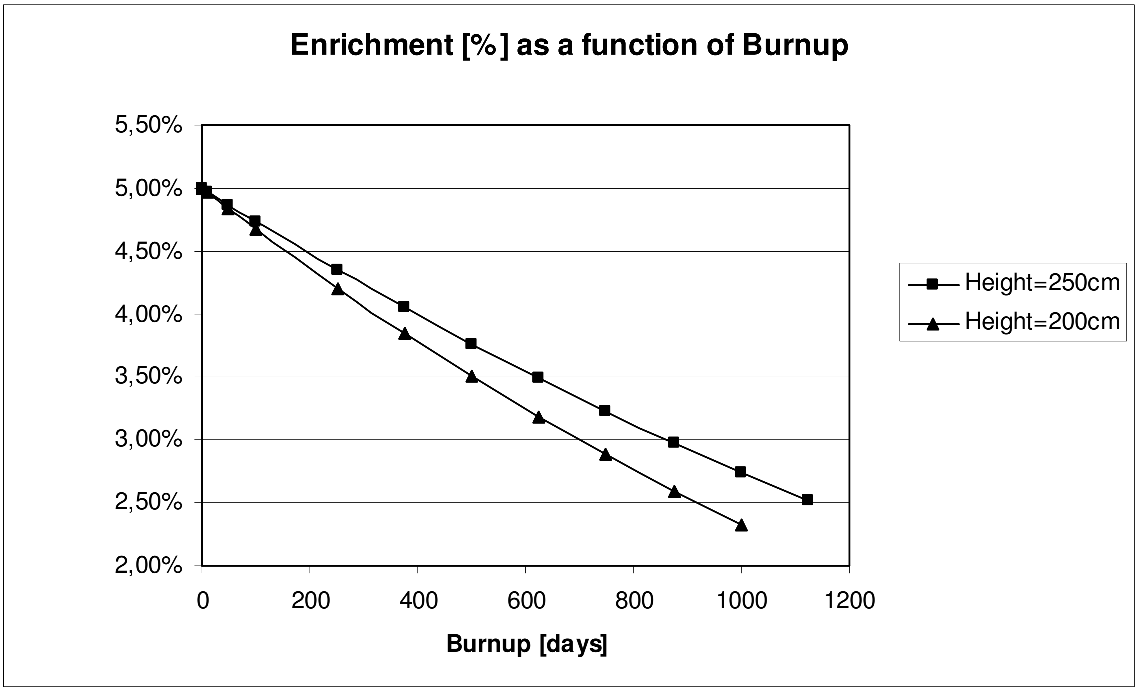

Figure 6-2: Enrichment as a function of Burnup

In order to have a core life of more then 7 years, a fuel enrichment of about 9% will be needed for a core height of 250cm (or 14% for a core height of 200cm). The maximal core life of 17 years can be obtained for fuel of 19% enrichment.

STARBUCS allows printing all fission products being generated during burnup. Figure 6-3 shows the total mass of Plutonium inside the reactor core as a function of burnup.

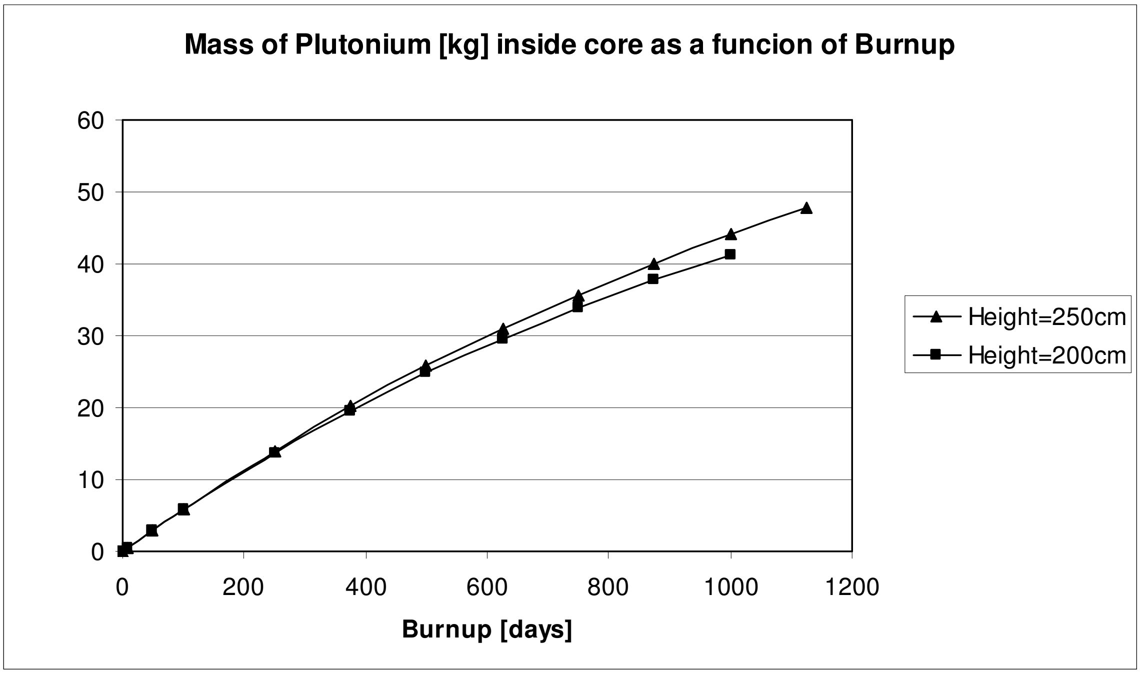

Figure 6-3: Pu mass as a function of Burnup

7 Conclusions

The preliminary neutronics calculations show that the expected behavior of the FBNR is similar to a conventional PWR. The core lifetime can be as long as 17 years should the customer being ready to pay for the fuel of 19% enrichment. The 9% enrichment provides a lifetime of 7 years. In practice, this is not necessary as the reactor design involves the existence of small fuel chamber that can easily be changed. A 5% enriched reactor will require a change of fuel chamber only once every 3 years. The refueling involves the connecting and disconnecting of a 5 m^3 fuel chamber to the reactor by a flange that is sealed by the safeguard authorities.

8 References

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9 Acknowledgement

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