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Nodal models of Pressurized Water Reactor core for control purposes – A comparison study

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HIGHLIGHTS

- Comparison of basic nodal and expanded multi-nodal models of the PWR core.
- Implementation of power distribution coefficients in expanded multi-nodal model.
- Methodology for calculation of power distribution coefficients.
- Comparison of simulation results with dedicated nuclear simulation software.
- New capabilities of expanded multi-nodal models in synthesis of control algorithms.

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ABSTRACT

The paper focuses on the presentation and comparison of basic nodal and expanded multi-nodal models of the Pressurized Water Reactor (PWR) core, which includes neutron kinetics, heat transfer between fuel and coolant, and internal and external reactivity feedback processes. In the expanded multi-nodal model, the authors introduce a novel approach to the implementation of thermal power distribution phenomena into the multi-node model of reactor core. This implementation has the form of thermal power distribution coefficients which approximate the thermal power generation profile in the reactor. It is assumed in the model that the thermal power distribution is proportional to the axial distribution of neutron flux in the un-rodded and rodded reactor core regions, as a result of control rod bank movements. In the paper, the authors propose a methodology to calculate those power distribution coefficients, which bases on numerical solutions of the transformed diffusion equations for the un-rodded and rodded reactor regions, respectively. Introducing power distribution coefficients into the expanded multi-nodal model allows to achieve advanced capabilities that can be efficiently used in design and synthesis of more advanced and complex control algorithms for PWR reactor core, for instance in the field of reactor temperature distribution control.

1. Introduction

The mathematical models of nuclear reactor processes can be divided into two main groups, the first of which is related to complex and accurate models, while the second represents less accurate, reduced and simple models. These two groups have different applications and purposes. The first group models may be used in design of reactor core, safety considerations, or detailed analyses of phenomena and nature of processes occurring in the nuclear reactor core. On the other hand, the less accurate models composing the second group can be used in control systems synthesis, and for education or training purposes. The second group models should also comply with several aspects which the first group is unable to fulfil, for instance easy implementation, convenient calculation time, or simple description. This paper focuses on the second group of models, with emphasis on nuclear reactor control systems and algorithm synthesis.

Several approaches can be efficiently used in that field, of which the lumped parameter models are most common and widely utilized. In Han (2000) the author presents dynamic models for the primary loop systems of nuclear power plants, which have the potential of fast running on personal computers. These models have been mainly made for thermal-hydraulic analysis purposes. Another example is the paper (Fazekas et al., 2007), in which the authors present simplified dynamic models of primary circuit

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elements of a nuclear power plant for control system design purposes. Also in Dong et al. (2009) the authors present lumped parameter dynamic models of primary loop elements for control system design and simulation, while in Karla et al. (2015) and Tarnawski and Karla (2016) the authors present an approach to building lumped model based low cost non-real-time and real-time nuclear reactor simulators. Lumped parameter models can also be utilized in advanced process control. Authors in Kulkowski et al. (2015) and Sokolski et al. (2016) presents simple and complex non-linear dynamic models of nuclear power plant steam turbine used for that purposes.

Another modelling approach is the usage of fractional calculus. In Espinosa-Paredes et al. (2011) and Nowak et al. (2014a,b) the authors present similar ways of modelling neutron point kinetics equations which play a significant role in the reduced models of nuclear reactor core. Also in Nowak et al. (2015) the authors introduce fractional order neutron kinetics equations, along with the integer order heat transfer model. The fractional order models are more complex than the lumped non-fractional parameter models, because of more sophisticated computational methods to be applied. On the other hand, they have an ability to mimic some physical processes in a more accurate way.

The last, but not least, field in simplified modelling of nuclear processes is nodal approach. Nodal methods divide the distributed systems into smaller parts (nodes) which can be modelled by ordinary differential equations. Due to a high number of computational nodes used, these models are most complex, yet still capable of performing calculations in convenient time. The nodal models are also interesting for their ability to model spatial relations in the modelled elements with little effort. The point kinetics approach is not proper in some types of nuclear reactors, as stated in Dong et al. (2010). The authors of that paper derive the nodal neutron kinetics model with corresponding nodal thermal-hydraulic models of nuclear reactor core. In Puchalski et al. (2016) the authors derive nodal thermal-hydraulic models of reactor core with corresponding power distribution coefficients, while in Sharma et al. (2003) and Tiwari et al. (1996), for control purposes of xenon induced spatial oscillations, the authors divide the nodal model of reactor core into a number of zones, which are then treated as small cores coupled through neutron diffusion – a concept of coupled-core kinetics.

In Zhang (2012) and Zhang et al. (2013) the authors compare nodal and distributed parameter models in the frequency domain and identify advantages and disadvantages of each model, while in Puchalski et al. (2015a) and Guimarães et al. (2008) the authors present simple and multi-nodal models of the U-tube steam generator, a crucial element in the nuclear power plant, for control and simulation purposes. For example, in Puchalski et al. (2015a,b) the multi-regional fuzzy controller, with local PID controllers and Takagi-Sugeno reasoning mechanism for U-tube steam generator water level control, and the membership functions tuning procedure is described.

The nodal models presented in the paper are based on the point kinetics model to describe the time-depended average neutron population including the delayed neutrons in the reactor core. Moreover, they make use of a nodal approach to describe the heat transfer between the reactor fuel and coolant, and the reactivity feedback influences related to the main internal and external mechanism of fuel and coolant temperature changes and control rod bank movement. In the paper the authors expand the nodal approach presented mainly in Kerlin (1978) and applied in Naghedolfeizi (1990), Kapernick (2015), Liu (2015) and Perillo, 2010 by adding special thermal power distribution coefficients to approximate thermal power generation distribution in the reactor core with respect to the reactor control rod bank movement.

This paper is organized as follows. In Section 2 the point kinetics model is briefly described, while in Section 3 basic nodal models of the PWR reactor core are described and compared in simulation. In Section 4 the extended multi-nodal model of PWR reactor core is described in detail, including: model equations, different cases of thermal power distribution coefficient calculations, comparison of simulation results for selected number of fuel nodes, and comparison with dedicated nuclear simulation and analysis software. Finally, conclusions are presented in Section 5.

2. Point kinetics model of the reactor core

In general, the behaviour of the distributed physical systems can be represented by lumped point models. Partial differential equations, commonly used to describe spatial systems, can be simplified to ordinary differential equations with a finite number of parameters. With this simplification, the space dependent nature of the process can be neglected by using averaged physical quantities. For the nuclear reactor core, the point model describes average values of the neutron density and the temperature of fuel or coolant, i.e. the physical quantities which are most important for reactor core operation.

In the paper, the point kinetics model of nuclear reactor core is used to describe the time-depended average neutron population, including six groups of delayed neutrons. It is featured by a well-known set of ordinary differential equations Duderstadt and Hamilton (1976)

$$\frac{d\bar{n}(t)}{dt} = \frac{\rho(t) - \beta}{\Lambda}\bar{n}(t) + \sum_{i=1}^{6}\lambda_i C_i(t), \tag{1}$$

$$\frac{dC_i(t)}{dt} = \frac{\beta_i}{\Lambda}\bar{n}(t) - \lambda_i C_i(t), \quad i = 1, \ldots, 6, \tag{2}$$

where $\bar{n}$ is the average neutron density, ρ is the reactivity, $\beta = \sum_{i=1}^{6}\beta_i$ is the total yield of the delayed neutron precursors, Λ is the average neutron generation time, λ_i are the decay constants of the delayed neutron precursors, C_i are the concentrations of the ith group of the delayed neutron precursors, β_i are yields of the delayed neutron precursors, and t is the time.

Taking into account the well-known fact that the thermal power generated in the reactor core is proportional to the neutron flux and the average neutron density

$$P_{TH}(t) \sim \phi \sim \bar{n}, \tag{3}$$

the thermal power generated in the reactor core is assumed to be calculated from the formula

$$P_{TH}(t) = \frac{\bar{n}(t)}{N_{0,N}} P_{TH,N}, \tag{4}$$

where P_{TH} is the reactor thermal power, $N_{0,N}$ is the nominal average neutron density at full power, $P_{TH,N}$ is the nominal power of the reactor core, and ϕ denotes the neutron flux. The values of numerical parameters of the PWR reactor point kinetic model are included in Appendix A. The heat transfer processes, and the reactivity feedback due to the fuel temperature, coolant temperature, and control rod bank movements, are also taken into account and described in detail in the following sections for the appropriate reactor nodal models.

3. Basic heat transfer nodal models of the reactor core

This section focuses on two basic nodal models of heat transfer in the reactor core, Fig. 1. These models are most popular and widely used in the field of control algorithm synthesis. The first nodal model includes one fuel node and one coolant node –