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Performance of natural uranium- and thorium-fueled fast breeder reactors (FBRs) for ^{233}U fissile production

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Abstract

The performance of natural uranium and thorium-fueled fast breeder reactors (FBRs) for producing ^{233}U fissile material, which does not exist in nature, is investigated. It is recognized that excess neutrons from FBRs with good neutron economic characteristics can be efficiently used for producing ^{233}U. Two distinct metallic fuel pins, one with natural uranium and another with natural thorium, are loaded into a large sodium-cooled FBR. ^{233}U and the associated-U isotopes are extracted from the thorium fuel pins. The FBR itself is self-sustained by plutonium produced in the uranium fuel pins. Under the equilibrium state, both uranium and thorium spent fuels are periodically discharged with a certain discharge rate and then separated. All discharged fission products are removed and all discharged actinides are returned to the FBRs except the discharged uranium utilized for fresh fuel of the other thorium-cycled reactors. ^{233}U-production rate of the FBRs as a function of both the uranium–thorium fuel pins fraction in the core and the discharge fuel burnup is estimated. The result shows that larger fraction of uranium pins is better for the FBR criticality while larger fraction of thorium fuel pins and lower fuel burnup give higher ^{233}U production rate.

Keywords: FBR; Mixture core; Thorium; U-233; Discharge constant; Fissile producer

1. Introduction

There are many reasons for the resurgence of interest in the thorium fuel cycle nowadays. The phenomenal increase in the price of uranium and thorium's abundance is about three times of uranium are probably the most compelling reasons. A better physical (neutronic) characteristic of ^{232}Th and the transmuted ^{233}U are attractive because several previous studies have shown some promising reactor physics performances of thorium-fueled core due to those characteristics. Thorium-fueled reactor is also an attractive tool to produce long-term nuclear energy with low radiotoxicity waste. The thorium reactors, however, remind a basic engineering problem, i.e. how the fissile ^{233}U enrichment can be provided for reactor operation because it is impossible to operate the reactors fueled only by natural thorium. Although there is a possibility to enrich the fuel with ^{235}U, but it will not be better than the enriched fuel with ^{233}U. Therefore, it appears a significant basic consideration to produce ^{233}U fissile material through fast breeder reactors (FBRs) by introducing thorium fuel pins in the reactor core.

In a conventional FBR, it is common that FBR core consists of core and blanket regions. Core is located in inner side and fueled by enriched uranium/plutonium, while blanket is located in outer side and usually fueled by natural uranium. In the present study, we propose a uranium and thorium mixture core of the FBR. Uranium and thorium fuel pins are arranged in the core without any blanket surrounding it. There are some advantages of this mixture core design: (i) it is more proliferation resistant than the conventional core because blanket region (in conventional core) is easier to be taken out from the core, (ii) it has less negative void reactivity coefficient, and (iii) it provides higher flux level in the thorium pins so that a higher fissile production can be achieved.

The performance of natural uranium and thorium-fueled fast breeder reactors for producing ^{233}U fissile material was investigated. Uranium-233, which does not exist in nature, has to be generated and accumulated in thorium fuel reactors. We recognize that excess neutrons from large FBRs with good

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neutron economic characteristics can be efficiently used for producing ^{233}U.

In the present study, two distinct metallic fuel pins, one with natural uranium and another with natural thorium, are loaded into a mixture core of large sodium-cooled FBR, where ^{233}U and the associated-U isotopes are extracted from only the thorium fuel pins. The metallic-fueled and sodium-cooled FBR is chosen because it provides a good and most realistic FBR design (Mizutani and Sekimoto, 1998). The FBR itself is self-sustained by the plutonium produced in the uranium fuel pins. ^{233}U production rate of the FBRs as a function of uranium–thorium fuel pin fraction in the core and of discharge fuel burnup was estimated and optimized.

2. Equilibrium fuel cycle scenario

From the early stage of our study on nuclear equilibrium state, equilibrium analysis of liquid metal fast reactors had been performed to achieve an optimized design in fuel and coolant materials composition of the reactor (Sekimoto and Takaki, 1991; Takaki and Sekimoto, 1992; Takaki et al., 1993). It was assumed that in a nuclear equilibrium state, the production and annihilation of each nuclide were balanced in the system for a long time period. Actinides confining and fission products discharging strategies were investigated to reduce radioactive wastes.

Fuel cycle scenario of the present study is shown in Fig. 1, where r_U and r_{FP} are the uranium- and fission product-discharge constants, respectively; which indicate discharge fractions of the corresponding nuclides from the FBR core per year. A large FBR, inside a nuclear park, is equipped with related nuclear fuel facilities, such as separation, storage and fabrication facilities. Natural uranium and thorium fuels are charged to the FBR in distinct fuel pin types. Under the equilibrium state, both the uranium and thorium spent fuels are periodically discharged with a certain discharge constant and then separated. After the separation process, all discharged fission products are removed and all actinides, excluding uranium, are returned to the FBR. Discharged ^{233}U and the associated-U isotopes are then used for fresh fuel of the other thorium-cycled reactors. The basic FBR design including cell parameters is shown in Table 1.

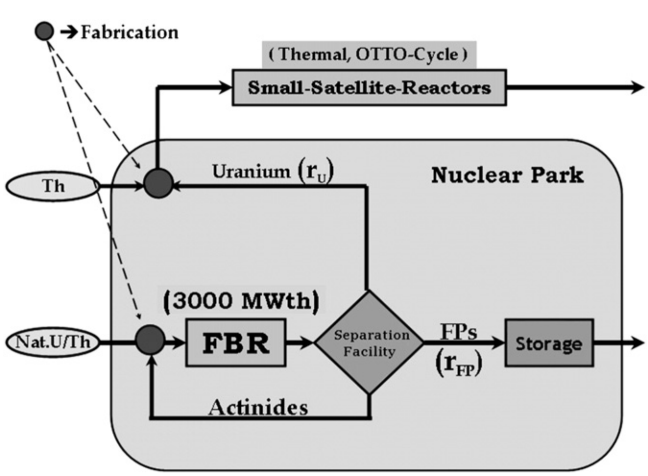

Fig. 1. System's scenario.

3. Calculation method

The Equilibrium Cell Iterative Calculation System (ECICS) code (Mizutani and Sekimoto, 1997) was used for the present calculation. The ECICS method employs an iterative procedure of cell calculation and equilibrium one as shown in Fig. 2 using SRAC-2005 (Okumura et al., 2005) and JENDL-3.2 cross-section library. The nuclear fuel cycle at the nuclear equilibrium state, called *equilibrium fuel cycle*, satisfies the following conditions.

- Number density of each nuclide in reactor does not change with time.
- Refueling process is a continuous process.

In these conditions, the number density of ith nuclide, n_i, should satisfy the following burnup equation:

$$\frac{dn_i}{dt} = -(\lambda_i + \phi\sigma_{a,i} + r_i)n_i + \sum_j \lambda_{j\to i} n_j + \phi \sum_j \sigma_{a,j\to i} n_j + s_i = 0 \tag{1}$$

where ϕ is the neutron flux, λ_i is the decay constant of ith nuclide, r_i is the discharge constant of ith nuclide, $\lambda_{j\to i}$ is the decay constant of jth nuclide to produce ith nuclide, $\sigma_{a,j\to i}$ is the microscopic absorption cross-section of jth nuclide to produce ith nuclide, s_i is the supply rate of ith nuclide, $\sigma_{a,i}$ is the microscopic absorption cross-section of ith nuclide.

Firstly, equilibrium calculation was performed with a proper initial guess of one-group constant set. Then, material balance equation in the fuel pellet was solved with the fixed total number density of all heavy materials and fission products. In the next step, one-group scalar neutron flux level, ϕ, was normalized by the given power density, P, in the fuel pellet:

Table 1
Basic FBR design and its cell parameters

Basic FBR design parameters	
Total power output (MWth)	3000
Power density (W/cm^3)	280
Initial HM inventory (t)	137
Coolant	Sodium
Fuel	(U and Th)Zr$^{10\%}$
Cladding	Ferritic-stainless steel
Cell parameters	
Theoretical density (g/cc)	15.90
Fuel-pellet diameter (mm)	7.09
Pin diameter (mm)	8.50
Pin pitch (mm)	9.85
Cladding thickness (mm)	0.48
Effective fuel volume ratio	0.325
Smear density	75%