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Possible scenarios for the transition to thorium fuel cycle in molten salt reactor by using enriched uranium

D.Y. Cui [a, b, c], X.X. Li [a, b], S.P. Xia [a, b, c], X.C. Zhao [a, b, c], C.G. Yu [a, b], J.G. Chen [a, b, c, *], X.Z. Cai [a, b, c, **]

[a] Shanghai Institute of Applied Physics, Chinese Academy of Sciences, Shanghai 201800, China
[b] CAS Innovative Academies in TMSR Energy System, Shanghai 201800, China
[c] University of Chinese Academy of Sciences, Beijing 101408, China

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A B S T R A C T

Molten Salt Reactor (MSR) with Th-^{233}U fuel cycle attracts more and more attention with its excellent performance such as desirable breeding capacity, low waste production and high inherent safety. Considering the fact that there is no available ^{233}U in the nature, it is necessary to analyze the fuel transition from enriched ^{235}U/Th to ^{233}U/Th and then give a flexible transition scenario for a graphite-moderated MSR. By employing an in-house developed tool which is based on SCALE6.1, two scenarios, a Breeding and Burning (B&B) scenario and a Pre-breeding scenario, are studied. The evolution of the inventories of main nuclides, net ^{233}U production and isothermal temperature coefficient are presented and discussed in the B&B scenario. It is found that the fuel transition can be achieved smoothly by using enriched uranium with greater than 40% concentration of ^{235}U. The fuel transition can still be accomplished with 20% enriched uranium but takes a long double time of about 79 years. Meanwhile, we perform an analysis of the Pre-breeding scenario and conclude that it is efficient to produce ^{233}U and the double time ranges from 2.07 years for the 10-day reprocessing to 10.7 years for the 180-day reprocessing. A comparison of these two scenarios is conducted, which indicates that the B&B scenario is more favorable than the Pre-breeding scenario from the aspect of resource utilization efficiency. Finally, a combined three-stage program for developing Th-based MSRs is proposed.

1. Introduction

Molten Salt Reactor (MSR), as one of the six advanced Generation IV nuclear systems, offers a number of advantages over its solid-fueled counterparts. The history of research and development (R&D) on MSR is briefly reviewed as follows. It was initially studied by Oak Ridge National Laboratory (ORNL) with an Aircraft Reactor Experiment (ARE) project (Cottrell et al., 1955), in which a reactor employing a molten fluoride salt containing highly enriched uranium (e = 93.4%) was built and firstly demonstrated the feasibility of liquid fuel. Inspired by the success of ARE, ORNL built a 8 MWth graphite-moderated reactor in the 1960s, namely the Molten Salt Reactor Experiment (MSRE) (Haubenreich and Engel, 1970), to attempt to apply the MSR concept to electricity production. This experimental reactor operated successfully from 1965 to 1969 with fuels of enriched ^{235}U, ^{233}U and ^{233}U/^{239}Pu at different times, which provided significant technical foundation for the later study. Afterwards, a 1 GWe graphite-moderated Molten Salt Breeder Reactor (MSBR) (Robertson, 1971; Bettis and Robertson, 1970) based on Th-^{233}U fuel cycle was designed. Unfortunately, the MSR development program at ORNL ceased in 1973 due to fund shortage and other reasons (MacPherson, 1985). In the later 1970s, a Denatured Molten Salt Reactor (DMSR) fueled with denatured ^{235}U for non-proliferation was designed (Abbott et al., 1979; Engel et al., 1980) with limited work continued at ORNL. Henceforth, some effort has been made to develop this promising technology and in 2002, a worldwide renewed interest of MSR was stimulated and it was selected as one of the six systems in Generation IV International Forum (GIF) for future advanced reactors R&D (Abrams et al., 2002). Nowadays, the research on MSR directed by ORNL has been further

* Corresponding author. Shanghai Institute of Applied Physics, Chinese Academy of Sciences, Shanghai 201800, China.
** Corresponding author. Shanghai Institute of Applied Physics, Chinese Academy of Sciences, Shanghai 201800, China.
E-mail addresses: chenjg@sinap.ac.cn (J.G. Chen), caixz@sinap.ac.cn (X.Z. Cai).

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developed to a serials of concepts (Serp et al., 2014), such as the FUJI-MSR in Japan (Adamovich et al., 2007; Shimazu, 2010), the MOSART in Russia (Ignatiev et al., 2007), the AMSTER (Vergnes and Lecarpentier, 2002) and MSFR (Rubiolo et al., 2013; Heuer et al., 2014) in France. The TMSR project in China (Xu, 2013; Xu et al., 2014), launched in 2011, are developing both the liquid-fuel TMSR and the solid-fuel TMSR for realizing effective thorium energy utilization within 20–30 years.

Many studies have proven the considerable merits of MSR with Th-^{233}U fuel cycle in aspects of safety, breeding capacity, low radioactive waste production and non-proliferation (Lung and Gremm, 1998; Nuttin et al., 2005; Hargraves and Moir, 2010). Meanwhile, thorium, as an appealing alternative nuclear fuel to uranium, is more abundant than uranium in the earth's crust (Vance, 2014) and thus it would contribute greatly to nuclear energy supply for global development if a sustainable Th-^{233}U fuel cycle could be realized in MSRs. Nevertheless, some challenges should be addressed in order to deploy a Th-based MSR successfully. One of the important and urgent tasks is to produce enough ^{233}U for startup of MSR. It is believed that two solutions are potentially available to produce ^{233}U. One is attaining ^{233}U from other systems, and the other is MSR self-breeding system.

As for other systems for ^{233}U production, the light water reactor, which can use a fissile fuel and prime the thorium fuel, is a typical example (Yun et al., 2010). A three-stage program (Vijayan et al., 2016; Jain, 2010) launched by India is another example. In this program, ^{233}U is produced from the plutonium fueled fast breeder reactors in stage-II and further employed in molten salt breeder reactors in stage-III for larger scale thorium utilization. Furthermore, both the accelerator driven system (ADS) (Barros et al., 2015; IAEA, 2015) and fusion-fission hybrids system (Moir, 2008) are promising options which can provide neutrons to breed ^{233}U just with the help of ^{232}Th. In Japan, an accelerator molten-salt breeder (AMSB) system was designed (Adamovich et al., 2007) and it can produce about 400 kg/year of ^{233}U. However, the technology of high current proton accelerator has not been proved. Also, the fusion-fission hybrids system featuring fissile fuel production and minimized waste disposal, is attractive but not applicable at present (Freidberg and Kadak, 2009; Ragheb, 2009).

With respect to the MSR self-breeding system, it can be started with currently available fissile material and produce ^{233}U with the help of ^{232}Th. At present, enriched uranium is commonly used in operating thermal reactors and importantly, it has been tested in ARE and MSRE. Therefore, it is expected to employ enriched uranium in such a MSR self-breeding system. Also, much work associated with using enriched uranium in MSRs has been conducted. In the ORNL's work on MSBR, 93% enriched uranium was used (Perry and Bauman, 1970) and the analyses concluded that the startup can be accomplished with enriched uranium and the 20-year average breeding ratio (BR) is close to that with ^{233}U. Regarding to the work of Nuttin et al. (2005), a single-fluid MSBR core was reevaluated and employed to breed ^{233}U with starter fuel of both 90% and 33% enriched uranium. During the operation, it is assumed that ^{233}U produced from extracted ^{233}Pa is fed back into the core to keep critical and the excess ^{233}U is stored outside the core. It was found that the double time ranged from 45 years for 90% enriched uranium to 70 years for 33% enriched uranium. However, as pointed out by Nuttin et al., the produced ^{233}U is not enough to offset its depletion during the early operation stage and the maximum deficit of ^{233}U is about 200 kg when 33% enriched uranium is used. Moreover, enriched uranium is also adopted in the FUJI-U3 (Sakuraba and Yoshioka, 2010). It indicates that the initial mass of ^{235}U is 2–4 times greater than that of ^{233}U started case and the initial conversion ratio is from 0.76 to 0.86 when ^{235}U enrichment varies from 5% to 20%. The further analysis of the performance over the reactor life has not yet been given.

The previous work has established a significant foundation for our work. However, some important issues, as mentioned above, are still needed to be addressed. Therefore, a detailed analysis on the MSR self-breeding system is performed in this work. Two scenarios are proposed according to the fact whether the ^{233}U produced from the decay of the extracted ^{233}Pa is added back to the reactor. If it is refueled, we name this scenario as the breeding and burning (B&B) scenario, otherwise, it is described as the Pre-breeding scenario. The comparison of these two scenarios is expected to establish a feasible strategy for the transition to thorium fuel cycle in a MSR.

The methodology for the core geometry and the calculation tool are introduced in Section 2. The B&B scenario with progressive replacement of ^{235}U by ^{233}U in the core, is discussed in detail in Section 3. Moreover, the Pre-breeding scenario will be presented in Section 4. A comparison of the two scenarios is presented in Section 5 and the conclusions are summarized in Section 6.

2. Methodology

2.1. The geometry and material descriptions

The reference core used in this work is based on the single-fluid, two-zone MSBR designed by ORNL (Robertson, 1971). In such a conceptual design, a 2250 MWth breeder reactor operating on thorium fuel cycle is presented with moderately good performances, such as the relatively high BR (1.06) and the low initial fissile load. A fast, continuous fuel salt processing is employed in the design in order to enhance the breeding capacity, although separation of thorium from lanthanide fission products is rather challenging within a short cycle time.

The quarter vertical section of our geometrical model is shown in Fig. 1. Table 1 summarizes the detailed geometry parameters. The core is divided into two parts: zone 1 and zone 2 with 13% and 37% salt fraction respectively. Both zones are consisted of a larger number of graphite hexagons (10 cm pitch), each pierced by a channel for salt circulation. The fuel salt used in both zones is the same and contains both fissile and fertile materials. The salt plenum above and below the core, is used to direct the salt flow and the radial annulus provides clearance when replacing the core assembly. Around the core, a 76 cm radial reflector and a 100 cm axial reflector are arranged to minimize the neutron leakage. The heat exchanger is used to simulate the primary loop of the reactor outside the vessel. The B_4C layer with thickness of 10 cm is to protect the vessel from neutron irradiation. The total salt volume is 48.8 m^3.

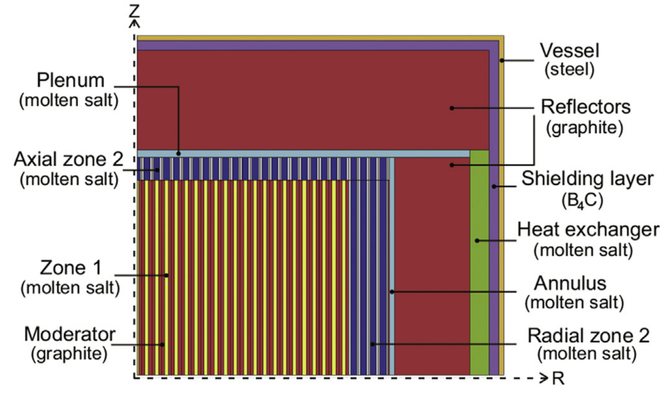

Fig. 1. Geometrical description for the quarter of the reactor core.