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Dynamics analysis of the graphite-moderated channel-type molten salt reactors based on Serpent/NTH3D-MSR

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ABSTRACT

The special characteristic of the Molten Salt Reactors (MSR) is that the liquid molten salt mixtures, acting like a coolant and fuel, are flowing through the core and the primary circuit. The delayed neutron precursors are driven by the fuel flow, which results in unique neutron dynamics characteristics of the MSR. The coupled neutronics and thermal-hydraulics code for MSR named NTH3D-MSR has been developed in this paper to simulate the dynamics characteristics of the MSR. In this code, the Predictor-Corrector Improved Quasi-Static (PC-IQS) method and the Nodal Green's Function Method (NGFM) were adopted to solve the neutron kinetics equation and neutron diffusion equation respectively, and the multi-channel thermal hydraulic model is developed to consider the thermal feedback. In addition, the few-group constants of the graphite-moderated channel-type molten salt reactor are generated by the continuous-energy Monte-Carlo code Serpent which can provide remarkable geometry flexibility against the deterministic lattice transport codes. In order to verify the NTH3D-MSR code, the experimental data collected in the European project MOST are adopted to validate the coupled neutronics and thermal-hydraulics model of the NTH3D-MSR code. The numerical results indicate that the Serpent/NTH3D-MSR code has the capability for the dynamics analysis for the graphite-moderated channel-type molten salt reactors.

1. Introduction

Many meaningful codes were developed to analyze the transient behavior and safety performance of the Molten Salt Reactor (MSR) (Zhang et al., 2009a; Zhang et al., 2009b; Krepel et al., 2005; Krepel et al., 2007). The neutronics models implemented in these codes include both Monte Carlo methods and deterministic methods. In the Monte Carlo respect, BUTE used a modified Monte Carlo code MCNP4C to take into account the transport of delayed neutron precursors (Kophazj et al., 2003), but the Monte Carlo code is time consuming. In the deterministic respect, the code DYN1D-MSR (Krepel et al., 2005) based on the solution of two-group one-dimensional neutron diffusion equation by using a nodal expansion method was developed by the Forschungszentrum Rossendorf (FZR), then the code DYN3D-MSR (Krepel et al., 2007) was developed and it allows transients simulation by 3D neutronics and parallel channel thermal-hydraulics of the MSR. The transient code Cinsf1D (Lecarpentier and Carpentier, 2013) based on the two-energy-group diffusion theory with one dimension of space was developed by the Electricité de France (EDF) to achieve critical or kinetics calculations for an MSR. The transient code (Dulla et al., 2003) based on diffusion theory with two-dimensional cylindrical geometry was developed by the Politecnico di Torino (POLITO) to analyze the kinetics for the MSR, and it adopted the quasi-static approximation associated to a modified point kinetic equation. The codes SIMMER (Shisheng et al., 2006) and SimADS (Schikorr, 2001) developed by Forschungszentrum Karlsruhe (FZK) also were used to carry out the dynamics calculations for the MSR, the code SIMMER is based on the neutron transport theory and the code SimADS is based on point kinetics model. In the following sections, results of the codes SIMMER and SimADS were identified by the marks FZK(a) and FZK(b), respectively. Recently, a coupled neutronics and thermal-hydraulics code MOREL was developed by the Xi'an Jiaotong University in China, in the MOREL code, the lattice code HELLOS was used to generated the homogenized few-group macro cross sections, and the variational nodal method was adopted to solve the neutron diffusion equation (Zhuang et al., 2015). A coupled neutronics and thermal-hydraulics code based on diffusion theory with two-dimensional cylindrical geometry was developed by the Shanghai Institute of Applied Physics of the Chinese Academy of Science to carry out the transient analysis for the liquid thorium-molten salt reactor 2 MW TMSR (Wei, 2017).

In order to carry out accurate and efficient transient analysis for the graphite-moderated channel-type molten salt reactors, a

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coupled neutronics and thermal-hydraulics in three-dimensional code for MSR named NTH3D-MSR is developed. There are two main differences between Serpent/NTH3D-MSR and the other dynamics codes. In the first place, different from the previous codes mentioned above which mainly adopted traditional Improved Quasi-Static (IQS) method (Ott and Meneley, 1969), the Predictor-Corrector Improved Quasi-Static (PC-IQS) method (Dulla et al., 2008; Alcaro et al., 2010) is used in the code NTH3D-MSR to factorize the neutron flux and DNP density into a weak time-dependent spatial shape function and a strong time-dependent amplitude function, and the shape function is generated from the predictor neutron flux which is obtained through the implicit difference method. In the PC-IQS method, the iterative calculation of shape function can be avoided, the predicted neutron flux would be corrected according to the amplitude function and then the final distribution of neutron flux would be generated. Therefore the PC-IQS method could obtain higher computation efficiency than the traditional IQS method during solving the neutron kinetics equation. In the second place, different from the famous code DYN3D-MSR which adopted the nodal expand method to solve the diffusion equation, while the Nodal Green's Function Method (NGFM) (Lawrence and Dorning, 1980) is used in the code NTH3D-MSR to obtain high accuracy and efficiency. This work is organized in the following manner: the theory of the neutronics and thermal-hydraulics calculation for the graphite-moderated channel-type molten salt reactors are described in Section 2. The verifications of the coupled neutronics and thermal-hydraulics code NTH3D-MSR are described in Section 3. Finally conclusion remarks are given in Section 4.

2. Theories and numerical model

2.1. "Two-step" calculation strategy

The whole core heterogeneous calculation is time-consuming, therefore the "two-step" calculation strategy (Hu et al., 2017) is adopted in this paper. The first step is the lattice calculation, and the second step is the core calculation. The lattice calculation is implemented by the continuous-energy Monte-Carlo code SERPENT (Leppänen, 2012) which can generate the few-group cross sections for the MSR.

Fig. 1 presents the lattice model of a channel-type molten salt reactor by SERPENT code. Then the few-group cross sections will be used by the NTH3D-MSR to execute whole core neutronics calculations of the channel-type molten salt reactor.

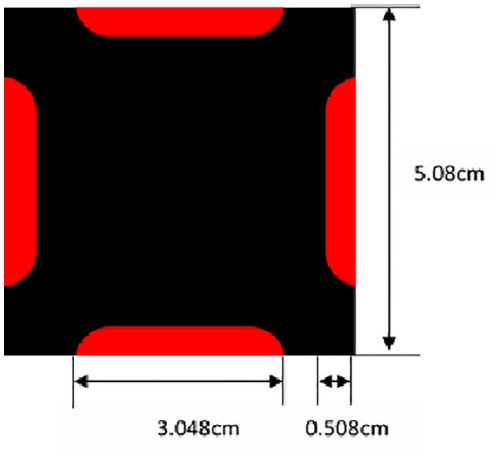

Fig. 1. The typical lattice of the graphite-moderated channel-type molten salt reactor (the black part represents the graphite, the red part indicates the liquid molten salt within the flow channel). (For interpretation of the references to colour in this figure legend, the reader is referred to the web version of this article.)

2.2. Full-core coupled neutronics and thermal-hydraulics calculation

2.2.1. Neutronics calculation

The time-dependent diffusion kinetics equations for MSR can be written by using standard notation:

$$\frac{1}{v_g}\frac{\partial \Phi_g}{\partial t}(r,t) = \nabla D_g(r,t)\nabla\Phi_g(r,t) - \sum_{t,g}(r,t)\Phi_g(r,t) + \sum_{g'=1}^{G}[\chi_{pg}(1-\beta)(\upsilon\sum_f)_{g'}(r,t) + \sum_{g'-g}(r,t)]\Phi_{g'}(r,t) + \sum_{i=1}^{I}\chi_{dgi}\lambda_i C_i(r,t) \quad (1)$$

$$\frac{\partial C_i(r,t)}{\partial t} = \sum_{g'=1}^{G}\beta_i \nu\Sigma_{fg'}(r,t)\Phi_{g'}(r,t) - \lambda_i C_i(r,t) - \frac{\partial[\mathbf{u}(r,t)C_i(r,t)]}{\partial r} \quad i = 1,2...,I \quad (2)$$

$$g = 1,2,...,G; \quad i = 1,2,...I$$

where v_g is the average velocity for the group g, unit: cm·s^{-1}; χ_{pg} is the fraction of the prompt neutrons enter to group g; χ_{dgi} is the probability of the delayed neutrons of family i which appear in the group g; λ_i is the decay constant of delayed neutron precursors of family i, unit: s^{-1}; β_i is the delayed neutron fraction of the family i; Φ is the scalar flux; C is the Delayed Neutron Precursors (DNP) density, $\boldsymbol{u}$ is the vector of the fuel velocity.

According to the factorization method, the neutron flux and the DNP density can be decomposed to an amplitude function and a shape function:

$$\Phi_g(\mathbf{r},t) = n(t)\cdot\phi_g(\mathbf{r},t)$$
$$C_i(\mathbf{r},t) = T_i(t)\cdot\varphi_i(\mathbf{r},t) \quad (3)$$

where the $n(t)$ and $T_i(t)$ is the amplitude function, and the $\phi_g(\mathbf{r},t)$ and $\varphi_i(\mathbf{r},t)$ is the shape function.

If introducing the normalization condition:

$$\sum_{g=1}^{G}\frac{1}{v_g(t)}\int_V \Phi_g^*(\mathbf{r})\phi_g(\mathbf{r},t)d\mathbf{r} = 1 \quad (4)$$

$$\sum_{g=1}^{G}\int_V \chi_{dgi}(\mathbf{r},t)\Phi_g^*(\mathbf{r})\varphi_i(\mathbf{r},t)d\mathbf{r} = 1 \quad (5)$$

Then the amplitude function equations can be written as:

$$\frac{dn(t)}{dt} = \frac{\rho(t)-\bar{\beta}(t)}{\Lambda(t)}n(t) + \sum_i \bar{\lambda}_i T_i(t) \quad (6)$$

$$\frac{dT_i(t)}{dt} = \frac{\bar{\beta}(t)}{\Lambda(t)}n(t) - \bar{\xi}_i T_i(t) \quad (7)$$

In addition, the implicit backward difference method was used to deal with the time variable of the kinetic Eqs. (1) and (2):

$$\nabla D_g(r,t_{n+1})\nabla\Phi_g(r,t_{n+1}) - \sum_{t,g}(r,t_{n+1})\Phi_g(r,t_{n+1}) + \sum_{g'=1}^{G}[\chi_g(1-\beta)(\upsilon\sum_f)_{g'}(r,t_{n+1}) + \sum_{g'-g}(r,t_{n+1})]\Phi_{g'}(r,t_{n+1}) + \sum_{i=1}^{I}\frac{\lambda_i\chi_{ig'}\beta_i\Delta t_n}{1+\lambda_i\Delta t_n}\sum_{g'=1}^{G}(\upsilon\sum_f)_{g'}(r,t_{n+1})\Phi_{g'}(r,t_{n+1}) - \frac{1}{v_g\Delta t_n}\Phi_g(r,t_{n+1}) = -\frac{1}{v_g\Delta t_l}\Phi_g(r,t_l) - \sum_{i=1}^{I}\frac{\lambda_i\chi_{ig}C_i(r,t_l)}{1+\lambda_i\Delta t_l} \quad (8)$$