Available online at www.sciencedirect.com

SCIENCE DIRECT®

Nuclear Engineering and Design 236 (2006) 615–634

Nuclear Engineering and Design

www.elsevier.com/locate/nucengdes

HTGR reactor physics and fuel cycle studies

J.C. Kuijper[a,*], X. Raepsaet[b], J.B.M. de Haas[a], W. von Lensa[c], U. Ohlig[c], H.-J. Ruetten[c], H. Brockmann[c], F. Damian[b], F. Dolci[b], W. Bernnat[d], J. Oppe[a], J.L. Kloosterman[e], N. Cerullo[f], G. Lomonaco[f], A. Negrini[g], J. Magill[h], R. Seiler[i]

[a] *NRG, Westerduinweg 3, P.O. Box 25, NL-1755 ZG Petten, The Netherlands*
[b] *Commisariat à l'Energie Atomique (CEA), Saclay/Cadarache, France*
[c] *Forschungszentrum Jülich (FZJ), Jülich, Germany*
[d] *Universität Stuttgart, Institut fuer Kernenergetik und Energiesysteme (IKE), Stuttgart, Germany*
[e] *Delft University of Technology (TUD), Delft, The Netherlands*
[f] *Universita di Pisa (UNIPI.DIMNP), Pisa, Italy*
[g] *Ansaldo Energia S.p.A., Genova, Italy*
[h] *European Commision, Institute for Transuranium Elements (JRC-ITU), Karlsruhe, Germany*
[i] *Paul Scherrer Institut (PSI), Villigen, Switzerland*

Received 6 October 2004; received in revised form 21 October 2005; accepted 31 October 2005

Abstract

The high-temperature gas-cooled reactor (HTGR) appears as a good candidate for the next generation of nuclear power plants. In the "HTR-N" project of the European Union Fifth Framework Program, analyses have been performed on a number of conceptual HTGR designs, derived from reference pebble-bed and hexagonal block-type HTGR types. It is shown that several HTGR concepts are quite promising as systems for the incineration of plutonium and possibly minor actinides.

These studies were mainly concerned with the investigation and intercomparison of the plutonium and actinide burning capabilities of a number of HTGR concepts and associated fuel cycles, with emphasis on the use of civil plutonium from spent LWR uranium fuel (first generation Pu) and from spent LWR MOX fuel (second generation Pu). Besides, the "HTR-N" project also included activities concerning the validation of computational tools and the qualification of models. Indeed, it is essential that validated analytical tools are available in the European nuclear community to perform conceptual design studies, industrial calculations (reload calculations and the associated core follow), safety analyses for licensing, etc., for new fuel cycles aiming at plutonium and minor actinide (MA) incineration/transmutation without multi-reprocessing of the discharged fuel.

These validation and qualification activities have been centred round the two HTGR systems currently in operation, viz. the HTR-10 and the HTTR. The re-calculation of the HTTR first criticality with a Monte Carlo neutron transport code now yields acceptable correspondence with experimental data. Also calculations by 3D diffusion theory codes yield acceptable results. Special attention, however, has to be given to the modelling of neutron streaming effects. For the HTR-10 the analyses focused on first criticality, temperature coefficients and control rod worth. Also in these studies a good correspondence between calculation and experiment is observed for the 3D diffusion theory codes.

---

1. Introduction

The European Research and Development (R&D) activities on high-temperature gas-cooled reactors (HTGR) concentrate on HTGR-related key technologies and innovation potentials with the objective to consolidate and advance modular HTGR technology for industrial application in the next decade and to explore new applications like hydrogen production and waste transmutation in the long-term. As the HTGR is a promising concept for the next generation of nuclear power reactors and nuclear process heat, the European nuclear community must have analytical tools capable to perform conceptual design studies, industrial calculations (reload calculations and the associ-

---

[*] Corresponding author at: NRG, Fuels, Actinides & Isotopes Group, Westerduinweg 3, P.O. Box 25, NL-1755 ZG Petten, The Netherlands.
Tel.: +31 224 564506; fax: +31 224 568490.
E-mail address: kuijper@nrg-nl.com (J.C. Kuijper).

0029-5493/$ – see front matter © 2006 Elsevier B.V. All rights reserved.
doi:10.1016/j.nucengdes.2005.10.021

ated core follow), safety analyses for licensing etc. for new fuel cycles aimed at plutonium and minor actinides (MA) transmutation by ultra-high burn-up without multi-reprocessing of the discharged fuel. As part of the European Union Fifth Framework Program the "HTR-N" project and the complementary activities in "HTR-N1" (HTR-N/N1 Contracts, 2000/2001) deal with high-temperature reactor nuclear physics, waste and fuel cycle studies and include 14 partner organisations. For simplicity the projects/contracts mentioned above will be further referred to as "HTR-N" in this article. An overview of the full set of activities within the projects mentioned above can be found in von Lensa et al. (2003).

This article focuses on the qualification, validation and improvement of computational tools and models for the analysis and design of HTGR cores, as well as on the application of these tools and models to analyse several HTGR concepts with different fuel cycles. The first subject is mainly centred around the two HTGR systems currently in operation, viz. the HTR-10 and the HTTR, and is also concerned with the influence of uncertainties in nuclear data and with the applicability of the codes for plutonium-based HTGR fuel at very high burn-up. The second subject concerns the analysis of some more advanced HTGR concepts and applications, e.g. the incineration of plutonium.

2. Contribution to core physics code and data qualification

The HTGR appears to be a promising concept for the next generation of nuclear power reactors. In this context, the European scientific community needs analytical tools to perform conceptual design studies and industrial plant sizing as best estimate or reference calculations. This implies in a near future, besides Monte Carlo codes, to develop methods based on multigroup diffusion and transport codes able to model the HTGR core with its inherent characteristics whichever the fuel cycle and core concept (pebble or prismatic).

A survey on the inherent HTGR characteristics shows that they have a strong impact on the core modelling related issues. Several points can be identified:

- The use of helium gas as coolant in HTGR leads to an important void fraction in the core and to a large neutron streaming effect.
- Due to the use of graphite as moderator, a large part of the neutron spectrum is epi-thermal. Therefore, the classical self-shielding treatment of the resonances amplifies the existing imperfections of the models, which conversely represent well-mastered uncertainties for other reactors.
- Compared to a cladded fuel pin, the fuel dispersion in micro-particles (coated kernels) has the potential to exploit (also taking advantage of the epi-thermal spectrum) very high burn-up. Uncertainties in fuel depletion calculations must be addressed.
- It is often mentioned that the HTGR is highly flexible and can fulfil a wide range of diverse fuel cycles through different physical parameters such as the fuel loadings (particle volume fraction in the graphite), the type of fuel, the burnable poisons, fissile/fertile fuel particle fraction, etc. The resulting core configurations are often strongly heterogeneous with important space dependent variations of the neutron spectrum.
- Finally, the fuel in a form of dispersed particles on the one hand and, the treatment of the pebble-bed core on the other, impose a stochastic approach of the geometry in the Monte Carlo calculations. This may conflict with the requirement of the absolutely unbreakable reference that constitutes the Monte Carlo method.

Core physics calculation tools are available today both for pebble-bed and block-type core. In order to take into account all the characteristics detailed above in the HTGR core physics studies, some calculation schemes have been developed in the past and continue to be improved. However, these codes and methods are validated for the former HTGR concept conditions and for a limited set of fuel types, such as uranium or U/thorium. Additional requirements appear today because the HTGR design evolutions and changes lead today to some new core configurations for which references do not exist, e.g.:

- Annular core geometry;
- Type of fuel (plutonium & minor actinides burning, waste minimization strategy);
- Ultra-high burn-up (e.g. up to more than 700 GWd/t).

Therefore, validation and qualification steps are always needed in order to be able to take into account these additional requirements. So the activities in this part of the "HTR-N" project are aiming at:

- Code validation;
- Qualification and improvement of the methods for modelling the HTGR.

HTTR and HTR-10 are two reactors recently started-up in Japan (1998) and in China (2000). Both reactors are representative of the HTGR concepts that are envisaged today: block-type and pebble-bed reactors. On the basis of these reactors activities have been performed demonstrating the capabilities of the European code systems as well as identifying the calculation method deficiencies or the lack of theoretical models. These activities have been reported in an earlier article (Raepsaet et al., 2003). Besides the activities concerning the HTTR and HTR-10, also re-calculations have been performed on the HTR-PROTEUS experiment at PSI, Villigen, Switzerland. Furthermore, studies have been performed on the influence of basic nuclear (cross-section) data, and on the applicability of the applied HTGR reactor physics code systems for HTGR plutonium-based fuel at very high burn-up. In the following sections the main results of these studies are presented. Further note that nothing is available today for qualifying the codes on plutonium or minor actinides fuels in an HTGR. A first step in his qualification process is presented by the "HTR Plutonium Cell Burnup Benchmark", an activity within the "HTR-N" project, which is described elsewhere (Kuijper et al., 2004).

2.1. HTTR

The high temperature engineering test reactor (HTTR) constructed at the JAERI-site at Oarai in Japan is a graphite moderated and helium gas-cooled reactor with an outlet temperature of 950 °C and a nominal thermal power of 30 MW (IAEA-TECDOC-1382, 2003). It is build to gain and upgrade the technology for high temperature reactors to be built in the future. It became critical at the end of 1998.

The active hexagonal core ($d = 2.30$ m and $h = 2.90$ m) consists of 30 fuel columns and 7 control rod guide columns and is surrounded by a reflector ($d = 4.25$ m and $h = 5.26$ m), which also contains 9 control rod columns and 3 irradiation columns. All is contained in a reactor pressure vessel measuring 13.2 m high and 5.5 m in diameter. The columns consist of stacks of hexagonal blocks 58 cm high and 36 cm wide and are made of graphite. A fuel block has 33 holes, which contain sleeves with fuel, leaving a gap to let pass the helium coolant along the fuel. The fuel is of the coated fuel particle type (TRISO, UO_2) embedded in graphite compacts, which are mounted in the sleeves. Up to 12 different enrichments, ranging from 3.3 to 9.8 w/o in U-235, are in use over the core. To control the over-reactivity burnable poison has been applied in the fuel blocks. The reactor operates in batch mode, so after each cycle part of the fuel will be replaced. The reactor is controlled by means of chains of cans with boron carbide, lowered into the holes in the control rod guide blocks. Helium at 395 °C and 40 bar is blown from the top downwards through the core.

As far as the HTTR is concerned, the activities within the "HTR-N" project have been launched following the great discrepancies observed on the international results of the HTTR-FC benchmark (IAEA-TECDOC-1382, 2003) in which the number of fuel columns to achieve criticality had to be predicted. The fuel columns were gradually loaded one after another from the outer region of the core. In these conditions, a thin annular core configuration was obtained in the course of loading (18 columns), the rest of the core being loaded with dummy fuel blocks. This specific geometry is very close to the one that can be encountered in current HTGR designs proposed today, i.e. GTHTR-300, GT-MHR and PBMR-SA. It represents one of the first opportunities to model such a core geometry and to be able to compare with the experiment. Finally, the excess reactivity for 18, 24, and 30 fuel columns in the core had to be evaluated and formed (or forms) also the subject of the benchmark HTTR-EX (IAEA-TECDOC-1382, 2003). It is noteworthy that an important analysis and interpretation of the former HTTR-FC benchmark results have been done in order to tentatively explain the discrepancies with the experiment. Then, different strong assumptions or physical hypothesis in the HTTR modelling have been identified and their effects quantified by the partners. A very good coherence has been observed between the code systems for quantifying the impact of three common physical effects. These investigations have been extensively reported in an earlier article (Raepsaet et al., 2003).

However, it must be pointed out that the observed discrepancies for the thin core decreased with increasing number of fuel columns in the core. Due to the large experimental error at 30 fuel columns loading (full core), the differences between the calculations and the experiment are within the error interval, whereas at the thin annular core assembly the discrepancies are still significant.

As a concluding remark, one could say that, based on the revised data of the HTTR benchmark, the re-calculation of the first criticality with the TRIPOLI-4 Monte Carlo code allowed to reduce the discrepancy by about a factor of two (from ∼2 to 1% $\Delta k/k$). On the other hand, the results obtained for the fully loaded core configuration is quite acceptable taking into account the uncertainties associated with the experimental values. The remaining deviation for the thin annular core (first criticality) may be explained by the uncertainties of the graphite impurities for which the impact is very important in this core configuration (dummy fuel blocks of pure graphite, with impurities, in the central part of the core).

Finally, the following procedures seem to be necessary for a better approach to the experimental results:

- Take into account the detailed heterogeneity of the burnable poisons- and fuel region in the whole core calculation;
- Use fine-group constants in the whole core diffusion calculation (FZJ) or consider the actual environment of the fuel bocks in the transport cell calculations (NRG) in order to accurately describe the core/reflector coupling;
- Consider the axially heterogeneous distribution of the burnable poisons by 2D cell calculations (FZJ) or by 3D diffusion calculations (CEA and NRG);
- Improve the treatment of the enhanced neutron streaming either by an adaptation of the diffusion constants to Monte Carlo calculations (FZJ) or by a leakage model combined with an analytical model (CEA).

The HTTR of JAERI will be operated at temperatures between 850 and 950 °C and a thermal output of 30 MW. The HTTR calculations presented so far have been performed at room temperature of the core, only. Additional calculations on the HTTR to be carried out in the "HTR-N" project had to take into account temperature feedback effects. Indeed, HTTR critical core configurations at elevated, homogeneously distributed temperature may be available in the near future. They represent a succession of critical states in which temperature feedback effects become more and more important.

However, before modelling these HTTR core configurations at different reactor power levels taking into account these temperature feedback effects can take place, firstly, control rod modelling related problems must be solved and then, a good accordance has to be achieved between the partners on the evaluation of the obtained reactivity worth for an homogeneous temperature variation in the reactor at a hot zero power (isothermal temperature coefficient). Therefore, the following activities have been carried out in the "HTR-N" project:

- Evaluation of the control rod worth for different core configurations at Hot Zero Power condition. Many configurations are available: scram reactivity of all the control rods, and scram

Table 1
Control rod worth in the HTTR (k_{eff} and $\Delta k/k$)

Control rod position	KENO	TRIPOLI4	CITATION	CRONOS	PANTHER	Experimental
178.9 cm	1.0093 ± 0.0006	1.00117 ± 0.00024	1.0031	1.00020	1.0088	Critical
177.6 cm		0.99972 ± 0.00038		0.99840		
Scram of reflector CRs	0.9178 ± 0.0005 (9.88%)	0.92215 ± 0.0004 (8.56%)	0.86535 (15.87%)	0.90245 (10.83%)		12% (±1.2)
Scram of all CRs	0.6809 ± 0.0005 (47.78%)	0.68396 ± 0.0003 (46.32%)	0.67937 (47.50%)	0.63982 (56.31%)	0.7317 (37.5%)	46% (±4.6)

- reactivity of the reflector control rods only (Raepsaet et al., 2004a);
- Calculation of isothermal temperature coefficients on the fully loaded core configuration at Hot Zero Power with fixed control rod position. The temperature coefficients have been determined for temperatures between 280 and 480 K by calculating the effective multiplication factors at 280, 300, 340, 380, 420, 460 and 480 K. The following expression was used to calculate the isothermal temperature reactivity coefficient at the mid-point temperatures (so at 290, 320, 360, 400, 440 and 470 K, respectively) (Raepsaet et al., 2004b):

$$\alpha_{12}(T_{1,2}) = \frac{(k_{T_1} - k_{T_2})/k_{T_1}k_{T_2}}{T_1 - T_2} \quad \text{with} \quad T_{1,2} = \frac{T_1 + T_2}{2}$$

The control rod ("CR") worth calculations at hot zero power conditions (30 fuel columns configuration) have been performed by the codes KENO (at IRI/TUD), TRIPOLI4 and CRONOS2 (at CEA), CITATION (at FZJ), WIMS/CITATION (at UNIPI) and PANTHER (at NRG). The main results are listed in Table 1.

It should be noted that criticality in the experiment was obtained with all the CRs inserted at 178.9 cm. In this configuration the calculated k_{eff} is slightly above 1 for all codes applied, which is favourable from a safety point of view (conservative calculation).

The reflector control rod worths are all in good agreement with the values given by the other international participants of the IAEA benchmark, nevertheless an underestimation of the control rod worth could be underscored compared to the experiment in case a scram of reflector CRs. Especially the 3D PANTHER calculations yield a very low control rod worth in comparison with the other codes. However, this can be explained by neutron streaming in the control rod holes and is to be recalculated by means of an isotropic cross-sections/diffusion coefficients.

Moreover, these results underscore the fact that discrepancies exist between the CR worths, as obtained from diffusion theory and Monte Carlo calculations and experiment, especially in this case where no equivalence factors have been used in order to respect either the flux or the absorption rates between the multigroup transport calculations and the broad group diffusion core calculations. Consequently, it is obvious that further investigations must be carried out in a near future in order to improve the CR modelling related problem especially for rods inserted in the reflector of an HTGR.

Concerning the calculation of the isothermal temperature coefficients it is found that these coefficients range from −15 to −16 pcm/K between 300 and 480 K. In Table 2 a comparison is shown (at $T_{1,2} = 400$ K) between the results obtained by the different code systems used by the project partners. It is noteworthy that the results are in relatively good accordance with the experiment where the values are comprised between −13 and −14 pcm/K at the same temperature. One can conclude that the isothermal temperature coefficients are overestimated by about 10% on average by the different codes.

2.2. HTR-10

The HTR-10 is a high temperature reactor build at the site of the Institute of Nuclear Energy Technology (INET) near Beijing in China (IAEA-TECDOC-1382, 2003). The reactor is of the pebble-bed type and has a thermal power of 10 MW. It reached criticality at the end of 2000 and full power in January 2003 with an outlet temperature of 700 °C. After the first phase of reactor experiments it is the intention to install, in the second phase, a direct gas turbine power generation unit in the primary loop to develop nuclear helium turbine technology.

The pebble-bed in the core cavity ($d = 1.80$ m and $h = 1.97$ m) consists of fuel pebbles, graphite balls ($d = 6$ cm) dispersed with coated fuel particles (TRISO, UO_2). Each pebble contains 5 g of uranium enriched to 17 w/o in U-235.

The core cavity is a void in the reflector with an effective thickness of 1.00 m, all is contained in an RPV 11.15 m high and 4.20 m in diameter (see Fig. 1). In the side reflector are holes, which guide the control rod system. In the bottom reflector is a 50 cm wide chute to carry off the fuel pebbles in a continues reload operation, in which pebbles are taken away at the bottom of the core and if still below the burn-up limit returned at the top of the reactor, else they are discarded and a fresh pebble will be added on the top. The helium coolant with an inlet temperature of 250 °C and pressure of 30 bar flows from the top downwards through the pebble-bed at a rate of 4.3 kg/s.

To achieve first criticality, the fuel discharge tube and the cone of the core bottom has been filled with graphite pebbles, only. Thus, the active core has de facto a cylindrical shape when adding fuel and graphite pebbles from the top except a conical heap-up on the surface of the pebble-bed core.

The core was filled with fuel pebbles from Chinese fabrication because the transport from FZJ to INET of the residual AVR pebbles used in the Swiss PROTEUS experiment was

Table 2
Isothermal temperature coefficients of the HTTR at 400 K

Temperature (K)	KENO (pcm/K)	PANTHER (pcm/K)	CRONOS (pcm/K)	CITATION (pcm/K)
400	−14.7	−15.2	−16.2	−17.2

CR group C, R1 and R2 inserted (178.9 cm).

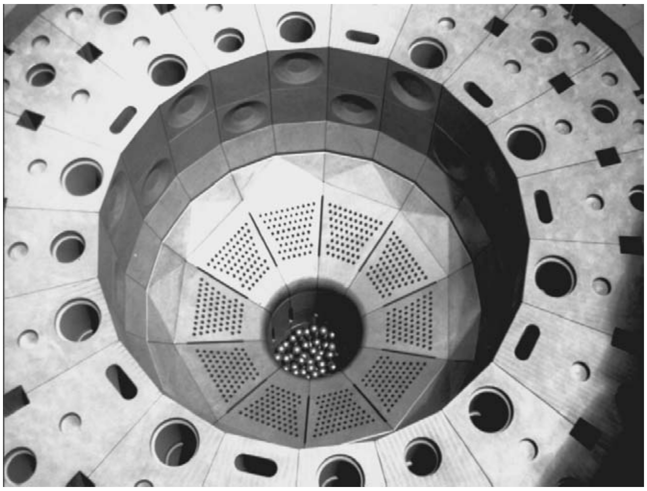

Fig. 1. HTR-10 core view.

delayed due to licensing problems. Nevertheless, the type of fuel and the core geometry in HTR-10 and PROTEUS is rather comparable. PROTEUS was charged with pebbles in different arrangements to investigate the influence of pebble relocations. In contrary to PROTEUS, HTR-10 will also deliver data on higher temperatures to determine the temperature coefficient for LEU fuel.

Despite the small power size of HTR-10, it is a nearly 1:1 scale test for a modular HTGR because the radial dimensions of the reflector blocks are identical to the commercial size. Therefore, HTR-10 can be seen as a representative test for the passive decay heat removal and for verification of codes especially with regard to the effectiveness of the shutdown systems.

Due to the similarities of HTR-10 and PROTEUS, no big deviations were expected for the cold first criticality. The Chinese partners predicated 16 759 pebbles from calculations based originally on European codes. The reactor finally got critical with 16 890 pebbles, which corresponds to an effective core height of 123.1 cm, assuming a pebble packing fraction of 0.61.

The first benchmark task was to evaluate the amount of pebbles, or the level of core loading, at which the reactor became critical. Loading started at the upper level of the bottom cone, which itself was filled with dummy (graphite only) balls.

For the benchmark all control rods were withdrawn. The *original* benchmark was defined before the actual first core criticality and has been revised for the real experimental configuration as the *deviated* benchmark (IAEA-TECDOC-1382). The differences are:

- Core temperature: 20 instead of 15 °C;
- Graphite density dummy balls: 1.73 instead of 1.84 g/cm^3;
- B-impurity in dummy balls: 1.30 instead of. 0.125 ppm;
- Core atmosphere: helium instead of humid air.

The code systems in use by the “HTR-N” partners NRG, FZJ and CEA were PANTHER, VSOP (CITATION) and TRIPOLI, respectively. PANTHER and VSOP are both codes based on 3D diffusion theory whereas TRIPOLI is a 3D Monte Carlo code. At INET also a version of VSOP has been used. Results for the different institutes are summarized in Table 3 and show a good agreement with the experimental value.

Table 3
Critical core level of the HTR-10

Code/model	Core level (cm)	
	Original benchmark	Deviated benchmark
Diffusion with VSOP (2D)	124.2	121.0
Diffusion with VSOP (3D)	126.8	123.3
Diffusion with PANTHER	125.3	122.1
Monte Carlo with TRIPOLI (cubic)	–	117.4
Monte Carlo with TRIPOLI (hex)	–	122.7
Experimental	–	123.1

The difference between the 2D and 3D VSOP calculations is that in the latter the control rod guide holes and shutdown KLAK holes has been considered explicitly. This gives an impression of the neutron streaming in those holes.

And the difference between the cubic and hexagonal TRIPOLI calculations is an adapted face-centred-cubic lattice (74%) of the pebbles in the core or a column hexagonal point-on-point arrangement of the pebbles (60.46% of packing fraction). The latter turns out to be more representative of the pebbles random distribution and highlights the influence of the pebble-bed description in the core model. Results are tabulated in Table 3 and show a good agreement with the experimental value. More detailed information on Monte Carlo modelling of the HTR-10 first criticality can be found in Chang et al. (2004).

Further calculational benchmark exercises concerned the isothermal temperature coefficient and the control rod worth. It should be noted that for the first item no experimental data have been made available yet, whereas only limited experimental information (at somewhat deviating state parameters) is available on the second item.

The benchmark calculations were to be performed with the entire reactor at the same (isothermal) temperatures of 20, 120 and 250 °C and a core height of 180 cm (full core) (IAEA-TECDOC-1382, 2003). By calculating the corresponding multiplication factors, the temperature coefficients could be calculated in the same way as was done for the HTTR (see Section 2.1). Results are listed in Table 4.

As no measured values are available, a comparison can be made with the values for the isothermal temperature coefficient as obtained by INET, the owner of the HTR-10. INET values for the original benchmark were obtained by means of VSOP. The

Table 4
HTR-10 isothermal temperature coefficients

T (°C)	INET	CEA	FZJ	NRG
k_{eff} for the different temperatures				
20	1.119747	1.14737	1.12665	1.11759
120	1.110435	–	1.11331	1.10846
250	1.095961	–	1.09588	1.09629
Isothermal temperature coefficient				
20–120	−7.49E−5	–	−1.06E−4	−7.37E−5
120–250	−9.15E−5	–	−1.10E−4	−7.70E−5

NRG values agree rather well with the values of INET, especially at lower temperature. The FZJ values agree rather well at higher temperatures, but at low temperatures it is rather high compared with the other participants. No reason could be given yet. The differences between the values at low and high temperature are about the same for FZJ and NRG and are small compared to the difference in the values of INET, leading to a stronger decline of coefficient with temperature.

Concerning the calculation of the control rod worth two different benchmark situations were proposed and performed:

- With the core level at 180.0 cm (full core) and a uniform reactor temperature of 20 °C; the insertion of all control rods ran from 119.2 to 394.2 cm;
- And with the core level at 126 cm (critical level) and a uniform reactor at of 20 °C; the insertion of all control rods ran from119.2 to 394.2 cm as well.

Also two different core situations with the core level at 126 and at 180.0 cm and with the reactor at 20 °C, a series of calculations has been done for different insertion fractions of only one control rod.

As the experimental condition of the reactor differed from the stated original benchmark condition, INET, CEA and FZJ performed these benchmark items for the experimental situation (the so-called *deviated* benchmark (IAEA-TECDOC-1382, 2003)). A summary of the obtained results is presented in Table 5.

As there are no experimental values known yet for the 10 control rod worths, a comparison can be made with the MCNP calculations done by INET, the owner of the HTR-10. If these MCNP calculations are taken as reference calculations the values of FZJ do compare very well with INET. The values of NRG are too low which can be attributed to the influence of neutron streaming in the rather wide holes which contain the control rods and the KLAK shut down system. FZJ values are corrected for this effect. In the CEA calculations, the neutron streaming effect cannot be invoked to explain the discrepancies. The CR worth has been calculated from the Cubic core model already used for the evaluation of the first criticality (shown in Table 3). That core model led to a reactivity overestimation of 1.5% due to the specific description of the arrangement of the pebbles in the core cavity. It is then obvious that the neutron spectrum near at the core/reflector interface and in the reflector itself is influenced by the pebble-bed model and can be far from the actual one, leading to differences in the CR worth estimation.

For the single control rod worth a value of 1.437% was measured for a core height of 123.86 cm and the deviated benchmark conditions. INET calculated the single rod worth as 1.448%, which is in very good agreement with the experiment, which gives confidence to these reference calculations. As for the scram reactivity the same applies for the single rod worth as concerns the neutron streaming to arrive at a lower rod worth for NRG.

2.3. HTR-PROTEUS

Benchmark calculations and cold critical experiments for fresh LEU-HTR pebbles were done at PSI in the critical facility PROTEUS (see Fig. 2) in the time period 1992–1997. The main goal of the program was to provide integral data for small and medium-sized LEU-HTR-systems related to:

- Reaction rate distributions and criticality;
- Worth of absorber rods which are located in the side reflector;
- The effects of accidental water ingress;
- Neutron streaming on the neutron balance.

The experimental results have been analyzed mainly with the MICROX-2/TWODANT calculational route. However, some shortcomings especially in calculating the reaction rate traverses have been identified.

In the framework of the "HTR-N" project, new calculations with the Monte Carlo code MCNP4B have been performed with respect to criticality and reaction rate distributions for two reference core configurations (de Haas et al., 2004; Seiler, 2004). Monte Carlo calculations with MCNP have already been performed during the HTR-PROTEUS program, but with poor

Table 5
Control rod worth in the HTR-10

Number of rods	INET	CEA	FZJ	NRG
Rod worth for the full core (180 cm) and original benchmark (%Δ*k*/*k*)				
10	16.56 ± 0.31	13.44 ± 0.26	16.60	11.86
1	1.413 ± 0.265	1.31 ± 0.29	1.563	–
Rod worth for the critical core (126 cm) and original benchmark (%Δ*k*/*k*)				
10	19.36 ± 0.44	13.80 ± 0.20	20.55	13.67
1	1.793 ± 0.371	0.28 ± 0.10	1.969	–

Number of rods	INET	FZJ
Rod worth for the full core (180 cm) and deviated benchmark (%Δ*k*/*k*)		
10	15.31	15.73
1	1.343	1.48
Rod worth for the critical core (126 cm) and deviated benchmark (%Δ*k*/*k*)		
10	18.28	19.31
1	1.571	1.86

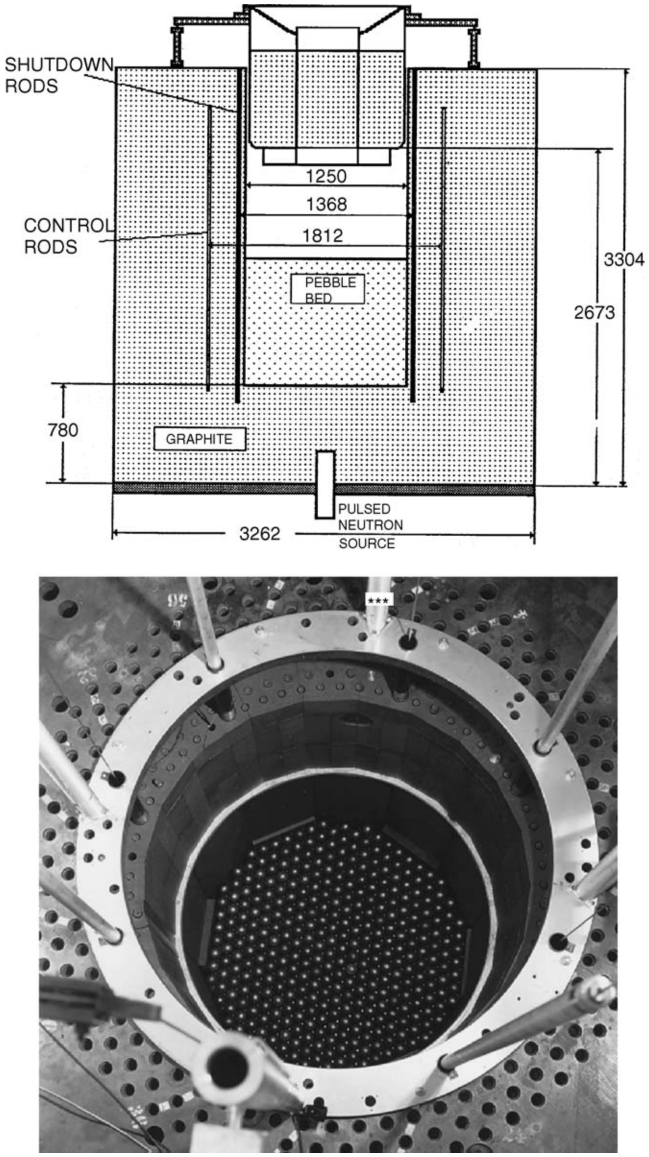

Fig. 2. Vertical cross-section of the HTR-PROTEUS configuration (dimensions in mm, top) and top view of the pebble-bed (bottom).

statistics in the low flux regions (lower and upper reflector). In the meantime, the measurements to estimate the absorption cross-section of the reflector-graphite were re-analyzed resulting in an increase of the graphite absorption cross-section from 4.09 to 4.47 mbarn. With new MCNP4B calculations, the statistical error could be reduced by a factor of two by calculating 5 million histories.

The cavity was partially filled with mixtures of moderator (pure graphite) and fuel (containing 16.7% enriched UO_2 TRISO-coated particles) pebbles, loaded either in deterministic or random arrangements to form the reactor core. Both pebble types had an outer diameter of 6.0 cm and a fuel region with a diameter of 4.7 cm. The “Arbeitsgemeinschaft Versuchsreaktor” (AVR) in Germany supplied the pebbles. Each fuel pebble contained about 1 g of ^{235}U in ∼9400 particles.

The presently reported results were calculated for the HTR-PROTEUS Cores 5 and 7. Core 5 has a rhombohedral pebble-lattice geometry with a fuel-to-moderator (F/M) pebble ratio of 2:1, corresponding to a C-to-^{235}U ratio of about 5670. This so-called column hexagonal point on point (CHPOP) pebble-bed arrangement had a filling factor of 0.6046, which is only slightly lower than a stochastic arrangement with a filling factor of 0.62. In order to improve the homogeneity of the core region, an ABCABC . . . loading scheme was adopted in which the layer pattern repeats every fourth layer. The packing frequency ABC was repeated up to layer 22. Each layer consists of 241 fuel pebbles and 120 moderator pebbles, however the position of the pebbles differed from layer to layer [9, 10]. The arrangement of the 23rd layer (top layer) was changed because too few fuel pebbles remained to form a complete layer. Therefore the remaining 138 fuel pebbles were arranged in a 2:1 lattice in the centre of this layer, with the surrounding area being filled with moderator pebbles.

Core 7 was similar to Core 5 but the vertical channels contained polyethylene rods (total of 654 rods) in order to simulate accidental moderation increase in terms of higher hydrogen density. The pebble-bed core height was reduced from 23 layers to 18 layers to yield a critical configuration. The pebble-layers of Core 7 were identical to those of Core 5 up to layer 17, and the top layer 18 similar to the top layer 23 of Core 5.

The deterministic models for the calculation of Cores 5 and 7 were based on use of the 2D transport-theory code TWODANT. The necessary macroscopic cross-sections for the doubly heterogeneous pebble-bed-lattices were derived using the MICROX-2 cell code in conjunction with its JEF-1 based data library. Corrections for inter-pebble streaming effects were made, in each case.

The Monte Carlo code MCNP4B was employed along with its ENDF/B-V based continuous-energy cross-section library. For Cores 5 and 7 a very detailed model was developed with the 12-sided polygon, absorber rod channels and the top reflector modelled in detail. Thereby, heterogeneity effects in the core region (particles/matrix/shell for the fuel pebble, moderator/fuel pebble arrangement for the lattice, and polyethylene rods in the case of Core 7) were all treated explicitly. But certain detailed aspects of the HTR-PROTEUS configurations have been omitted in order to facilitate a more straightforward modelling of the experiments. The most important single item, in this context, is represented by the partly inserted control rods which have not been described and had an experimentally determined worth (inserted) of about 84 and 48 cents in Cores 5 and 7, respectively. Considering the other detailed features (e.g. the instrumentation channels, etc.), which have been omitted, one has estimated that corrections of 1109 and 670 pcm (TWODANT) and 834 and 505 pcm (MCNP) need to be applied for the two configurations. The “experimental” k_{eff} values to be used as reference for the presently described TWODANT/MCNP-models for Cores 5 and 7 (without any shutdown rod inserted) are thus 1.0111/1.0083 and 1.0067/1.0051, respectively.

Tables 6 and 7 show the comparison of calculated and measured values for the system reactivity k_{eff}. As mentioned before, the reactivity was calculated for a system without partially inserted control rods. Only the absorber rod channels and the air gaps of the driver-fuel channels in the side reflector have

Table 6
Calculated and Measured k_{eff} values for cores 5 and 7 (HTR-PROTEUS)

Core	Experimental	TWODANT	MCNP4B
Measured and calculated k_{eff} values			
5	1.0111/1.0083 ± 0.0005	0.998	0.99502 ± 0.00044
7	1.0067/1.0051 ± 0.0005	1.010	1.00268 ± 0.00033

been modelled for the MCNP calculations, the experimental k_{eff} values were corrected accordingly. It can be seen that the calculations agree well with the experiments in Core 7 but underestimate k_{eff} of Core 5. This could be an indication that the polyethylene rods can be smeared into the inter-pebble void, but that streaming corrections, which have to be applied for the core region, are not treated correct in the deterministic model.

A comparison of calculated with experimental axial reaction rate distributions shows a good agreement with MCNP4B (see Fig. 3) and a satisfactory agreement with TWODANT, especially in the low-flux regions (lower and upper axial reflectors). The distributions were normalised to unity in the centre of the pebble-bed.

2.4. Conclusions—core physics codes and data qualification

From the calculations and re-calculations of the HTTR and HTR-10 (benchmark) configurations it can be learned that Monte Carlo codes are sensitive to the way the in principle stochastic "lattice" of the pebbles in the core are modelled in a more regular lattice (HTR-10). The adaptation of a highly compact cubic lattice (74%) by randomly removing pebbles to achieve the actual packing fraction (61%) can lead to high inter-pebble streaming in the pebble-bed (pseudo cavity) or local moderating ratio different than in the experiment especially at the core/reflector interface, leading to differences in core reactivities. For the diffusion codes care has to be taken on how big holes, like control rod guide holes, are modelled. These holes also give rise to pronounced neutron streaming inside and affect the control rod worth (HTTR and HTR-10). Also care has to be taken on how the core heterogeneities have to be modelled. This is the case for the burnable poison rods in the HTTR, where axial detail has to be taken into account. There is a tendency that temperature coefficients and single control rod worths are slightly overestimated by diffusion/transport codes compared to Monte Carlo or measurements (HTTR and HTR-10). All together do streaming effects in voids play an important role in graphite reflectors so further investigations have to be performed in the future.

Table 7
Comparison of calculated and measured k_{eff} values for cores 5 and 7 (HTR-PROTEUS)

Core	Experimental	TWODANT	MCNP4B
Measured and calculated/experimental k_{eff} values			
5	1.0111/1.0083 ± 0.0005	0.987	0.987 ± 0.00044
7	1.0067/1.0051 ± 0.0005	1.003	0.998 ± 0.00033

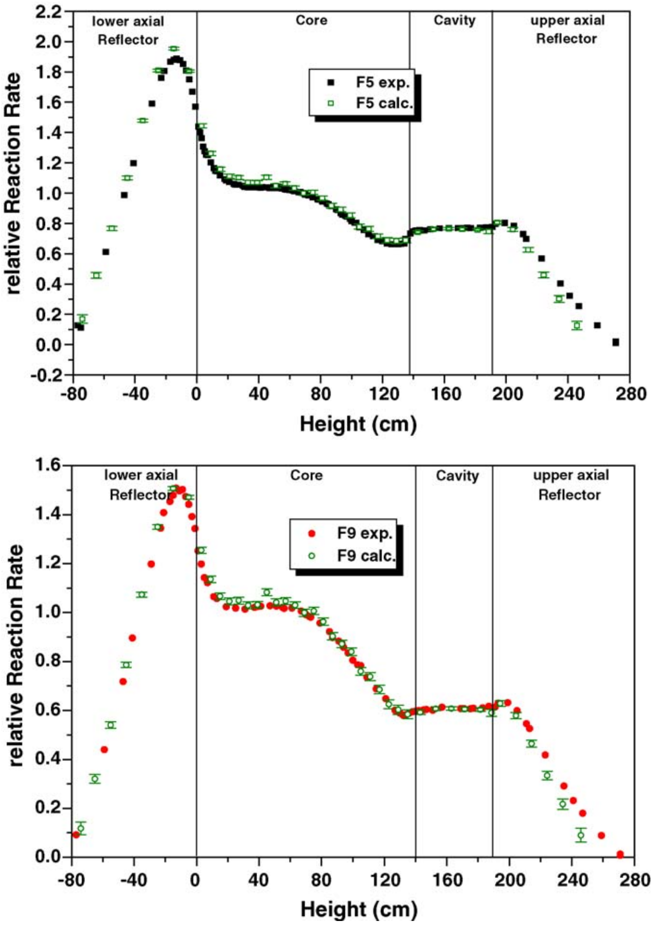

Fig. 3. Experimental and calculated (MCNP4B) axial reaction rate traverses of fission in ^{235}U (F5) and ^{239}Pu (F9) in Core 5 (closed points denote experimental data, open points denote calculations).

Deterministic and stochastic calculations have been performed with MICROX/TWODANT and MCNP4B for an HTR-PROTEUS core configuration with (Core 7) and without (Core 5) simulated water ingress. The system reactivity (k_{eff}) could be well calculated for Core 7, but was underestimated for Core 5. This can be an indication that water ingress can be well simulated with (heterogeneous) polyethylene rods. The axial reaction rates calculated with MCNP4B are in good agreement with the measurements especially in the lower reflector. The calculations with TWODANT were less satisfactory, indicating the need for an exact modelling of the core/reflector region at the bottom of the pebble-bed.

Two other activities within the "HTR-N" project concern a first step towards the qualification of HTGR core physics codes for the use of Pu-based fuel at high burn-up ("HTR Plutonium Cell Burnup Benchmark"), and the investigation of the influence of uncertainties in nuclear data, respectively. Results form these activities have been reported elsewhere (Kuijper et al., 2004; Oppe and Kuijper, 2004; Dolci, 2003; Bernnat et al., 2003a,b; Difilippo et al., 2002; Young and Huffman, 1964), but for completeness their main conclusions are listed below.

Concerning the "HTR Plutonium Cell Burnup Benchmark", generally a good agreement, up to a burn-up of approximately

600 MWd/kgHM, is found between the results of three out of four participants, representing four out of five code systems. The reasons for deviations of the single partner have been identified, and meanwhile corrected. The remaining differences in results between the three participants can be largely attributed to differences in modelling of reaction paths in the different code systems, which are amplified by the unusually high flux levels typical to this particular benchmark exercise (Kuijper et al., 2004).

Investigations on the influence of nuclear data uncertainties on results of calculational reactor physics analyses lead to the conclusion that, for the calculation of criticality parameters, a good agreement is obtained between JEF-2.2 and JENDL-3.2 based calculations for a broad range of moderation ratios, whereas ENDF/B-VI (release 5) based calculations lead to an underestimation for low moderation ratios. From sensitivity and uncertainty calculations both at low and high burn-up of the fuel it can be concluded that the uncertainty in calculated reactivity is about 0.6% for weapon grade (WG) Pu fuel. A need for actualized covariance data was found since there rather few reliable processed data available for thermal systems. Observed large differences in scattering law and frequency distribution data for the scattering of thermal neutrons have no significant influence on calculated neutron spectra and integral parameters (Dolci, 2003; Bernnat et al., 2003a,b; Difilippo et al., 2002; Young and Huffman, 1964).

3. Several HTGR concepts with different fuel cycles

An important part of the activities within the “HTR-N” project was dedicated to the analyses, by the code systems also used for the analyses presented in Section 2, of several HTGR concepts. These studies were mainly concerned with the investigation and intercomparison of the plutonium and actinide burning capabilities of a number of HTGR concepts and associated fuel cycles, with emphasis on the use of civil plutonium from spent LWR uranium fuel (first generation Pu) and from spent LWR MOX fuel (second generation Pu). Two main types of HTGRs under investigation are the hexagonal block-type reactor with batch-wise reloading and the continuously loaded pebble-bed reactor. In conjunction with the reactor types also a number of different fuel types (e.g. Pu-based and Pu/Th-based) and associated fuel cycles have been investigated. In addition, studies have been conducted on the optimization of the power size of pebble-bed HTGRs (employing an annular core geometry), the optimization of burnable poison particle designs (mainly required for batch-loaded HTGRs) and the more exotic concept of the spectrum transmitter.

3.1. Reference core and fuel designs

For a meaningful assessment and intercomparison of HTGR concepts a common basis has been defined and agreed upon by the partners in the “HTR-N” project (Kuijper et al., 2002). This common basis includes the definition of a reference pebble-bed reactor (“flat bottom” “HTR-MODUL”) with continuous re-loading (“MEDUL”) of fuel elements (Reutler and Lohnert, 1983), the definition of a reference hexagonal block-type reactor (“GT-MHR”) (General Atomics, 1996), the definition of a reference TRISO coated particle (kernel diameter, composition and thickness of coatings), the definition of reference first and second generation Pu-composition and the definition of a set of transmutation (plutonium and minor actinide reduction) and safety related common output parameters to be calculated for each of the concepts and cases under study by the partners (Kuijper et al., 2002).

A common feature for the pebble-bed and block-type HTGR designs is the use of coated particle (CP) fuel. Main parameters of the PuO_2-loaded CP fuel are given in Table 8. Detailed information on other CP fuel types, which have been investigated in these studies, can be found in Kuijper et al. (2002). The assumed initial isotopic composition of first and second generation Pu is presented in Table 9.

The main dimensions and other parameters of the reference continuous reload pebble-bed reactor are presented in Fig. 4 and Table 10. This reference reactor is a simplified version of the “HTR-MODUL” design (Reutler and Lohnert, 1983). For example the conically shaped defuelling chute is not modelled and consequently a uniform vertical flow velocity distribution

Table 8
General parameters of PuO_2-containing coated particle fuel

Kernel diameter of coated particle	0.240 mm
Kernel material (fuel)	PuO_2
Density of kernel material	10.4 g/cm^3
Coating materials (inner to outer)	C/C/SiC/C
Coating thickness (inner to outer)	0.095/0.040/0.035/0.040 mm
Density of coating material (inner to outer)	1.05/1.90/3.18/1.90 g/cm^3

Table 9
Isotopic composition of first and second generation plutonium in the HTR Pu cell burn-up benchmark (wt.%)

Isotope	First generation (original) “A”	First generation (alternative) “B”	Second generation “C”
^{238}Pu	1	2.59	4.9
^{239}Pu	62	53.85	26.9
^{240}Pu	24	23.66	34.3
^{241}Pu	8	13.13	15.3
^{242}Pu	5	6.78	18.6

Table 10
General parameters of the HTR-MODUL-based reference reactor

Nominal power	200 MWth
Power density in the core	3.0 MW/m^3
Thermal efficiency	40% (assumed in FZJ calculations)
Core height	9.43 m
Core diameter	3.0 m
Number of pebbles	5394 per m^3
He core inlet temperature	250 °C
He core outlet temperature	700 °C
System pressure	60 bar
He mass flow rate	85.55 kg/s
Basic graphite density (in reflectors)	1.80 g/cm^3
Pebble diameter	6.0 cm
Diameter of fuel zone (matrix/coated particles)	5.0 cm
Graphite density (matrix and outer shell)	1.75 g/cm^3

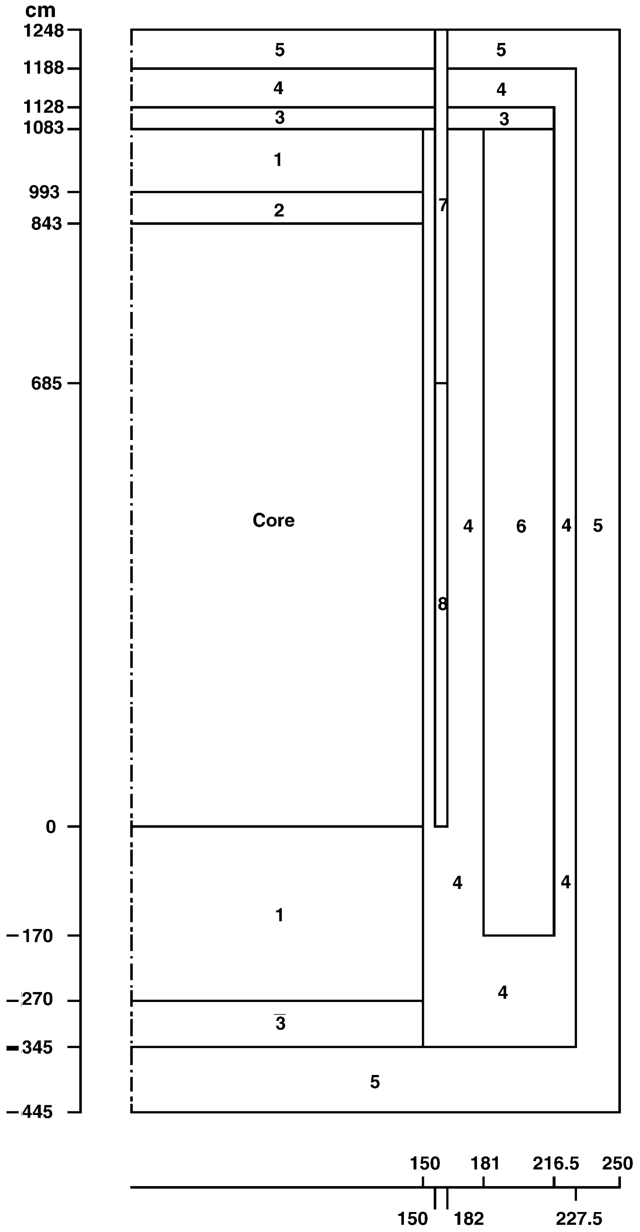

Fig. 4. Main dimensions and material regions of the calculational model of the HTR-MODUL. Dimensions are in centimetre. The core is a random stacking of the well-known 6 cm fuel balls ('pebbles'). The conical defuelling chute below the core is not modelled. Other (more or less homogenised) material regions in the model are: (1) reflector (graphite); (2) void (so He gas); (3) homogenised void and graphite; (4) reflector (graphite); (5) carbon bricks; (6) reflector with coolant channels; (7) reflector with control rod channels; (8) reflector (graphite).

of the pebbles over the entire radius of the core is assumed. In fact there is a rather uniform velocity distribution from the top of the pebble-bed to the bottom, except for the conical region but this is an area with a relative low power density, thus justifying the "flat bottom" option as a simplification. As in the original HTR-MODUL design, in our analyses the "MEDUL" (German: "MEhrfach DUrchLauf"–multi-pass) fuelling strategy was assumed as well.

Table 11
General parameters of the GT-MHR-based reference reactor

Power	600 MWth
Thermal efficiency	48% (assumed in CEA calculations)
Loading factor	0.85 (assumed in CEA calculations)
Power density in active zone	6.6 MW/m^3
Inlet/outlet temperature	490/850 °C
Height of active zone	8 m
Equivalent diameter of active zone	2.96/4.84 m
Height of axial reflectors	1.3 m
Number of columns in the annular core	102
Standard fuel elements	720 (10 per column)
Control fuel elements	300 (10 per column)
Control rods in core	12 (start-up) and 18 (shutdown)
Control rods in reflector	36 (core operation)
Type of fuel loaded into core	PuO$_x$
Fuel composition	Only one type of particle

The investigation of fuel cycle studies for block-type HTGR cores was performed on the basis of the gas turbine modular helium-cooled reactor (GT-MHR) concept. The main features of the GT-MHR core (General Atomics, 1996) are indicated in Table 11 and Fig. 5. The core of the GT-MHR consists of 102 columns of fuel comprising 72 standard element columns and 30 control element columns. The reflector and fuel columns consist of stacks of prismatic blocks with a height of 80 and 36.0 cm across opposite sides. The core of the GT-MHR also includes a reflector at the top and the bottom with a height of 130 cm.

3.2. Continuous reload pebble-bed type HTGR

Starting from the reference pebble-bed reactor, NRG and FZJ investigated on the feasibility of burning of first and second generation plutonium in such a reactor. By 3D reactor calculations, combining neutronics and pebble-bed HTGR core thermal-hydraulics, several loading schemes, including some containing mixtures of fuel pebbles containing different CP fuel types, were investigated, focusing on Pu incineration capabilities and parameters concerning the safety of a reactor loaded as such (e.g. maximum power densities and temperature reactivity coefficients). The investigations by IKE concerned the optimization of the power size of the reactor, considering a number of different core layout designs.

3.2.1. Plutonium incineration capability

The NRG analyses on the Pu-loaded HTR-MODUL presented in this report were performed by means of the WIMS/PANTHERMIX code system. Since a number of years NRG is developing the HTGR reactor physics code system WIMS/PANTHERMIX, based on the well-known lattice code WIMS (versions 7 and 8), the 3D steady-state and transient core physics code PANTHER and the 2D R-Z HTGR thermal-hydraulics code THERMIX-DIREKT. At NRG the PANTHER code has been interfaced with THERMIX-DIREKT to enable consistent core follow and transient analyses on both pebble-bed and block-type HTGR systems. Further information can be found in de Haas and Kuijper (2005).

NRG has implemented the reference pebble-bed reactor ("HTR-MODUL") in their PANTHERMIX code system and has performed some initial studies on the OTTO (once through then out) loading scheme with UO$_2$ fuel (7.8% enriched) and first generation (pure) PuO$_2$, with 7 g per pebble of initial heavy

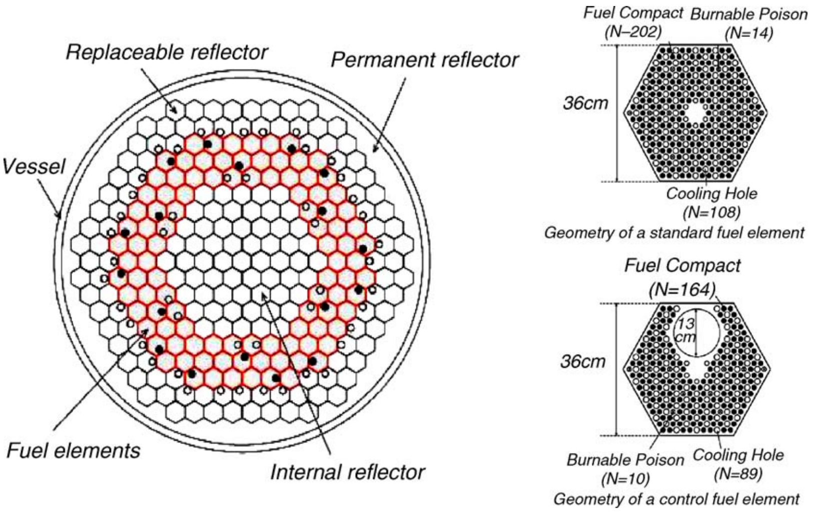

Fig. 5. Geometry of the 600 MWth GT-MHR core, with details of a standard fuel element and a control element.

metal mass. It was concluded that in the equilibrium state, after 2000 days of operation, 415 (fresh) pebbles are needed per day to maintain criticality. In this state the maximum power density in the core is 11.84 MW/m^3, the maximum burn-up in the core is 77.5 MWd/kg and the maximum (fuel) temperature in the core is 1072.5 K.

Further studies have been executed concerning the use of first and second generation (pure) Pu in an HTR-MODUL in continuous recycling mode, focussing on the influence of the heavy metal mass per pebble and the selected discharge burn-up on the values of the common parameters agreed upon. The pebble circulation rate was kept constant at 3 kg (initial) Pu per day, throughout all NRG calculations. Some results from these studies are shown in Table 12. In this table the calculated "case" is described by the coding "*Pu-x-mass-mod*", in which "*x*" indicates the Pu type (1—first generation, 2—second generation), "*mass*" indicates the amount of Pu per fresh (unit: grams) and "*mod*" the number of admixed moderator pebbles (pure graphite) per fuel pebble (either 1 or 0). For first generation Pu a further distinction is made between composition "A" (70% fissile) and "B" (67% fissile) (see Table 9). Further information can be found in de Haas and Kuijper (2005).

From these investigations it can be concluded that the reactor can be made critical at *beginning of life* with all investigated fuel types containing first generation Pu. However, only the fuel pebbles containing 2 g Pu, without admixed moderator pebbles, lead to a sufficiently negative temperature coefficient in the equilibrium situation. For the first generation Pu cases the average burn-up of the permanently discharged pebbles is about 750 MWd/kg. An appreciable reduction of about 85% of the original plutonium can be achieved. Note that quite similar results are found for the two types of first generation Pu, which indicates a relative insensitivity of the results to the *exact* plutonium vector.

For second generation plutonium the situation is somewhat less favourable. The burn-up of the permanently discharged pebbles has to be reduced to about 440 MWd/kg in order to retain a negative temperature coefficient at equilibrium. In this case, the

Table 12
Results from calculations, by NRG, on the use of pure first and second generation Pu (oxide) in a pebble-bed HTR in continuous reload operation mode

Case	Pu-fiss (%)	Pu feed (g/d)	BU cycle (GWd/t)	BU disc (GWd/t)	k_{eff} (BOL)	k_{eff} (equiv.)	$\alpha(T)$ (BOL)	$\alpha(T)$ (equiv.)	Rem. HM(%)	Rem. Pu (%)
Pu-1-1.00-0	70.0	255.8	373	781.8	1.2825	1.0717	−3.25E−05	−4.01E−06	23.4	18.0
Pu-1-2.00-1	70.0	255.8	373	781.8	1.2931	1.0705	−2.95E−05	8.73E−07	23.4	18.1
Pu-1-2.00-0	70.0	254.1	361	787.0	1.1892	1.0353	−5.41E−05	−4.60E−05	23.3	15.9
Pu-1-1.00-0	67.0	266.3	384	751.1	1.2961	1.0706	−1.94E−05	1.64E−05	23.4	16.7
Pu-1-2.00-1	67.0	266.3	384	751.1	1.3044	1.0686	−1.72E−05	2.08E−05	23.5	16.8
Pu-1-2.00-0	67.0	270.1	366	740.4	1.2111	1.0575	−4.17E−05	−3.28E−05	23.4	14.8
Pu-2-0.75-0	42.2	256.0	376	781.4	1.1647	0.8411	2.58E−05	9.80E−05		
Pu-2-1.00-0	42.2	256.2	363	780.7	1.1436	0.8991	6.10E−07	5.43E−05	22.6	6.6
Pu-2-2.00-1	42.2	266.3	383	751.1	1.1509	0.8789	9.66E−06	7.54E−05	22.6	6.7
Pu-2-1.50-0	42.2	256.3	377	780.4	1.0964	0.9192	−2.40E−05	7.10E−06	22.6	6.8
Pu-2-3.00-1	42.2	268.2	382	745.8	1.1124	0.9156	−1.71E−05	2.07E−05	22.6	6.9
Pu-2-2.00-0	42.2	271.0	379	738.0	1.0560	0.9250	−4.64E−05	−2.32E−05	22.7	6.9
Pu-2-3.00-0	42.2	282.9	358	706.9	1.0013	0.9210	−7.35E−05	−4.87E−05	22.7	7.5
Pu-2-1.50-0	42.2	456.6	195	438.1	1.0964	1.0068	−2.40E−05	−2.39E−05	54.8	46.5
Pu-2-2.00-0	42.2	444.6	197	449.8	1.0560	0.9831	−4.64E−05	−4.17E−05	55.1	44.9

Table 13
Fuelling strategy of the HTR-MODUL reactor for burning first generation LWR Pu

	(a) High Pu-burning ratio		(b) Low residual Pu in discharged fuel	
	Pu-FE (50%)	U/Th-FE (50%)	Pu-FE (50%)	U/Th-FE (50%)
Pu	3 g		3 g	–
Th	–	18 g	–	16.7 g
U (HEU)	–	2 g	–	3.3 g
Incore time		7.3 F.P. years		11.0 F.P. years
Fractional power production	65%	35%	52%	48%
HM-burn-up	595 MWd/kg	58 MWd/kg	700 MWd/kg	116 MWd/kg
Average		128 MWd/kg		192 MWd/kg

reduction of only about 50% of the original plutonium can be achieved.

Similar investigations concerning first and second generation plutonium have been conducted by FZJ. The numerical investigations within this study have been performed by means of the V.S.O.P.(99) code. In the FZJ calculations concerning first generation plutonium pebble-bed core is assumed to be fuelled according to the two-pebble concept (see Table 13). One type of pebbles (Pu-FE) contains PuO_2-coated particles with a diameter of 0.24 mm having a total of 3 g plutonium per pebble (first generation Pu, composition “A”, see Table 9). The assumed maximum attainable burn-up of this particle is 800 MWd/kg. The second pebble type (U/Th-FE) contains 20 g $(HEU\text{-}Th)O_2$ in the form of larger coated particles (diameter 0.5 mm). The assumed maximum attainable burn-up of this particle is 120 MWd/kg. On the one hand the addition of uranium to the thorium is necessary to sustain criticality – depending on the desired burn-up of the fuel –, and, on the other hand, in order to achieve a prompt temperature increase of the resonance absorber, thorium, in case of an increase of the neutron flux, thus causing a prompt negative reactivity feedback. The uranium is highly enriched (93%) in order to minimize the build-up of Pu.

A strategy for burning Pu can be optimised in view of two principal objectives. Today’s main goal probably should be to reduce the separated amounts of Pu as soon as possible. This – in other words – means to maximize the amount of Pu depleted in nuclear reactors per unit of produced energy, which is equivalent to maximizing of the fractional power production by Pu in the reactors. Another important aspect, however, comes up with a view to intermediate storage of burned fuel and to final disposal of fuel without Pu-separation, as well as with respect to the non-proliferation aspect. From these points of view the minimization of residual Pu in the discharged fuel elements should be the main goal of the fuelling strategy. Here, the high burn-up, which is achievable in case of HTGR fuel elements, is a feature of particular importance. The positive features of this fuelling strategy, of course, imply the need to handle highly enriched uranium.

In Table 13 a comparison is shown of the two fuelling strategies indicated above (cases “a” and “b”) for the incineration of spent first generation LWR-Pu in the considered HTGR core. The first strategy is designed to achieve a high Pu-burning ratio, the second one to achieve an especially small amount of residual Pu in the discharged fuel elements. Table 14 displays the corresponding mass balance of the plutonium and of the fissile uranium. Detailed further information can be found in Ruetten and Haas (2002).

Both cases apply two kinds of fuel elements, as it has been described above. About half the reactor power is produced by fissions of the Pu. The charged Pu is depleted by 81% (Table 14, case “b”) and about 500 kg Pu are incinerated per GWa of produced electrical energy, assuming the efficiency 0.4 for the HTGR power plant. In case “a” the burn-up period of the fuel elements is reduced from 11 down to 7.3 years of full power, and thus the average burn-up of the fuel is lowered to a standard operation value of the German AVR reactor. In consequence the amount of Pu burned per GWa_{el} increases by 30%. On the other hand, the residual Pu of the discharged fuel also increases from 19 to 31% of the initial amount. The requirement of uranium is similar in both cases.

A parametric study on the temperature coefficients of an HTGR for Pu-burning showed the need for a relatively large Pu-

Table 14
Mass balances for the HTR-MODUL burning first generation LWR Pu

	(a) High Pu-burning ratio		(b) Low residual Pu in discharged fuel	
	Pu-FE (50%)	U/Th-FE (50%)	Pu-FE (50%)	U/Th-FE (50%)
Pu charged (kg/GWa_{el})	929	–	615	–
Pu discharged (kg/GWa_{el})	265	23	93	26
Pu burned (kg/GWa_{el})	664	−23	522	−26
	641 net		496 net	
Pu burned/Pu charged	0.69		0.81	
U^{235} charged (kg/GWa_{el})	–	578	–	624
U^{235} discharged (kg/GWa_{el})	–	251	–	171
U^{233} produced (kg/GWa_{el})	–	161	–	116

load of the fuel elements, favourably about 3 g Pu. The result is a "hard" thermal neutron spectrum, which favours the parasitic absorption of neutrons in the resonance of the ^{240}Pu-absorption cross-section at the energy 1 eV. Its increase with the moderator temperature dominates some others – partly contrary – spectral effects. Thus, the value of the moderator coefficient is strongly influenced by the fraction of ^{240}Pu in the fuel. Nevertheless, the temperature reactivity coefficients of the reactor (both Doppler and moderator coefficient) were found to be sufficiently negative over the whole applied range of reactor operation.

Second generation plutonium contains a distinctly lower fraction of fissile plutonium (about 40–50%) compared to plutonium of the first generation (about 70%) (also see Table 9). Following their investigations concerning first generation plutonium, FZJ has concluded a study on continuous reload pebble-bed reactors loaded with a mixture of second generation PuO_2 and $(U\text{-}Th)O_2$, comparing a number of different fuelling strategies. These strategies involved different combinations of the following fuel element types:

- Pu, Type 1: 3 g plutonium 2. Generation/fuel element;
- Pu, Type 2: 1 g plutonium 2. Generation/fuel element;
- Pu, Type 3: 0.5 g plutonium 2. Generation/fuel element;
- Th, Type 1: 20 g (Th + HEU)-MOX/fuel element;
- Th, Type 2: 10 g (Th + HEU)-MOX/fuel element;
- U: 10 g U (20% ^{235}U)/fuel element.

Some results of these investigations are shown in Table 15. The composition of the second generation plutonium differs (238/239/240/241/242 = 5/36/35/10/14 wt.%) slightly from the definition in the "Common Parameters" document (Kuijper et al., 2002). However, the results, as presented in Table 15, show a good agreement with those from the NRG studies, for the pure ("100%") Pu case. The combination of thorium and plutonium allows for a slightly higher burn-up of the permanently discharged second generation Pu fuel elements, leading to a somewhat higher reduction of the initial Pu content.

Pu incineration by means of a use of low-enriched uranium as basic fuel turns out to be the by far most unfavourable variant of the regarded fuelling strategies. Here, the production of new plutonium by the breeding effect of ^{238}U is nearly as big as the destruction rate by fissions. As consequence, the amount of plutonium in the unloaded fuel elements is lower by only 14% compared to the start of the irradiation.

3.2.2. *Maximum power size*

IKE has adopted and actualized the ZIRKUS program system to model pebble-bed reactors with annular core and performed fuel cycle equilibrium calculations with different core sizes and reload cycles with uranium oxide fuel. The goal of investigations is the optimization of power of a pebble-bed HTGR under constraints of limitation of maximum fuel temperature during a depressurisation accident. Starting point of the investigation was the HTR-MODUL reactor with 200 MWth and LEU fuel. Further calculations were performed for annular cores with increased power up to 400 MWth.

The maximisation of the power size of modular pebble-bed HTGRs under the constraints that defined temperatures of fuel and structure components will not be exceeded even under all kinds of loss of coolant accidents is an important task for developing of inherent safe and economic nuclear reactors. For HTR pebble-bed reactors the maximum power under these constraints can be achieved by several design and reload concepts:

- Reload strategy of fuel or moderator spheres;
- Annular core with inner reflector column of moderator spheres;
- Annular core with solid inner reflector column;
- Core height;
- Thermal isolation of the core to limit the temperature of the pressure vessel during LOCA.

For all concepts additionally to the main constraints requested from inherent safety principle, all safety related parameters such as reactivity coefficients, shutdown margin, maximum fuel temperature etc. must lie inside of distinct limits to guarantee safety under operating and accidental conditions. The power conversion can be realised by a RANKINE or a BRAYTON cycle. The coolant will be in all cases He. Typical mayor design parameters for pebble-bed HTRs are given in Table 16.

The HTR-MODUL and PBMR designs with dynamic middle column only need one outlet for the operating elements (fuel or moderator elements), The concept with compact solid inner reflector needs at least three outlets. The difference between the MODUL concept and the PBMR concept with dynamic inner column is the reload strategy. The PBMR concept reloads into the inner cylindrical part of the core pure moderator elements, which allows a total higher power since the maximum fuel temperature under DLOCA conditions is lower compared to the

Table 15
Comparison of different fuelling strategies for the incineration of second generation plutonium in pebble-bed HTGRs

Fuel elements	Heavy metal burn-up (MWd/t)	Average burn-up (MWd/t)	Pu charged (kg/GWa$_{el}$)	Plutonium burned (kg/GWa$_{el}$)	Ratio Pu burned/ Pu charged (%)
50% Pu, Type 1	522000	174000	683	450	66
50% Th, Type 1	122000				
50% Pu, Type 1	495000	200500	1048	643	61
50% Th, Type 2	112000				
100% Pu, Type 2	428000	–	2050	1020	50
100% Pu, Type 3	416000	–	2093	968	46
50% Pu, Type 1	145000	55000	3725	503	14
50% U	30000				

Table 16
Overview of major design parameters for selected modular HTGR concepts

Reactor	HTR-MODUL	PBMR dynamic inner column	PBMR solid inner column
Thermal power (MW)	200	302	400
Core layout	Cylindrical	Annular core with dynamic middle column	Annular core with compact middle column
Outer diameter of core (m)	3	3.7	3.7
Inner diameter of core (m)	–	2	2
Height of core (m)	9.4	9.3	11
Diameter of RPV (m)	6	6.2	6.2
Inlet/outlet temperature (°C)	250/700	500/900	500/900
Coolant	Helium	Helium	Helium

MODUL concept with cylindrical active core. The MODUL concept can only vary the reload strategy or the core height to increase the total power. The influence of the number of reloads of fuel elements on the maximum fuel temperature is shown Fig. 6.

The larger the number of reloads, the lower is the peaking factor of the axial power distribution and hence the lower the maximum temperature after a DLOCA accident. This means the power can be increased if 15 instead of 5 reloads are planned. For the case of a strategy, which reloads into the inner part fuel elements with higher burn-up and into the annular part, fresh fuel and fuel elements with lower burn-up, the radial power distribution can be flattened and correspondingly the radial peak factors kept lower than in the original MODUL concept. This allows also to increase the total power while the maximum fuel temperature under DLOCA conditions remains below the limit. The disadvantage is the necessity of more feed channels compared to the one of the MODUL. The maximum power, however, can be achieved if the inner part of the core is inactive. The disadvantage is the radial temperature profile in the core during operation and the very high thermal flux in the thermal column, which can be problematical if a fresh fuel element enters this region. A variable reload concept with higher irradiated fuel in the middle column avoids this problem and the problem of mixing of very different He temperatures at core outlet.

Comparing concepts with dynamic and solid inner columns with inactive material regarding the maximum fuel temperature after DLOCA an advantage for the solid inner part can be seen. For both designs under considerations, the behaviour is quite similar. The reactor starts to heat up due to the decay heat. The initial temperature profile under operating conditions, with maximum temperatures at the core exit, is transformed in an axially essentially symmetric profile imposed by the heat source distribution. The maximum temperatures in the core continuously increase, until they reach a maximum around 2.5 days (see Fig. 7), when the decreasing decay heat can be removed from the core region by heat conduction and radiation. The time, when the maximum fuel temperature is approached, marks also a transition from transient to quasi-steady behaviour. This can especially be seen from the radial temperature profile in the core. During heat-up, the temperatures in the unheated middle column lag behind the temperatures in the annular core. When the maximum temperature is approached, the radial profile over the central column vanishes. The subsequent cool-down then follows a quasi-steady behaviour, in which the developed temperature profile is practically maintained.

For the DLOCA case, the differences between the designs with dynamic and fixed middle column are relatively small. The maximum fuel temperature remains within acceptable limits. The higher temperatures reached in the case with dynamic middle column are mainly caused by the lower thermal iner-

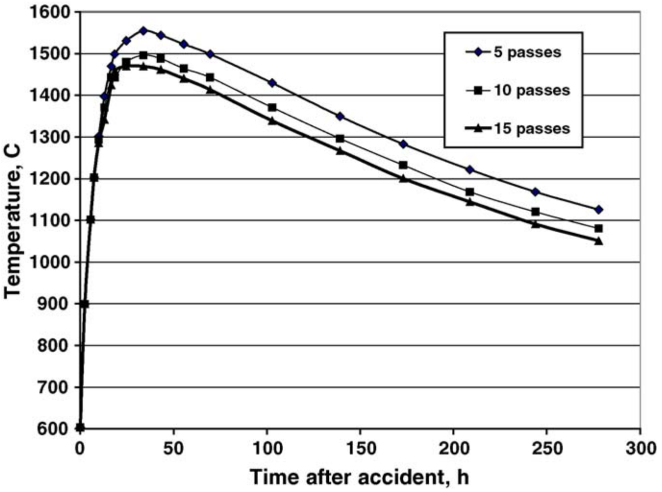

Fig. 6. Maximum fuel temperature as a function of time after depressurisation for different recycle passes for HTR-MODUL (200 MWth).

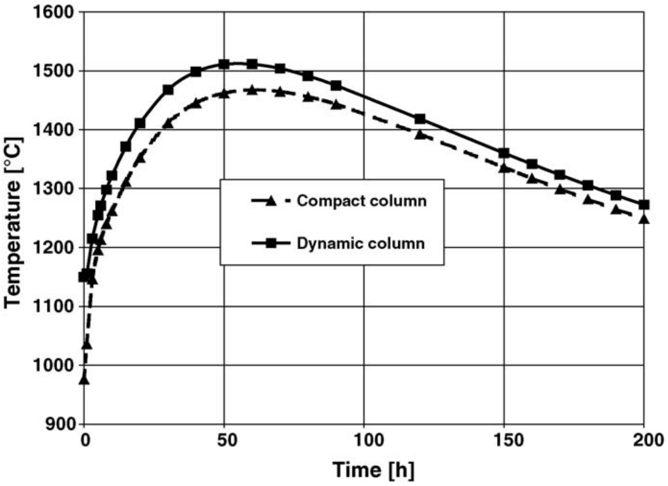

Fig. 7. Comparison of maximum fuel temperature versus time for pebble-bed HTGR designs with annular core in the DLOCA case.

tia of the pebbles compared to a massive graphite column. Thus, the maximum temperature is reached at an earlier time, when the decay heat to be removed is still at a higher level.

In the case of a failure of the circulation of coolant is assumed and the reactor is supposed to be shut down but remains under pressure (PLOCA). A natural convection flow develops under the influence of driving temperature differences, which in contrast to the DLOCA case strongly affects the decay heat redistribution. Due to a large initial radial temperature gradient, which results from operational conditions with cooled middle column and hot core annulus, a strong natural circulation loop has developed after 4 s. Helium rises in the outer hot annulus, heating up the upper parts, and flows downwards through the central part, releasing heat to the cold middle column. As a result, the hot spot moves from its initial position at the bottom towards the top of the core.

While the radial temperature difference between core and middle column is successively reduced, a second convection loop develops due to the increasing radial temperature gradient between core and reflector, which is cooled by conduction and radiation. The heat flux redistributed through the two loops exceeds at around 2.6 h the local decay power in the hot spot region, resulting in the first peak and subsequent decrease of the maximum fuel temperatures. Finally, the first circulation loop practically disappears due to the continuous heat up of the middle column. With only the reflector as major heat sink, the heat redistribution by convection is less effective. This leads to an at first renewed increase of the maximum temperature, until the core finally cools down with decreasing decay power. Although the fuel temperature remains within prefixed limits during the DLOCA accident the reference design with 425 MWth cannot be considered as optimum in terms of inherent safety. The mechanical stability of the reactor pressure vessel e.g. cannot be guaranteed if it is exposed to too high temperatures for a long time. At this point a second safety criterion regarding this fact is needed. A design with a thermal isolation of the reflector, keeping both the maximal accident temperature of the fuel and the maximal temperature of the RPV below distinct limits. More details of these investigations are shown in Ben Said et al. (2004).

3.3. Batch-wise reload hexagonal block-type HTGR

For the GT-MHR-based reference reactor, CEA investigated the Pu (and minor actinide) incineration capability. It is noteworthy that to give information such as cycle length, mass balance, peak power, core flux distribution, etc. for a specific block-type HTGR loaded with plutonium fuels suppose an important *optimization stage of the core*: use burnable poison or not, flattening the flux distribution in the annular zone (different filling fraction in the compact close to the reflector, different enrichment, burnable poison in the reflector, etc.), number of the control rods, etc. Moreover, this optimization stage must be based on an equilibrium fuel cycle assuming a specific fuel-reshuffling scheme. For the present analysis, simplest methods than 3D core burn-up calculations were employed. 2D transport detailed calculations allowed to compute the fuel depletion. Nevertheless, in order to get the fuel element discharged burn-up, the core reactivity was calculated during fuel depletion using a simplified 2D annular core configuration on which also transport calculations have been done. It is important to note that all these calculations have been performed without taking into account temperature feedback. The same 2D annular core configuration was used for the temperature coefficient estimations. The plutonium and minor actinides balances were calculated considering a thermal efficiency of 48% and a loading factor of 0.85.

Intimately connected to the batch-wise reload (hexagonal block-type) HTGR is the use of burnable poison, e.g. to flatten the reactivity-to-time behaviour of the core or to improve the temperature reactivity coefficients. A detailed study was performed on the optimization of the burnable poison particle design, in combination with different HTGR fuel types.

3.3.1. Plutonium incineration capability

The investigation of fuel cycle studies for block-type HTGR cores was performed for *first generation* and *second generation* plutonium-based fuel cycles. The core neutronic analysis presented here is essentially based on a specific calculation process employing the APOLLO2 transport code. The formalism used to solve the multi-group Boltzmann transport equation is either the integral-equation (P_{ij}: collision probability 1D and 2D) or integral-differential-equation (S_n: discrete ordinates and nodal methods in 2D). The standard 172-group cross-section library issued mainly from JEF 2.2 is used in the present study. Detailed information can be found in Damian and Raepsaet (2004a).

The calculations have been performed in fundamental mode (critical buckling), considering a linear anisotropic collision hypothesis for the calculation of the graphite diffusion coefficient. In order to evaluate the fuel element discharged burn-up, the core k_{eff} needs to be calculated. The core reactivity during fuel depletion is calculated using a simplified 2D core configuration. Although the calculations are performed using a simplified modelling, it allows making an accurate calculation of the radial leakage during core depletion. After all, the core volumic leakage (3D leakage) was evaluated using the radial leakage issued from the simplified core calculation and considering a constant axial leakage value (1500 pcm in all cases). For the different fuel types feeding the reactor, the discharged burn-up was determined in order to achieve a reactivity margin of 2000 pcm at the end of cycle ($k_{\text{eff}} = 1.02$) embracing the possible uncertainties.

Preliminary investigations showed that:

- The fuel cycle length increases linearly with the mass of plutonium loaded into the core;
- There is an optimum for the fuel fed into the core with respect to the discharge burn-up, which allows using at best the plutonium (see Fig. 8; case “A” is first generation, case “B” is second generation).

Indeed, an increase of the total mass fed into the core has been analyzed for both types of plutonium fuel. All the results are gathered in Table 17. Whatever the plutonium isotopic content

Table 17
Plutonium and minor actinides balance for first and second generation plutonium fuel in batch-wise fuelled hexagonal block-type HTGR

Type of fuel	First generation plutonium (66.2%)				
Mass of fuel loaded into the core (kg)	701	900	1200	1500	1800
Plutonium balance					
%	−67.4	−71.3	−74.4	−75.4	−75.1
Pu_f/Pu_{total} at EOL (%)	28.3	28.6	30.0	32.7	36.7
Minor actinides balance					
In percentage of metal burnt	8.3	9.2	10.2	11.1	12.0

Type of fuel	Second generation plutonium (42.2%)		
Mass of fuel loaded into the core (kg)	700	900	1100
Equilibrium cycle length	180	234	275
Average discharged BU	460.7	468.0	450.7
Plutonium balance			
%	−56.1	−58.2	−57.6
kg/TWhe	−107.4	−110.3	−113.5
Pu_f/Pu_{total} at EOL (%)	19.35	22.4	27.0
Minor actinides balance			
Americium (kg/TWhe)	+13.57	+14.88	+16.93
Curium (kg/TWhe)	+3.90	+5.29	+6.43
Total (kg/TWhe)	+17.47	+20.17	+23.36
In percentage of metal burnt	16.3	18.3	20.5

is, the fuel cycle length is proportional to the total mass loaded into the core. The higher the plutonium loaded into the core, the longer the fuel cycle length. Nevertheless, an increase of the plutonium loaded into the core will be limited by technological and physical criteria. For example, the particles volume fraction in the compact represents a technological limit to the plutonium loading capacity. Besides, the reactivity margin at the beginning of cycle appears as a physical limit to the use of highly degraded plutonium or important fuel loading. In fact, higher plutonium loading imply an increase of great absorbers like ^{240}Pu in a similar core geometry and reduce the reactivity margin although the fissile isotopes content increases. By increasing the loaded fuel mass, the neutron spectrum becomes harder and favours the neutron absorption in the fertile isotopes. It should be noted that if the plutonium balance reaches an optimum with respect to the plutonium loaded into the core, it is not the case with the minor actinide balance, which increases linearly with the mass of plutonium. One could have thought that maximize the burn-up should minimize both discharged masses of Pu and minor actinides. In fact, as shown in Table 17, the production of minor actinides raises continuously with the Pu-loading. Consequently, the optimum burn-up obtained from the critical calculations, which leads to an optimum of the plutonium consumption with respect to the fuel loading, can be explained as follow:

- Despite a smaller initial reactivity, the increasing of the Pu-loading leads to a neutron spectrum hardening that will enhance the Pu conversion and thus increase the cycle length (then the burn-up);
- At a certain level of Pu-loadings a too hard neutron spectrum (deteriorating the fission rate) and the important amount of minor actinides in the fuel will limit again the cycle length and thus the burn-up.

Therefore, for each isotopic Pu-composition, an optimum Pu-loading exists that maximizes the burn-up and then minimizes the Pu-discharge despite a constant MA-discharge mass increasing.

Finally, as far as the first generation plutonium is concerned, the temperature effect (Doppler and moderator) has also been evaluated on the fuel element geometry (see Table 18). The Doppler coefficient given in the table is an average value between 20 and 900 °C. As far as the moderator temperature coefficient is concerned, the calculated value is an average between 20 and 500 °C. Despite the strong decrease of the moderator temperature coefficient during fuel irradiation, the results have shown that the global core temperature effect is negative and therefore self-stabilizing, with a fuel management by 1/3rd where the average core burn-up ranges roughly from 200 and 400 GWd/t between the beginning and the end of cycle.

Further studies have also been conducted concerning the incineration of minor actinides in prismatic block HTGRs. Final conclusions on this application cannot be drawn yet, as the assembly based calculations do not provide for sufficient accuracy (Damian and Raepsaet, 2004b).

Prismatic block-type HTGRs have a flexible core that can fulfil a wide range of diverse fuel cycles. The use of a wide spectrum of plutonium isotopic compositions prove HTGR potentials to use at best the plutonium as fuel without generating large amounts of minor actinides. However, the analysis has been done without really taking into account the common fuel particle performance limits (burn-up, fast fluence, temperature). It is obvious that such long cycles and associated high level of Pu-destruction will be possible only if burn-ups as high as 700 GWd/t and fluences in the order of 12 n/kb (a factor 2 with the common requirements) sustained by the fuel particles will be technologically feasible. The use of high-burn-up plutonium

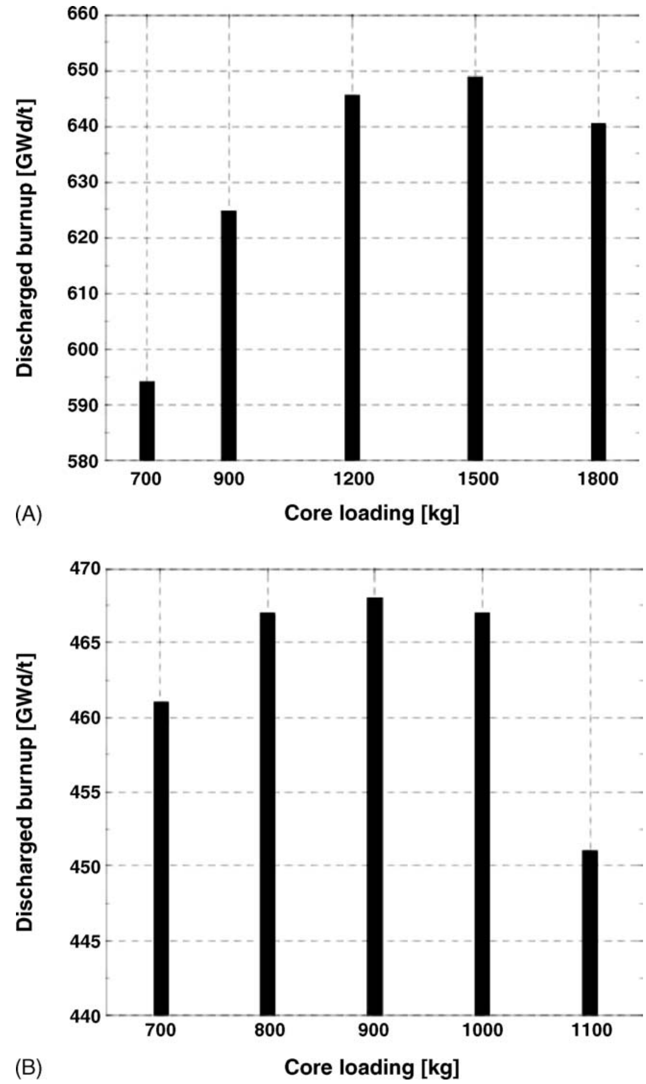

Fig. 8. Dependence of burn-up on core loading of the batch-wise loaded hexagonal block-type HTGR (case “A” is first generation plutonium, case “B” is second generation).

particles cannot be regarded as proven technology today and important fuel characterisation program including irradiation will be required to demonstrate that a burn-up of about 80% “fissions per initial metal atom” (FIMA) can be achieved for the Pu-particles without an inadmissible failure rate of the fuel coating.

It should be stressed that precaution must be taken with regard to the preliminary results given in the previous section. Indeed, the indicated mass balances have not been estimated from 3D full core calculations and remain to be confirmed. Nevertheless, such a 3D core calculation is inferred that a core optimization approach close to conceptual design studies is needed for a block-type reactor fully loaded with plutonium fuel. This has not been carried out in the present analysis.

Moreover, it is noticeable that further detailed core physic analyses will be required in the future in order to assess the dynamic features of such a reactor, as is also the case for the pebble-bed HTGR (Section 3.2.1). Additional studies concerning also the reactivity control aspects, the temperature coefficients, the decay heat associated with plutonium fuel, the appropriate fuel management and the associated power distributions related issues (especially important in the case of the plutonium use) should allow to precise that pure plutonium cycles will respect the current high level of safety of the HTGR.

3.3.2. *Optimization of burnable poison design*

A batch-wise fuel load scheme in HTGRs can be combined with a burnable poison in a heterogeneous way by mixing burnable poison particles (small spherical particles made of burnable poison, in the remainder abbreviated as BPPs) in the fuel elements. By varying the diameter of the BPPs and the number of these particles per fuel pebble, it is possible to tailor the reactivity-to-time curve.

Such a batch-loading scheme in HTGRs combined with BPPs has some attractive properties not offered by the continuous loading scheme. Burn-up calculations have been performed on a standard HTGR fuel pebble with a radius of 3 cm containing 9 g of enriched uranium or 1 g of first-grade plutonium, together with spherical BPPs made of B_4C highly enriched in ^{10}B or Gd_2O_3 containing natural Gd. The calculations aim at obtaining a flat reactivity-to-time curve for a batch-wise-loaded HTGR by varying the radius of the BPP and the number of particles per fuel pebble.

With BPPs mixed in the fuel of an HTGR, it is possible to control the excess reactivity present at beginning of life. For 8% enriched UO_2 fuel, mixing 1070 BPPs containing B_4C with radius of 75 μm through the fuel zone of a standard HTGR

Table 18
Doppler and moderator temperature coefficient for the first generation Pu in batch-wise fuelled hexagonal block-type HTGR

Burn-up (GWd/t)	701 kg	900 kg	1200 kg	1500 kg	1800 kg
Doppler coefficient (pcm/°C)					
	Without erbium				
0	−2.76	−3.13	−3.49	−3.67	−3.70
	−0.98	−1.00	−0.92	−1.04	−1.14
Variable	625 GWd/t	650 GWd/t	650 GWd/t	650 GWd/t	650 GWd/t
Graphite temperature coefficient (pcm/°C)					
	Without erbium				
0	−2.29	−2.14	−1.91	−1.69	−1.46
	+8.15	+6.33	+4.47	+2.86	+1.74
Variable	625 GWd/t	650 GWd/t	650 GWd/t	650 GWd/t	650 GWd/t

fuel pebble with outer radius of 3 cm, the reactivity swing is 2% at a k_∞ of 1.1. This means the burnable poison occupies a volume 60,000 less than that of the fuel pebble (FVR = 60,000).

Using Gd_2O_3 as a burnable poison gives an optimum radius of about 840 μm and an FVR of only 5000. This latter number corresponds to 9 BPPs per fuel pebble. The low number for the FVR reflects the fact that the natural Gd in the particle absorbs fewer neutrons despite the fact that the thermal cross-sections of the ^{155}Gd and ^{157}Gd isotopes are much larger than that of the ^{10}B. This is due to the relatively large microscopic absorption cross-section of ^{10}B in the epi-thermal range and the high atomic number density of the boron in B_4C. For the Gd_2O_3 particles, the resulting reactivity swing is 3%, which is very similar to that obtained with the B_4C particles. The bigger size of the Gd_2O_3 particles could be advantageous for the manufacturing process of the BPPs.

The B_4C particles used in UO_2 fuel (radius between 70 and 90 μm) can also be used to reduce the reactivity swing in PuO_2 particles. The reactivity swing at a target k_∞ of 1.1 is about 4% for BPPs with radius of 85 μm and an FVR of 27,500 (corresponding to 1600 BPPs per fuel pebble). The uniform temperature coefficient is comparable to that of the UO_2 fuel (−7 to −8 pcm/K). More results can be found in Kloosterman (2003a,b).

3.4. Spectrum transmitter

The disposal of nuclear waste is one of the major problems to be solved to guarantee a future for the nuclear industry. For this reason, the incineration of plutonium and minor actinides (MA) is probably the most interesting and effective option in reducing the radio-toxicity of the wastes produced by the nuclear fuel cycle.

An alternative solution to fast reactor or ADS is to make use of thin fissile films as flux converters to generate regions with fast fluxes inside a thermal reactor and thereby improve their incineration capabilities. The basic idea is to isolate some regions inside the reactor by de-coupling them from the main core with a flux converter. Provided that no moderating material is present inside these regions, the flux there will be prevalently fast and allow a more effective incineration of minor actinides.

The scope of this work is to analyse the feasibility of fast islands in thermal reactor, by giving a rough estimation on basic dimensioning, flux conversion and incineration performances. Presently, the main conclusions are as follows (Magill and Peerani, 2001a):

- It is possible to obtain fast islands inside the cores of thermal reactors by coating special assemblies with thin films of fissile material. These special assemblies have to be moderator-free.
- The special assemblies could be loaded with minor actinides to enhance the incineration rates in the fast spectrum.
- The fast flux inside the thermal islands is improved by a factor ranging from 2 to 10, depending on the reactor type and on the film material and thickness. This improves considerably the capabilities of MA transmutation.
- In a PWR the realization of a fast island with the same dimensions of a standard fuel element is possible from the neutronic point of view. Nevertheless, since in this kind of reactor water is both coolant and moderator, the condition requiring no moderator inside the fast island leads to a severe heat removal problem.
- Intrinsic to the HTGR concept is the fact that the moderator (graphite) and the coolant (gas) are distinct. It follows that heat can be easily removed without introducing any significant neutron moderation.
- The pebble dimension in pebble-bed HTGR is not optimal for the fast island concept. In fact, since the minimum thickness of the fissile film is imposed by neutronic conditions to be at least 1 mm, the fast island should have reasonably large dimensions in order to keep as low as possible the ratio between the fissile mass in the film and the MA loaded in the fast island.
- Block-type HTGR seems to offer the best conditions for an optimal design of a fast island.
- Typical incineration rates in fast islands are two to three times higher than the corresponding rates in thermal reactors.

Following the above results it seems worthwhile to go on the analysis to assess definitely the feasibility of the fast island concept. The most immediately required further steps are:

- Optimization of the MA assembly geometry;
- Analysis of the local effects close to the interface due to the flux perturbation induced by the presence of the fast island and of the fissile layer;
- Thermal-hydraulic analysis to verify the capability to remove the heat produced inside the fast island and in the fissile layer;
- Investigation of the impact on the main reactor safety parameters (feedback effects, dynamic behaviour).

A world patent has been granted on the spectrum transmitter concept (Magill and Peerani, 2001b).

3.5. Conclusions—HTGR concepts

A number of conceptual HTR designs were analyzed with respect to their capability to incinerate plutonium and minor actinides, while maintaining favourable safety characteristics. The basis for these investigations was provided by two reference reactors, representing the two main HTR designs, viz. HTR-MODUL (continuous reload pebble-bed) and GT-MHR (batch-wise reload hexagonal block). The investigations show quite promising results concerning the incineration (reduction) of especially first, but also of second generation plutonium, for both HTR concepts. It should be noted, however, that only an indication of the favourable safety characteristics was calculated in the form of sufficiently negative temperature reactivity coefficients. Future R&D work should address the actual dynamic properties of such Pu-loaded HTR cores under both operational and accident conditions.

Furthermore, in the analysis of the Pu-burning capabilities of the several HTR concepts it was assumed that the fuel is able to withstand very high burn-ups, in the range of 700 MWd/kg

Table 19
Comparison of the Pu incineration in the different studies

First generation	NRG (pebble-bed)	FZJ (pebble-bed)		CEA (batch-wise)
		High rate	Low residual	
Discharge burn-up (MWd/kg)	750	595	700	640
Pu-balance (%)	85	69	81	64
Features	Pure Pu (2 g Pu)	Pu + (Th + HEU)	Pu + (Th + HEU)	Erbium BP
Second generation	**NRG (pebble-bed)**	**FZJ (pebble-bed)**		**CEA (batch-wise)**
		Pu + (Th + HEU)	Pure Pu	
Discharge burn-up (MWd/kg)	445	495	428	470
Pu-balance (%)	55	61	50	58
Features	Pure Pu (2 g Pu)	3 g Pu/pebble	1 g Pu/pebble	Erbium BP

or higher. However, as in particular the use of high burn-up plutonium particles cannot be regarded as proven technology, an irradiation program will be required to:

- Demonstrate that a burn-up equal about 80% “fissions per initial metal atom” (FIMA) can be achieved for the Pu-loaded coated particles without an inadmissible failure rate of the fuel coating;
- Investigate the fission product retention of both the fuel element variants at a temperature level, which might occur in a loss-of-coolant accident.

For batch-wise (re-) loaded HTRs the use of burnable poison enables flattening of the reactivity-to-time behaviour of the reactor and the improvement of temperature reactivity coefficients. The investigations demonstrate the capabilities of burnable poison particles, containing either boron or gadolinium in the form of either small spheres or cylinders, to achieve these goals. Optimization of the behaviour is quite well possible by varying the diameter of the particles and/or the number of particles per fuel element.

A comparison of the Pu incinerating capacities of the different fuelling strategies of the of the pebble-bed and batch-wise fuelled reactors can be seen in Table 19, with the Pu-balance as Pu burned/Pu charged, showing a slight decrease in capability for the batch-wise fuelled reactor because of the neutron consumption by the burnable poison.

Investigations concerning the more exotic concept of the “spectrum transmitter” (thermal reactor containing “fast islands”—assemblies coated with thin fissile material) show that the block-type HTR seems to offer the best conditions for an optimal design of a fast island, and that the typical incineration rates in fast islands are two to three times higher than the corresponding rates in thermal reactors.

Acknowledgement

The work presented in this article was partly funded by the European Union Fifth Framework Program, under contract nos. FIKI-CT-2000-00020 “HTR-N” and FIKI-CT-2001-00169 “HTR-N1”.

References

Bernnat, W., Mattes, M., Keinert, J., 2003a. Validation of cross section libraries for analysis of thermal reactors. HTR-N/N1 Project Report. IKE, Germany.

Bernnat, W., Mattes, M., Keinert, J., 2003b. Neutron thermalisation in reactor-grade polycrystalline graphite. HTR-N/N1 Project Report. IKE, Germany.

Ben Said, N., Buck, M., Bernnat, W., Lohnert, G., 2004. The impact of design on the decay heat removal capability of a modular pebble bed HTR. In: Proceedings of the Second International Topical Meeting on High Temperature Reactor Technology (HTR’2004), Friendship Hotel, Beijing, PR China, September 22–24.

Chang, H., Raepsaet, X., et al., 2004. Analysis of HTR-10 first criticality with Monte Carlo code TRIPOLI-4.3. In: Proceedings of the Second International Topical Meeting on High Temperature Reactor Technology (HTR’2004), Friendship Hotel, Beijing, PR China, September 22–24.

Damian, F., Raepsaet, X., 2004a. Feasibility of burning first and second generation plutonium in prismatic block HTRs. HTR-N/N1 Project Report. CEA, France.

Damian, F., Raepsaet, X., 2004b. Feasibility of burning minor actinides in prismatic block HTRs. HTR-N/N1 Project Report. CEA, France.

Difilippo, F.C., Renier, J.P., Hawari, A.I., 2002. Benchmarks for the scattering kernel for graphite. In: International Conference on the New Frontiers of Nuclear Technology: Reactor Physics, Safety and High-Performance Computing (PHYSOR’2002), Sheraton Walker Hill Hotel, Seoul, Korea, October 7–10.

Dolci, F., 2003. Sensitivity analysis and estimation of major impact of nuclear data uncertainties to reactivity & mass balance. HTR-N/N1 Project Report. CEA, France/IKE, Germany.

de Haas, J.B.M., et al., 2004. Summary report of WP2 in HTR-N1. HTR-N/N1 Project Report. NRG, The Netherlands.

de Haas, J.B.M., Kuijper, J.C., 2005. Feasibility of burning first and second generation plutonium in pebble bed HTRs. Nucl. Technol. 151, 192–200.

General Atomics, 1996. Gas Turbine Modular Helium Reactor conceptual design report, July 1996.

HTR-N/N1 Contracts, 2000/2001. HTR-N—high-temperature reactor [nuclear] physics [waste] and fuel cycle studies. European Union Fifth Framework Program contracts no. FIKI-CT-2000-00020 and FIKI-CT-2001-00169, signed 31 August 2000 and 18 September 2001, respectively.

IAEA-TECDOC-1382, 2003. Evaluation of high temperature gas cooled reactor performance: benchmark analyses related to initial testing of the HTTR and HTR-10. IAEA, Vienna.

Kloosterman, J.L., 2003a. Design of burnable poison particles to reduce the reactivity swing in batch-wise fuelled high temperature reactors (fuel cycles with burnable poison). HTR-N/N1 Project Report. Delft University of Technology, The Netherlands.

Kloosterman, J.L., 2003b. Application of boron and gadolinium burnable poison particles in UO2 and PUO2 fuels in HTRs. Ann. Nucl. Energy. 30, 1807–1819.

Kuijper, J.C., et al., 2002. Definition of common parameters and reference HTR designs for HTR-N[1]. HTR-N/N1 Project Report. NRG, The Netherlands.

Kuijper, J.C., et al., 2004. HTR-N plutonium cell burnup benchmark: definition, results & intercomparison. In: Proceedings of the PHYSOR'2004 on the Physics of Fuel Cycles and Advanced Nuclear Systems: Global Developments, April 25–29, Chicago, IL, USA. On CD-ROM, American Nuclear Society, LaGrange Park, IL, USA.

Magill, J., Peerani, P., 2001a. Improving the incineration of minor actinides in thermal reactors by generation of fast islands with thin fissile films. Report JRC-ITU-TN-2001/10. JRC-ITU, Germany.

Magill, J., Peerani, P., 2001b. Method of incineration of minor actinides in nuclear reactors, 1 November 2001. Patent International Application Number WO 01/82306 A2 and PCT/EP01/04573.

Oppe, J., Kuijper, J.C., 2004. Plutonium HTR cell burnup benchmark-several calculations and their results. Report 20780/03.55862/P. NRG, Petten, The Netherlands.

Raepsaet, X., Damian, F., Ohlig, U.A., Brockmann, H.J., de Haas, J.B.M., Wallerbos, E.M., 2003. Analysis of the European results on the HTTR's core physics benchmarks. Nucl. Eng. Des. 222, 173–187.

Raepsaet, X., et al., 2004a. HTR-N1-control rod position and scram reactivity in the HTTR. HTR-N/N1 Project Report. CEA, France.

Raepsaet, X., et al., 2004b. HTR-N1-temperature coefficients evaluation in the HTTR. HTR-N/N1 Project Report. CEA, France.

Reutler, H., Lohnert, G.H., 1983. The modular high-temperature reactor. Nucl. Technol. 62, 22–30.

Ruetten, H.J., Haas, K.A., 2002. Incineration of LWR-plutonium in a modular HTR using thorium-based fuel. HTR-N/N1 Project Report. FZJ, Germany.

Seiler, R., 2004. Further analysis of HTR-PROTEUS experiments. HTR-N/N1 Project Report. PSI, Switzerland.

von Lensa, W., et al., 2003. European programme on high temperature reactor nuclear physics waste and fuel cycle studies. In: Proceedings of the 2003 International Congress on Advances in Nuclear Power Plants (ICAPP'03), Cordoba, Spain, May 4–7.

Young, J.C., Huffman, D., 1964. Experimental and theoretical neutron spectra. Report GA-5319. Gulf General Atomics, USA.