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HTR steady state and transient thermal analyses

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ABSTRACT

A thermal model to simulate the steady state and transient behavior of the 10 MW pebble bed high temperature gas cooled reactor (HTR-10) is presented in this work. The helium cooled HTR-10 was designed, constructed and operated by the Institute of Nuclear and New Energy Technology (INET), in China. In this study, a simulation is performed using the RELAP5 code. 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. The simulation is demonstrated to be in good agreement, showing that the developed model is capable of reproducing the thermal behavior of the HTR-10 in steady state and transient operation conditions.

Introduction

The Pebble Bed High Temperature Gas cooled Reactor (HTGR-PM) is seen as one of the best candidates for the next generation reactors [1]. In particular, the High Temperature gas cooled Reactor (HTR-10) is a 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 establish 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.

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 [2]. This is the modular concept. 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. HTR-10 safety verification experiments were performed in 2003 [3].

The HTR-10 reactor core is cooled by helium gas, moderated by graphite and uses Uranium spherical fuel elements

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

Parameter	Value
Equivalent diameter, cm	180.0
Average height, cm	197.0
Volume, m^3	5.0
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	50.0

(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. 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. Table 1 gives some geometrical characteristics of the HTR-10 reactor core. This reactor is loaded with German type fuel elements with coated particles. The reactor equilibrium core contains about 27,000 fuel elements. The design parameters of the fuel elements and the dummy balls are given in Table 2. The main thermal parameters of the HTR-10 core are show in Table 3.

Developed model in the RELAP5 code

Investigations and development of models for the HTR-10 have been extensively done using several codes as verified, for example, in Ref. [2] and in Ref. [4]. Some studies have also done with the aim to model pebble bed reactors with the RELAP5-3D code [5]. In this work, RELAP5-3D version 3.0.0 has been used to perform the HTR-10 model and 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 [6]. However, in the present model, the neutron kinetic model was not used and only the point reactor kinetics model was considered in the calculations. In the future, with the thermal model adequately verified, a neutron kinetic/thermal coupled simulation will be performed inserting adequate macroscopic cross sections in the neutronic part of the code.

Table 2 – Design 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

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

Parameter	Value
Reactor thermal power, MW	10.0
Primary helium pressure, MPa	3.0
Average helium temperature at reactor outlet, °C	700.0
Average helium temperature at reactor inlet, °C	250.0
Helium mass flow rate at full power, kg/s	4.32

In the model developed, seven thermal hydraulic channels were considered to represent the core. Seven pipes and seven corresponding heat structures (HS) have been modeled. The average quantity of fuel pebbles corresponding to each modeled thermal channel was calculated. 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 to match the channel volumes. All HS have 12 radial meshes. The radial meshes were divided into 6 intervals in the fuel region and 6 intervals representing the graphite region. The dummy pebbles have not been simulated.

The RELAP5-3D model is illustrated in Fig. 1, where time dependent volume components TMDPVOL 500 and 600 represent, respectively, the inlet and outlet plena. The SJ 400 and SJ 300 are single junctions. The coolant channels were represented by the component of the type pipe and were divided in axial volumes of 0.1 m. The last volume of channels 201, 202, 204, 205 and 206 are 0.05, 0.075, 0.025, 0.05 and 0.075 m, respectively. To simulate the helium cross flow between the thermal channels, single junctions were used to interconnect the volumes at the same level. Time dependent junctions (components 101, 102, 103, 104, 105 and 107) were inserted in the model to control the mass flow in the core. The channels were modeled according to Fig. 2, which shows how the channels have been defined for the model. The heat structures are represented in the nodalization (Fig. 1) as the gray part in each thermal channel reaching the height of 1.8 m from the core top (coolant inlet).

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, such as 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 the working fluid only exists in one phase and behaves like an ideal gas [7]. 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".

Results of steady state calculation

In Table 4 are shown some results obtained for steady state behavior. As it can be verified, the parameters analyzed reached the permanent regime according with the reference

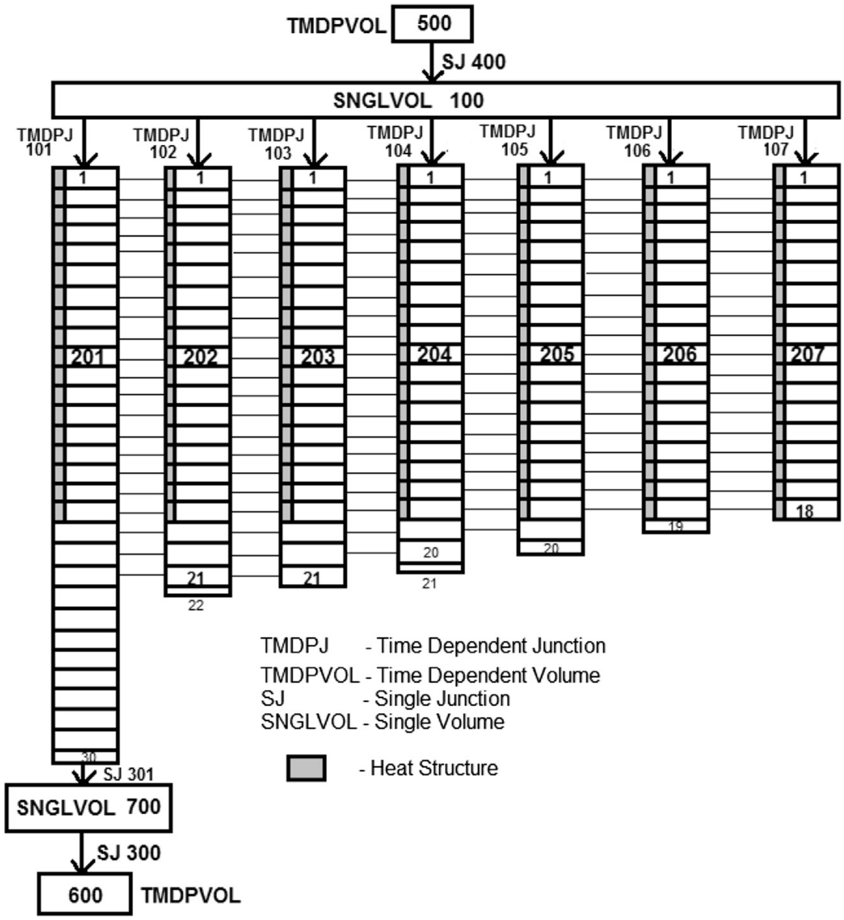

Fig. 1 – RELAP5-3D model of HTR-10 core.

data obtained from International Atomic Energy Agency document [2].

The inlet and outlet coolant temperature time evolution are presented in Fig. 3. The coolant inlet temperature in the steady state is 522.99 K (249.84 °C) and the outlet is 974.80 K (701.65 °C), very close to the reference values presented in Table 3. The increase of coolant temperature along the core is then 451.8 °C in the calculation.

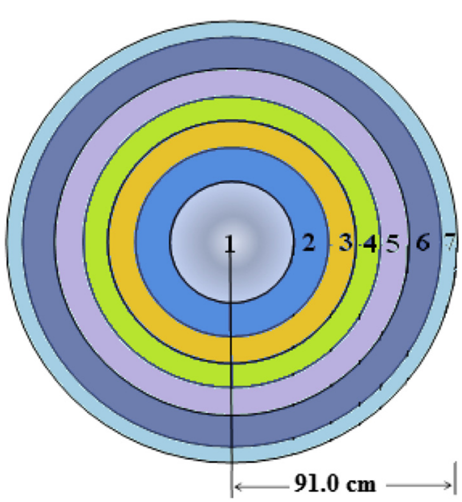

Fig. 2 – HTR-10 thermal channels nodalization in RELAP5-3D (core upper view).

Fig. 4 presents a comparison between the core fuel centre temperature data obtained in this work and those found by the participants of the benchmark [2]. Only the maximum and minimum results of the benchmark, represented in Fig. 4 by Benchmark 2 and Benchmark 1, respectively, are represented. The fuel temperature calculated in this work rises along the core with expected average values although its behavior does not follow the benchmark exactly. Investigations are being performed to find the possible causes, mainly to the fact that the fuel temperature in the initial height for the present model is underestimated in relation to the benchmark results.

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

Parameter	Reference data	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
Core temperature increase (°C)	450.0	451.8	0.4

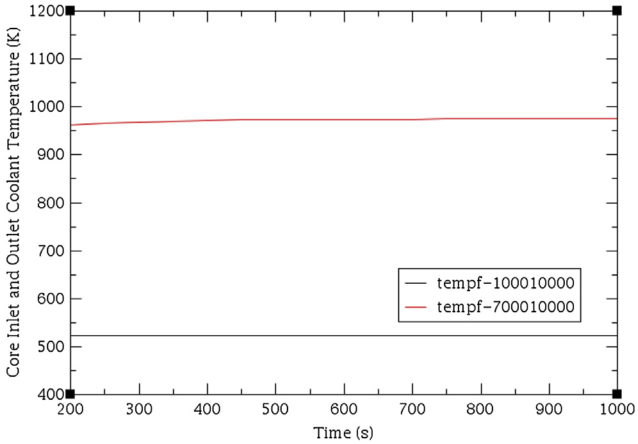

Fig. 3 – Core coolant inlet and outlet temperatures time evolution.

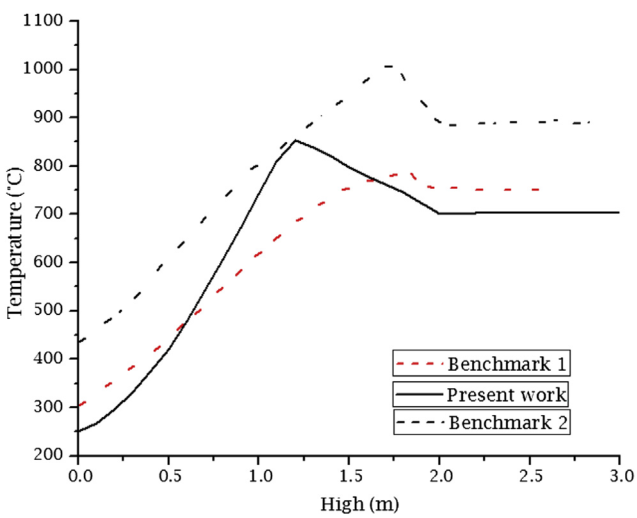

Fig. 4 – Comparison of axial core centre temperature profiles.

Results of loss of coolant calculation

Following the safety requirement of the fourth generation of nuclear reactors, the reactor must remove the decay heat passively from the core under any designed accident condition keeping the maximum fuel temperature below 1873.0 K to contain all fission products inside the SiC layer of TRISO coated fuel [3]. An accident of loss of helium flow has been simulated to verify the behavior of the core. To simulate this transient, the coolant mass flow rate decreased from about 0.4 kg/s to zero in time dependent junction 101 (see Fig. 1) that feeds the thermal channel 201. This transient begins at 500 s of steady state calculation. Fig. 5 shows the decrease of the coolant mass flow rate in the junction 101 while the others junctions (102, 103, 104, 105, 106 and 107) remain with the same mass flow rate. The values of the mass flow rates are different because the seven channels have different flow areas.

The coolant temperature in channel 201 starts to increase after the beginning of the transient at 500 s. Although the

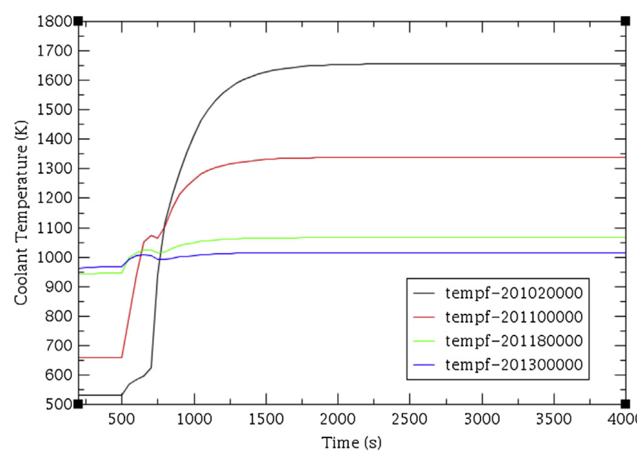

Fig. 6 – Coolant temperature in some axial points of the thermal channel 201.

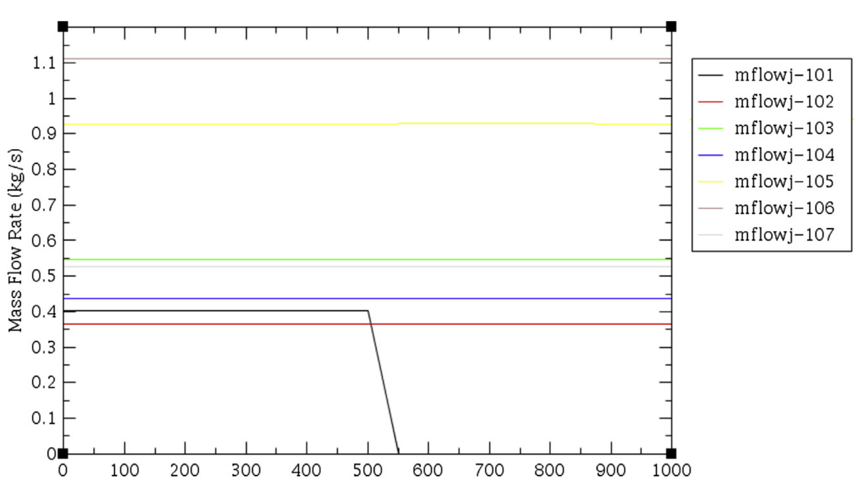

Fig. 5 – Coolant mass flow in the inlet of the core thermal channels.

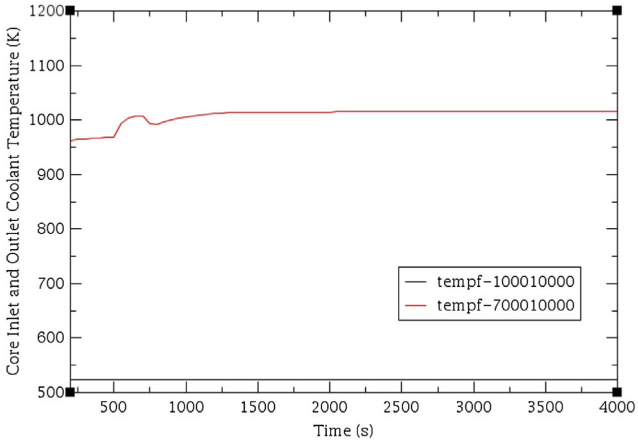

Fig. 7 – Coolant temperature at inlet and outlet of the core.

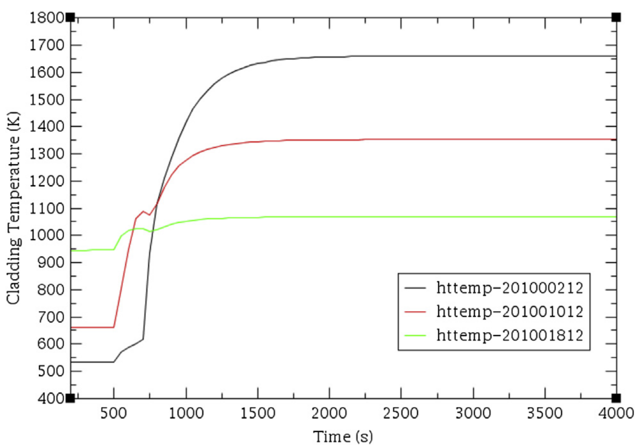

Fig. 8 – Cladding temperature of the heat structure 201 at axial levels 02, 10 and 18.

channel was able to receive helium from the others channels by the coolant cross flow junctions, this was not enough to avoid the high coolant temperatures in the upper part of the channel as shown in Fig. 6. The maximum temperature value in channel 201 is observed in volume 2 corresponding to about 1658.7 K. However, it is possible to observe in Fig. 7 that the outlet core coolant temperature does not increase considerably. It reaches about 1014.9 K, only 40.1 K higher than the steady state operation condition.

It was observed in the simulation that the fuel and cladding temperatures reached maximum values of about 1660.0 K which is below the fuel melting point. This value is below the maximum safety limit for the fuel temperature of 1873.0 K as described before. The shutdown action was not considered during the transient simulation. The cladding temperature time evolution is shown in Fig. 8. The fuel temperature presented values very close to the cladding temperature.

Conclusions

The HTR-10 core has been simulated using the RELAP5-3D code with a point kinetics model. The results presented similar thermal behavior in comparison with the data from the IAEA reference document. The cross-flow model inserted in the core seems to work well since the temperature distribution reached a steady state behavior as expected. In addition a loss of flow transient in the core was simulated. The results indicated that after the accident the fuel temperature reaches a new steady state with a temperature value below the safety limit.

Future work consists in incorporating more reactor details beyond the core in the model and also to simulate several possible transient events. To reach more realistic results the tridimensional representation of the core will be also developed. A future step 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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