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Studies on the Continuous Reload Pebble Bed HTRs
Summary Report

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Summary Report

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SUMMARY

The core physics investigations at NRG and FZJ, as part of the activities within the 'HTR-N' project of the European Fifth Framework Program, are focussed on the incineration of pure first and second generation Pu in the reference pebble bed HTR ('HTR–MODUL') with a continuous reload ('MEDUL' – German: 'MEhrfach DUrchLauf', multi pass) fuelling strategy, in which the spherical fuel elements, or pebbles, pass through the core a number of times before being permanently discharged.

For pebbles fuelled with different plutonium loadings, the feasibility of a sustained fuel cycle under nominal reactor conditions were investigated by calculation of the reactivity and temperature coefficients of the reactor. The HTR–MODUL was found to be a very effective reactor to reduce the stockpile of first generation plutonium. It reduces the amount of plutonium to about one sixth of the original, and also reduces the risk of proliferation by denaturing the plutonium vector. For second generation plutonium the incineration is less favourable, as the amount of plutonium is only halved.

At IKE the maximum power for uranium fuelled HTRs has been investigated. As the decay heat removal capabilities are important safety features of the Modular Pebble Bed HTR, the reactor can be designed and optimised that even under loss of cooling accidents the decay heat can be removed by passive means without exceeding defined temperature limits for fuel and structure components. Such a design yields limitation of power output, however. The optimisation of power under constraints of limitation of fuel and structure temperature is therefore an important task.

Several designs with cylindrical and annular cores were investigated assuming both Pressurised and Depressurised Loss of Coolant Accidents (P/DLOCA).
Annular Cores can produce higher power output without exceeding fuel temperature limits during DLOCA accidents, since heat storage effect of the inner fuel free column have an influence on the maximum fuel temperature by increasing the time at which this temperature is reached. A thermal isolation of the core can decrease the maximum temperature of the pressure vessel during a LOCA. Optimum is reached if both maximum temperature of fuel and RPV remains within corresponding limits.

INTRODUCTION

Large amounts of civil plutonium have accumulated in the world in burnt fuel from the operation of nuclear power reactors. Until the year 2000 they amounted to about 1400 metric tonne.
The aim of the research reported here was to find a fuelling strategy for:

- Burning as much plutonium as possible per unit of produced energy
- Minimizing the residual amount of plutonium in the fuel for long term waste disposal

In the 'HTR-N' project of the European Fifth Framework Program investigations are being made concerning the feasibility of burning plutonium in High Temperature Gas-Cooled Reactors (HTR). The capability to reach ultra high burnup (up to 800 MWd/kgHM) with (TRISO/BISO) coated particles containing actinides, which has already been demonstrated in the early seventies, opens the way to burn e.g. military or civil plutonium in a way that no intermediate or further reprocessing is required. In addition the HTR coated particle fuel may already be well suited for final disposal due to the ceramic coating retaining the fission products from the biosphere.

A substantial part of the 'HTR-N' project is devoted to investigation and intercomparison of the plutonium and actinide burning capabilities of a number of HTR concepts and associated fuel cycles. This 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 HTR's 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 are under investigation.

For a meaningful assessment and intercomparison of HTR concepts a common basis has been defined and agreed upon by the partners in the 'HTR-N' project. This common basis includes the definition of a reference pebble bed reactor ('flat bottom' 'HTR-MODUL') with continuous re-loading ('MEDUL') of fuel elements, the definition of a reference hexagonal block type reactor ('GT-MHR'), 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.

This report is a summary of the HTR-N reports:

HTR-N-03/11-D-3.2.1 : *Feasibility of burning first and second generation Pu in a continuous reload HTR*, by J.B.M. de Haas and J.C. Kuijper, from NRG, The Netherlands.

HTR-N-02/05-D-3.2.2 : *Incineration of LWR-Plutonium (1. Generation and 2. Generation) in a Modular HTR using Thorium-based Fuel*, by H.J. Ruetten and K.A. Haas, from FZJ, Germany.

HTR-N-04/06-D-3.2.3 : *U based fuel and maximum power size*, by W.Bernnat, M. Mattes, J. Keinert, N. Ben Said, from IKE, Germany.

For the full details and references one is referred to these reports and to the report:

HTR-N-02/09-D-3.1.1: *Definition of common parameters and reference HTR design for HTR-N[1]*, by J.C. Kuijper et al.

1 REFERENCE REACTOR AND FUEL ELEMENT

The High Temperature Pebble-bed Reactor (HTR), which has been chosen as the basis for this study, is a close approximation of the HTR-MODUL-reactor, as it was designed in Germany by the INTERATOM Company. The small size of this reactor was required in order to avoid a release of fission products from the fuel elements by an inherent restriction of the fuel temperature to less than 1600 °C even in case of a complete loss of active cooling. The schematic of the reactor is illustrated in figure 1. The reactor core consists of a statistical filling of about 360000 pebble-shaped fuel elements, which are circulated about ten times through the core, until they reach their final burnup. It is surrounded by the graphite reflectors, which contain the control devices and the ducts for the supply of Helium, which is the cooling gas.

As indicated in the introduction, the results of the studies on Pu incineration are to be inter-compared with those of other partners in the 'HTR-N' project. Therefore the reference pebble bed HTR was employed in this activity. A simplification on the HTR-MODUL design is: the conical shaped defuelling chute is not modelled and consequently a uniform vertical flow velocity distribution of the pebbles over the entire radius of the core is assumed. As in the original HTR-MODUL design, in our analyses the 'MEDUL' (German: 'MEhrfach DUrchLauf' – multi-pass) fuelling strategy was assumed as well. Main parameters on the reference pebble bed reactor are presented in figure 1 and table 1.

Table 1: 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 per m^3	5394 per m^3
He core inlet temperature	250 degr. C
He core outlet temperature	700 degr. C
System pressure	60 bar
He mass flow rate	85.55 kg/s
Basic graphite density (in reflectors)	1.80 g/cm^3

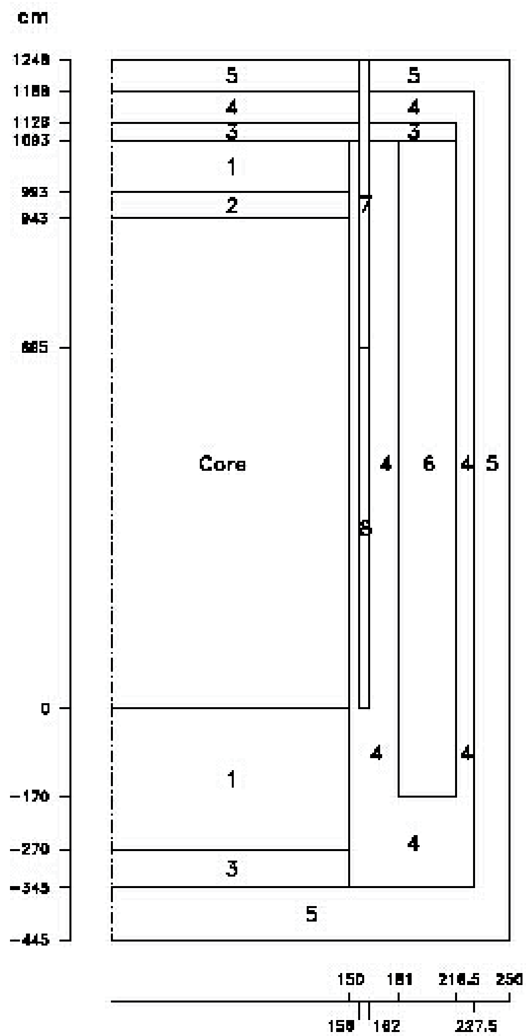

Figure 1: Main dimensions and material regions in the calculational model of the HTR-MODUL. Dimensions are in cm. The core is a random stacking of the well-known 6 cm fuel balls (‘pebbles’). The conical defuelling chute below the core is not modeled. Other (more or less homogenized) material regions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

The fuel is loaded in the form of coated particles, which are embedded in the graphite matrix of the fuel elements. In this way, an exceptionally high heavy metal burnup can be achieved.

The different fuel materials can be loaded into different types of coated particles. In a pebble-bed HTR the different types of fuel particles – e.g. Pu on the one hand and Th and U on the other hand – can also be loaded into different fuel elements. As they are continuously loaded and unloaded, these elements can even have different numbers of passes through the core – if desired – until they will reach their final burnup. On-line burnup measurements and the discrimination of LEU- vs. Th-fuel, respectively, have been successfully practiced for many years at the AVR reactor in Jülich (FRG), it was provided for the THTR-plant (FRG) and for the MODUL-200 HTR plant.

General information on the fuel pebbles containing coated particles is presented in table 2. More specific – case-dependent – data on the fuel, such as the isotopic composition of the Pu, is given in the sections.

Table 2: 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

Starting from the reference pebble-bed reactor, NRG and FZJ investigated the feasibility of the burning of first and second generation plutonium in such a reactor. By 3-D reactor calculations, combining neutronics and pebble-bed HTR core thermal-hydraulics, several loading schemes, including some containing mixtures of fuel pebbles containing different coated particle 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).
In the FZJ study also U-Th fuel has been admixed with the pure Pu fuel. For this option the HTR has some unique features, which make this system especially suitable to burn Pu on the basis of thorium-fuel.

2 PLUTONIUM INCINERATION - NRG-STUDIES

2.1 CALCULATIONAL METHODOLOGY

The 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 HTR reactor physics code system WIMS/PANTHERMIX, based on the well-known lattice code WIMS (versions 7 and 8), with nuclear data based on JEF-2.2, the 3-D steady-state and transient core physics code PANTHER and the 2-D R-Z HTR thermal hydraulics code THERMIX-DIREKT. At NRG the PANTHER code has been interfaced with THERMIX-DIREKT to enable core follow and transient analyses on both pebble bed and block type HTR systems. The plutonium vectors as used in the study are listed in table 3. An example of the power density and temperature distribution of the MEDUL reactor at equilibrium and fuelled with 2.0 gram second generation Pu per pebble is shown in Figs. 2 and 3. White lines indicate the core location, the dashed line indicates the pebble bed surface. From these figures one can notice the “up-swing” in power and the resulting temperature increase for the core region adjecent to the reflectors, because of the usual thermal fluence rate peaking in the reflectors. This peaking is caused by the slowing down process of the neutrons in a low absorptive medium.

Table 3: Plutonium vectors.

Nuclide	Composition of 1st generation Pu [weight %]		Composition of 2nd generation Pu [weight %]
^{238}Pu	1.0	2.59	4.90
^{239}Pu	62.0	53.85	26.90
^{240}Pu	24.0	23.66	34.30
^{241}Pu	8.0	13.13	15.30
^{242}Pu	5.0	6.78	18.6
Pu_{fiss}/Pu_{tot}	70%	67 %	42.2 %

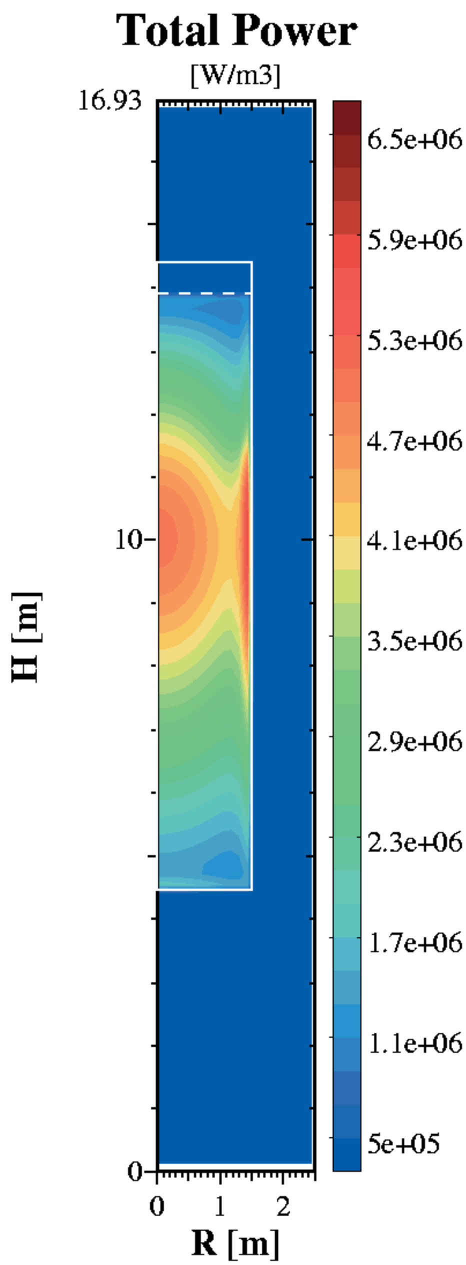

Figure 2: Power density.

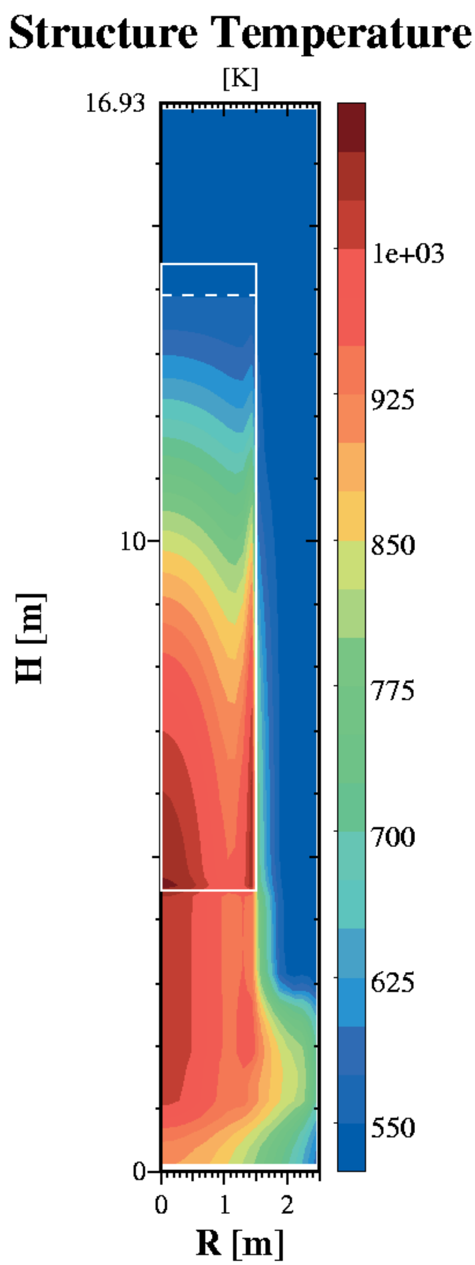

Figure 3: Reactor temperature.

2.2 RESULTS

The NRG analyses on the Pu-loaded HTR-MODUL presented in this report were performed by means of the WIMS/PANTHERMIX code system.
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 1st generation (pure) PuO_2, with 7 grams per pebble of initial heavy 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 1st and 2nd generation (pure) Pu in a 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 4. 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 3).

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

Table 4: 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	BU cycl	BU disc	k_{eff}	k_{eff}	α(T)	α(T)	rem. HM	rem. Pu
	%	g/d	GWd/t	GWd/t	BOL	equi	BOL	equi	%	%
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**

3 PLUTONIUM INCINERATION – FZJ STUDIES

3.1 CALCULATIONAL METHODOLOGY

The numerical investigations within this study have been performed by means of V.S.O.P(99). This computer code system has been developed for the comprehensive numerical simulation of the physics of thermal reactors. It includes processing of cross sections, the set-up of the reactor and of the fuel element, repeated neutron spectrum evaluation, neutron diffusion calculation in two or three dimensions, fuel burnup, fuel shuffling, reactor control and thermal hydraulics. The involved cross section libraries have been derived from the evaluated nuclear data files ENDF/B-V and JEF-1. The code can simulate the reactor operation from the initial core towards the equilibrium core.

3.2 RESULTS

The simultaneous use of different coated particle variants was originally developed for the high temperature reactors using prismatic fuel elements, as they were applied in the Fort. St. Vrain plant. Here, the fissile material was used in the form of small, so called “feed” particles reaching a burnup of about 90 % of its initial amount. Larger particles -“breed” particles- were made of ThO_2.

In the present study the pebble-bed core is fuelled by use of a two-pebble-concept (table 5). 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. 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). On the one hand the addition of uranium to the thorium is necessary to sustain criticality – depending on the desired burnup 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. While the (U-TH)-load of the fuel elements for the THTR-plant was 11g, values between 16 g and 20 g were provided in the frame of the German HHT-project (Hochtemperaturreactor mit Heliumturbine) and the reference maximum burnup was 120 GWd/t. Elements containing up to 30 g heavy metal were developed within the same project. A restriction to 20 g, however, should limit the fraction of coated particles which will be damaged during the fabrication process of the fuel elements to 10^{-4}.

Table 5: Fuelling strategy of the HTR-MODUL reactor for burning(1st generation) LWR Pu.

	a) High Pu-burning ratio fuel		*b) Low residual discharged Pu*	
	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 Prod.	65 %	35 %	52 %	48 %
HM-burnup	595 MWd/kg	58 MWd/kg	700 MWd/kg	116 MWd/kg
Average	128 MWd/kg		192 MWd/kg	

Table 6: Mass Balances for the HTR-MODUL burning (1st generation) LWR-Pu.
a) High Pu-burning ratio fuel b) Low residual discharged Pu

	Pu-FE (50%)	U/Th-FE (50%)	Pu-FE (50%)	U /Th-FE (50%)
Pu charged	929 kg/GWa_{el}	--	615 kg/GWa_{el}	--
Pu dischgd.	265 kg/GWa_{el}	23	93 kg/Gwa_{el}	26
Pu burned	664 kg/GWa_{el}	-23	522 kg/GWa_{el}	-26
		Σ 641		Σ 496
Pu-burned / Pu-charged	0.69		0.81	
U^{235} charged	--kg/GWa_{el}	578	--kg/GWa_{el}	624
U^{235} discharged	--kg/GWa_{el}	251	--kg/GWa_{el}	171
U^{233} produced	--kg/GWa_{el}	161	--kg/GWa_{el}	116

As in particular the use of high-burnup plutonium particles cannot be regarded as proven technology, an irradiation program will be required to:

- demonstrate that a burnup equal about 80 % “fissions per initial metal atom” (FIMA) can be achieved for the Pu (feed) particles without an inadmissible failure rate of the fuel coating,
- qualify (U-TH)-mixed oxide fuel elements loaded with 20g heavy metal,
- investigate the fission product retention of both the fuel element variants at a temperature level, which might occur in a loss-of-coolant accident.

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 burnup, which is achievable in case of HTR 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 the FZJ calculations concerning first generation plutonium the pebble-bed core is assumed to be fuelled according to the two-pebble-concept. In table 5 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 HTR 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 6 displays the corresponding mass balance of the plutonium and of the fissile uranium.
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 6, case “b”) and about 500 kg Pu are incinerated per GWa of produced electrical energy, assuming the efficiency 0.4 for the HTR 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 a HTR for Pu-burning showed the need for a relatively large Pu-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 (table 7).

Table 7: Temperature Coefficients ($\Delta K_{eff} / \Delta T$) at operating temperature
(Design for high Pu – burning ratio)

Doppler coefficient	$-1.71 \bullet 10^{-5}$ / K
Moderator coefficient	$-4.46 \bullet 10^{-5}$ / K
Total Temp.- coefficient	$- 6.17 \bullet 10^{-5}$ / K

Second generation plutonium contains a distinctly lower fraction of fissile plutonium (about 40% - 50%) compared to plutonium of the 1st generation (about 70%) (also see table 3). Following their investigations concerning first generation plutonium, FZJ has concluded a study on continuous reload pebble bed reactors loaded with a mixture of 2nd 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 — 3g Plutonium 2. Generation / fuel element
- Pu, Type 2 — 1g Plutonium 2. Generation / fuel element
- Pu, Type 3 — 0.5g Plutonium 2. Generation / fuel element
- Th, Type 1 — 20g (Th + HEU)-MOX / fuel element
- Th, Type 2 — 10g (Th + HEU)-MOX / fuel element
- U: — 10g U, (20% ^{235}U) / fuel element.

Some results of these investigations are shown in table 8 The composition of the second generation plutonium differs (238/239/240/241/242 = 5/36/35/10/14 weight %) slightly from the definition in the "Common Parameters" document. However, the results, as presented in table 8, 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 the use of low-enriched uranium as driving 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.

Table 8: Comparison of different fuelling strategies for the incineration of second generation plutonium in pebble-bed HTRs.

Fuel elements	Heavy metal burn-up (MWd/t)	Average burn-up (MWd/to)	Pu charged (kg/GWa-el)	Plutonium burned (kg/GWa-el)	Ratio Pu-burned / Pu-charged (%)
50% Pu, Typ1 50% Th, Typ1	522 000 122 000	174 000	683	450	66
50% Pu, Typ1 50% Th, Typ2	495 000 112 000	200 500	1048	643	61
100% Pu, Typ2	428 000	---	2050	1020	50
100% Pu, Typ3	416 000	---	2093	968	46
50% Pu, Typ1 50% U	145 000 30 000	55 000	3725	503	14

4 MAXIMUM POWER SIZE – IKE STUDIES

IKE has adopted and actualised 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 (with 7.0 g U per pebble and 7.87% in U-235). The goal of investigations is the optimisation of power of a pebble bed HTR 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.
Further studies were performed by IKE for multiple number of fuel element passes through the pebble bed core with continuous reload of fuel elements and various core designs. Cylindrical core designs as well as annular cores with graphite pebbles in the central column and a solid inner column were considered. For all designs the important safety parameters like reactivity coefficients (moderator, fuel, water ingress) were calculated too. The core size, the power, the uranium load and the number of fuel element passes were varied and the corresponding equilibrium state and afterheat distribution for accident investigations were calculated by means of the ZIRKUS system. The maximum fuel temperature after a loss of coolant accident (DLOCA) was then determined for all variations. The results show the potential for raising the total thermal power using an annular core design regarding the feature that the maximum fuel temperature after DLOCA remains below 1620 ^{o}C and further safety, feasibility and economical constraints. The rise of the total power is limited by economical constraints and of course by fabrication limits of the RPV. Furthermore there is also a limit due to the maximum RPV temperature, which is expected during a DLOCA. Finally, also the core height is limited due to expected axial Xe-oscillations in very high HTR cores.

The maximisation of the power size of modular pebble bed HTRs 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 following table 9:

Table 9.Overview of major design parameters for selected modular high temperature reactor concepts.

Reactor	HTR-MODUL	PBMR-302	PBMR-400
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

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-302 concept 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 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 shows Fig. 4. 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 peak factors for such a power distribution is remarkably 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 as in the PBMR-302 concept. 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 reaches 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.

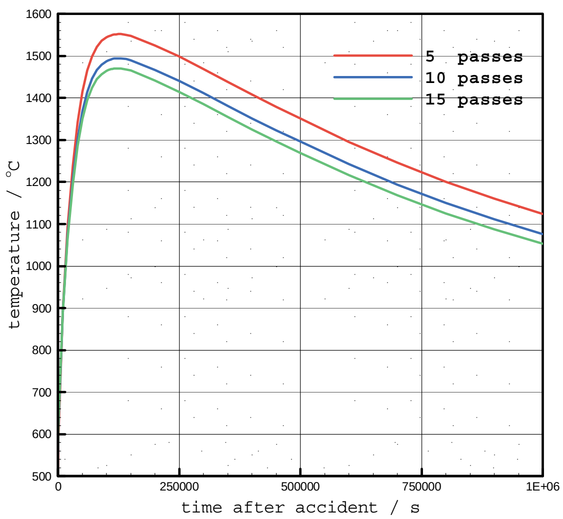

Figure 4: Maximum fuel temperature as a function of time after depressurisation for different recycle passes for MODUL (200 MWth)

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 (425 MW assumed), 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. 5), 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 inertia 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.

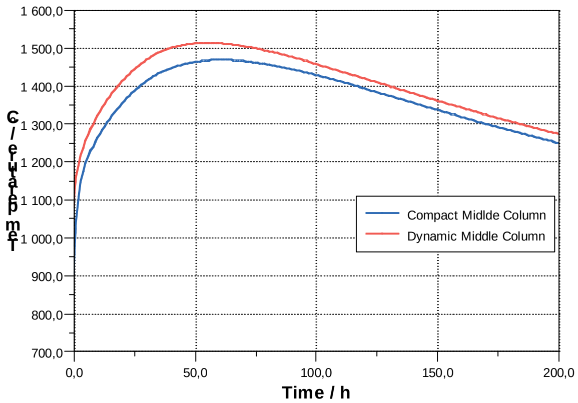

Figure 5: Comparison of the maximum fuel temperature versus time for designs with an annular core in the DLOCA case.

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. This can be observed from the development calculated for the design with dynamic middle column given in Fig. 6 which shows the gas flow and temperature distribution at different times. 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 sec. 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.

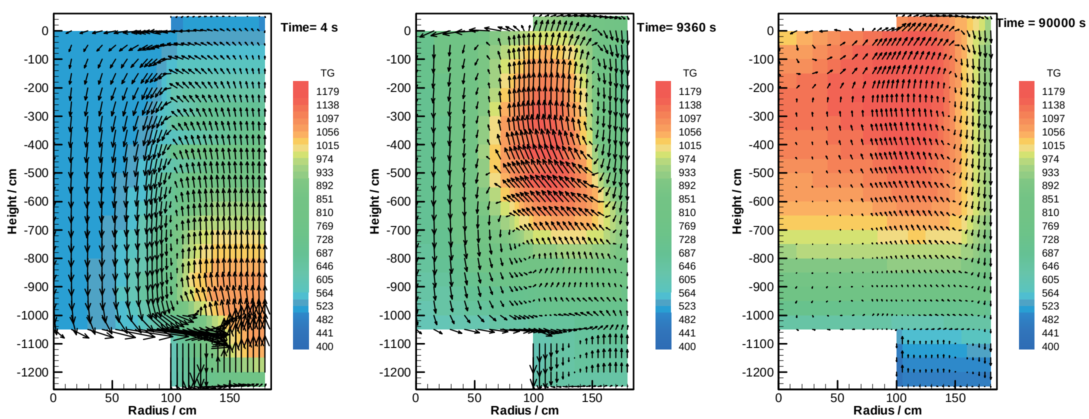

Figure 6: Gas velocity (arrows) and gas temperature (color shade) distributions at different times in the PLOCA case for the annular core design with dynamic middle column.

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 (see Fig. 6, middle), 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 (see Fig. 6, right). 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 can not be considered as optimum in terms of inherent safety. The mechanical stability of the Reactor Pressure Vessel e.g. can not be guaranteed if it is expand to too high temperatures for a long time (Fig. 7). At this point a second safety criterion is needed. A more optimised design with a thermal isolation of the reflector as shown in Fig. 7 (right).

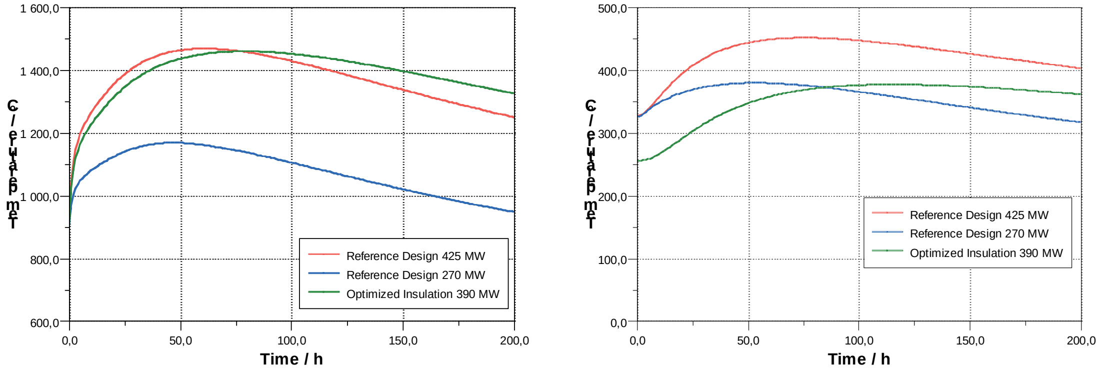

Figure 7: Comparison of maximum fuel temperature (left) and maximum RPV temperature (right) versus time for the reference geometry cases and for the case with optimized insulation.

The potential to maximise the thermal power with annular core designs while keeping the safety features has been investigated. Particularly, the potential of producing more power by increasing the diameter of the middle column is quantified. Additional technical complexities anticipated if this potential is tapped are mentioned and discussed.

Finally, it is investigated how the performance of a design is affected if an other safety criterion in addition to the maximal accident temperature of the fuel, e.g. the maximal accident temperature of the RPV, is included in the optimisation. The increase of core height will be limited by the pressure drop in core and by Xe oscillations, which excite power profiles, which can lead to high operational fuel temperatures.

5. CONCLUSIONS

A number of fuelling strategies for the continuous reloaded Pebble Bed Reactor were analyzed with respect to their capability to incinerate plutonium and minor actinides, while maintaining favorable safety characteristics. The investigations show quite promising results concerning the incineration (reduction) of especially first, but also of second generation plutonium. It should be noted, however, that only an indication of the favorable 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 HTR it was assumed that the fuel is able to withstand very high burn-ups, in the range of 700 MWd/kg 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.

Pu-incineration by means of the use of low-enriched uranium as driving fuel turns out to be an 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.

For a safe reactor design the behaviour of different design options for the modular pebble bed HTR during loss of forced cooling accidents has been analysed. It is shown, that if properly designed, the modular pebble bed HTR can be considered as inherent safe in the sense that the maximal fuel temperature never exceed limits during both pressurised and depressurised loss of coolant accidents.

The potential to maximise the thermal power with annular core designs while keeping the safety features is investigated. Particularly, the potential of producing more power by increasing the diameter of the middle column is quantified. Additional technical complexities anticipated if this potential is tapped are mentioned and discussed.
Finally, it is investigated how the performance of a design is affected if an other safety criterion in addition to the maximal accident temperature of the fuel, e.g. the maximal accident temperature of the RPV, is included in the optimisation. The increase of core height will be limited by the pressure drop in core and by Xe oscillations which excite power profiles which can lead to high operational fuel temperatures.

If the technical feasibility of solid inner columns is investigated in detail, such cores can lead to comparable high power of up to 400 MW without loss of the main inherent safety features.