Research and Analytical Investigation of Criticality and Graphite Flow Channel Quantity in Molten Salt Reactors

Authors: Miao, Longxin, Yang, Dr. Zhen, Yin, Dejian, Yao, Mr. Yongkui, Yang, Dr. Zhen

Date: 2025-06-30T14:45:31+00:00

Abstract

The molten salt reactor, one of the six fourth-generation reactor types, employs liquid nuclear fuel and graphite as a moderator. Graphite also serves as a core structural component, forming fuel salt channels in a honeycomb-like configuration. To ensure effective core design management and nuclear safety oversight, it is essential to analyze the complex relationship among the graphite core structure, fuel salt channels, and critical mass. This study utilizes software-based simulation calculations to investigate the effects of varying the number of graphite flow channels and core dimensions on molten salt reactor reactivity by modeling the core size and graphite flow channels. The results demonstrate that the design dimensions of the graphite core are closely correlated with the total uranium mass in the reaction zone. Notably, the required critical total uranium mass decreases gradually as the number of fuel salt channels increases, eventually reaching a plateau followed by a slight decline. Based on the characteristics of graphite flow channels' influence on reactor reactivity, we summarize the regulatory requirements for graphite components from a legal perspective. Furthermore, we provide recommendations for improving manufacturing and installation quality, preventing accidental criticality incidents, and closely monitoring safety parameters from a regulatory standpoint. It is concluded that an optimal design value exists for the number of flow channels in the graphite components of liquid-fueled molten salt reactors with respect to reactivity.

Full Text

Research and Analytical Investigation on the Criticality of Molten Salt Reactors and the Quantity of Graphite Flow Channels

Longxin Miao[1], Zhen Yang[1],[2],[3]*, Dejian Yin[1], Yongkui Yao[1]

[1] North-Western China Regional Office of Nuclear and Radiation Safety Inspection, NNSA, Lanzhou 730020, China
[2] University of Chinese Academy of Sciences, Beijing 100049, China
[3] Shanghai Institute of Applied Physics, Chinese Academy of Sciences, Shanghai 201800, China

Abstract

The molten salt reactor, one of the six fourth-generation reactor designs, utilizes liquid nuclear fuel with graphite serving as both moderator and a core structural component that forms honeycomb-like fuel salt channels. Effective core design management and nuclear safety supervision require careful analysis of the intricate relationships among the graphite core structure, fuel salt channels, and critical mass. This study employs software simulation to analyze how varying numbers of graphite flow channels and core dimensions affect molten salt reactor reactivity through core modeling. The findings reveal that graphite core design dimensions correlate closely with total uranium mass in the reaction zone. Notably, the critical total uranium mass decreases gradually with increasing numbers of fuel salt channels, eventually plateauing before experiencing a slight decline. Drawing upon these characteristics of graphite flow channel influence on reactor reactivity, we summarize regulatory requirements for graphite components from a legal perspective and offer recommendations for enhancing manufacturing and installation quality, preventing accidental criticality events, and closely monitoring safety parameters from a regulatory standpoint. We conclude that an optimal design value exists for the number of flow channels in graphite components of liquid-fuel molten salt reactors from a reactivity perspective.

Keywords: Molten salt reactor, Fuel salt, Graphite, Nuclear criticality, Regulation

Introduction

The core of a liquid-fuel molten salt reactor differs fundamentally from those of existing light water reactors (LWRs), heavy water reactors (HWRs), and high-temperature gas-cooled reactors (HTGRs). Its core fuel consists of high-temperature liquid fuel molten salt that simultaneously serves as the primary circuit coolant, operates at atmospheric pressure, and uses graphite as a moderator [1,2]. Currently, only Oak Ridge National Laboratory (ORNL) in the

United States has achieved full-life-cycle operation of a molten salt reactor, specifically the Molten Salt Reactor Experiment (MSRE) [3]. The MSRE is an 8 MWt reactor where molten fluoride salts circulate through graphite channels at 1200 °F (648.89 °C) to demonstrate key molten salt reactor characteristics [4,5], including boundary material corrosion, graphite irradiation creep, off-gas treatment, negative temperature feedback, fuel salt pump reliability, redox potential equilibrium, fuel salt fluorination treatment, and operational verification of ^{235}U, ^{233}U, and ^{239}Pu [6]. Consequently, this research uses the MSRE as a reference.

The primary circuit fuel salt in its initial critical state comprises LiF-BeF_2-ZrF_4-UF_4 [7,8] with a molar ratio of 65%-29.1%-5%-0.9% [9], a melting point of 434 °C, and a density of 2.26 $g \cdot cm^{-3}$ [10]. By establishing a cylindrical core model with initial parameters for core diameter and height, analyzing reactivity changes from single-channel to 225-channel configurations, calculating critical fuel loading using the "golden section" iteration method, and simulating the effective multiplication factor (keff) with the SCALE6.1 (Standardized Computer Analyses for Licensing Evaluation) program, the results demonstrate that the total uranium mass required for criticality decreases sharply initially and then gradually stabilizes as the number of channels increases, with a slight increase observed at 16 channels. This proves that an optimal reactivity design value exists for the number of graphite flow channels. Meanwhile, safety recommendations are proposed from a regulatory perspective, providing a safety basis for small modular molten salt reactor design.

According to the Regulations of the People's Republic of China on the Control of Nuclear Materials and the practical requirements of commercial nuclear power projects, the maximum design value for ^{235}U mass abundance in civil nuclear materials is 20% [11]. Therefore, the ^{235}U mass abundance in the nuclear fuel salt for the following models is designed as 20% (corresponding to a molar enrichment of 20.2%), rather than the 33% enrichment used in the original MSRE design.

2 Modeling

The fuel salt temperature within the core is set at 650 °C, and the core is approximated as a cylindrical vessel model [12,13] with graphite filling it in a specific proportion [14]. After the reactor reaches stable operation, fuel salt circulation can carry delayed neutron precursors out of the core [15,16], whose influence on reactivity depends on fuel salt flow rate and the length of the extracore loop [17]. However, this research is simplified to focus solely on the initial fuel loading and critical conditions, where no delayed neutron precursors are generated, and the impact of loop fuel salt flow on delayed neutrons can be neglected.

2.1 Fuel Salt Parameters

Based on the above data, the number of atoms of each nuclide is calculated as follows:

$$0.65m_{LiF}N_A + 0.291m_{BeF_2}N_A + 0.05m_{ZrF_4}N_A + 0.009m_{UF_4}N_A = 2.26 \times 10^6$$

where N_A is Avogadro's constant (6.022$\times$10^{23}$ mol^{-1}), and m_{LiF}, m_{BeF_2}, m_{ZrF_4}, and m_{UF_4} are the molecular weights of LiF, BeF_2, ZrF_4, and UF_4, respectively. The total number of cations N_0 is calculated to be approximately 3.25958$\times$10^{28}$ m^{-3}. The number of nuclides of each element in the homogeneous mixture medium per unit volume is shown in Table 1 .

The sum of the densities of all anions and cations in the fuel salt is:

$$N_{total} = \sum N_i = 8.04464 \times 10^{28}$$

2.2 Core Geometric Parameters

The MSRE core diameter is 54 inches (1.372 m), and the moderator zone height is 64 inches (1.626 m). It is constructed with graphite rods of 2-inch (0.051 m) square cross-section. As shown in Figure 1 [Figure 1: see original paper], 260 graphite rods can be placed [18,19]. The mass of fuel salt loaded in the core is calculated as follows:

$$V_{total} = \pi \left(\frac{D}{2}\right)^2 H = 3.14 \times 1.626 \times \left(\frac{1.372}{2}\right)^2 = 2.40 \text{ m}^3$$

$$V_{fuel\ salt} = \pi \left(\frac{D}{2}\right)^2 H - na^{2H} = 1.626 \times \left(3.14 \times \left(\frac{1.372}{2}\right)^2 - 260 \times 0.051^2\right) = 1.30 \,\text{m}^3$$

$$m_{fuel\ salt} = V_{fuel\ salt}\rho = 1.3 \times 2.26 \times 10^3 = 2945 \text{ kg}$$

According to neutron moderation and reflection characteristics and safety conservatism, it is assumed that the graphite is seamlessly attached to the inner wall of the reactor vessel, and liquid fuel salt can only pass through internal channels surrounded by graphite [20].

First, referring to the molten salt mass in the MSRE moderator zone [21], reactivity changes are analyzed for configurations with 1 channel (radius 50.5 cm), 4 channels (radius 25.24 cm), 9 channels (radius 16.83 cm), 16 channels (radius 12.62 cm), 25 channels (radius 10.1 cm), 100 channels (radius 5.0 cm), and 225 channels (radius 3.4 cm). For convenient grid modeling and multi-channel cross-section calculation, the core size is fixed at a diameter of 2 m (the original MSRE core diameter was 1.372 m) and a height of 1.626 m. The fuel salt loading volume of the core is 1.3 m^3, with a corresponding loading mass of 2945 kg. The cross-sectional area of the fuel salt channel in the core is 0.80 m^2. As shown in Figures 2 [Figure 2: see original paper] and 3 [Figure 3: see original paper], which present top views of the core cross-section, the black rings represent graphite and the inner circles represent fuel salt channels. The channels do not overlap in the cross-section.

Without changing the fuel salt ratio, the diameter and height are proportionally varied based on the aforementioned core size as the initial value, and the "golden section" method is used for point-by-point iterative approximation calculation, as shown in Figure 4 [Figure 4: see original paper], to obtain the loading amount under critical conditions.

The graphite density [22] is 1.6\$×10^{3}\$ kg/m^3. Lithium is assumed to be pure, meaning the ^{7}Li abundance is 100%.

The SCALE6.1 calculation tool is employed. The SCALE6.1 program system is a standardized computer analysis package for licensing evaluation developed and maintained by Oak Ridge National Laboratory in the United States. It is primarily used for nuclear reactor physics calculations, critical safety analyses, and radiation shielding calculations, making it suitable for reactor criticality calculations. The effective multiplication factor keff is calculated, and the parameters in Table 2 are obtained.

3.1 Single-Channel Core

SCALE6.1 calculations show that for a core size with 1.372 m diameter, 1.626 m height, and a channel radius of 0.5046 m, keff is 0.6809. For the designed core and fuel ratio, 20% enriched ^{235}U is insufficient to achieve criticality. The iterative proportional approximation calculation for this core is shown in Table 3 .

If the initial core diameter is 1.372 m, a single-channel graphite core with 20% enriched ^{235}U requires proportional iteration to achieve criticality at a core height of 3.07 m, radius of 1.3 m, and fuel salt channel radius of 0.95 m—representing 1.89 times the original designed dimensions. The total loading volume of the active zone is 8.78 m^3, the total fuel salt mass is 19,850 kg, and the total uranium mass is 1014 kg.

Therefore, the decision not to use the original MSRE core diameter of 1.372 m is reasonable. Similarly, when the core size is 2 m diameter, 1.626 m height, and channel radius of 0.5046 m, keff is 0.87483. For the core and fuel ratio designed in this paper, 20% enriched ^{235}U remains insufficient for criticality. The iterative approximation calculation is shown in Table 4 .

For a single-channel graphite core with 20% enriched ^{235}U, the critical core height is 2.185 m, the radius is 1.344 m, and the fuel salt channel radius is 0.678 m—1.344 times the set initial core size. The active zone loading volume is 3.153 m^3, the total fuel salt mass is 7127 kg, and the total uranium mass is 364 kg.

3.2 Core with 4 Flow Channels

Calculations show that for a core size of 2 m diameter, 1.626 m height, and each flow channel radius of 0.2524 m, keff is 1.01. For the initial core size and fuel ratio, 20% enriched ^{235}U yields supercriticality. The iterative approximation calculation is shown in Table 5 .

For a 4-channel graphite core with 20% enriched ^{235}U, the critical core height is 1.6 m, the radius is 0.985 m, and the fuel salt flow channel radius is 0.249 m—0.985 times the set initial core size. The total active zone loading volume is 1.24 m^3, the total mass is 2810 kg, and the total uranium mass is 144 kg.

3.3 Core with 9 Flow Channels

Calculations show that for a core size of 2 m diameter, 1.626 m height, and each flow channel radius of 0.168 m, keff is 1.056. For the initial core size and fuel ratio, 20% enriched ^{235}U yields supercriticality. The iterative approximation calculation is shown in Table 6 .

For a 9-channel graphite core with 20% enriched ^{235}U, the critical core height is 1.51 m, the radius is 0.928 m, and the fuel salt flow channel radius is 0.156 m—0.928 times the set initial core size. The total active zone loading volume is 1.04 m^3, the total mass is 2350 kg, and the total uranium mass is 120 kg.

3.4 Core with 16 Flow Channels

Calculations show that for a core size of 2 m diameter, 1.626 m height, and each flow channel radius of 0.126 m, keff is 1.089. For the initial core size and fuel ratio, 20% enriched ^{235}U yields supercriticality. The iterative approximation calculation is shown in Table 7 .

For a 16-channel graphite core with 20% enriched ^{235}U, the critical core height is 1.46 m, the radius is 0.895 m, and the fuel salt flow channel radius is 0.113 m—0.895 times the set initial core size. The total active zone loading volume is 0.93 m^3, the total mass is 2110 kg, and the total uranium mass is 108 kg.

3.5 Core with 25 Flow Channels

Calculations show that for a core size of 2 m diameter, 1.626 m height, and each flow channel radius of 0.101 m, keff is 1.085. For the initial core size and fuel ratio, 20% enriched ^{235}U yields supercriticality. The iterative approximation calculation is shown in Table 8 .

For a 25-channel graphite core with 20% enriched ^{235}U, the critical core height is 1.46 m, the radius is 0.9 m, and the fuel salt flow channel radius is 0.091 m—0.899 times the set initial core size. The total active zone loading volume is 0.94 m^3, the total mass is 2130 kg, and the total uranium mass is 109 kg.

3.6 Core with 100 Flow Channels

Calculations show that for a core size of 2 m diameter, 1.626 m height, and each flow channel radius of 0.051 m, keff is 1.086. For the initial core size and fuel ratio, 20% enriched ^{235}U yields supercriticality. The iterative approximation calculation is shown in Table 9 .

For a 100-channel graphite core with 20% enriched ^{235}U, the critical core height is 1.46 m, the radius is 0.9 m, and the fuel salt flow channel radius is 0.046 m—0.901 times the set initial core size. The total active zone loading volume is 0.95 m^3, the total mass is 2150 kg, and the total uranium mass is 110 kg.

3.7 Core with 225 Flow Channels

Calculations show that for a core size of 2 m diameter, 1.626 m height, and each flow channel radius of 0.034 m, keff is 1.075. For the initial core size and fuel ratio, 20% enriched ^{235}U yields supercriticality. The iterative approximation calculation is shown in Table 10 .

For a 225-channel graphite core with 20% enriched ^{235}U, the critical core height is 1.48 m, the radius is 0.91 m, and the fuel salt flow channel radius is 0.031 m—0.909 times the set initial core size. The total active zone loading volume is 0.98 m^3, the total mass is 2210 kg, and the total uranium mass is 113 kg.

4 Results and Discussion

This study calculated parameters including core size and total uranium mass required for criticality under conditions of 650 °C core temperature, 100% ^{7}Li abundance, and 20% ^{235}U abundance for graphite cores with various numbers of flow channels. Results for configurations ranging from single-channel to 225 channels show that the total uranium mass required for criticality decreases sharply initially and then gradually slows with increasing channel numbers. The single-channel configuration requires the largest uranium mass, which decreases to 144 kg at 4 channels and further drops to 108 kg at 16 channels, after which a slight increase occurs and the mass stabilizes at 109–113 kg for configurations up to 225 channels. Core dimensions (diameter, height) and fuel salt volume gradually decrease as the number of flow channels increases.

The single-channel core exhibits the largest volume. At 16 channels, the core radius and height are reduced to 89.5% and 89.8% of the initial size, respectively, compressing the volume to 29.5% of the single-channel core, after which dimensional changes stabilize. In terms of reactivity, the initial keff of the single-channel core is less than 1, indicating subcriticality, requiring significant core enlargement to achieve criticality. Initial keff values for cores with more than 4 channels are supercritical, though reactivity gains gradually saturate with increasing channel numbers.

The relationship between critical total uranium mass and the number of graphite flow channels is illustrated in Figure 5 [Figure 5: see original paper], with the following analytical characteristics: (1) The core can be made very compact, with height and diameter within 2 m, active zone loading of about 3000 kg or less, and total uranium mass of about 150 kg or less, making it more suitable for small modular construction compared to conventional water reactors. (2) For this reactor type, due to limitations of graphite moderation and neutron distribution, except for the impractically large single-channel critical volume, multi-channel configurations show similar critical volumes and masses. However, as the number of graphite flow channels increases, manufacturing complexity and cost also rise. Therefore, the optimal number and distribution of graphite core flow channels can be designed based on cost and other factors. (3) For the core designed in this paper, when the number of flow channels exceeds 16, the critical mass of nuclear fuel increases slightly, but the total uranium mass in the core active zone essentially stabilizes at approximately 110 kg.

Consequently, when designing fuel salt flow channels for this core, a range of channel numbers exists that provides stable reactivity influence, which can reduce reactivity deviations caused by graphite damage during construction and operation.

4.1 Discussion on Regulatory Strategies

The relationship between reactivity changes and the number of graphite flow channels provides valuable reference for liquid-fuel molten salt reactor design and nuclear safety supervision. As a novel reactor type, establishing a nuclear safety experience feedback system and emphasizing experimental work during operation and commissioning stages is essential. (1) Core graphite, serving as a crucial in-core component functioning as both moderator and fuel salt channel, should be assigned the highest nuclear safety level in molten salt reactor design. Manufacturing and installation quality must be strengthened to prevent damage during graphite installation in the construction phase and wear during reactor operation, which could affect reactivity control and operational lifespan. Graphite in molten salt reactors should be classified as safety-class components, which is also the classification used in Chinese regulatory practice for molten salt reactors. According to the Nuclear Safety Law of the People's Republic of China, if a molten salt reactor operator adjusts design, manufacturing, or installation parameters such as the diameter, number, or spacing distribution of core graphite flow channels, such changes must be reported and approved as required. (2) According to the Regulations on the Supervision and Administration of Civil Nuclear Safety Equipment, design, manufacturing, and installation units for core graphite must immediately take measures to address major quality issues in their activities and report them promptly. Greater attention should be paid to graphite components due to their specialized installation technology, vulnerability to collision damage, and high cleanliness requirements. (3) When calculating reactor core critical parameters, the total uranium mass in the core active region is predominantly considered. In practical regulatory oversight, for a fixed-core reactor, monitoring the concentration of ^{235}U in the fuel salt suffices to determine its keff. Consequently, alterations in reactor core structure may cause reactivity to deviate from design specifications. According to the Regulations on Nuclear Safety Reporting for Research Reactor Operating Organizations, research reactor operators must report operational events to the National Nuclear Safety Administration during the operational phase, including collapse of graphite components within the reactor core, notable radiation-induced swelling, functional degradation, or damage.

4.2 Discussion on Safety Recommendations

For the novel graphite-moderated liquid-fuel molten salt reactor, the following nuclear safety-related recommendations are proposed: (1) According to the above calculations, when approaching criticality at keff values of 0.995–0.997, the reactor is only about 1 kg of fuel salt away from the critical state. Since the error in loading fuel salt through a quantitative tank (several kilograms) is relatively large, operators face significant risks when attempting to achieve criticality by continuously adding fuel salt at this stage. To ensure proper supervision of the approach-to-criticality control point, adopting more precise reactivity control measures to achieve criticality is recommended. (2) When a molten salt reactor approaches criticality, the reactivity to be introduced is very small. At this stage, monitoring process parameters such as neutron count, fuel salt temperature, and flow rate is necessary to prevent accidental criticality. Additionally, radiation protection management for personnel in the control area must be strengthened to prevent excessive neutron irradiation of personnel caused by accidental criticality.

5 Conclusion

Liquid-fuel molten salt reactors primarily rely on graphite moderation, with fuel salt flowing through graphite component channels and forming a circulation loop with fuel salt channels outside the core. Chain fission reactions occur within the graphite component flow channels. When the number of graphite component flow channels is 1, the total uranium mass required for criticality in the active zone is the largest, significantly exceeding that required for other channel configurations. The total uranium mass required for criticality then decreases sharply as the number of flow channels increases. When the number reaches 16, the total uranium mass required for criticality in the active zone begins to increase slightly, though the upward trend is relatively gentle. The total uranium mass required for criticality is proportional to core size.

Therefore, for liquid-fuel molten salt reactors, the number of flow channels in graphite components can be optimized to minimize the uranium loading required for criticality and reduce core design dimensions.

References

1. M. S. Cheng, Z. M. Dai. Preliminary safety analysis of molten salt breeder reactor [J]. Nuclear Techniques, 2013, 36(6): 060601. doi: 10.11889/j.0253-3219.2013.hjs.36.060601 (in Chinese)

2. X. Z. Cai, Z. M. Dai, H. J. Xu, Thorium molten salt reactor nuclear energy system[J]. Physics, 2016, 45(9): 578-590. doi: CNKI:SUN:WLZZ.0.2016-09-004 (in Chinese)

3. M. H. Jiang, H. J. Xu, Z. M. Dai. Advanced Fission Energy Program-TMSR Nuclear Energy System [J]. Bulletin of Chinese Academy of Sciences, 2012, 27(3): 366-374. doi: 10.3969/j.issn.1000-3045.2012.03.016 (in Chinese)

4. G. M. Sun, M. S. Cheng, Z. M. Dai. Preliminary analysis of fuel management for a small modular molten salt fast reactor [J]. Nuclear Techniques, 2016, 39(7): 070603. doi: 10.11889/j.0253-3219.2016.hjs.39.070603 (in Chinese)

5. X. D. Zuo, M. S. Cheng, Y. Q. Dai et al., Flow field effect of delayed neutron precursors in liquid-fueled molten salt reactors[J].Nucl. Sci. Tech., 2022, 33(8):1-17.doi:10.1007/s41365-022-01084-0

6. J. H. Wang, M. S. Cheng, Z. M. Dai. Design and analysis of adaptive power control system for solid fuel molten salt reactor [J]. Nuclear Techniques, 2017, 40(9): 090602. doi: 10.11889/j.0253-3219.2017.hjs.40.090602 (in Chinese)

7. S. H. Yu, Y. F. Liu, P. Yang et al., Neutronics analysis for MSR cell with different fuel salt channel geometries [J]. Nucl. Sci. Tech., 2021, 32(1): 9. doi: 10.1007/s41365-020-00844-0

8. Z. H. Zhang, X. B. Xia, X. W. Zhu et al., Source terms calculation for the MSRE with on-line removing radioactive gases[J]. Nuclear techniques, 2014, 37(2): 020603 doi: 10.11889/j.0253-3219.2014.hjs.37.020603 (in Chinese)

9. H. H. Yu, Y. F. Liu , P,Yang, et al,. Neutronics analysis for MSR cell with different fuel salt channel geometries[J].Nucl. Sci. Tech.,2021,32(1):9. doi:10.1007/s41365-020-00844-0

10. S. H. Yu, Y. F. Liu, P. Yang et al., Effect analysis of core structure changes on reactivity in molten salt experimental reactor [J]. Nuclear Techniques, 2019, 42(2): 020603. doi: 10.11889/j.0253-3219.2019.hjs.42.020603 (in Chinese)

11. Center for Nuclear and Radiation Safety, Ministry of Environmental Protection. Laws and regulations related to nuclear safety[M]. Revise ed. Beijing: China Atomic Energy Press,2018 (in Chinese)

12. Haubenreich P N, Engel J R. Experience with the molten-salt reactor experiment[J]. Nucl. Appl. Technol., 1970, 8(2): 118-136. doi: 10.13182/nt8-2-118

13. Robertson R C. MSRE design and operation report Part I: description of reactor design[R]. ORNL, 1965. doi: 10.2172/4654707

14. S. H. Yu, X. X. Li, Y. F. Liu et al,. Study on core physical parameters for molten salt experimental reactor[J]. Nuclear Techniques, 2019,42(03):1-6. doi: 10.11889/j.0253-3219.2019.hjs.42.030604 (in Chinese)

15. Z. Yang, Z. M. Dai, Yang Zou et al,. Nuclear safety analysis of molten salt reactor criticality with changes in 7Li abundance[J]. Nuclear Techniques, 2024,47(12):120603. doi:10.11889/j.0253-3219.2024.hjs.47.120603 CSTR: 32193.14.hjs.CN31-1342/TL.2024.47.120603 (in Chinese)

16. J. Cai, X. B. Xia, K.Chen et al,. Study of fluid fuel influence on delayed neutron in Molten Salt Reactor[J]. Nuclear techniques, 2014, 37(3): 030603 doi: 10.11889/j.0253-3219.2014.hjs.37.030603 (in Chinese)

17. X. X. Li,Y. W. Ma,C. G. Yu et al,. Effects of fuel salt composition on fuel salt temperature coefficient (FSTC) for an under-moderated molten salt reactor (MSR)[J]. Nucl. Sci. Tech., 2018, 29(8): 110. doi: 10.1007/s41365-018-0458-1

18. Q. Yang,S. H. Yu, Y, Zhou et al,. Quantitative similarity analysis of the fluoride salt-cooled pebble bed reactor with HTR-10 and MSRE [J]. Nuclear Techniques, 2016, 39(1): 010601. doi: 10.11889/j.0253-3219.2016.hjs.39.010601 (in Chinese)

19. Y. J. Wang,Y. X. Qu,H. Y. Fu et al,. Separation of rare earth fission products from LiF-BeF2 molten salt by sulfide precipitation[J]. Nuclear Techniques,2024,47(1):010302. doi: 10.11889/j.0253-3219.2024.hjs.47.010302 (in Chinese)

20. P.Yang,Y. Dai,Y. Zou et al,. Application of global variance reduction method to calculate a high-resolution fast neutron flux distribution for TMSR-SF1[J]. Nucl. Sci. Tech., 2019, 30(8): 125. doi: 10.1007/s41365-019-0650-y

21. X. X. Li, D. Y. Cui, Y. W. Ma et al,. Influence of 235U enrichment on the moderator temperature coefficient of reactivity in a graphite-moderated molten salt reactor[J]. Nucl. Sci. Tech., 2019, 30(11): 166. doi: 10.1007/s41365-019-0694-z

22. Z. S. Xie. Nuclear reactor physics analysis[M].Xi'an: Xi'an Jiao Tong University Press, 2004. (in Chinese)