Steady state investigation on neutronics of a molten salt reactor considering the flow effect of fuel salt *

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Abstract The Molten Salt Reactor (MSR), one of the 'Generation IV' concepts, is a liquid-fuel reactor, which is different from the conventional reactors using solid fissile materials due to the flow effect of fuel salt. The study on its neutronics considering the fuel salt flow, which is the base of the thermal-hydraulic calculation and safety analysis, must be done. In this paper, the theoretical model on neutronics under steady condition for a single-liquid-fueled MSR is conducted and calculated by numerical method. The neutronics model consists of two group neutron diffusion equations for fast and thermal neutron fluxes, and balance equations for six-group delayed neutron precursors considering the flow effect of fuel salt. The spatial discretization of the above models is based on the finite volume method, and the discretization equations are computed by the source iteration method. The distributions of neutron fluxes and the distributions of the delayed neutron precursors in the core are obtained. The numerical calculated results show that, the fuel salt flow has little effect on the distribution of fast and thermal neutron fluxes and the effective multiplication factor; however, it affects the distribution of the delayed neutron precursors significantly, especially the long-lived one. In addition, it could be found that the delayed neutron precursors influence the neutronics slightly under the steady condition.

Key words MSR, steady state, neutronics, flow effect, delayed neutron precursors

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1 Introduction

The Molten Salt Reactor (MSR) is an old concept, which was first proposed by Bettis and Briant of Oak Ridge National Laboratory in the late 1940s to develop a nuclear engine for a military jet aircraft. In 1954, the 2.5 MW Aircraft Reactor Experiment (ARE) was carried out successfully, and then the Molten Salt Reactor Experiment (MSRE) followed at a power level of 8 MW for 13000 equivalent full-power hours from 1956 to 1968[1]. These two prototype reactors established the basic technologies for the MSR, and offered three main advantages: 1) excellent neutron economy; 2) inherent safety features; and 3) continuous or in-batch reprocessing.

These three advantages make the MSR attractive also for the Generation IV International Forum (GIF), and have drawn attention of many researchers again. In the European Union, the reduction of long-lived wastes and transmutation of the minor actinides (MAs) are being experimented under the project of the molten salt reactor technology (MOST)[2]. In Russia, the molten salt advanced reactor transmuter (MOSART) has been developed to burn Pu and MAs[3, 4]. In addition, the SIMMER code[5, 6], which was originally developed for fast reactor safety analysis by JNC-FZK-CEA, is being extended for neutronics and thermo-hydraulics analysis of the MSR. However, the molten fuel salt at high temperature of MSR is not only coolant, but also nuclear heat source, which is very different from the traditional reactors with solid fuels. Therefore, there are few reactor design theories and safety analysis methods which could be referenced for the MSR.

The fuel salt flow makes the MSR very different from the conventional reactors using solid fissile materials, and makes the neutronics and thermal-hydraulic coupled strongly, which plays the important role in the research of reactor safety analysis. The fuel salt flow decreases the number of delayed

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neutron precursors in the reactor core, which affect the distribution of the neutron fluxes. Therefore, it's necessary to consider the flow effect of fuel salt when analyzing the reactor neutronic characteristics. In the present paper, a theoretical investigation on steady neutronics for a designed MSR is conducted, and calculated by numerical method. The multi-group diffusion theory is used to develop the neutron diffusion equations and balance equations of the delayed neutron precursors considering the flow effect of fuel salt. The assembly calculation code DRAGON is adopted to compute the group constants for the neutron diffusion equations.

2 System description

The schematic diagram of the MSR system is shown in Fig. 1. The ternary system of the LiF-NaF-BeF_2 functioning as the reactor fuel solvent, coolant and also moderator has fissile, fertile, and fission products in the primary loop which works around 873.15 K at the inlet and goes out of the core at 1073.15 K. The high temperature fuel salt transfers nuclear heat to the secondary salt $NaBF_4$-NaF of the primary heat exchanger. Then the secondary salt transfers the heat to helium for the electricity generation or hydrogen production.

In this paper, the fractions of the components LiF, NaF and BeF_2 in the reactor fuel solvent are proposed as 15%, 58% and 27% respectively, and 1% mol UF_4 dissolves in it. The reactor core is simplified as a cavity with graphite reflector surrounding it. The radius of the cavity is 1.7 m, and the height of which is 3.8 m. The thickness of the surrounding graphite reflector is 0.5 m. This analysis keeps the thermal output of the Molten Salt Reactor 690 MW constant.

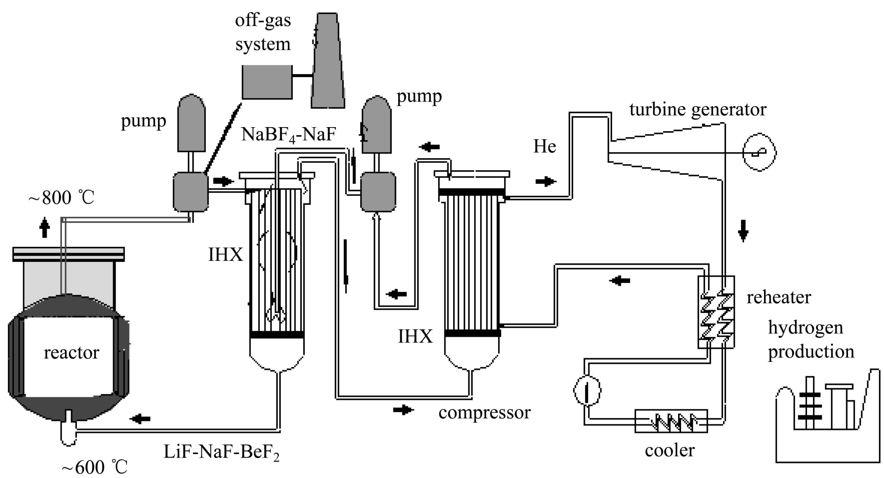

Fig. 1. Schematic diagram of the MSR.

3 Theoretical model

3.1 Steady state model of neutronics

In the present study, the multi-group diffusion theory is adopted to found neutronics model under steady condition, which consists of neutron diffusion equations for neutron fluxes and balance equations for the delayed neutron precursors. These equations are derived from the basic conservation of the number of particles in a control volume. The MSR is a liquid-fuel reactor, which is very different from the conventional solid reactors due to the effect of the fuel salt. Therefore, the flow effect must be considered, especially for the delayed neutron precursors.

The neutron fluxes are classified into two groups, the fast and the thermal neutron fluxes, by a threshold neutron kinetic energy 4.0 eV. The neutron fission spectrums of prompt neutron and delayed neutron are assumed as $\chi_{\mathrm{p},1}=1$, $\chi_{\mathrm{p},2}=0$, $\chi_{\mathrm{d},1,i}=1$, and $\chi_{\mathrm{d},2,i}=0$. The two-group neutron diffusion equations for the fast and the thermal neutron fluxes are derived as follows,

$$-\frac{\partial}{\partial z}\left(D_1\frac{\partial \Phi_1}{\partial z}\right)-\frac{1}{r}\frac{\partial}{\partial r}\left(D_1 r\frac{\partial \Phi_1}{\partial r}\right)+\Sigma_{r,1}\cdot\Phi_1= \frac{1}{k_{\mathrm{eff}}}\sum_{g=1}^{2}(1-\beta_{\mathrm{t}})\cdot(\nu\Sigma_{\mathrm{f}})_g\cdot\Phi_g \tag{1}$$

$$+\sum_{i=1}^{I}\lambda_i\cdot C_i-\frac{1}{v}\left[\frac{\partial(\Phi_1\cdot u)}{\partial z}+\frac{1}{r}\frac{\partial(\Phi_1\cdot w)}{\partial r}\right]-$$

$$\frac{\partial}{\partial z}\left(D_2\frac{\partial \Phi_2}{\partial z}\right)-\frac{1}{r}\frac{\partial}{\partial r}\left(D_2 r\frac{\partial \Phi_2}{\partial r}\right)+\Sigma_{r,2}\cdot\Phi_2= \Sigma_{\mathrm{s},1\text{—}2}\cdot\Phi_1-\frac{1}{v}\left[\frac{\partial(\Phi_2\cdot u)}{\partial z}+\frac{1}{r}\frac{\partial(\Phi_2\cdot w)}{\partial r}\right]. \tag{2}$$

The delayed neutron precursors are classified into six groups by half-life periods. The balance equation of the i-th delayed neutron precursor density C_i (i=1—6) is derived by the same method as neutron fluxes.

$$\left[\frac{\partial(C_i\cdot u)}{\partial z}+\frac{1}{r}\frac{\partial(C_i\cdot w)}{\partial r}\right]=\frac{1}{k_{\text{eff}}}\sum_{g=1}^{2}\beta_i\cdot(\nu\Sigma_{\text{f}})_g\cdot\Phi_g-\lambda_i C_i\ . \tag{3}$$

The effective neutron multiplication factor k_{eff} is defined as

$$k_{\text{eff}}^{(n)}=\frac{\iiint\left[(\nu\Sigma_{\text{f}})_1\cdot\Phi_1^{(n)}+(\nu\Sigma_{\text{f}})_2\cdot\Phi_2^{(n)}\right]\text{d}V}{1/k_{\text{eff}}^{(n-1)}\iiint\left[(\nu\Sigma_{\text{f}})_1\cdot\Phi_1^{(n-1)}+(\nu\Sigma_{\text{f}})_2\cdot\Phi_2^{(n-1)}\right]\text{d}V}, \tag{4}$$

where, Φ represents the neutron flux, and D, Σ_r, ν, Σ_{f}, $\Sigma_{\text{s},1—2}$ are the group constants of neutron diffusion equations presenting diffusion coefficient, removal cross section, average number of neutrons emitted per fission, fission cross section and scatter cross section from group 1 to group 2. u and w are the axial and the radial velocities of the fuel salt, which v is the neutron velocity, z and r represent the axial and radial direction. λ_i denotes the decay constant of i-th delayed neutron precursors. β_i represents the fraction of i-th delayed neutron precursor, and $\beta_t=\sum_{i=1}^{L}\beta_i$. Detail data of β_i and λ_i are listed in Table 1.

Table 1. The delayed neutron fraction and the precursor decay constant.

group	λ_i	$\beta_i(\times10^{-5})$
1	0.0124	22.3
2	0.0305	145.7
3	0.111	130.7
4	0.301	262.8
5	1.14	76.6
6	3.01	28.0

The subscripts 1 and 2 denote the fast neutron flux and the thermal neutron flux respectively, and 1—2 means from the fast neutron flux to the thermal neutron flux. The superscript n in Eq. (4) denotes the n-th source iteration time in solving the diffusion equations.

The flow effect of fuel salt is reflected by the convection items in the model, which are the last items in Eq. (1) and Eq. (2) and the first item in Eq. (3). Because the velocity value of fuel salt is far less than that of the neutron (i.e. $u\ll v$ and $w\ll v$), the convection items in Eq. (1) and Eq. (2) can be neglected. However, the flow effect of fuel salt must be considered for the delayed neutron precursors in Eq. (3). In this paper, the velocity in the main flow direction is given as 20 cm/s, and the radial velocity is assumed as 0. Therefore, the steady state model of neutronics could be simplified as follows,

$$-\frac{\partial}{\partial z}\left(D_1\frac{\partial\Phi_1}{\partial z}\right)-\frac{1}{r}\frac{\partial}{\partial r}\left(D_1 r\frac{\partial\Phi_1}{\partial r}\right)+\Sigma_{r,1}\cdot\Phi_1=$$
$$\frac{1}{k_{\text{eff}}}\sum_{g=1}^{2}(1-\beta_t)\cdot(\nu\Sigma_{\text{f}})_g\cdot\Phi_g+\sum_{i=1}^{I}\lambda_i\cdot C_i\ , \tag{5}$$

$$-\frac{\partial}{\partial z}\left(D_2\frac{\partial\Phi_2}{\partial z}\right)-\frac{1}{r}\frac{\partial}{\partial r}\left(D_2 r\frac{\partial\Phi_2}{\partial r}\right)+$$
$$\Sigma_{r,2}\cdot\Phi_2=\Sigma_{\text{s},1—2}\cdot\Phi_1, \tag{6}$$

$$\frac{\partial(C_i\cdot u)}{\partial z}=\frac{1}{k_{\text{eff}}}\sum_{g=1}^{2}\beta_i\cdot(\nu\Sigma_{\text{f}})_g\cdot\Phi_g-\lambda_i\cdot C_i\ . \tag{7}$$

The boundary conditions for the fast and the thermal fluxes are set zero on the outside surfaces of the graphite reflector, and the symmetric boundary at the center line. As for the delayed neutron precursors, the fuel salt outflows from the reactor core, passes through the external loop, then re-inflows the core bottom. If defining the circulate time of fuel salt in the external loop as τ_{loop}, the re-inflow delayed neutron precursors $C_{i,\text{in}}$ are deduced as

$$C_{i,\text{in}}=C_{i,\text{out}}\cdot\exp(-\lambda_i\cdot\tau_{\text{loop}}). \tag{8}$$

3.2 Calculation of group constants

The core of the designed MSR is a cavity surrounding with graphite reflector, where the fuel salt flows through. Therefore, the core is approximately a fuel element of large size. The assembly calculation code DRAGON[7] is adopted to calculate the group constants in Eq. (1) to Eq. (4) for the Molten Salt Reactor. In order to improve the calculation accuracy, the multi-group library iaea172, published by IAEA in 2006, which is a 172-group library generated from ENDF/B-6, is used in this study.

3.3 Numerical method

The neutron fluxes equations and balance equations of delayed neutron precursors are discretized by the control volume method, which could ensure the conservation of the special discretization equations. The diffusion items are discretized by the central difference method, and the power-law scheme for convection items. And the source iteration method[8] is adopted to solve these discretization equations.

4 Results and discussion

4.1 Verification of the numerical model

The numerical model for neutronics which is developed for the MSR considering the flow effect of the fuel salt, also could be used for that without flow by assuming the velocity is zero. In this paper, the

IAEA five-region benchmark and TWIGL benchmark are adopted to verify the numerical model. The calculated results are listed in Table 2, which verify the validity of the model for neutronics presented in this study.

Table 2. Verification results of benchmarks.

k_{eff}	benchmark value	calculated value	relative error
IAEA	1.592740	1.592296	0.028%
TWIGL	0.913214	0.913175	0.00427%

4.2 Neutron fluxes distributions

Figure 2 shows the flow effect of fuel salt on the distributions of fast and thermal neutron fluxes, in which the solid lines indicate the results considering the fuel salt flow ($u = 20$ cm/s), while the dashed lines represent the results that disregard the fuel salt flow. Fig. 2(a) shows the fast and the thermal fluxes along axial direction at the central axis, in which very few differences are found between the results under these two different conditions. Fig. 2(b) illustrates the neutron fluxes along the radial direction at $z = 2.4$ m, from which it can be found that the fast and the thermal neutron fluxes calculated under the condition of axial velocity equal to 20 cm/s are nearly the same with that calculated when the fuel salt does not flow.

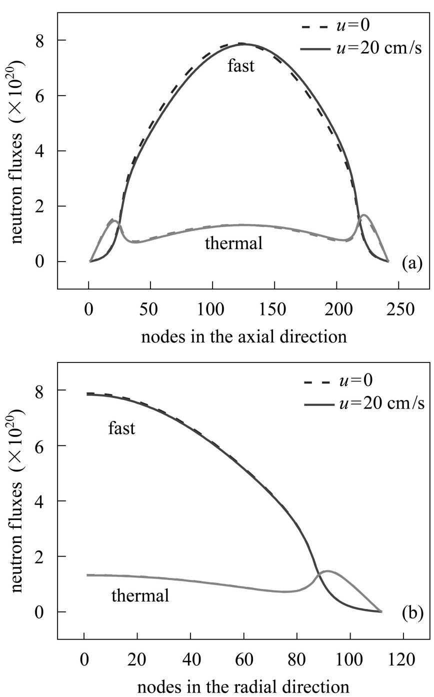

Fig. 2. Flow effect of fuel salt on the neutron fluxes distributions.

The flow effect of fuel salt on the effective multiplication factor k_{eff} is presented in Table 3. This table shows that k_{eff} decreases slowly with the velocity increasing. This result suggests that the fuel salt flow has little influence on the effective multiplication factor. That's because the fuel salt flow affect the fast and the thermal neutron fluxes very slightly.

Table 3. Flow effect on the effective multiplication factor.

u/(cm/s)	k_{eff}	u/(cm/s)	k_{eff}
0.0	1.30462	80.0	1.29932
20.0	1.30184	100.0	1.29894
50.0	1.30017		

The effect of delayed neutron precursors on k_{eff} and the distribution of fast and thermal neutron fluxes is shown in Fig. 3. The results considering the delayed neutron precursors are represented by the solid lines, while those without considering the delayed neutron precursors are represented by the dashed lines. Fig. 3(a) displays the k_{eff} varying with the number of iteration under two different conditions, which shows that k_{eff} is coincident with each other. From Fig. 3(b) and (c), it can be easily found that the fast and the thermal neutron fluxes along the axial and the radial direction in these two different cases are similar respectively.

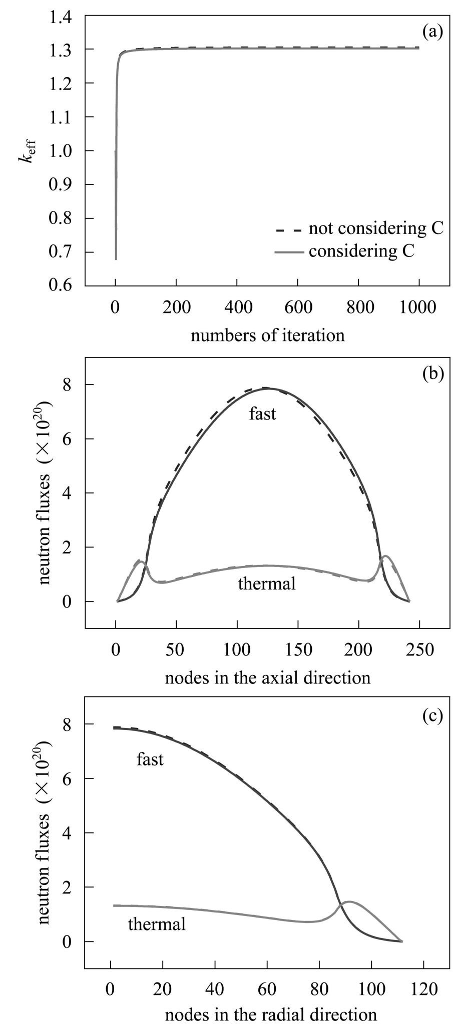

Fig. 3. Effect of the delayed neutron precursors.

4.3 Precursors distributions

The distributions of delayed neutron precursors C_i (i=1—6) considering the fuel salt flow are shown

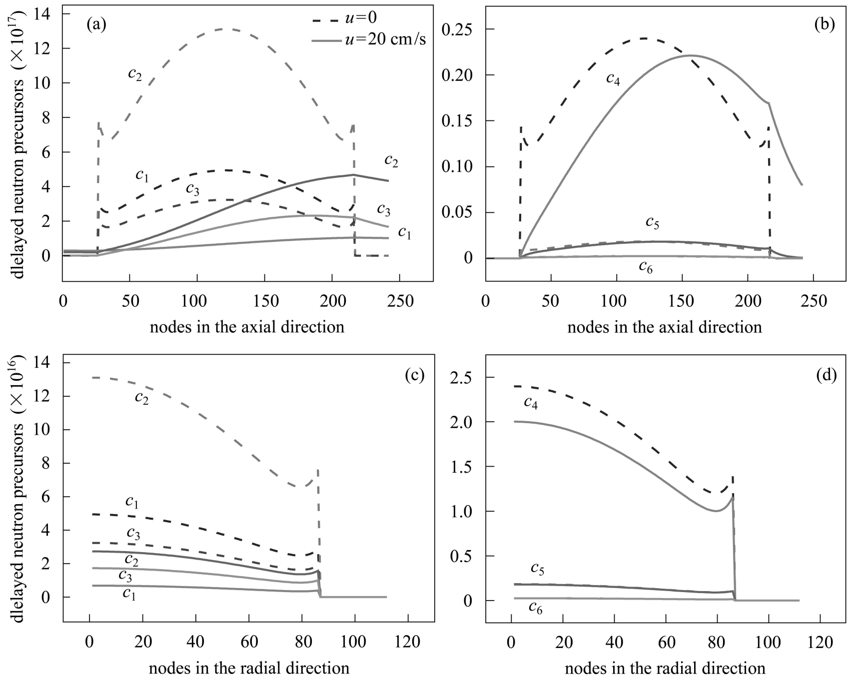

Fig. 4. Flow effect of fuel salt on the delayed neutron precursors distributions.

by the solid lines in Fig. 4. In order to study the flow effect on the delayed neutron precursors, the results that disregard the fuel salt flow are also displayed by the dashed lines in the figures. Figs. 4(a) and (b) show the delayed neutron precursors along the axial direction. From these two figures, it could be found that the flow influences the delayed neutron precursor No.C_1—C_4 greatly, and the larger the decay constants, the greater the influence of the fuel salt flow. The delayed neutron precursors along the radial direction are shown in Fig. 4(c) and (d), in which the same conclusion could be found. Comparing the results of the same group under different conditions, it also can be found that the effect of fuel salt flow on the delayed neutron precursors is greater with larger decay constant.

5 Conclusions

In this study, the theoretical model of neutronics for the Molten Salt Reactor is developed and calculated by the numerical method. The model adopts two-group diffusion equations and six-group delayed neutron precursor equations which consider the flow effect of the fuel salt. The main results obtained in this research are as follows,

(1) The fuel salt flow has little effect on the distribution of fast and thermal neutron fluxes and effective multiplication factor;

(2) The fuel salt flow affects the distribution of the delayed neutron precursors significantly, and the smaller the delay constant, the greater the influence of the flow;

(3) The delayed neutron precursors influence the neutronics slightly under the steady condition.

References

1 Rosenthal M W, Kasten P R, Briggs R B. Nuclear Application & Technology, 1970, **8**(2): 107

2 Dulla S, Ravetto P, Rostagno M M. Neutron Kinetics of Fluid-Fuel Systems by the Quasi-Static Method. Nuclear Energy, 2004, **31**: 1709

3 Ignatiev V, Feynberg O, Myasnikov A et al. Mosart Fuels and Container Materials Study: Case for Na, Li, Be/F Solvent System. In: Proceedings of the 2003 ANS/ENS International Winter Meeting. New Orleans: American Nuclear Society, 2003

4 Merzlyakov V, Ignatiev V. Measurement of Transport Properties for Molten Na, Li, Be/F Mixture. In: the 11th International Topical Meeting on Nuclear Reactor Thermal-Hydraulics. Avignon: American Nuclear Society, 2005

5 Wang S, Rineiski A, Maschek W. Nuclear Engineering and Design, 2006, **236**: 1580

6 Kondo, Sa, Tobita Y, Morita Y et al. Current Status and Validation of the SIMMER-III LMFR Safety Analysis Code. In: Proceedings of the ICONE-7. Tokyo: Japan Society of Mechanical Engineering, 1999

7 Marleau G, Hebert A, Roy R. A User's Guide for DRAGON 3.05C. Canada: Ecole Polytechnique de Montreal, 2006. 1—175

8 XIE Z S. Numerical Calculation of Nuclear Reactor Neutronics. Beijing: Atomic Energy Press, 1997, 29—31 (in Chinese)