Fundamental Research on Molten Salt Reactors

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Abstract — **The new concept Molten Salt Reactor (MSR) is the only liquid-fuels reactor in the six candidates of the generation IV advanced nuclear reactor. MSR is presently revisited all around the world because of its remarkable advantages in safety, economics, continuable development of the fissile resources, and proliferation resistance of nuclear. However, the flow effect of the fuel salt in MSR makes it very different from the conventional reactors using solid fuels, and induces many new challenges from the perspective of both theoretical models and solution methods. This research focused on four fundamental issues of MSR research by founding theoretical models and developing numerical codes, after intensive investigation on research fronties of MSRs. First, the state-equation method was firstly introduced to study the static thermalphysic properties of the ternary molten salt system used in MSR. The densities, enthalpy, entropy and specific heat were obtained. Second, the general neutron diffusion equations for the circulating fuels were established, in which the group constants and other nuclear parameters were prepared by the DRAGON code. The neutron physics analysis code was designed to solve the founded models, and the influences of the main parameters on the reactor physics were analyzed. Third, the flow and heat transfer models were founded and solved by developing the thermal hydraulics analysis code. The characteristics of the flow and heat transfer in MSR were obtained. In addition, the two codes above were coupled to research the reaction between the neutron physics and thermal hydraulics in MSR. Last, a safety analysis code was programed based on the proper theoretical models. Three types including six transient accidents of MSR were analyze. This project serves some useful information and method for the MSR design and research.**

Keywords — **molten salt reactor, thermalphysical properties, neutron physics, thermal hydraulics, safety analysis**

I. INTRODUCTION

The MSR concept was first proposed by Oak Ridge National Laboratory (ORNL) and has been researched and developed extensively in their projects of the aircraft reactor experiment (ARE) and the molten salt reactor experiment (MSRE) from 1940s to 1960s[1,2]. These two projects established the basic technologies for MSR, and demonstrated the main advances of molten salt reactor, including good neutron economy, inherent safety and on-line refueling, processing and fission product removal.

These advantages of MSR, changing goals for advanced reactors and new developed technologies make MSR attractive also for the present Generation IV International Forum (GIF), and being one of the six candidates for the Generation IV Reactor. Therefore, worldwide research activities are being conducted to study and develop new concept molten salt reactors for different attractive features of current renewed relevance. Some new concepts of the molten salt reactor have been proposed, such as the small molten salt reactor (SMSR)[3], the actinides molten salt transmuter (AMSTER)[4], the molten salt actinide recycler and transmuter system (MOSART)[5] and thorium molten salt reactor (TMSR)[6].

In order to provide more basic understanding and information for MSRs, in the present study, the thermalphysic properties of the molten salt used in MSR, the neutron physics, thermal hydraulics and safety analysis of the MSR were researched by founding theoretical models and designing numerical codes.

II. ACTUAL WORK

A. Evaluation of Static Thermophysical Properties of Molten Salt Systems

Molten fluoride salts are suggested as the fuel solvent in MSRs mainly owing to their good neutron properties (low neutron cross sections, radiation stability, negative temperature coefficient), thermal and transport properties (low melting point, thermal stability, low vapor pressure, adequate heat transfer and viscosity), and chemical properties (high solubility of fuel components, compatibility with container and moderator materials, ease of fuel processing). The thermalphysical properties of the molten salt system are the base of the studies of the neutron physics, thermal hydraulics and safety analysis for MSRs. In this research, the state-equation of a molten salt system was described by modifying the Peng-Robinson equation as shown in Eq. (1) (in which *d, b* are constants). The densities of the molten salt system and its components are evaluated by this equation directly. Based on the equation of state, the other static thermalphysical properties such as the enthalpy, entropy and heat capacity are estimated by the residual function method and the fugacity coefficient method respectively.

$$p = \frac{RT}{V-b} - \frac{d}{V(V+b)+b(V-b)} \tag{1}$$

B. Neutron Physics Analysis

MSRs are characterized by using fuel salts in liquid phase rather than solid fuels as in the current generation of nuclear reactors. The liquid-fuels in MSRs circulate in the reactor primary loop with drifting the delayed neutron precursors out of the core, which makes the neutron physics of MSRs are different from that of the solid-fuels reactors. The multi-group diffusion theory is adopted to deduce the neutron physic mode of the MSRs, which consists of neutron diffusion equations for neutron fluxes and balance equations for delayed neutron precursors. The founded model takes the following form

$$\frac{1}{v_g}\cdot\frac{\partial\phi_g}{\partial t}+\frac{1}{v_g}\nabla(U\phi_g)=\nabla\cdot D_g\nabla\phi_g+\sum_{g'=1}^{g-1}\phi_{g'}\cdot\Sigma_{g'\to g}-\phi_g\cdot\Sigma_{r,g}$$
$$+\chi_{p,g}\cdot(1-\sum_{i=1}^{I}\beta_i)\cdot\sum_{g=1}^{G}(\nu\Sigma_f)_g\cdot\phi_g+\sum_{i=1}^{I}\chi_{d,g,i}\cdot\lambda_i\cdot C_i \tag{2}$$

$$\frac{\partial C_i}{\partial t}+\nabla(UC_i)=\beta_i\cdot\sum_{g=1}^{G}\cdot(\nu\Sigma_f)_g\cdot\phi_g-\lambda_i\cdot C_i \tag{3}$$

In the present research, the group constants in the above control equations were calculated by the DRAGON code.

Because the delayed neutron precursors can move out the core with the circulating fuel salt, the effective delayed neutron fraction in MSRs must be paid more attention on. In this research, the value of the effective delayed neutron fraction for the precursor family *i* was deduced by generalizing the diffusion equation into point kinetic equation and weighted by the neutron importance. The equation of neutron importance and the effective delayed neutron fractions are as follows.

$$\nabla\cdot D_g\nabla\phi_g^* + \sum_{n=1,n\neq g}^{G}\Sigma_{g\to n}\phi_n^*$$
$$-\Sigma_{r,g}\phi_g^*+(1-\beta)(\nu\Sigma_f)_g\sum_{n=1}^{G}\chi_{p,n}\phi_n^*=0 \tag{4}$$

$$\tilde{\beta}_i=\frac{(\phi^*,\beta_i\chi_{di}(E)F\phi)}{(\phi^*,\chi(E)F\phi)} \tag{5}$$

C. Thermal Hydraulic Analysis

Briant and Weinberg [7] researched the feasibility of the molten fluorides as power reactor fuels, and they conclude that there was no decisive difference between water and molten fluorides from the flow and heat transfer viewpoint. Therefore, the flow and heat transfer models of the fuel salt in the MSR core can be generally founded based on the fundamental laws of the mass, momentum and energy conservation equations. In the present research, the standard $k-\varepsilon$ model [8] was adopted for the turbulent flow.

D. Safety Analysis

Point kinetic model is always used for engineering safety and stability analyses in the solid-fuel reactors. However, the conventional point kinetic model for the solid-fuel reactors is not suitable for the MSRs because of the fuel salt circulation. In this research, a heuristic point kinetic model is established for the MSRs, in which the delayed neutron precursors are split into two parts – ones in the core and the others in the external loop of the core. The control equations are as follows.

$$\frac{dn(t)}{dt}=\frac{(\rho(t)-\tilde{\beta}_{eff})}{\Lambda}n(t)+\sum_{i=1}^{6}\lambda_i c_{c,i} \tag{6}$$

$$\frac{dc_{c,i}}{dt}=\frac{\beta_i}{\Lambda}n(t)-\lambda_i c_{c,i}+c_{l,i}\frac{1}{\tau_l}(\frac{V_l}{V_c})-c_{c,i}\frac{1}{\tau_c} \tag{7}$$

$$\frac{dc_{l,i}}{dt}=-\lambda_i c_{l,i}+c_{c,i}\frac{1}{\tau_c}(\frac{V_c}{V_l})-c_{l,i}\frac{1}{\tau_l} \tag{8}$$

III. Representative Results

The founded models were calculated by developing a modular code, which was applied to the MOSART to verify the validation of the models and the code. Some representative results were shown as following.

Figure 1 displays the density of the ternary molten salt system 15%LiF-58%NaF-27%BeF_2 used in MOSART primary loop, from which it can be found that the calculated results are agree with the experimental data. It could be concluded that the modified Peng-Robinson equation can be applied to evaluate the density of the molten salt systems, and it is recommended that it be used as the basis to estimate the enthalpy, entropy and heat capacity of the molten salt systems.

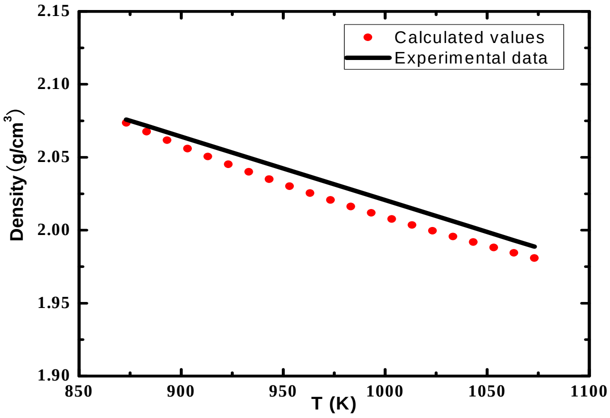


Fig.1 Density of the ternary system 15%LiF-58%NaF-17%BeF_2

The effective multiplication factors calculated by different institutes are listed in Table 1, from which it can be seen that the calculated result is consistent with those calculated by other institutes.

Table 1 Effective multiplication factor

Institute	Codes	k_{eff}
BME	MCNP4C + JEFF3.1	1.00905

FZK	2D560gr. + JEFF3.0	0.99285
NRG	MCNP4C + JEFF3.1	1.00887
Polito	2D4 gr. + JEFF3.1	0.99595
RRC-KI	MCNP4B+ENDF5,6	0.99791
SCK-CEN	MCNPX250	1.00904
XJTU	NPAC-XJTU	0.99994

The Doppler coefficient calculated from k_{eff} at 900K and 1500K of the fuel salt temperatures is *-1.607667 pcm/K*, which is in reasonable agreement with the value *-1.53 pcm/K* of FZK. Fig. 2 shows the k_{eff} changing with the fuel salt temperature, which illustrates MOSART is a fast-spectrum reactor.

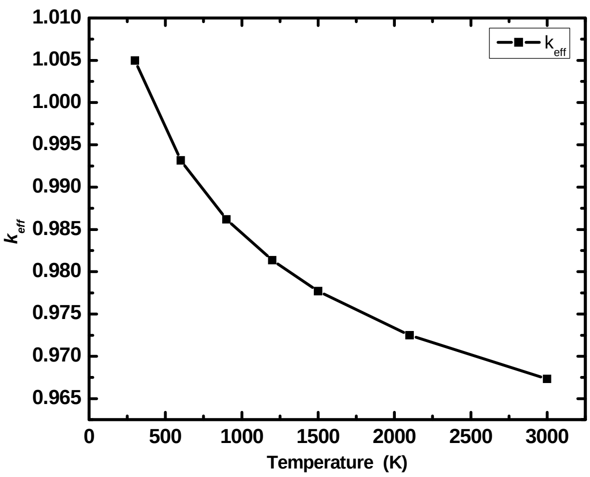


Fig.2 Effective multiplication factor of MOSART vs temperature

Figure 3 and Fig. 4 show the velocity and temperature fields in the MOSART core. The velocity field shows that the design of conic outlet can avoid stagnant regions and reverse flow in the core. And the temperature field is corresponding to the velocity field.

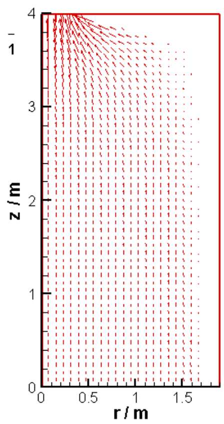


Fig.3 Velocity field

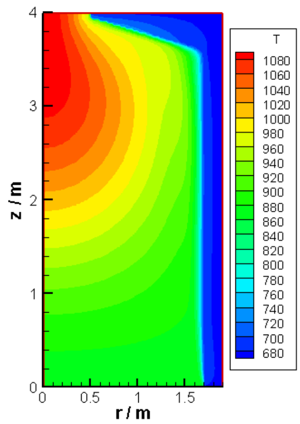


Fig. 4 Temperature field

The safety analysis model was firstly validated by the reactivity variation of MSRE in the pump coastdown experiment. Fig.5 displays the comparison of the calculated results and the experimental results, from which it is obviously found that the calculated results are consistent with the experimental data.

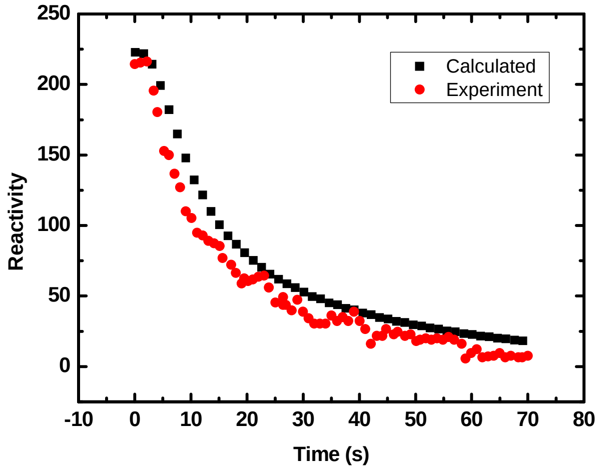
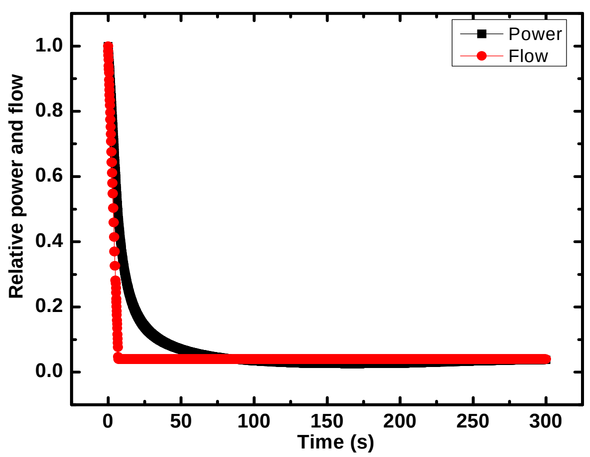


Fig.5 Reactivity in the pump coastdown condition of MSRE

Figure 6 and Fig. 7 depict the calculated results of the relative power (related to the rated power) and temperatures in the unprotected loss of flow in MOSART, in which the flow decrease to 4% of the rated value in seven seconds. Fig. 6 shows that the power decreases to 4% of the rated power, which is good corresponding to the flow decreasement. The average and outlet temperature of the fuel salt increase before 50 seconds into the transient, then decrease and finally stabilize nearby the initial values. The maximum temperature at the outlet is about 875°C, which is in the limitation of the MOSART. The average temperature of the graphite almost has no change because of its thermal inertia.

Figure 8 and Fig.9 illustrate the comparison between the calculated results and those from FZK, from which it can be found that the calculated results are in high agreement with those calculated by FZK.



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ig.6 Relative power and flow in the ULOF of MOSART

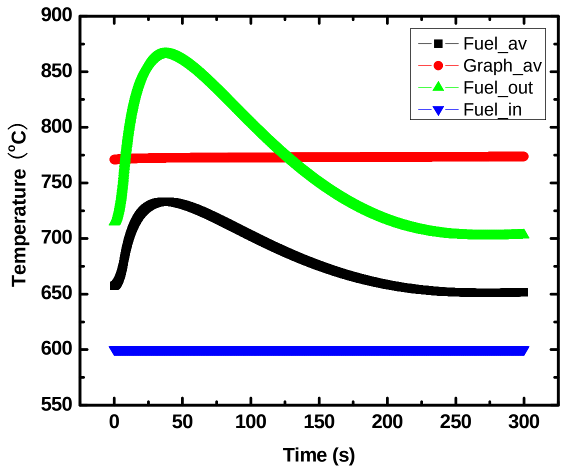

Fig.7 Temperatures in the ULOF of MOSART

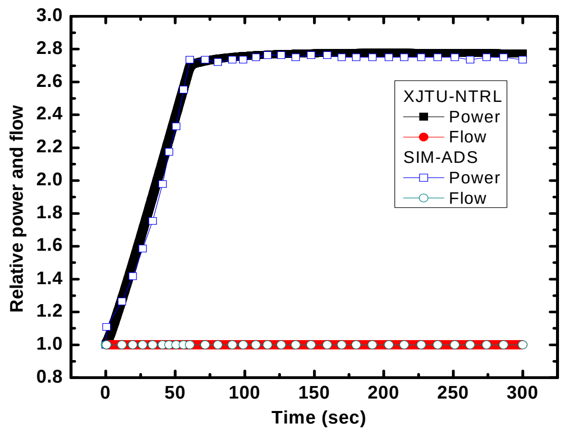

Fig.8 Relative power and flow in the UOC accident of MOSART

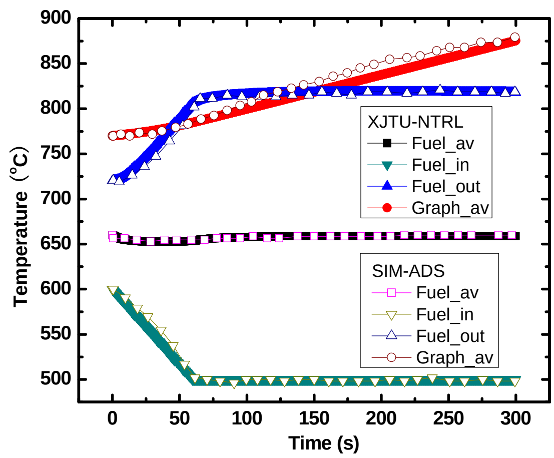

Fig.9 Temperatures in the UOC accident of MOSART

IV. Conclusion

In the present research, the themalphysical properties of the molten slats, neutron physics, thermal hydraulics and safety characteristics of the MSRs are studied by founding the theoretical models and developing numerical codes. The founded models and designed codes are applied to MOSART and the calculated results are compared with those benchmarks. Some representative results are shown in this presentation. From the results and comparison, it can be concluded that the established models are applicable to the molten salt reactors, and the designed codes can be easily applied to the other molten salt reactors because of its modular construction.

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