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THE INVESTIGATION OF BURNUP CHARACTERISTICS USING THE SERPENT MONTE CARLO CODE FOR A SODIUM COOLED FAST REACTOR

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In this research, we investigated the burnup characteristics and the conversion of fertile ^{232}Th into fissile ^{233}U in the core of a Sodium-Cooled Fast Reactor (SFR). The SFR fuel assemblies were designed for burning ^{232}Th fuel (fuel pin 1) and ^{233}U fuel (fuel pin 2) and include mixed minor actinide compositions. Monte Carlo simulations were performed using Serpent Code1.1.19 to compare with CRAM (Chebyshev Rational Approximation Method) and TTA (Transmutation Trajectory Analysis) method in the burnup calculation mode. The total heating power generated in the system was assumed to be 2000 MWth. During the reactor operation period of 600 days, the effective multiplication factor (keff) was between 0.964 and 0.954 and peaking factor is 1.88867.

KEYWORDS : Generation IV Reactor, SFR, Serpent Code, Thorium, Monte Carlo Calculation

1. INTRODUCTION

The main issues concerning nuclear energy development are economics, nuclear safety, nuclear waste management and limited uranium resources. Owing to a new generation of nuclear energy systems there are many plans to solve the problems. Many different technologies have been examined for nuclear waste transmutation and energy production, including an Accelerator Driven System (ADS) and a variety of reactors [1 – 4]. The new nuclear reactors (Generation IV) are expected to start being deployed by 2030 [5,6]. The Generation IV reactors will provide economic competitiveness, enhanced safety, minimal radioactive waste generation and proliferation resistance. In addition, Generation IV nuclear energy systems will provide sustainable energy generation that meets the clean air objectives and promotes long-term availability of the systems and effective fuel utilization for worldwide energy production [7].

The existing once though nuclear fuel cycle is a major concern due to the limited uranium resources worldwide. The thorium fuel cycle has several challenges which need to be resolved before thorium could be introduced in commercial nuclear power reactors. Depending on the type of reactor, there is a big different between thorium based nuclear fuels and fuel elements. Thorium-232 is a "fertile" material that can be transmuted by neutron irradiation into ^{233}U, a fissionable material. The Th–U cycle represents an alternative: this cycle can be realized with not only a (epi)thermal spectrum, but also with a fast spectrum. In order to preserve and extend the lifetime of nuclear resources, fast reactors are an option [8].

There is always a need for a topping fissile material in order to achieve extended burnup in the once though thorium fuel cycle in thermal reactors. The best topping material is ^{233}U for this purpose. Fuel utilization is a key topic in new reactor development. Most of today's nuclear power plants are LWRs, which have a low conversion ratio, thus much more fissile material is burned than produced [8,9].

2. SFR CONCEPT

The Sodium-Cooled Fast Reactor (SFR) which is one of six reactor concepts favored by the Gen-IV program has been studied and developed in many countries such as the U.S., Russia, France, U. K. and Japan [10,11]. The SFR system has sodium cooled core, a fast neutron spectrum and a closed fuel cycle for efficient managements of actinides (Np, Cm, Am…). The SFR concept has the most comprehensive technological system as a result of the past experience gained from the worldwide operation of several experiments, prototypes and commercial size reactors. This experience with the design and operation of such systems has shown that they can be operated reliably. However, the important technology gaps for the SFR are in an area of development of fuel fabrication and in reactor safety. For SFR, one major safety issue is the high sensitivity to the sodium void effect that may induce positive reactivity [12].

SFR fuels generally contain a relatively small fraction of minor actinides and a small amount of fission products. The mixed oxide fuel type (U, Pu)O_2 is commonly used in the SFR. The SFR's fast spectrum also makes it possible to use available fertile and fissile materials like ^{232}Th and ^{233}U. ^{232}Th is less fissile than ^{238}U due to a higher fission threshold energy in the fast neutron spectrum. In addition, a thorium based fuel cycle does not produce minor actinides but is associated with other isotopes like ^{231}Pa, ^{229}Th and ^{230}U which would have long term radiological impacts [9,12].

3. SERPENT MONTE CARLO CODE

The Monte Carlo method is based on the generation of a sequence of random numbers, which are used together with statistical laws to simulate the desired process. Monte Carlo particle transport methods are conventionally based on a ray-tracing algorithm using complicated geometry. The development of Serpent which is the Monte Carlo lattice physics code started in 2004 for use in Monte Carlo reactor physics calculations. Serpent is a new continuous -energy Monte Carlo code, developed at VTT under the working title “Probabilistic Scattering Game”. The Serpent code is mainly intended for lattice physics calculations [13]. User defined tallies can be set up for calculating integral flux and reaction rates in cells, materials and universes. Serpent uses ENDF format interaction data, read from ACE format cross section libraries. Burnup calculation requires radioactive decay data and neutron-induced and spontaneous fission product yields. These files are read in the raw ENDF format [14]. Three types of cross sections are available in the data files. First, continuous-energy neutron cross sections contain all necessary reaction cross sections, together with energy and angular distributions, fission neutron yields and delayed neutron parameters for the actual transport simulation. Second, dosimetry cross sections exist for a large variety of materials and can be used with detectors but not in physical materials included in the transport calculation. Third, thermal scattering cross sections are used to replace the low-energy free-gas elastic scattering reactions for some important bound moderator nuclides. All calculation outputs are written in Matlab files (m-format) to simplify the simultaneous post-processing of several calculation cases. The code also has a geometry plotter feature and a reaction rate mesh plotter. The Serpent code has two methods (TTA and CRAM) for solving the Bateman equations describing the changes in the isotopic compositions caused by neutron-induced reactions and radioactive decay [15].

4. CRAM AND TTA METHOD

The radioactive transformation of many radionuclides often yields a product that is also radioactive. The radioactive product in turn undergoes transformation to produce yet another radioactive product and so on until stability is achieved. The number of atoms of each member of a radioactive series at any time t can be obtained by solving a system of differential equations which relates each product N_1, N_2, N_3,..., N_i with corresponding disintegration constants λ_1, λ_2, λ_3,..., λ_i. Each series begins with a parent nuclide N1, which has a rate of transformation [16].

$$\frac{dN_1}{dt} = -\lambda_1 N_1 \tag{1}$$

The Bateman equation is a set of first order differential equations describing the abundances and activities in a decay chain as a function of time, based on the decay rates and initial abundances. Assuming zero concentrations of all daughters at time zero $N_1(0)\neq 0$ and $N_i(0)=0$ when i > 1 the concentration of *n*th nuclide after time t was given by Bateman[17].

$$N_n(t) = \frac{N_1(0)}{\lambda_n} \sum_{i=1}^{n} (\lambda_i \alpha_i \exp[-\lambda_i t]) \tag{2}$$

where

$$\alpha_i = \prod_{\substack{j=1 \\ j\neq 1}}^{n} \frac{\lambda_i}{(\lambda_j - \lambda_i)} \tag{3}$$

The time evolution of the nuclide concentrations during the transmutation process can be obtained by solving the set of Bateman's equations. The solving method is based on the resolution of the transmutation chain, which is nonlinear, into a set of linear chains -the equations of which- can be solved analytically [18]. Transmutation trajectory analysis (TTA), also known as the linear chains method, is an alternative method for solving the decay and transmutation equations. In this method, one linear chain represents one transmutation trajectory between a destroyed nuclide and a produced one [19]. The decay and transmutation equations can be written in a matrix exponential function. The matrix exponential notation is

$$e^{At} = \sum_{m=0}^{\infty} \frac{1}{m!} (At)^m \tag{4}$$

The matrix exponential methods are based on different numerical approximations which in a general case cannot be evaluated exactly. Numerous methods have been developed for evaluating the matrix exponential. The Chebyshev rational approximation method (CRAM) is a new matrix exponential method, the matrix exponential power series with instant decay and secular equilibrium approximations for short-lived nuclides [20].

5. SFR CORE MODEL DESIGN

The calculations and system design were performed using the Serpent Monte Carlo code. A number of calculations were simulated with this code in order to obtain neutronic and burn-up characteristics of the SFR model. The SFR has been designed by a geometry plotter using these calculations. A parametric survey has been done to find out the adequate SFR core design while the accommodating the design requirements. During the first stage of the SFR design study, the SFR was designed as having a 4.6 m height and a 2.3 m radius. The active core height and radius were restricted as 2.6 m and 1.78 m respectively. The distributions of the fuel pin cell for a bundle and core specifications are given as shown two dimensional in Figure 1 and Table 1. Table 1 describes the construction materials, fuels and burnup parameters for SFR system. The SFR was designed for management of high-level wastes and, in particular, management of plutonium and other actinides. In this study, the Fuel Pin 1 was composed of ^{232}Th and minor actinides. There were also ^{233}U and minor actinide fuel isotopes in Fuel Pin 2. The centers of the fuel pin bundle were composed of sodium cooled

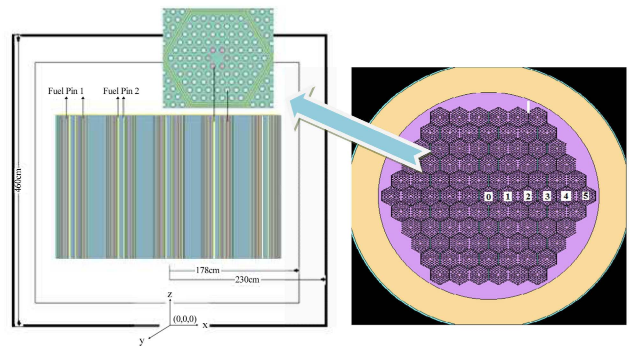


Fig. 1. The Vertical and Horizontal Geometry Specifications of SFR System

Table 1. The Specifications of SFR System and Burnup Parameters

Burnup mode	CRAM and TTA
Message Passing Interface (MPI) tasks	8
Running time(minute)	131
Total number of burnup steps	12
Burnup days	600
Depletion day steps	0.10-60-120-180-240-300-360-420-480-540-600
Number of simulated source neutrons	10^7
The number of source neutrons per cycle	20000
The number of active cycles run	500
The number of inactive cycles run	20
Fuel 1 pin and Fuel 2 pin inner/outer radius	0.75 cm / 0.95 cm
Fuel Pin Height	260 cm
Fuel 1 pin and Fuel 2 isotopes	^{232}Th / ^{233}U; $^{237-239}U$; $^{235-239}Np$; $^{236-244}Pu$; $^{241-243}Am$; $^{232-248}Cm$
Fuel density(g/cm^3)	10.457
Number of fuel 1 in asemblies	84
Number of fuel 2 in asemblies	6
Pin numbers in bundle	90
Number of asemblies in core	91
Total pin number in system	8190
Total thermal power (W)	2.0E+9
Structural Material	HT9
Reflector	B_4C

channels. The construction materials of the fuel clads and system were made from stainless steel (HT9). The core was surrounded by a region of reflector (B_4C). There were no control rods in SFR model.

6. NEUTRONIC AND BURNUP CALCULATIONS

The neutronic and burnup calculations were completed using the Message Passing Interface (MPI) on a double-processor 2.4 GHz Intel Xeon PC workstation. During calculations, 8 MPI tasks were shared on a 48 GB memory space. An irradiation period of 600 days was chosen in 2000 MWth SFR for burnup calculations. The evaluated cross section library used in the calculations was selected as ENDF/B-VII. The neutronic behaviors of the SFR have been investigated in the fuel pins. Neutronic characteristics are important when the SFR is operated as a fast neutron spectrum. The neutronic performance of fuel 1 and fuel 2 are shown in Figure 2. These calculations were carried out to examine the neutronic properties of the SFR and methods for the evaluation of reactor parameters including reaction rate distribution. The neutronic calculations were achieved for CRAM and TTA calculations according to neutron flux and burnup in fuel 1 and fuel 2. These results show almost the same characteristics for fuel 1 and fuel 2.

In Figure 1, the radial geometry specifications of the SFR core can be seen for lattice numbers. The hexagonal fuel assemblies were divided as radial in an array of six areas. Radial power distribution was plotted depending on each lattice as shown Figure 3. In the center of core, calculated peaking factor was calculated to be 1.88867. The results are consistent as the burn time interval increases. The core power distribution is reduced by the radial axes.

Fig. 4 shows effective multiplication factors for different burnup modes in SFR system. The results have been obtained subcritical because of the used breeding thorium fuel in the core. The effective multiplication factor (keff) changes from 0.964 to 0.954 during the reactor operation period.

The Monte Carlo analysis was performed in order to obtain burnup characteristics of the SFR. According to burnup, Figure 5(a,b) shows ^{233}U atomic density change in fuel 1 and fuel 2. At first, ^{233}U atomic density is not

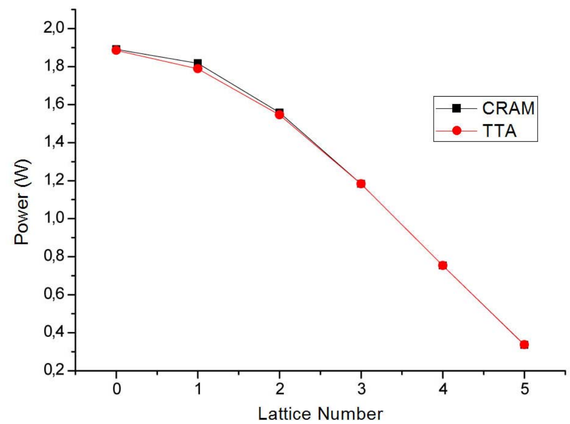


Fig. 3. Radial Power Distribution in Lattice

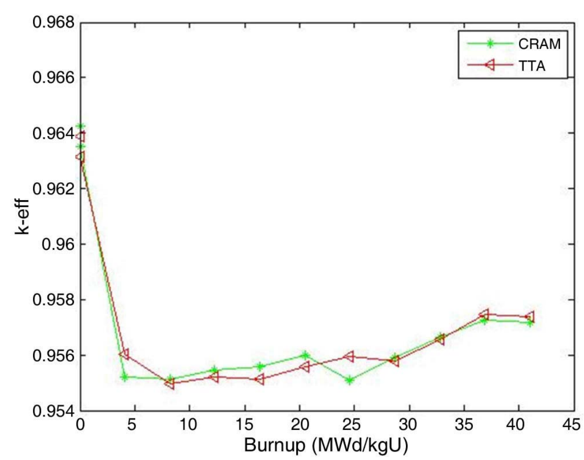


Fig. 4. Effective Multiplication Factor for SFR

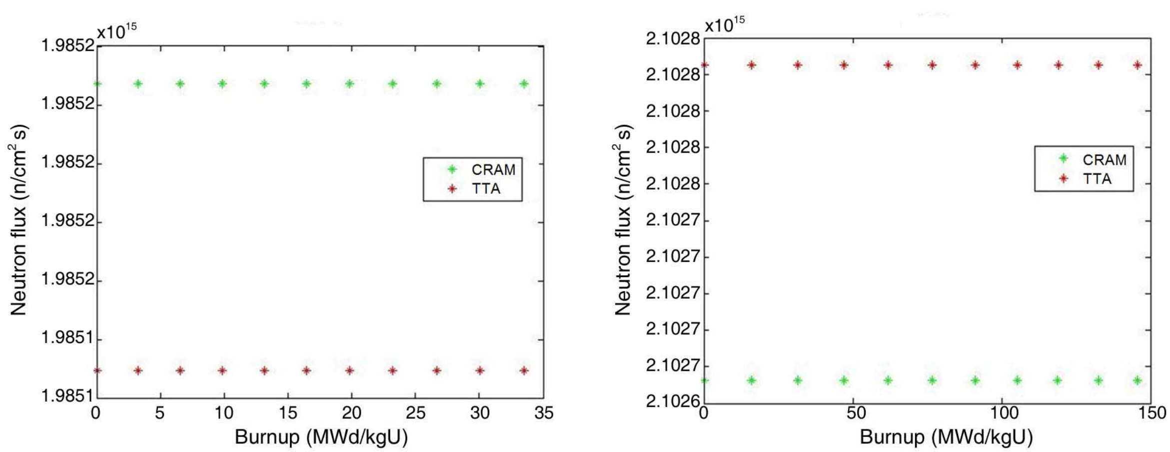


Fig. 2. Neutron Flux and Burnup Calculations for CRAM and TTA in (a) Fuel 1 and (b) Fuel 2

available in fuel 1. However, it has been increasing during the irradiation period due to $^{232}Th(n,g)^{233}U$ neutron capture reaction. On the other hand, ^{233}U which is a fissile nuclide was used as fuel and hence, atomic density has been decreasing over time. Figure 5c also shows ^{232}Th depletion in fuel 1. As regards minor actinide, ^{237}Np, ^{239}Pu, ^{241}Am and ^{244}Cm mass changes according to total burnup are shown in Figure 6. The depletion rate is mainly related to the fuel type and to the fast neutron spectrum. It also depends on the general core design and performances, as these relate to the fuel lattice, the fuel management and the power distribution.

7. CONCLUSIONS

We have performed an analysis of the neutronic and burn-up characteristics of the SFR model. We have investigated the potential for transmutation of $^{232}Th/^{233}U$ in sodium cooled fast reactor. In calculations, the core configurations were investigated for both CRAM and TTA. As regards radial power distribution, there is no significant difference between CRAM and TTA for SFR

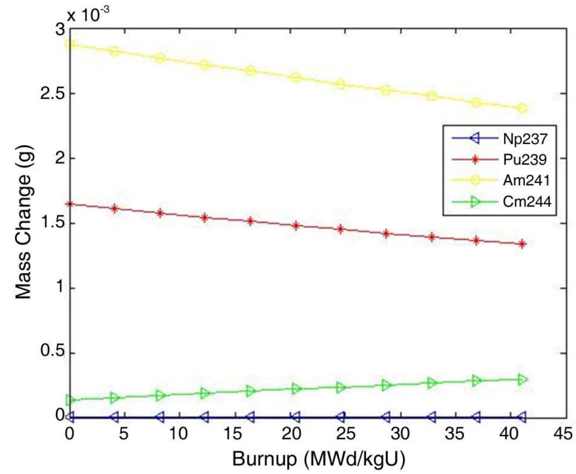
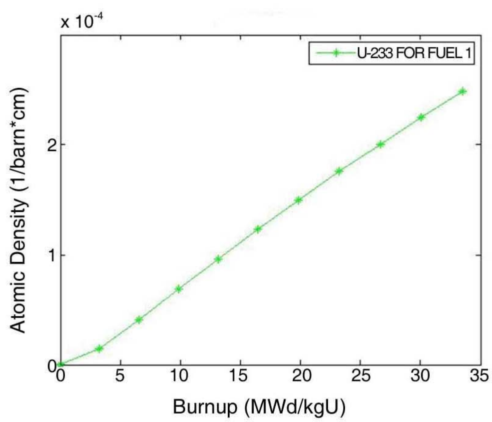
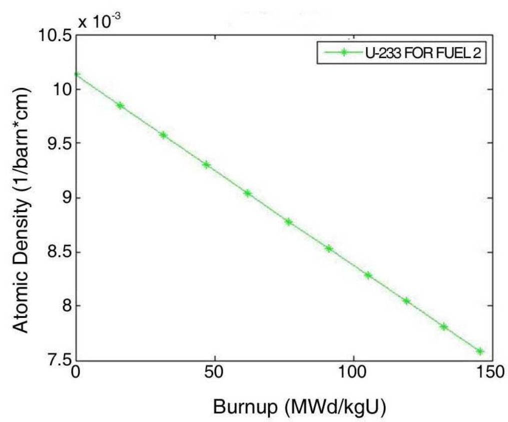


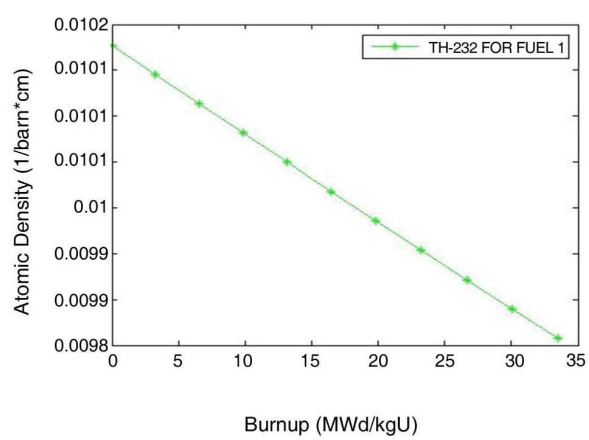
Fig. 6. Minor Actinides (^{237}Np, ^{239}Pu, ^{241}Am and ^{244}Cm) Mass Change in SFR System



(a)



(b)



(c)

Fig. 5. ^{232}Th and ^{233}U Atomic Density Change in Fuel 1 and Fuel 2

as compared to lattice numbers. The better neutron economy of the pin lattice of SFR provides good $^{232}Th/^{233}U$ conversion and transmutation minor actinides. This also means that an SFR has a larger actinide burn-up rate. The SFR consume significant amounts of minor actinides for 600 days in the core.

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