HIGH PERFORMANCE COMPUTING FOR NUCLEAR REACTOR DESIGN AND SAFETY APPLICATIONS

Lecture, NCBJ, Poland

A. Shams* - NRG

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EU DuC=N

Presentation Plan

- Introduction: NRG-NCBJ Collaboration
- Applications

  - Pressurized Thermal Shock

  - Bare rod bundle
- Achievements: NRG-NCBJ Collaboration (so far)

Introduction: NRG-NCBJ Collaboration

NRG-NCBJ Collaboration: Phase-I

- 2013

- P. Prusinski

NRG-NCBJ Collaboration: Phase-I

- 2013

- Head of the team:

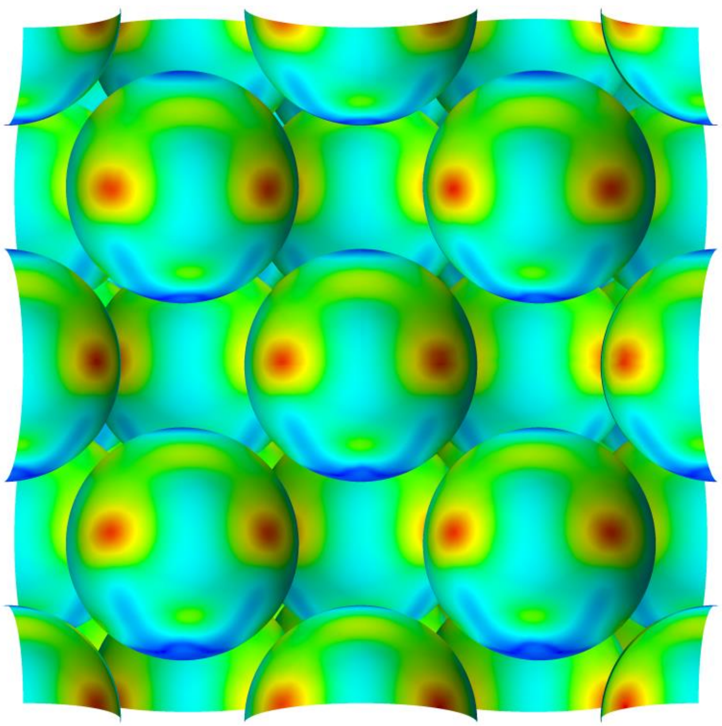
- **S. Potempski**

- Other NCBJ Colleagues:

- P. Prusinski
- T. Kwiatkowski

NRG-NCBJ Collaboration: Phase-I

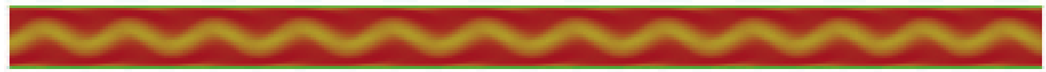
- 2013
- Pebble bed
- Head of the team:
- **S. Potempski**
- Other NCBJ Colleagues:
- P. Prusinski
- T. Kwiatkowski



Simely Pebbles: RANS @ STAR-CCM+

NRG-NCBJ Collaboration: Phase-I

- 2013
- Pebble bed
- Rod bundle



- Head of the team:
- **S. Potempski**

- Other NCBJ Colleagues:
- P. Prusinski
- T. Kwiatkowski

CALIBRATION OF HIGH FIDELITY BARE ROD BUNDLE SIMULATIONS FOR VARIOUS PRANDTL FLUIDS

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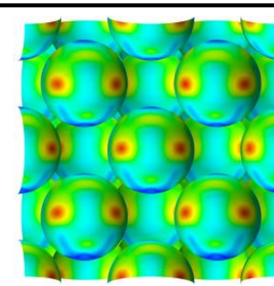
ABSTRACT

One of the key factors in the design of liquid metal cooled fast reactors (LMFRs) refers to the detailed knowledge of the coolant flow in the sub-channels of the fuel assemblies in the reactor's core. In fact, a thorough experimental investigation of the flow mixing and heat transport in such sub-channels is often impossible or quite costly to be performed. Due to these limitations, computational fluid dynamics (CFD) has recently become a valuable tool in the study of the dynamics of the flow within the sub-channel regions of typical nuclear fuel assemblies.

NURETH-16, 2015, Chicago, USA



NRG-NCBJ Collaboration: Phase-II

- 2016 (Formal)

- 2017-2019

- Pressurized Thermal Shock
- Bare Rod Bundle

NRG-NCBJ Collaboration Signing: 2016

NRG-NCBJ Collaboration: Phase-II

- 2016 (Formal)

- 2016-2019
- Pressurized Thermal Shock
- Bare Rod Bundle

NRG-NCBJ Collaboration Signing: 2015

- Main contact person:

- NRG: A. Shams
- NCBJ: T. Kwiatkowski

NRG-NCBJ Collaboration: Phase-II

- 2016 (Formal)

- 2016-2019
- Pressurized Thermal Shock
- Bare Rod Bundle

- Main contact person:

- NRG: A. Shams
- NCBJ: T. Kwiatkowski

ERMSAR 2017 – Photo Credit: Michał Spirzewski

The Godfather

NRG-NCBJ Collaboration: Phase-II

- 2016 (Formal)

- 2016-2019

- Pressurized Thermal Shock
- Bare Rod Bundle

- Main contact person:

- NRG: A. Shams
- NCBJ: T. Kwiatkowski

The Silent Warrior

NRG-NCBJ Collaboration: Phase-II

- 2016 (Formal)

- 2016-2019
- Pressurized Thermal Shock
- Bare Rod Bundle

- Main contact person:

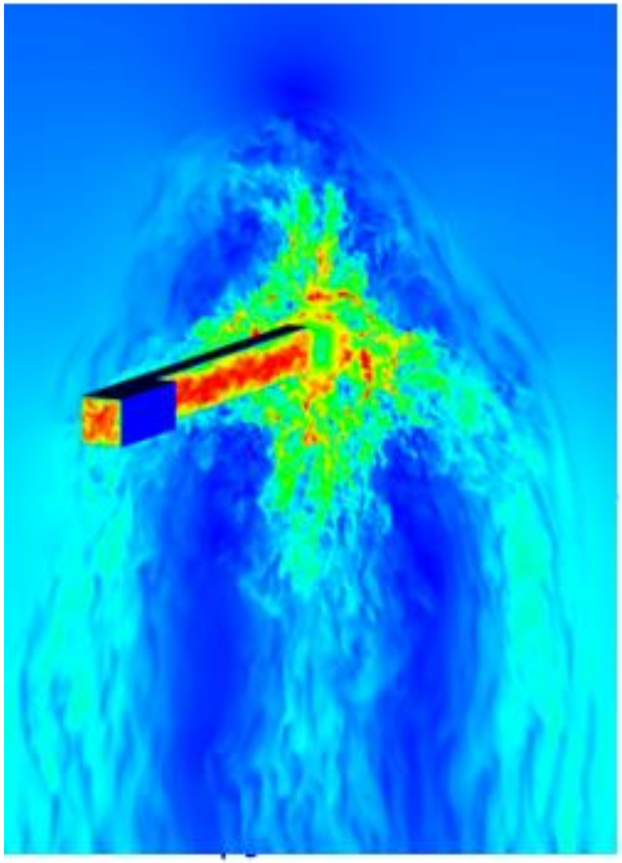
- NRG: A. Shams
- NCBJ: T. Kwiatkowski

Świerk Computing Centre – main building

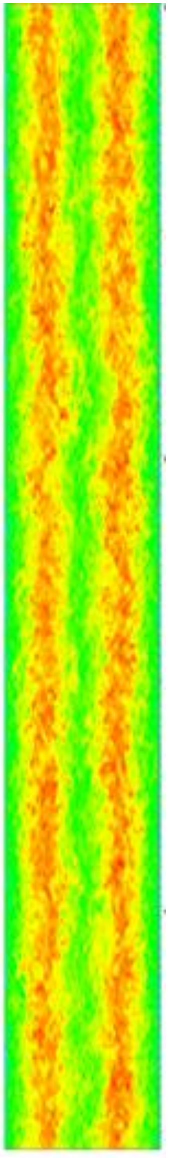
The Catalyst

Prof. W. Wislicki

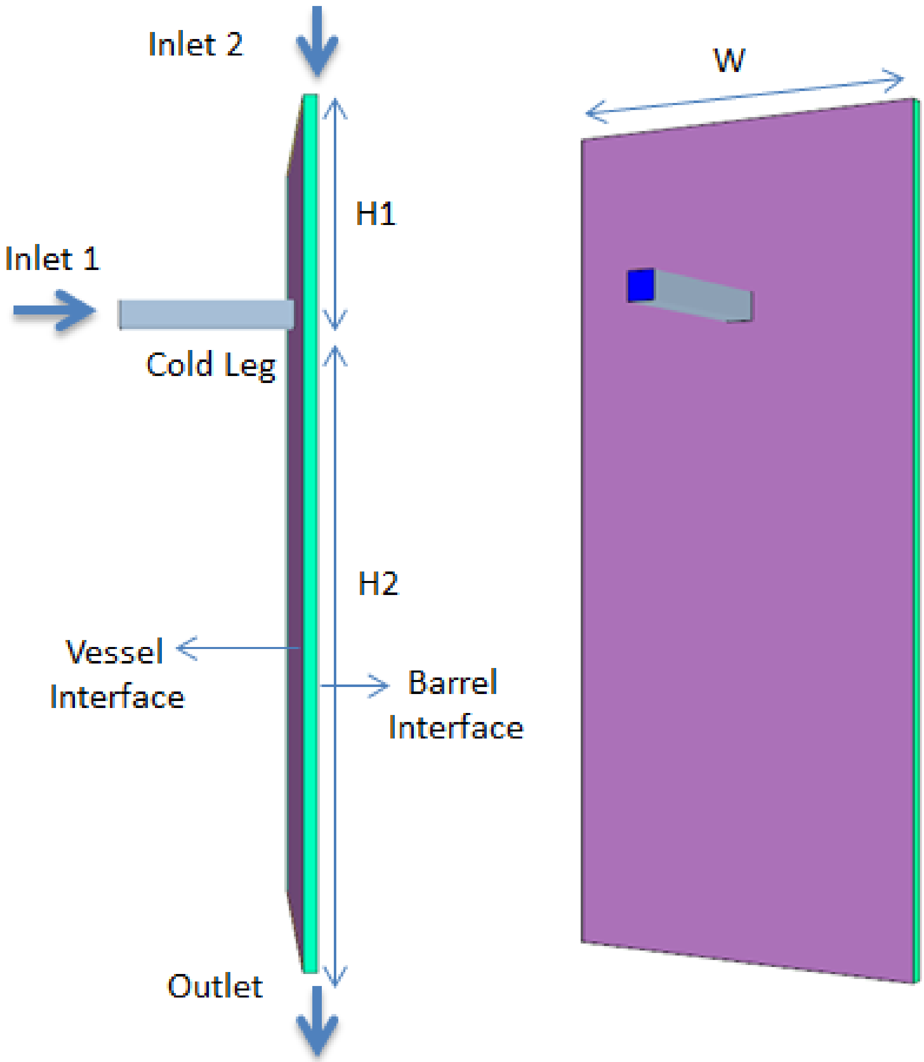
A. Padee, T. Jarosiewicz, P. Wasiuk



Applications



Pressurized Thermal Shock

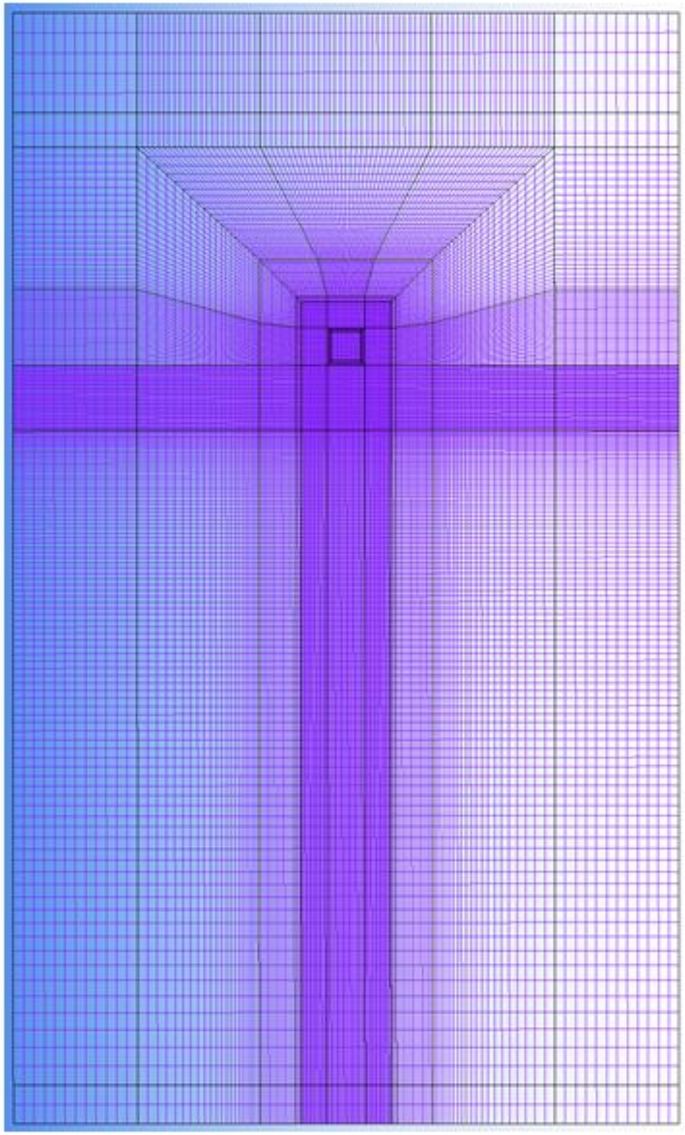


Parameters	Description
Inlet 1	Instead of a round pipe, a duct is considered for the inlet section. In addition, the flow parameters are calibrated to match $Re_\tau = 180$
Inlet 2	In the absence of the buoyancy effects, an artificial inlet 2 is imposed to force the inlet 1 flow towards the down-comer
Cold Leg	The length of the cold leg is 6 times the equivalent diameter (D) of the duct
H1	The upper height of the domain is 10D
H2	The lower height of the down-comer is 23.3D
W	The width of the domain is 20D
Vessel Interface	Instead of a conjugate heat transfer, two boundary conditions are imposed (i) iso-thermal (2) adiabatic
Barrel Interface	Two boundary conditions are imposed (i) iso-thermal (2) adiabatic
Outlet	Pressure outlet

Relevant ref.

Shams et al. (2015): NED - Shams & Komen (2017): IAEA - Shams & Komen (2018): FTaC

Pressurized Thermal Shock



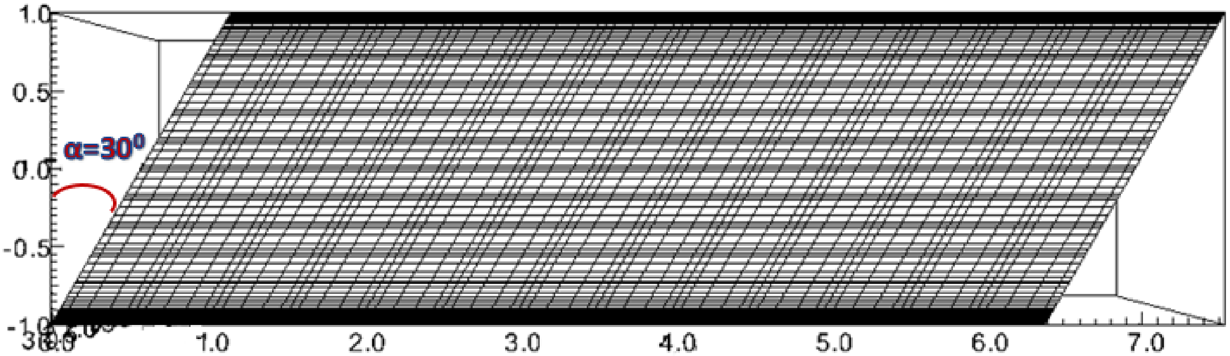
Code – NEK5000
- Spectral Element Code

Mesh:
- 0.75 Million Elements
- 162 Million Degrees of Freedom

Computational Power:
- 40 Million core hours

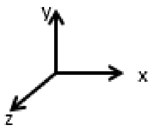
Relevant ref.
Shams & Komen (2018): FTaC

NEK5000: DNS of Skewed meshes

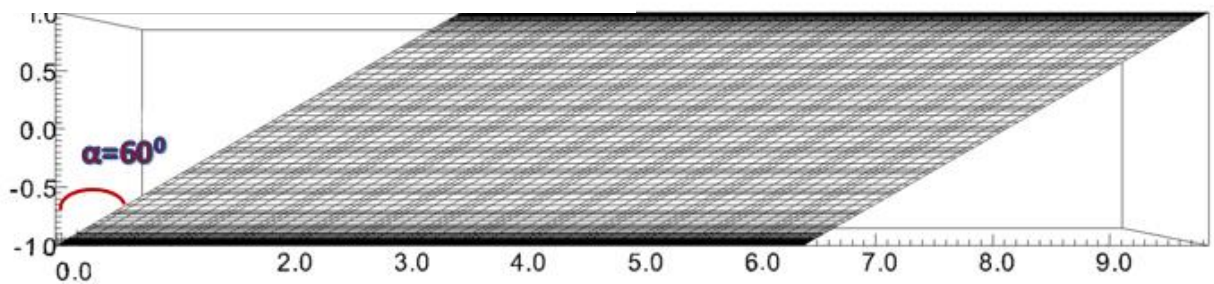


Front view

Re$_\tau$ = 180

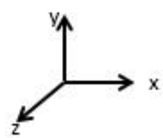


- $\alpha=30^0$; the angle of twist about the z-axis
- $\beta=0^0$; the angle of twist about the x-axis



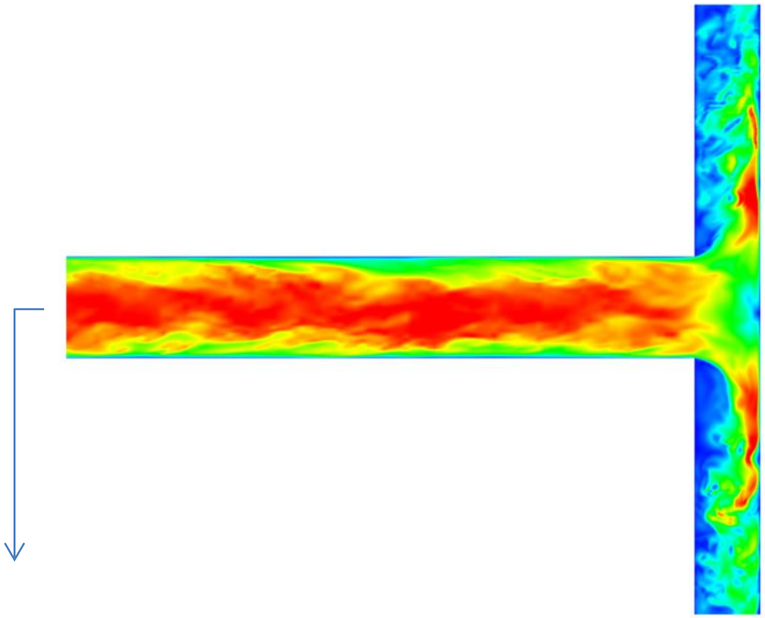
Front view

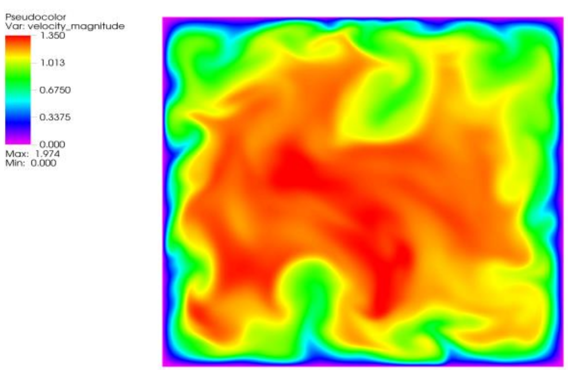
Relevant ref.
Shams & Komen (2018): FTaC

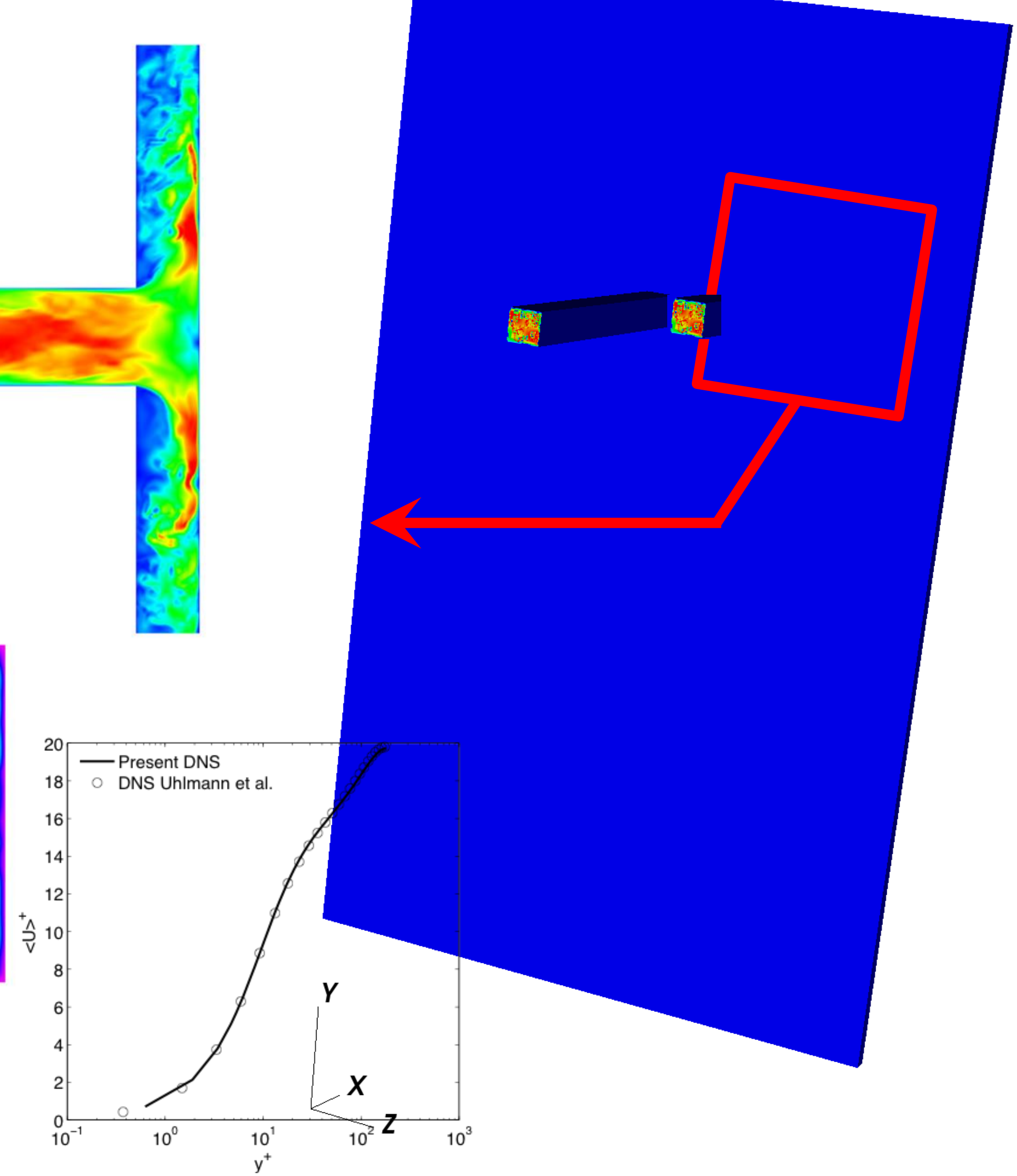


- $\alpha=60^0$; the angle of twist about the z-axis
- $\beta=0^0$; the angle of twist about the x-axis

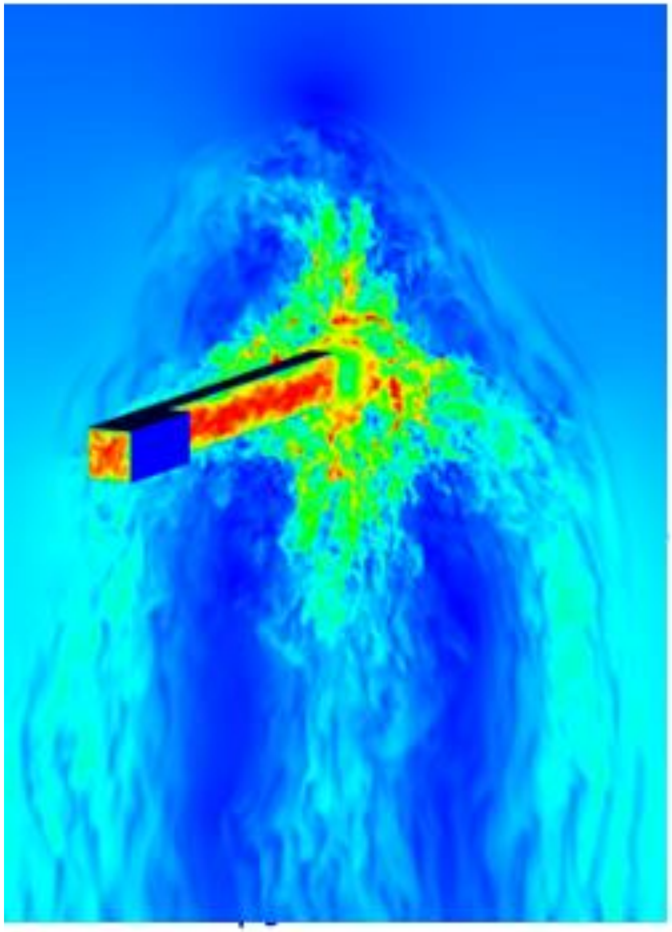
Pressurized Thermal Shock

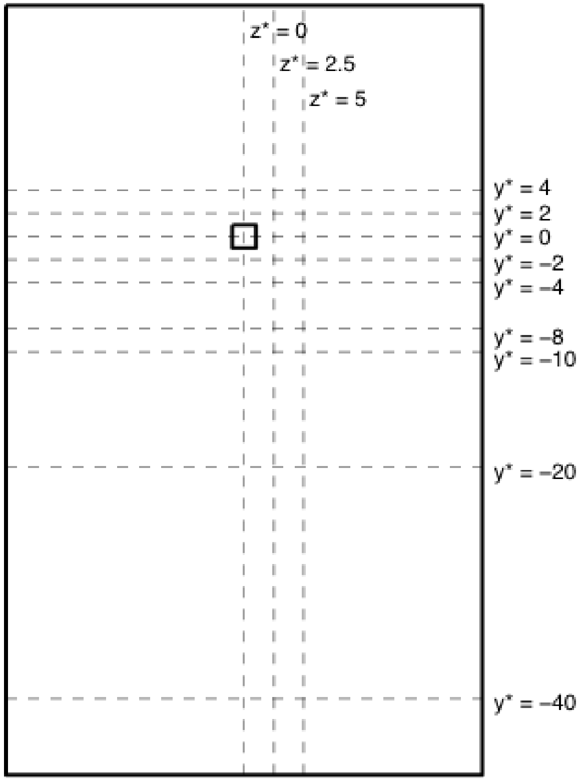


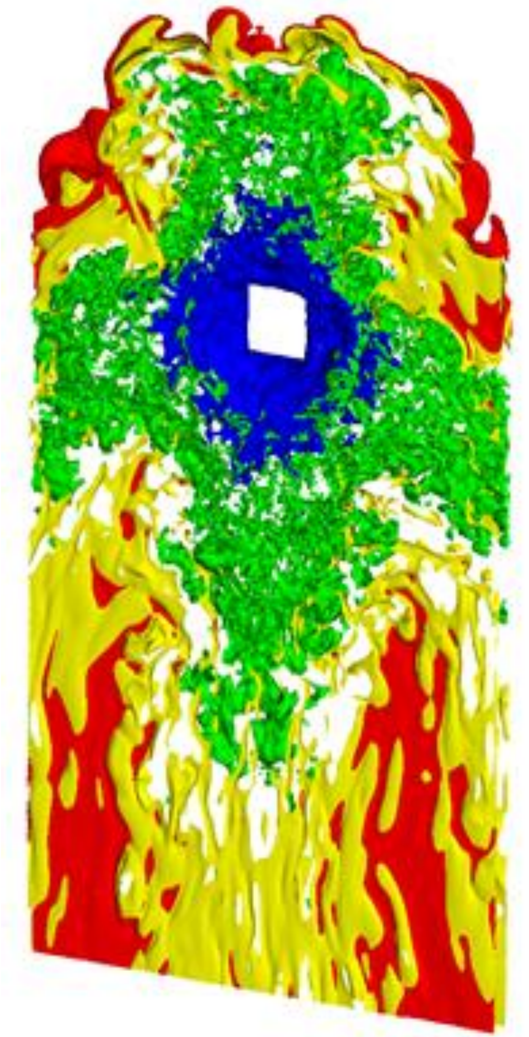




Pressurized Thermal Shock: Flow Field

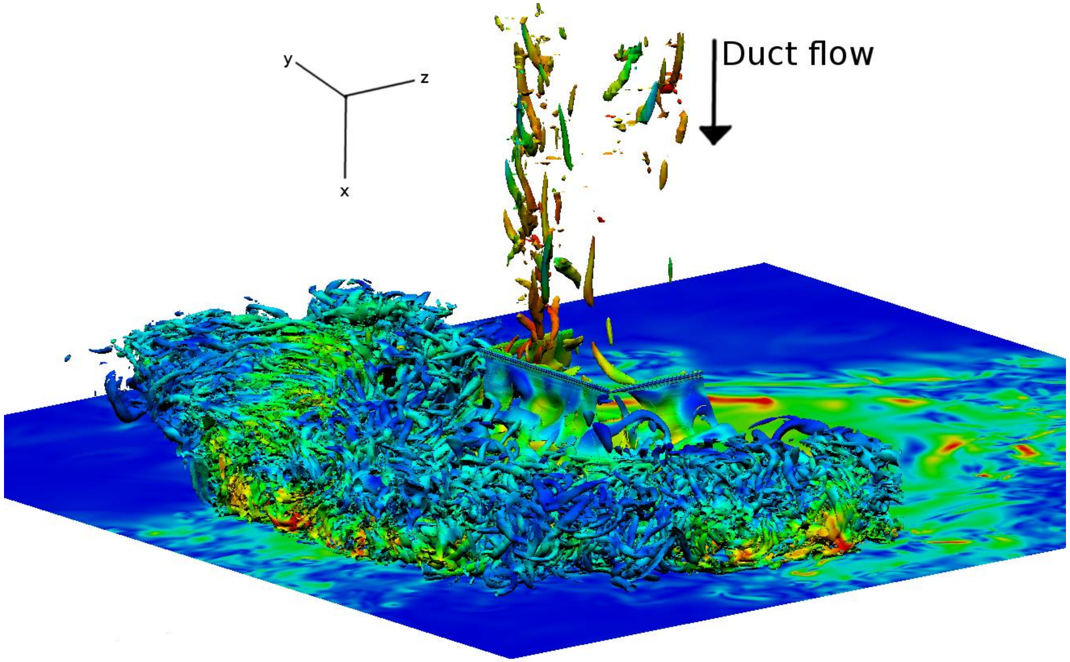


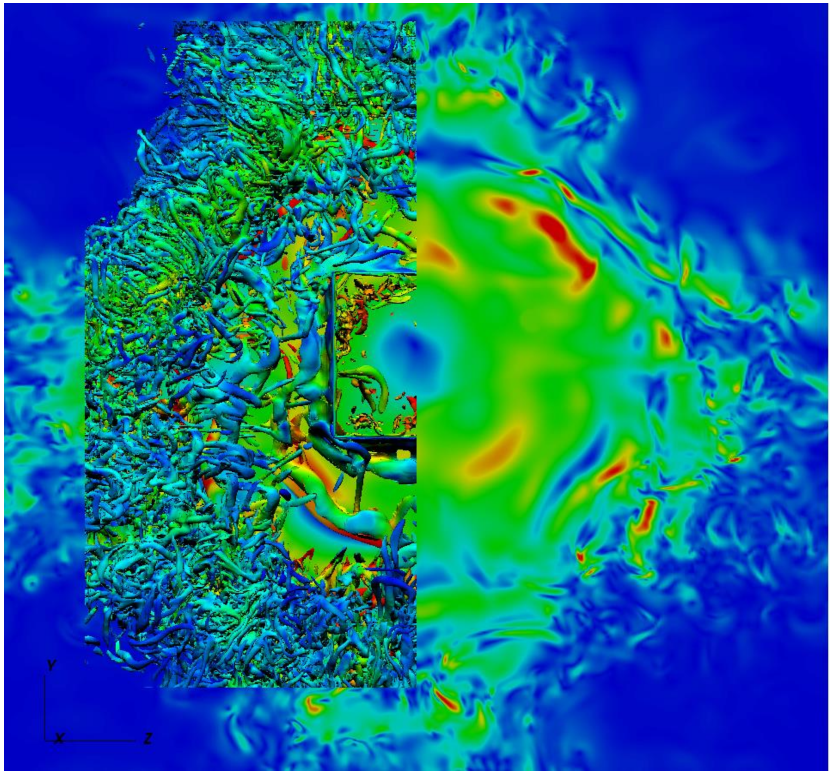




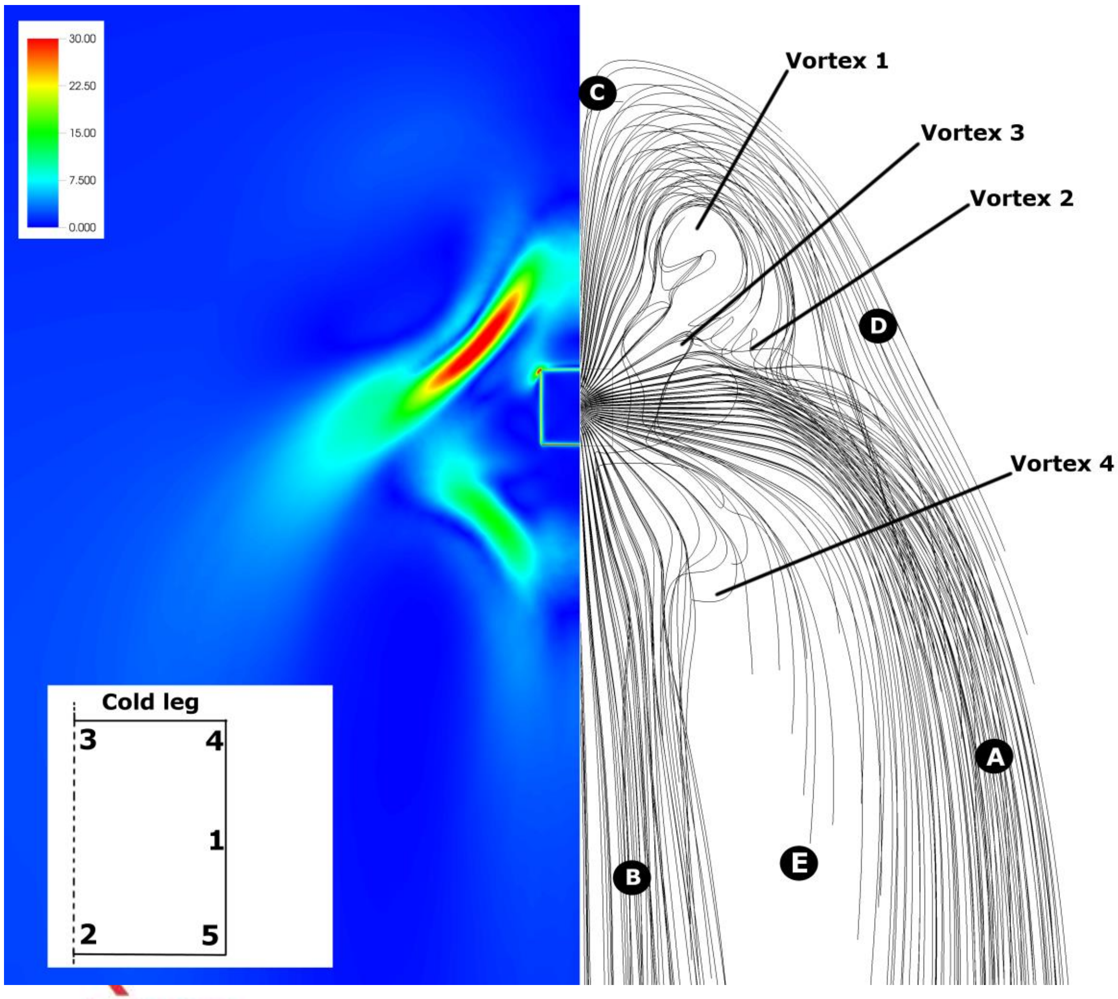
NRG

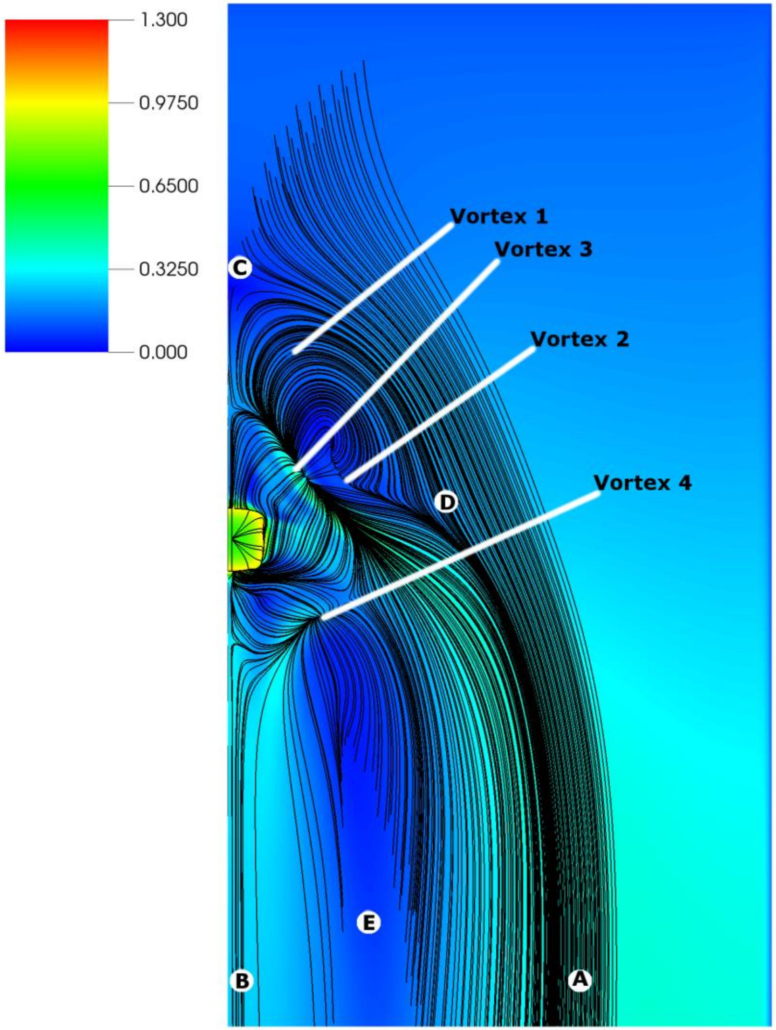
Pressurized Thermal Shock: Flow Field



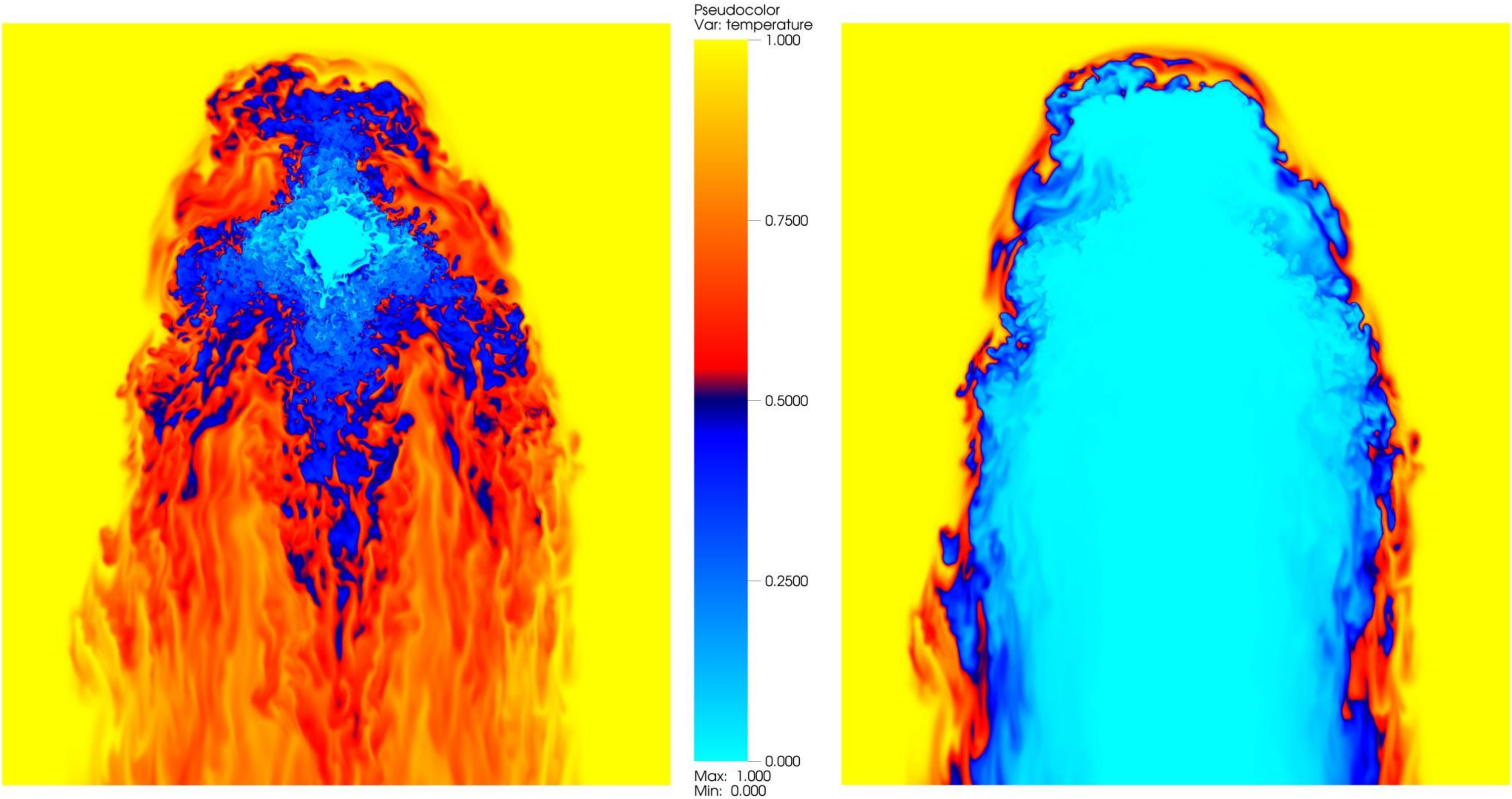


Pressurized Thermal Shock: Flow Field





Pressurized Thermal Shock: Thermal Field



iso-thermal

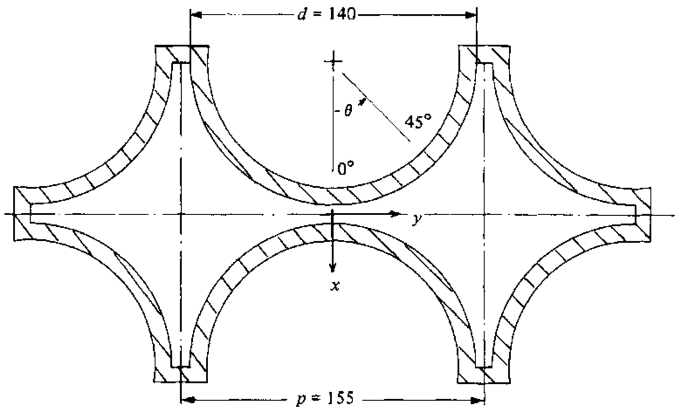
iso-flux

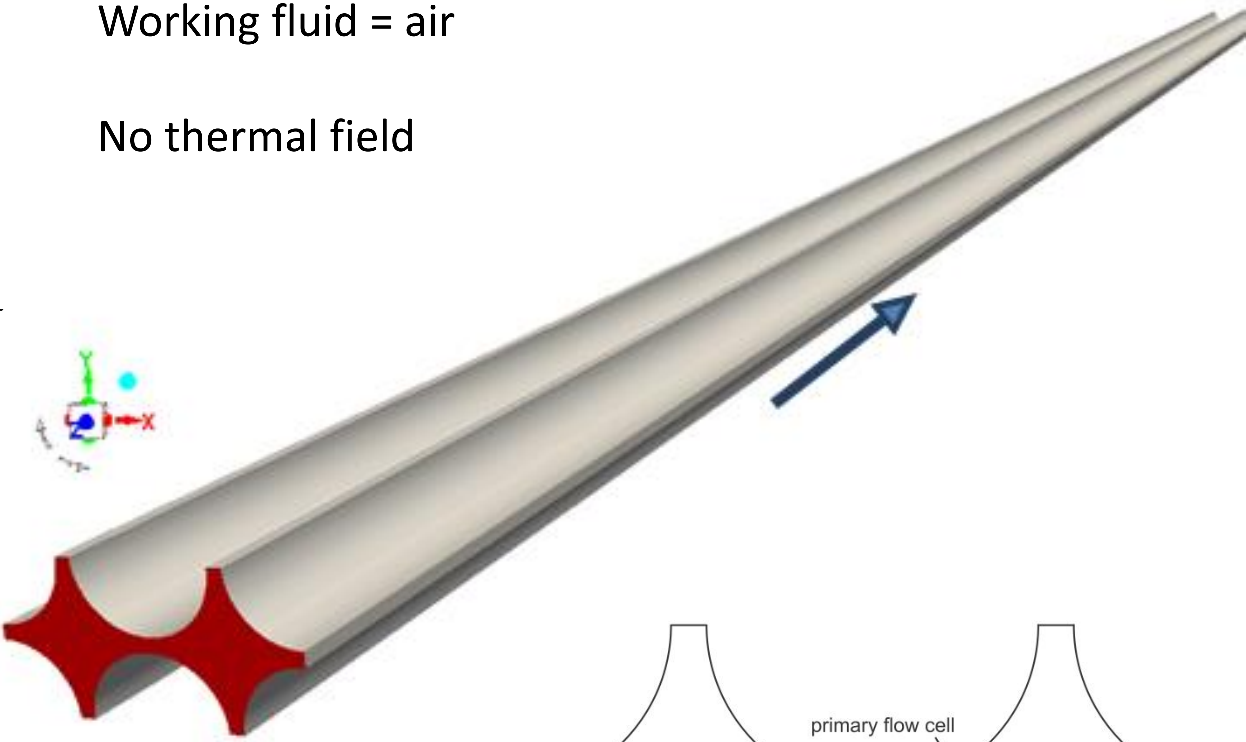
Bare rod bundle: Hooper case

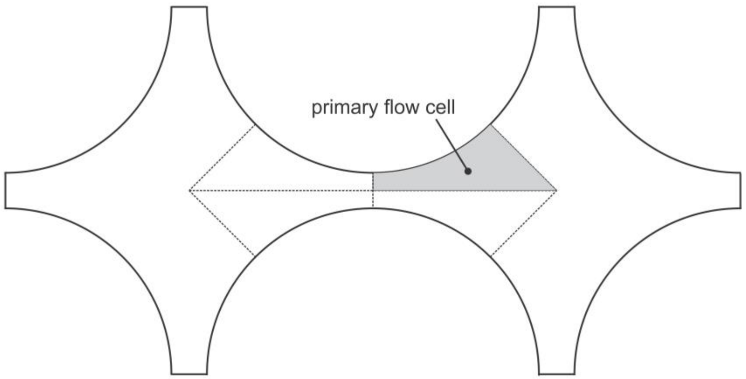
$Re_{bulk} \approx 49000$

Working fluid = air

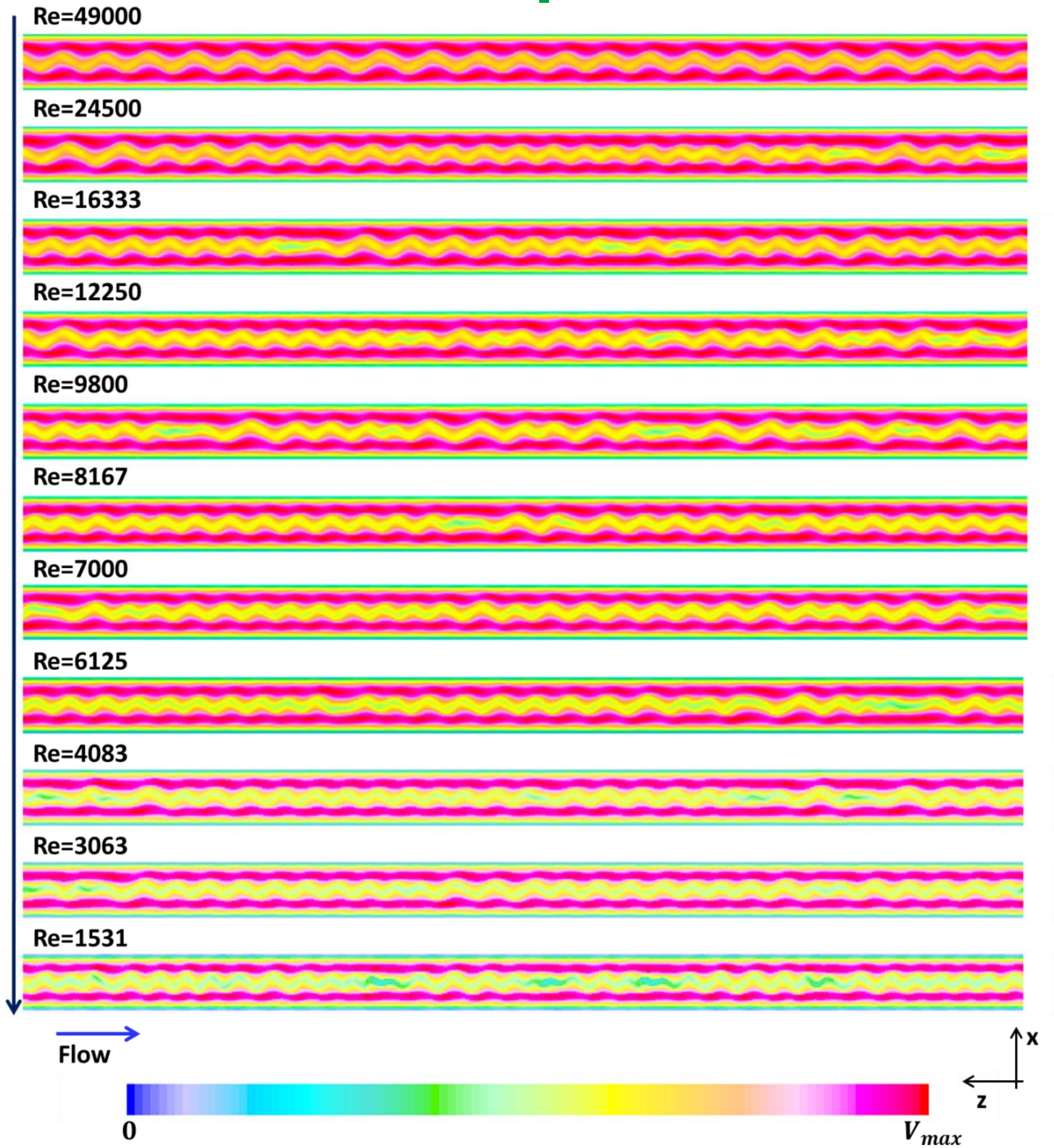
No thermal field



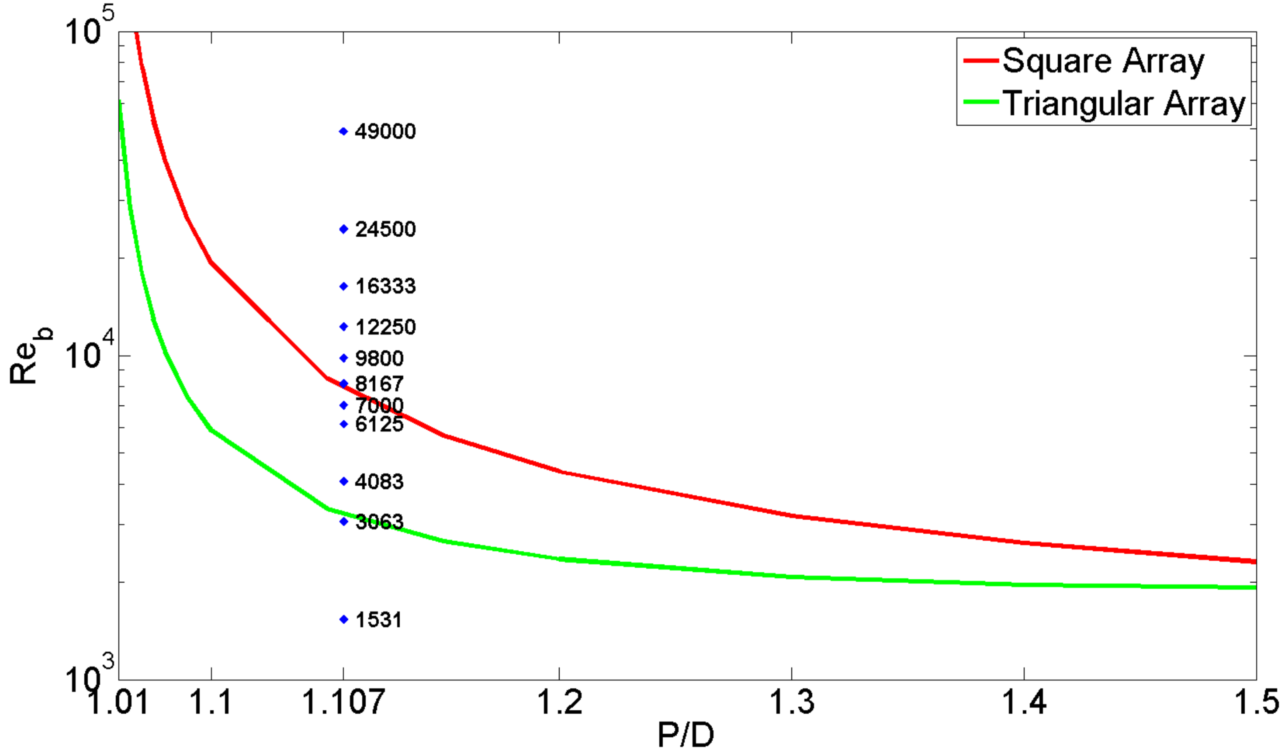




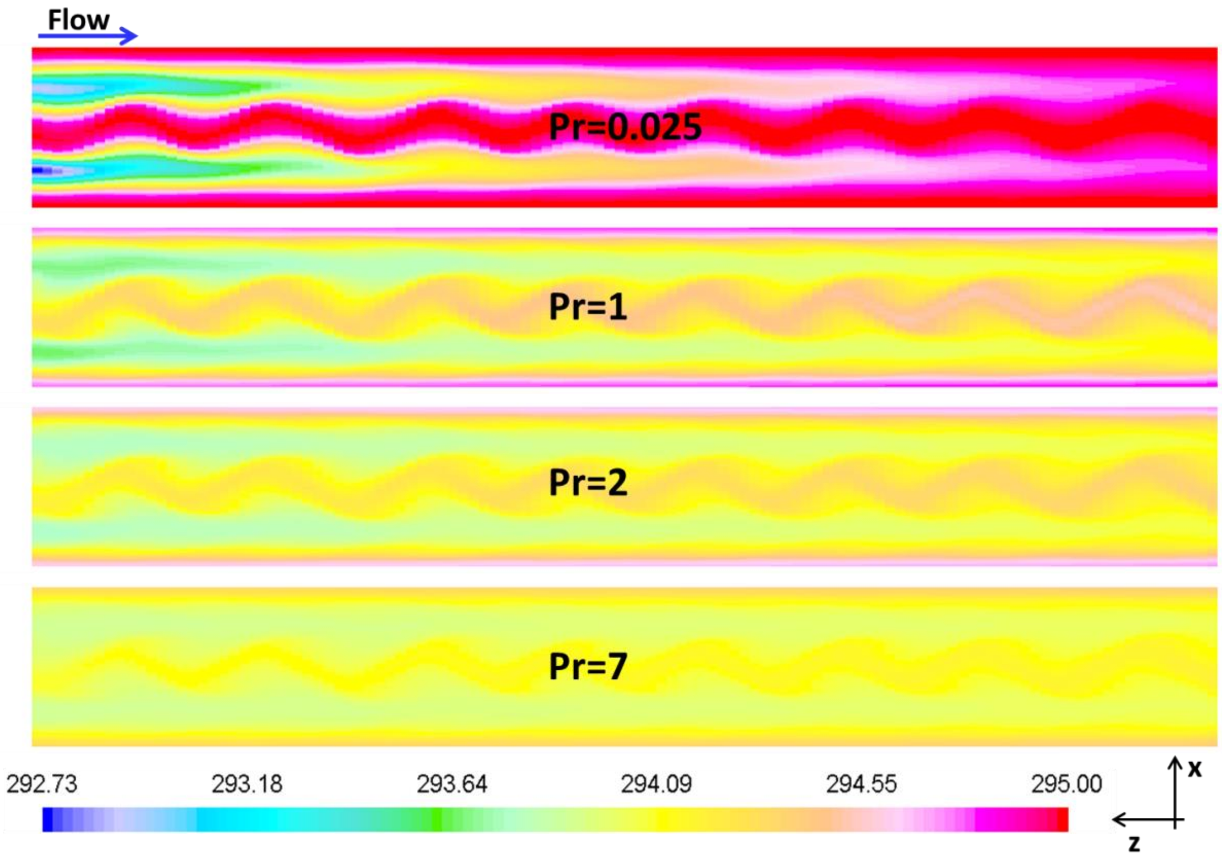
Bare rod bundle: Hooper case calibration



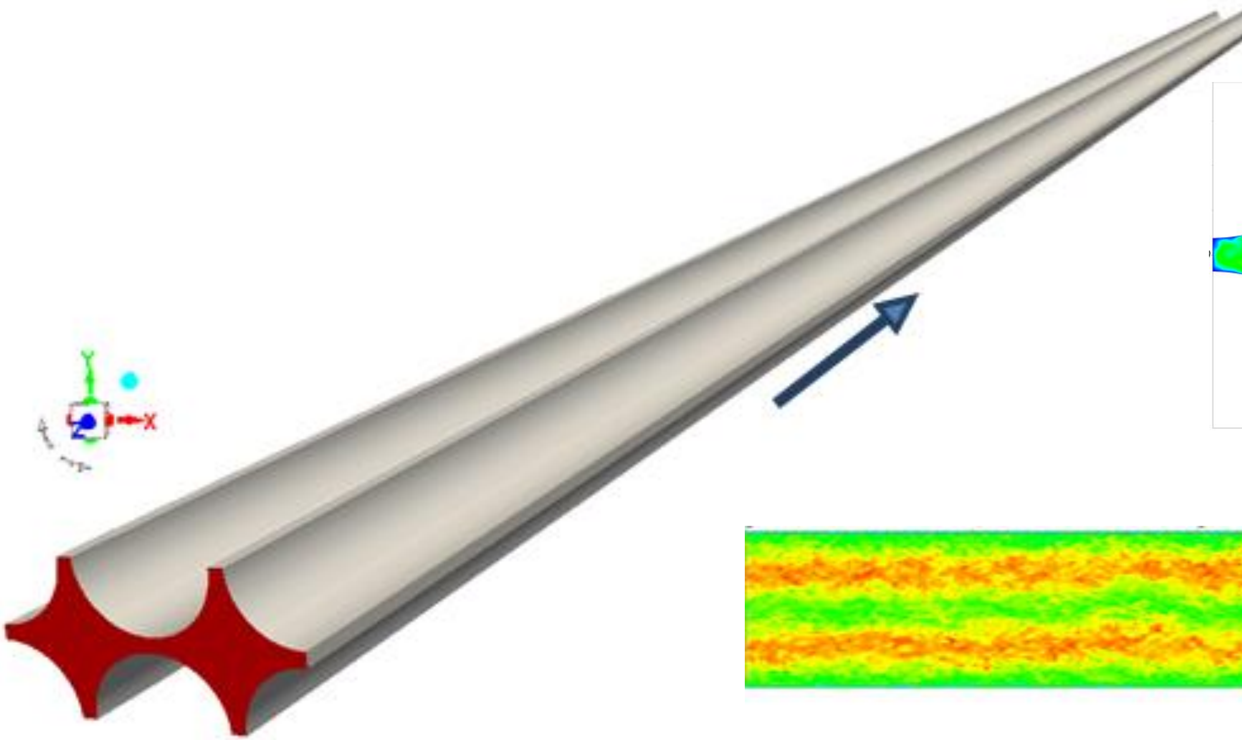
Bare rod bundle: Hooper case calibration



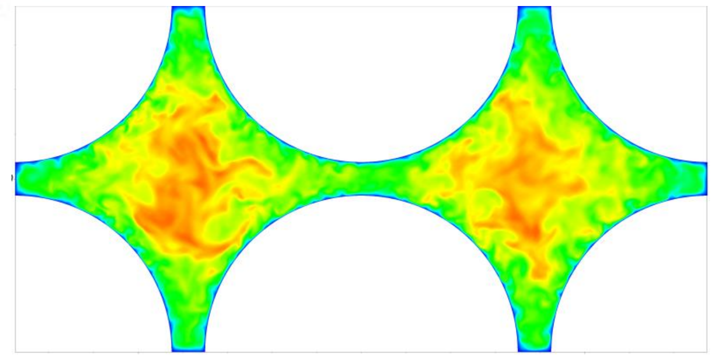
Bare rod bundle: Hooper case calibration

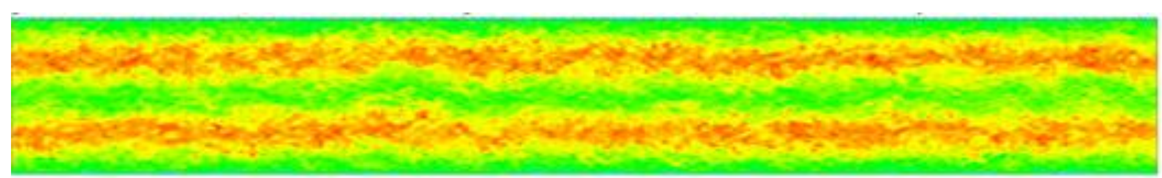


Bare rod bundle: Hooper case DNS



> 65 million core hrs





$Re_\tau \sim 600$

i) Iso-flux ii) iso-thermal

Pr = 7, 2, 1 & 0.025

Code – NEK5000
- Spectral Element Code

Mesh:
- 1 Million Elements
- 660 Million Degrees of Freedom

Achievements: NRG-NCBJ Collaboration (so far)

2015 - NURETH-16

CALIBRATION OF HIGH FIDELITY BARE ROD BUNDLE SIMULATIONS FOR VARIOUS PRANDTL FLUIDS

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ABSTRACT

One of the key factors in the design of liquid metal cooled fast reactors (LMFRs) refers to the detailed knowledge of the coolant flow in the sub-channels of the fuel assemblies in the reactor's core. In fact, a thorough experimental investigation of the flow mixing and heat transport in such sub-channels is often impossible or quite costly to be performed. Due to these limitations, computational fluid dynamics (CFD) has recently become a valuable tool in the study of the dynamics of the flow within the sub-channel regions of typical nuclear fuel assemblies.

2018 – ANE @ IF=1.38

Annals of Nuclear Energy 121 (2018) 146–161

Contents lists available at ScienceDirect

Annals of Nuclear Energy

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

Towards the Direct Numerical Simulation of a closely-spaced bare rod bundle

Afaque Shams [a,*], Tomasz Kwiatkowski [b]

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ARTICLE INFO

Article history:
Received 1 March 2018
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ABSTRACT

The aim of the present research work is to design a numerical experiment for a closely-spaced bare rod bundle in order to perform a DNS, which will serve as a reference for the verification purpose. The considered geometric design is based on the well-known Hooper experiment, which contains a bare rod bundle with pitch-to-diameter ratio of P/D = 1.107. However, performing a DNS computation corresponding

2018 – ICONE26

Proceedings of the 2018 26th International Conference on Nuclear Engineering
ICONE26
July 22-26, 2018, London, England

ICONE26-81049

DESIGN OF A CLOSELY-SPACED ROD BUNDLE FOR A REFERENCE DIRECT NUMERICAL SIMULATION

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Petten, The Netherlands

Tomasz Kwiatkowski
National Centre for Nuclear Research
Otwock-Swierk, Poland

2018 – IJHMT @ IF=3.89

International Journal of Heat and Mass Transfer 127 (2018) 345–358

Contents lists available at ScienceDirect

International Journal of Heat and Mass Transfer

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

The influence of low Prandtl numbers on the turbulent mixed convection in an horizontal channel flow: DNS and assessment of RANS turbulence models

Dante De Santis [a], Andrea De Santis [a], Afaque Shams [a,*], Tomasz Kwiatkowski [b]

[a] *Nuclear Research and Consultancy Group NRG, Westerduinweg 3, 1755 LE Petten, The Netherlands*
[b] *National Center for Nuclear Research, Andrzeja Soltana 7, 05-400 Otwock-Świerk, Poland*

ARTICLE INFO

Article history:
Received 15 March 2018
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Accepted 30 July 2018

ABSTRACT

In this paper, the turbulent mixed convection in a channel flow with differentially heated walls is considered for a fixed Richardson number ($Ri_b = 0.5$) and three different Prandtl numbers ($Pr = 1$, 0.1 and 0.01). Numerical simulations are conducted assuming constant fluid properties and the effect of the buoyancy is taken into account by means of the Boussinesq approximation. Direct Numerical Simulations (DNS) are

NRG

2018 – ATH18

HIGH PERFORMANCE COMPUTING FOR NUCLEAR REACTOR DESIGN AND SAFETY APPLICATIONS

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ABSTRACT

This paper presents technical aspects of large-scale direct numerical simulation (DNS) using high performance computing (HPC) cluster. It is divided in two parts. The first part contains a detailed description of HPC infrastructure used for the task, located in Świerk Computing Centre (CIŚ), Poland. The description includes hardware configuration, software used for on-demand deployment of dedicated subclusters, and finally queuing and storage systems. The second part presents the large-scale simulations

2019 – ATH18 > Nuclear Technology: under-review

HIGH PERFORMANCE COMPUTING FOR NUCLEAR REACTOR DESIGN AND SAFETY APPLICATIONS

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ABSTRACT

This paper presents technical aspects of large-scale direct numerical simulation (DNS) using high performance computing (HPC) cluster. It is divided in two parts. The first part contains a detailed description of HPC infrastructure used for the task, located in Świerk Computing Centre (CIŚ), Poland. The description includes hardware configuration, software used for on-demand deployment of dedicated subclusters, and finally queuing and storage systems. The second part presents the large-scale simulations

2019 – IJHMT @ IF=3.89

International Journal of Heat and Mass Transfer 135 (2019) 517–540

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International Journal of Heat and Mass Transfer

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

ELSEVIER

Direct numerical simulation of flow and heat transfer in a simplified pressurized thermal shock scenario

Check for updates

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ARTICLE INFO

Article history:
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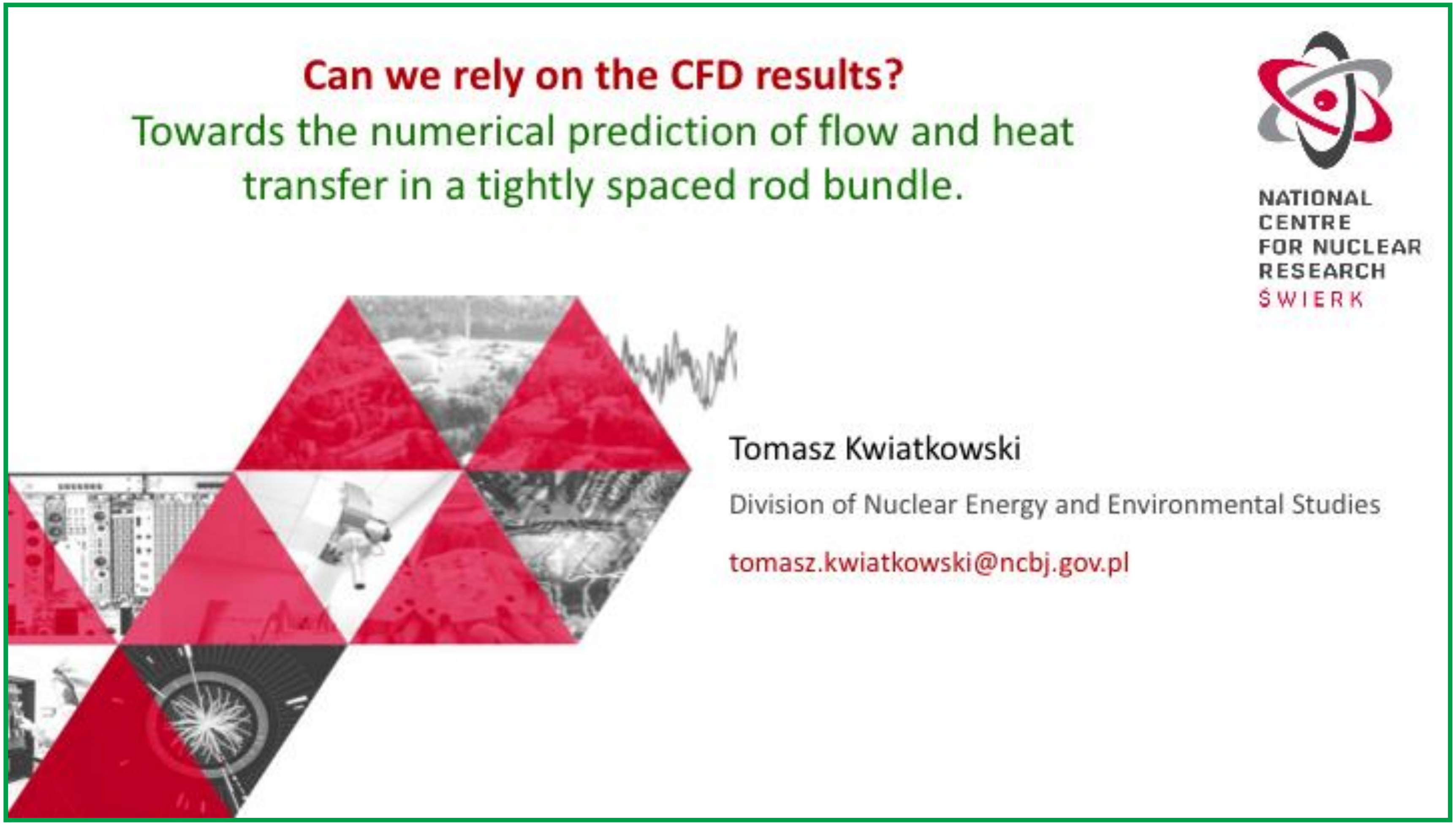
ABSTRACT

Direct numerical simulation (DNS) is carried out using a high order spectral element method in order to analyse the flow and thermal fields in a Pressurized Thermal Shock (PTS) scenario. The adopted configuration is representative of a simplified reactor pressure vessel. It consists of a square duct that intersects perpendicularly with a rectangular vessel (downcomer). Cold water is injected from the square duct. It impinges against the wall of the downcomer that contains water at higher temperature, and a mixing

2019 – NUGENIA Forum



2019 – FISA - 9

4-7 June 2019
Pitesti, Romania

FISA 2019
9th European Commission Conference on EURATOM Research and Training in Safety of Reactor Systems

Towards Validation of RANS CFD Approach for Flow and Heat Transfer in a Closely-spaced Bare Rod Bundle

Tomasz Kwiatkowski[1] and Afaque Shams[2]

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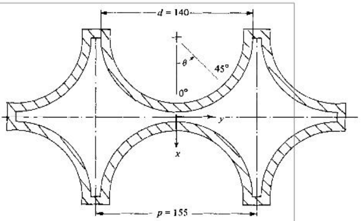
Introduction

The coolant flowing through the channels within the fuel assembly removes the heat generated by the nuclear fission. In an ideal scenario, the temperature distribution through the fuel assemblies should remain uniformly distributed under normal operating conditions of a nuclear reactor. However, in reality this does not happen and accordingly it leads to inter-subchannel mixing phenomena. A detailed knowledge of flow and temperature in a fuel assembly has always been a major factor in the design and reliable operation of existing and new nuclear systems.

Step 1 - design a numerical experiment



a) Selection of the geometry [2]
b) Scaling of the the Reynolds number
c) Assesment of the stream-wise lengths
d) Introduction of the thermal fields:
   - 4 Pr fluids (Pr = 0.025, 1, 2, 7)
   - constant temperature and heat flux at the rods
e) ANSYS Fluent v17.2.0
f) URANS: k-ω SST model

Achievements

Drive

Persistent

Hard Working

Respectful

Humble

Mr. Consistent

Questions?