Ph.D. thesis booklet

Neutronics analysis of demonstration and experimental fast reactors with transport methods

Zoltán István Böröczki

Supervisor: Dr. Máté Szieberth

MŰEGYETEM 1782

Budapest University of Technology and Economics

Institute of Nuclear Techniques

Budapest, Hungary

2022

Introduction

Fast reactors are under extensive research to facilitate their commercial deployment and their introduction to the nuclear fuel cycle in the future. The use of these reactors offers improved sustainability of nuclear energy production with the extension of current reserves, while they also help reduce the amount of nuclear waste. At the Gen. IV International Forum in 2000 [1], six concepts were selected for further development, including sodium-, lead-, and gas-cooled fast reactor designs. Within the 7th European Framework Program (FP7), the objective of the Lead-cooled European Advanced Demonstration Reactor (LEADER) project was to design the Advanced Lead-cooled Fast Reactor Demonstrator (ALFRED)[2]. The construction and successful operation of ALFRED would be a milestone in the process of developing the Lead-cooled Fast Reactor (LFR) concept. Gas-cooled Fast Reactors (GFR) are also in extensive research, and the helium-cooled fast reactor, the ALLEGRO, could be the first demonstrator. The reactor was originally developed by CEA until 2009, then the V4G4 Centre of Excellence continued the preparation [3].

Due to the limited experience with fast reactors, the designs heavily rely on the results of numerical simulations. In particular, sensitivity and uncertainty analysis and transient simulations have a huge importance in identifying fundamental physical processes and critical points of the systems under development.

Objectives

Lower-order approximations of the angular-dependent flux, such as diffusion theory, are often applied for perturbation theory calculations or transient simulations. However, a wide range of problems is known where diffusion theory does not provide adequate results, and the application of higher-order transport approximation is recommended. One may easily find examples like [4], [5], [6], [7] and [8] in the field of fast reactor calculations.

The main focus of my research was to perform sensitivity and uncertainty calculations and transient simulations for fast spectrum reactors in higher ordered transport approximation; therefore, one of my main aims was to develop a new tool and compare its capabilities with other available codes for these types of analysis.

The newly developed code uses the PARTISN [9] deterministic discrete ordinate neutron transport solver to perform its calculations. Deterministic calculation necessitates group constants which are mainly applied for diffusion calculation. Developing and investigating procedures for group constant generation and comparing the available methods was also one of my main aims.

With the development of my own code, I aimed to provide a more accurate tool for sensitivity and uncertainty calculations for fast reactors. The development required different types of validation calculations, where I investigated the effect of the angular and spatial discretisation of the angular-dependent flux solution on the results. I have also implemented two time discretisation scheme into my code to perform time-dependent neutron transport calculations on fast reactors.

Methods

The dissertation presents the applicability of the SEnTRi code in various areas of the safety analysis of generation IV. reactor concepts. The effect of the angular and spatial discretisation of the angular-dependent flux on the results of perturbation theory calculations in sensitivity and uncertainty analysis is also investigated. Furthermore, the extension of the SEnTRi code has begun to perform coupled thermal-hydraulic calculations on fast reactors. Two different time discretisation scheme was implemented into the code to solve the time-dependent neutron transport equation, and the extension to take into account the thermal hydraulic feedback was done.

In order to perform a deterministic calculation, group constants must be prepared previously. The assumptions and the modelling during the homogenisations and the energy group condensation will highly affect the final group constants and, therefore, the accuracy of the deterministic calculations. The majority of the codes available for homogenised group constant generation for deterministic transport calculations apply the approximation of scalar flux weighting during energy group condensation of higher-order anisotropic scattering matrices. This condensation significantly affects the final results of the higher ordered transport calculations and is investigated in the dissertation.

The effect of the angular and spatial discretisation on the results of perturbation theory calculations is also studied with the help of the SEnTRi code. The accuracy of the calculations with the different angular flux discretisation options, namely discrete ordinates representation and spherical harmonics, are crucial from the viewpoint of perturbation theory and transient simulations.

Sensitivity and uncertainty calculations were performed for the Comet Critical Assembly and the ALFRED reactor core. Reactivity coefficients and their uncertainties originating from the nuclear data were determined using several codes and methods, and the properties of these fundamentally different calculations are presented.

Reactor kinetic codes are crucial in safety assessment. Validation calculations of the SEnTRi for a low power transient measured at the BME Training Reactor were performed, and the extension of the code began to perform coupled thermal hydraulic calculations on fast reactors. To test the newly developed features of the code, the simulation of an asymmetrical control assembly withdrawal transient of the ALLEGRO reactor was chosen, and the effect of applying diffusion or higher-order transport approximation was also investigated.

New scientific results

The new scientific results presented in the thesis can be summarized in the form of the following propositions:

Statement 1. I have investigated the bias introduced by the scalar flux-weighted linearly anisotropic scattering matrices, commonly used instead of current-weighted ones, in discrete ordinate neutron transport calculations, and showed that it is especially significant in small-sized, high-leakage cores with few group approximation. In such cases, the results showed larger deviations from the reference Monte Carlo calculations than the diffusion approximation. I have shown that in two energy groups, the results of the discrete ordinate calculations can be significantly improved with diagonal first-order scattering matrices derived from transport cross-sections. With an increasing number of groups, the bias from scalar-flux-weighting diminishes, and the scattering matrix estimated from the transport cross-section introduces more bias. [P1] [P3] [P5]

Statement 2. I have demonstrated with exact perturbation theory calculations based on the PARTISN discrete ordinate neutron transport code that both the conversion to spherical harmonics angular description and the use of cylindrical geometry introduce deviation in the calculation of the reactivity worth. This can only be eliminated by increasing the order of angular representation, which has to be considered when conversion between angular representations or cylindrical geometry is applied in perturbation theory calculations. [P6]

Statement 3. I have developed the SEnTRi code to determine the sensitivity coefficients with the help of the PARTISN discrete ordinate neutron transport code. Validation calculation was performed on the COMET lead-containing fast critical assembly with the Serpent, SCALE and SEnTRi codes with excellent agreement. I have shown that the inelastic scattering has an essential contribution to the positive void coefficient of the lead, which is reduced by the negative contribution of the elastic scattering. Therefore, the appropriate angular and spatial discretizations are essential to estimate the sensitivity coefficients accurately. [P2] [P4] [P8]

Statement 4. I performed sensitivity and uncertainty analysis and determined three reactivity feedback coefficients and their nuclear data uncertainties of the ALFRED reactor with the SEnTRi code and compared them with the results of the calculations of the TSUNAMI sequence of SCALE and Serpent. I have shown that radial leakage has an essential contribution to the lead void coefficient, and the inadequate spatial or angular discretization of the flux can lead to relevant distortion in the results. In the uncertainty calculations, the deterministic SEnTRi results were in good agreement with the Monte Carlo codes and provided much lower variances. [P2] [P4] [P8] [P9]

Statement 5. I have implemented a first-order fully implicit time discretization scheme into the SEnTRi code to perform time-dependent neutron transport calculations in three-dimensional geometry with the PARTISN discrete ordinate neutron transport code. The method was validated with the transient measurements of the BME Training Reactor and a comparison with the GUARDYAN Monte Carlo-based reactor kinetic code. I have determined the parameters of a hypothetical experiment for the Training Reactor, which would be suitable to test spatial-dependent reactor dynamics calculations. [P7]

Statement 6. I implemented a predictor-corrector quasi-static transient scheme into the SEnTRi code that can perform a time-dependent neutron transport calculation using the PARTISN discrete ordinate neutron transport code. The code was extended with the treatment of parameterized cross-section libraries and volume mixing to consider thermohydraulics feedback during a simulation and prepare the code for coupling with a system code. It was tested using an unprotected rod withdrawal transient on the ALLEGRO core applying a precalculated temporal temperature variation obtained from the KIKO3DMG calculation, and the effect of the diffusion and transport approximations was investigated. It was shown that the transport approximation affects the power distribution during the transient, although the effect is small and its importance could only be determined with fully coupled neutron transport and thermal hydraulics calculations.

List of publications

The work presented in the thesis is based on the following publications:

[P1] B. Babcsány, I. Pós, Z. I. Böröczki, D. P. Kis. "Hybrid finite-element-based numerical solution of the SP_3 equations – SP_3 solution of two- and three-dimensional VVER reactor problems". *Annals of Nuclear Energy*, 173:109117, 2022.

[P2] Z. I. Böröczki, Á. Aranyosy, M. Szieberth. "Comparison of calculation methods for lead fast reactor reactivity effects". *Proceedings of FR22*, Vienna, Austria, April 19-22, 2022.

[P3] B. Babcsány, Z. I. Böröczki, J. E. Maróti, M. Szieberth. "On the Effect of Scalar Flux Weighting of Linearly Anisotropic Scattering Matrices in Few-Group Transport Calculations". *Proceedings of PHYSOR 2022*, Pittsburgh, USA, May 15-20, 2022.

[P4] Z. I. Böröczki, Á. Aranyosy, M. Szieberth. "Comparison of calculation methods for lead fast reactor reactivity effects". *Annals of Nuclear Energy*, 171:109042, 2022.

[P5] A. Sz. Ványi, B. Babcsány, Z. I. Böröczki, A. Horváth, M. Hursin, M. Szieberth, Sz. Czifrus. "Steady-state neutronic measurements and comprehensive numerical analysis for the BME Training Reactor". *Annals of Nuclear Energy*, 155:108144, 2021.

[P6] Z. I. Böröczki, M. Szieberth, F. Gabrielli, A. Rineiski. "On the effect of angular and spatial discretization on perturbation calculations". *Journal of Computational and Theoretical Transport*, 5:347-363 2020.

[P7] Z. I. Böröczki, G. Klujber, G. Tolnai, B. Molnár, D. Legrady, F. Gabrielli, A. Rineiski, M. Szieberth. "Simulation of a research reactor reactivity transient with deterministic and GPU-assisted Monte Carlo reactor kinetics codes". *The European Physical Journal Plus*, 135:281 2020.

[P8] Z. I. Böröczki. "Sensitivity analysis with the PARTISN discrete ordinates neutron transport code". *Proceedings of the PhD workshop of the Physics Doctoral School at the Faculty of Science Budapest University of Technology and Economics (ed. F. Simon)*, Budapest, Hungary, ISBN: 978-963-313-293-7, July 06, 2018.

[P9] P. German, Á. Aranyosy, Z. I. Böröczki, M. Szieberth. "Sensitivity and Uncertainty studies for the Alfred lead cooled fast reactor core". *Proceedings of PHYSOR 2018*, Cancun, Mexico, April 22-26, 2018.

Bibliography

[1] U. S. Department of Energy. "A technology roadmap for generation IV nuclear energy systems". *Nuclear Energy Research Advisory Committee and the Generation IV International Forum*, 2002.

[2] A. Alemberti, M. Frogheri, and L. Mansani. "The Lead Fast Reactor: Demonstrator ALFRED and ELFR Design". pages 233–247, Paris, France, March 4-7, 2013.

[3] L. Bělovskỳ, J. Gadó, B. Hatala, A. Vasile, and G. Wrochna. "The ALLEGRO experimental gas cooled fast reactor project". In *Proceedings of International Conference on Fast Reactors and Related Fuel Cycles Next Generation Nuclear Systems for Sustainable Development*, pages 26–29, Yekaterinburg, Czech Republic, March 20, 2017.

[4] E. T. Tomlinson. "Transport-diffusion comparisons for small core LMFBR disruptive accidents". Technical report, ORNL/CSD/TM–38, USA: Oak Ridge National Laboratory (ORNL), 1977.

[5] Y. I. Kim, A. Stanculescu, P. Finck, R. N. Hill, and K. N. Grimm. *"BN-600 Hybrid Core Benchmark Analyses"*. Number 1623 in TECDOC Series. International Atomic Energy Agency, Vienna, Austria, 2010.

[6] K. Masatoshi, T. Yasushi, N. Akito, and Y. Mitsuaki. "Sensitivity analyses for small fast reactor nuclear characteristics with a discrete ordinate transport calculation method". In *Proceedings of PHYSOR 2010*, Pittsburgh, USA, May 9-14, 2010.

[7] A. Riyas, K. Devan, and P. Mohanakrishnan. "Perturbation analysis of prototype fast breeder reactor equilibrium core using IGCAR and ERANOS code systems". *Nuclear Engineering and Design*, 255:112–122, 2013.

[8] Y. Zheng, Y. Xiao, and H. Wu. "Application of the virtual density theory in fast reactor analysis based on the neutron transport calculation". *Nuclear Engineering and Design*, 320:200–206, 2017.

[9] R. E. Alcouffe, R. S. Baker, J. A. Dahl, S. A. Turner, and R. C. Ward. "PARTISN: A Time-Dependent, Parallel Neutral Particle Transport Code System, Version 7.72". Technical report, LA-UR-08-7258, USA: Los Alamos National Laboratory, 2008.