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# A PRELIMINARY NEUTRONIC EVALUATION OF THE HIGH TEMPERATURE GAS-COOLED TEST REACTOR HTR-10 USING THE SCALE 6.0 CODE

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## ABSTRACT

The High Temperature Gas-cooled Test Reactor HTR-10 is a 10 MW modular pebble bed type reactor, which core is filled with 27,000 spherical fuel elements, e.g. TRISO coated particles. This reactor was built by the Institute of Nuclear Energy Technology (INET), Tsinghua University, China, and its first criticality was attained on December 1, 2000. The main objectives of the HTR-10 are to verify and demonstrate the technical and safety features of the modular HTGR (High Temperature Gas-cooled Reactor) and to establish an experimental base for developing nuclear process heat applications. In this work, using the Standardized Computer Analysis for Licensing Evaluation (SCALE) 6.0, a nuclear code developed by Oak Ridge National Laboratory (ORNL), the HTR-10 first critical core is modeled by the DEN/UFMG. The $k_{eff}$ was obtained and compared with the reference value obtained by the Idaho National Laboratory. The result presents good agreement with experimental value. The goal is to validate the DEN/UFMG model to be applied in transmutation studies changing the fuel.

## 1. INTRODUCTION

The graphite-moderated helium gas-cooled reactor HTR-10 is a small pebble-bed test reactor intended to develop pebble-bed reactor (PBR) technology in China, with power of 10 MWt. It is used to test and develop fuel, verify PBR safety features, demonstrate combined electricity production and co-generation of heat, and provide experience in PBR design, construction, and operation. Its construction was started in 1995 and it reached its first criticality in December 2000. It was operated in full power condition in January 2003 [1].

To investigate neutronic behavior concerning the HTR-10 in future studies, a benchmark should be successfully reproduced to validate geometrical parameters, using the SCALE6 nuclear code, providing more reliable results. The goal is to reproduce part of the HTR-10 benchmark [1] using the SCALE6 to support future studies using reprocessed fuels. In this work, it was used CSAS6/KENO-VI control sequence of SCALE [3,4,5] and the fine-group 238GROUPNDF5 cross-section library.

## 2. OVERVIEW

The HTR-10 is located at the Institute of Nuclear and New Energy Technology (INET), a research institute of Tsinghua University in Beijing and its project was approved by the Chinese State Council in March 1992, ground was broken in 1994, and construction was completed in 2000, when the initial criticality was achieved. Figure 1 shows the regions of HTR-10 initial configuration [1].

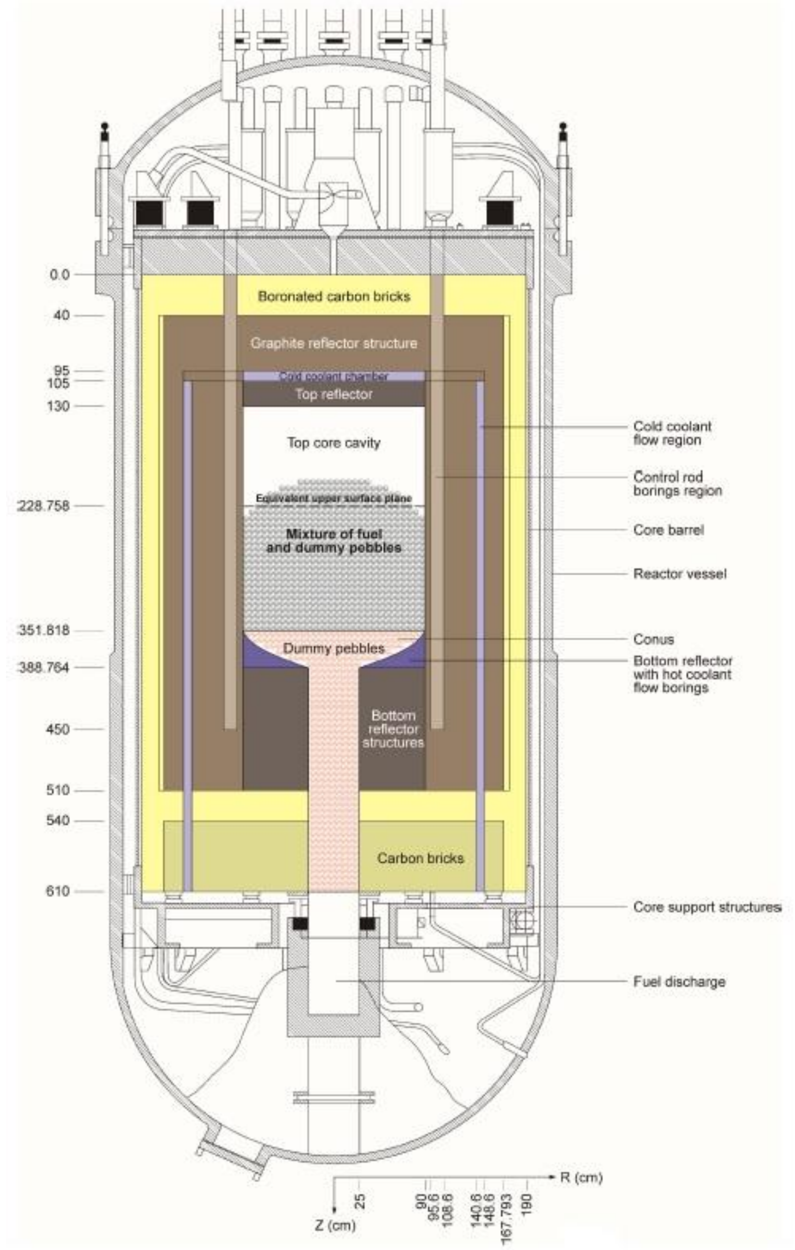
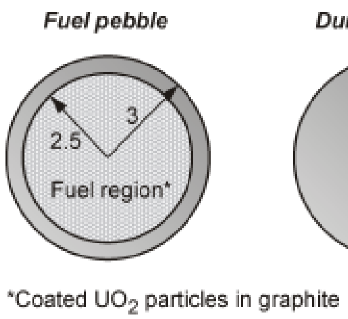



**Figure 1: Regions of HTR-10 Initial Configuration (dimensions are in cm).**

The HTR-10 is loaded with German type spherical fuel elements (6.0 cm in diameter) with coated particles. The reactor equilibrium core contains about 27,000 spherical fuel elements forming a pebble bed with 180.0 cm in diameter and 197.0 cm in average height [2]. The spherical fuel elements move through the reactor core in a multi-pass pattern, where dummy balls (graphite balls without nuclear fuel) and fuel balls are mixed in different percentages. After the first criticality is reached, mixed balls of the same ratio will be further loaded to full core in order to make the reactor capable of being operated at full power; the full core has an estimated volume of 5 $m^3$[2].

Figure 2 shows fuel and dummy pebbles, while figure 3 shows TRISO fuel particle [1].



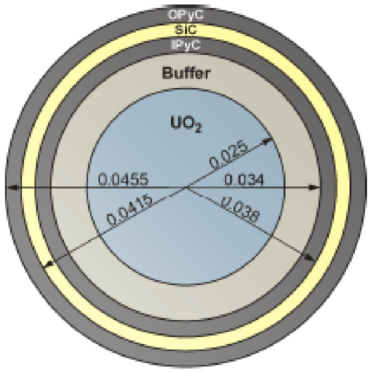




**Figure 2: Fuel and dummy pebbles (dimensions are in cm).**

**Figure 3: TRISO fuel particle (dimensions are in cm).**

The average number of TRISO fuel particles per pebble is 8,335 for a packing fraction of 61% [1]. Table 1 shows the major design parameters of the HTR-10 [6].

**Table 1: Major design parameters of the HTR-10**

| *Item* | *Unit* | *Value* |
|---|---|---|
| Thermal power | MW | 10 |
| Reactor core diameter | cm | 180 |
| Average core height | cm | 197 |
| Primary helium pressure | MPa | 3.0 |
| Average helium temperature at reactor inlet/outlet | ºC | 250/700 |
| Helium mass flow rate at full power | $kg.s^{-1}$ | 4.3 |
| Average core power density | $MW.m^{-3}$ | 2 |
| Power peaking factor | | 1.54 |
| Number of control rods in side reflector | | 10 |
| Number of absorber ball units in side reflector | | 7 |
| Nuclear fuel | | $UO_2$ |
| Heavy metal loading per fuel element | g | 5.0 |
| Enrichment of fresh fuel element | % | 17.0 |
| Number of fuel elements in core | | 27,000 |
| Fuel management | Multi-pass | |
| Average residence time of 1 fuel element in core | EFPD | 1,080 |
| Maximum power rating of fuel element | kW | 0.57 |
| Maximum fuel temperature (normal operation) | ºC | 919 |
| Maximum burn-up | $MWd.tHM^{-1}$ | 87,072 |
| Average burn-up | $MWd.tHM^{-1}$ | 80,000 |
| Maximum thermal flux in core (E>1.86 eV) | $n.cm^{-2}.s^{-1}$ | $3.43x10^{-13}$ |
| Maximum fast flux in core (E>1 MeV) | $n.cm^{-2}.s^{-1}$ | $2.77x10^{-13}$ |

## 3. METHODOLOGY

The reactor core was modeled to achieve the highest level of accuracy available in the literature according to all geometrical and material composition data described in reference [1]. Literature describes two models of HTR-10: the simplified and the high-fidelity model. The main differences between these two models are that the first one represents the upper surface of the core as a horizontal plane and radial zones containing borings are azimuthally homogenized, but the total number of pebbles in the core remains the same in both models [1].

Figure 4 shows the zones of HTR-10 for high-fidelity model [1].

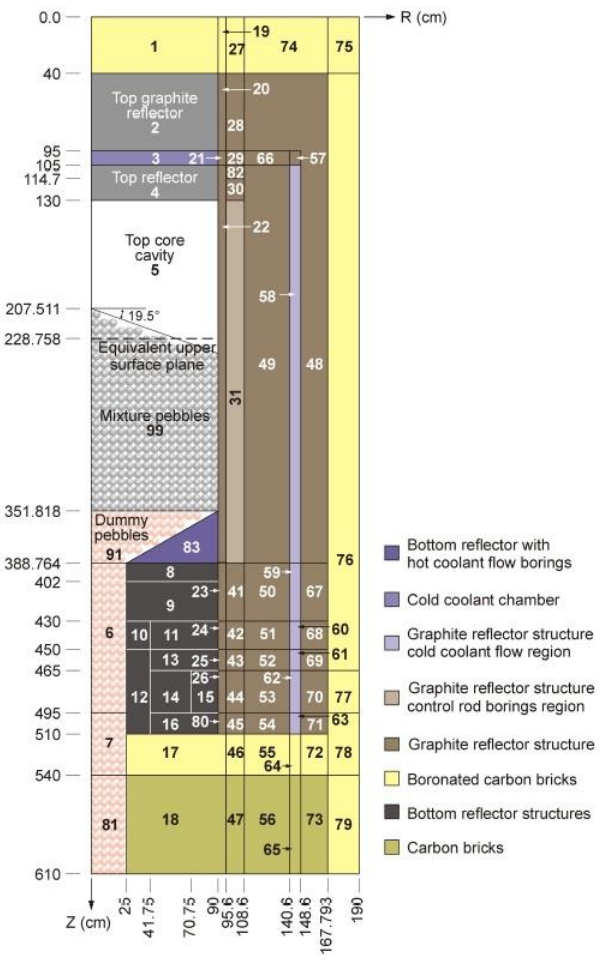



**Figure 4. Zones of HTR-10 for high-fidelity model (dimensions are in cm).**

It was modeled the following structures: boronated carbon bricks, bottom reflector structures, bottom reflector with hot coolant flow borings, carbon bricks, cold coolant chamber, cold coolant flow region, control rods regions, dummy pebbles – graphite, dummy pebbles, fuel structure (kernel, buffer, inner pyrolitic carbon, SiC layer and outer pyrolitic carbon), graphite matrix, graphite reflector structures, pebble shell – graphite, top core cavity, top graphite reflector and top reflector. It was simulated 107 different mixtures, with 317 composition specifications. The regular fuel kernel simulated was 1.4982202E+01 wt% $^{235}$U, 7.3148525E+01 wt% $^{238}U$, 1.1868920E+01 wt% $^{16}O$, 6.4973281E-05 wt% $^{10}B$ and 2.8755105E-04 wt% $^{11}B$. The uranium mass per pebble is 5 g and the equivalent boron in uranium is 4 ppm. The fuel and the dummy pebble were simulated with the dimensions showed in figure [2] while the TRISO particles were simulated with the dimensions presented in figure [3]. The atomic densities used were: 10.4 g.cm$^{-3}$ for fuel kernel, 1.1 g.cm$^{-3}$ for buffer layer, 1.9 g.cm$^{-3}$ for inner pyrolitic carbon layer, 3.18 g.cm$^{-3}$ for SiC layer, 1.9 g.cm$^{-3}$ for outer pyrolitic carbon layer and 1.73 g.cm$^{-3}$ for graphite matrix.

## 4. RESULTS AND DISCUSSION

It was used the 238-group neutron energy library in the simulations, but other three libraries were tested: the 44-group, the continuous energy library version 6 and the continuous energy library version 7. Values of $k_{eff}$ were not obtained with these three libraries because the 44-group does not have the isotopes $^{28}Si$ and $^{40}Ar$ and the continuous energy libraries cannot be used with doubly-heterogeneous systems.

Table 2 was adapted from reference [1] and shows experimental and calculated values of $k_{eff}$ with two different nuclear codes.

**Table 2: Calculated $k_{eff}$ for high-fidelity model**

| Case | $k_{eff}$ |
|---|---|
| Experimental results [1] | 1.00000 |
| MCNP result [1] | 1.01190 ± 0.00021 |
| SCALE 6.0 results (238 groups) | 1.01366 ± 0.00028 |

To identify the impact of different nuclear data and code-specific numerical approximations, the standard deviation (SD) and relative standard deviation (RSD), were calculated respectively as ($N_i$ is the value of $k_{eff}$):

$$Average = \frac{1}{I}\sum_{i=1}^{I} N_i \qquad (1)$$

$$SD = \sqrt{\frac{\sum_{i=1}^{I}(N_i - Average)^2}{I}} \qquad (2)$$

$$RSD = \frac{SD}{Average} \qquad (3)$$

Table 3 shows the results. The good agreements between the results obtained encourage the continuity of the studies.

**Table 3: SD and RSD**

| Case | SD | RSD(%) |
|---|---|---|
| MCNP; SCALE | 0.001245 | 0.1229 |
| MCNP; SCALE; Experimental | 0.007431 | 0.7368 |

The simulations were done in a personal computer with an Intel(R) Core(TM) Quad CPU Q8200 2.33 GHz, 8 GB RAM and Microsoft Windows 7 Ultimate 64 bits installed. The computation time was 3,048.64 minutes (50.81 hours) for SCALE 6.0 with 238-group neutron energy library.

## 5. CONCLUSIONS

With the use of the CSAS6 control module of ORNL SCALE 6.0 nuclear code and the 238-group neutron energy library, a very complex geometrical high-fidelity model of the Chinese HTR-10 based on cited references was modeled and evaluated. All the original geometrical and material composition characteristics were maintained in details. The $k_{eff}$ obtained in the simulations is according to the benchmark. In a preliminary analysis, it can conclude that the HTR-10 modeled by DEN/UFMG using SCALE 6.0 can be used to predict HTR-10 $k_{eff}$ with its standard fuel.


## ACKNOWLEDGMENTS

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