ANALYSIS OF PROTECTED RIA AND LOFA IN PLATE TYPE RESEARCH REACTOR USING COUPLED NEUTRONICS THERMAL-HYDRAULICS SYSTEM CODE

Marat Margulis and Erez Gilad*
The Unit of Nuclear Engineering
Ben-Gurion University of the Negev, Beer-Sheva 84105, Israel
maratm@post.bgu.ac.il; gilade@bgu.ac.il

ABSTRACT

The application of Best Estimate (coupled neutron kinetics/thermal-hydraulics - NK/TH) codes for research reactors safety analyses has gained considerable momentum during the past decade. This activity is largely facilitated by the high level of technological maturity and expertise attained by these techniques as NPPs safety technology and is largely driven by IAEA activities. The present study belongs in this framework, where a coupled NK/TH code (THERMO-T) was developed and applied to the analysis of protected reactivity insertion (RIA) and loss of flow (LOFA) accidents in a typical research reactor with standard MTR plate type fuel assemblies. The coupling is realized by considering the neutronic reactivity feedbacks of the fuel and coolant temperatures and a heat generation model for the reactor power. The neutron flux in the reactor core is solved by applying the point reactor kinetic equations, using radial and axial power distributions calculated from a 3D full core model by the three-dimensional continuous-energy Monte Carlo reactor physics code Serpent. The evolution of temporal and spatial distributions of both fuel and coolant temperatures is calculated for all fuel channels using a finite volumes time implicit numerical scheme for solving a three conservation equations model. In this study, three different thermal hydraulic models of the code are evaluated, as well as its sensitivity to different heat transfer correlations.

KEYWORDS
Research reactor, transients, safety analysis, LOFA, RIA, coupled system code

1. INTRODUCTION

Research Reactors (RRs) are developed and built primarily as test facilities and neutron generators for vast range of scientific, industrial and medical purposes. Unlike commercial nuclear power plants (NPPs), RRs are characterized by small core size, low total thermal power, high power density, low fuel and clad temperatures and low system pressure. Furthermore, the different fuel composition, geometric configuration and different ranges of relevant operational parameters constitute different neutronic and thermal-hydraulics designs [1-4].

As a result, these reactors must meet different safety requirements and unique safety features to ensure their safe utilization in nominal and off-nominal operation conditions and safe shutdown in case of an emergency or an accident. The reactor safety analysis report is frequently updated and must include the analysis of a wide variety of safety related scenarios. Furthermore, the uniqueness of each RR and its experimental systems makes the standardization of design, operation and licensing of RRs almost impractical, unlike commercial NPPs [2,5].

* Corresponding author

19. T. Hamidouche, A. Bousbia-Salah, E.K. Si-Ahmed, M.Y. Mokeddem and F.D'Auria, "Application of coupled code technique to a safety analysis of a standard MTR research reactor," *Nucl. Eng. Des.* **239**(10), pp. 2104-2118 (2009).
20. IAEA, "Research Reactor Core Conversion from the Use of High-Enriched Uranium to the Use of Low Enriched Uranium Fuels Guidebook," IAEA-TECDOC-233 (1980).
21. IAEA, "Research reactor core conversion guidebook," IAEA-TECDOC-643 (1992).
22. J. Leppänen, "Development of a new Monte Carlo reactor physics code," D.Sc. Thesis, Helsinki University of Technology, (2007).
23. M.A. Gaheen, S. Elaraby, M.N. Aly and M.S. Nagy, "Simulation and analysis of IAEA benchmark transients," *Prog. Nucl. Enrgy* **49**(3), pp. 217-229 (2007).
24. U. Grundmann, U. Rohde, and S. Mittag, "DYN3D – Three-dimensional core model for steady-state and transient analysis of thermal reactors," *Proceedings of the 2000 ANS International Topical Meeting on Advances in Reactor Physics and Mathematics and Computations into the Next Millenium (PHYSOR 2000)*, Pittsburgh, USA, May 7-11 2000, (2000).
25. G.R. Keepin, *Physics of nuclear kinetics*, Addison-Wesley, Reading, Mass. (1965).
26. C.P. Tzanos, "Predictions of the heat transfer coefficient by correlations and turbulence models," *Nucl. Technol.* **183**(1), pp. 88-100 (2013).
27. C.A Sleicher and M.W. Rouse, “A convenient correlation for heat transfer to constant and variable property fluids in turbulent pipe flow”, *Int. J. Heat Mass Transfer* **18**, pp 677-683(1975).
28. E.N. Sieder and G.E. Tate, “Heat transfer and pressure drop of liquids in tubes”, *Ind. Eng. Chem.,* **28**(12), pp. 1439-1435(1936).
29. F.W. Dittus and L.M.K. Boelter,”Heat transfer in automobile radiators of the tubular type”, *Int. Comm. Heat Mass Transfer*, **12**, pp.3-22(1985).
30. B.W. Petukhov, “Heat transfer and function in turbulent pipe flow with variable physical properties” in J.P. Harlett and T.F. Ivrine (eds), *Advances in Heat Transfer,* Academic Press, New York (1970).