On the modelling of thermal-hydraulics and neutronics coupling for designing of a prismatic HTGR core with the HTTR fuel block structure.

NATIONAL CENTRE FOR NUCLEAR RESEARCH ŚWIERK

Michał Górkiewicz

Division of Nuclear Energy and Environmental Studies

Michal.Gorkiewicz@ncbj.gov.pl

Fundusze Europejskie
Wiedza Edukacja Rozwój

Rzeczpospolita Polska

Unia Europejska
Europejski Fundusz Społeczny

New reactor concepts and safety analyses for the Polish Nuclear Energy Program
POWR.03.02.00-00.I005/17

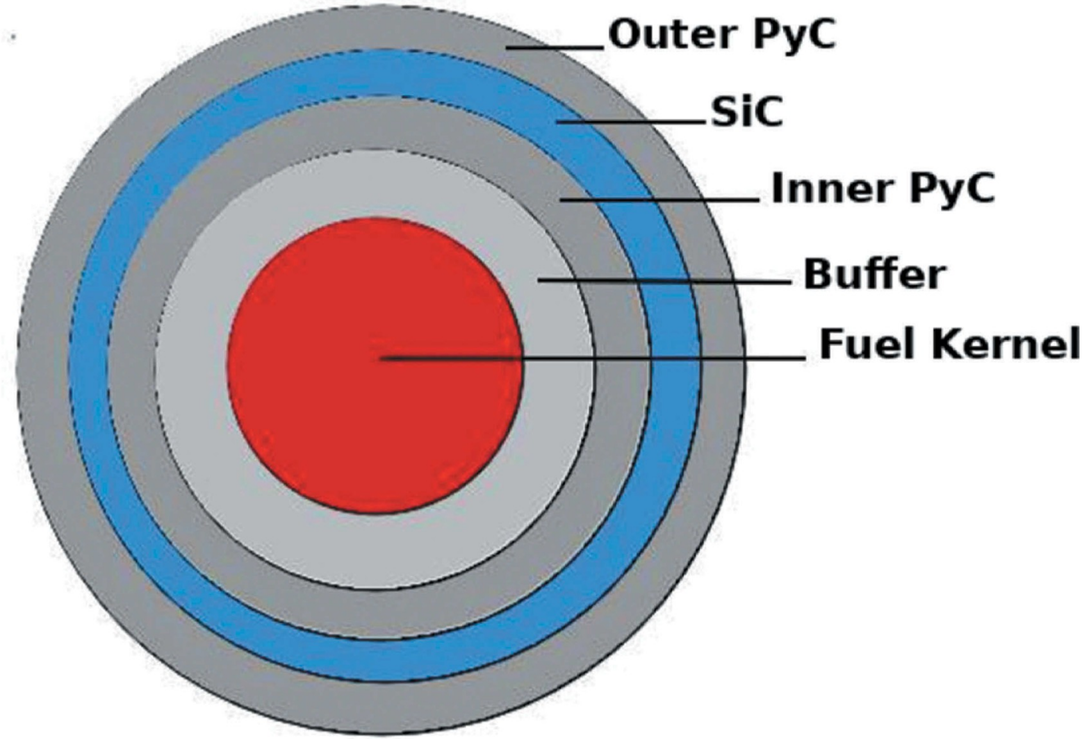
Outline

- Introduction
- Challenges in coupled calculations for HTGRs
- Application of coupled calculations
- Current work status
- The HTTR performance simulation approach
- Ideas for modelling of thermal-hydraulics

Introduction

- PhD topic:
  Development and validation of coupled neutronic and CFD calculations for HTR applications
- Supervisor
  prof. Dr hab. inż. Jerzy Cetnar
- HTGR
  - Reactor concept
  - High Temperature Gas-cooled Reactor
  - Helium as coolant
  - Graphite as moderator
  - Fuel in TRISO particles
- HTTR
  - Existing reactor
  - High Temperature engineering Test Reactor
  - The Oarai Research and Development Center, Japan
  - Thermal power 30 MW
  - Maximum outlet temperature 950 °C

Introduction



Source: Baghdasaryan N., el. at.: Review of Progress in Coated Fuel Particle Performance Analysis, Nuclear Science and Engineering, 2019

Challenges in neutronic calculation

- Deep neutron thermalisation
- Hard neutron spectrum
- Large migration length
- Local neutron spectrum is strongly influenced by:
  - Control rods
  - Burnable poisons
  - Reflectors
- High neutron flux gradients

Nuclear Properties

Core	Modular - HTGR	LWR
Power density, w/cc	5.8-6.6	58 - 105
Linear heat rate, kW/ft	1.6	19
Avg. therm-neutron energy, eV	0.22	0.17
Average Uranium Enrichment	15.5%	4.00%
Moderator (at 0.025 eV)	Graphite	Water
Diffusion Coefficient D, cm	0.86	0.16
Diffusion Length L, cm	54	2.75
Migration length M, cm	57	6
Collisions to thermalize	~18	~1
Σa (cm^{-1})	0.00029	0.022
Σs c(m^{-1})	0.41	3.45

Source: HTGR Technology Course for the Nuclear Regulatory Commission Module 5b, Idaho National Laboratory, 2010

Challenges in neutronic and thermal-hydraulic calculation

- Neutronic
- Power peaks
- Double heterogeneity
  - Caused by fine structure of compacts filled with TRISO particles
  - Highly structured geometrical model is needed to account for neutron spectra effects that occur in the fuel due to resonant cross sections
- Neutronic cross section dependence on temperature

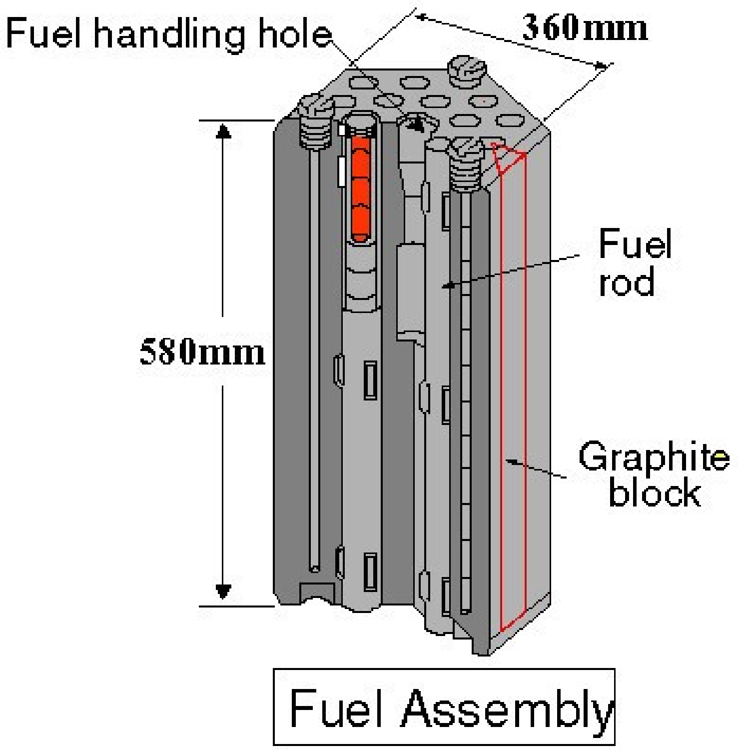
- Thermal-hydraulic
- Hot spots
  - Maximum temperature limit due to TRISO particles failures
  - Necessity to identify temperature peaks in the fuel
- Bypass flow

Application

- Full core analyses
- Accurate power profile lets one obtain accurate temperature profile, thus identify hot spots and estimate TRISO particles safety conditions
- Neutronic cross-sections are being updated according the temperature profile
- Updated power profile
- Including burnup calculation one can simulate the reactor fuel cycle and estimate optimal design by applying burnable poisons, control rods operation and shuffling scheme

Present work status

- Interface for data exchange was initially developed employing the Python programming language
- A 3D model of 1/6th fuel rod was created in the OpenFoam CFD software basing on the HTTR design
- chtMultiRegionFoam solver is used in order to calculate temperature distribution in fluids and solids

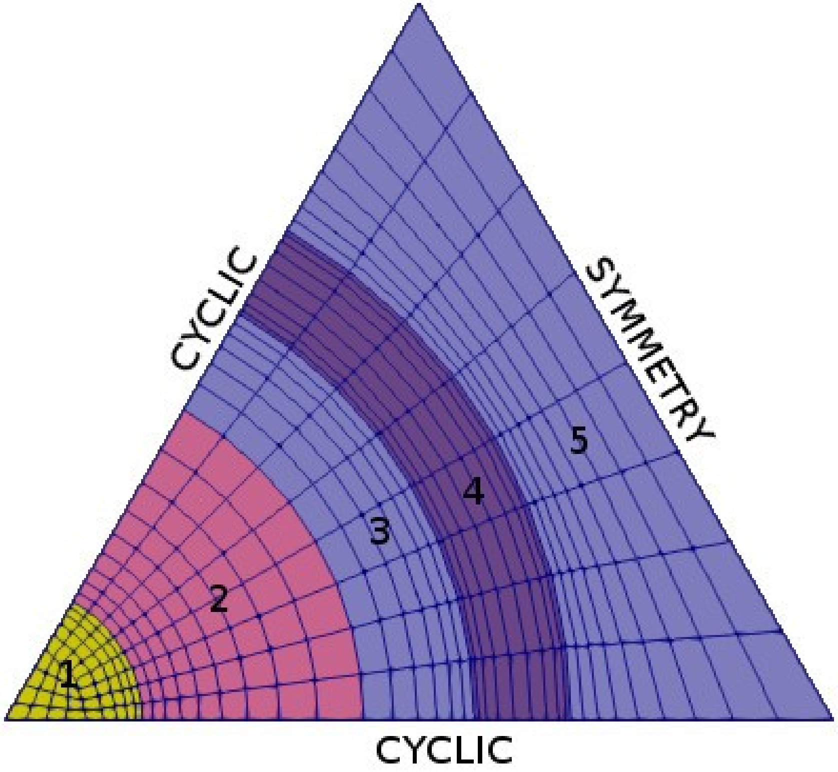


Source:https://httr.jaea.go.jp/eng/D/D2-1.html

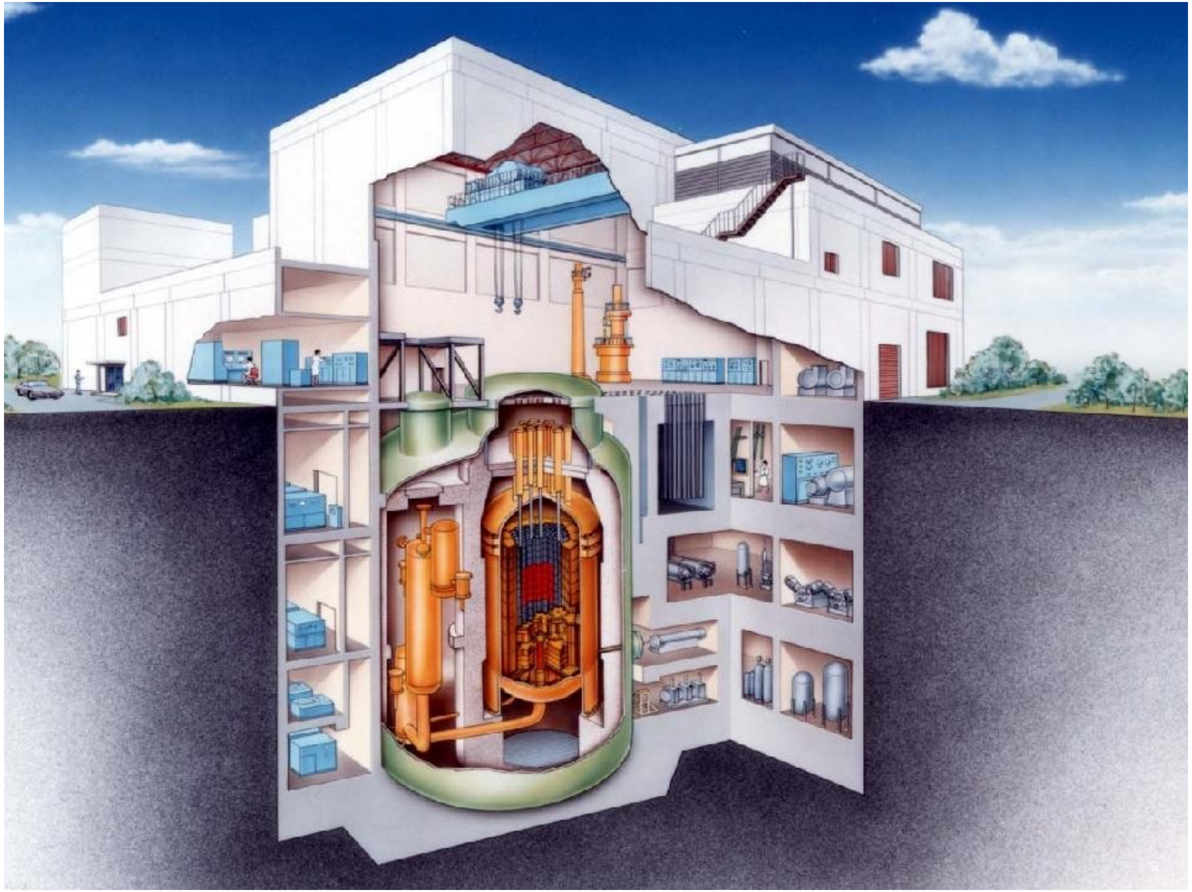
Numerical setup

- 1. Helium void; 2. Fuel; 3. Graphite sleeve; 4. Coolant; 5. Graphite matrix
- Mesh generated with Gmsh software
- Cyclic and symmetry boundary conditions applied
- The model is still under development

- Expand to entire core?



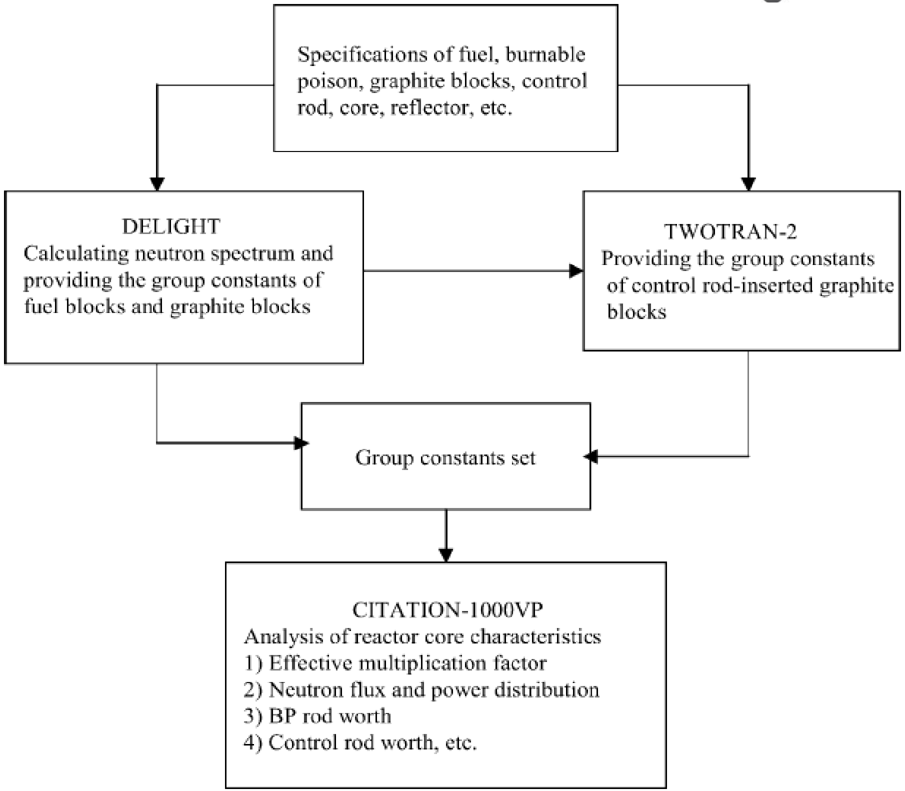
The OpenFoam model mesh



Source:https://httr.jaea.go.jp/eng/index.html

HTTR Nuclear design

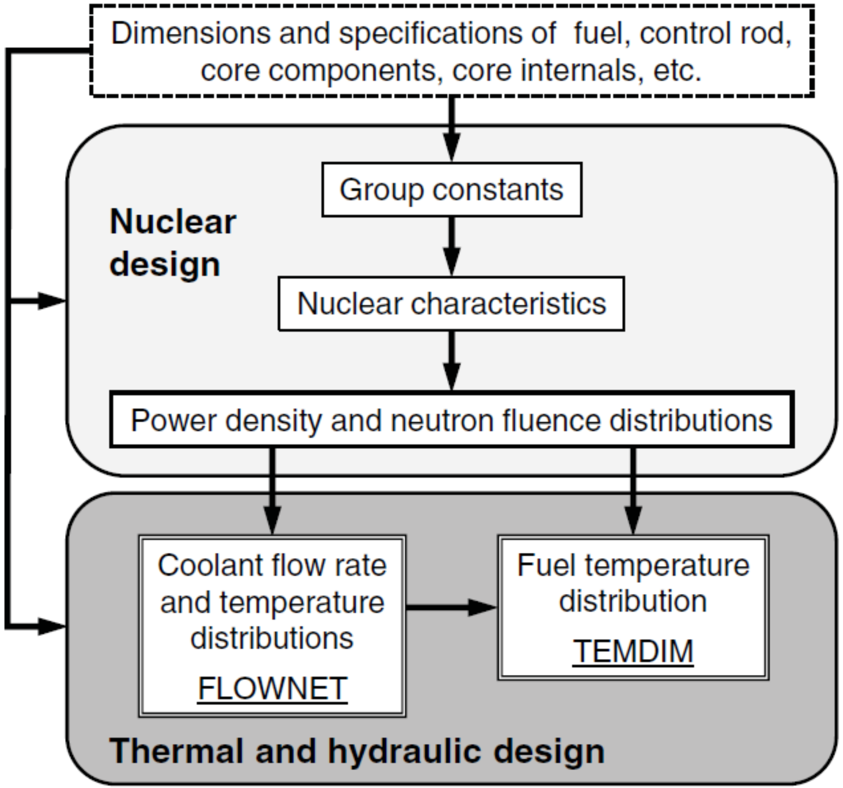
- DELIGHT calculates the resonance, neutron spectrum, fuel cell, burnable poison cell, criticality, and burnup calculations.
- TWOTRAN-2 is employed to find the shielding factors of the control rods and give the average group constants of a graphite block
- CITATION-1000VP performes the whole core calculations



Source: Fujimoto N. el. al., Nuclear Design, Nuclear Engineering and Design, 233/2004, 26

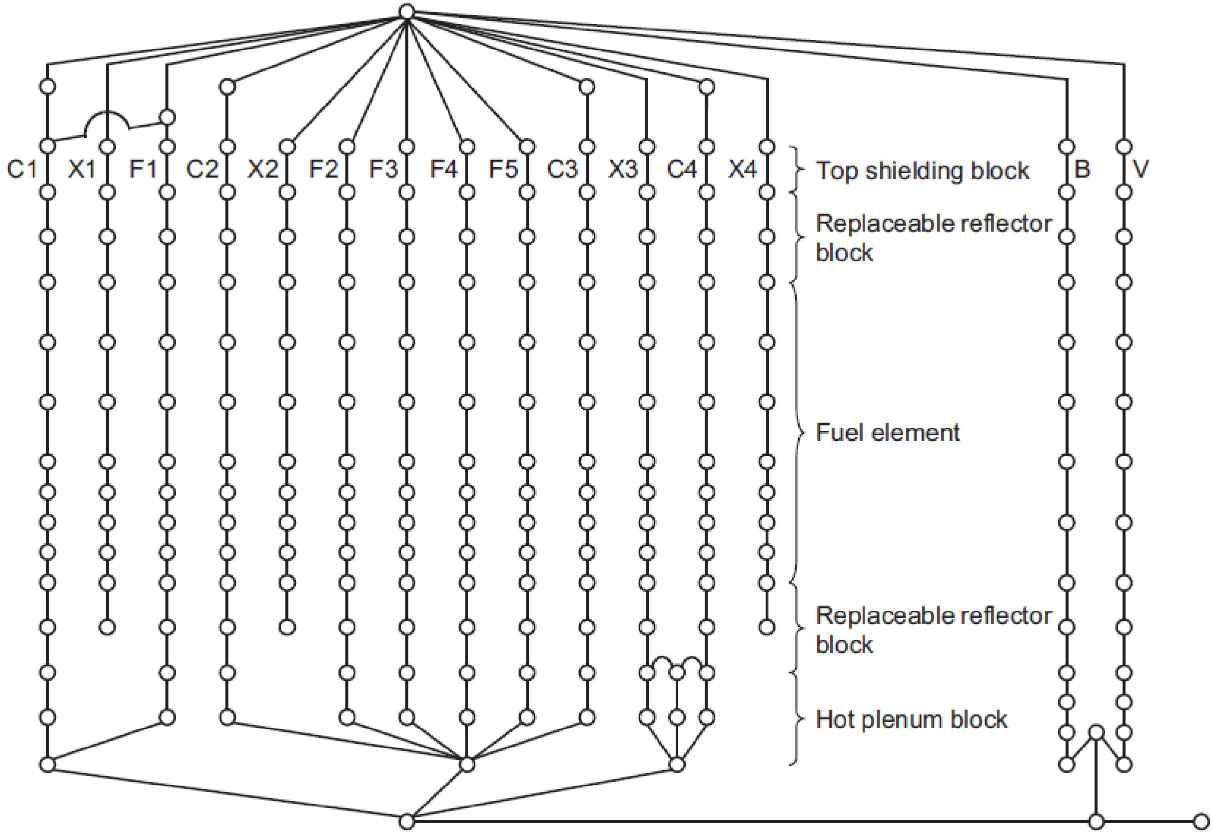
HTTR Thermal-hydraulic design

- FLOWNET model consists on 1D flow branches and pressure nodes. Coolant flow rate distribution is determined on the basis of the power density and neutron fluence distributions obtained by the nuclear design and the dimensions of the core components and internals.
- Bypass flow is included and is calculated on the basis of the core temperature and neutron fluence distributions and the thermal expansion of the components



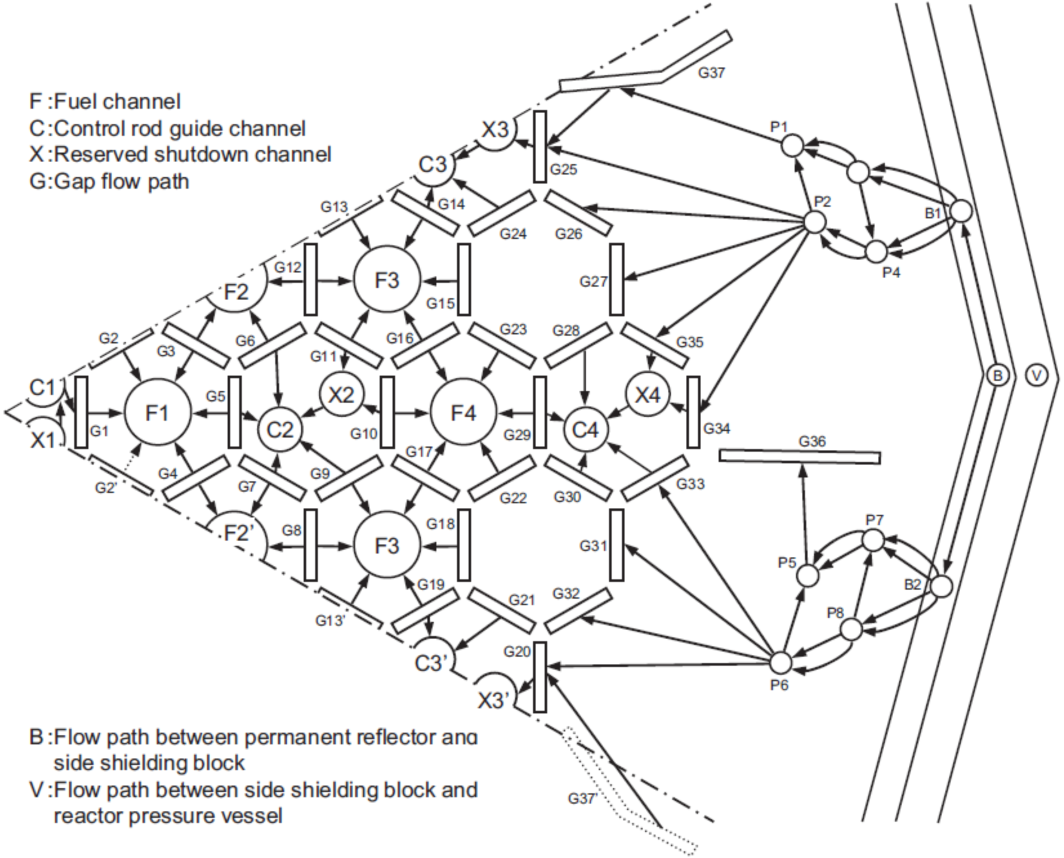
Source: Inaba Y. et. al, Evaluation of maximum fuel temperature in HTTR, Journal of Nuclear Science and Technology, 51/2014, 1337

Flownet



Source: Inaba Y. et. al, Evaluation of maximum fuel temperature in HTTR, Journal of Nuclear Science and Technology, 51/2014, 1338

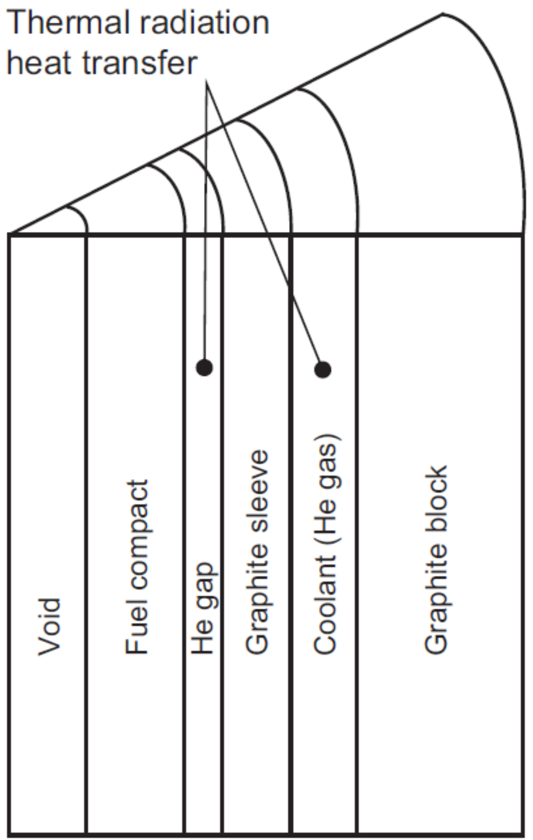
Flownet



Source: Inaba Y. et. al, Evaluation of maximum fuel temperature in HTTR, Journal of Nuclear Science and Technology, 51/2014, 1338

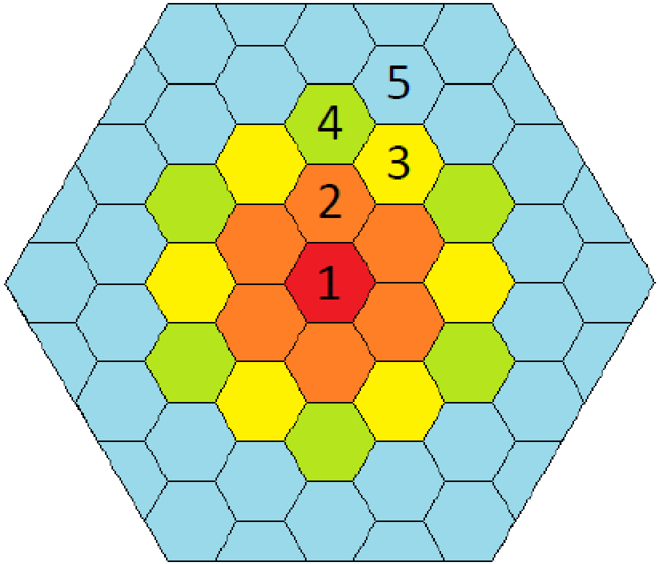
TEMDIM

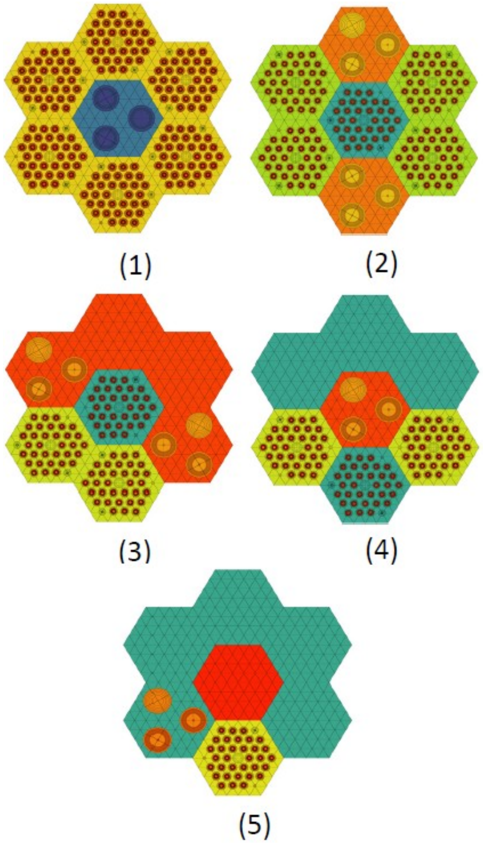
- The thermal radiation heat transfer between the fuel compact and the graphite sleeve and that between the graphite sleeve and the graphite block are considered.
- The periphery of the model is surrounded by the adiabatic boundary in order to achieve conservative solutions.
- The hot spot factors are considered in the fuel temperature analysis in order to evaluate the maximum fuel temperature with a sufficient margin.



Source: Inaba Y. et. al, Evaluation of maximum fuel temperature in HTTR, Journal of Nuclear Science and Technology, 51/2014, 1339

Supercell concept





Source: Sen S. et. al., Super-Homogenization-Corrected Cross-Section Generation for High-Temperature Reactors, Idaho National Laboratory, 2017

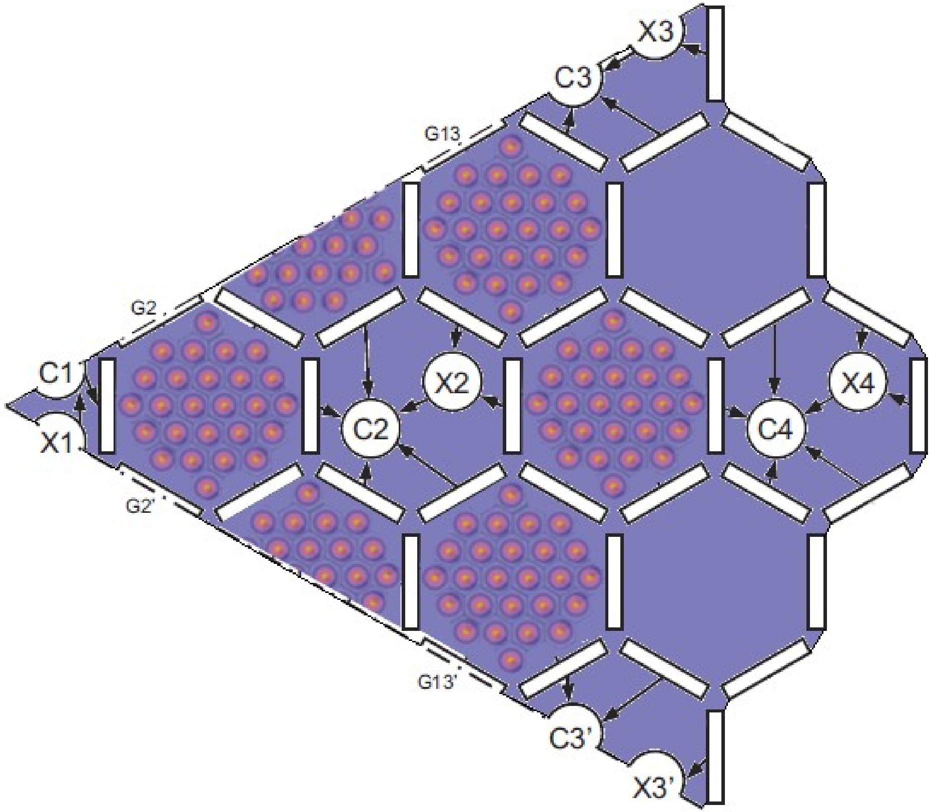
Ideas for modelling of thermal-hydraulics for neutronic coupling

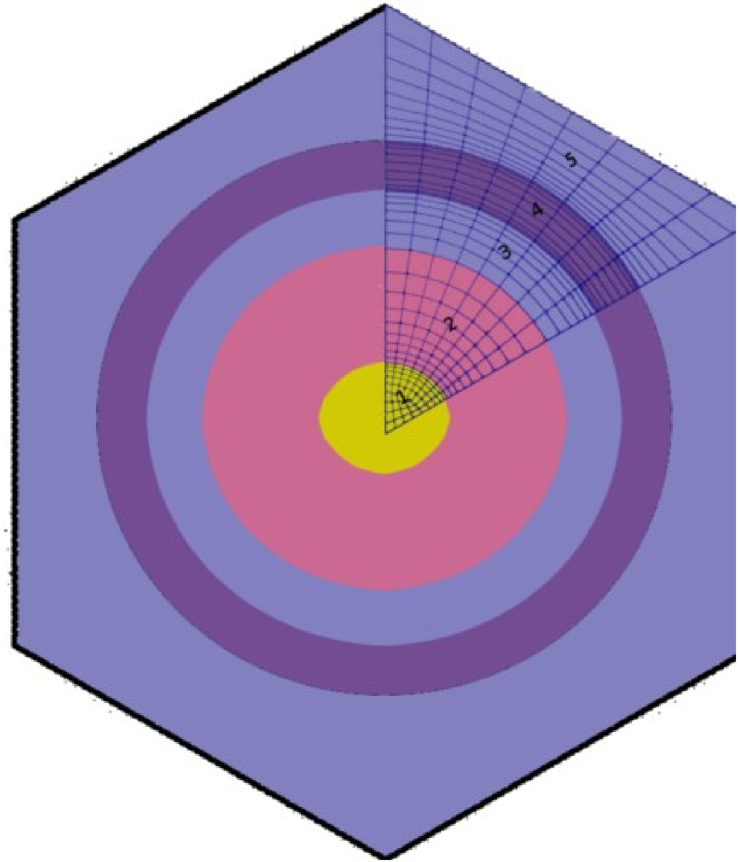
- In case of HTGR it is essential to take into consideration several neutron effects and temperature feedback during normal operation. For that purpose 1D thermal-hydraulic models are often sufficient.
- Nevertheless, with 3D models hot-spots can be identified more accurately. Moreover, transients and accident conditions can be simulated.

- Why not both?

- A model consisting on branches of simple 3D CFD models could be created.
- Doing so, the effort would be shifted from the CFD to data exchange interface.

Idea for modelling of thermal-hydraulics in prismatic HTGR core





Exapmle of similar approach

DEVELOPMENT OF A CORE THERMO-FLUID ANALYSIS CODE FOR PRISMATIC GAS COOLED REACTORS

NAM-IL TAK*, SUNG NAM LEE, MIN-HWAN KIM, HONG SIK LIM, and JAE MAN NOH
Korea Atomic Energy Research Institute
Daedeok-Daero 989-111, Yuseong-gu, Daejeon 305-353, Korea
*Corresponding author. E-mail : takni@kaeri.re.kr

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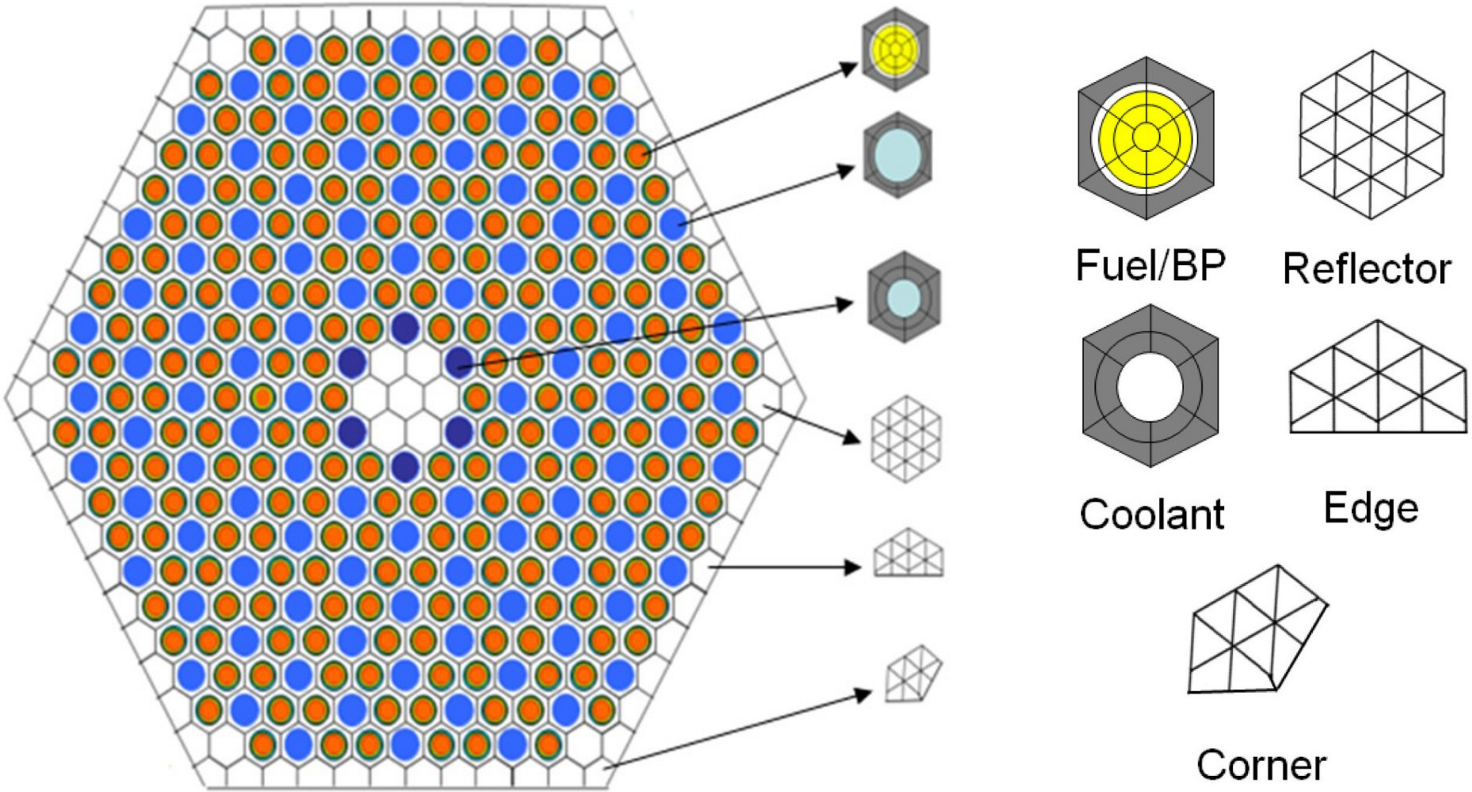
A new computer code, named CORONA (Core Reliable Optimization and thermo-fluid Network Analysis), was developed for the core thermo-fluid analysis of a prismatic gas cooled reactor. The CORONA code is targeted for whole-core thermo-fluid analysis of a prismatic gas cooled reactor, with fast computation and reasonable accuracy. In order to achieve this target, the development of CORONA focused on (1) an efficient numerical method, (2) efficient grid generation, and (3) parallel computation. The key idea for the efficient numerical method of CORONA is to solve a three-dimensional solid heat conduction equation combined with one-dimensional fluid flow network equations. The typical difficulties in generating computational grids for a whole core analysis were overcome by using a basic unit cell concept. A fast calculation was finally achieved by a block-wise parallel computation method. The objective of the present paper is to summarize the motivation and strategy, numerical approaches, verification and validation, parallel computation, and perspective of the CORONA code.

KEYWORDS : Prismatic Core, Fuel Temperature, Fluid Flow Network, Gas Cooled Reactor, CORONA, VHTR

CORONA sub-channel code



Source: Tak N. et. al.: DEVELOPMENT OF A CORE THERMO-FLUID ANALYSIS CODE FOR PRISMATIC GAS COOLED REACTORS, Nuclear Engineering and Technology, 46/2014

Literature

1. Cetnar J. et. al.: *Advanced burnup assessments in prismatic HTR for Pu/MA/Th utilization using MCB system*, AGH, 2013
2. Inaba Y. et. al.: *Evaluation of maximum fuel temperature in HTTR*, Journal of Nuclear Science and Technology, 51/2014, 1336-1344
3. Sen S. et. al.: *Super-Homogenization-Corrected Cross-Section Generation for High-Temperature Reactors*, Idaho National Laboratory, 2017
4. Sterbentz J.W. el. al.: *Reactor Physics Parametric and Depletion Studies in Support of TRISO Particle Fuel Specification for the Next Generation Nuclear Plant*, Idaho National Engineering and Environmental Laboratory, 2004
5. Fujimoto N. el. al.: *Nuclear Design*, Nuclear Engineering and Design, 233/2004, 23 – 36
6. Ortensi J.: *Supercell Depletion Studies for Prismatic High Temperature Reactors*, Idaho National Laboratory, 2012
7. Tak N. et. al.: *DEVELOPMENT OF A CORE THERMO-FLUID ANALYSIS CODE FOR PRISMATIC GAS COOLED REACTORS*, Nuclear Engineering and Technology, 46/2014, 641 - 654

Thank you for attention

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