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Abstract

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Full Text

Preamble

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Development of a Three-Dimensional Multi-Physics Code for Molten Salt Fast Reactor

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The Molten Salt Reactor (MSR) was selected as one of the six innovative nuclear reactors by the Generation IV International Forum (GIF). The circulating fuel in the can-type molten salt fast reactor makes the neutronics and thermohydraulics of the reactor strongly coupled and different from that of traditional solid-fuel reactors.

In the present paper, a new coupling model is presented that physically describes the inherent relations between the neutron flux, the delayed neutron precursor, the heat transfer, and the turbulent flow. Based on this model, integrating nuclear data processing, CAD modeling, structured and unstructured mesh technology, data analysis and visualization application, a three-dimensional steady-state simulation code system (MSR3DS) for the can-type molten salt fast reactor is developed and validated. In order to demonstrate the ability of the code, the three-dimensional distributions of the velocity, the neutron flux, the delayed neutron precursor, and the temperature were obtained for the simplified MOlten Salt Advanced Reactor Transmuter (MOSART) using this code. The results indicate that the MSR3DS code can provide a feasible description of multi-physical coupling phenomena in can-type molten salt fast reactor. Furthermore, the code can well predict the flow effect of fuel salt and the transport effect of the turbulent diffusion.

Keywords: Molten salt fast reactor, Turbulent model, Delayed neutron precursor, Neutronics, Thermohydraulics, Turbulent diffusion DOI: 10.13538/j.1001-8042/nst.25.010601

Introduction

The first molten salt reactor (MSR) was developed in the late 1940s for aircraft propulsion at Oak Ridge National Laboratory (ORNL). The Aircraft Reactor Experiment (ARE) [1] operated successfully in 1954. In 1965, the Molten Salt Reactor Experiment (MSRE) [2] went critical, and after six months of successful operation the enriched U-235 was removed and replaced by denatured U-233 as fissile fuel. In October 1966, the MSRE was the first reactor to reach criticality with U-233. A detailed 1000 MWe engineering conceptual design of a Molten Salt Breeder Reactor (MSBR) [3] was developed in the 1970s. Even though the concept looked promising, the studies were stopped in 1976. MSR has the potential to meet the goals of Generation IV reactors and high-level waste transmutation programs, offering advantages over solid fuel reactors.

Consequently, many countries are interested in developing new MSR concepts, such as the FUJI series [4], Actinides Molten Salt TransmutER (AMSTER)

[5], MOSART [6], Fast Spectrum Molten Salt Reactor (FS-MSR) [7], European Molten Salt Fast Reactor (MSFR) [8], and Small Mobile Molten Salt Reactor (SM-MSR) [9]. In 2010, the Chinese Academy of Sciences restarted the thorium molten salt reactor project and established the Center for Thorium Molten Salt Reactor System (CTMSRS) at the Shanghai Institute of Applied Physics, and CTMSRS completed the conceptual design of a Liquid Fuel Thorium Molten Salt Reactor (LF-TMSR).

The Can-type Molten Salt Fast Reactor (CMSFR) can be employed to consume actinides from light water reactor (LWR) fuel or, alternatively, to extend fissile resource availability through U/Pu and Th/U breeding. CMSFRs are highly flexible and can be configured into modified open or full-recycle configurations. Since 2005, the GIF has focused on CMSFR rather than thermal spectrum molten salt reactors. Many works have been conducted to investigate the complex behavior of the CMSFR. Wang et al. [10] extended the SIMMER-III code for simulating MOSART with additional thermohydraulic and neutronic modules in two-dimensional (2D) axial-symmetric geometry. Nicolino et al. [11] developed a new approach to describe the strong coupling between neutronics and thermofluid dynamics with particular focus on the MOSART molten salt fast reactor in 2D axial-symmetric geometry. Zhang et al. [12] provided a theoretical model coupling flow, heat transfer, and neutronics in 2D axial-symmetric geometry and calculated the steady-state characteristics of a molten salt fast reactor without graphite moderator in the core.

In all these studies, the transport effect of turbulent diffusion was not taken into account for the delayed neutron precursor (DNP) concentration. In this paper, we present a new model to describe the inherent relations between neutronics and thermohydraulics. Based on this model, a three-dimensional steady-state simulation code (MSR3DS) for the molten salt fast reactor is developed and validated. Using the multi-physical coupling code, the three-dimensional distributions of velocity, neutron fluxes, DNPs, and temperature are obtained for a simplified MOSART.

Mathematical Model

More recently, several fluoride molten salt fast reactors have been proposed as part of the Gen IV program. These include the European MOSART reactor using LiF/NaF/BeF_2/(TRU)F_3 as a fuel salt, and a European MSFR concept using LiF/NaF/(TRU)F_3 or LiF/NaF/(U+Th)F_4. In this paper, to establish the theoretical model and develop the simulation code, a core configuration shown in Fig. 1 [Figure 1: see original paper] is adopted based on the core configuration of MOSART. The main parameters of the core are listed in Table 1 . The mass proportions at equilibrium in a finite critical core with 20 cm graphite reflector are given in Table 2 . Table 3 lists six-group delayed neutron fractions and the precursor decay constants.

Thermohydraulics Model

The fuel salt flow in the molten salt fast reactor without moderator in the core (like MOSART and MSFR) is considered turbulent flow. Their Reynolds numbers in the core region can exceed 10^5. Many models can be used to describe turbulent flow. The most accurate method is direct numerical simulation (DNS), which computes the mean flow and all turbulent velocity fluctuations. However, DNS is highly costly in terms of computing resources, so this method is not used for industrial flow computations. Large eddy simulation is an intermediate form of turbulence calculations that tracks the behavior of larger eddies. The effects on the resolved flow due to the smallest, unresolved eddies are included by means of a sub-grid scale model in the large eddy method. Consequently, the demands on computing resources in terms of storage and volume of calculations are large.

The Reynolds-averaged Navier-Stokes (RANS) equations focus on the mean flow and the effects of turbulence on mean flow properties. The computing resources required for reasonably accurate flow computations are modest, making this approach the mainstay of engineering flow calculations in engineering practice.

In this study, the flow, heat transfer, and turbulent characteristics in the core were obtained by solving the following three-dimensional, incompressible, steady-state Navier-Stokes equations and turbulence model. The RANS equations with Boussinesq's closure hypothesis and the standard k- turbulence model [13] were adopted.

The continuity equation is:

$$\nabla \cdot \mathbf{U} = 0$$

The momentum conservation equation is:

$$\rho_{fs}(\mathbf{U} \cdot \nabla)\mathbf{U} = -\nabla \cdot \left(\rho_{fs} k\mathbf{I} + \nabla \cdot \left[(\mu + \mu_t)\left(\nabla\mathbf{U} + (\nabla\mathbf{U})^T\right)\right]\right)$$

The transport equations for k and in the standard k- turbulence model are:

$$\rho_{fs}\mathbf{U} \cdot \nabla k = \nabla \cdot \left(\frac{\mu_t}{\sigma_k}\nabla k\right) + G - \rho_{fs}\varepsilon$$

$$\rho_{fs}\mathbf{U} \cdot \nabla \varepsilon = \nabla \cdot \left(\frac{\mu_t}{\sigma_\varepsilon}\nabla \varepsilon\right) + C_{\varepsilon 1}\frac{\varepsilon}{k}G - C_{\varepsilon 2}\rho_{fs}\frac{\varepsilon^2}{k}$$

where $\mathbf{U}$ represents the mean velocity vector, and k, ε, ρ_{fs}, μ, μ_t, and $\mathbf{I}$ are turbulent kinetic energy, turbulent dissipation rate, fuel salt density, dynamic viscosity, turbulent dynamic viscosity, and identity matrix, respectively. The equations contain five constants: $C_{\varepsilon 1}$, $C_{\varepsilon 2}$, C_μ, σ_k, and σ_ε. The standard k- turbulence model employs values for these constants that are determined by comprehensive data fitting for a wide range of turbulent flows.

The energy conservation equation of the fuel salt expressed by temperature:

$$\rho_{fs} c_{p,fs} \nabla \cdot (\mathbf{U} T_{fs}) = \nabla \cdot \left[\lambda_{T,fs} + c_{p,fs} \frac{\mu_t}{Pr_t} \right] \nabla T_{fs} + s_{fs}$$

The energy conservation equation of the reflector expressed by temperature:

$$\nabla \cdot \lambda_{T,ref} \nabla T_{ref} + s_{ref} = 0$$

The inner heat sources of fuel salt (s_{fs}) and the reflector (s_{ref}) in the equations are calculated using the neutron fission reactions:

$$s_{fs} = \gamma E_f \sum_g (\phi_g \cdot \Sigma_{f,g})$$

$$s_{ref} = (1-\gamma) E_f \sum_g (\phi_g \cdot \Sigma_{f,g})$$

where T_{fs}, T_{ref}, $\lambda_{T,fs}$, $\lambda_{T,ref}$, $c_{p,fs}$, and Pr_t respectively represent the temperature of the fuel salt and the reflector, thermal conductivity of the fuel salt and the reflector, specific heat capacity of the fuel salt, and turbulent Prandtl number; E_f, ϕ_g, $\Sigma_{f,g}$, and γ are energy released from each fission reaction, neutron flux for group g, fission cross-section for group g, and the fraction of power released into fuel molten salt, respectively.

Neutronics Model

According to the basic conservation of neutron number and DNP concentration in a control volume with multi-group diffusion theory, the neutronics model of CMSFR can be derived.

The diffusion equation of the neutron for group g [11]:

$$\nabla \cdot [\mathbf{U} \phi_g(\mathbf{r},t)] = \nabla \cdot D_{n,g}(\mathbf{r}) \nabla \phi_g(\mathbf{r},t) - \Sigma_{r,g}(\mathbf{r}) \phi_g(\mathbf{r},t) + \chi_{p,g}(1-\beta) \sum_{g'} (\nu\Sigma)_{f,g'}(\mathbf{r}) \phi_{g'}(\mathbf{r},t) + \sum_i \chi_{d,i,g} \lambda_i C_i(\mathbf{r},t) + \sum_{g' \neq g}$$

The delayed neutron precursors are classified into six groups by half-life periods. The balance equation of the DNP concentration for group i:

$$\frac{\partial C_i(\mathbf{r},t)}{\partial t} + \nabla \cdot [\mathbf{U} C_i(\mathbf{r},t)] = \nabla \cdot \left(D_{c,i} + \frac{\mu_t}{\rho_{fs} Sc_t} \right) \nabla C_i(\mathbf{r},t) + \beta_i \sum_{g'} (\nu\Sigma)_{f,g'}(\mathbf{r}) \phi_{g'}(\mathbf{r},t) - \lambda_i C_i(\mathbf{r},t)$$

The second terms in the equations indicate the flow effect of the fuel molten salt on the neutron fluxes and DNPs. The third term in the DNP equation includes two parts: the $\nabla \cdot D_{c,i} \nabla C_i(\mathbf{r},t)$ part represents molecular diffusion of DNP group i, and the $\nabla \cdot \left(\frac{\mu_t}{\rho_{fs} Sc_t} \right) \nabla C_i(\mathbf{r},t)$ part represents turbulent diffusion

of DNP group i. In liquids and for turbulent flow, molecular diffusion can be neglected compared to turbulent diffusion.

In the above equations, u_g, C_i, $D_{n,g}$, $D_{c,i}$, β_i, β, ν, $\Sigma_{g'\to g}$, $\chi_{p,g}$, $\chi_{p,i,g}$, and Sc_t respectively represent neutron mean velocity for group g, the DNP for group i, neutron diffusion coefficient for energy group g, DNP diffusion coefficient for group i, the fraction of delayed neutron for group i, the total fraction of delayed neutron, the average number of neutrons produced in energy group g, scatter cross section from energy group g' to group g, the fission spectrums of prompt neutron for group g, the fission spectrums of delayed neutron for energy group g and delayed neutron group i, and turbulent Schmidt number.

In steady-state conditions, the time-dependent terms can be removed and the effective multiplication factor k_{eff} is introduced. Therefore, the 3D steady-state neutronics model for molten salt fast reactor can be obtained.

The neutron diffusion equations in the fuel salt:

$$\nabla\cdot[\mathbf{U}\phi_{1,fs}(\mathbf{r})] = \nabla\cdot D_{n,1,fs}(\mathbf{r})\nabla\phi_{1,fs}(\mathbf{r}) - \Sigma_{r,1,fs}(\mathbf{r})\phi_{1,fs}(\mathbf{r}) + \frac{\chi_{p,1}(1-\beta)}{k_{eff}}\sum_{g'}(\nu\Sigma)_{f,g',fs}(\mathbf{r})\phi_{g',fs}(\mathbf{r}) + \sum_i \lambda_i C_i(\mathbf{r}) +$$

$$\nabla\cdot[\mathbf{U}\phi_{2,fs}(\mathbf{r})] = \nabla\cdot D_{n,2,fs}(\mathbf{r})\nabla\phi_{1,fs}(\mathbf{r}) - \Sigma_{r,1,fs}(\mathbf{r})\phi_{1,fs}(\mathbf{r}) + \Sigma_{1\to 2,fs}(\mathbf{r})\phi_{1,fs}(\mathbf{r})$$

$$\nabla\cdot[\mathbf{U}\phi_{g}(\mathbf{r})] = \nabla\cdot D_{n,g}(\mathbf{r})\nabla\phi_{g}(\mathbf{r}) - \Sigma_{r,g}(\mathbf{r})\phi_{g}(\mathbf{r}) + \frac{\chi_{p,g}(1-\beta)}{k_{eff}}\sum_{g'}(\nu\Sigma)_{f,g'}(\mathbf{r})\phi_{g'}(\mathbf{r}) + \sum_i \chi_{d,i,g}\lambda_i C_i(\mathbf{r}) + \sum_{g'\neq g}\Sigma_{g'\to g}($$

The transport equations for six-group DNPs in the fuel salt:

$$\nabla\cdot[\mathbf{U}C_i(\mathbf{r})] = \nabla\cdot\left(D_{c,i} + \frac{\mu_t}{\rho_{fs}Sc_t}\right)\nabla C_i(\mathbf{r}) + \frac{\beta_i}{k_{eff}}\sum_{g'}(\nu\Sigma)_{f,g'}(\mathbf{r})\phi_{g'}(\mathbf{r}) - \lambda_i C_i(\mathbf{r})$$

The neutron diffusion equations in the reflector:

$$\nabla\cdot D_{n,1,ref}(\mathbf{r})\nabla\phi_{1,ref}(\mathbf{r}) - \Sigma_{r,1,ref}(\mathbf{r})\phi_{1,ref}(\mathbf{r}) = 0$$

$$\nabla\cdot D_{n,2,ref}(\mathbf{r})\nabla\phi_{1,ref}(\mathbf{r}) - \Sigma_{r,2,ref}(\mathbf{r})\phi_{1,ref}(\mathbf{r}) + \Sigma_{1\to 2,ref}(\mathbf{r})\phi_{1,ref}(\mathbf{r}) = 0$$

The effective multiplication factor k_{eff} can be computed from:

$$k_{eff} = \frac{\iiint \sum_g (\nu\Sigma_f)_{g,fs}\phi^n_{g,fs}dV}{\iiint \sum_g (\nu\Sigma_f)_g\phi^{n-1}_g dV}$$

where the subscripts fs and ref respectively represent the fuel salt and the reflector; $\Sigma_{r,1,fs}$, $\Sigma_{r,2,fs}$, $\Sigma_{r,1,ref}$, and $\Sigma_{r,2,ref}$ represent the removal cross sections, and the superscripts n and $n-1$ represent the iteration number.

In this work, the steady neutronics model consists of two-group ($G = 2$) neutron diffusion equations for fast and thermal neutron fluxes in the fuel molten salt and reflector, and transport equations for six-group ($I = 6$) DNPs in the fuel salt.

Boundary Conditions

Thermohydraulics

1. **Inlet boundary**: At the inlet, the velocity, turbulent kinetic energy, turbulent dissipation rate, and temperature are specified:

$$u_z = u_{inlet}, \quad k = \frac{3}{2}(I u_{inlet})^2, \quad \varepsilon = C_\mu^{0.75}\frac{k^{1.5}}{l}, \quad T = T_{inlet}$$

   where u_{inlet} is given as 0.5 m/s; the turbulence intensity I is 10%; C_μ is a constant assigned 0.09; l is length scale; and T_{inlet} is imposed at 873.15 K.

2. **Outlet boundary**: At the outlet, a free outflow is assumed. A Neumann boundary condition is used for velocity, turbulent kinetic energy, turbulent dissipation, and temperature.

3. **Symmetry planes**: Symmetry boundary conditions are set for velocity, turbulent kinetic energy, turbulent dissipation, and temperature.

4. **Wall boundary**: The boundary condition at the inner wall is treated by the wall function method for velocity, turbulent kinetic energy, turbulent dissipation, and temperature. At the outer wall, the temperature is set as a constant (680 K).

Neutronics

1. **Inlet boundary**: The fast and thermal neutron fluxes are imposed as vacuum boundaries. Due to the DNP decay characteristics and residence time in the external loop, DNPs may return to the inlet. The DNPs at the inlet are:

$$C_{i,inlet} = C_{i,outlet} \cdot e^{-\lambda_i \tau}$$

   where $C_{i,outlet}$ is DNP group i at the outlet and τ is the residence time out of the core for fuel salt.

2. **Outlet boundary**: Vacuum boundary conditions are applied for neutron fluxes at the outlet, and DNPs adopt a zero-gradient boundary condition.

3. **Symmetry plane**: Neutron fluxes and DNPs are all set to symmetry boundary conditions.

4. **Wall boundary**: At the inner wall, neutron fluxes use coupling boundary conditions and DNPs are set to vacuum conditions. At the outer wall, vacuum conditions are applied for neutron fluxes.

Thermophysical Properties and Group Constants

Generally, the thermophysical properties in thermohydraulics and group constants in neutronics both depend on temperature. Ignatiev et al. [6] provided properties for the $LiF/NaF/BeF_2$ solvent system including density, thermal conductivity, viscosity, and heat capacity:

$$\rho_{fs}(\mathrm{kg \cdot m^{-3}}) = 2518 - 0.406T$$

$$\lambda_{T,fs}(\mathrm{W \cdot m^{-1} \cdot K^{-1}}) = 0.0429 + 0.0009T$$

$$\mu(\mathrm{Pa \cdot s}) = 0.001e^{-0.9942+1603.2/T}$$

$$c_{p,fs}(\mathrm{J \cdot kg^{-1} \cdot K^{-1}}) = 2090$$

The thermophysical properties of the graphite reflector are considered constant (Table 4).

The MSR3DS code system uses the DRAGON4 [14] code based on XMAS-172 format libraries produced from ENDF-VII.1 nuclear data to generate two-group macroscopic cross-sections, diffusion coefficients, and neutron group velocities under different temperatures. The relationships between group constants and temperature are fitted using 5th-order polynomial curves:

$$Y_c(T) = A_0 + A_1(T - T_{avg}) + A_2(T - T_{avg})^2 + A_3(T - T_{avg})^3 + A_4(T - T_{avg})^4 + A_5(T - T_{avg})^5$$

where Y_c denotes the group constants and T_{avg} is the reference temperature. A_0–A_5 are fitting constants.

Numerical Method

The finite volume method (FVM) is widely employed for solving computational fluid dynamics (CFD) problems in engineering. The solution domain is subdivided into a finite number of small control volumes and conservation equations are applied to each control volume. FVM can handle complex geometries. In this paper, FVM is used for spatial discretization of all equations.

This discretization process results in a system of linear algebraic equations that must be solved. For the discretized equations of steady neutron diffusion and heat transfer, the Gauss-Seidel iterative method is employed. The Preconditioned Bi-Conjugate Gradient Method (PBiCG) is used to solve the discretized equations for DNPs, momentum, turbulent kinetic energy, and turbulent dissipation. The Preconditioned Conjugate Gradient Method (PCG) is applied to the discretized pressure correction equations [15].

The SIMPLE [13] algorithm provides a method for calculating pressure and velocities. The acronym SIMPLE stands for Semi-Implicit Method for Pressure-Linked Equations. Originally proposed by Patankar and Spalding, it is essentially a guess-and-correct procedure for pressure calculation. It is an iterative method, and when other scalars are coupled to the momentum equations, the calculation is performed sequentially. The sequence of operations in the SIMPLE algorithm is shown in Fig. 2 [Figure 2: see original paper]. Fig. 3 [Figure 3: see original paper] shows the program flow diagram of the coupled solver in the MSR3DS code.

Description of the MSR3DS Code

The entire program diagram of the MSR3DS code is shown in Fig. 4 [Figure 4: see original paper]. The MSR3DS code system consists of pre-processing, coupled solver, and post-processing modules. The pre-processing module contains CAD 3D modeling, meshing, and generation of group constants. The CAD 3D modeling and meshing tools use SolidWorks and ANSYS ICEM CFD software, respectively.

DRAGON4 generates group constants under different temperatures for the MSR3DS code. DRAGON4 is open-source and contains a multi-group iterator designed to control various algorithms for solving the neutron transport equation. The SYBIL option solves the integral transport equation using the collision probability method for simple one-dimensional (1D) geometries (plane, cylindrical, or spherical) and the interface current method for 2D Cartesian or hexagonal assemblies. The EXCELL option solves the integral transport equation using the collision probability method for general 2D geometries and 3D assemblies. The MCCG option solves the integro-differential transport equation using the long characteristics method for general 2D and 3D geometries.

The ParaView [16] code serves as the post-processing module in the MSR3DS code. ParaView is an open-source, multi-platform data analysis and visualization application that can quickly build visualizations and analyze data using qualitative and quantitative techniques. Data exploration can be performed interactively in 3D or programmatically using ParaView's batch processing capabilities. Furthermore, it is easily integrated into the MSR3DS code.

Results and Discussion

Code Validation

The 3D TWIGL Seed/Blanket problem [17] is adopted to benchmark the neutronics calculation in the MSR3DS code. The effective multiplication factors calculated by CITATION [18] and MSR3DS are listed in Table 5 , with a relative error of only 0.00147%. The fluxes normalized to 1 MWth are shown in Figs. 5 and 6. The results verify the validity of the neutronics model presented in this study.

To evaluate the flow and heat transfer calculations in the MSR3DS code, a simple pipe case simplified from MOSART is used (Fig. 7 [Figure 7: see original paper]). The thermophysical properties of the molten salt and graphite at 873.15 K are applied in the validation process. Similarly, we assume the inlet velocity is 0.5 m/s for the flow calculation validation. The results are compared in Figs. 8 and 9. The figures show that results calculated by MSR3DS agree well with those from Fluent, confirming the validity of the flow calculation.

For heat transfer validation, the inlet temperature is assumed to be 873.15 K, while the outlet and other outer boundaries are at 300 K. Figs. 10 and 11 show that temperatures obtained by MSR3DS are in accord with those from Fluent, indicating that the heat transfer calculation in MSR3DS is acceptable in an engineering context.

Distributions of Calculated Physical Fields in the Core

Using the multiphysical coupling code, we calculated the 3D distributions of velocity, turbulent kinematic viscosity, fast and thermal neutron fluxes, DNPs, and temperature in the core under different conditions.

Figures 12-13 show the 3D distributions of velocity and turbulent kinematic viscosity. As shown in Fig. 13, the turbulent kinematic viscosity at the outlet is far greater than at other locations.

Figures 14-15 show the 3D distributions of fast and thermal neutron flux without flow (a), with convective term (b), and with both convective and turbulent diffusion terms at $Sc_t = 0.7$ (c) in the balance equations for six-group DNPs. The turbulent Schmidt number is set to 0.7 as the default value in ANSYS Fluent. The core temperature is set to 900 K under no-flow conditions. Figs. 14-15(a) and (b) show that fuel salt flow has little effect on the distribution of fast and thermal neutron fluxes. Comparing Figs. 14-15(b) with Figs. 14-15(c), the turbulent diffusion term hardly affects the distribution of fast and thermal neutron fluxes.

Figures 16-21 display the 3D distributions of DNPs without flow (a), with convective term (b), and with both convective and turbulent diffusion terms at $Sc_t = 0.7$ (c). The convective term significantly affects DNP distribution, as shown in Figs. 16-21(b), with smaller delay constants showing greater flow influence. After considering the turbulent diffusion term, the transport effect of turbulent diffusion reduces DNP concentrations and observably changes their distribution, again with smaller delay constants showing greater turbulent diffusion influence.

To calculate the temperature distribution in the core, the turbulent Prandtl number is set to 0.85, matching the default value in ANSYS Fluent. The calculated temperature distributions with convective term (a) and with both convective and turbulent diffusion terms at $Sc_t = 0.7$ (b) are shown in Fig. 22 [Figure 22: see original paper]. The results indicate that the temperature distribution in

the core remains essentially unchanged because DNPs affect neutron flux only slightly under steady-state conditions. However, because the transport effect of the turbulent diffusion term in the energy conservation equation becomes increasingly strong with the distribution of turbulent kinematic viscosity, the fuel temperature at the core outlet decreases slightly in both cases.

Conclusion

To investigate the complex behavior of the core in CMSFR, this research presents a new multi-physical coupling model including turbulent diffusion. The model physically describes the mutual dependence among neutron flux, delayed neutron precursor (DNP), heat transfer, and turbulent flow. The neutronics model consists of two-group neutron diffusion equations for fast and thermal neutron fluxes considering fuel salt flow effects, and balance equations for six-group DNPs considering both fuel salt flow effects and turbulent diffusion transport effects. In thermohydraulics, the RANS equations with Boussinesq's closure hypothesis and the standard k- turbulence model were adopted. Based on this model, integrating open-source DRAGON4 and ParaView code with CAD modeling and structured/unstructured mesh technology, a 3D multi-physical coupling steady-state code system was developed and validated.

To demonstrate the code's capability, 3D distributions of velocity, temperature, neutron flux, and DNPs were obtained for a simplified MOSART under steady-state conditions. The main results are:

1. Fuel salt flow has little effect on the distribution of fast and thermal neutron fluxes, and the turbulent diffusion term hardly affects their distribution.
2. The convective term significantly affects DNP distribution, with smaller delay constants showing greater flow influence. Turbulent diffusion reduces DNP concentrations and observably changes their distribution, with smaller delay constants showing greater turbulent diffusion influence.
3. The turbulent diffusion term in the balance equations for six-group DNPs does not change the temperature distribution in the core, but the transport effect of the turbulent diffusion term in the energy conservation equation has a strong effect on temperature distribution.

Therefore, the MSR3DS code system can be applied to simulate main physical fields and describe multi-physical coupling phenomena in the core of molten salt fast reactors, and can well reflect both the flow effect of the convective term and the transport effect of the turbulent diffusion term, which are peculiar to can-type molten salt reactors.

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Note: Figure translations are in progress. See original paper for figures.