EUROPEAN COMMISSION
5th EURATOM FRAMEWORK PROGRAMME 1998-2002
KEY ACTION : NUCLEAR FISSION

HTR-N

HTR-N & N1
High-Temperature Reactor Physics and Fuel Cycle Studies
(EC-funded Projects: FIKI-CT-2000-00020 & FIKI-CT-2001-00169)

Work Package: 1	HTR-N project document No.: HTR-N-03/06-D-1.3.1				Q.A. level: 1
Task: <1.3.1>	Document type: Deliverable				Document status: Final
Issued by: CEA	original document No.:	Annex pg. 0	Rev. 0	Q.A. level	Dissemination level: PU

Document Title

Calculational Results on the HTR-10 First Criticality

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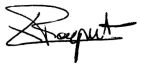
Notes

Electronic filing: ***SINTER***

Electronic file format: ***PDF file***

Rev.	Date	Short Description	Prepared by	Partners' Review	Control WP Manager	Approval Project Manager
						
0	5-06-2003	first issue	CEA	FZJ / NRG	CEA X. Raepsaet	FZJ von Lensa

Chronology and history of revisions

Rev.	Date	Description
0	5-06-2003	First version

I – INTRODUCTION

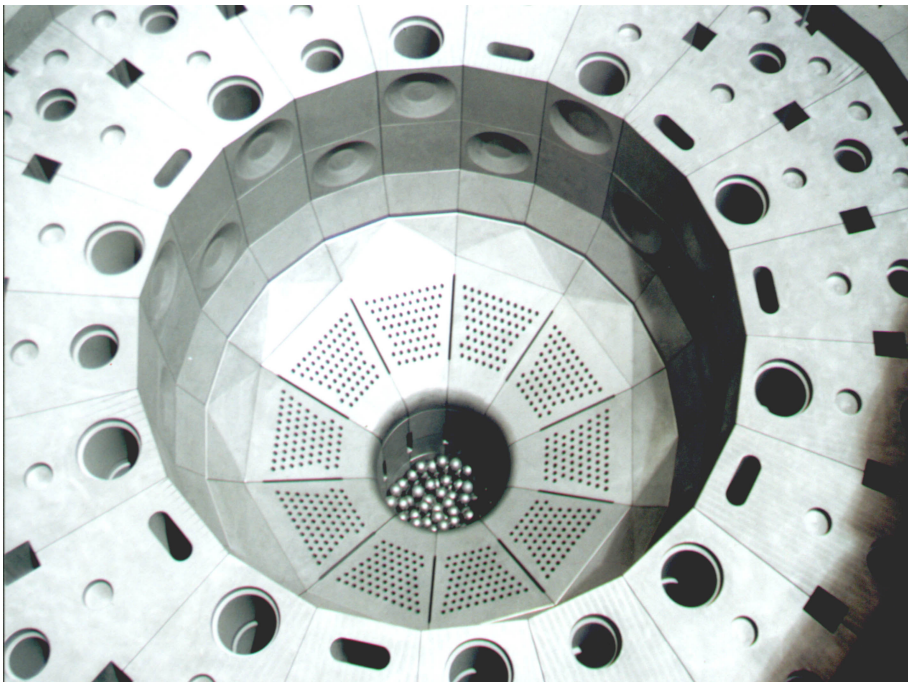
This report constitutes a deliverable for the European contract HTR-N. It provides a summary of the works performed for the **task 1.3** of the **Work Package 1 (WP1)** during one and half year. Three partners are involved in this work package: FZJ in Germany, NRG in Netherlands and the CEA (work package manager).

Core physics calculation tools for HTR modelling are available in the European participating organisations both for **pebble bed** and **block-type** fuel. They are validated for the former HTR concept conditions and a limited set of fuel types, such as uranium or thorium. However, new annular core configurations, ultra high-burn-up, actinide burning or waste minimisation strategies impose additional requirements. For all these reasons decribed above, the objectives of the **WP1** are:

- to contribute to the code validation taking into account these new requirements
- to qualify and improve the methods for modeling the HTGR

The work of the **WP1** is based on the HTTR and HTR-10 reactors recently started-up and for which benchmarks have been proposed through the IAEA (CRP-5). The HTTR with annular core configurations at first criticality and the HTR-10 provide experimental data for the validation of the codes in an extended spectrum of fuel cycles and core geometries. The case of the HTTR corresponds to the **task 1.1 and 1.2** of the **WP1** and the results are gathered in the final reports :
HTR-N-02/5-D-1.1.1 and HTR-N-02/5-D-1.2.1

As shown below, the 10 MWth reactor HTR-10 at INET is a Chinese prototype pebble bed gas cooled reactor. The reactor will provide important data as it is the first reactor that is designed to demonstrate the self-acting decay heat removal principle of modular HTR. He has reached criticality at the end of 2000 and the major European codes have been used to calculate this first criticality (subject of this report). For a description and the main data of the reactor, reference is made to the Benchmark Description [1].



The present comparative synthesis of the calculations performed with different codes systems commonly used in Europe relies on the works of U. Ohlig and H. Brockmann from FZJ, J.B.M. de Haas from NRG and F. Damian at CEA.

After a brief presentation of the calculation scheme used to model the HTR-10, both kind of results obtained with **original benchmark data** and **revised data** are presented and discussed. Diffusion and Monte Carlo codes have been used. Diffusion calculations lead to some quite acceptable results compare to the experiment. As far as the Monte Carlo calculations are concerned, the preliminary approach for modelling the HTR-10 underscores an overestimation of the core reactivity due to the hypotheses made for describing the stochastic geometry.

II - DESCRIPTION OF THE DIFFERENT HTR-10 CORE MODELS

II.1 - CALCULATIONAL METHOD WITH VSOP CODE SYSTEM

The benchmark problems are calculated using the following parts of the VSOP code system [2]: the ZUT [3], GAM-1 [4], THERMOS [5], and the CITATION code [6]. The code system considers the following main features of pebble bed reactors:

- the double heterogeneous nature of the spherical fuel elements with the coated particles,
- the streaming correction of the diffusion constant in the pebble bed,
- detailed leakage feedback when using few group homogenised cross sections in the whole core diffusion calculation,
- the use of anisotropic diffusion constants in the upper cavity and in the channels of the control rods and of the small absorber balls.

Library, nuclear data and calculational scheme

Self-shielded cross sections in the resolved resonance region of U^{238} are generated by the ZUT code taking into account the double heterogeneity of the fuel elements. The resonance parameters processed by the ZUT code are taken from the **JEF-1** data file [7].

Spectrum calculations are performed by a combination of the epithermal spectral code GAM-1 and the thermal cell code THERMOS using two fine group nuclear data libraries [8] based on JEF-1 and ENDF/B-V: an epithermal library with a **68 energy group** structure ranging from 10 MeV to 0.414 eV, and a thermal library with **30 fine energy** groups in the energy range from 2.05 eV to 10^{-5} eV.

The **0-d spectrum code** GAM-1 applies the P_1–approximation to homogenised spectrum zones. Neutron leakage between adjacent spectrum zones is considered by buckling terms determined from the fast and epithermal leakage values of the whole core diffusion calculation and transferred to the spectrum calculation. At the beginning all **buckling terms are zero**. Condensation of the fine energy groups is performed to three broad groups used in the diffusion calculation.

The thermal cell code THERMOS performs **1-d transport** spectrum calculations using a heterogeneous cell model. The grain structure of the coated particles inside the fuel elements is taken into account. The neutron exchange between neighbouring spectrum zones is considered in the form of albedo terms at the cell surface. They are calculated from the thermal leakage value of the diffusion calculation and booked into the fine thermal energy groups. At the beginning of the leakage iteration a white boundary condition is assumed at the outer surface of the cell. A cell-weighted condensation of the 30 fine group cross sections is performed to one broad energy group used in the diffusion calculation. The broad group structure is given in Table 1.

Group	Upper Energy Boundaries (eV)
1	$1.0\ 10^{7}$
2	$1.111 \cdot 10^{5}$
3	29.0
4	1.86

Table 1: Group Structure in the VSOP Diffusion Calculations

The eigenvalues and flux distributions of the whole reactor are calculated by the diffusion code CITATION in 2-d r,z and 3-d r,φ,z geometry. A streaming correction of the diffusion constants in the pebble bed is calculated according to Lieberoth [9]. In the upper void region the method of

Gerwin and Scherer [10] is applied. The geometric dimensions of the fuel elements and core components and the atom number densities of the materials given in [1] are used.

Cell calculations with THERMOS

According to the initial core loading of the HTR-10, -the inner cone region at the bottom of the core is filled with graphite balls, then, a mixture of fuel and graphite balls in the ratio of 0.57 to 0.43 is loaded gradually to approach first criticality- two kinds of spherical unit cell models are chosen: a fuel and a dummy (graphite) unit cell model.

The dummy unit cell model consists of two zones: a graphite zone with r = 3 cm, and a void corresponding to the filling fraction f = 0.61 of the dummy balls in the cone region.

The fuel unit cell model shown in Fig.1 consists of 4 zones: a fuel zone, containing the coated particles (cp) and the graphite matrix, the graphite shell, a helium or air gap as third zone corresponding to the packing fraction of 0.61, and a graphite/ helium or air mixed zone as mock-up for the moderator pebbles and their corresponding void space. The coated particles grain structure inside the fuel zone is treated by effective macroscopic total cross sections replacing the homogenised macroscopic cross sections of the matrix and the coated particles. The k_∞-value of the fuel cell calculation is: 1.7475 at T= 293 K using a white boundary condition.

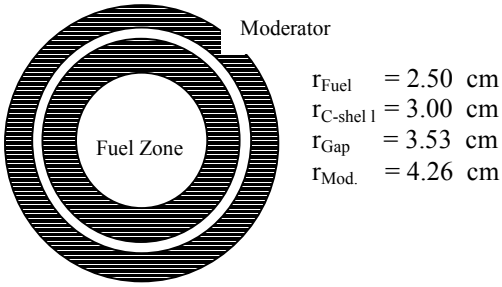


Fig 1: 1D Spherical Fuel Cell for THERMOS

Whole Reactor Calculations

Using the **4-group** constants from the GAM-1 and THERMOS cell calculations the whole HTR-10 reactor is modelled with the CITATION diffusion code. Two geometric models are chosen in order to compare the results: a **2-d r,z** model as presented in Fig. 4 of Ref. [1] and a **3-d r,φ,z** model. The assembly is modelled by dividing the volume into different spectrum zones each comprising several material compositions given in Table 2 of Ref. [1]. There are 16 different spectrum zones. The active core and the adjacent reflector regions are subdivided into smaller spectrum zones in order to take into account the core/reflector coupling accurately during the leakage iteration.

In the 3-d calculations the boring holes of the ten control rods and the seven channels for the small absorber balls are explicitly taken into account using anisotropic streaming coefficients in the remaining voids. All other channels or holes, as e.g. the helium flow channels, the hot gas duct, the three irradiation channels, are neglected. The control rods in the reflector in their withdrawn position (the lower end of the control rods is at the axial position of 114.7 cm) are also considered in the 3-d geometry.

II.2 - CALCULATIONAL METHOD WITH WIMS/PANTHER

The HTR-10 has been modelled in the PANTHERMIX code [11], a combination of the 3-D diffusion reactor code PANTHER 5.1 [12] coupled to the 2-D thermal hydraulics code THERMIX./DIREKT [13] The nuclear data necessary for the PANTHER code has been generated by means of the WIMS8 [14] code system.

Cell Calculations

The calculations with WIMS8 has been performed with an adapted version of the standard **172 groups** 1997 WIMS library based on **JEF-2.2**. Adaptations has been made for Pu, Am and Cm isotopes to extend the range of temperatures and of the potential scattering to improve the resonance treatment.

The method followed to overcome the impossibility of modelling the pebbles on coated fuel particle (CFP) scale is to model **a cylindrical cell with an equivalent radius**. This method has been proved adequate in the PROTEUS benchmark. The spherical pebble is transformed into an infinite long cylinder but with the same mean chord length. The mean chord length of a concave body is given by the simple relation: 4 times the ration of volume over surface. So the fuelled part of the pebble (radius 2.5cm) is translated into a cylinder of 1.667 cm radius and contains the fuel kernels, coatings and matrix graphite. This cylinder is surrounded by an annulus of radius 2.191 cm to accommodate the unfuelled shell of 0.5 cm based on conservation of volume ratio. Finally the outer cylinder which contains the admixed moderator pebbles and the void between the pebbles this radius amounts to 3.716 cm in our model.

These dimensions and densities are fed into the WIMS module WPROCOL to produce collision probabilities for use in the resonance treatment, for U235, U238 and Pu239 in WRES after an approximate resonance treatment in WHEAD for all other resonant isotopes not treated explicitly in WPROCOL/WRES. This treatment is based on equivalence theory with calculated potential scatter cross section applied on slabs with thickness according to mean chord lengths of the fuel kernel, coatings and matrix graphite belonging to the kernel. In this approximation a Dancoff factor has been used which is derived analytically to account for the double heterogeneity of kernels and pebbles [15].

After smearing and condensation of the pebble and kernel materials to one material and in **16 neutron energy groups**, a **pseudo reactor calculation** has been started in as well the axial direction (infinite radius) as in the radial direction (infinite length) in the real reactor dimensions. This has been done for the case without control rods (unrodded) and with control rods inserted (rodded).

Three kinds of control rod banks have been used to answer to the whole benchmark :

- the normal control rods to govern the power level,
- the KLAK system for cold shut-down of the reactor and
- a ‘gas’ control rod bank. This is a control rod bank, which comprises all core channels in the PANTHER model. This bank in the unrodded state returns the nuclear data of the fuel in the unrodded axial meshes of the core, in the rodded state it returns the nuclear data of the Helium void on top of the pebble-bed in the rodded axial intervals. By moving in and out this “gas” rod one can simulate the level of the pebble-bed height in the core. The gas space above the pebble bed has ‘artificial’ and anisotropic diffusion coefficients obtained according to the Gerwin-Scherrer method [10].

After these pseudo reactor calculations, the materials comprised in the reactor were smeared and condensed into **2 neutron groups**, one thermal and one fast neutron group, for the PANTHER full core calculations. The division between the thermal and the fast energy group was at 2.1 eV.

Core Calculations

The HTR-10 has been approximated in the diffusion code PANTHER in the **3-D X-Y-Z** mode with radial 861 square meshes (channels) of 11.48 cm by 11.48 cm and 52 axial intervals of different height but over the core height all 11.10 cm high. The full model comprised 193 core channels, 668 reflector channels, 20 core layers (plus 8 layers to model the bottom cone) 14 bottom reflector layers and 10 top reflector layers. The size of the square mesh has been chosen such that concentric 'rings' can be composed from these meshes in a manner that those rings fit snugly with the main radial dimensions of the reactor structures (Fig. 2). Each mesh in the PANTHER model contains a distinct material with a corresponding set of nuclear data.

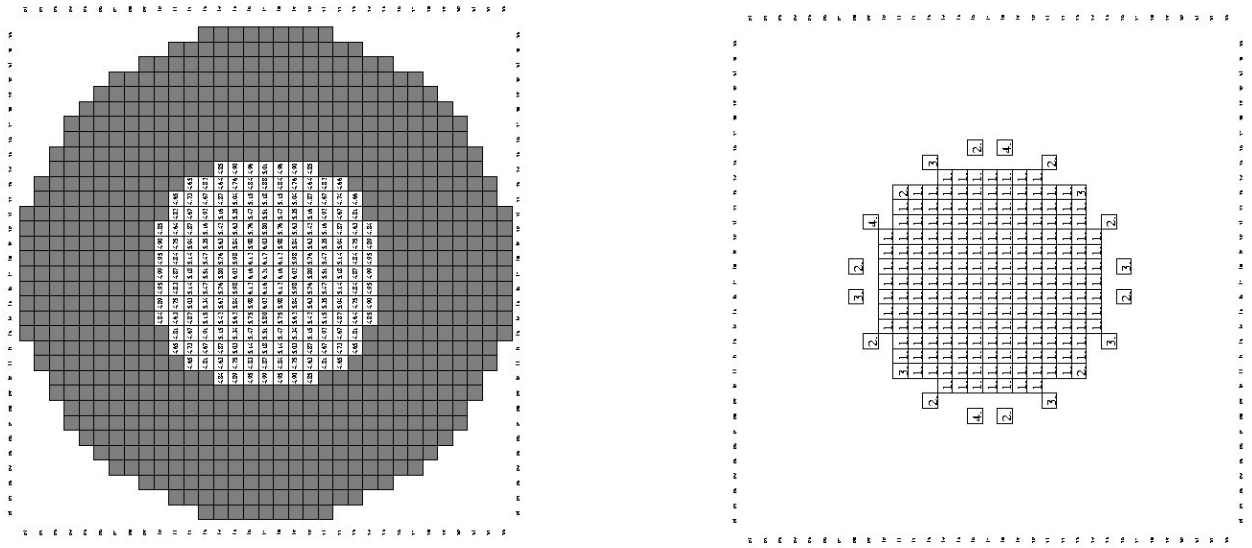


Fig 2: PANTHER: layout of the core and reflector channels and layout of the rod types. The "gas"rod bank, which insertion determines the core height, is indicated by a 1, the normal control rods in the reflector are numbered with a 2, the KLAK system by a 3 and the instrumentation channels are designated by a 4.

II.3 – A MONTE CARLO MODEL FOR THE HTR-10

Codes and calculation scheme

In the following HTR-10 calculations, the French reactor physics code system SAPHYR has been used. SAPHYR gathers several codes developed at CEA like APOLLO2 [16] (transport) based on a database produced with THEMIS/NJOY, CRONOS2 (diffusion-transport), FLICA4 (3D- thermal hydraulics), ..., which are interconnected. Besides, the Monte-Carlo code TRIPOLI4 [17] has also been used throughout the study.

All the HTR-10 problems proposed in reference [1] have been treated considering a calculation scheme based on a Transport – Monte-Carlo method. Figure 3 illustrates the general procedure. The standard **172-group** or **point-wise cross sections** library issued mainly from **JEF-2.2** are used for the calculations.

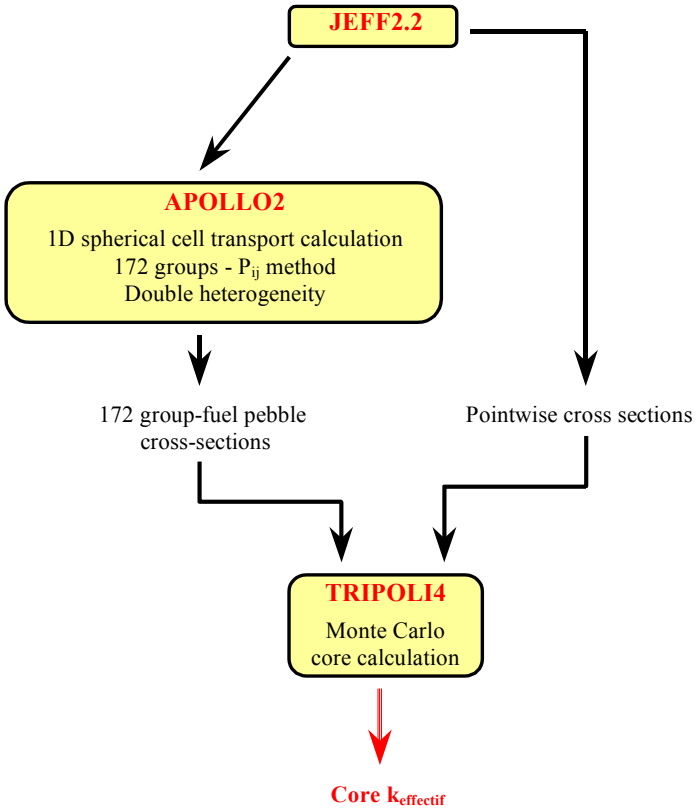


Fig. 3: Description of the Transport – Monte-Carlo calculation scheme

Fuel pebble calculation (APOLLO2)

Knowing that the stochastic geometries calculations (coated fuel particles - CFP - randomly distributed in the inner zone of the fuel pebble) are not available in the Monte-Carlo code TRIPOLI4, a **1D-cell calculation** has been performed as a first step (Figure 4). It takes into account a precise spherical description of the particles with their coatings, which themselves fill the spherical fuel zone of the pebble. The self-shielding of the uranium isotopes is calculated during this calculation step. A collision probability method is used to solve the transport equation with **172 energy groups**.

The **critical buckling search** allows taking into account the neutron leakage by the addition of a homogeneous leakage term in the form of $DB^2\Phi$. The extra region of the spherical cell is representative of the helium coolant plus the moderator pebbles (dummy balls) loaded into the core with the fuel pebbles. The volume of the extra-region has been calculated by considering a volumetric filling fraction of pebbles (fuel and dummy balls) of 0.61 in the core and a 57:43 fuel-to-moderator pebble percent ratio. The obtained multiplication factor and migration area are given in Table 2.

	$k_{infinity}$	M^2 [cm^2]
Enrichment of U235 (weight) 17%	1.71926	1298.0

Table 2: 1D spherical – Transport calculation results in APOLLO2

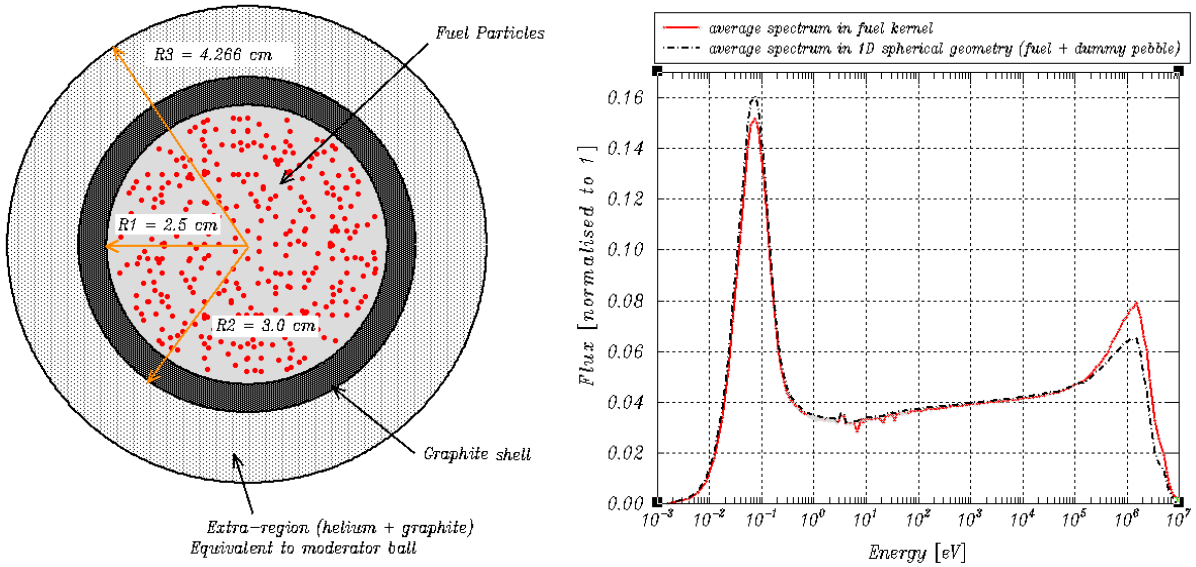


Fig. 4: Spherical Fuel Pebble Model in APOLLO2 and average neutron spectrum in fuel kernel and in fuel pebble surrounded by extra-region

This first stage provides **172-group fuel region averaged cross sections** for Monte-Carlo core calculations (TRIPOLI4). The fuel homogenisation is performed with respect to the pebble bed description in the Monte-Carlo code TRIPOLI4. Indeed, for this benchmark, two types of pebble bed modelling have been analysed. In the first one, the core calculations have been performed considering an homogenised pebble bed (mixture of fuel, moderator pebbles and coolant). In the second one, each pebble has been modelled in the core. For this last case, the fuel pebble calculation in APOLLO2 provided the homogenized cross sections for the fuel pebble inner zone (radius = 2.5 cm).

Therefore, in the core calculations performed by TRIPOLI4, pointwise cross sections are used everywhere in the core except in the fuel pebble region where the multigroup cross sections have been generated with APOLLO2.

Core calculation with TRIPOLI4

Two sets of calculations have been performed on the core geometry. In both cases, all the core components are explicitly modelled (coolant and control rod channels…). The major difference concerns the pebble bed modelling (fuel and dummy balls modelling).

In the first modelling, the pebble bed is represented with an **homogeneous medium** as shown on Figure 5. This homogeneous medium is a mixture of fuel, moderator pebbles and coolant. The equivalent pebble bed cross sections are homogeneous cross sections issued from APOLLO2 calculations (homogenisation of the 1D-spherical geometry). This modelling will be called **"Simplified Pebble Bed modelling"**.

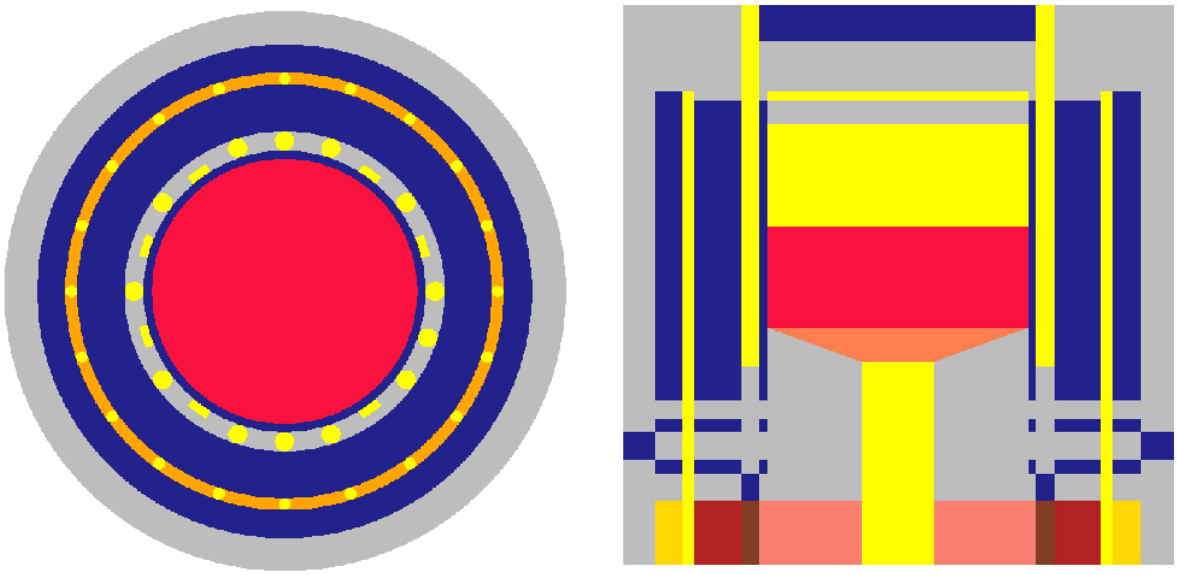


Fig. 5: Core modelling in Monte-Carlo code TRIPOLI4 ("Simplified PB modelling")

In the second set of calculations, all the pebbles (fuel and moderator pebbles) loaded into the core have been described explicitly. Considering a Cubic-Face-Center lattice (theoretical filling fraction of 74%), the fuel and dummy pebbles have been randomly placed at lattice nodes to achieve an average filling fraction of 61% (Figure 6).

Besides, the specification of a 57:43 fuel-to-moderator pebble percent ratio for the initial HTR-10 core loading has been respected. For the calculations, the homogenised cross-sections for the fuel pebble inner-zone have been evaluated by APOLLO2. For the moderator pebble and the fuel pebble outer region, pointwise cross sections are used. This modelling will be called **"Improved Pebble Bed modelling"**.

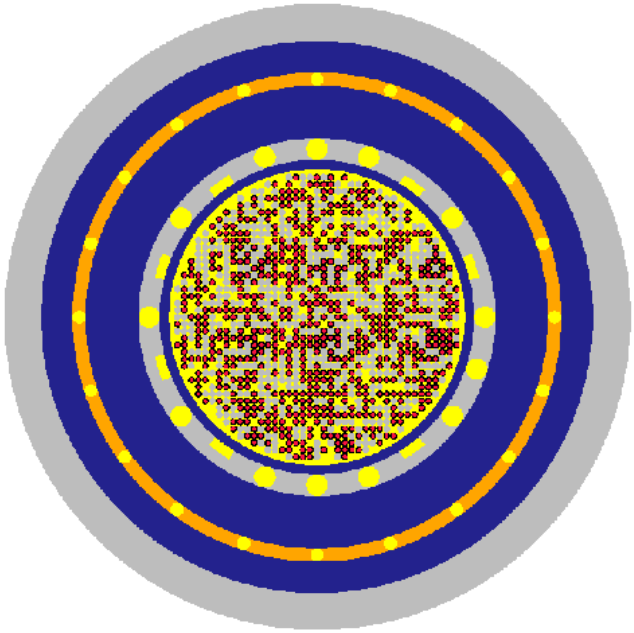


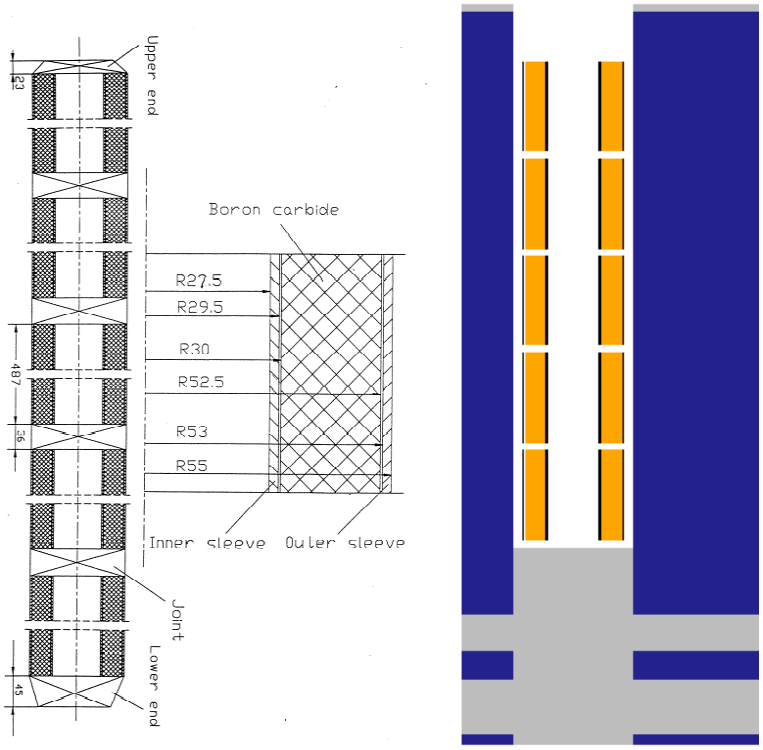
Fig.6: Core modelling in Monte-Carlo code TRIPOLI4 ("Improved PB modelling")

Void channels modelling

In the preliminary results given by CEA, all the calculations were performed in TRIPOLI4 without modelling the void channels (control rods and coolant channels). The consequence was an over-estimation of the core reactivity during the criticality approach. For all the calculations presented in this paper, the void channels have been described exactly in the Monte-Carlo code TRIPOLI4. The cavities modelling impact on reactivity (streaming effect) has also been evaluated for an intermediate core configuration close to the critical core configuration.

Control rods modelling

Ten control rods are designed in the side reflector of HTR-10. Each control rod contains five B4C ring segments, which are housed in the place between an inner and outer sleeve of stainless steel. In TRIPOLI4, the control rods have been modelled explicitly (Figure 7). It means that the annular geometry of the control rod has been modelled as well as the joints between the segments (3,6 cm height).



Control rod design | **Control rod modelling in TRIPOLI4**

Fig. 7: Control Rods modelling in Monte-Carlo code TRIPOLI4

III - HTR-10 CRITICAL CORE LEVEL EVALUATION

III.1 - BENCHMARK RESULTS WITH VSOP

As mentioned above, two series of diffusion calculations are performed for each of the different core loading heights, needed to determine the critical core height:

- one in 2-d geometry considering the empty CR- and KLAK-channels by homogenised atom densities as given in Ref. 1,
- and a second one in 3-d geometry taking into account the neutron streaming in the channels of the CR's and KLAK's and the CR insertion into the side reflector until an axial position of 114.7 cm.

The results of both series, presented in Table 3 and Fig. 8, differ considerably.

H_{core} (cm)	k_{eff}-Values at T= 20^0 C		
	2-d geometry	Δk	3-d geometry
180.12	1.13725	0.0106	1.12665
170.12	1.11870		
160.11	1.09836		
150.10	1.07545	0.0106	1.06483
140.09	1.04919		
130.09	1.01961	0.0087	1.01087
126	1.00639	0.0088	0.99736
120.08	0.98601	0.0089	0.97714
H_{crit} (cm)	**124.2**		**126.8**

Table 3: Results of the Diffusion Calculations for the Original Benchmark Problem B1

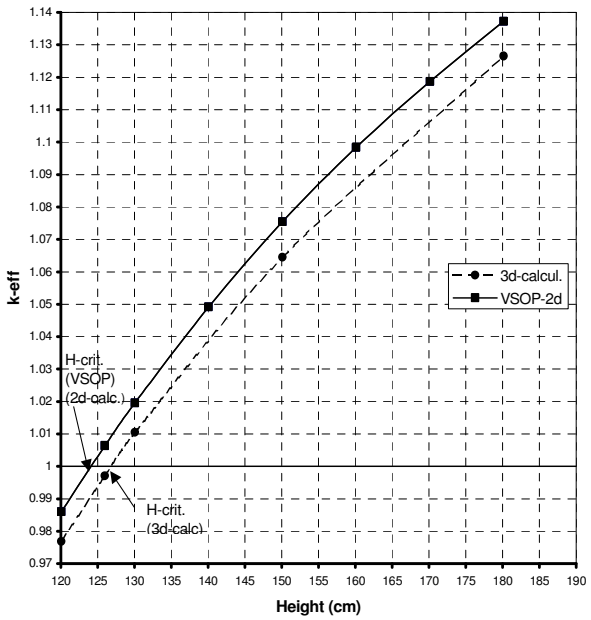


Fig. 8: k_{eff} versus Core Height of the HTR-10 (Original Benchmark Problem B1)

The consideration of the large holes with their increased neutron streaming causes a difference in k_{eff} of about $\Delta k \approx 0.01$. Consequently, the corresponding critical core heights differ about 2.5 cm:

when considering the CR-channels by homogenised atom densities in the 2-d diffusion calculation the critical loading height is: $\mathbf{H_{crit} = 124.2\ cm}$, corresponding to 9716 fuel and 7330 dummy balls, but when considering the CR- and KLAK-boring holes explicitly in the 3-d diffusion calculation the critical height amounts to**:** $\mathbf{H_{crit} = 126.8\ cm}$, corresponding to 9912 fuel and 7477 dummy balls. The latter result is in excellent agreement with the MCNP Monte Carlo result of the INET[18].

Experimental results and revised data

It turns out that the HTR-10 got critical with 16890 pebbles in the core, 9627 fuel pebbles and 7263 moderator pebbles, under air condition and being loaded with another kind of moderator pebbles than defined in the original benchmark description. The core temperature was 15^0 C and the critical core height was: $\mathbf{H_{crit.} = 123.06\ cm}$. Compared to the description of the original benchmark problems three conditions have been changed:

- Density of dummy balls: from **1.73** to **1.84 g/cm^3**
- Boron equivalent of impurities in dummy ball: from **1.3** to **0.125 ppm**
- Core atmosphere at initial criticality: from **Helium** to **Air**

Calculational Results for the Deviated Benchmark Problems with VSOP

When considering these **revised data** in the unit cell calculations and in the whole core diffusion calculations the first criticality is determined to:

- in the case of the 2-d diffusion calculations: $\mathbf{H_{crit.} = 121.0\ cm}$,
- in the case of the 3-d diffusion calculations: $\mathbf{H_{crit.} = 123.3\ cm.}$

The latter result is in excellent agreement with the experiment.

III.2 - BENCHMARK PROBLEM B1 WITH PANTHER

Under **original benchmark conditions**, so with an isothermal reactor temperature of 20 °C, the control rods at 114.7 cm and atmospheric Helium in the coolant spaces, a critical search on the core level (‘gas’rod insertion) has been performed by PANTHER. The critical core level was found to be:

$$\mathbf{H_{crit} = 125.3\ cm,}$$

above the bottom cone filled with moderator balls. With the **revised data** the critical core height becomes equal to **122.1 cm**.

III.3 - BENCHMARK RESULTS WITH TRIPOLI4

Void channels modelling impact (streaming effect)

Some calculations have been performed with the Monte-Carlo code TRIPOLI4 in order to evaluate the impact of the void cavity modelling on the core neutron leakage. The results are presented in Table 4. In the first calculation (case a), the void channels (coolant and control rods channels) are homogenised with the surrounding graphite. In the second calculation, all the cavities have been described exactly in TRIPOLI4 (case b). In the selected configuration (the height of the pebble bed is approximatively 123.0 cm), the streaming effect has been evaluated to **1.48%Δk/k**.

	TRIPOLI4 Case (a) Homogenised cavity	**TRIPOLI4** Case (b) Exact modelling	**Δk [%Δk/k]**
Core $k_{eff} \pm \sigma$ Pebble bed height = 123.06 cm	1.03586 ± 0.00034	1.01960 ± 0.00043	1.48

Table 4: Evaluation of the streaming effect in HTR-10

Analysis of the benchmark data impact

Considering the **revised benchmark data**, some calculations have been performed with TRIPOLI4 in order to evaluate the impact on the core k_{eff}. The results that are presented in Table 5 are related to a core loading height of 123.06 cm. The decrease of the boron equivalent impurities in the dummy balls induces an increase of the core reactivity higher than the capture increase of air atmosphere. The impact on core reactivity amounts to **0.70%.**

	TRIPOLI4 Helium	**TRIPOLI4** Air + impurities + density	**Δk [%Δk/k]**
Core $k_{eff} \pm \sigma$ Pebble bed height = 123.06 cm	1.01960 ± 0.00043	1.02669 ± 0.00048	0.70

Table5: Benchmark data impact

Initial criticality

The aim of the first benchmark problem was to evaluate the amount of loading (given in loading height, starting from the upper surface of the conus region) for the first criticality, under air atmosphere and core temperature of 20 °C, without any control rods being inserted. The core multiplication factor has been calculated for different core loading heights with TRIPOLI4 by using the two types of modelling described below. The results are summarized in Figure 9.

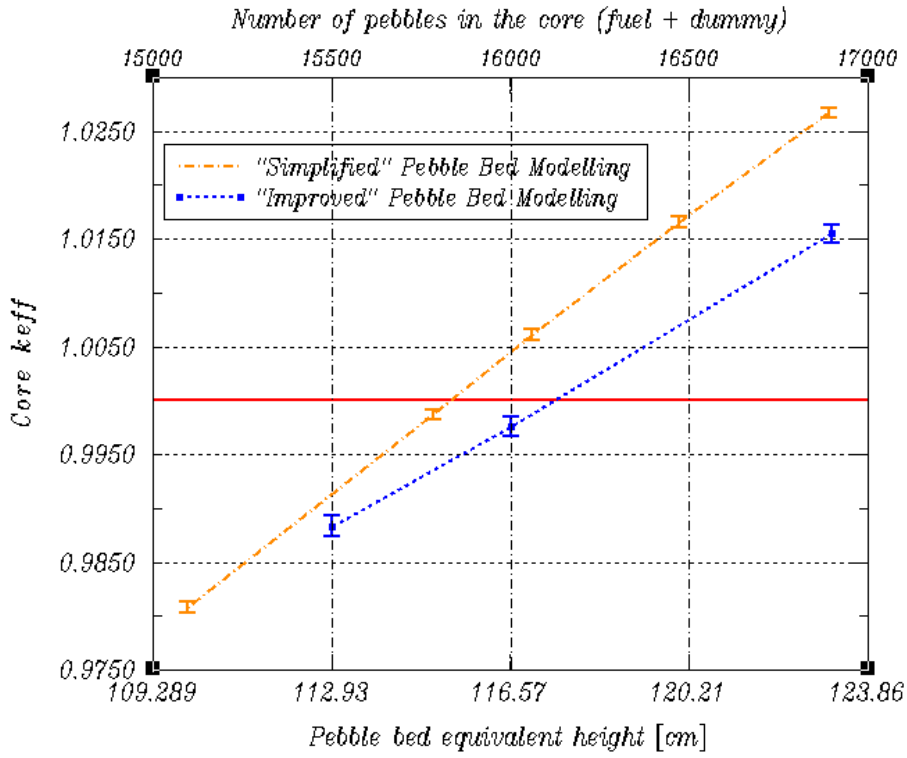


Fig. 9: Calculated k_{eff} vs loading height for HTR-10 initial criticality

On the one hand, it should be noticed that for the core configuration close to the criticality, the core effective multiplication coefficient increases linearly with the number of pebbles loaded. On the other hand, the pebble bed modelling has a strong impact on the core k_{eff}. For example, the discrepancy between the simplified and the improved pebble bed modelling reaches 0.9% Δk/k for a

core loading height of 123.133 cm. This discrepancy is lower when the number of pebbles loaded into the core decreases but in all cases, the core k_{eff} obtained with the improved PB modelling is lower than the one obtained with the simplified PB modelling.

In order to evaluate the impact of the pebble bed modelling on the results, calculations have been performed in a simplified core configuration (infinite medium with white boundary condition). It allowed evaluating the impact of the pebble bed geometrical description without considering core physical effects such as streaming effect and spectrum transient at the interface between core and reflector. The results are gathered in Table 6. The $k_{infinity}$ calculated with both modelling (homogenised pebble bed or randomly distributed pebble bed) are very closed. It shows that the two types of pebble bed modelling are equivalent in infinite medium.

Pebble bed in infinite boundary conditions	**TRIPOLI4 Simplified PB** modelling	**TRIPOLI4 Improved PB** modelling	**Δk [%Δk/k]**
$\mathbf{k_{infinity} \pm \sigma}$	1.76431 ± 0.00051	1.76155 ± 0.00125	0.15

Table 6: Pebble bed modelling impact in infinite medium

After all, the discrepancies observed between the two modelling on the core geometry could be explained by:

- an underestimated streaming effect in the case of an homogenised pebble bed modelling,
- an overestimation of the fission rates in the region close to the reflector in the case of the *"Simplified BP Modelling"*. The pebble bed geometrical modelling seems to have a strong impact in the regions where the neutron spectrum transients are important (at the interface between active core and reflector). Besides, this phenomenon increases when the size of the core decreases (the radius of the active core is 90 cm).

As far as the benchmark problem is concerned, the critical heights calculated with the two models are presented in Table 7. With the **simplified PB modelling**, the criticality is achieved with a loading height of **115.36 cm**. This loading height is equivalent to **15833 pebbles** (fuel and moderator pebbles) loaded into the core. With the **improved PB modelling**, the criticality is achieved with a loading height of **117.37 cm** that corresponds to **16108 pebbles** loaded.

B1	**TRIPOLI4 Simplified PB** modelling	**TRIPOLI4 Improved PB** modelling
Critical height [cm]	**115.36**	**117.37**
Number of pebbles loaded		
Total	**15833**	**16108**
Fuel	~ 9025	~ 9182
Moderator	~ 6808	~ 6926

Table 7: Critical height for the differentHTR-10 TRIPOLI4 modelling

IV - CONCLUSION

The HTR-10's core physics benchmarks have been treated on one hand, with two *Transport-Diffusion* calculation schemes WIMS/PANTHER and VSOP(CITATION) and on the other hand with a *Transport-Monte-Carlo* calculation scheme (APOLLO2 – TRIPOLI4). The calculational results of the original and the deviated benchmark problems are summarised in Table 8.

Critical level in cm	Defined bench.	Revised bench.
Diffusion with VSOP(2D)	124.2 cm	121.0cm
Diffusion with VSOP(3D)	126.8 cm	123.3 cm
Diffusion with PANTHER	125.3cm	122.1 cm
Monte Carlo with TRIPOLI		117.4 cm

Table 8: HTR-10 benchmark results. Critical core level.

It can be noticed that there is a discrepancy of about 1% between the 2-d VSOP and the 3-d VSOP calculations considering the neutron streaming in the channels of the control rods and small absorber balls explicitly. However, it has to be pointed out that **diffusion calculations agree well** with each other and with the experiment.

As far as the Monte Carlo calculations are concerned, two different modelling for the pebble bed geometrical description have been considered. In the "*Simplified PB Modelling*", the pebble bed has been represented by a homogeneous medium. In the "*Improved PB Modelling*", each pebble has been represented in the core (moderator and fuel pebbles). In both case, the double heterogeneity (fuel particles in a graphite matrix) and the self-shielding of the heavy nuclides have been treated by APOLLO2. Due to the streaming effect and the small size of the core (90 cm in radius), the *"Simplified PB modelling"* always overestimates the core $k_{effective}$. As a consequence, the critical pebble bed height calculated with the *"Simplified PB Modelling"* is much more lower than the one calculated with the *"Improved PB Modelling"***.**

Nevertheless, this **preliminary approach for modelling the stochastic geometry** of the HTR-10 in the **Monte Carlo** calculations leads to an **overestimation of the core reactivity**. The reason for this difference has to be investigated furthermore, but recent publications [19, 20] underscored that significant impact could be related to, on one hand the geometric description of the pebble bed in the core cavity and, on the other hand the geometric description of the particules insides the pebbles.

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