Simulation of Natural Convection in the Molten Salt Reactor Experiment Using GAMMA+

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1. Introduction

The Molten Salt Reactor Experiment (MSRE) [1] is an experimental nuclear reactor designed for a thermal power of 10 MW. It is a graphite-moderated, molten-salt fueled, thermal reactor. It was designed, built, and operated at the Oak Ridge National Laboratory (ORNL) in the 1960s. In the MSRE, the nuclear fuel and the primary coolant are the same fluid.

In this work, a natural convection heat removal test [2] performed in the MSRE was analyzed by using the GAMMA+ code [3]. The results of the GAMMA+ code were compared with the measured data and four kinds of point-kinetics (PK) models specially developed for a molten-salt fueled reactor (MSR) were tested.

2. Natural Convection Test in MSRE

Fig. 1 shows the layout and major components of the MSRE. Nuclear fission reaction occurs at the active region in the reactor vessel which is connected by the piping, the fuel pump, and the heat exchanger. The fuel salt was $LiF\text{-}BeF_2\text{-}ZrF_4\text{-}UF_4$. The liquid fuel-salt in the primary loop was circulated by the fuel pump.

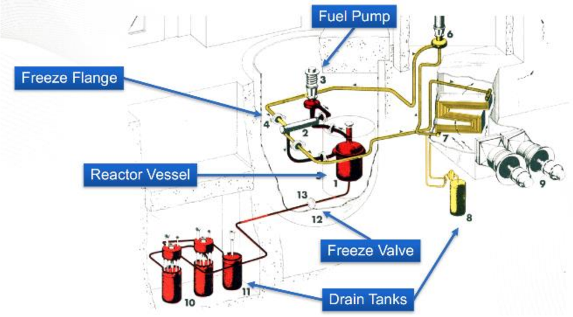

Fig. 1. Layout of MSRE [1].

As shown in Fig. 2, the reactor core was formed of 617 2-in x 2-in graphite stringers. Stringers were mounted in a vertical, close-packed array which formed vertical fuel salt channels (~1140 equivalent fluid channels).

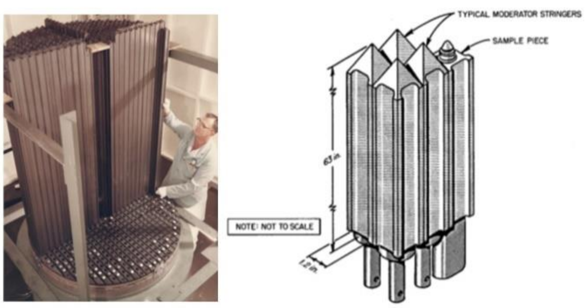

Fig. 2 Graphite moderator of MSRE [1].

The primary loop was cooled by the coolant salt loop via the salt-to-salt shell and tube heat exchanger. The coolant salt was $LiF\text{-}BeF_2$. And the coolant salt loop was cooled by outside air via the air-cooled radiator. Two blowers were used to supply air to the radiator.

A natural convection test was performed to investigate the heat removal characteristics of the MSRE by using natural convection flow of the fuel salt. Forced circulation in the coolant salt loop was maintained during the experiment. The heat removal rate in the air radiator was increased in steps keeping the reactor critical. The reactor power was solely controlled by inherent feedback of the MSRE.

3. GAMMA+ Model

Fig. 3 shows the GAMMA+ nodalization to simulate the natural convection test of the MSRE. Unfortunately, detailed cooling conditions in the air radiator are not available. Therefore, thermo-fluid conditions of the coolant salt loop were used as boundary conditions and the air-radiator was not considered in the present work.

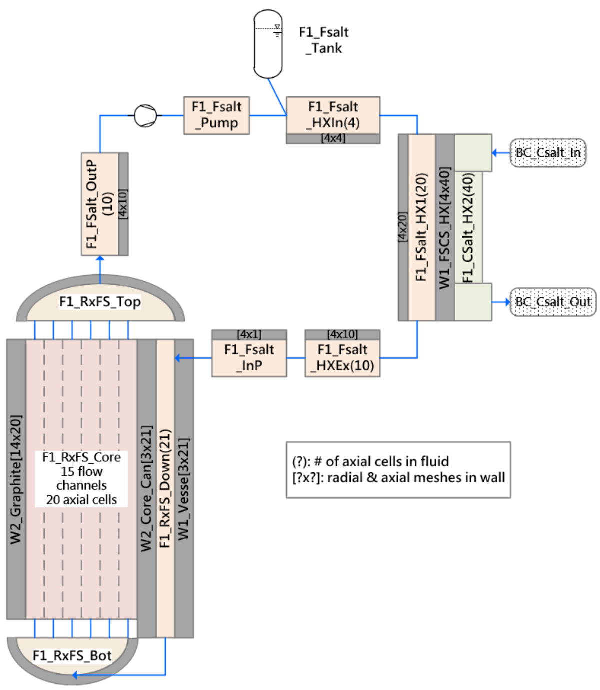

Fig. 3. GAMMA+ nodalization to simulate natural convection in the MSRE.

Four kinds of PK models are available in GAMMA+ to simulate MSRs. Eqs. (1) and (2) are used in PK type 1 and PK type 2. These models (which are traditional in MSR applications) are modified version of the PK model of solid-fuel reactors. They consider the fuel transit time and the decay of the delayed-neutron precursors during the fuel salt is not in the core.

$$\frac{dP}{dt} = \frac{\rho-\beta_{eff}}{\Lambda} P + \sum_{i=1}^{6} \lambda_i C_i \quad (1)$$

$$\frac{dC_i}{dt} = \frac{\beta_i}{\Lambda} P - \lambda_i C_i - \frac{C_i}{\tau_c} + \frac{C_i(t-\tau_l)}{\tau_c} e^{-\lambda_i \tau_l} \quad (2)$$

where τ_c = fuel transit time in the core, τ_l = fuel transit time in the loop. On the other hand, Eqs. (3)-(5) are used in PK type 3 and PK type 4. The idea of these models was proposed by Guo et al. [4]. They separate the concentrations of the delayed-neutron precursors into two parts (i.e., inside the core ($C_{c,i}$) and outside the core ($C_{l,i}$)).

$$\frac{dP}{dt} = \frac{\rho-\beta_{eff}}{\Lambda} P + \sum_{i=1}^{6} \lambda_i C_i \quad (3)$$

$$\frac{dC_{c,i}}{dt} = \frac{\beta_i}{\Lambda} P - \lambda_i C_{c,i} - \frac{C_{c,i}}{\tau_c} + \frac{V_l}{V_c}\frac{1}{\tau_l} C_{l,i}(t-\tau_l) \quad (4)$$

$$\frac{dC_{l,i}}{dt} = -\lambda_i C_{l,i} - \frac{C_{l,i}}{\tau_l} + \frac{V_c}{V_l}\frac{1}{\tau_c} C_{c,i} \quad (5)$$

Depending on the definition of the core, the PK models are further classified into four sets. Fig. 4 explains the concepts of the four models. The graphite core region is used for PK type 1 and PK type 3 whereas the entire vessel region is used for PK type 2 and PK type 4.

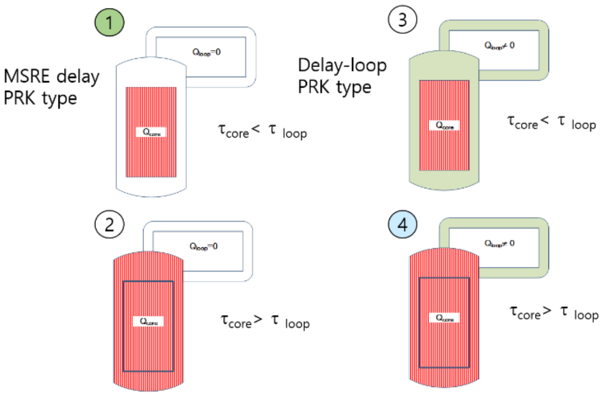

Fig. 4. Four kinds of point kinetics models in GAMMA+.

The fuel and graphite temperature coefficients obtained by the Serpent calculations [5] for the MSRE with U-233 fuel were adopted in this work for thermal feedback.

4. Results and Discussions

It is expected that the flowrate of the coolant salt was gradually increased with that of the air in the radiator. Therefore, it was assumed that the flowrate of the coolant salt was linearly increased with time until the reactor power reached the peak (~250 min.). Fig. 5 explains this assumption. The peak flowrate was determined by series of GAMMA+ calculations. As shown in Fig. 6, it was found that the GAMMA+ calculation with the peak value of 6.3 kg/s produces the best agreement with the measured data.

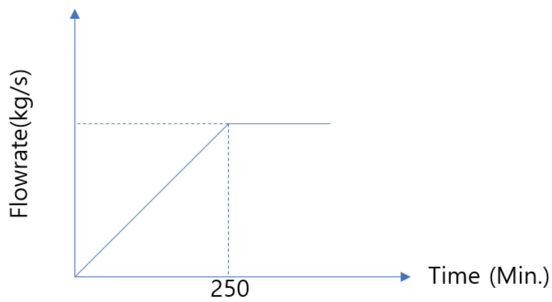

Fig. 5. The assumed flowrate of coolant salt.

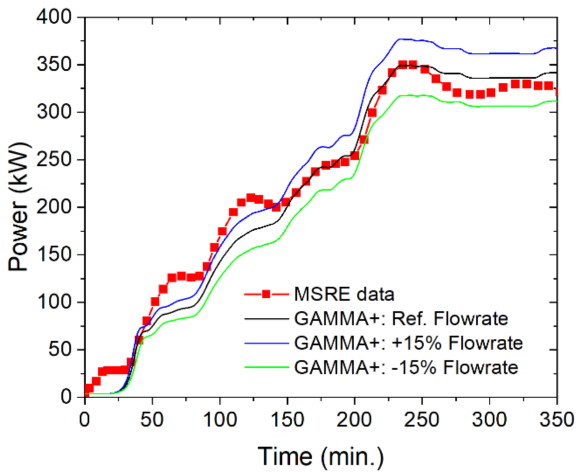

Fig. 6. The sensitivity result of GAMMA+ calculations: coolant salt flowrate vs. reactor power (based on PK type 1).

Figs. 7~9 show the results of the GAMMA+ calculations with the peak coolant salt flowrate of 6.3 kg/s. The figures show good agreements between the GAMMA+ results and the measured data. The figures also show that the results of the GAMMA+ calculations are not largely affected by the choice of the PK model. Such a result is reasonable since the speed of the fuel salt is so slow that most of the delayed-neutron precursors are decayed in the core. The core transit time of the fuel salt was calculated to be 227 second at the end of the calculation (= 350 min.). It should be noted that the largest half-life of the delayed-neutron precursor group is less than 1 minute.

A slight over-prediction was obtained in Fig. 8. It seems that the reasons of such differences are mainly from the uncertainties in the assumed flowrate of the coolant salt.

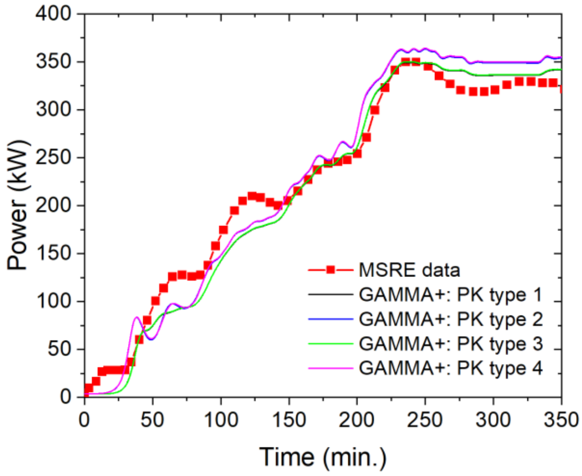

Fig. 7. The predicted reactor power and comparison with measured data.

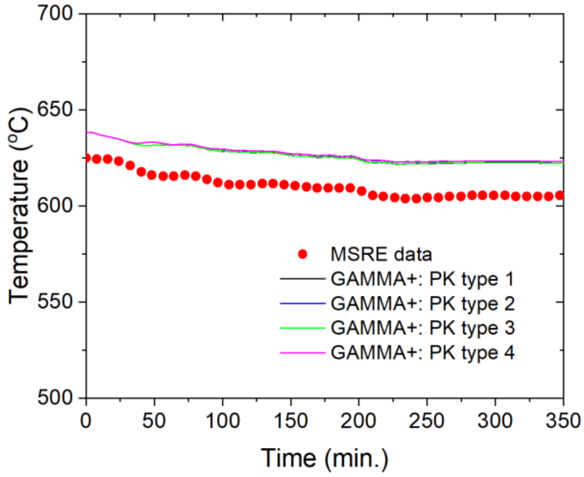

Fig. 8. The predicted inlet temperature of fuel salt and comparison with measured data.

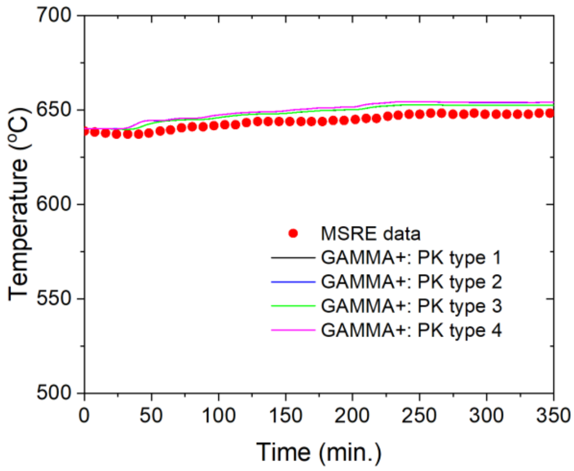

Fig. 9 The predicted outlet temperature of fuel salt and comparison with measured data.

5. Conclusions

In this paper, GAMMA+ calculations were carried out for the thermal convection heat removal test of the MSRE. Good agreements were found between the GAMMA+ results and the measured data. The applicability and usefulness of the GAMMA+ code for the thermo-fluid analyses of MSRs are well illustrated in this work. It is also confirmed that in the thermo-fluid conditions of thermal convection test of the MSRE, the choice of the PK model is not crucial in the GAMMA+ calculations.

Intensive and extensive researches are required to develop an MSR in Korea. These activities include the enhancement of physical models of the GAMMA+ and their verification and validation studies. It is envisaged that the GAMMA+ will be the major tool for the thermo-fluid and safety analysis of MSRs in KAERI.

Acknowledgements

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REFERENCES

[1] R. C. Robertson, MSRE Design and Operations Report Part I, Description of Reactor Design, ORNL-TM-728, Oak Ridge National Laboratory, 1965.01.

[2] M. W. Rosenthal, Molten-Salt Reactor Program Semiannual Progress Report for Period Ending February 28, 1969, ORNL-4396, Oak Ridge National Laboratory,1969.08.

[3] H. S. Lim, GAMMA+ 2.0 Volume II: Theory Manual, KAERI/TR-8662/2021, Korea Atomic Energy Research Institute, 2021.

[4] Z. Guo et al., "Simulation of Unprotected Loss of Heat Sink and Combination of Events Accidents for a Molten Salt Reactor," *Ann. Nucl. Energy*, Vol. 53, pp. 309-319, 2013.

[5] T. Hanusek, R. M. Juan, "Analysis of the Power and Temperature Distribution in Molten Salt Reactors with TRACE. Application to the MSRE," *Ann. Nucl. Energy*, Vol. 157, 108208, 2021.