MŰEGYETEM 1782

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Experimental and numerical multi-physics analysis for the BME Training Reactor

Ph.D. thesis booklet

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Budapest
2023

Introduction

The comprehensive numerical analysis of nuclear reactors involves several fields and disciplines, such as reactor physics, thermal-hydraulics, coolant chemistry, etc. In the past, each of these aspects was treated separately by stand-alone single-physics codes, some of which were loosely coupled [1, 2]. In the 21st century, motivated by safety purposes and supported by the ever-increasing computational capacity, a new multi-physics approach is being realised in numerical reactor analysis. Novel multi-physics codes include several tightly coupled single-physics solvers within the same framework, which ensures that high-resolution, high-fidelity results can be obtained. Nowadays numerous novel multi-physics reactor analysis codes exist, such as the MOOSE project of Idaho National Laboratory [3], GeN-Foam of École Polytechnique Fédérale de Lausanne [4] or VERA of the Consortium for Advanced Simulation of Light Water Reactors [5]. On the other hand, the number of relevant and accessible validation experiments is low [6]. Such a lack is essentially due to the fact that there are only a few appropriate and available reactors for this purpose. Most of the research reactors are low power facilities and, therefore, they are often not suitable for the measurement of thermal-hydraulic feedback. On the other hand, commercial power reactors and research reactors are generally not available for such measurements.

The Training Reactor of Budapest University of Technology and Economics (BME TR) is a small-core, pool-type reactor of 100 kW thermal power, which ensures accessibility for instrumentation and its power is sufficiently high to produce measurable thermal-hydraulic feedback in response to reactor physics transients. Thus, BME TR is a suitable facility for designing and performing code validation experiments. The reactor is primarily cooled by natural circulation, therefore, similar conditions to normal operation and design basis accident scenarios of certain small modular reactors can be modelled as well.

Objectives

The doctoral dissertation consists of neutronic and thermal-hydraulic experiments performed on BME TR as well as corresponding modelling and simulations carried out with the Serpent 2, PARCS and TRACE calculation codes. The goal of the performed experiments was to provide validation measurements for computer codes that were developed to model small-core, pool-type, natural circulation cooled research or small modular reactors. The corresponding modelling work serves multiple purposes. It provides initial calculation models for the experiments, demonstrates a traditional neutronics–thermal hydraulics coupling and, finally, contributes to the modelling experiences on neutronic diffusion calculations and thermal-hydraulic system code analysis.

Methods

The reactor physics modelling of the Training Reactor was carried out with stochastic and deterministic calculation tools as well. The steady-state reference model was developed in the Serpent 2 Monte Carlo burnup calculation code. The initial model geometry and material composition were based on MCNP models that were developed earlier in the Institute of Nuclear Technique. The final model includes several new details and geometry-related corrections, while the fuel composition was updated according to new burnup calculations. The choice of Serpent 2 as reference code was also motivated by its capability to generate group constants for deterministic reactor physics core calculations. Using Serpent 2, both full-core and traditional two-step method group constants were generated from the reference model.

The deterministic neutronic modelling of the reactor was carried out with the PARCS three-dimensional nodal diffusion code. Throughout the calculations, the so-called hybrid nodal kernel was applied. The nodal core geometry consisted of a 8×9 Cartesian lattice of fuel and moderator assemblies, where each radial calculation node was representing an assembly. The Serpent 2 generated group constants were further processed with the GenPMAXS code that transforms the data into a format that can be directly read by PARCS.

The thermal hydraulics modelling of the Training Reactor was entirely realised in the U.S. NRC system code TRACE. The code uses a component-based approach, where the physical elements of the reactor or experiment can be represented by a corresponding TRACE component, which can be appropriately nodalised and parameterised. The TRACE model of BME TR mainly consisted of a three-dimensional VESSEL component representing the reactor vessel and 24 connecting CHANNEL components, each corresponding to a fuel assembly. The atmospheric pressure was set by the connecting SINGLE JUNCTION and BREAK components, while the vessel wall was modelled by an appropriately nodalised HEAT STRUCTURE unit. The walls and the structural elements within the reactor vessel were modelled using flow area restrictions and cell volume factors. In stand-alone mode, the assembly radial power distribution was input from Serpent 2 calculations and the power time series were supplied from measurements, while in coupled mode, all reactor physics related data were supplied by PARCS.

The reactor physics related measurements were performed using various detectors. The relative axial neutron flux measurements were carried out with the activation method using a Dy(5%)-Al alloy wire. The reactivity worth measurements were performed using the built-in ex-core CFUL08 fission chambers and evaluated with the method of inverse kinetics. The reactor power was monitored using KNK-53M boron-lined, gamma-compensated ionisation chambers.

The thermal hydraulics measurements were carried out utilising several type K, class 1 thermocouples with exposed hot junction and a type J sensor of class 1 precision with ungrounded hot junction. During the measurements, the thermocouples were connected to a National Instrument NI-9213 card. In the NI LabVIEW 2020 software, a graphical interface was created to display a graph of the thermocouple signals as well as the calculated mean temperature values and standard deviations for each sensor.

New scientific results

I. I have developed a guideline and procedure to generate diffusion coefficients for small-core, high-leakage reactors that take into account the neutron spectrum of the modelled material region. For fuel regions that are moderated with water, the reference current-weighted in-scatter or the flux-limited approximations, while for the water moderator regions, the reference-current weighted in-scatter approach is to be used. The reference-current spectra can be determined by Monte Carlo simulations of the given one-dimensional fuel–moderator systems. The calculations performed according to these guidelines yielded better power distributions for the investigated systems than simulations utilizing the traditional approaches. [P2, P3]

II. I have shown that the small-core, high-leakage Training Reactor of BME can be accurately modelled in terms of power distribution at steady-state by the deterministic nodal diffusion code PARCS, for which the group constants were generated by the Serpent 2 Monte Carlo code. In the case of calculations that used the full-core group constant set, the maximum relative assembly power difference was 2.93%, while the average was 1.01% compared to the reference Monte Carlo simulations. On the other hand, using the group constant set of the traditional two-step approach yielded maximum 8.82% and average 1.71% relative differences compared to the reference calculations. [P1, P3, P5]

III. I have demonstrated that the small-core, pool-type, natural circulation cooled Training Reactor of BME can be effectively modelled with the U.S. NRC system code TRACE. Together with colleagues, we have designed and performed transient measurements of various lengths (20–90 minutes), power levels (1–100 kW nominal power) and types, while the coolant temperature was measured in several axial and radial positions. I have evaluated and compared the measured and the calculated temperature profiles, which showed good qualitative and quantitative agreement. The difference between the measured and simulated temperature values was generally within 1 °C. [P4]

IV. I have investigated the transient analysis capabilities of the coupled TRACE/PARCS and TRACE/Point Kinetics code systems for the small-core, high-leakage, pool-type, natural circulation cooled Training Reactor of BME. We have designed and performed transient measurements with feedback of various lengths, powers and types in the Training Reactor of BME, where the outlet temperature of several fuel assemblies and the power of the reactor were simultaneously measured. I concluded that both coupled simulations significantly overestimate the transient peak reactor powers, which is probably due to the heat transfer correlations implemented in the code being outside of their validated range. [P5]

V. I have performed a new power calibration for the Training Reactor of BME using the foil activation and a calorimetry based method. The foil activation measurements were carried out earlier in the institute, while the corresponding numerical simulation was performed with the Serpent 2 Monte Carlo code. The calorimetry based estimation relied on three long, constant high power irradiations during which the temperature of the reactor vessel was monitored by thermocouples in several radial and axial positions. The estimations yielded the reactor power to displayed power ratios of 1.44 ± 0.03 and 1.43 ± 0.08 for the activation and calorimetry methods, respectively. [P6]

List of publications

[P1] 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*, Vol. 155, 2021.

[P2] A. Sz. Ványi, M. Hursin, Sz. Czifrus, “Investigation of Recently Introduced Diffusion Coefficient Generation Methods”, *Proceedings of NENE2021*, Bled, Slovenia, September 6–9, 2021.

[P3] A. Sz. Ványi, M. Hursin, Sz. Czifrus, “Analysis of diffusion coefficient correction methods applied for small-core, high-leakage reactors”, *Annals of Nuclear Energy*, Vol. 174, 2022.

[P4] A. Sz. Ványi, M. Hursin, A. Aszódi, L. Adorján, B. Biró, B. Magyar, P. Mészáros, T. Bozsó, Sz. Czifrus, “Thermal-hydraulic measurements and TRACE system code analysis performed on the natural circulation cooled BME Training Reactor”, *Annals of Nuclear Energy*, Vol. 189, 2023.

[P5] A. Sz. Ványi, M. Hursin, Sz. Czifrus, “Analysis of transient measurements with thermal feedback and coupled TRACE/PARCS calculations performed on the BME Training Reactor”, *Annals of Nuclear Energy*, Vol. 194, 2023.

[P6] András Szabolcs Ványi: Three-dimensional Monte Carlo reactor physics calculation and foil activation measurement based power calibration of the BME Training Reactor (Original Hungarian title: A BME oktatóreaktor háromdimenziós Monte-Carlo reaktorfizikai számításokon és fóliaaktivációs méréseken alapuló teljesítménykalibrációja), BME-NTI-1007/2023 research report, BME NTI, 2023.

References

[1] W. A. Rhoades and R. L. Childs, "TORT: A Three-Dimensional Discrete Ordinates Neutron/Photon Transport Code," *Nuclear Science and Engineering*, vol. 107, no. 4, pp. 397–398, 1991.

[2] U. Grundmann and U. Rohde, "DYN3D - A 3-dimensional core model for steady state and transient analysis in Thermal reactors," *Proc. Int. Conf. On the Physics of Reactors "PHYSOR 96", Mito (Japan)*, 1996.

[3] D. Gaston, C. Newman, G. Hansen, and D. Lebrun-Grandié, "MOOSE: A parallel computational framework for coupled systems of nonlinear equations," *Nuclear Engineering and Design*, vol. 239, pp. 1768–1778, 2009.

[4] C. Fiorina, I. Clifford, M. Aufiero, and K. Mikityuk, "GeN-Foam: a novel OpenFOAM based multi-physics solver for 2D/3D transient analysis of nuclear reactors," *Nuclear Engineering and Design*, vol. 294, pp. 24–37, 2015.

[5] R. Schmidt, K. Belcourt, R. Hooper, R. Pawlowski, K. Clarno, S. Simunovic, S. Slattery, J. Turner, and S. Palmtag, "An approach for coupled-code multiphysics core simulations from a common input," *Annals of Nuclear Energy*, vol. 84, pp. 140–152, 2015.

[6] T. Valentine, M. Avramova, M. Fleming, M. Hursin, K. Ivanov, A. Petruzzi, U. Rohatgi, and K. Velkov, "Overview of the OECD-NEA Expert Group on Multi-physics Experimental Data, Benchmarks and Validation," PHYSOR2020 – International Conference on Physics of Reactors: Transition to a Scalable Nuclear Future; Cambridge, UK, March 28 - April 2, 2020.