



# Editorial: Reactor Physics: Methods and Applications

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## Editorial on the Research Topic

### Reactor Physics: Methods and Applications

The goal of reactor physics is to investigate the interactions of neutrons and matter in nuclear reactors using both analytical and numerical models. Nuclear physics modeling can be used to obtain critical neutronic information about the reactor core, such as the multiplication factor and the neutron flux (fission power) distribution. This field has profited from the development of both deterministic and Monte Carlo approaches. The deterministic technique employs a multi-step process for solving a problem that involves selectively combining a variety of neutronic models based on reactor type and application objectives. The Monte Carlo method is a straightforward and accurate approach for simulating neutron motions within a nuclear reactor. To comprehend the neutronic properties of nuclear reactors, reactor physics approaches are required, as they act as the foundation for reactor design and analysis.

In recent decades, reactor physics methods have advanced at an unprecedented rate. These advancements have had a profound effect on the nuclear energy industry. This study area examines advanced mathematical and numerical modeling techniques used in the field of reactor physics in order to provide an update on current reactor physics methodologies. This Research Topic comprises 26 papers on a variety of research topics, including nuclear data processing and resonance calculation models, cross-section homogenization techniques, steady-state and transient neutron transport methods, Monte Carlo approaches and applications, nuclear reactor design and analysis, and methods for sensitivity and uncertainty analysis. We classify and describe the substance of the featured papers in “*Reactor Physics: Methods and Applications*” in the sections that follow.

“*Reactor Physics: Methods and Applications*” contains three papers on nuclear data processing and model-based resonance computation. Choi et al. developed the pointwise energy slowing-down technique (PSM) to account for the nonuniformity of the fuel pellet’s material composition and temperature profile. This method eliminates the need for a pre-generated table and instead directly calculates the collision probability in all subdivided regions of the fuel pellet when solving the slowing-down equation. Extensive comparative analysis was conducted using models that simulated a variety of conceivable operating situations for a light water reactor (LWR) design. PSM predicts the eigenvalue with errors of over 2000 pcm for the pin-cell problem with steep temperature profiles and material compositions, whereas PSM-CPM predicts the eigenvalue accurately with errors of less than 100 pcm. Zu et al. investigated the effect of traditional approaches on thermal neutron scattering data using zirconium hydride as

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an example. The numerical results indicate that the incident energy grid has a significant effect on the eigenvalue, and that considering both coherent and incoherent elastic scattering simultaneously has a tens-of-pcm effect on the eigenvalue.

Advanced homogenization approaches are required for reliable prediction of nuclear reactor neutronic characteristics. To account for the environment effect in whole-core pin-by-pin computations, Zhang et al. conducted a systematic analysis of the relative errors of pin-cell homogenized group constants and determined the significance of adjusting pin-cell discontinuity factors (PDF) of the thermal group. The relationship between the thermal group PDF and the core parameters is functionalized and analyzed using the least-square method. Lei et al. investigated the connection between the reactivity calculation deviation and the optical length in order to determine the influence of double heterogeneity on homogenizing neutron cross sections for dispersed particle type fuels. They hypothesized and examined the two-step ring reactivity-equivalent physical transformation (TRRPT). It was demonstrated that the TRRPT method is more accurate in calculating reactivity and has a broader transformation range than the standard improved reactivity-equivalent physical transformation (IRPT).

The steady-state and time-dependent particle transport equations have formed the bedrock of the computational reactor physics area. Among them, the combined fission matrix theory has been praised for its high efficiency, fidelity, and resolution in solving the neutron transport equation in thermal nuclear reactors. He et al. investigated the viability of the combined fission matrix theory in fast reactors, concluding that approximations to the fission matrix elements will result in considerable errors in fast reactors. The combined fission matrix theory's application to fast reactors requires additional changes and refinement. Zhao et al. developed the SHARK nuclear reactor neutronics code. The code comprises a treatment of construct solid geometry (CSG), a method for subgroup resonance, and a two-dimensional/1-dimensional method of characteristics (MOC). The C5G7, BEAVRS, and VERA benchmarks are discussed numerically. Kang et al. presented a multilevel predictor-corrector quasi-static technique (AML-PCQM) for pin-resolved neutron kinetics problems in transient instances. The approach is established by integrating neutron transport, multi-group coarse mesh finite difference (CMFD), one-group coarse mesh finite difference (CMFD), and the point-kinetics equation. Additionally, one work on neutron transport acceleration has been approved for this Research Topic. Xu et al. applied the sign preservation rule from the field of numerical heat transfer to the CMFD framework in order to address the stability difficulties in CMFD. The updated method, dubbed rCMFD, outperformed previous CMFD methods in terms of efficiency.

As is widely known, Monte Carlo (MC) methods are used to solve the particle transport equation due to their precision and capacity to handle complex geometries. Ma et al. compared the single-node performance of history-based and event-based multigroup MC algorithms on CPUs and GPUs. This research may shed light on the proper selection of techniques for parallelizing MC codes on various architectures. Li et al.

used the MC approach to model and analyze the initial criticality of HTR-PM. This work is noteworthy for its use of the discrete element method (DEM) code LAMMPS to explicitly describe randomly packed TRI-structural ISotropic (TRISO) particles.

Computational reactor physics advances have significantly increased the breadth of nuclear reactor design and analysis. Zhao et al. used the RMC Monte Carlo code to perform pre-conceptual design of spectral-shift control rods in a small lead-based reactor. They observed improvements in the burnup depth and fuel usage rate. Wang et al. designed the scheme and analyzed the data for a major physical experiment involving a hexagonal casing type fuel reactor. The performance of fully ceramic microencapsulated fuel in a supercritical CO<sub>2</sub> (S-CO<sub>2</sub>) cooled reactor was investigated by Lu et al. To maximize neutron moderation in the high-performance S-CO<sub>2</sub> reactor assembly, the arrangement of moderator rods and fuel enrichment partition is investigated. Xu et al. investigated the properties of burnable poisons and neutronics in a long-life PWR plate fuel assembly. It was demonstrated that enriched 157Gd, enriched 167Er, B4C, 231Pa, PACS-J, PACS-Er, and PACS-Pa can be chosen and mixed as burnable poisons for plate-fuel assemblies without incurring large reactivity penalties. She et al. used the in-house deterministic code PANGU to simulate the HTR-10 reactor. In all steady-state power periods, numerical comparisons to measured data revealed good agreement. The difference in  $k_{\text{eff}}$  was within 500 pcm, and the difference in coolant outlet temperature was less than 5°C. Additionally, it was discovered that graphite impurity has minor impacts at the conclusion of the operating history, resulting in an discrepancy of up to 1,500 pcm.

The cost of the accompanying experiments for validating the computer models is a barrier to the deployment of reactor design improvements. To address this Research Topic, a criterion is required for determining whether a particular experiment, whether past or future, is relevant to the application of interest. Ma et al. introduced a generalized perturbation theory (GPT) based on an implicit sensitivity calculation method for evaluating the sensitivity and uncertainty associated with reactor physics modeling. Seo et al. stressed the concept of experimental relevance in their study, which extends the fundamental similarity score to account for the influence of previous experiments and associated experimental uncertainties. The suggested metric, dubbed ACCURE, is validated through a series of criticality experiments in order to determine the relevance of a group of tests to a certain application. Using a tritium breeding blanket, Qu et al. investigated the density perturbation calculation approach. Li et al. suggested a lightweight verification approach for nuclear reactor codes based on the metamorphic connection. The advantage of this method is that it determines the accuracy of the code by examining if the program meets the metamorphic relation, rather than manually solving or benchmarking the code. By combining observational data and a reduced model, the generalized empirical interpolation method (GEIM) is frequently employed to estimate the physical field. Gong et al. applied a smooth restriction on the GEIM to address the observation noise problem. The model constraints the

H1 semi-norm of the reconstructed field of the reduced model and proves to be efficient.

## AUTHOR CONTRIBUTIONS

All authors listed have made a substantial, direct, and intellectual contribution to the work and approved it for publication.

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