Book Review

Nuclear Reactor Thermal-Hydraulics: Past, Present and Future, 1st edition. By Pradip Saha. ASME Press, New York (2017). LCCN 2016051235, ISBN 9780791861288.

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Providing a study of a complex power plant technology, with a phenomenological description of its operational principles and an overview of engineering design methods, current deployments, and future development is a very challenging task that requires substantial author skills and experience. An excellent example of succeeding with such a task is the monograph *Nuclear Reactor Thermal-Hydraulics: Past, Present and Future*, written by Dr. Pradip Saha. The purpose of the monograph is completely fulfilled: It is a concise but comprehensive account of thermal hydraulics in nuclear power reactors during steady states, transients, and accident conditions. Primary focus is on water-cooled reactors including pressurized water reactors (PWRs), boiling water reactors (BWRs), and pressurized heavy water reactors (PHWRs). Brief comments are also made on the thermal hydraulics of more promising small modular reactors (SMRs) and Generation IV reactors.

The monograph is presented in 140 pages and divided into five chapters—Chapter 1: "Introduction," Chapter 2: "Steady State Reactor Thermal-Hydraulics," Chapter 3: "Nuclear Reactor Safety Systems," Chapter 4: "Nuclear Reactor Safety Analysis," and Chapter 5: "Summary and Conclusions." The presented material is documented by 229 references.

Chapter 1 presents the current status of nuclear power plant utilization, principles of nuclear fission and heat generation, nuclear reactor designs and classification, and the role of thermal hydraulics in reactor steady-state and transient operating conditions as well as in accident conditions. Desired features of the reactor coolant are also included.

Physical phenomena of steady-state reactor thermal hydraulics and modeling tools for their simulation and analyses are presented in Chapter 2. Provided are clear and comprehensive presentations of single-phase and two-phase boiling flows of coolants in the reactor core, including important topics of boiling two-phase flow, such as the void-quality relationship, pressure drop in two-phase flow, departure from nucleate boiling (DNB) and boiling transitions, critical heat flux and critical power enhancement, and margin to critical conditions. Dr. Saha's unique contribution in Chapter 2 is the presentation of analytical methods for the thermal-hydraulic analysis of nuclear fuel subchannels. He describes the traditional approach based on the mass, momentum, and energy balance equations for two adjacent channels and mixing models for cross flow between channels. Related experimental investigations and results on rod bundle thermal hydraulics, which supported the development of thermal-hydraulic subchannel models, are also presented. The study is further extended to the two-fluid, three-field approach based on the balance equations for vapor, liquid droplets, and liquid film flows; and related interface transfer models, such as models for entrainment and deposition, which are especially useful in cases of annular flow simulations and dryout predictions under high void fractions in nuclear fuel rod bundles.

Chapter 3 presents design and thermal-hydraulic features of nuclear reactor safety systems together with their role in providing safety during abnormal and accident conditions of nuclear power plants. The historical development and design features of Generation II, Generation III, and Generation III+ nuclear power plants and related safety systems are reviewed. Distinctive features of active and passive safety systems are presented. Besides PWRs, BWRs, and PHWRs, the presented comprehensive study also includes Russian design of the VVER-1000 and SMRs.

Evaluation of nuclear power plant safety philosophy and development of computer tools for nuclear reactor safety analyses are presented in Chapter 4. Included are the conservative thermal-hydraulic analysis of nuclear reactor safety based on loss-of-coolant accidents (LOCAs) up to the mid-1970s, the development of the best-estimate methodology up to 1990, and the consolidation of the best-estimate methodology up to the present day. The main thermal-hydraulic phenomena (phases) and effects during LOCAs and the abilities of computer codes to predict them are discussed. Developments and applications of safety analysis codes, such as RELAP, TRAC, CATHARE, COBRA, CATHENA, and their modes are discussed and evaluated. The three-dimensional effects of reactor core flooding and cooling during LOCAs are included, such as those results obtained in the Upper Plenum Test Facility. Basic challenges to the development of models incorporated in the computer codes for the safety analyses are outlined, for example, the prediction of the interfacial area transport equation. Modern developments and applications of computational fluid dynamics (CFD) and computational multi-fluid dynamics (CMFD) tools, as well as efforts toward multiscale analyses of reactor thermal hydraulics are also presented. Activity of the Consortium for Advanced Simulation of Light Water Reactors is presented together with examples of the consortium efforts to improve the capabilities of thermal-hydraulic phenomena and effect predictions, such as boiling crises (DNB) by the CFD approach. Chapter 4 is concluded with the presentation of thermal-hydraulic analysis applications to Generation IV nuclear reactors.

Chapter 5 outlines the material presented in the monograph as well as presents an outlook for the development and deployment of new Generation IV reactors.

Nuclear Reactor Thermal-Hydraulics: Past, Present and Future is a valuable contribution to the engineering literature on nuclear energy and nuclear reactors. Students can effectively acquaint themselves with the broad area of nuclear reactor thermal hydraulics, while experienced engineers and academics may refresh their knowledge, inform themselves about certain topics, and get a systematic overview of this essential field for the exploitation of nuclear energy and further development of nuclear engineering and design. This monograph is highly recommended for students, experienced engineers, and academics alike.

About the reviewer: Vladimir D. Stevanovic (PhD, Thermal Power Engineering, University of Belgrade) is the professor of Thermal and Nuclear Power Engineering at the University of Belgrade, Faculty of Mechanical Engineering. His research interest is in the field of nuclear and coal-fired steam generator thermal hydraulics, two-phase flow, and CMFD. He has developed computer codes for thermal-hydraulic, three-dimensional analyses of horizontal and vertical steam generators in VVER and PWR nuclear power plants, predictions of radiolytic gas accumulation in nonvented steam pipelines in BWR safety systems, and simulations of water hammers in single-phase and two-phase systems, which have been applied in industry. Professor Stevanovic has been the principal investigator in industrial projects for the power and energy efficiency upgrade in thermal power plants and safety analyses of nuclear power plant incidents. He has authored or coauthored more than 50 archival peer-reviewed journal papers and more than 200 conference papers and technical reports.