

# Sensitivity Study on the Natural Convection Test in the Molten Salt Reactor Experiment using the GAMMA+ Code

Hyun-Sik Park <sup>a\*</sup>, Nam-il Tak, and Hong-Sik Lim

<sup>a</sup> Korea Atomic Energy Research Institute, 989-111 Daedeokdaero, Yuseong, Daejeon, 34057, Korea

\*Corresponding author: hspark@kaeri.re.kr

\***Keywords:** MSR, MSRE, GAMMA+, Natural Convection

## 1. Introduction

Molten-salt fueled reactor (MSR) has different reactor physics from other solid fueled reactors. It is due to the fact that the fission product nuclides including delayed-neutron precursors circulate the entire fuel system, from the core to the loop and vice versa.

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 neutron 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.

The system analysis code, GAMMA+ code [2], has three kinds of point-kinetics (PK) models for predicting neutronic behaviors of MSR named as: 1) decay-term point reactor kinetics (DT-PRK), 2) delay-loop point reactor kinetics (DL-PRK), and 3) nuclide groups transport kinetics (NTK). Previously a natural convection heat removal test and a reactivity insertion test performed in the MSRE were analyzed using the GAMMA+ code [3, 4]. Also the steady-state and transient benchmarks of GOTHIC to the MSRE were performed [5]. They included the simulation results on the pump start-up and coast-down transient tests and natural convection test, which demonstrated the GOTHIC capabilities for modeling MSR design with circulating fuel.

In this paper, the natural convection heat removal test [6] performed in the MSRE were analyzed using the GAMMA+ code. The results of the GAMMA+ code were compared with the measured data and the calculations with two sensitivity parameters of different active core regions and different heat transfer correlations in free convection conditions.

## 2. Natural Convection Test in MSRE

### 2.1 MSRE System

Fig. 1 shows the layout and major components of the MSRE. Nuclear fission reaction occurs at the entire region within the reactor vessel which is connected by the piping, the fuel pump, and the heat exchanger. The fuel salt was LiF-BeF<sub>2</sub>-ZrF<sub>4</sub>-UF<sub>4</sub>. The liquid fuel-salt in the primary loop was circulated by the fuel pump.

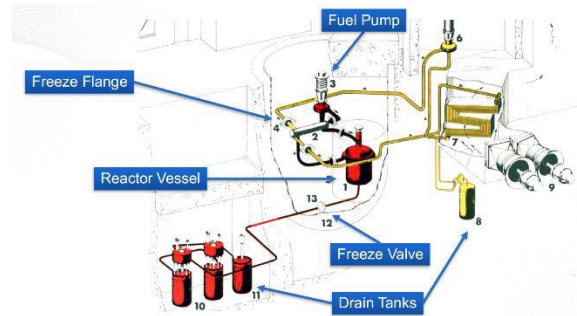


Fig. 1. Layout of MSRE [1].

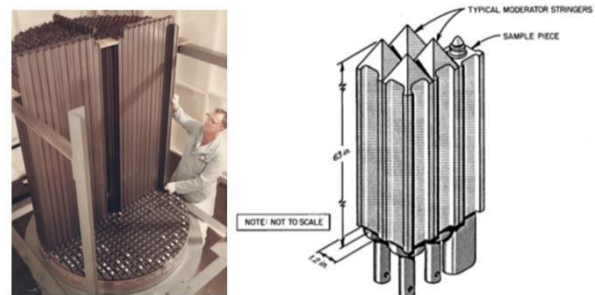


Fig. 2 Graphite moderator of MSRE [1].

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 (~1,140 equivalent fluid channels).

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-BeF<sub>2</sub>. The coolant salt loop was cooled by the outside air via the air-cooled radiator. Two blowers were used to supply air to the radiator.

### 2.2 Natural Convection Heat Removal Test

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.

During lower-power testing with  $^{233}\text{U}$  fuel, a natural convection test was performed with no control rod motion such that reactor power was responding to the heat load demand from the radiator. With forced circulation in the coolant salt loop, the radiator heat removal rate was increased in steps by incrementally opening the radiator door(s); each radiator adjustment was made after the reactor power rose as a result of the previous adjustment, thereby approaching equilibrium. The available data from the test are reactor power, heat exchanger fuel salt inlet and outlet temperatures, and heat exchanger coolant salt inlet and outlet temperatures, along with the timing of movements of the radiator doors [5].

### 3. GAMMA+ Simulation on the Natural Convection Test

#### 3.1 GAMMA+ Input Models

GAMMA+ input models are constructed with fuel salt system, as shown in Fig. 3. Base input model refers the ORNL documents [1, 6]. Since the detailed cooling conditions in the air radiator are not available, in the present sensitivity study the thermo-fluid conditions of the coolant salt loop were used as boundary conditions and the air-radiator was not considered in the present work.

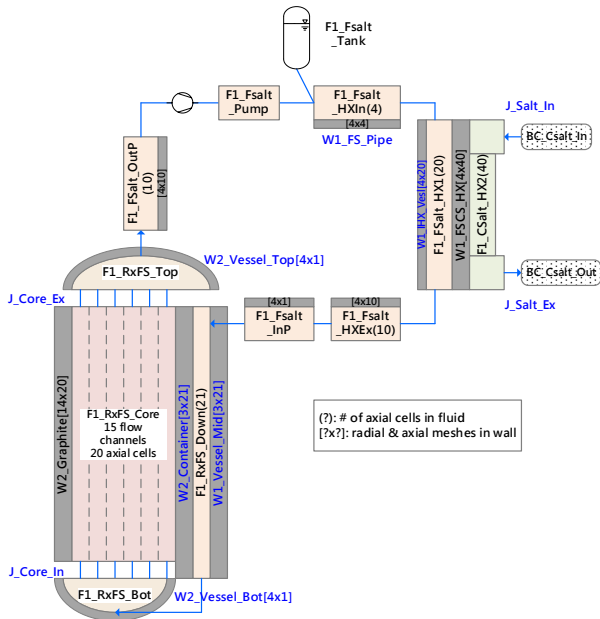


Fig. 3. GAMMA+ nodalization to simulate the fuel-pump test in the MSRE.

#### 3.2 PK models in the GAMMA+ code

As explained by Yoon et al [4], there are efforts to calculate accurately the fission power by obtaining the concentration of the delayed-neutron precursors in the core and the loop. It can be estimated that the fission power and spatial concentrations significantly influenced the occupied region of the core. However,

the determination of the core region is not clear in the MSRE system since the fission reaction occurs at the graphite region as well as outside the graphite region in the reactor vessel (i.e., top plenum, bottom plenum, and downcomer). It would be confused whether the core region should be defined as from the downcomer to the top plenum or any partial region in the vessel. In the present study, three kinds of active fission regions are considered: 1) the graphite core region only (Core), 2) the extended core region including half of the bottom and top plena (Ext-Half), and 3) the extended core region including all the bottom and top plena (Ext-Full).

Three kinds of PK models are available in the GAMMA+ code to simulate the neutronic behavior of MSRs. The concept of PK type 1(DT-PRK, Decay-term point reactor kinetics) in the GAMMA+ code was developed by ORNL [7]. The delayed-neutron precursors can decay when they are in the loop before they re-enter to the core. Equations (1) and (2) are used in PK type 1. 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  $P$ ,  $\rho$ ,  $\beta$ ,  $\Lambda$ ,  $C_i$ ,  $\lambda_i$ ,  $\tau_c$ ,  $\tau_l$  are the fission power, the reactivity, the effective fraction of delayed neutrons, the effective prompted neutron lifetime, the concentration of the delayed-neutron precursors, the decay constant of delayed-neutron precursors, the fuel transit time in the core, and the fuel transit time in the loop, respectively. The last two terms in the equation (2) are also applied to poison and decay heat nuclides in the GAMMA+ code. The limitation of DT-PRK is that it only solves for the core power from the concentration of the precursors in the core.

D. Zhang et al. [8] suggested additional equations to obtain the concentration of the delayed-neutron precursors in the loop as following equations of Equations (3)-(5). In the PK type 2(DL-PRK, Delay-loop point reactor kinetics) of the GAMMA+ code the concentrations of the delayed-neutron precursors are separated into two parts, inside the core and outside the core.

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

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

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

where  $C_{c,i}$ ,  $C_{l,i}$ ,  $V_c$ ,  $V_l$  are concentrations of the delayed-neutron precursors in the core and the loop, and volumes of the core and the loop. The last two terms in the equation (4) and the equation (5) are also implemented to poison and decay heat nuclides in the GAMMA+ code.

DL-PRK has capabilities to calculate the decay heat in the loop and the concentration of the delayed-neutron precursors in the loop. However, the data of the entire loop is represented as the one point. So the PK type 3(NTK, Nuclide groups transport kinetics) is suggested to calculate the concentration of the delayed-neutron precursors in every cell as following equations:

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

$$\frac{dC_{i,j}}{dt} = \frac{\beta_i}{\Lambda} P - \lambda_i C_{i,j} + \frac{1}{V_j} [(uAC_i)_{in} - (uAC_i)_{ex}] \quad (7)$$

where j is the index of the cell, V is the volume, u is the velocity, and A is the flow area.

Depending on the definition of the core region (the graphite core region = Core, the extended core region including half of the bottom and top plena = Ext-Half, the extended core region including all the bottom and top plena = Ext-Full), the PK models are further classified. The fuel and graphite temperature coefficients obtained by the Serpent calculations [9] for the MSRE with U-233 fuel were adopted in this work for thermal feedback.

### 3.3 Sensitivity on the PK models in GAMMA+

In the present study the PK models of type 3 (NTK) are simulated using the GAMMA+ code since it is the most advanced one among three kinetics models implemented in the current GAMMA+ version.

It is very important that the appropriate core region is properly defined to calculate the fission of molten salt fuel accurately throughout the core and loop regions, and thus the core region is extended from the graphite core region only to some parts of the bottom and top plena. It is because the core region is not fixed as the molten salt coolant is circulating the primary loop. Through some sensitivity calculation, it is considered that the NTK model simulates the MSRE data well when the core region is extended to include half of the bottom and top plena.

In the reference case (Core) the graphite core region is included only. However, in the full-extended case (Ext-Full) all the bottom and top plena are included to the active fission region and in the half-extended case (Ext-Half) half of the lower and upper plena are considered for the point kinetics calculation. Fig. 4 shows the simulation results on the natural convection test of the MSRE. [6]

As shown in Fig. 4, the effect of the PK model on the predicted power is not significant. The results is

reasonable since the impact on the fuel transit time is small due to low flow velocity during natural convection. It should be noted that most of the delayed neutron precursors are decayed at the core region where they are generated.

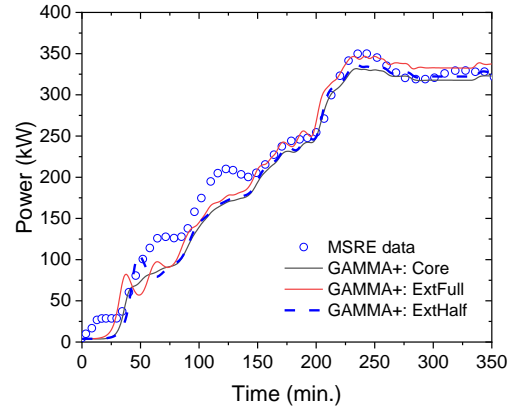


Fig. 4. GAMMA+ simulation results on the natural convection heat removal test of the MSRE: Sensitivity of PK models.

### 3.4 Heat transfer models in the GAMMA+ code

In the GAMMA+ code the heat transfer package includes gas mixture heat transfer correlations, two-phase heat transfer correlations, heat transfer enhancement devices, liquid metal heat transfer correlations and molten salt heat transfer correlations. Among them the molten salt heat transfer correlations include Sohal correlation [10] and Garon correlation [11] for forced and free convection conditions, respectively, in GAMMA+ 2.1 as follows.

However, it is estimated that the Garon correlation under-predicts the heat transfer under the molten salt conditions. The sensitivity of three correlations on the free convection condition, which are McAdams correlation [12], Garon correlation, and Yoder correlation [13], is investigated.

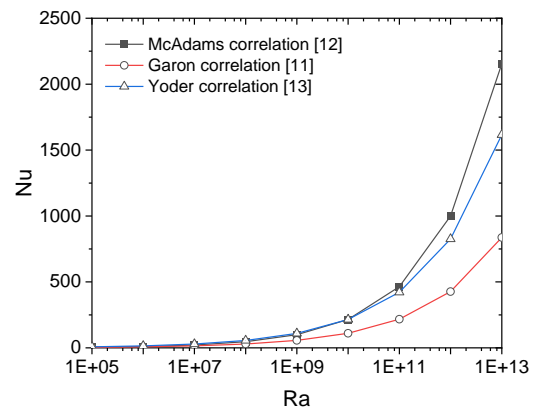


Fig. 5. The dependency of three correlations on Rayleigh number

The dependency of three correlations on Rayleigh number is expressed in Fig. 5, which shows that the values calculated from Yoder correlation is between those from McAdams and Garon correlations.

Four correlations are as follows.

$$\text{Sohal correlation: } Nu_N = 0.024 Re_j^{0.807} Pr_j^{0.301} \quad (8)$$

$$\text{Garon correlation: } Nu_N = 0.13 Ra_j^{0.293} \quad (9)$$

$$\text{McAdams correlation: } Nu_N = 0.10 Ra_j^{1/3} \quad (10)$$

$$\text{Yoder correlation: } Nu_N = 0.2601 Ra_j^{0.2918} \quad (11)$$

### 3.5 Sensitivity on the heat transfer models in GAMMA+

In the present study the natural convection test in MSRE are simulated using the three different free convection heat transfer models in the GAMMA+ code since there seems to be high dependency on the natural convection phenomena occurred in the molten salt circulating conditions. As shown in Fig. 6, the power simulated using the Yoder correlation agrees the test data better.

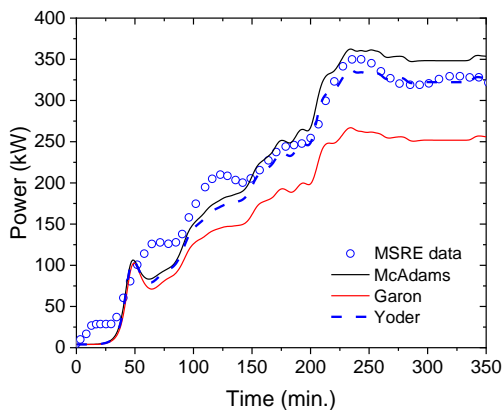


Fig. 6. GAMMA+ simulation results on the natural convection heat removal test of the MSRE: Sensitivity of heat transfer correlations.

## 4. Conclusions

In this paper, two sensitivity calculations on different active core regions and heat transfer models were performed against the natural convection heat removal test of the MSRE with the GAMMA+ code. First of all, the PK models of type 3 (NTK) was used and three different active core regions of core only, extended half (half of top and bottom plena included) and extended full (all the top and bottom plena included, or all the vessel). The impact of the selection of the active core was found to be small. Second, three different heat transfer correlations were used to simulate the natural convection test. The GAMMA+ code best simulate the MSRE test data best using the Yoder correlation.

## ACKNOWLEDGEMENT

This work was supported by Korea Research Institute for defense Technology planning and advancement(KRIT) grant funded by the Korea government(DAPA(Defense Acquisition Program Administration)) (KRIT-CT-22-017, Next Generation Multi-Purpose High Power Generation Technology(Liquid Fueled Heat Supply Module Design Technology), 2022)

## REFERENCES

- [1] R. C. Robertson, MSRE Design and Operations Report Part I, Description of Reactor Design, ORNL-TM-728, Oak Ridge National Laboratory, January 1965.
- [2] H. S. Lim, GAMMA+ 2.0 Volume II: Theory Manual, KAERI/TR-8662/2021, Korea Atomic Energy Research Institute, 2021.
- [3] N. I. Tak, et al., Simulation of Natural Convection in the Molten Salt Reactor Experiment Using GAMMA+, Transactions of the Korean Nuclear Society Autumn Meeting, Changwon, Korea, October 20-21, 2022.
- [4] S. H. Yoon, et al., Validation of GAMMA+ code using reactivity insertion tests of MSRE, Transactions of the Korean Nuclear Society Spring Meeting, Jeju, Korea, May 18-19, 2023.
- [5] R. C. Harvill, et al., Steady-State and Transient Benchmarks of GOTHIC to the Molten Salt Reactor Experiment, Nuclear Technology, Vol. 208, pp.70-99, January 2022.
- [6] M. W. Rosenthal, Molten-Salt Reactor Program Semiannual Progress Report for Period Ending February 28, 1969, ORNL-4396, Oak Ridge National Laboratory, August 1969.
- [7] O. W. Burke, F.H.S. Clark, Analysis of Transients in the MSRE system with 233U fuel, ORNL-4397, 1969.
- [8] D. L. Zhang, et al., Development of a steady state analysis code for a molten salt reactor, Annals of Nuclear Energy, Vol. 36, pp. 590-603, 2009.
- [9] 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.
- [10] M.S. Sohal, et al., Conceptual Design of Forced Convection Molten Salt Heat Transfer Testing Loop, INL/EXT-10-19908, 2010.
- [11] A.M. Garon, R.J. Goldstein, Velocity and Heat Transfer Measurements in Thermal Convection, Phys Fluids, 96, 1818-1825, 1973.
- [12] W.H. McAdams, Heat Transmission, 3rd edition, McGraw-Hill, New York, 1954.
- [13] G.L. Yoder, et al., Natural Convection Heat Transfer Experiments in Fluoride Salt, Journal of Heat Transfer, Transactions of the ASME, 140, No. 042501, 2018.