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Benchmark exercise for fluid flow simulations in a liquid metal fast reactor fuel assembly

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HIGHLIGHTS

- A EUROTAM-US INERI consortium has performed a benchmark exercise related to fast reactor assembly simulations.
- LES calculations for a wire-wrapped rod bundle are compared with RANS calculations.
- Results show good agreement for velocity and cross flows.

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ABSTRACT

As part of a U.S. Department of Energy International Nuclear Energy Research Initiative (I-NERI), Argonne National Laboratory (Argonne) is collaborating with the Dutch Nuclear Research and consultancy Group (NRG), the Belgian Nuclear Research Centre (SCK·CEN), and Ghent University (UGent) in Belgium to perform and compare a series of fuel-pin-bundle calculations representative of a fast reactor core. A wire-wrapped fuel bundle is a complex configuration for which little data is available for verification and validation of new simulation tools.

UGent and NRG performed their simulations with commercially available computational fluid dynamics (CFD) codes. The high-fidelity Argonne large-eddy simulations were performed with Nek5000, used for CFD in the Simulation-based High-efficiency Advanced Reactor Prototyping (SHARP) suite. SHARP is a versatile tool that is being developed to model the core of a wide variety of reactor types under various scenarios. It is intended both to serve as a surrogate for physical experiments and to provide insight into experimental results.

Comparison of the results obtained by the different participants with the reference Nek5000 results shows good agreement, especially for the cross-flow data. The comparison also helps highlight issues with current modeling approaches.

The results of the study will be valuable in the design and licensing process of MYRRHA, a flexible fast research reactor under design at SCK·CEN that features wire-wrapped fuel bundles cooled by lead-bismuth eutectic.

1. Introduction

Nuclear power plays an important role in power generation, producing about 16% of the total electricity worldwide. The rapidly growing energy demand suggests an even more important role for nuclear power in the future energy supply, as projected by the World Energy Outlook 2013 (IAE, 2013). Arguably, the accident at the Fukushima Daiichi nuclear power plant in Japan in March 2011 did have a minor effect on the future demand for nuclear power. Nevertheless, IAE (2013) calculates that nuclear power will be maintaining a 12% share of electricity generation globally by 2035, with expansion mainly in Asia. In Europe, in the Vision Report

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Nomenclature

n	polynomial order, in the spectral-element method
D_h	hydraulic diameter
np	total number of collocation points, in the spectral-element method
p	number of processors
P	center-to-center pin pitch
g	gap size ($g = P–D$)
P/D	pitch-to-diameter ratio
F	flat-to-flat
H	axial wire pitch
H/D	ratio of axial wire pitch to diameter
u_{bulk}	bulk velocity
Re	Reynolds number, based on hydraulic diameter
Re_D	Reynolds number, based on pin diameter
ν	kinematic viscosity
x,y,z	Cartesian coordinates
u,v,w	velocity components in the Cartesian coordinates
u',v',w'	root mean square of the velocity fluctuations in the Cartesian coordinates
k	turbulent kinetic energy
ε	dissipation of turbulent kinetic energy
ω	rate of dissipation of the turbulent kinetic energy
E	average error

(SNE-TP, 2007) of the Sustainable Nuclear Energy Technology Platform (SNETP), a large role is attributed to the deployment of fast reactors. The preferred option is the sodium-cooled fast reactor, with the lead-cooled fast reactor as one of the two backups. Clearly, then, liquid metals will be important in the development of future nuclear energy technologies. A detailed overview of the status of fast reactor development is given in the IAEA (2012) report.

Thermal hydraulics is one of the key scientific factors in the design and safety analysis of liquid metal-cooled reactors. To solve thermal-hydraulic issues, nuclear engineers apply experiments, analytical and empirical correlations, and system thermal hydraulics codes or subchannel codes. Additionally, computational fluid dynamics (CFD) techniques are becoming increasingly integrated in the daily practice of the thermal-hydraulics researchers and designers. Roelofs et al. (2013b) summarize the current status and future challenges for CFD application to liquid-metal-cooled fast reactors. They show that for many liquid metal fast reactor thermal-hydraulic issues, the validation of CFD techniques is and will remain a key issue. In general, they underscore the simultaneous need for developments with respect to experiments including measurement techniques and numerical simulations.

Under the U.S. Department of Energy's International Nuclear Energy Research Initiative (I-NERI), Argonne National Laboratory (Argonne) collaborates with three Euratom members: the Dutch Nuclear Research and consultancy Group (NRG), the Belgian Nuclear Research Centre (SCK·CEN), and Ghent University (UGent) in Belgium on simulations of nuclear reactor core flows. The aim is to share data produced by the partners involved in order to systematically cross-verify fluid-dynamic simulations in liquid-metal-cooled nuclear reactor fuel assemblies. This collaboration focuses on code-to-code CFD comparisons in the absence of CFD-grade experimental data for wire-wrapped fuel assemblies.

Most liquid-metal-cooled fast reactor designs employ wire wraps as spacers between the individual pins in a rod bundle. Yet although many experiments have been performed, Roelofs et al. (2013a) clearly demonstrate that CFD-grade validation data is not available. New thermal-hydraulic experiments are under preparation in Germany, Italy, and the United States to fill this gap. To gain confidence in their employed Reynolds-averaged Navier–Stokes (RANS) approaches, the partners in this collaboration compare their results from RANS approaches with data from high-fidelity large eddy simulation (LES) performed at Argonne in a blind benchmark. Explanations of the various CFD modeling approaches are given by Roelofs et al. (2013a). These explanations basically show that direct numerical simulation and LES can provide high-fidelity reference data for comparison with more pragmatic RANS or hybrid approaches. The current paper describes the simulation efforts shared by the collaborating partners. Discrepancies or concerns with current prediction technologies are identified. Furthermore, an experimental plan for validation is under preparation taking into account the concerns that emerged from this collaboration.

The study is part of the code validation and verification approach developed in the licensing process of the Multi-purpose hYbrid Research Reactor for High-tech Applications (MYRRHA) under design at SCK·CEN (Abderrahim, 2012). MYRRHA is a flexible fast spectrum research reactor with wire-wrapped fuel bundles cooled by lead-bismuth eutectic. MYRRHA is identified as the European technology pilot plant for the lead-cooled fast reactor (LFR), which is one of the Generation IV reactor concepts (SNE-TP, 2010).

Argonne performed several wire-wrapped analyses as part of the Nuclear Engineering Advanced Modeling and Simulation (NEAMS) initiative (Pointer et al., 2008, 2009; Smith et al., 2008). In particular, a simulation of a 7-pin wire-wrapped fuel bundle was performed with the LES code Nek5000. These calculations are used as the basis for a blind benchmark calculation among the collaborating partners. We emphasize that although pin counts in actual liquid metal assemblies are much higher, code comparisons in small assemblies are useful for building confidence in numerical and modeling practices at an early stage. Code-to-code comparisons in large assemblies are computationally expensive and present significant logistic challenges because of the large amount of data and the presence of multiple physical phenomena. Nevertheless, such comparisons in progressively larger assemblies are planned (see Section 6).

The present article describes the Nek5000 code (Section 2), briefly discusses the benchmark exercise (Section 3), and presents the comparisons of the benchmark datasets (Section 4) with the Argonne high-fidelity reference LES data. The results confirm previous findings (i.e., the good performance of the k–ω SST model in this type of modeling) and highlight the importance of consistent geometry in code-to-code comparisons.

2. Computational tools

Here we describe in detail SHARP and Nek5000 the tools used to perform the reference LES calculations.

2.1. SHARP suite

The Simulaton-based high-Fidelity Advanced Reactor Prototyping (SHARP) project (Siegel et al., 2007; Mahadevan et al., 2014) at Argonne is a multidivisional effort to develop a modern set of design and analysis tools for advanced nuclear reactors. SHARP is an integral part of the NEAMS Reactor Product Line. With the SHARP suite, users construct complex virtual reactor models that accurately integrate the governing physics in order to evaluate the performance of the reactor in a wide variety of operational or accident scenarios. Alternatively, SHARP users may construct highly detailed component models using high-fidelity methods that rely on few or no engineering models or approximations. The thermal-hydraulic high-fidelity component of SHARP is Nek5000 (Fischer et al., 2007, 2008), a high-order spectral-element code ideally suited for LES or