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Fuel cycle analysis of molten salt reactors based on coupled neutronics and thermal-hydraulics calculations

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ABSTRACT

A new fuel cycle analysis code named FAMOS was developed for molten salt reactor (MSR) applications by extending the DIF3D code to model the transport of delayed neutron precursors and by coupling the extended DIF3D code with an in-house multi-channel thermal-hydraulics solver and an in-house nuclide depletion solver for the simulation of fuel depletion, reprocessing and feeding. FAMOS can also calculate the effective delayed neutron fraction accounting for the motion of fuel salt, the neutron generation time and the reactivity feedback coefficients. In order to verify the FAMOS code, the molten salt fast reactor (MSFR) proposed in the EVOL (Evaluation and Viability of Liquid Fuel Fast Reactor System) project was analyzed using FAMOS and the calculation results were compared with the reported results. The numerical results indicate that FAMOS is capable of simulating the fuel cycle process of MSRs with good accuracy. FAMOS was applied to the fuel cycle analysis of the molten salt breeder reactor (MSBR) as well. The effects of the temperature distribution and the drift of delayed neutron precursors on the neutronics characteristics and fuel cycle performances of MSBR were evaluated using three different calculation models. The drift of delayed neutron precursors and the temperature distribution shift the axial power distribution upward and downward, respectively, and the latter effect is stronger than the former one. However, the impacts of the temperature distribution and the drift of delayed neutron precursors on the depletion and buildup behaviors of actinides and fission products in MSRs are negligible. It is also possible to achieve a ^{233}U conversion ratio larger than 1.0 without protactinium removal and thus to operate the MSBR by only feeding ^{232}Th, except for the initial start-up stage.

1. Introduction

Molten Salt Reactors (MSRs) with the fuel dissolved into the liquid salt are selected as one of the six Generation-IV reactor types due to the excellent characteristics in terms of sustainability, economy, passive safety and resource utilization (Serp et al., 2014). The conversion ratio of MSRs can be flexibly adjusted by changing the fissile-to-fertile material ratio in the fuel salt. MSRs eliminate the fuel element fabrication and transport, and excess reactivity can be minimized by adjusting the fuel salt composition. Large negative temperature and void reactivity coefficients of the fuel salt ensure the passive safety of MSRs. In addition, the passive core drain system is generally considered for improving the safety of MSRs. High temperature and large power density can be reached without high internal pressures. Deep burnup can be achieved in MSRs by removing the fission products from the salt to reduce the parasitic neutron capture and feeding the fissile and fertile materials in either continuous or batchwise operation (Holcomb et al., 2011).

In order to simulate the fuel cycle process of MSRs, the unique physical characteristics of MSRs should be considered accurately. First, the significant thermal expansion property of liquid fuel tightly couples the neutronics and thermal-hydraulics characteristics so that the power density distribution and the fuel salt temperature distribution can influence each other significantly. The short circulation time (several seconds to tens of seconds) and the mixing of fuel salt in the primary circuit allow the fuel salt composition in the primary circuit system (excluding short-lived fission products) to be treated homogeneous with a variable density distribution of fuel salt. On the other hand, the delayed neutron precursors generated in the core may decay in positions of low importance and even out of the core. The drift of the delayed neutron precursors reduces the effective delayed neutron fraction signifi-

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cantly (Aufiero et al., 2014). Thus, the motion of the delayed neutron precursors should be considered explicitly in the neutronics calculation.

During the past few years, various fuel cycle analysis codes were developed to analyze the neutronics characteristics and fuel cycle performances of MSRs. In most of the previous works, under the assumption of the homogeneous fuel salt composition in the primary circuit system, the primary circuit system was treated as a single point in the depletion calculation and the fuel salt reprocessing and feeding processes were simulated by continuous or batchwise extraction and addition of materials. For example, Nuttin et al. (2005) developed a code for MSR fuel cycle analysis by coupling the Monte Carlo code MCNP (Briesmeister, 1997) with the in-house depletion code REM and simulated the continuous extraction of fission products as a fictitious decay process. Aufiero et al. (2013) extended the Serpent code (Leppanen et al., 2015) to analyze the fuel cycle of MSRs by simulating the fuel reprocessing and external feeding as continuous processes. Powers et al. (2013) and Sheu et al. (2013) developed the fuel cycle analysis procedures for MSRs based on SCALE/TRITON (ORNL, 2011) and modeled the batch-wise fuel salt reprocessing and feeding processes by directly modifying the fuel salt composition in the core between two successive burn steps. Doligez et al. (2014) coupled the depletion process in the primary circuit system with the radioactive decay and chemical extraction processes in the reprocessing units by introducing a nuclide exchange term between two connecting units in the depletion equations.

In these previous MSR fuel cycle studies, thermal feedback was not explicitly considered despite of the tightly coupled neutronics and thermal-hydraulics characteristics in MSRs. The motion of the delayed neutron precursors was also neglected in the previous fuel cycle analyses. In addition, they treated the primary circuit system as a single point in the depletion analysis using the core averaged microscopic reaction rates. In this study, a new code for MSR fuel cycle analysis named FAMOS (Fuel cycle Analysis code for MOlten Salt reactors) was developed in order to analyze the fuel cycle process of MSRs considering thermal feedback explicitly. The deterministic neutronics code DIF3D (Derstine, 1984) was extended to solve the coupled system of neutron diffusion and precursor convection equations simultaneously for MSR analysis (Park et al., 2017), and the extended DIF3D code was coupled with an in-house multi-channel thermal-hydraulics code for liquid-fueled reactors and an in-house nuclide depletion solver.

In the fuel cycle analysis with FAMOS, the motion of delayed neutron precursors is explicitly modeled in core neutronics calculation using the extended neutronics solver DIF3D and thermal feedbacks are taken into account by coupled neutronics and thermal-hydraulics calculations. In the depletion analysis, the primary circuit system is represented by two points: one for the core and the other for the external loop. The nuclide transmutations in the core are calculated using the core-averaged one-group microscopic cross sections obtained from the coupled neutronics and thermal-hydraulics calculations. In the external loop, the fuel reprocessing and feeding processes as well as radioactive decays are considered. The nuclide inventories in the core and in the external loop are coupled through the flow continuity conditions. At the user-specified time points during the fuel cycle, FAMOS also calculates safety parameters such as the delayed neutron fraction, neutron generation time, and reactivity feedback coefficients. The effective delayed neutron fraction is calculated using the distributions of delayed neutron precursor concentrations obtained from the coupled neutronics and thermal-hydraulics calculations.

Verification tests of the FAMOS code were performed using the Molten Salt Fast Reactor (MSFR) benchmark problem proposed in the EVOL (Evaluation and Viability of Liquid Fuel Fast Reactor System) project (Merle-Lucotte et al., 2011). The static neutronics results and the fuel salt evolution results were compared with those obtained with other codes. The FAMOS code was applied to the fuel cycle analysis of the Molten Salt Breeder Reactor (MSBR) designed by the Oak Ridge National Laboratory (ORNL) (Robertson et al., 1971) as well. The effects of the temperature distribution and the motion of delayed neutron precursors on the neutronics and fuel cycle performances of MSBR were assessed. The impact of the protactinium removal from the fuel salt on the fuel cycle performance was also evaluated for MSBR.

This paper is organized as follows. In Section 2, the methodology adopted in the FAMOS code is discussed focused on the fuel cycle analysis and the evaluation of safety parameters. Section 3 presents the numerical results for the MSFR and MSBR test problems. Finally, conclusions are drawn in Section 4.

2. Methodology

2.1. Fuel cycle analysis

The FAMOS code consists of three main computational modules as shown in Fig. 1: neutronics calculation module, multi-channel thermal-hydraulics module, and depletion calculation module. Homogenized multi-group microscopic cross sections are prepared for user-specified spatial regions as a function of fuel salt temperature and density (and moderator temperature if moderator is employed) using the Monte Carlo code OpenMC (Romano et al., 2015). In order to account for the thermal feedback effect on the fuel cycle analysis, the core-averaged one-group microscopic cross sections of individual nuclides are determined by the coupled neutronics and thermal-hydraulics calculation with the extended DIF3D code and an in-house multi-channel thermal-hydraulics code.

The coupled neutronics and thermal-hydraulics calculation procedure is shown in the dashed line box in Fig. 1. First, the power and neutron flux distributions are obtained from the neutronics calculation with an initial guess of the fuel salt temperature and density distributions (and possibly the moderator temperature distribution). The region-dependent, multi-group microscopic cross sections required for the DIF3D calculation are obtained by interpolating the tabulated cross sections for specified fuel salt temperature and density (and possibly the moderator temperature). Using the obtained power distribution, the fuel salt temperature distribution and the corresponding fuel density distribution are determined by the multi-channel thermal-hydraulics calculation. If moderator is used, the moderator temperature distribution is also calculated. With the updated microscopic cross sections and the nuclide densities in each region, the neutronics calculation is performed to obtain new power and neutron flux distributions. The neutronics and thermal-hydraulics calculations are iterated until the power and fuel salt temperature distributions (and possibly moderator temperature distribution) converge.

Using the converged flux solution, core-averaged one-group microscopic cross sections are calculated. With the core-averaged microscopic cross sections and radioactive decay constants, the nuclide transmutation matrix for the fuel depletion in the core is constructed. The transmutation matrix for the external loop is also constructed considering fuel reprocessing and external feed rates as well as radioactive decays. The nuclide exchanges between the reactor core and the external loop are modeled by the flow continuity conditions as described in detail in Section 2.1.3. The resulting system of depletion equations is solved using the matrix exponential solver EXPOKIT (Sidje, 1998). In order to account for the flux variation during a burn step, the nuclide densities at the end of burn step is iteratively calculated as discussed in Section 2.1.3. To take into account the dependency of microscopic