Development of a transient neutronics and thermal-hydraulics coupling method for TRIGA reactor

Authors: Yi-Qing Zhang, Please provide the Simplified Chinese text (including any LaTeX and tags) that you would like me to translate into English., Cheng-Wei Liu, Ya-Nan Zhao, TaoYu, "QianGuo" is written in Hanyu Pinyin (the Romanization system for Mandarin Chinese), but without the corresponding Chinese characters it is ambiguous. It could correspond to several different names or words, such as:

- 前国
- 千国
- 乾国
- 钱国
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  ···or many other character combinations pronounced "Qián Guó," "Qiān Guó," etc.

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2. Whether you want a **meaning-based translation** (e.g., "Thousand Nations") or a **name transliteration** (keeping "Qian Guo" as a proper noun).

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Date: 2026-01-20T11:19:03+00:00

Abstract

The interaction and feedback between three-dimensional neutronics and thermal hydraulics are of great significance in reactor safety analysis, particularly for TRIGA reactors. Owing to the pulse-transient operating characteristics of TRIGA reactors, the power can change by six to eight orders of magnitude within an extremely short time; this behavior differs substantially from that of PWRs and poses challenges for conventional neutronics methods.

To accurately describe the transient behavior during control rod insertion or withdrawal, a novel time-dependent particle transport algorithm based on a combined and moving geometry approach is developed and implemented in the neutronics code MagicMC, a Monte Carlo particle transport code developed by the Nuclear Energy and Application Laboratory (NEAL). In conjunction with a subchannel model, this work proposes coupled neutronics and thermal-hydraulics methods for high-fidelity simulation of the TRIGA reactor.

First, a steady-state coupling method is established based on an over-relaxation iteration scheme, in which the number of neutrons in the Monte Carlo simulation is adaptively controlled according to the convergence behavior. Subsequently, a transient coupling method is proposed based on a semi-implicit coupling strategy, and a dynamically varying time-step scheme is designed for the coupling iteration process to achieve robust and efficient convergence. The parameter mapping strategy between neutronics and thermal hydraulics is constructed using one-to-one mapping and volume-weighting methods.

To verify the reliability of the proposed methods, the JSI TRIGA Mark II reactor is selected as a validation benchmark. The coupled simulation results show good agreement with the experimental data of the JSI TRIGA Mark II reactor, demonstrating that the coupling methods can achieve high-fidelity numerical simulation of the TRIGA reactor. Therefore, the coupling strategies proposed in this paper can provide technical support for reactor experiments and the safe operation of TRIGA reactors.

Full Text

Preamble

Development of a Transient Neutronics and Thermal-Hydraulics Coupling Method for TRIGA Reactors

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The interaction and feedback between three-dimensional neutronics and thermal hydraulics are of great significance in reactor safety analyses, particularly for TRIGA reactors. Owing to the TRIGA reactor' s pulse-transient operation status, the power changes by six to eight orders of magnitude within an extremely short duration. This operation is significantly different from that of PWRs and imposes challenges for conventional neutronics methods. To describe the transient status of rod insertion or withdrawal, a novel time-dependent particle transport algorithm based on the combing and moving geometry methods is developed and integrated into the neutronics code MagicMC, which is a Monte Carlo particle transport code developed by the Nuclear Energy and Application Laboratory (NEAL). Combined with a subchannel model, this work presents neutronics and thermal-hydraulics coupling methods for high-fidelity simulation of TRIGA reactors. First, a steady-state coupling method is established based on overrelaxation iteration, and the number of neutrons in the Monte Carlo simulation is adaptively controlled according to convergence. Subsequently, a transient coupling method is proposed based on the semi-implicit coupling strategy, and a dynamically changing time-step strategy is designed for the coupling iterative process to achieve reasonable convergence. The parameter mapping strategy between neutronics and thermal hydraulics was constructed using one-to-one mapping and volume weight methods. To verify the reliability of the methods, a JSI TRIGA Mark II reactor was selected as the validation benchmark. The coupling results were in good agreement with the experimental data of the JSI TRIGA Mark II reactor, and the coupling methods achieved high-fidelity numerical simulation of the TRIGA reactor. Therefore, the coupling methods proposed in this paper can provide technical support for reactor experiments and the safe operation of TRIGA reactors.

Keywords: Neutronics; Thermal-hydraulics; Time-dependent Monte Carlo; Transient coupling; TRIGA reactor

INTRODUCTION

Training Research Isotope General Atomics (TRIGA) is a type of pool-type nuclear research reactor generally used for neutron activation analysis and radiation hardness studies [1–3]. TRIGA has a small core size, strong neutron leakage, and extremely uneven core power and temperature distribution [4]. In addition, owing to the selection of U-ZrH as fuel, it exhibits an extremely high prompt negative temperature coefficient [5], which has led to the development of a special transient operating condition—pulse transient. During the pulse transient, the power changes by six to eight orders of magnitude in a short time (10 ms) [6]. With such strong variations in the core physics fields, high-fidelity transient neutronics and thermal-hydraulic coupling provides a necessary solution to ensure reliable operation.

In general, transient coupling can be divided into explicit, semi-implicit, and implicit coupling according to the different situations of the intercode communication process [7]. This coupling strategy is also applicable to TRIGA reactors. [Figure 1: see original paper] shows the working principles of the different transient couplings. Explicit coupling does not consider convergence in the time step but only data exchange at the beginning of the time step, and the Operator Splitting method [8, 9] is widely used in explicit coupling. Semi-implicit coupling requires iterative calculations at each time step to ensure calculation accuracy, which is usually based on the Picard iteration method [10–12]. Implicit coupling solves transient problems from the perspective of simultaneous neutronics and thermal-hydraulic equations, with data exchange between the solving steps. Because of this property, achieving full implicit coupling between independent codes of different computational domains is challenging. The JFNK method is typically used for realizing implicit coupling [13, 14]. Because of the temporal behavior of transient coupling, current neutronics solutions in TRIGA reactor coupled systems usually use the deterministic method [6, 15, 16], which simplifies the geometry and neutron energy group; thus, obtaining a high-fidelity solution has been challenging in transient coupling.

With advancing computational power, the Monte Carlo (MC) method is increasingly applied in the coupling of transient neutronics and thermal-hydraulics, owing to its precise geometric representation and high-fidelity computational accuracy. Two approaches analyze temporal neutron behavior in MC transient neutron transport methods: the quasi-static method [17, 18] and the time-dependent Monte Carlo (TDMC) method [19, 20]. The quasi-static method solves time-dependent neutron flux as a factorization of the time-independent shape function and time-dependent amplitude function. The amplitude function variation is calculated using the point kinetics model [17], which can suffer from inaccuracy due to discretization [20]. The MC code MagicMC [21], developed by the Nuclear Energy and Application Laboratory (NEAL) at the University of South China, implemented a TDMC particle transport algorithm based on the combing method [22] and moving geometry method. By coupling MagicMC with SubChanFlow, this work presents high-fidelity transient coupling methods for TRIGA reactors, developing an adaptive time-step strategy with semi-implicit transient coupling for TRIGA reactor pulsed transients.

The goal of this study is to establish a high-fidelity neutronics/thermal-hydraulics transient coupling method for TRIGA reactors. First, a novel time-dependent MC particle transport algorithm was established based on the combing and moving geometry methods. The neutronics/thermal-hydraulics coupling method was then developed using MagicMC and SubChanFlow for TRIGA reactors. 1) In the second section, the codes used in the coupling are introduced, the steady-state and transient coupling methods used in this study are proposed, and the convergence of the transient coupling time step is constrained. 2) In the third section, based on the coupling methods, the MC and subchannel code models of the JSI TRIGA Mark II reactor are created, and the relevant geometry and material data of the models are described. 3)

The fourth section presents the results and experimental comparisons. The experimental data from the JSI TRIGA MARK II reactor were compared and verified using the coupling methods developed in this study. The results show that the coupling method proposed in this study can complete the steady-state and transient simulations of a TRIGA reactor with high accuracy.

II. NEUTRONICS/THERMAL-HYDRAULICS COUPLING METHOD BASED ON MAGICMC AND SUBCHANFLOW

In this study, the MC code MagicMC and subchannel code SubChanFlow (SCF) [23] were used as solvers for neutronics and thermal-hydraulics calculations. The interface for data exchange between codes was built using the C++ language. Serial calculations are performed between the MC and subchannel codes, and the exchange of data fields uses a strategy that combines one-to-one grid mapping and the volume weight method [24]. To ensure computational speed and convergence of the transient calculation, this study adds a time-step optimization module to the transient coupling process and uses semi-implicit coupling to deal with the transient process.

A. Neutronics Analysis Code MagicMC

MagicMC is a neutron/photon transport calculation code based on the MC method, which can complete the modeling of a variety of complex geometries. It has developed a special time-dependent particle transport module for transient neutronics/thermal-hydraulics coupling calculations. MagicMC coupled with the on-the-fly cross-section generation method can quickly and accurately evaluate the impact of the Doppler effect [25] on resonance cross sections.

Owing to the TRIGA reactor' s pulse-transient operation status, the reactor changes from critical to supercritical and then to subcritical and finally reverts to its critical condition within an extremely short duration. To realize an accurate description of the pulse process induced by the continuous insertion or withdrawal of the pulse rod, MagicMC not only developed a time-dependent geometric movement function (moving geometry method) to describe rod movement in the reactor, but also calculated the time-dependent power changes under supercritical status through efficient control of the particle population with the combing method. The functional modules of the moving geometry, time-dependent particle transport, on-the-fly cross-section generation, and time-dependent tally developed by MagicMC for neutronic/thermal-hydraulic coupling are shown in [Figure 2: see original paper].

1. Moving Geometry Method In this study, to realize MC calculations with moving geometries under TRIGA reactor transient conditions, MagicMC decomposes the MC model into fixed and moving geometry parts. When a particle enters the moving geometry region, the positions of the moving geome-

tries are recalculated based on the current particle time variable and moving geometry motion parameters (i.e., translation and rotation), and the particle transport calculation is restarted. At the end of the transport time step, the particle time variable and positions of the moving geometries are recalculated, the cell parameters are updated, and the next transport step is started. This cycle is repeated until the particles disappear or leave the moving geometry region. [Figure 3: see original paper] illustrates the particle transport process in a moving geometry.

2. Time-Dependent Particle Transport with the Combing Method In this study, the combing method is introduced as a population control algorithm, which realizes control of the number of particles by combining particle weights when the neutron population increases drastically in the TRIGA pulse-transient operation status. Assuming a total of K particles in the system at moment t, the total weight of the particles accumulated by the system is given by

$$\sum_{i=1}^{K} w_i = N$$

A data set was generated according to the following method:

$$c_m = \xi + (m-1)\frac{N}{K}$$

where ξ is a uniformly distributed random number from 0 to 1 and N is the number of particles to be maintained.

When c_m falls into the weight interval w_i, the corresponding particle is copied and assigned a new weight value:

$$w_i' = \frac{N}{K}$$

Therefore, the weights of the particles compressed using the combing method are conserved in the statistics.

B. Subchannel Analysis Code SubChanFlow

SubChanFlow (SCF) is a subchannel-level thermal code for steady-state and transient analyses of reactors developed at KIT [26]. It is written entirely in Fortran [27]. Mass conservation, energy conservation, axial momentum conservation, and transverse momentum conservation equations are used to construct the solution model. At present, SCF can define a variety of coolant fluids such as water, lead, lead-bismuth, sodium, etc. Simultaneously, based on the subchannel method, SCF can flexibly divide the control volume in the reactor. In

theory, SCF can complete the model construction of all rod-fuel reactors. [Figure 5: see original paper] shows the input information and computational flow of SCF.

C. Neutronics/Thermal-Hydraulics Coupling Method

1. Steady-State Coupling Method The steady-state coupling process uses an overrelaxed iterative method. Steady-state coupling starts with an initial thermal-hydraulic calculation with the power set to a uniform distribution. The calculated temperature and coolant density fields are then passed to the input model of the MagicMC code, and the fuel temperature and coolant density are reset according to the mapping relationship, as shown in Eq. (8). The MagicMC code performs neutron transport to calculate the power distribution and passes it to the SCF model for correction. Thus, the iterative calculation outputs the results when the convergence conditions are satisfied. The steady-state MagicMC-SCF coupling method is illustrated in [Figure 6: see original paper].

This method uses the infinite norm of the fuel temperature, calculated twice using adjacent iterations, as the convergence condition [28, 29], as shown in Eq. (9):

$$T^n_{\text{fuel}}(i,j,k) = (1-\omega)T^{n-1}_{\text{fuel}}(i,j,k) + \omega T^n_{\text{fuel\_scf}}(i,j,k)$$

$$\epsilon_{L^\infty} = \max\left\{\left|\frac{T^n_{\text{fuel}}(i,j,k) - T^{n-1}_{\text{fuel}}(i,j,k)}{T^n_{\text{fuel}}(i,j,k)}\right|\right\}$$

where T^n_{fuel} represents the temperature of the reactor fuel, ω is the relaxation factor ranging from 1 to 2, $T^n_{\text{fuel\_scf}}$ is the calculated fuel temperature from SCF, n indicates the number of iterations, i, j, k represent the position information of the fuel element, ϵ_{L^∞} is the relative error calculated at the nth iteration, and ε is the convergence criterion.

In the MC calculations, the number of simulated particles strongly influences the convergence of the results. If the number of particles in the calculation is not controlled, it can lead to the method falling into a situation where it cannot converge when the convergence conditions are set too harshly. In addition, considering the error in the initial assumption of uniformly distributed power, an adaptive population control strategy was developed to adjust the number of neutrons in the MC calculations to satisfy the convergence condition in neutronics:

$$N_n = \begin{cases} N_{n-1}\left\lceil 1 + 10(\epsilon_{L^\infty})\right\rceil & \text{if } n \le n_{\max} \\ N_{n-1} & \text{if } n > n_{\max} \end{cases}$$

where N_n is the number of particles simulated in the nth iteration of the computational simulation, and $n_{\max}$ is the maximum number of iterations used to eliminate errors caused by uniformly distributed power.

2. Transient Coupling Method In transient analysis, because the time-dependent changes of neutronics and thermal-hydraulics make the coupling process relatively complex, semi-implicit coupling is used as the time-coupling strategy for MagicMC-SCF coupling in this work. The transient-coupling method is illustrated in [Figure 7: see original paper].

The method uses the following equations:

$$T_{\text{fuel}}^{t,n'}(i,j,k) = \begin{cases} (1-\omega)T_{\text{fuel}}^{t,n'-1}(i,j,k) + \omega T_{\text{fuel\_scf}}^{t,n'}(i,j,k) & \text{if } n' = 1 \\ T_{\text{fuel\_scf}}^{t,n'}(i,j,k) & \text{if } n' > 1 \end{cases}$$

$$L_T^{\infty} = \max\left\{\left|T_{\text{fuel}}^{t,n'}(i,j,k) - T_{\text{fuel}}^{t-1}(i,j,k)\right|\right\}$$

where t is a time point, Δt is the time step, L_T^{∞} is the infinite norm of the temperature field between adjacent time steps, $T_{\text{fuel\_scf}}^{t,n'}$ is the direct result of the coupling method in the n'th iteration of the calculation at moment t, a is the time step increase factor, b is the time step shortening factor, $N_{\max}$ is the maximum number of iterations, and $\Delta T_{\min}$ and $\Delta T_{\max}$ are semi-implicit iterative judgment conditions.

As time advances, the code first calculates the infinite norm L_T^{∞} of the temperature field between adjacent time steps, and determines the convergence of the data in a single time step through semi-implicit coupling. The specific method is as follows:

a) When $L_T^{\infty} < \Delta T_{\min}$, the data between adjacent time steps are considered to have good convergence and can be output directly. This method carries out transient coupling to advance and lengthen the time step to reduce the number of data exchanges.

b) When $L_T^{\infty} > \Delta T_{\max}$, the coupling method directly considers that the coupling result has not converged, reduces the time step, and returns to the calculation.

c) When $\Delta T_{\min} < L_T^{\infty} < \Delta T_{\max}$, the coupling data is considered to be between "convergence"and "non-convergence,"and over-relaxation iteration will be carried out in the time step at this time. During the iteration of this time step, the method still performs computation and judgment of L_T^{∞}. If the conditions in (a) or (b) are satisfied, the method prioritizes the execution of the time-step computation strategy in (a) or (b). If the conditions in (a) and (b) are not satisfied, iterations are performed based on the temperatures computed in Eq. (11) until convergence, after which time advancement is performed.

Because the transient coupling calculation is a process advancing with time, in practice, the neutron source calculated at different time steps will theoretically change. The coupling method updates the neutron source in each time step to minimize its influence on the coupling results.

III. NEUTRONICS/THERMAL-HYDRAULICS MODELS FOR TRIGA REACTOR

The TRIGA Mark II research reactor at JSI is a typical 250 kW TRIGA reactor, which is used for various applications such as neutron activation analysis, neutron radiography and tomography, education and training, radiation hardness studies, and benchmark experiments for the verification and validation of computer codes [1–3].

Figure 8: see original paper shows the overall design of the JSI TRIGA Mark II reactor [16, 30]. The core of the TRIGA reactor is placed at the bottom of an open tank (atmospheric pressure) with a 5 m water column above it. The core has a cylindrical configuration with 91 designed locations to accommodate fuel elements or other components, such as control rods, a neutron source, and irradiation channels. Figure 8: see original paper presents the dimensions of the JSI TRIGA reactor cores. The elements were arranged in six concentric rings: A, B, C, D, E, and F, each with 1, 6, 12, 18, 24, and 30 locations, respectively [31, 32].

A. Neutronics Model

Two different core arrangements are selected as numerical models for steady and transient states. Core 134 is selected as the steady-state model, and core 231 is selected as the research object for the transient model. The core structure diagram constructed using MagicMC is shown in Figure 8: see original paper. In core 134, all control rods except the regulating rod are fully withdrawn. In core 231, both the safety and shim rods are fully withdrawn, the transient rod (i.e., pulse rod) is fully inserted into the core, and the regulating rod is partially withdrawn to make the core exactly critical.

For the neutronics model, to smooth the temperature and power of the neutronics model, a zirconium rod is designed at the center of the fuel element, and small sections of graphite are designed at the upper and lower ends of the fuel element. The reactor core is reactively controlled by four control rods: a safety rod, shim rod, regulation rod, and transient rod. Except for the transient rod, all the other control rods are designed with a fueled-follower. Figure 8: see original paper shows the design scheme of the fueled-follower control rod and fuel element. Cores 134 and 231 are designed with 12 wt% uranium and U-235 with 20% enrichment fuel elements. lists the materials and dimensions used in the neutronics model.

B. Thermal-Hydraulics Model

The subchannel division method generally uses the coolant channel as the center [33]. In 2021, Manuel Garcia of KIT analyzed the two partitioning methods centered on the coolant channel and fuel element using the VVER benchmark and concluded that negligible difference existed between the results of the two partitioning methods [29]. Because the TRIGA Mark II reactor uses ring-filled fuel elements for the reactor design, this study adopts a subchannel division method centered on the fuel elements. The core sub-channel division is illustrated in Figure 8: see original paper.

The TRIGA reactor uses natural circulation to cool the reactor. The thermodynamic parameters of fuel U-ZrH are shown in Eq. (13) and Eq. (14), and the thermodynamic parameters of stainless steel are given by Eq. (15) and Eq. (16) [18, 34]:

$$C_v(\mathrm{UZrH}) = 4.17 \times 10^3 T^2 + 2.04 \times 10^6$$

$$k(\mathrm{UZrH}) = 0.0075T + 17.58$$

$$C_v(\mathrm{S.S}) = 2.88 \times 10^3 T^2 - 9.645 \times 10^5 T + 1.037 \times 10^8$$

$$k(\mathrm{S.S}) = -3.3 \times 10^{-6} T^2 + 1.70 \times 10^{-2} T + 14.280$$

where C_v is the heat capacity per unit volume in units of $\mathrm{J/(m^3 \cdot K)}$, and k is the thermal conductivity in units of $\mathrm{W/(m \cdot K)}$.

C. Spatial Mapping Method

Spatial mapping methods can be divided into volume weighting and one-to-one mapping methods. Because no power is released from the coolant during actual calculation, the data processing for the fuel element and coolant differs. The main data used for mapping are power, temperature, and density. The following method is used to map the data:

a) In the mapping of power data, the calculation is completed using the MagicMC code, ignoring the release of power in the coolant, mapping only the power data of the fuel element, and the mapping mode is one-to-one mapping. This simplification is verified in the power density calculation of steady-state coupling.

b) In the mapping of temperature data, we divided the temperature data into the temperature of the rod bundle and that of the coolant. For the rod bundle, a one-to-one mapping method is used to map the temperatures of the fuel, control rod, cladding, and other structures. [Figure 9:

see original paper] shows ring B as an example of a radial temperature-mapping scheme, which is divided into 10 nodes in the axial direction for data mapping.

c) For the coolant density, the volume weight method is used to map the data and calculate the average density according to the position of the coolant ring. In the mapping of reactor material density, the changes in the density of the fuel and cladding structure are ignored, and the density does not change during the coupling calculation process.

IV. EXPERIMENTAL AND NUMERICAL RESULTS COMPARISONS

A. Steady-State Coupling Results

The steady-state calculation used core 134 as the research analysis model, and two different power levels, 125 and 250 kW, were calculated. The two power levels were regulated by the position of the regulating rod, and the remaining control rods were fully withdrawn. At 125 and 250 kW, the regulating rod was inserted up to 19.4 and 13.7 cm, respectively; the coolant inlet temperature was 21.6 °C at both power levels. lists the information on k_{eff} for the steady-state coupling of the two power levels.

compares the calculated and experimental temperatures at certain fuel locations. These temperatures are located 7 mm from the center of the fuel element in the radial direction, and their axial position is at half the height of the fuel meat. The accuracy of absolute temperature measurements in the experiments, due to calibration uncertainty, is estimated at $±$3 °C [31]. The temperatures calculated in this study are consistent with the experimental values. The main error arises from the necessary model simplification required for code-based computation.

Figure 10: see original paper show the pin-wise distribution of radial power factor. The standard deviation of all power factors did not exceed 0.003. It can be observed that the power factor of ring B is higher than that of ring A for both power levels, and this result follows the value of the fuel elements of rings A and B in [31]. Because the steady-state power is controlled using only the insertion behavior of rod R, the power factors of rod R and its nearby elements differ significantly at the two power levels. In addition, the pinwise power peak factors at 125 kW and 250 kW are 1.3734 and 1.3403, respectively, which verifies that the deeper the R-rod is inserted, the more uneven the core power distribution. To demonstrate high-fidelity power distribution, Figure 11: see original paper show a finer radial power density distribution in the active zone of the core at 125 kW and 250 kW using an x-y grid of 500$×$500.

The normalized power distribution is calculated in the axial direction of the reactor, and the results are shown in Figure 11: see original paper. The relative axial position of only the active zone of the core is calculated, with the bottom

corresponding to relative position 0 and the top corresponding to relative position 1. The axial power peaking factors at 125 and 250 kW are 1.288687 and 1.300017, respectively.

Figure 11: see original paper shows a comparison of the calculated and experimental values of the fuel temperature for different rings.

B. Pulse Transient Coupling Results

The transient calculations use core 231 as the analysis model. During the transient calculation, the safety and shim rods are fully withdrawn, and reactivity is introduced by withdrawing the transient rod for a short time (80 ms). The integral value of the transient rod under core 231 is calculated to select the correct position for the transient rod. The reactivity worth of the transient rod is calculated using the positive-period method [35], as shown in Eq. (17), and the error propagated during the calculation is given by Eq. (18) [36]:

$$\rho = \frac{(1 - k_{\text{eff0}}) - (1 - k_{\text{eff}})}{k_{\text{eff0}}}$$

$$\Delta\rho = \left(\left(\frac{\Delta k_{\text{eff}}}{k_{\text{eff0}}} \right)^2 + \left(\frac{\Delta k_{\text{eff0}}}{k_{\text{eff0}}} \right)^2 \right)^{1/2}$$

where k_{eff0} denotes the effective multiplication factor. Subsequently, one of the control rods is withdrawn at a certain position by calculating a new k_{eff}. Δk_{eff0} and Δk_{eff} are the fractional statistical error estimates for k_{eff0} and k_{eff}, respectively.

The integral worth of the transient rod in core 231 is shown in Figure 12: see original paper. Compared to the reactivity value of the transient rod measured in the experiment [6], the integral value of the transient rod is roughly consistent with the experimental value, and the main error arises from the uncertainty of the model simplification construction and nuclear data.

Based on the calculated integral worth of the core 231 transient rod, this study used a coupled code to perform a reactivity insertion pulse transient simulation. The reactor started with an initial power of 100 W and coolant temperature of 21.6 °C. The length required for the transient rod to be withdrawn was calculated based on the integral value of the transient rod, as shown in Figure 12: see original paper. The fixed-source mode was used to calculate the power change in the reactor in a single time step. The shortest timescale of the power change in a single step was 0.002 ms (the minimum scale of the solving step).

The pulse power as a function of time is shown in Figure 12: see original paper. To analyze the error source, the peak power of the power curve, calculated using the coupling code, was aligned with the experimental value. compares the calculated and experimental values of the key parameters of the pulse-coupling

model. The experimental data used in this study were obtained from the JSI public TRIGA pulse experiment, which can be found at http://trigapulse.ijs.si/.

By comparing the power curve and pulse parameters, the coupling method proposed in this work was found to accurately simulate the pulse transient of the TRIGA reactor. The calculated and experimental power change curves are generally consistent. Small differences exist only in regions where the power varies relatively slowly because the presence of delayed neutrons in these regions affects the power variations. In the two simulations, the relative errors between the calculated and experimental values of the peak power were 0.705% and 1.834%, respectively, and the relative error of the maximum fuel temperature did not exceed 4.5%.

Figure 13: see original paper display the temperature distribution at different time steps under two transient coupling calculations. The temperature decreased from the inside to the outside. However, the positions of the center and transient control rod do not have fission power release; therefore, heat cannot be transferred to these locations in a short pulse time. Consequently, their temperatures were significantly lower than those of other parts of the core.

Because the coupling code uses MC for 3D neutronics calculations, it can output neutron flux distributions for different energy groups at different time steps. The central guide tube was selected as the analysis model, and the WIMS 69-group and WIMS 172-group structure [37] were used as the energy group division method. The flux distribution per lethargy of groups 69 and 172 at the central guide tube when the peak power was reached under the two transient conditions is shown in Figure 12: see original paper. The fluxes per lethargy for the different energy-group structures were in good agreement. Because the two-pulse calculations had little effect on the core structure, the relative flux distributions under different energy groups were basically the same. A distinct difference is observed only in the flux magnitude per lethargy, mainly because of the difference in peak power owing to the magnitude of the reactivity insertion.

C. Computing Environment

All calculations were performed on a computer with an Intel Core i5-13600KF CPU and 48 GB RAM. The calculations were performed using Linux with the CPU operating at a stable frequency of 4.6 GHz. An OpenMP parallel programming model with 20 threads was used for the computations.

In the steady-state coupling, the calculations were performed with 100 inactive cycles followed by 1000 active cycles, with 5000 neutrons per generation. The convergence condition (ϵ_{L^∞}) for the coupled system is set to 0.002. For 125 kW operation, the coupled system converges after 59 iterations, and the number of neutrons computed per generation increases to 22,533. For 250 kW operation, the convergence condition is reached after 49 iterations, and the number of neutrons computed per generation increases to 26,629. The runtime for both calculations was approximately 3 h.

In transient coupling, MagicMC used 1.5 million particles in each time bin, and the time bins under a single coupling time step were evenly divided into 100. In the adaptive time-step strategy, we set $\Delta T_{\max}$ and $\Delta T_{\min}$ to 6 °C and 2 °C, respectively, which controlled the time-step behavior of the coupled system. Semi-implicit calculations at a single time step were considered to converge when L_T^{∞} was less than 2 °C. The calculated runtime was approximately 28 h with 2.75$ reac- tive insertion and approximately 30h with 3$ reactivity insertion, and the minimum time step for coupling was 0.0004 s.

V. CONCLUSIONS

To describe the typical transient conditions of the TRIGA reactor in detail, a neutronics/thermal-hydraulics coupling method based on MC and subchannel methods was used in this study. Functional modules of moving geometry and time-dependent particle transport were developed using MagicMC to satisfy the 3D transient neutronics simulation of the TRIGA reactor. Two coupling methods are proposed for the steady-state and pulse transient of the TRIGA reactor: 1) The steady-state coupling method is based on the over-relaxation iterative method, and the temperature field is chosen as the convergence condition of the coupling calculation to successively approach the steady-state solution. 2) The transient coupling method uses semi-implicit coupling to control the length of the time step and iteration process using convergence evaluation criteria. Simultaneously, the source distribution is adaptively updated using MC iterations to ensure an accurate description of the source term. Based on the above coupling methods, the JSI TRIGA Mark II reactor was selected as the analytical model for validation. The numerical results show that the steady-state and transient neutronics/thermal-hydraulics coupling method proposed in this study can successfully realize the coupling simulation of the corresponding operating conditions, which can provide technical support for reactor experiments and safe operation of the TRIGA reactor.

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