Annals of Nuclear Energy 103 (2017) 238–250

Contents lists available at ScienceDirect

Annals of Nuclear Energy

journal homepage: www.elsevier.com/locate/anucene

Investigation on the potential use of thorium as fuel for the Sodium-cooled Fast Reactor

H. György *, Sz. Czifrus

Budapest University of Technology and Economics, Budapest, Hungary

ARTICLE INFO

Article history:
Received 26 October 2016
Received in revised form 16 January 2017
Accepted 19 January 2017

Keywords:
Burnup
Gen IV
SFR
Thorium

ABSTRACT

Generation IV reactors are planned to be an evolutionary step in the history of nuclear power plants. Although they have many advantageous properties, five of the six concepts are designed for the utilization of uranium–plutonium fuel. In this paper, the Sodium-cooled Fast Reactor is investigated regarding the potential application of thorium. The basis of the investigation is the European concept of SFR, which is studied with thorium-containing fuel compositions. Two different approaches are presented with the results of several full-core burnup calculations performed by the Monte Carlo code Serpent 2. The presented results are the multiplication factor changes, delayed neutron fractions, fuel temperature and void coefficients and fissile isotope vectors as well. The results can help to determine how ^{233}U could be produced in this type of reactors and how it could be used as alternative fuel in SFR.

1. Introduction

Today nuclear reactors operate mainly with uranium–plutonium cycle. Since the beginning of nuclear power development, thorium was considered an alternative fuel option for reactors (OECD, 2015). It contains only fertile isotopes which can be converted to fissile ^{233}U. The amount of thorium available in nature is more than three times that of uranium. This property makes it a significant alternative fuel for nuclear reactors (OECD, 2015; Belle and Berman, 1984). Generation IV reactors are widely researched today. They are expected to be deployed commercially around 2030 or 2040 (Generation IV International Forum, 2015). These reactors are based on the evolution of nuclear reactors, thus containing many improvements. The Generation IV International Forum selected six promising reactor-types: the Very-High-Temperature Reactor (VHTR), the SuperCritical-Water-cooled Reactor (SCWR), the Sodium-cooled Fast Reactor (SFR), the Gas-cooled Fast Reactor (GFR), the Molten Salt Reactor (MSR) and the Lead-cooled Fast Reactor (LFR) (Generation IV International Forum, 2015). Each of these reactors has a number of different designs. A previous investigation (György and Czifrus, 2016) showed that SFR reactors can be potentially suitable for the utilization of thorium. In this paper the European concept of the Sodium-cooled Fast Reactor is modified to investigate how thorium could be applied.

2. Thorium

In natural form thorium consists essentially one isotope, ^{232}Th. This is a fertile material, which, after absorbing a neutron, becomes ^{233}Th and decays into ^{233}Pa through beta decay. This protactinium isotope is an intermediate nucleus similar to ^{239}Np in the case of U-Pu cycle. This isotope, in turn, has a significant neutron capture cross-section, which leads to non-fissile ^{234}U. In addition, ^{233}Pa has a half-life of 27 days and after a beta decay it becomes fissile ^{233}U. Because of these properties, ^{233}Pa can be a significant neutron poison in the case of thorium utilization (Belle and Berman, 1984).

Thorium-dioxide is one of the most refractory and chemically non-reactive solid substances available. Compared to uranium-dioxide, its melting point and also its thermal conductivity are higher. In addition, it is not subject to oxidation beyond stoichiometric ThO_2. Experimental results showed that the dimensional behaviour of (Th,Pu) O_2 is similar to that of UO_2 up to about 38 MWd/kg (OECD, 2015).

The thorium cycle has a few disadvantages as well. One is that during the thorium cycle more fission gas is produced per fission; however, based on experiences, thorium dioxide can retain more of these gases than uranium dioxide can. Another important issue is connected to recycling. The produced ^{233}U always contains ^{232}U as a contaminant. This uranium isotope has a half-life of 69 years and its daughter products are intense gamma and alpha emitters

* Corresponding author at: Institute of Nuclear Techniques, Budapest University of Technology and Economics, Budapest, Hungary.

E-mail addresses: hunor.gyorgy@reak.bme.hu (H. György), czifrus@reak.bme.hu (Sz. Czifrus).

http://dx.doi.org/10.1016/j.anucene.2017.01.030
0306-4549/

with short half-lives (e.g. ^{208}Tl emits gamma radiation with 2.6 MeV energy). As a consequence, the radioactivity increases with time for bred uranium isotopes (Vijayan et al., 2013). This requires that these materials should be handled with remote controlled equipment and appropriate shielding. The available information is also less complete than that on UO_2 although various irradiation programmes are carried out, are ongoing or planned nowadays (OECD, 2015; Belle and Berman, 1984).

For the use of thorium, a number of extensive studies have been developed. Different methods suggested for the utilization of thorium can be found in the literature, e.g. for Light Water Reactors (LWR), Heavy Water Reactors (HWR), Spectral-Shift Controlled Reactors (SSCR), which are modified LWRs using light water and heavy water mix, High-Temperature Gas-Cooled Reactors (HTGR), Light Water Breeding Reactors (LWBR), which are also modified LWRs, Molten Salt Reactors (MSR), Fast Breeder Reactors (FBR), CANDU Reactors (Trauger, 1978; Teller, 1978; IAEA, 2003). There is another promising method of utilization of thorium, namely its use in accelerator driven subcritical systems e.g. the Thorium-Driven Fast-Neutron Energy Amplifier (Rubbia, 2013).

In the past, several facilities were used to research thorium, e.g. Elk River BWR, Edison Indian Point-1 PWR, Shippingport PWR, Peach Bottom HTR, Fort St Vrain Reactor in Platteville HTGR, DRAGON HTGR in England, AVR and THTR pebble bed reactors in Germany etc. (OECD, 2015; Sehgal, 2013). In Russia and in France the use of thorium was studied in fast neutron spectrum as well (e.g. in BN-800). The ability of iso-generation of fissile material was demonstrated; however, the performance achieved in this field was lower than that of uranium–plutonium cycle. It is caused by the fission cross section of thorium, which is about three times smaller for fast neutrons than that of ^{238}U (Greneche, 2013). Nowadays theoretical as well as experimental research programs, such as the irradiation program in the Halden reactor aim to determine how thorium could be optimally utilized in nuclear reactors (OECD, 2015; Sehgal, 2013).

Recently, the interest in the thorium cycle has increased and many research programmes on thorium containing reactors are carried out, e.g. (Lopez-Solis et al., 2016; Brown et al., 2015; Lindley et al., 2014; Mohamed and Badawi, 2016).

3. Sodium-cooled Fast Reactor

The main objective of the Generation IV reactors is the improvement in areas of sustainability, safety and reliability, economic competitiveness, proliferation resistance and physical protection (Generation IV International Forum, 2015). The Generation IV International Forum selected six systems out of nearly 100 concepts in 2002 which meet the requirements (Generation IV International Forum, 2015; GEN, 2014). These mostly include fast reactors. Fast reactors with a closed fuel cycle allow for significantly improved exploitation of natural resources and help with a substantial reduction in the amount of waste and their decay heat as well since the long-term repository spaces are limited. The SFR can be an appropriate choice to fulfil the requirements for future reactors. It is also important to mention that sodium-cooled reactors have operated in the past in Europe such as Rapsodie, Phénix, Superphénix, etc. (Fiorini and Vasile, 2011).

The use of sodium as reactor coolant has numerous advantages. The boiling point is high (892 °C). In addition, sodium (similarly to lead) has high heat capacity and thermal conductivity, which can improve the safety of these reactors. Besides, sodium has low absorption cross-section and the moderation is almost negligible. It is also important that this metal is inherently compatible with stainless steel. Compared to lead, sodium-cooled systems require smaller pin pitch and therefore higher fast flux can be produced. On the other hand, in this way slightly lower enrichment is feasible (smaller core with higher power density). Although sodium has many advantages, it also has an important issue, ie. the chemical reactivity with air and with water (Fanning, 2007).

In this paper, the European concept of SFR is investigated. In the International Collaborative Project 24 partners are working together to create a sodium-cooled fast reactor. Two different core layouts were designed: an oxide core layout and a carbide fuel core layout. This paper focuses only on the SFR oxide core layout. The investigated reference core design is the one specified by the SFR Benchmark Task Force of OECD/NEA Working Party on Reactor Systems (WPRS) (Blanchet et al., 2011). The designed thermal power of the chosen concept is 3600 MW. The cycle length is planned to be 410 effective full power days with one fifth reloading scheme. The average burnup is going be around 100 GWd/tHM.

The core can be divided into an inner and an outer core zone with 225 and 228 fuel assemblies, respectively. The geometric parameters of the sub-assemblies can be seen in Table 1. The fuel assemblies are surrounded by 270 radial reflectors. Every fuel assembly has an EM10 wrapper tube, which is an unstabilized 9% Cr fully tempered martensitic alloy. This alloy has an excellent dimensional stability under irradiation and it was already investigated in the Phénix reactor (Yvon et al., 2015). The number of fuel pins is 271, being designed with (U,Pu) O_2 pellets. Due to the high achievable burnup and the neutron dose, oxide dispersion strengthened (ODS) steel (Yvon et al., 2015) was chosen as cladding. Axially, the fuel pin contains two gas plena and steel pellets as axial reflectors are located under and above the active part. The plutonium content is axially different for both the fuel assemblies which are in the inner and in the outer core. The active length can be divided into five axial concentration sets. For the inner core, the average Pu content is 15.7% while for the outer core it is 17.5%.

4. Calculations

For the burnup calculations a three-dimensional continuous-energy Monte Carlo reactor physics burnup calculation code, the Serpent 2 was used. Regarding the speed in burnup calculations, the performance of this code is better than other general-purpose Monte Carlo codes because it uses the Woodcock delta-tracking method and a single unionized energy grid for all microscopic and macroscopic cross sections. For depletion calculations, besides the conventional algorithms (Euler and predictor–corrector method with linear interpolation for the corrector calculation) higher order methods are available as well. The Bateman depletion equations are solved with CRAM matrix exponential method. The results of the program were compared with benchmarks and it was compared to other codes as well (Leppänen, 2013; Pusa and Leppänen, 2010; Leppänen et al., 2013). The code is widely used

Table 1
SFR fuel sub-assembly dimensions.

Central hole radius	0.1257 cm
Fuel pellet radius	0.4742 cm
Inner clad diameter	0.9786 cm
Outer clad diameter	1.0838 cm
Pin to pin distance	1.1897 cm
Sub-assembly pitch	21.2205 cm
Sub-assembly duct wall thickness	0.4525 cm
Active core height	100.56 cm
Lower gas plenum	89.91 cm
Upper gas plenum	10.05 cm
Lower axial reflector	30.17 cm
Upper axial reflector	80.45 cm