Analysis on reactivity initiated transient from control rod failure events of a molten salt reactor

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In a molten salt reactor (MSR), the fuel is dissolved in fluoride salt. In this paper, the reactivity worth and reactivity initiated transient of Molten-Salt Reactor Experiment (MSRE) in the control rod failure events are analyzed. The point kinetic coupling heat-transfer model with decay character of six-group delayed neutron precursors due to the fuel motion is applied. The relative power and temperature transient under reactivity step and ramp initiated at different power levels are studied. The results show that the reactor power and temperature increase to a maximum, where they begin to decrease to stable values. Comparing with full power level, the transient result at low power level is more serious. The results are of help in our study on safety characteristics of an MSR system.

Keywords: Reactivity initiated transient, Fluid fuel, Molten-salt reactor experiment

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I. INTRODUCTION

Molten salt reactor (MSR) is a class of nuclear fission reactors in which the fuel is fluid molten salt mixture. This type of reactor was originated for the U.S. Aircraft Nuclear Propulsion (ANP) at Oak Ridge National Laboratory (ORNL) in 1940s. The 2.5 MWth aircraft reactor experiment (ARE) was carried out in 1954, and the molten-salt reactor experiment (MSRE) was constructed in 1960s and operated until 1969 [1]. Also, a detailed 1000 MWe engineering conceptual design of Molten Salt Breeder Reactor (MSBR) was developed [2]. Having advantages of good neutron economy, inherent safety, online reprocessing, less radioactive waste, nuclear nonproliferation etc., MSR is one of the six advanced reactor types for future nuclear energy systems in the Generation IV International Forum (GIF) [3], attracting many research groups in recent years. New MSR concepts proposed include AMSTER, FUJI, MOSART, and MSFR [4–7]. In 2011, the Chinese Academy of Sciences also launched the project for technologies related to Thorium-based Molten Salt Reactor (TMSR) [8].

In MSR, the fuel is dissolved in fluoride salt and most of fission energy is released into the molten salt. Due to the fluid fuel, a part of delayed neutron precursors drift out of the reactor and decay in the loop. So, the reactivity and effective delayed neutron fraction of MSR differ from solid fuel reactors. Haubenreich *et al*. [9] performed the reactivity initiated transient analysis with simple Murgatroyd model for MSRE. In safety analysis of MSBR, Shimazu [10] used the point-kinetic model of two-group delayed neutron precursors. Recently, Zhang *et al*. [11] developed the point-kinetic model of six-group delayed neutron precursors to analyze the MOSART safety characteristics.

The control rod failure event is an important postulated initiating events for safety analysis of nuclear reactor [12], which may lead to the step reactivity initiated and ramp reactivity initiated in the core. In this paper, for studying reactivity initiated transient analysis of an MSR during control rod failure events, a point kinetic coupling heat-transfer model is applied. In this model, the flow effect of fuel salt with six-group delayed neutron precursors and character of the delayed neutrons precursors are considered. The transient of relative power and temperature in the control rod failure events of MSRE are analyzed at different power levels. The study reveals safety characteristics of the MSRE and provides information for design of the MSR system.

II. SYSTEM DESCRIPTION

MSRE was built in 1960s to demonstrate practicality of key technologies of molten-salt reactors. It ran 9000 and 2500 equivalent full power hours with ^{235}U and ^{233}U, respectively, in the fuel salt. The reactor power was designed at 10 MWth, using fuel salt of $LiF-BeF_2-ZrF_4-UF_4$ (65-29.2-5-0.8, mol%) and coolant salt of $LiF-BeF_2$(66-34, mol%), with a graphite moderator and all salt-contacting parts made of Hastelloy N [13]. Reactor heat was transferred from fuel salt to coolant salt by a heat exchange, and dissipated to the atmosphere by the radiator. Table 1 lists the principal physical parameters.

In the MSRE, the reactivity should be adjusted at normal operation due to variation of the reactor power, xenon poison, samarium poison, delayed neutron losses, burn-up, and entrained gas. Therefore, the control rods are withdrawn to compensate for various effects of reactivity. Table 2 shows the calculated and measured reactivity worth of control rods. The worth of single control rod is 2.11% δk/k in the initial critical concentration, which meets the need of reactivity adjustment and retained the big margin. In the design of MSRE, three control rods were used to adjust the reactivity with the maximum speed of 1.27 cm/s corresponding to the ramp reactivity 0.04% δk/k/s, and the value for withdrawing any single rod was restricted at 0.35% δk/k. Thus, in the reactivity ini-

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TABLE 1. Principal physical parameters of MSRE [14]

Prompt neutron lifetime (s)	2.4×10^{-4}	Mass velocity of fuel (kg/s)	174.13
Core inlet temperature(°C)	635	Core outlet temperature (°C)	662.78
Fuel specific heat (J/kg °C)	1967.83	Residence time in the core (s)	9.37
Graphite specific heat (J/kg °C)	1758.49	Residence time in the loop (s)	16.45
Graphite-fuel heat transfer (J/s °C)	3.6×10^{4}	Fuel density (648.89 °C), (g/cm^3)	2.3
Fraction of heat generation		Temperature coefficients of reactivity	
Fuel (%)	93.3	Fuel ((δk/k)/°C)	-5.45×10^{-5}
Graphite (%)	6.7	Graphite ((δk/k)/°C)	-6.05×10^{-5}

TABLE 2. Reactivity worth (δk/k, in %) of control rods [15]

Control rods inserted	Initial critical concentration		1.1× Initial critical concentration	
	Calculated	Measured	Calculated	Measured
One rod	2.11	2.26	1.94	2.08
Three rods	5.46	5.59	—	—

tiated transient analysis for the control rod failure events of MSRE, the maximum value of step reactivity is 0.35% δk/k and the ramp reactivity is 0.04% δk/k/s within 10 seconds.

III. MATHEMATICAL MODELS AND SOLUTION METHOD

A. Point kinetic model

In MSR, while part of the delayed neutron precursors drift out of the core and decay in the loop, some flow back into the core. The point kinetics model used in solid fuel reactors is not suitable for MSR. In this paper, we use the fluid fuel point kinetic model based on the drift of delayed neutron precursors [10]. The decay of six-group delayed neutrons in the loop and their returning back to the core are considered. The delayed neutron constants are showed in Table 3. The effective fraction value β_{eff} and the circulation lag term $C_i(t-\tau_1)$ are adopted. The point kinetic equations for MSR can be written as:

$$\frac{\mathrm{d}P(t)}{\mathrm{d}t}=\frac{\rho(t)-\beta_{\text{eff}}}{\Lambda}N(t)+\sum_{i=1}^{6}\lambda_iC_i(t),\tag{1}$$

$$\begin{aligned}\frac{\mathrm{d}C_i(t)}{\mathrm{d}t}=&\frac{\beta_{\rm i}}{\Lambda}P(t)-\lambda_iC_i(t)-\frac{1}{\tau_c}C_i(t)\\&+\frac{\exp(-\lambda_i\tau_i)}{\tau_c}C_i(t-\tau_i),\end{aligned}\tag{2}$$

where, C_i is precursor concentration of delayed neutron; β_i is delayed neutron fraction; β_{eff} is effective fraction of delayed neutron; $P(t)$ is core power level; λ_i is decay constant of delayed neutron; Λ is lifetime of prompt neutron; $\tau_{\rm L}$ are $\tau_{\rm c}$ transit time of fuel in the loop and in reactor, respectively; ρ is reactivity initiated value; $\alpha_{\rm f}$ and $\alpha_{\rm g}$ are temperature coefficients of reactivity of fuel and graphite, respectively; $T_{\rm f}$ and $T_{\rm g}$ are average temperature of fuel and graphite, respectively.

TABLE 3. Delayed neutron constants for ^{235}U fuel [16]

Groups	$T_{1/2}$(s)	λ_i	$\beta_i(10^{-4})$
1	55.9	0.0124	2.23
2	22.7	0.0305	14.57
3	6.22	0.1114	13.07
4	2.30	0.3013	26.28
5	0.61	1.140	7.66
6	0.23	3.010	2.80

The reactivity feedback, $\rho(t)$, due to changes in temperature and reactor power, can be written as

$$\rho(t)=\rho_0+\rho_{\text{in}}+\alpha_{\rm f}(T_{\rm f}-T_{\rm f0})+\alpha_{\rm g}(T_{\rm g}-T_{\rm g0}),\tag{3}$$

where, ρ_0 is reactivity value in steady state; ρ_{in} is reactivity initiated value; $\alpha_{\rm f}$ and $\alpha_{\rm g}$ are temperature coefficients of reactivity of fuel and graphite, respectively; $T_{\rm f}$ and $T_{\rm g}$ are average temperature of fuel and graphite, respectively; and $T_{\rm f0}$ and $T_{\rm g0}$ are temperature of fuel and graphite in steady state, respectively.

B. Heat transfer model

According to the principle of energy conservation, a heat transfer model with an external loop of fuel circulation and a heat exchanger is established. Thermal energy is generated inside the core and transported out of the core by the circulating fuel in external loop. As showed in Fig. 1, the reactor core zone is divided into a graphite lump and two fuel salt lumps along the axial direction. The heat sink Q is introduced to demonstrate heat characteristics of the primary loop which removes the core heat.

The heat generation in reactor core, heat transport due to mass flow of the fuel, and the heat released inside the heat exchanger are taken in consideration. The mass flow of fuel and heat sink is simply considered as constant in the model.

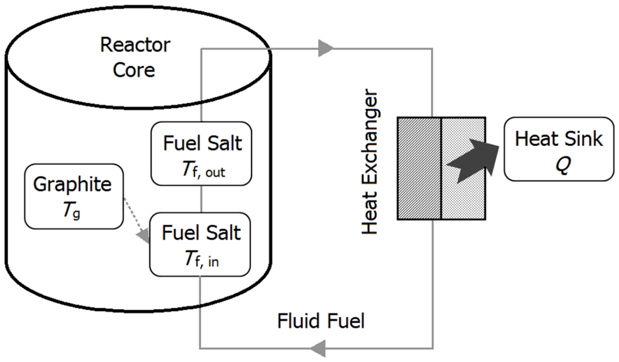

Fig. 1. Lump parameter model of MSRE

The corresponding equations can be written as:

$$M_gC_{p,g}dT_g/dt = F_gP + h_{fg}A_{fg}(T_f - T_g), \tag{4}$$

$$\begin{aligned} M_fC_{p,f}dT_{f.in}/dt =& F_fP - wC_{p,f}A_{fg}(T_{f.in} - T_{in}) \\ &+ h_{fg}A_{fg}(T_g - T_{f.in}), \end{aligned} \tag{5}$$

$$dT_{f,out}/dt = F_fP - wC_{p,f}(T_{f,out} - T_{in}), \tag{6}$$

$$M_fC_{p,f}dT_{f,h}/dt = h_{fg}A_{fg}(T_{f,h} - T_g) - Q, \tag{7}$$

where P is core power level; h_{fg} is graphite-fuel heat transfer coefficient; A_{fg} is graphite-fuel heat transfer area; F is fraction of heat generation; M_f and M_g are mass of fuel salt and graphite, respectively; w is mass flow of fuel salt, $C_{p,f}$ and $C_{p,g}$ are specific heat of fuel salt and graphite, respectively; and T_f and T_g are temperature of fuel salt and graphite, respectively.

C. Solution method

The point kinetic coupling heat-transfer model is a stiff differential equation with delay function. DDE23 solver of Matlab software is an effective and reliable method for solution of the delay function, which applies the variable step size to appropriate iterated by the implicit formula [17]. It can be used to solve the stiff differential equation. Thus, the model equations are solved by using DDE23 solver, and the initial value of models is steady state of MSRE at normal operation.

IV. EXPERIMENTAL VALIDATION

In operation of MSRE, experiments were performed to investigate safety characters of the reactor. The dynamic transients of start-up and coast-down with fuel pump in MSRE were performed to evaluate in zero-power operation conditions. In these transients, the turn-on or turn-off status of fuel pump would impact on the circulation of delayed neutron precursors, hence the loss of reactivity. Thus the reactivity should keep the power at a normal level by adjusting the control rod. The results of dynamic transients in calculations and experiments are presented in Fig. 2 [18, 19].

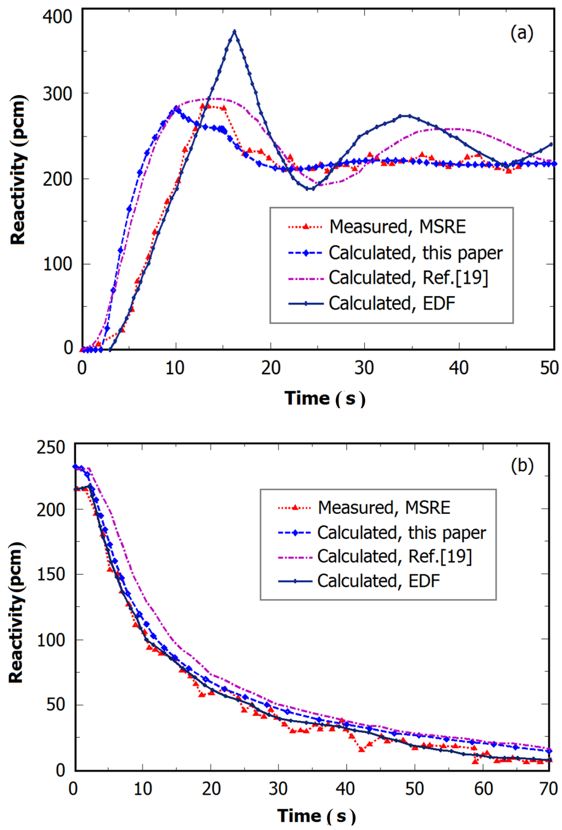

Fig. 2. (Color online) Reactivity inserted during fuel pump start-up (a) and coast-down (b) transients in the MSRE.

From Fig. 2, in the start-up transient, the fuel-salt flow from no motion towards its normal speed in 10 seconds, and some delayed neutron precursors of the loop can reenter the core, hence the increase of reactivity in the core. In the fuel coast-down transient, the fuel-salt flow from the critical condition decrease to zero in 15 seconds, and delayed neutron precursors stop leaving the core, hence the need of inserting control rod to keep the reactor core critical. Our calculation results are in reasonable agreement with the reference data, though there are differences that may be caused by different choices of the flow-rate that controlled by the pump speed.

V. RESULTS AND DISCUSSION

A. Reactivity initiated at full power level

The MSRE is assumed to run at full power equilibrium condition, without any change of the heat sink. The 0.35% δk/k step reactivity and 0.04% δk/k/s for 10 seconds ramp reactivity is initiated. Fig. 3 shows the relative power and temperatures transient results during step reactivity initiated. The relative power rises rapidly to a peak value of about 7.2 times, where it begins to decrease until the steady state. The average

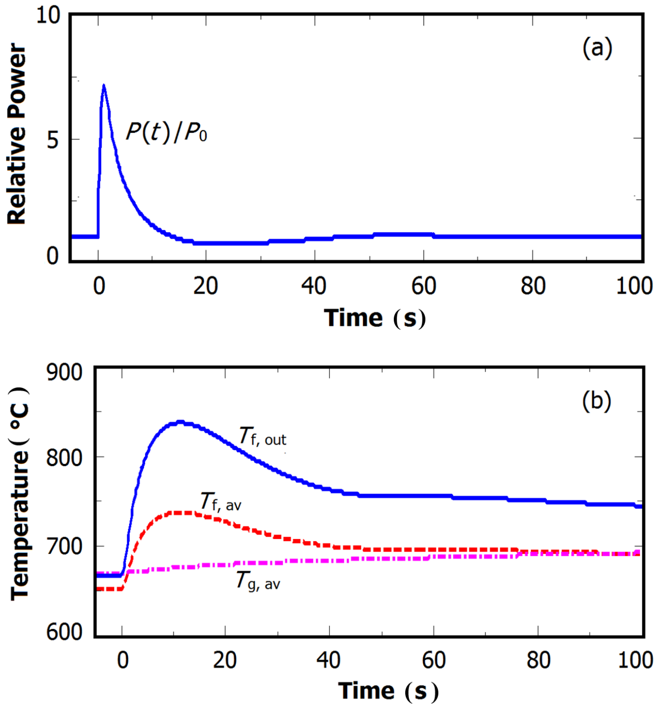

Fig. 3. (Color online) Reactivity step initiated at full power of MSRE: relative power (a) and temperatures of the fuel and graphite (b) during the transients.

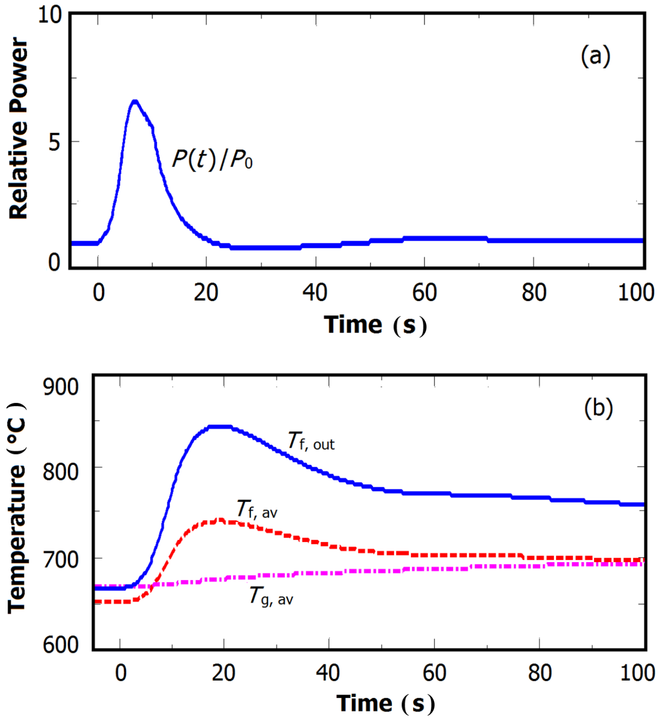

Fig. 4. (Color online) Reactivity ramp initiated at full power of MSRE: relative power (a) and temperatures of the fuel and graphite (b) during the transients.

and outlet temperatures of fuel increase quickly in 10 s to a maximum of 736.6 °C and 838.5 °C, respectively, decreasing gradually to their steady state, but the average temperatures of graphite increase a little.

Under the ramp reactivity initiated value of 0.04% δk/k/s for 10 seconds, the results of relative power and temperature transient in reactor are showed in Fig. 4. The temperature transient rises to the maximum at 20 s. It is longer than step reactivity initiated because of its slow reactivity initiated velocity. The maximum outlet temperature of fuel is 844.3 °C, a little higher than that of the step reactivity initiated. This is because the total initiated value of reactivity is 0.4% δk/k in the 10 seconds. The temperature transient values are below the allowable temperature for Hastelloy N [20].

B. Reactivity initiated at low power level

The same initiated value of step reactivity and ramp reactivity are analyzed with MSRE is assumed to be critical at the power level of 1 kWth. Under this condition, the fuel is static with no heat removal from the reactor. Figs. 5 and 6 show the results of relative power and temperatures in reactivity initiated transient at this low power. The relative power and temperature increase to the maximum, where they decrease gradually to the steady state. This is similar to the results at full power level. In the step reactivity initiated transient, the maximum average and outlet temperature of molten fuel is 771.4 °C and 847.8 °C, respectively, in 40 seconds, while

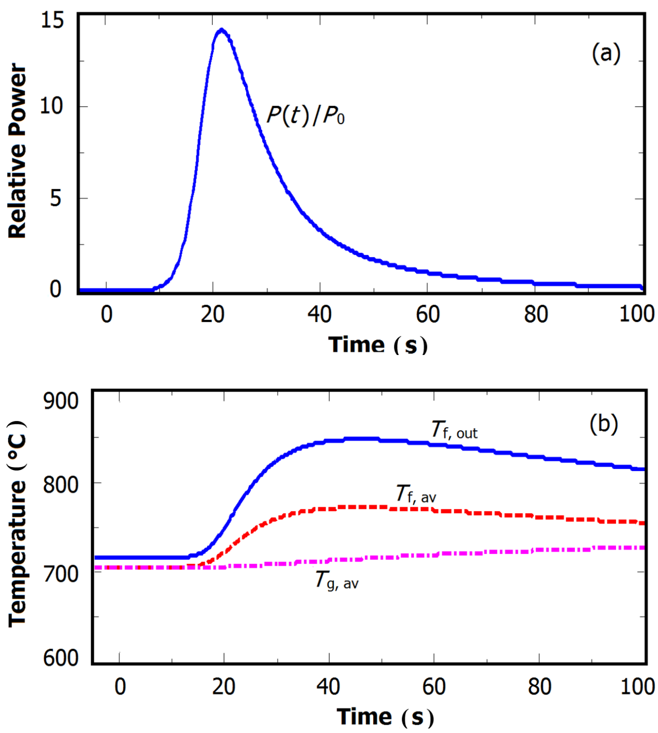

Fig. 5. (Color online) Reactivity step initiated at low power of MSRE: relative power (a) and temperatures of the fuel and graphite (b) during the transients.

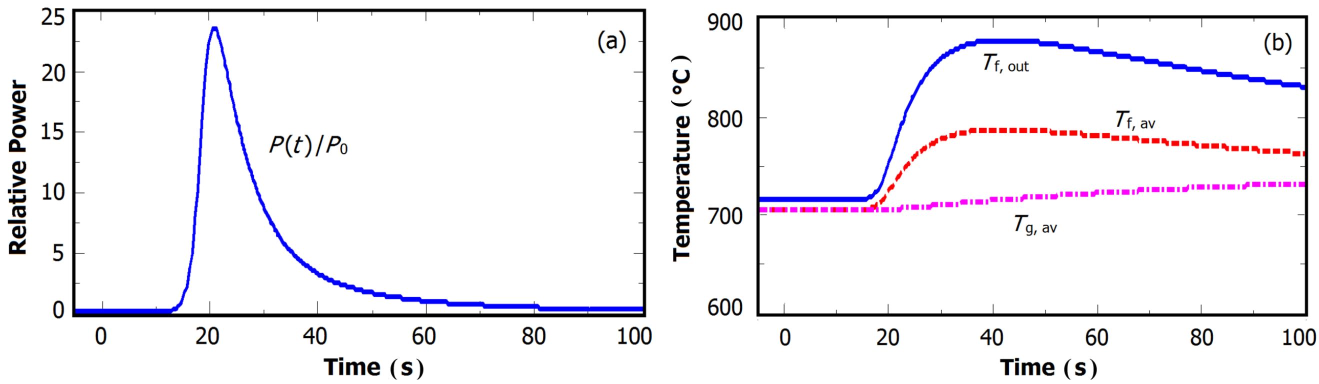

Fig. 6. (Color online) Reactivity ramp initiated at low power of MSRE: relative power (a) and temperatures of the fuel and graphite (b) during the transients.

in the ramp reactivity initiated transient, the maximum average temperature and outlet temperature of molten fuel are 785.2 °C and 877.5 °C, respectively, in 45 seconds. The results show that the transient result of relative power and temperatures at low power level is more serious than that of full power level. The reason is that the losses of delayed neutron fraction happen at the full power with the circulating fuel in the core and loop. But the values of temperature transient are below the maximum allowable temperature of Hastelloy N, too. The time of maximum transient values at low power level is longer than that at the full power, hence longer time to control the reactor during the abnormal condition.

VI. CONCLUSION

The control rod failure events, which may cause abnormal reactivity initiated in a nuclear reactor, should be studied in reactor safety analysis. In this paper, the reactivity initiated transient of MSRE is studied by applying the fluid-fuel point kinetic model coupled with a heat transfer model in the control rod failure events. Decay character of six-group delayed neutron precursors in the loop is considered. The relative power and temperature transient under reactivity step and ramp initiated at different power levels are analyzed. The results show that reactor power and temperature rise to a peak value and decrease gradually to a steady state. The transient result of relative power and temperatures at low-power level is more serious than that of full power level because of the fluid fuel. However, this has little effect on safety of MSRE. This study helps us to understand the power and temperature transient of the MSRE, and provides information for design of our MSR system. Still, we shall optimize the mathematical model, towards its better applications in safety analysis of the MSR system.

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