Annals of Nuclear Energy 122 (2018) 207–216

Contents lists available at ScienceDirect

Annals of Nuclear Energy

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

A study of the impact of mixed mode of fuel salt in plenum on steady-state performance of channel-type molten salt reactor

Kun Zhuang [a,*], Liangzhi Cao [b]

[a] College of Material Science and Technology, Nanjing University of Aeronautics and Astronautics, Nanjing, JiangSu 211106, PR China
[b] School of Nuclear Science and Technology, Xi'an Jiaotong University, Xi'an, ShannXi 710049, PR China

ARTICLE INFO

Article history:
Received 29 April 2018
Received in revised form 17 July 2018
Accepted 30 August 2018

Keywords:
Molten salt reactor
Mix mode in plenum
Neutronics
Thermal-hydraulics

ABSTRACT

In this study, a coupled neutronics and thermal-hydraulics (THs) code named MOREL2.0, based on multi-group diffusion theory and multi-channel THs model, was employed to analyze a channel-type molten salt reactor TMSR-LF and the impact of different fuel mix mode in plenum on its steady-state performance was also discussed. MOREL2.0 adopts "two-step" calculation scheme, in which few-group homogenized parameters are generated by lattice code PIJ, and diffusion equation considering the drift of delayed neutron precursors is coupled with multi-channel THs model, and the least square method is implemented for cross-section feedback in coupling scheme. The numerical results indicate that TMSR-LF has negative reactivity coefficients in term of inlet fuel temperature and core power level, and the response of k_{eff} to mass flow is different for cases of nominal power and zero power, and k_{eff} decreases with the rise of time of fuel salt spent in external loop and gradually tends to steady. Different mix mode in plenum have a greater impact on the mass flow distribution and delayed neutron precursors distribution, but less impact on core lumped parameter such as k_{eff} and effective delayed neutron fraction.

1. Introduction

A Molten Salt Reactor (MSR) is characterized by adopting molten fluoride salts in which fissile and/or fertile elements are dissolved. The liquid fuel salt serves as coolant simultaneously and circulates in the primary circuit through the core and heat exchanger. The concept of MSR was first proposed in an air reactor experiment (Bettis et al., 1957) by Oak Ridge National Laboratory in the late of 1940s. Then the successful operation of a molten salt reactor experiment (MSRE) demonstrated the feasibility of MSR technology (Haubenreich, 1969). After that, the concept design of molten salt breeder reactor (MSBR) was developed to investigate the possibility of MSR acting as a breeder reactor (Rosenthal et al., 1972).

Renewed interests in further MSR study emerged after MSR being one of six candidates in the Generation IV reactor system that aims to develop nuclear reactor with better safety, proliferation resistant and economic characteristics than light water reactor (LWR) (Buckthorpe, 2017). The EURATOM 5th, 6th and 7th Framework Program (MOST, ALISIA, EVOL project) were carried out to review MSRs technology based on MSRE operation experience and confirmed whether MSR have the potential to be a breeder or burner (Brovchenko and Merle-Lucotte, 2013; MOST, 2004). Those projects leaded to the design of two fast spectrum MSR concepts: Molten Salt Actinide Recycler & Transmuter (MOSART) (Ignatiev et al., 2007) and Molten Salt Fast Reactor (MSFR) (Fiorina et al., 2014). The design of thermal spectrum MSR system with LiF-BeF_2 carrier salt was implemented for development of structural material under SPHINX project in the Czech Republic (Hron et al., 2002). In Japan, a MSR concept was proposed for production of fissile U233 by a combination of fission power reactor of MSR (MSR-FUJI) and accelerator molten salt reactor (AMSR) (Furukawa et al., 2008). In 2011, Chinese Academy of Sciences launched a project of Thorium Molten Salt Reactor Nuclear Energy System (TMSR) aiming at realizing effective thorium utilization and hydrogen production based on Liquid-Fuel TMSR (TMSR-LF) and Solid-Fuel TMSR (TMSR-SF) during coming 20–30 years (Jiang et al., 2012). The R&D for graphite-moderated TMSR-LF provides fundamental and valuable guidance in TMSR program and promote innovative concept design of MSR.

Lots of MSR studies have been carried out in several literatures. Kophazi et al. modified the MCNP4C (A Monte Carlo N-Particle Transport Code, Version 4C) code to calculate the loss of delayed neutron precursors (DNPs) considering the transport of DNPs (Kópházi et al., 2003). Collins et al. analyzed MSRE power distribution, coolant temperature, and DNP concentrations based on MPACT + CTF tool (Collins et al., 2017). Yamamoto et al. calculated

* Corresponding author at: 29 Jiangjun Avenue, Nanjing 211106, PR China.
E-mail address: kzhuang@nuaa.edu.cn (K. Zhuang).

https://doi.org/10.1016/j.anucene.2018.08.048
0306-4549/

the peaks of the neutron fluxes and DNP distributions for a small molten salt reactor by coupling two-group diffusion model and energy conservation equations (Yamamoto et al., 2005). Zhang et al. studied the distributions of the velocity, temperature, neutron fluxes, and DNPs for a non-structured MSR based on a steady-state analysis code (Zhang et al., 2009b). Alexander et al. developed a MSR analysis code based on multi-physics MOOSE framework and verified it by MSRE experiment results and other coupled models (Lindsay et al., 2018). Dulla et al. studied the effect of fluid dynamics on reactivity and effective delayed neutron fractions by using of coupled fluid-dynamics models and neutronics diffusion models (Dulla and Ravetto, 2007). Shi et al. performed MSBR transient analysis in terms of load demand change, variation of primary flow and secondary flow by coupling point kinetic model and RELAP5 code (Shi et al., 2016). Li and Wang performed MSFR transient analysis by using of fully coupled neutronics and thermal-hydraulics (THs) codes (SIMMER and COUPLE) for several accident conditions, such as unprotected loss of heat sink (ULOHS), unprotected loss of flow (ULOF), and unprotected transient over power (UTOP) (Li et al., 2015). Křepel et al. analyzed transient characteristics of MSBR by DYN3D-MSR code under conditions of unprotected pump coast-down, reactivity insertion, and overcooling of the fuel. Zhang et al and Guo et al performed the transient analysis of MOSART under state of ULOF, ULOHS and unprotected overcooling accident based on coupled point kinetic model and single-channel TH model (Guo et al., 2013; Zhang et al., 2009a).

It can be seen that some researchers studied non-structured MSRs by coupling three-dimensional (3D) DNP flow model and 3D THs model (Dulla et al., Zhang et al. and Wang et al.), and other studies focused on steady and transient analysis of channel-type MSRs based on point reactor model, one-dimensional (1D) DNP flow model and 1D multi-channel THs model (Kophazi et al., Yamamoto et al., Zhang and Guo et al., Křepel et al., Shi et al.). Normally, non-structured lower and upper plenum locate respectively below and above the graphite channels in channel-type MSRs such as MSRE and TMSR-LF to distribute and collect fuel slat. Fuel salt flows axially in graphite channel and flows unrestrictedly in lower and upper plenum. Consequently, 1D DNP flow model and 1D THs model for the simulation of fuel salt flow in graphite channel are acceptable, and the simulation of fuel salt flow in lower and upper plenum requires a 3D model. However, almost all studies of channel-type MSR ignored the unrestricted flow of fuel salt in plenum and adopted 1D DNP model and 1D THs model for both graphite channel region and plenum region. The influence of mixed model of fuel salt in plenum region on performance of channel-type MSR is rarely found in published literatures. In order to provide profound understanding of this influence, two extreme cases, named non-uniform mix model and uniform mix model, were introduced into an in-house development MSR analysis code MOREL2.0 (Zhuang et al., 2015; Zhuang et al., 2017). In non-uniform mix model, the lower and upper plenum are divided into several separate flow channels and fuel salt in different channels does not mix with each other. Conversely, the fuel salt in lower and upper plenum is assumed to mix ideally for uniform mix model. In present study, the modified MOREL2.0 code was employed to analyze the influence of mixed mode of fuel salt in plenum on steady-state performance of TMSR-LF, including loss of effective delayed neutron fraction, DNP and flux distribution, mass flow distribution, fuel salt and graphite temperature distribution, reactor power and inlet fuel temperature reactivity coefficients, and the relationship between effective multiplication factor (k_{eff}) and time of fuel salt spent in external loop. The numerical results indicate that TMSR-LF has negative reactivity coefficients in term of inlet fuel temperature and core power level, and the response of k_{eff} to mass flow is different for case of nominal power and zero power, and k_{eff} decreases with the rise of time of fuel salt spent in external loop and gradually tends to steady. Different mix mode in plenum have a greater impact on the mass flow distribution and delayed neutron precursors distribution, but less impact on core lumped parameter such as k_{eff} and effective delayed neutron fraction.

The neutronics models adopted in MOREL2.0 are briefly introduced in Section 2. The descriptions of TMSR-LF, numerical results and some discussions are presented in Section3. Finally, some conclusions are summarized in Section 4.

2. Brief introduction to simulation tool MOREL2.0

2.1. Description of MOREL2.0

In our previous study, a fully 3D steady/transient analysis code MOREL2.0 for channel-type MSR has been developed by coupling multi-group neutron diffusion model and a multi-channel THs model (Zhuang et al., 2015; Zhuang et al., 2017). MOREL2.0 code system adopts the "two-step" calculation scheme, as shown in Figs. 1, 1) few-group homogenized macroscopic cross-sections at discrete operation states (different fuel salt temperature and graphite temperature) are produced by the a 2D transport lattice code PIJ based on JENDL-3.3 nuclear data library; 2) multi-group diffusion equations considering the drift of delayed neutron precursors are coupled with multi-channel THs model to provide spatial distribution of power density, mass flow and temperature; 3) the least square method (LSF) is implemented for cross-section feedback in coupling scheme. In neutronics and THs coupling calculation, an extended neutron diffusion solver based on variational nodal method (VNM) for liquid-fuel molten salt reactor is employed to obtain power density distribution with assumption of uniform fuel and graphite temperature in the beginning iteration. Then, the new temperature distributions are calculated by multi-channel THs calculation based on new power density. And the diffusion calculation is performed in the next iteration with new region-dependent homogenized parameters that are updated based on the region-

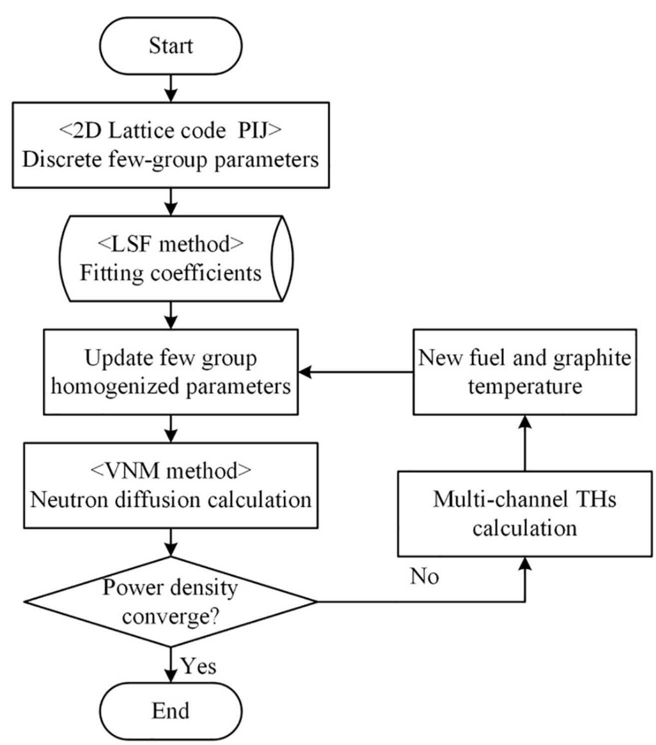

Fig. 1. The steady calculation scheme in MOREL2.0 code.