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Steady thermal hydraulic analysis for a molten salt reactor

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Abstract The Molten Salt Reactor (MSR) can meet the demand of transmutation and breeding. In this study, theoretical calculation of steady thermal hydraulic characteristics of a graphite-moderated channel type MSR is conducted. The DRAGON code is adopted to calculate the axial and radial power factor firstly. The flow and heat transfer model in the fuel salt and graphite are developed on basis of the fundamental mass, momentum and energy equations. The results show the detailed flow distribution in the core, and the temperature profiles of the fuel salt, inner and outer wall in the nine typical elements along the axial flow direction are also obtained.

Key words Molten salt reactor, Thermal hydraulics, Steady characteristics, Numerical simulation

CLC number TL426

1 Introduction

The concept of Molten Salt Reactor (MSR) was first proposed by Bettis and Briant of Oak Ridge National Laboratory in 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 8MW for 13 000 equivalent full-power hours from 1956 to 1968[1]. The two prototype reactors established basic technologies for MSR, which have advantages of excellent neutron economy, inherent safety features and continuous or in-batch reprocessing.

The advantages make the MSR attractive to the Generation IV International Forum (GIF), and have drawn attention of many researchers again. In the European Union, the reduction of long-life 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 in a high temperature MSR is not only coolant, but also nuclear heat source. This 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.

In this study, a theoretical investigation on the steady thermal hydraulic characteristics of a graphite-moderated channel type MSR is conducted. The DRAGON code is adopted to compute the axial and radial power factor firstly. Based on the calculation, the flow distribution in the core, and the temperature profiles of the fuel salt and inner/outer wall in the nine representative elements are simulated by numerical method.

2 System descriptions

A schematic diagram of the MSR is shown in Fig.1[7]. The ternary system of $LiF\text{-}NaF\text{-}BeF_2$,

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functioning as the reactor fuel solvent, coolant and moderator, has fissile and fertile fission products in the primary loop, which works at 873.15 K at the inlet and at 1073.15 K at the core exit. 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 electricity generation or hydrogen production.

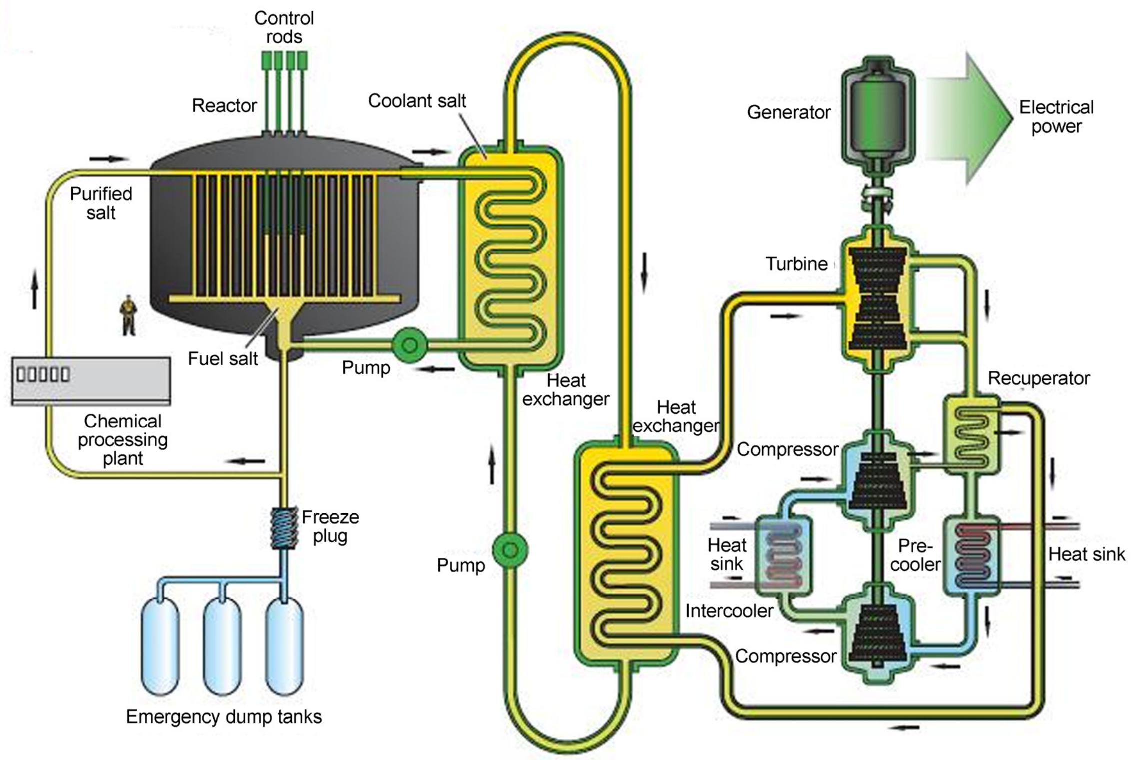

Fig.1 Schematic diagram of the Molten Salt Reactor system.

The designed MSR core consists of 199 one-dimensional parallel coolant channels in a form of hexagonal graphite blocks, each with a central fuel channel (Fig. 2). The active core height is 6.5 m, and the radius is 3.6 m. Fig. 3 shows the schematic diagram of a single graphite element, in which the channel diameter is 0.2 m and the distance between the opposite sides of the hexagonal graphite is 0.4 m.

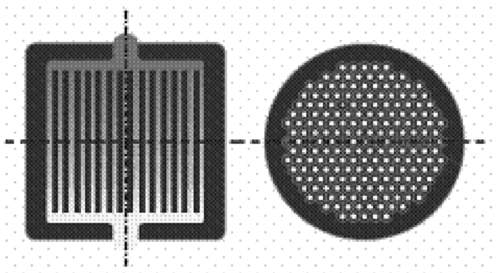

Fig.2 The axial and radial cut of the MSR core represented by 199 hexagonal graphite channels.

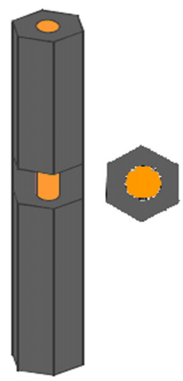

Fig.3 Schematic diagram of a graphite element.

3 Theoretical model

3.1 Calculation of power factor

The 199 graphite elements are divided into 9 groups (Table 1) according to the distance between the element and the core center, and every element is parted to 13 same control volumes in axial direction.

We assume that the characteristics of the graphite elements in the same group are the same, and only one element in a group is calculated, which is defined as element i(i=1~9), while the axial control volume is defined as j(j=1~13). The DARGON[8] code is used to calculate the axial and radial power factor of the MSR. Being originally developed for the assembly calculation of CANDO, it can calculate the neutron flux and group constants for almost any assembly or core with different geometry forms. In order to improve the calculation accuracy, multi-group library IAEA172 is used, which is a 172-group library generated from ENDF/B-6 and suitable for almost all kinds of reactors.

Table 1 The groups of 199 graphite elements

Group No.	Element's number	Distance / m
1	1	0~0.2
2	6	0.2~0.5196
3	12	0.5196~0.866
4	18	0.866~1.3
5	24	1.3~1.7
6	30	1.7~2.0
7	36	2.0~2.4
8	48	2.4~2.8
9	24	2.8~3.6

3.2 Calculation of the fuel salt flow distribution and temperature profiles

A one-dimensional single-phase flow model is developed to simulate the flow and heat transfer of fuel salt in the element channels. The model is based on the fundamental conservation principles, i.e. the mass, momentum and energy conservation equations:

$$\frac{\partial(\rho u)}{\partial z}=0 \tag{1}$$

$$\frac{\partial p}{\partial z}=-\frac{\partial(\rho u^2)}{\partial z}-\rho g-\frac{\partial p_{\mathrm{fric}}}{\partial z} \tag{2}$$

$$\frac{\partial(\rho u c_{\mathrm{p}} T_{\mathrm{f}})}{\partial z}=Q_{\mathrm{f}} \tag{3}$$

where ρ and c_{p} are the fuel salt density and heat capacity; p, u, T and z denote the pressure, velocity, temperature and the flow direction. The subscript f means the fuel salt, fric is the friction pressure drop, and Q_{f} is the fission energy released in the fuel salt.

3.3 Calculation of the inner and outer wall temperature profiles in graphite

The element is a hexagonal graphite block with a central fuel channel. In order to simplify the calculation, the element is assumed to be an equivalent of a column with a central channel, and the inner and outer radius is R_1 and R_2 respectively. The heat transfer in the graphite is described by the following energy equation.

$$(\frac{1}{r})\frac{\mathrm{d}}{\mathrm{d}r}(\lambda r\frac{\mathrm{d}T_{\mathrm{g}}}{\mathrm{d}r})+Q_{\mathrm{g}}=0 \tag{4}$$

And the inner and outer wall boundary conditions are defined respectively as

1） $$\lambda_{\mathrm{g}}\left.\frac{\mathrm{d}T_{\mathrm{g}}(r)}{\mathrm{d}r}\right|_{r=R_1}=\alpha[T_{\mathrm{f}}-T_{\mathrm{g}}(r)|_{r=R_1}]$$

2） $$\left.\frac{\mathrm{d}T_{\mathrm{g}}(r)}{\mathrm{d}r}\right|_{r=R_2}=0$$

where r is the radial direction; the subscript g represents graphite; λ is the thermal conductivity of the graphite; α is the heat transfer coefficient; and Q_{g} is the fission energy released in the graphite, which is usually 10% of the fission energy[9].

3.4 The calculation method and procedure

The theoretical models are applied to every control volumes. Eqs. (1) to (4) are resolved in the divided control volumes, and then the numerical method is used to simulate the steady thermal hydraulic characteristics of the whole core as the program flow diagram shown in Fig.4.

The discretized forms of Eqs. (1) to (3) are derived as shown below.

$$W_i=\mathrm{const} \tag{5}$$

$$p_{i,j}=p_{i,j-1}-\Delta p_{i,j} \tag{6}$$

$$\Delta p_{i,j}=(\Delta p_{\mathrm{f}})_{i,j}+(\Delta p_{\mathrm{a}})_{i,j}+(\Delta p_{\mathrm{el}})_{i,j}+(\Delta p_{\mathrm{c}})_{i,j} \tag{7}$$

$$(T_{\mathrm{f}})_{i,j}=(T_{\mathrm{f}})_{i,j-1}+\frac{(Q_{\mathrm{f}})_{i,j}}{W_i\cdot c_{\mathrm{p}}} \tag{8}$$

Under the given boundary conditions, Eq. (4) can be resolved by analysis method [10] given as

$$T_{\mathrm{g}}(r)=T_{\mathrm{f}}+\frac{Q_{\mathrm{g}}}{2\alpha}(\frac{R_2^2}{R_1}-R_1)+\frac{Q_{\mathrm{g}}}{2\lambda}\left[\frac{R_1^2-r^2}{2}+R_2^2\ln(\frac{r}{R_1})\right] \quad (9)$$

From Eq. (9), the inner and outer wall temperatures in the control volumes of a channel can be described as

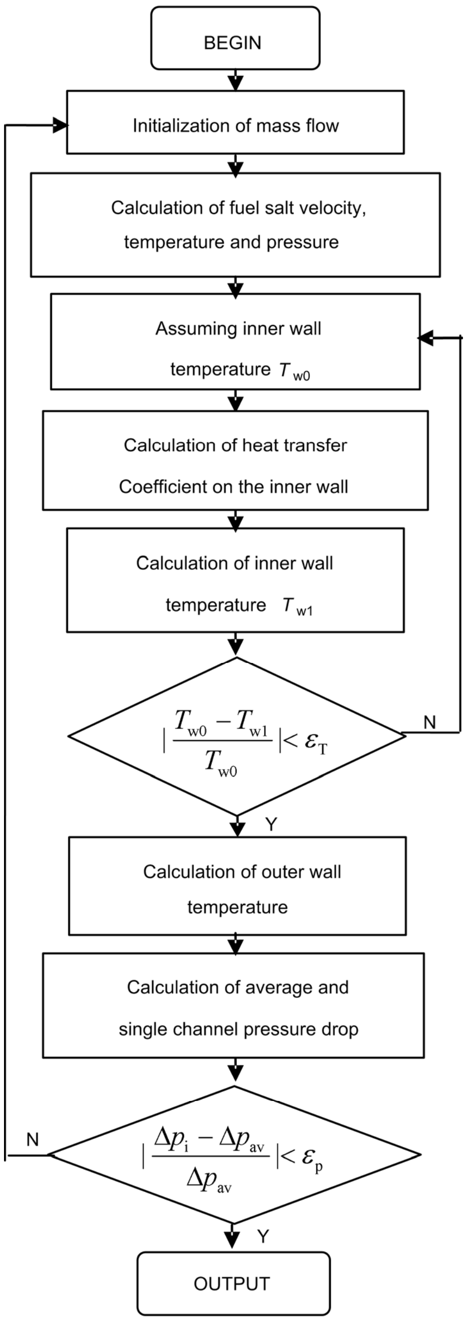

Fig.4 The program flow diagram.

$$(T_{R_1})_{i,j}=T_{\mathrm{g}}(R_1)_{i,j}=(T_{\mathrm{f}})_{i,j}+\frac{(Q_{\mathrm{g}})_{i,j}}{2\alpha_{i,j}}(\frac{R_2^2}{R_1}-R_1) \quad (10)$$

$$(T_{R_2})_{i,j}=T_{\mathrm{g}}(R_2)_{i,j}=(T_{R_1})_{i,j}+\frac{(Q_{\mathrm{g}})_{i,j}}{2\lambda}[\frac{R_1^2-R_2^2}{2}+R_2^2\ln(\frac{R_2}{R_1})] \quad (11)$$

where Δp_{f} is the friction pressure drop; Δp_{a} is the acceleration pressure drop; Δp_{el} is the gravity pressure drop; Δp_{c} is the local pressure drop; T_{f} is the fuel salt temperature; and W represents the mass flow. The subscripts i and j represent the element number and control volume number, respectively.

4 Results and discussion

The working condition of the MSR is shown in Table 2.

Table 2 Working condition of the MSR

Parameter	Value
Power / MW	56.4
Inlet temperature / K	873.15
Inlet pressure / × 10^5Pa	1
Core flow / kg·s^{-1}	10000

The axial and radial power factors are calculated by the DRAGON code (Fig.5 and Fig.6). Fig.5 shows that the axial power factor distributes as a cosine line. Since the core configuration is nearly uniform, the calculation result is similar to the analysis result of uniform cylinder bare core. In Fig.6, it could be found that the radial factor is dependent upon the element position. The more close to core center the element is, the higher the power factor is. However, the element No.8 had the lowest factor, because the most elements were arranged at that position.

Fig.7 shows the mass flow in every typical element. The mass flow in the central element is the largest, and it decreases with increasing distance between the channel and the core center. The mass flow element No.8 is the lowest, which is corresponding to the power factor. Considering both the radial power factor and the element arrangement, it could be found that the mass flow in the channel is related to its position and power load.

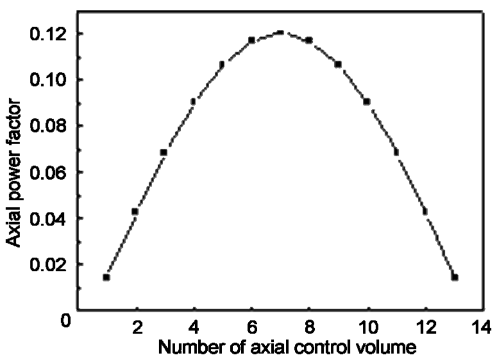

Fig.5 The axial power factor.

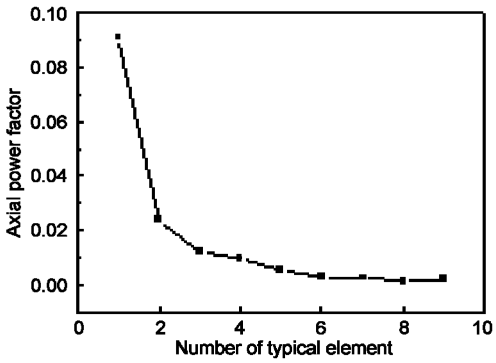

Fig.6 The radial power factor.

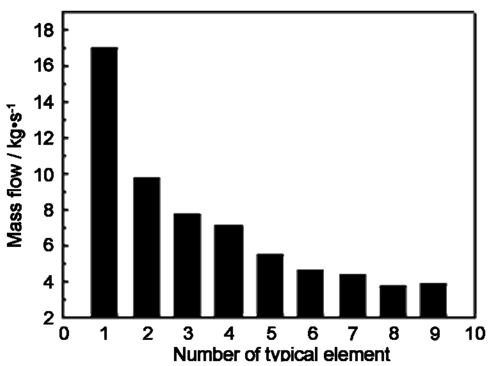

Fig.7 The mass flow in 9 typical elements.

Fig.8 represents the pressure drop along the axial flow direction. Pressure dropped about 0.133×10^5Pa through the core under the calculated working condition.

Figs.9~11 depict the fuel salt, inner and outer wall temperature profiles in 9 typical elements along the axial flow direction. From the figures, it can be seen that the temperatures increase with the axial distance. Element 1 has the maximum temperature increase because of its maximum power load. Element 8 has the minimum temperature increase due to its minimum mass flow.

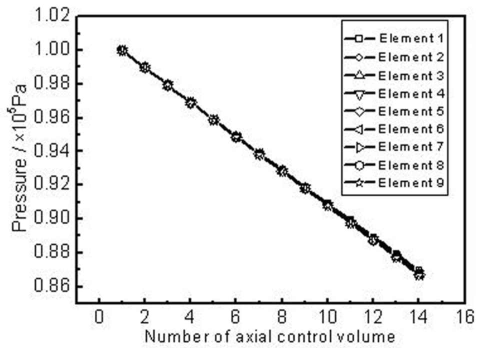

Fig.8 The axial pressure profile in 9 typical channels.

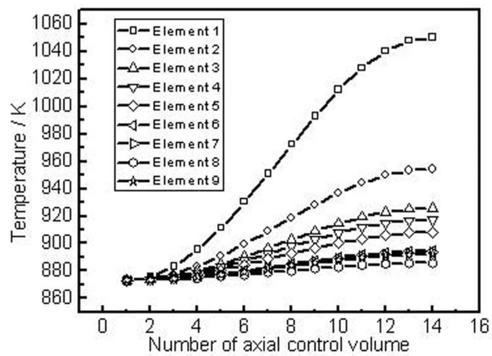

Fig.9 Axial fuel salt temperature profiles.

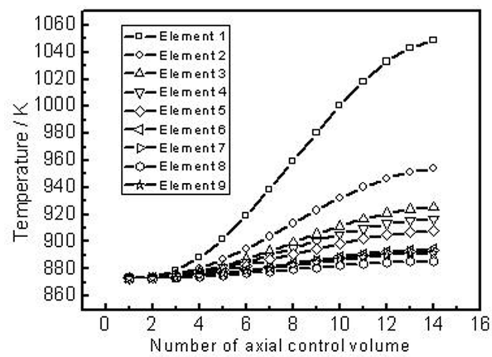

Fig.10 Axial inner wall temperature profiles.

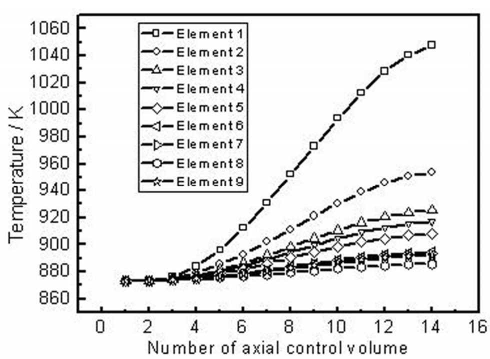

Fig.11 Axial outer wall temperature profiles.

Fig.12 represents the comparison of the fuel salt, inner and outer wall temperature profiles in element No.1. From this figure, it can be found that, the fuel salt has the maximum temperature, while the outer wall has the minimum temperature in the same control volume, because the fission energy releasing into the fuel salt is much more than that releasing into the graphite.

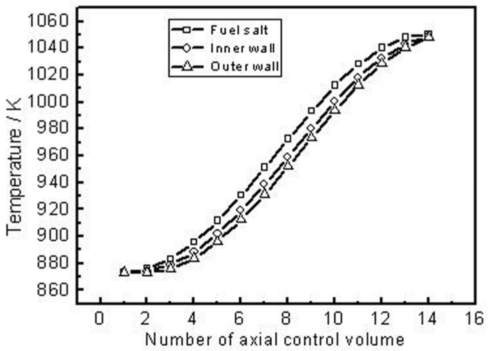

Fig.12 The axial fuel salt, inner and outer wall temperature profiles in element No.1.

5 Conclusions

In this work, theoretical calculation of steady thermal hydraulic characteristics of a graphite-moderated channel type MSR is conducted. The results are summarized as follows:

1) The mass flow in 9 typical elements is related to the channel position and power load.

2) The element position greatly affects the temperature increase of fuel salt, inner and outer wall.

3) In the same element, the temperature behavior of fuel salt, inner and outer wall along axial flow direction is similar, although the fuel salt has the maximum temperature, and the outer wall has the minimum temperature in the same control volume.

References

1 Rosenthal M W, Kasten P R, Briggs R B. Nuclear Application & Technology, 1970, **8**:107-117.

2 Dulla S, Ravetto P, Rostagno M M. Annals of Nuclear Energy, 2004, **31**:1709-1733.

3 Ignatiev V. MOSART fuels and container materials study: case for Na, Li, Be/F solvent system (2003b), Proc. 2003 ANS/ENS International Winter Meeting (GLOBAL 2003), New Orleans, USA.

4 Merzlyakov V, Ignatiev V. Measurement of transport properties for molten Na,Li,Be/F mixture, the 11th International Topical Meeting on Nuclear Reactor Thermal-Hydraulics, 2005, Avignon, France.

5 Wang S. Molten salt related developments for SIMMER, the 12th International SIMMER Review Meeting, 2004, Grenoble, France.

6 Kondo S. Current status and validation of the SIMMER-III LMFR safety analysis code, Proc. ICONE-7, 1999, Tokyo, Japan.

7 Generation IV International Forum. A Technology Roadmap for Generation IV Nuclear Energy System, GIF-002-00: 33-37, 2002.

8 Marleau G, Hebert A, Roy R. A user guide for DRAGON 3.05C, Institute de Genie Nuclearire, Department de Genie Physique, Ecole Polytechnique de Montreal, 2006.

9 Jiri K, Ulrich G, Ulrich R. DYN1D-MSR dynamics code for molten salt reactors, Annals of Nuclear Energy, 2005, **32**:1799-1824.

10 Huang X Q, Gong D H, Li H. Journal of Guizhou University of Technology (Natural Science Edition), 2001, **30**:9-12.