# Thermal Hydraulics Analyses on the Core of HTR-10 in Steady State and Transient Conditions using Fluent 6.3 3-Dimensional Model for Indonesian RDE Design Development

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## 1. Introduction

National Atomic Energy Agency of Indonesia (BATAN) has chosen the High Temperature Gas-cooled Reactor (HTGR) for energy fulfilment solution in Indonesia. HTGR is a high-level safety Gen. IV power reactor, no melting core when an accident occurs, which very suitable for Indonesia. Indonesian HTGR development is initiated by the development of its experimental type, named *Reaktor Daya Eksperimental* (RDE). RDE refers to the Chinese HTR-10 design that had reached full power operation in 2003. Various researches have been conducted to prepare the RDE design to meet the HTGR safety system. This research is aimed to understand the characteristics of primary fluid flow and heat transfer in the core of HTR-10 and in that respect it is possible to obtain important parameters that can be used in RDE design. HTR-10 core modelling, research methods, and the results are described herein.

## 2. Methods

### *2.1 HTR-10 Overview*

HTR-10 was designed, constructed and operated by the Institute of Nuclear and New Energy Technology (INET), Tsinghua University, as a major project in the energy sector of the Chinese National High Technology Programme. The main design parameters of HTR-10 are shown in Table I.

Table I: Main design parameters of HTR-10 [1].

| Parameter | Value |
|---|---|
| Reactor thermal power, $MW_{th}$ | 10 |
| Primary helium pressure, MPa | 3 |
| Average helium temperature at reactor outlet, °C | 700 |
| Average helium temperature at reactor inlet, °C | 250 |
| Helium mass flow rate at full power, kg/s | 4.32 |

The HTR-10 reactor core is cooled by helium gas, moderated by graphite and uses Uranium spherical fuel elements (TRISO). The bottom of the core filled by dummy balls, on the upper part filled by fuel elements, the hollow space around the ball filled by helium gas, which constitutes the pebble bed. The design parameters of fuel elements and dummy balls are given in Table II.

Table II: Design parameters of fuel elements, dummy balls and loading ratio [1].

| Parameter | Value |
|---|---|
| **Fuel element** | |
| Diameter of ball, cm | 6.0 |
| Diameter of fuel zone, cm | 5.0 |
| Density of graphite in matrix and outer shell, $g/cm^3$ | 1.73 |
| Heavy metal (uranium) loading (weight) per ball, g | 5.0 |
| Enrichment of U-235 (weight), % | 17 |
| **Dummy Balls** | |
| Diameter of ball, cm | 6.0 |
| Density of graphite, $g/cm^3$ | 1.73 |
| Loading ratio of fuel balls to dummy balls | 57:43 |

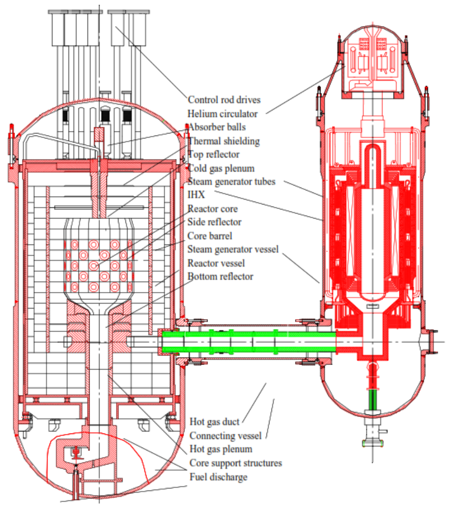

Fig. 1. HTR-10 Primary System [1].

HTR-10 primary systems are presented in Fig.1. The reactor consists of the Reactor Pressure Vessel (RPV), internal graphite and carbon brick components, metallic components, fuel elements, control rods and their driving mechanisms, small absorber ball shut-down system, fuel charging, and discharging system components. Graphite reflectors, categorized as the top reflector, the side reflector, and the bottom reflector, enclose the active core zone. Radially, the side reflector structure is divided into the inner graphite zone and the outer boronated carbon brick zone. The inner graphite zone serves as a neutron reflector of the active core and the outer carbon bricks play the role of thermal insulator and neutron absorber [1].

The graphite blocks are connected and integrated by graphite pin keys, whose role includes positioning and helium leakage reduction. There are 20 channels near the active core zone in the side reflector, including 10 control rod channels, 7 absorber ball channels, and 3 reserved irradiation channels. 20 cold helium channels are designed at the outer part of the side reflector graphite blocks. The hot helium plenum is located in the bottom reflector [1].

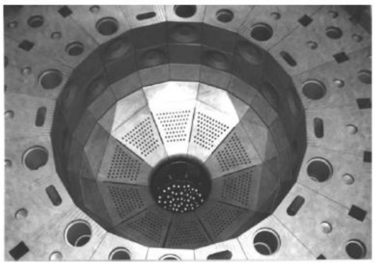

Fig. 2. HTR-10 Core [2].

Fig. 2 presents the HTR-10 Core. The upper part of the reactor core has a cylindrical geometry and the lower part is cone-shaped. Part of the helium coolant bypasses the main flow path, 2% to control rod channels and absorber ball channels, only 87% of the Rated Coolant Flow Rate (RCFR) effectively cools the fuel elements in the core [3]. Table III gives some geometrical characteristics of the HTR-10 reactor core.

Table III: Geometrical characteristics of the HTR-10 reactor core [1].

| Parameter | Value |
|---|---|
| Equivalent diameter, cm | 180 |
| Average height, cm | 197 |
| Volume, $m^3$ | 5 |
| Volumetric filling fraction of balls in the core | 0.61 |
| Height of the empty cavity above the pebble bed, cm | 41.7 |
| Diameter of fuel discharging tube, cm | 50 |

*2.2 3-D Modeling of HTR-10 Core*

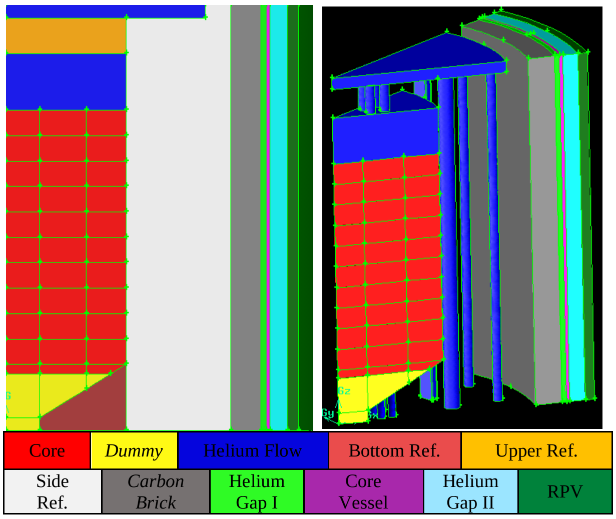


Fig. 3. Core modelling: vertical 2D and 3D without graphite.

The core modelling is presented in Fig. 3. Technical data of HTR-10 core such as materials, geometry and the others obtained from the benchmark [1]. Core inlets and core outlets were simplified by equalizing its flow area to adjust the quality of the mesh in those areas. Core area, which was analyzed, limited by cold helium plenum (Z = -75 cm) in the top axial direction, small plenum (Z = 227 cm) in the bottom axial direction and RPV (R = 218 cm) in the outer radial direction. 10 reactor core radial partitions were simplified into 3 parts by adjusting the volume of the core partition. Core modelling used 1 of 10 parts of the core that is identical so it still represents the entire core.

Cold helium flow through the model inlet is 0.38448 kg/s which flows into two parts, 0.00864 kg/s (2%) to the control rod channel and absorber ball channel, and 0.37584 kg/s (87%) to the reactor core.

*2.3 Solver FLUENT 6.3 code*

FLUENT solves the thermal hydraulics cases by connecting three fundamental principles to describe flow movements, i.e. the laws of conservation of mass, momentum, and energy [4].

Contiunity : $$\frac{\partial \rho}{\partial t} + \nabla.(\rho \boldsymbol{u}) = 0 \quad (1)$$

Momentum : $$\rho \frac{D\boldsymbol{u}}{Dt} = \nabla.\tau_{ij} - \nabla p + \rho F \quad (2)$$

Energy : $$\rho \frac{De}{Dt} + p(\nabla.\boldsymbol{u}) = \frac{\partial Q}{\partial t} - \nabla.\mathrm{q} + \Phi \quad (3)$$

with $\boldsymbol{\rho}$ as fluid density, $\boldsymbol{u}$ as a velocity vector, $\boldsymbol{\tau_{ij}}$ as tensor stress, **p** as pressure, **F** as body forces, **e** as internal energy, **Q** as the heat source, **t** as time, **Φ** as dissipation, and $\boldsymbol{\nabla.q}$ due to conduction. Conduction heat transfer can be approximated using Fourier's law to get the **q** value.

$$q = -\lambda \nabla \mathrm{T} \quad (4)$$

with **λ** as conductivity and **T** as temperature [4].

SIMPLE with default composition was used as the Solver. The discretization methods which used were standard for pressure, and first-order upwind for momentum, turbulent kinetic energy, turbulent dissipation rate, energy, and the discrete ordinates (DO) radiation model. The DO radiation method and geometric complexity give complicated iteration completion, so, the value of convergence of continuity, the velocity vector of direction x, direction y, and direction z, discrete ordinates, k, and epsilon were set at $10^{-4}$ and $10^{-5}$ for energy.

The core was modelled as a porous medium for configuration of fuel elements, dummy balls, and helium flow through the core. Porous medium in FLUENT initialized by entering viscous resistance (permeability) values to represent the viscous loss term and inertial resistance values to represent the inertial lost term. Those values were obtained by the Ergun Equation (5) approach to packed bed modelling with its correlation to the porous medium homogeneous momentum equation (6) [5], i.e.:

$$\frac{|\Delta p|}{L} = \frac{150\mu}{D_p^2}\frac{(1-\gamma)^2}{\gamma^3}u_\infty + \frac{1{,}75\rho}{D_p}\frac{(1-\gamma)}{\gamma^3}u_\infty^2 \quad (5)$$

$$S_i = -\left(\frac{\mu}{\mathrm{a}}u_i + C_2\frac{1}{2}\rho|u|u_i\right) \quad (6)$$

with **a** as permeability and $\boldsymbol{C_2}$ as inertial resistance, thus:

$$\mathrm{a} = -\left(\frac{{D_p}^2}{150}\frac{\gamma^3}{(1-\gamma)^2}\right) \quad (7)$$

$$C_2 = -\left(\frac{3{,}5}{D_p}\frac{(1-\gamma)}{\gamma^3}\right) \quad (8)$$

with $\boldsymbol{D_p}$ as average fuel diameter and $\boldsymbol{\gamma}$ as a void fraction in the porous cell [5].

## 3. Results and Discussions

### *3.1 Steady-State FPIC HTR-10*

The analysis of the steady-state condition used HTR-10 data on the Full Power Initial Core (FPIC) condition based on benchmarks [1].

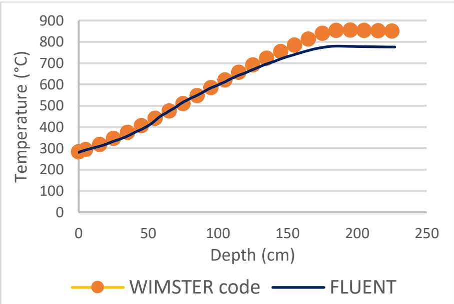


Fig. 4. Comparison of axial core temperature distribution (R = 0 cm).

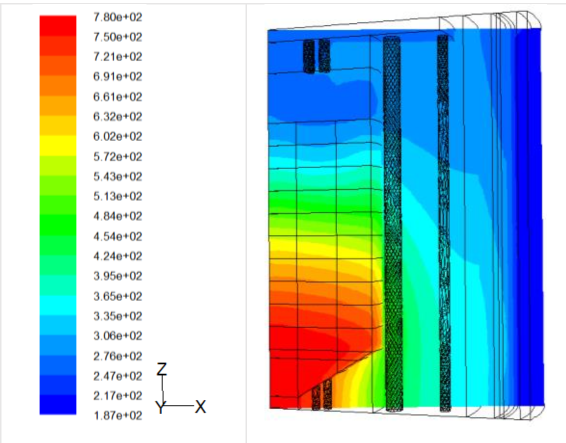


Fig. 5. Contour of temperature distribution by Sweep Surface Y = 0 cm.

Fig. 4 presents a comparison between axial core temperature distribution at the radial core centre (R = 0 cm) obtained in this work using FLUENT 6.3 and that found by the participant of the benchmark that using WIMSTER code. Core temperature distribution in this work increases along the core that has similar behaviour with the benchmark, however, this work result has a lower temperature especially at depths greater than 150 cm, caused by differences power density of the central radial core volume partition between the two models. Simplified radial core partition in this work reduces the average core density value in that partition volume, thus diminish power generation.

The contour of temperature distribution by sweep surface Y = 0 cm in Fig. 5 describes temperature distribution from radial core centre modelling to the RPV. The result, temperature distribution rises along the core due to heat accumulation that was taken by helium gas along axial of the core. Radially, the core temperature distribution is lower due to heat transfer by conduction and radiation to the reflector, core vessel, and RPV. The maximum temperature of the core, core vessel and RPV, respectively 779.96°C, 242.74°C, 194.56°C.

Core outlet flow data are important for the thermal hydraulic analysis of the HTR-10 reactor, especially for hot helium plenum and hot gas tubes. Those data are given in Table IV with the core outlet modelling in Fig. 6, which can be used in the design and safety analysis of RDE.

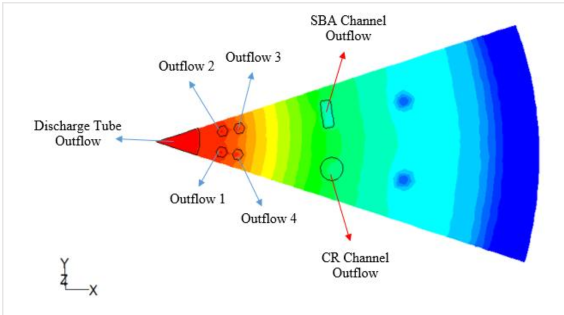


Fig. 6. Core outlet modelling by Sweep Surface Z = 227 cm.

Table IV: Core outlet data.

| Outlet | Mass flow rate (kg/s) | Pressure Drop (Pa) | Temperature (°C) |
|---|---|---|---|
| Discharged Tube | 0.14022 | 1846.33 | 768.28 |
| Outflow 1 | 0.05931 | 1869.85 | 740.94 |
| Outflow 2 | 0.05888 | 1872.81 | 741.03 |
| Outflow 3 | 0.05888 | 1744.76 | 712.40 |
| Outflow 4 | 0.05888 | 1714.57 | 712.09 |
| CR Channel | 0.00161 | 455.40 | 426.42 |
| SBA Channel | 0.00703 | 455.76 | 395.02 |

### *3.2 ATWS – PLOFC 50%*

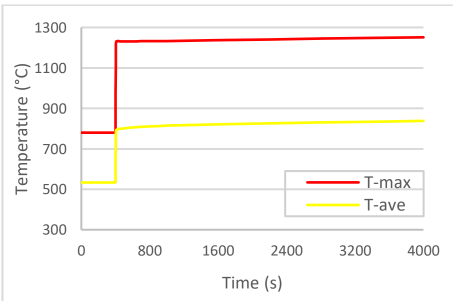


Fig. 7. Maximum (T-max) and Average (T-ave) core temperature.

In ATWS conditions, the steady state core was given changes with Pressurized Lost of Forced Cooling (PLOFC) transient conditions. In this work, PLOFC is simulating an accident loss of helium mass flow rate by 50%. The input is a step signal given in the 400th second with 4000 seconds total simulation duration.

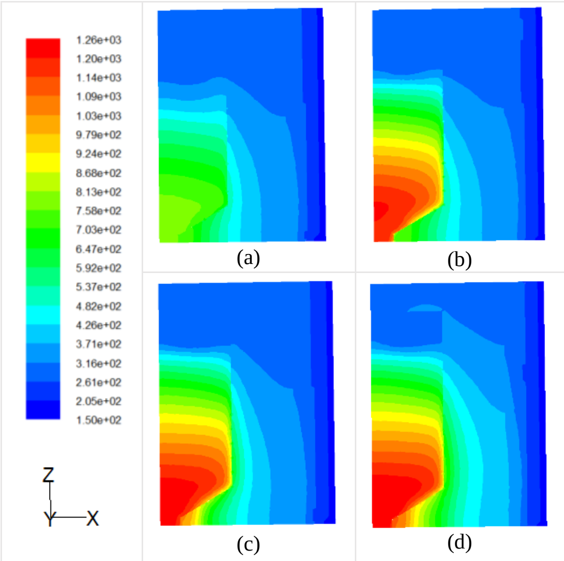


Fig. 8. Contour of temperature distribution by Sweep Surface Y = 0 cm at: (a) t = 400 s, (b) t = 403 s, (c) t = 1000 s and (d) t = 4000 s.

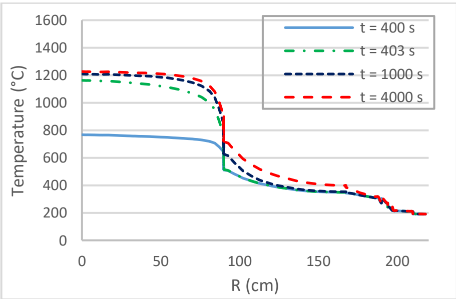


Fig. 9. Radial temperature distribution (Z = 170 cm).

Core temperature highly increased in the first few seconds of transient as shown in Fig.7 because the reduction of coolant flow rate into the core decreased heat transfer integrity especially in convection, but heat generation has a constant value. Highly increased temperature only occur shortly because the system immediately finds a steady flow rate condition on the core, thus the heat transfer starts to be constant especially in convection. Furthermore, the temperature still increased due to heat taking by the primary coolant flow was no longer able to compensate heat generation that occurs each second. The increased is relatively small each second due to the reliability of heat transfer by conduction of reactor materials, fuel elements, graphite, and helium gas, which has large heat capacity and increased conductivity when temperature increased resulting in faster heat transfer when the temperature increased, that showed by increased temperature of reflector and carbon brick (90 < R < 190) in the 1000th second and 4000th second (see Figs. 8 and 9). Those showed the advantageous feature of HTR-10 which has slow thermal transient. To be noted that neutron calculations are not included in this work, thus there is no negative feedback of fuel temperature.

Core temperature reached maximum values of about 1251.09°C which is below the maximum safety for the fuel temperature (1600°C). To be noted, the core temperature in this work presents the equilibrium temperature between the fuel temperature and the helium temperature which was the impact of FLUENT 6.3 limitations, thus fuel temperature is actually higher than the core temperature. It can be explained by the value of fuel conductivity which is much higher than the value of helium conductivity. However, the value of fuel conductivity more dominant to the effective conductivity of the core concludes that the fuel temperature is closer to the core temperature than the temperature of helium. The maximum temperature of the core vessel and RPV are below each its safety limit of 425°C and 375°C, respectively 245,91°C and 195,77°C.

## 4. Conclusions

The HTR-10 core has been simulated using the FLUENT 6.3 with 3-Dimensional modelling with a porous medium approach for the core. The results presented temperature distribution reached a steady state behaviour as expected, similar thermal behaviour in comparison with the data from the IAEA reference document. It also obtained some core outlet parameters that are useful in the analysis of hot plenum gas and hot gas tube designs. In addition, an ATWS – PLOFC 50% in the core was simulated. The results indicated that after the accident, core temperature increases significantly but just for short time, and then it still increased but relatively small for each second Those showed the advantageous feature of HTR-10 which has slow thermal transient. The maximum temperature of the core, core vessel and RPV in this ATWS are below its safety of temperature limits.

## REFERENCES

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