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THERMAL MODELING OF THE HTR-10 USING THE RELAP5-3D CODE

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ABSTRACT

Several efforts have been considered in the development of the modular High Temperature Gas cooled Reactor (HTGR) planned to be a safe and efficient nuclear energy source for the production of electricity and industrial applications. In this work, the RELAP5-3D thermal hydraulic code was used to simulate the steady state behavior of the 10 MW pebble bed high temperature gas cooled reactor (HTR-10), designed, constructed and operated by the Institute of Nuclear and New Energy Technology (INET), in China. The reactor core is cooled by helium gas. In the simulation, results of temperature distribution within the pebble bed, inlet and outlet coolant temperatures, coolant mass flow, and others parameters have been compared with the data available in a benchmark document published by the International Atomic Energy Agency (IAEA) in 2013. This initial study demonstrates that the RELAP5-3D model is capable to reproduce the thermal behavior of the HTR-10.

INTRODUCTION

The researches with high-temperature-gas-cooled reactors (HTGR) have been developed for nearly 50 years and this type of reactor is one of the candidates for the future nuclear power systems. The pebble-bed high-temperature-gas-cooled reactor (HTGR-PM) is seen as one of the best candidates for the next generation reactors [1].

The HTR-10 is one small reactor, with thermal power of 10 MW, developed in China for study and demonstration of the technical and safety of the modular HTGR-PM and to establishment of the experimental bases for developing processes. The aims of the HTR-10 are: to acquire the experience of HTGR design, construction and operation; to carry out the irradiation tests for fuel elements; to verify the inherent safety of the modular HTGR; to demonstrate the electricity/heat co-generation and steam/gas turbine combine cycle and to develop the high temperature process utilizations [2]. The reactor core and the steam generator are housed in two separate steel pressure vessels connected by a vessel comprised of concentric piping with the innermost pipe being the hot gas duct. This is the modular concept.

The design of the HTR-10 began in 1992 and its construction in 1995. It reached criticality in December 2000 and full power operation in January 2003. All the process was supported by the Chinese National High Technology Program and was built by the Institute of Nuclear Energy Technology (INET), Tsinghua University. During the operation of the HTR-10, five safety verification experiments were performed in October 2003 to verify and to demonstrate the safety features of this kind of reactor [3] and to establish an experimental base for developing of high temperature nuclear process applications.

The HTR-10 reactor core is cooled by helium gas, moderated by graphite and uses Uranium spherical fuel elements (TRISO). The upper part of the reactor core has a cylindrical geometry and the lower part is cone-shaped. In the initial core, fuel elements and graphite dummy balls (graphite balls without nuclear fuel) constitute the pebble bed. The lower part of the core has only dummy balls. There is a discharging tube below the coned core to unload the fuel elements. Fig. 1 shows the HTR-10 core configuration [4]. Part of the helium coolant bypasses the main flow path, only 87% of the Rated Coolant Flow Rate (RCFR) effectively cools the fuel elements in the core.

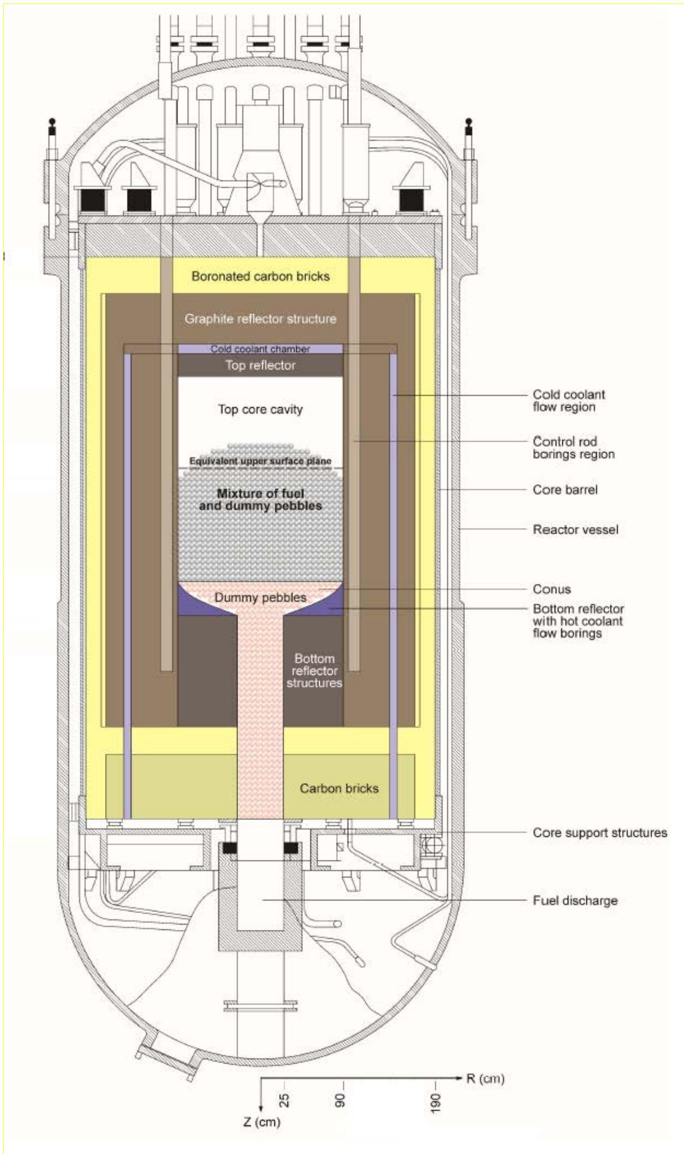

Figure 1 – HTR-10 core reactor configuration [4].

Table 1 gives some geometrical characteristics of the HTR-10 reactor core.

Table 1 - Geometrical characteristics of the HTR-10 reactor core [2]

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

This reactor is loaded with German type fuel elements with coated particles. The reactor equilibrium core contains about 27,000 fuel elements. The fuel elements and dummy balls are illustrated in Fig. 2. The design parameters are given in Table 2.

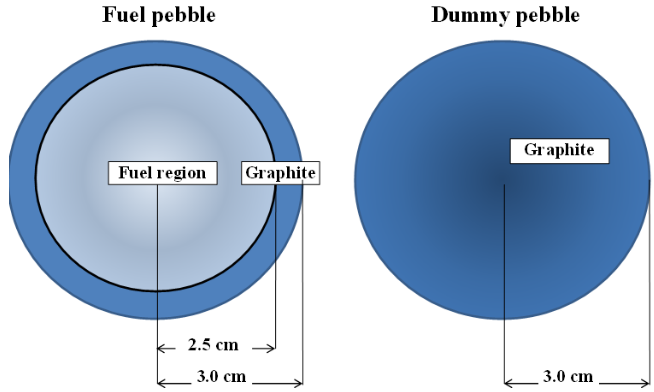

Figure 1 – Fuel elements and dummy balls.

Table 2 - Design parameters of fuel elements, dummy balls and loading ratio [2].

Parameter	Value
Fuel Element	
Diameter of ball, cm	6.0
Diameter of fuel zone, cm	5.0
Fuel	UO_2
Enrichment of U-238 (weight), %	17
Heavy metal (uranium) loading (weight) per ball, g	5.0
Density of graphite in matrix and outer shell, g/cm^3	1.73
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

The main thermal parameters of the HTR-10 core are show in Table 3.

Table 3 - Main thermal parameters of the HTR-10 [2].

Parameter	Value
Reactor thermal power, MW	10
Primary helium pressure, MPa	3.0
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

Investigations and development of models for the HTR-10 have been extensively done using several codes as verified, for example, in [2] and [6]. Some studies are also done with the aim to model pebble bed reactors with RELAP5-3D code [7]. In this way, the main aim of this work is demonstrate the capability of RELAP5-3D code to reproduce the thermal behavior of the HTR-10. An initial core model has been presented and analyzed using the RELAP5-3D.

THE RELAP5-3D MODELING

In this work, RELAP5-3D version 3.0, has been used to perform the simulations. The most prominent attribute that distinguishes the RELAP5-3D code from the previous versions is the fully integrated, multi-dimensional thermal hydraulic and neutron kinetic modeling capability [5]. There are two options for the computation of the reactor power in the RELAP5-3D code. The first option is the point reactor kinetics model that was implemented in previous versions of RELAP5. The second option is a multi-dimensional neutron kinetics model based on the NESTLE code developed at North Carolina State University. RELAP5-3D was modified to call the appropriate NESTLE subroutines depending upon the options chosen by the user. The neutron kinetics model in NESTLE and RELAP5-3D uses the few-group neutron diffusion equations. Two or four energy groups can be utilized, with all groups being thermal groups if desired. Point kinetics model was used in the simulations.

In the model developed, seven thermal hydraulic channels were considered to model the core. Seven pipes and seven corresponding heat structures (HS) have been modeled. It was calculated the average quantity of fuel pebbles corresponding to each modeled thermal hydraulic channel. The volumes of each fuel pebble were summed and the total volume corresponds to one cylindrical volume representing the HS of the channel. The heat structure simulates the power source of the channel and each one was axially divided according with the same quantity of the channel volumes. All HS have 12 radial meshes. The radial meshes were divided being 6 intervals to the fuel region and 6 intervals representing the graphite region. The dummy pebbles have not been simulated. In this word, the point reactor kinetics option was used in the calculations.

The coolant channels were represented by the component of the type pipe and were divided in axial volumes of 0.1 meters, being that the last volume of the channels 201, 202, 204, 205 and 206 has 0.05, 0.075, 0.025, 0.05 and 0.075 meters, respectively. To simulate the helium cross flow between the channels, single junctions were used to interconnect the volumes at the same level. Two time dependent volumes represent the inlet and outlet plenum.

The RELAP5 code was originally designed to simulate light water reactors (LWR). The hydrodynamic model is two-fluid model for flow of a two-phase steam-water mixture that allows noncondensible components as, for example, helium, in the steam phase and/or a soluble component in the water phase. In this way it is possible to use RELAP5 with only helium and no steam. Then working fluid only exists in one phase and behaves like an ideal gas [8]. Such criterions were used in the present model. The loss coefficients in the channels were adjusted to give the adequate mass flow rate. The RELAP5 card number 110 was defined as “helium”.

The RELAP5-3D model is illustrated in Fig. 3, where the time dependent volume components TMDPVOL 500 and 600 represent, respectively, the inlet and outlet plena. The SJ 400 and SJ 300 are single junctions and the pipes from 201 up to 207 represent the coolant channels. The channels were modeled in according with the Fig. 4, which shows how the channels have been defined to perform the model.

The HS are represented in the nodalization (Fig. 3) as the gray part in each thermal channel reaching the high of 1.8 meters from the core top (coolant inlet).

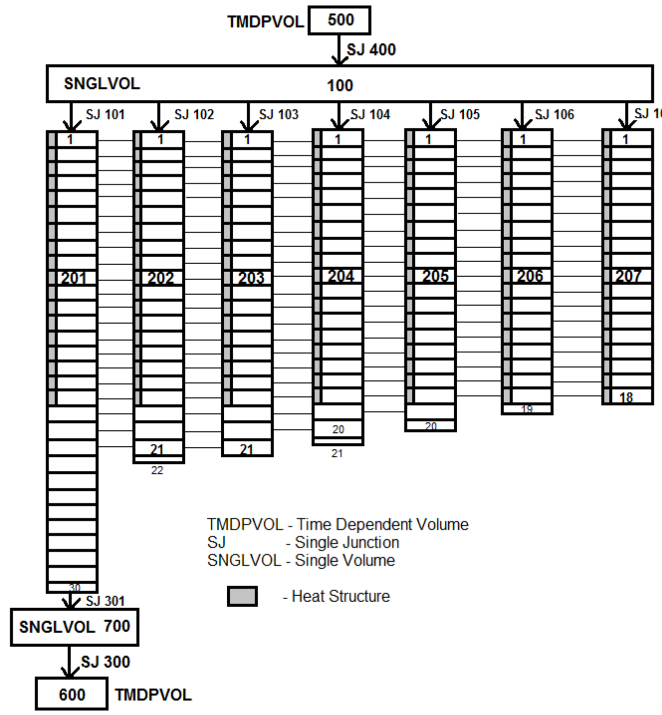

Figure 3 – RELAP5-3D model of HTR-10 core.

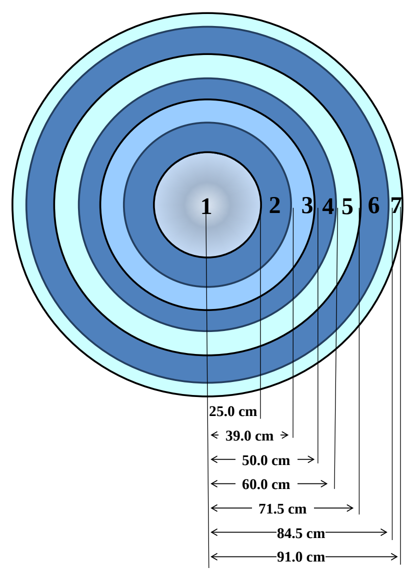

Figure 4 – HTR-10 channels nodalization in RELAP5-3D (core upper view).

STEADY STATE RESULTS

Table 4 shows some results obtained with this initial RELAP5 model for steady state behavior. As it can be verified, the parameters analyzed reached the permanent regime according with the reference data (from [2]).

To illustrate such results, some graphics are also presented. The inlet and outlet coolant temperature along the time is presented in Fig. 5. The coolant inlet temperature in the steady state is 522.99 K (249.84 ^{o}C) and the outlet is 974.80 K (701.65 ^{o}C) that are very close to the reference values presented in Table 3. The increase of coolant temperature along the core is then 451.8 ^{o}C in the calculation.

Table 4 – RELAP5-3D steady state results for HTR-10 (core) in comparison with the reference data [2].

Parameter	Reference Data [2]	RELAP5 -3D	Difference (%)
Power (MW)	10.0	10.0	0
Mass Flow Rate (kg/s)	4.32	4.26	1.4
Core Helium Pressure (MPa)	3.0	3.0	0
Temperature increase along the core (°C)	450.0	451.8	0.4

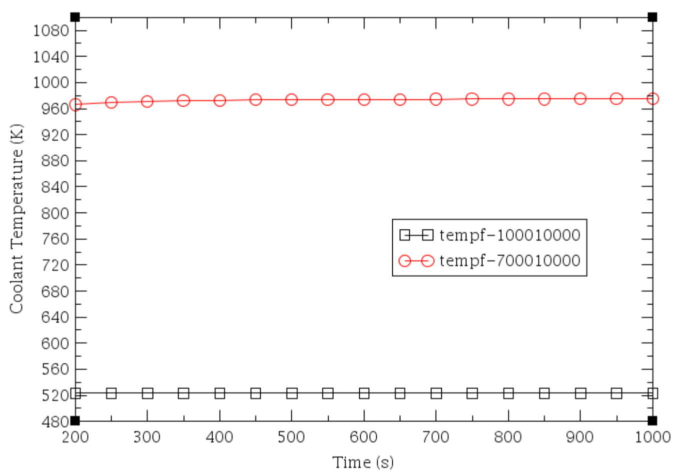

Figure 5 – Core coolant inlet and outlet temperatures time evolution.

The behavior of the coolant mass flow rate is presented in Fig. 6. In the steady state the helium mass flow rate is 4.26 kg/s. This value is close to the reference data (4.32 kg/s) presented in the benchmark [2] as can be verified in the Table 4.

In addition, Fig. 7 shows the core coolant pressure in the channel number 201. The coolant pressure practically remains constant along the core.

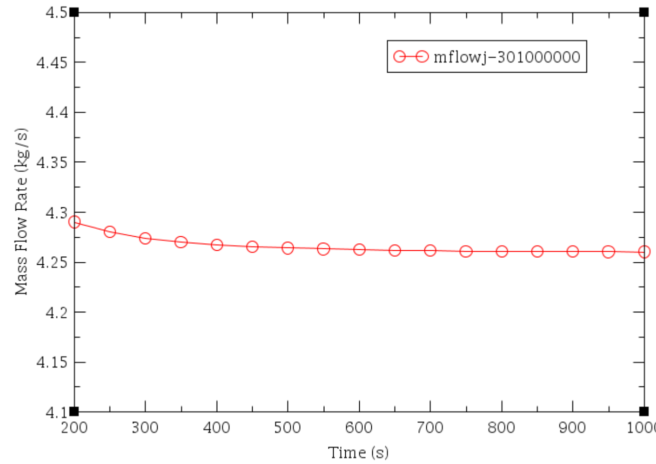

Figure 6 – Coolant mass flow rate

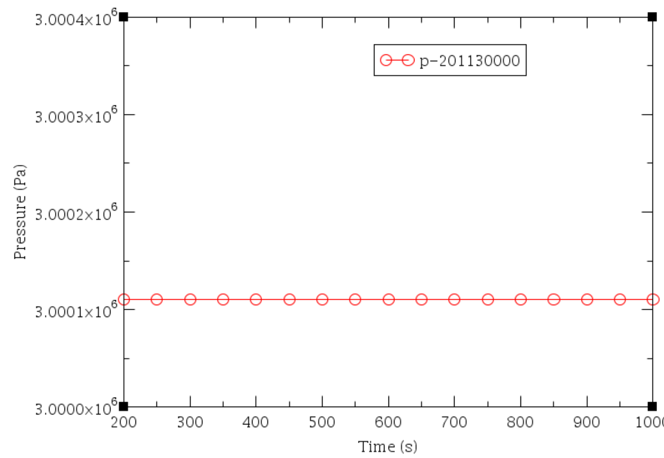

Figure 7 – Coolant pressure time evolution – channel 201.

Fig. 8 illustrates the fuel temperature in five axial levels of the heat structure 201 associated to the thermal channel 201. As it can be verified by the Fig. 9, the level 12 (1.2 meters below the top) presented higher value of temperature of 1161.5 K (= 888.4 °C). This value is inside of the range found by the benchmark participants that varies from 881.0 °C up to 1026.0 °C.

Fig. 10 presents a comparison between the core fuel centre temperature data obtained in this work and those found by the participants (maximum and minimum values) of the benchmark [2]. The fuel temperature calculated in this work rises along the core with expected average values but its behavior does not follow exactly the same as that presented in the benchmark. Investigations are being performed to find the possible causes, mainly to the fact that the fuel temperature in the initial high for the present model is underestimated in relation to the benchmark results.

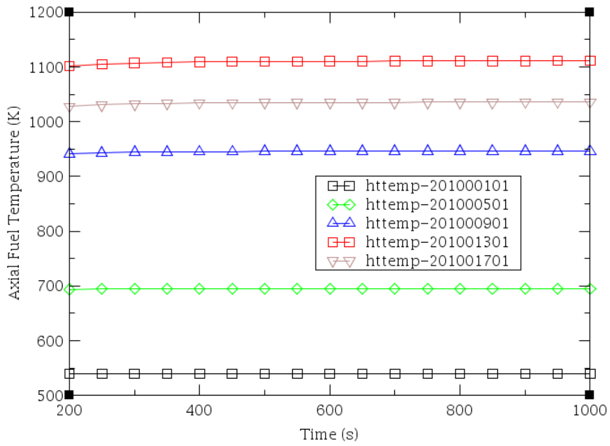

Figure 8 – Axial fuel temperature time evolution for the HS 201.

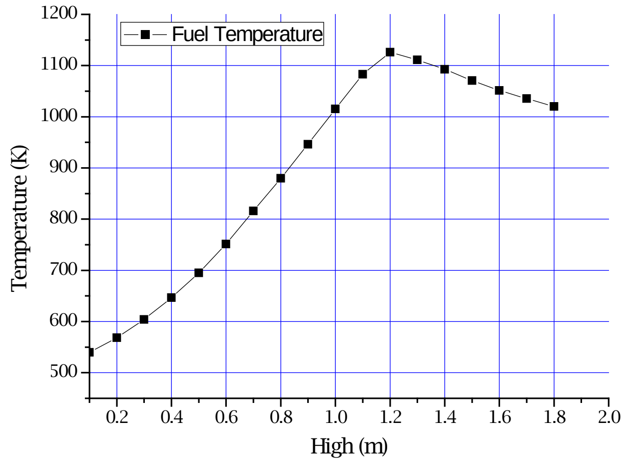

Figure 9 – Centre axial fuel temperature along the channel 201.

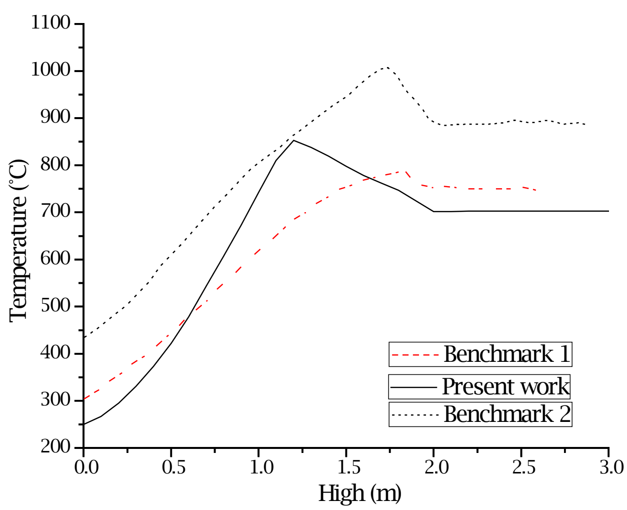

Figure 10 – Comparison of axial core centre temperature profiles.

CONCLUSIONS

In this study the core of the HTR-10 was simulated using the RELAP5-3D. The results of this initial work presented similar thermal behavior in comparison with the data from the reference document used to perform the model. However, investigations are necessary to find the possible causes for fuel temperature underestimated values in the core inlet in relation to the values found in the benchmark.

The cross-flow model inserted in the core seems to work well since the temperature distribution reached a steady state behavior as expected. Future work consists in to incorporate more reactor details beyond the core in the model and also to simulate transient events. To reach more realistic results the tridimensional representation of the core will be also developed. The future idea is to incorporate the neutronic parameters in the RELAP5-3D model to perform a 3D neutron kinetic/thermal coupling calculation of the HTR-10.

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