Progress in Nuclear Energy 109 (2018) 1–11

Contents lists available at ScienceDirect

Progress in Nuclear Energy

journal homepage: www.elsevier.com/locate/pnucene

Simulations of SPERT-IV D12/15 transient experiments using the system code THERMO-T

Marat Margulis, Erez Gilad*

The Unit of Nuclear Engineering, Ben-Gurion University of the Negev, Beer-Sheva, 84105, Israel

ARTICLE INFO

Keywords:
THERMO-T
System code
Transient analysis
RIA
SPERT-IV
Serpent

ABSTRACT

Best-estimate codes, which couple neutronic and thermal-hydraulic solvers, are mainly used for safety analyses of nuclear power plants. During the past decade, the application of these codes to research reactors gained considerable interest and many improvements were presented to them. The increasing interest in the application of best-estimate codes to safety analyses of research reactor is largely driven by advancements in this field concerning power reactors and the diffusion of knowledge and capabilities to smaller, more diverse systems. The current study is a continuous effort in this framework and presents the coupled neutronic and thermal-hydraulic code development for the analysis of protected and unprotected transient behavior of research reactors. The coupling between neutronic and thermal-hydraulic processes is realized by considering the mutual feedbacks between them; the fuel and coolant properties (temperatures and density) variation affect the core's reactivity and hence the neutronic fission chain reaction, which in turn affects the fuel and coolant properties via a heat generation model for the reactor's power. More specifically, this study deals with the extension of the thermal-hydraulic model to the two-phase flow regime of the THERMO-T code. The extended THERMO-T model is validated against experimental results from the SPERT-IV, which was driven mainly by the coolant density reactivity feedback. This allows a more accurate evaluation of the adequacy of available and relevant two-phase flow models and correlations, which are selected from the domain of large power reactors. This is done in order to encourage and ensure standardization of modeling procedure of all types of reactors as part of the international community's continuous efforts towards this goal.

1. Introduction

Research Reactors (RRs) are unique systems that support a variety of nuclear research needs, including basic nuclear physics and neutron physics, neutron diffraction, material properties, radiation studies, health applications, and more. One of the major roles of RRs is to support research needs of commercial nuclear power reactors. The characteristics of RRs are usually more flexible than those of commercial Nuclear Power Plants (NPPs) and they operate at low thermal power levels (usually not exceeding 100 MW_{th}) and small core sizes, which lead to high power densities, low temperatures of the fuel and clad, and low system pressure (close to atmospheric). Furthermore, the fuel composition and geometric design can be highly heterogeneous. These unique characteristics lead to a variety of different neutronic and thermal-hydraulic designs (D'Auria and Bousbia-Salah, 2006; Hamidouche et al., 2008; Adorni et al., 2006; Adorni et al., 2007), which dictate a wide range of different and unique safety requirements in order to ensure their safe operation. The diversity of different designs make the standardization of operation, regulation and licensing almost impractical (Hamidouche et al., 2008; Costa et al., 2011).

A modeling challenge in calculating power excursion transients in RR is to demonstrate that the numerical models are conservative with respect to the safety limits, i.e., provide a sort of “safe side” approach that would ensure overestimation of damage-indicating parameters (i.e., power, cladding temperature). Employing a conservative approach (approximations and correlations) is common practice in NPPs analysis (due to lack of experimental data, among other reasons), which ensure that the design and operation safety margins are not exceeded. In recent years, the international community acknowledged the importance of implementing the established knowledge and methodologies used for NPPs safety analyses to safety analyses of RRs (Hamidouche et al., 2008; Hamidouche et al., 2004; IAEA, 2007; IAEA, 2008).

In order to test the capabilities of different codes and models in analyzing RRs operation and transients, several benchmark problems were proposed by the International Atomic Energy Agency (IAEA). One of the first benchmarks proposed is the 10 MW_{th} Material Test Reactor

* Corresponding author.
E-mail address: gilade@bgu.ac.il (E. Gilad).

https://doi.org/10.1016/j.pnucene.2018.07.005
Received 14 February 2018; Received in revised form 13 June 2018; Accepted 16 July 2018

Acronyms

DNB	Departure from Nucleate Boiling
HEU	Highly Enriched Uranium
IAEA	International Atomic Energy Association
LOFA	Loss-of-Flow Accident
MC	Monte Carlo
MTR	Material Test Reactor
NEA	Nuclear Energy Agency
NPP	Nuclear Power Plant
RIA	Reactivity Insertion Accident
RR	Research Reactor
SIMPLE	Semi-Implicit Method for Pressure Linked Equations
SPERT	Special Power Excursion Reactor Test

(MTR) (IAEA, 1980; IAEA, 1992), which included information for code verification against burnup calculations, static power and flux distributions and information regarding transient analysis of Reactivity Insertion Accidents (RIA) and Loss-of-Flow Accidents (LOFA) for different fuel compositions (high and low enriched uranium). This benchmark has been introduced in the framework of the Reduced Enrichment for Research and Test Reactors (RERTR) Program. This benchmark is a purely numerical approximation of a hypothesized MTR core, which was utilized for the verification of the system code THERMO-T (Margulis and Gilad, 2016a).

In recent years, the IAEA 10 MW_{th} MTR benchmark tends to be considered obsolete to some extent for code validation. This is in view of recent activities of the IAEA and the Nuclear Energy Agency (NEA), which aim at introducing proper benchmark problems that would be based on experimental data deduced from experiments performed in RRs. A series of such benchmarks were made available in the framework of the IAEA CRP 1496 (IAEA, 2013) and published in 2015 as IAEA Technical Reports Series No. 480 (IAEA, 2015). The report contains experimental data gathered from different RRs such as ETRR-2 (Egypt), IEA-R1 (Brazil), Minerve (France), SPERT III and IV (USA), and more. The report includes both RIA and LOFA experimental measured data and is intended to be used as code validation benchmark (Chatzidakis et al., 2013, 2014; Hainoun et al., 2014). This work focuses on the SPERT-IV destructive test series.

The SPERT-IV experiment aimed at studying the unique dynamic behavior of a RR system by the performance and analysis of reactor kinetic experiments. The SPERT-IV D-12/25 core was the final aluminum plate-type core studied as part of the Special Power Excursion Reactor Test (SPERT) project. The experimental details are summarized in section 2. The experiments were designed to push the RR system to its limits, with the final experiment of a complete withdrawal of all the control rods. The characteristics of the experiment provide a good (yet challenging) platform for the evaluation of the two-phase flow models utilized in the different codes. This is a result of the utilization of Highly Enriched Uranium (HEU) fuel, which practically eliminates the Doppler reactivity coefficient and emphasizes reactivity coefficient of the coolant/moderator.

The main goal of this paper is to estimate the performance of the correlation implemented in the THERMO-T system code (Margulis and Gilad, 2015, 2016b). In previous studies, the THERMO-T was compared to state of the art codes such as RELAP5, PARET, RETRAC-PC, and COBRA-EN, in the frame of the IAEA 10 MWth MTR benchmark (Margulis and Gilad, 2016b). However, those studies did not include two-phase flow capabilities comparison nor did they include any experimental measured data. Thus, in support of the IAEA activities, the focus in this article is put on the utilization of common practice methodology for the analysis of two-phase flow in commercial power systems (Todreas and Kazimi, 1990a) in the THERMO-T system code for RRs analysis. This is made through the validation of those models against experimental data available from SPERT-IV program.

The current work falls in line with efforts of the nuclear community to utilize experimental data for code validation, lead by the activities of the IAEA and the OECD's Nuclear Energy Agency (NEA). In recent years, as a result of the activities of the two agencies, a substantial experimental data for code validation became available (IAEA, 2015; NEA, 2017), and more data to become available in the near future, e.g., the BEAVRS (Horelik et al., 2013), MSRE (Fratoni, 2017) and SNEAK-12 (Margulis et al., 2017) benchmark problems. However, efforts to produce higher quality experimental data for codes and design validation are constantly under investigation. For example, high representative experimental programs that will provide reactor designers and operators with experimental feedback. One such a program is currently under investigation in a collaboration between CEA Cadarache and Ben-Gurion University of the Negev on the studies of neutronic characteristics during severe accidents in Gen-IV reactors (Margulis et al., 2018).

2. Methodology

This section summarizes the tools, methods and models that are utilized in this work. It includes a short description of the SPERT-IV core, a short description of the Serpent Monte Carlo (MC) code and an overview of the THERMO-T code extended to two-phase flow and heat transfer models.

2.1. SPERT-IV d-12/25

SPERT-IV was a light water cooled and moderated pool-type reactor, with upward forced and natural convection cooling. The core was composed of 25 fuel assemblies, 20 standard, and 5 control fuel assemblies. The different fuel assemblies are placed in a 5 × 5 section of the 9 × 9 support grid, as shown in Figs. 1 and 2. The reactor was loaded with highly enriched uranium (HEU) plate-type fuel cased in an aluminum cladding (UAl_x-Al). Each standard fuel assembly contained 12 fuel plates, housed in an aluminum assembly can. The four control assemblies and the single transient rod are made of double-blade control rods of the same style, but with a different operational direction. The control rods were extracted in the upward direction, while the

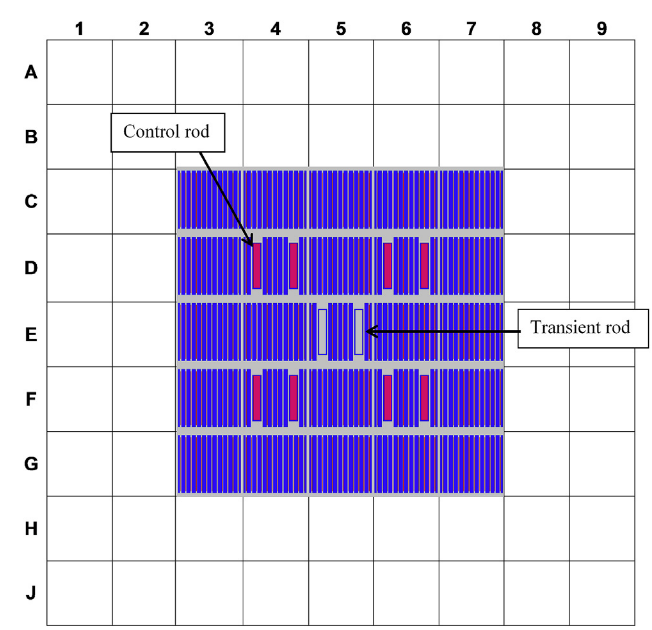

Fig. 1. Schematic representation of the SPERT-IV D-12/25 core loading.