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

HTR-N

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

Work Package: 3	**HTR-N project document No.:** HTR-N-04/10-D-3.0.0				**Q.A. level: 1**
Task: all	**Document type:** Deliverable				**Document status:** final
Issued by: NRG	**original document No.:** 20570/04.63015/P	**Annex pg.** 0	**Rev.** B	**Q.A. level** A	**Dissemination level:** PU

Document Title

Summary Report on Work Package 3 of HTR-N

Author(s)

J.B.M. de Haas[A)] (editor),
J.C. Kuijper[A)], J. Oppe[A)], X. Raepsaet[B)], F. Damian[B)],
H.J. Ruetten[C)],W. Bernnat[D)], J.L. Kloosterman[E)], J. Magill[F)], A. Negrini[G)].

Organisation, Country

[A)]NRG, The Netherlands
[B)]CEA, France, [C)]FZJ, Germany, [D)]IKE, Germany,
[E)]IRI/TUD,The Netherlands, [F)]ITU, Germany, [G)]Ansaldo, Italy.

Notes

Electronic filing: SINTER

Electronic file format: Microsoft Word 97

Rev.	Date	Short Description	Prepared by	Partners' Review	Control WP Manager	Approval Project Manager
1	01-12-2004	Final issue	JH			
0	15 11 2004	first issue	NRG J. de Haas	ANSALDO A. Negrini	NRG J. Kuijper	FZJ W. von Lensa

Chronology and history of revisions

Rev.	Date	Description
0	15 11 2004	First draft
1	01 12 2004	Final issue after review by partners and ANSALDO

DISTRIBUTION

EXTERNAL (SINTER database)

HTR-N Partners
HTR-N Coordinator (3 printed copies)

INTERNAL (NRG network disk)
NRG-DIR
R.J. Stol (1 printed copy)
A.M. Versteegh (1 printed copy)
NRG-PPT
A.I. van Heek (1 printed copy)
NRG-FAI
R.P.C. Schram
M.J. den Exter
K. Bakker
M.C. Duijvestijn
V.M. Smit-Groen
J.B.M. de Haas (1 printed copies)
R. Klein Meulekamp
J.C. Kuijper (3 printed copies)
D.F. da Cruz
A. Hogenbirk
A.J. Koning
S.C. van der Marck
J. Oppe
B.J. Pijlgroms
R.C.L. van der Stad

Distribution is in digital form (NRG network disk or SINTER database) unless stated otherwise.

HTR-N	Work Package: 3	HTR-N project document No.: HTR-N-04/10-D-3.0.0	Rev. 1
	Task: 3.2-4	Document type: EC deliverable	

Summary Report on Work Package 3 of HTR-N

J.B.M. de Haas[A)] (Editor),
J.C. Kuijper[A)], J. Oppe[A)], X. Raepsaet[B)], F. Damian[B)],
H.J. Ruetten[C)],W. Bernnat[D)], J.L. Kloosterman[E)], J. Magill[F)], A. Negrini[G)].

Petten, 18 October 2004

20570/04.63015/P

[A)]NRG, The Netherlands, [B)]CEA, France, [C)]FZJ, Germany,
[D)]IKE, Germany, [E)]IRI/TUD, The Netherlands, [F)]ITU, Germany, [G)]Ansaldo, Italy

Under the contract of the European Union (5th Framework Program) no. FIKI-CT-2000-00020

rev. no	date	description
B	01 12 2004	Final HTR-N report version
A	15 11 2004	1st draft

author : J.B.M. de Haas	JH	reviewed :	J.C. Kuijper
page(s) 30		approved :	R.P.C. Schram

TABLE OF CONTENTS

SUMMARY

The High-Temperature gas-cooled Reactor (HTR) is a promising concept for the next generation of nuclear power plants, and it is essential that validated analytical tools are available in the European nuclear community. This 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. In the “HTR-N” project analyses have been performed on a number of conceptual HTR designs, derived from reference pebble bed and hexagonal block type HTR types. It is shown that several HTR concepts are quite promising as systems for the incineration of plutonium and possibly minor actinides.

1. INTRODUCTION

The European Research and Development (R&D) activities on High Temperature gas-cooled Reactors (HTR) concentrate on HTR-related key technologies and innovation potentials with the objective to consolidate and advance modular HTR technology for industrial application in the next decade and to explore new applications like hydrogen production and waste transmutation in the long-term. A collaborative programme on different items like fuel, materials, components, licensing has been established within the European HTR Technology Network (HTR-TN). As part of the European Union Fifth Framework Program the “HTR-N” project [1] and the complementary activities in “HTR-N1” 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 report.

As the HTR 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 associated 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. In addition, it is necessary to identify the HTR-specific waste streams and to find measures for their minimization and treatment.

This report summarizes on the application of computational tools and models for the analysis of several HTR concepts with different fuel cycles. This is mainly centred around the two HTR systems currently in operation: the continuous reload (HTR-10) and the batch wise fuelled (HTTR) HTR types. The analyses are mainly concerned with the application of plutonium-based HTR fuel at very high burn-up, and also with a some more advanced HTR concepts dedicated to the incinerating of plutonium and minor actinides.

According to the HTR-N Q.A. guidelines a Peer Review has been provided for (draft versions of) most of the reports delivered in the work package.

This report is a summary of the reports 2-10 from the references, where all individual details and references can be found.

2. REFERENCE CORE AND FUEL DESIGNS

An important part of the activities within the “HTR-N” project was dedicated to the analyses, by the code systems, of several HTR concepts. These studies mainly concern the investigation and intercomparison of the plutonium and actinide burning capabilities of a number of HTR 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 HTRs 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 optimisation of the power size of pebble bed HTRs (employing an annular core geometry), the optimisation of burnable poison particle designs (mainly required for batch-loaded HTRs) and the more exotic concept of the spectrum transmitter.

2.1 REFERENCE FUEL DESIGNS

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 [2]. This common basis includes the definition of a reference pebble bed reactor (“flat bottom” “HTR-MODUL”) with continuous re-loading (“MEDUL”) of fuel elements and the definition of a reference hexagonal block type reactor (“GT-MHR”) Also for the fuel, the definition of a reference TRISO coated particle and the definition of reference first and second generation Pu composition.
A common feature for the pebble-bed and block type HTR designs is the use of coated particle (CP) fuel. Main parameters of the PuO_2-loaded CP fuel are given in table 2.1.1. Detailed information on other CP fuel types, which have been investigated in these studies, can be found in [2]. The initial isotopic composition of first and second generation Pu is presented in table 2.1.0.
The main parameters for the spherical HTR fuel element employed in the analyses are the following:

- Diameter of the fuel element: 6.0 cm
- Diameter of the central fuel zone: 5.0 cm
- Graphite density: 1.75 g/cm^3

The central 5 cm diameter fuel zone contains coated particles, embedded in a graphite matrix. The main parameters of the coated particles are as follows (table 2.1.1):

Table 2.1.0: Isotopic composition of first and second generation plutonium in these studies (weight %).

Isotope	1st generation (original) “A”	1st generation (alternative) “B”	2nd 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 2.1.1: 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

2.2 REFERENCE CONTINUOUS RELOAD HTR

The main dimensions and other parameters of the reference continuous reload pebble-bed reactor are presented in figure 2.2.1 and table 2.2.1. This reference reactor is a simplified version of the “HTR-MODUL” design [2], e.g. the conically 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.). 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.

Table 2.2.1: 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
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

As in the original HTR-MODUL design, in our analyses the “MEDUL” (German: “MEhrfach DUrchLauf” – multi-pass) fuelling strategy was assumed as well.

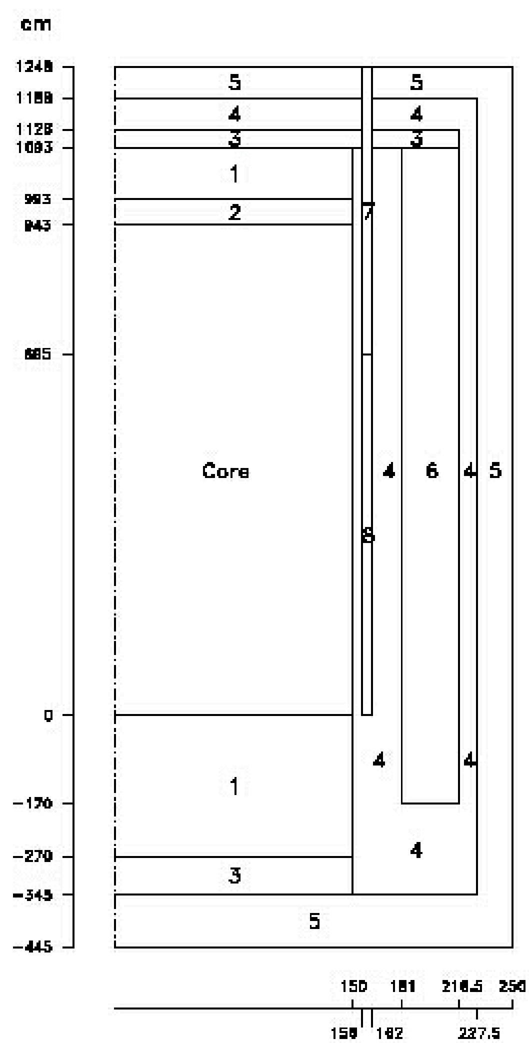

Figure 2.2.1: Main dimensions and material regions of the calculational model of the HTR-MODUL. Dimensions are in cm. The core is a random stacking of the well-known 6 cm fuel balls (‘pebbles’). The conical defuelling chute below the core is not modelled. Other (more or less homogenised) material regions in the model are: (1) reflector (graphite), (2) Helium plenum above pebble bed (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).

2.3 REFERENCE BATCH WISE RELOAD HTR

The investigation of fuel cycle studies for block type HTR 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 [2] are indicated in table 2.3.1 and figures 2.3.1, 2.3.2 and 2.3.3. 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 cm 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.

Table 2.3.1: General parameters of the GT-MHR-based reference reactor

Power	600 MWth
Thermal efficiency	48%
Loading factor	0.85
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

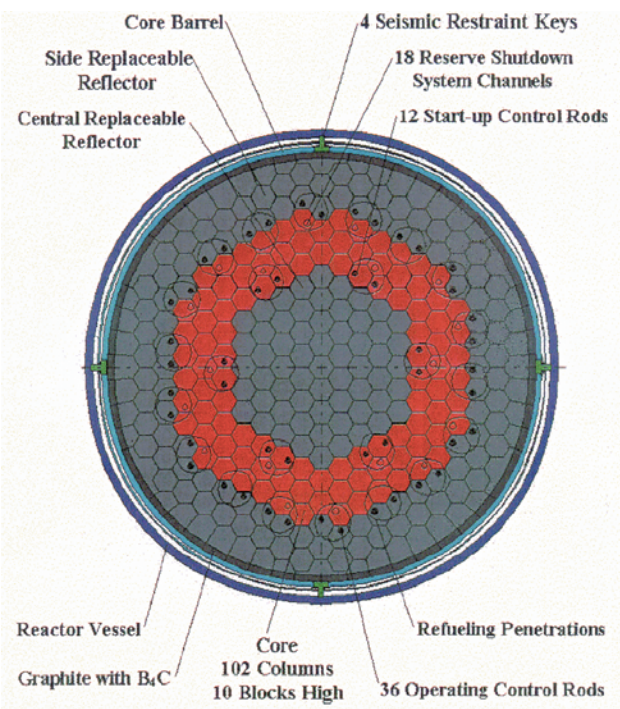

Figure 2.3.1: Geometry of the 600 MWth GT-MHR core

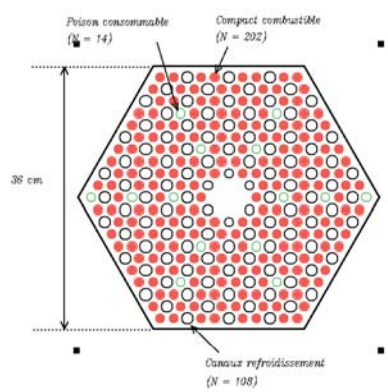

Figure 2.3.2: Geometry of a standard fuel element

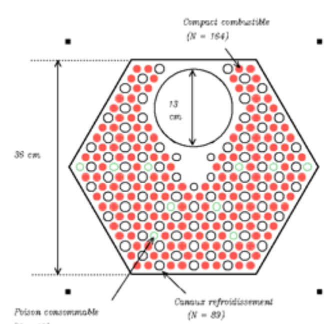

Figure 2.3.3: Geometry of a control fuel element

3. CONTINUOUS RELOAD PEBBLE BED TYPE HTR

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 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.

3.1 PLUTONUM INCINERATION CAPABILITY

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 [3] 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 3.1.1. 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)

Table 3.1.1: 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**

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 2.1.0). Further information can be found in [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, the 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 3.1.2). 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 (1st generation Pu, composition "A", see table 2.1.0). The assumed maximum attainable burn-up of this particle is 800 MWd/kg. The second pebble type (U/Th-FE) contains 20 g (HEU-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. Further details can be found in [4].

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 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.

Table 3.1.2: Fuelling strategy of the HTR-MODUL reactor for burning (1st 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 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 3.1.3: Mass Balances for the HTR-MODUL burning (1st 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	929	kg/GWa_{el}	--	615	kg/GWa_{el}	--
Pu discharged.	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

In table 3.1.2 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 3.1.3 displays the corresponding mass balance of the plutonium and of the fissile uranium. Detailed further information can be found in [4].

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 3.1.3, 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.

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 2.1.0). 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 3.1.4. 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 [2]. However, the results, as presented in table 3.1.4, 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 unfavorable 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 3.1.4: Comparison of different fuelling strategies for the incineration of second generation plutonium in pebble-bed HTRs.

Fuel elements	Heavy metal burnup (MWd/t)	Average burnup (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

3.2 MAXIMUM POWER SIZE

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 (7.0 g U per pebble and 7.87% U-235) [5]. 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.

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 3.2.1:

Table 3.2.1: Overview of major design parameters for selected modular HTR 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 is shown Fig. 3.2.1.

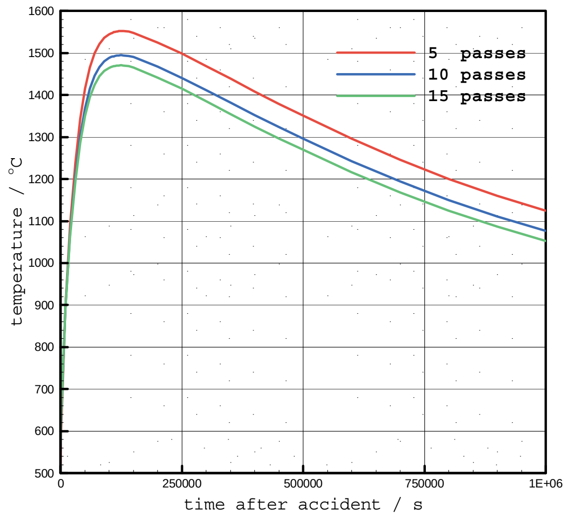

Figure 3.2.1: Maximum fuel temperature as a function of time after depressurisation for different recycle passes for MODUL (200 MWth).

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.

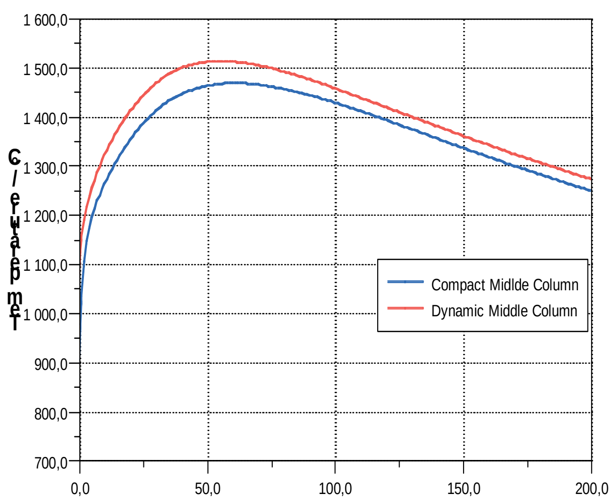

Figure 3.2.2: Comparison of maximum fuel temperature versus time for designs with annular core in the DLOCA case.

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. 3.2.2), 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.

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. 3 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 4s. 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.

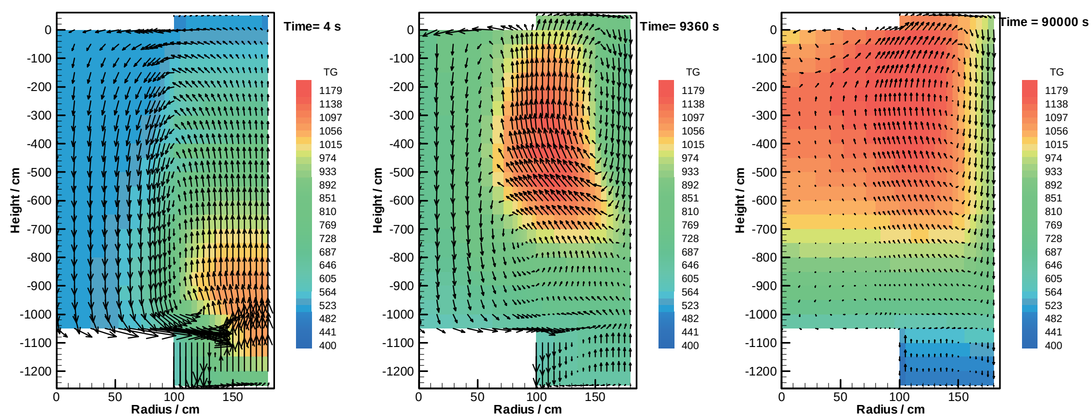

Figure 3.2.3: Gas velocity (arrows) and gas temperature (color shade) distributions at different times in the PLOCA case for the annular core design with dynamic middle column.

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. 3.2.3, 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. 3.2.3, 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 3.2.4). At this point a second safety criterion is needed. A more optimised design with a thermal isolation of the reflector is shown in Fig. 3.2.4 (right).

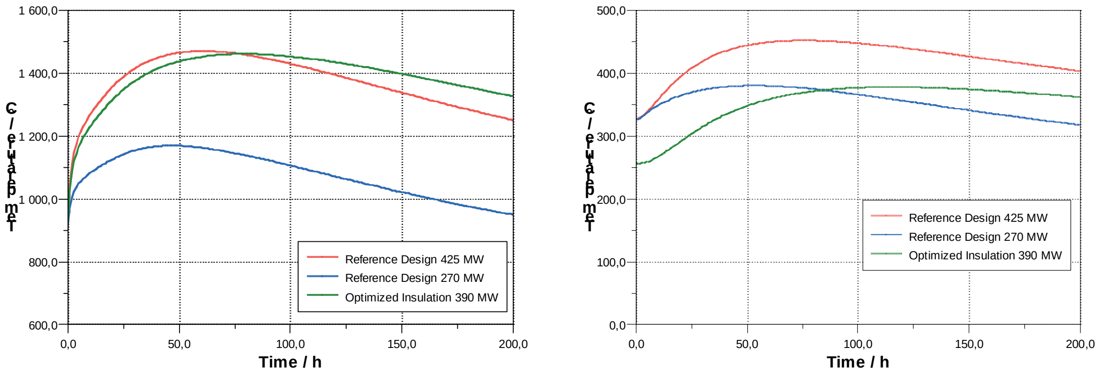

Figure 3.2.4: Comparison of maximum fuel temperature (left) and maximum RPV temperature (right) versus time for the reference geometry cases and for the case with optimized insulation.

The potential to maximise the thermal power with annular core designs while keeping the safety features 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.

4. BATCH-WISE RELOAD HEXAGONAL BLOCK TYPE HTR

For the GT-MHR-based reference reactor [2], 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 HTR loaded with plutonium fuels suppose an important optimisation 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 optimisation stage must be based on an equilibrium fuel cycle assuming a specific fuel-reshuffling scheme. For the present analysis, simplest methods than 3-D core burn-up calculations were employed. 2-D 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 2-D 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 2-D 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) HTR is the use of burnable poison, e.g. to flatten the reactivity-to-time behavior 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 HTR fuel types.

4.1 PLUTONIUM INCINERATION CAPABILITY

The investigation of fuel cycle studies for block type HTR cores was performed for 1st generation and 2nd generation plutonium based fuel cycles. Detailed information can be found in [6].

The calculations have been performed in fundamental mode (critical buckling), considering a linear and isotropic 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 2-D 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 volumetric leakage (3-D leakage) as 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_{eff} = 1.02) embracing the possible uncertainties.

Preliminary investigations showed that:

- 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. 4.1.1)
- The fuel cycle length increases linearly with the mass of plutonium loaded into the core; 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 the table 4.1.1. Whatever the plutonium isotopic content 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 as 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 as well.

By increasing the loaded fuel mass, the neutron spectrum becomes harder and favors the neutron absorption in the fertile isotopes. It should be noted that if the plutonium balance reaches an optimum with respect to the

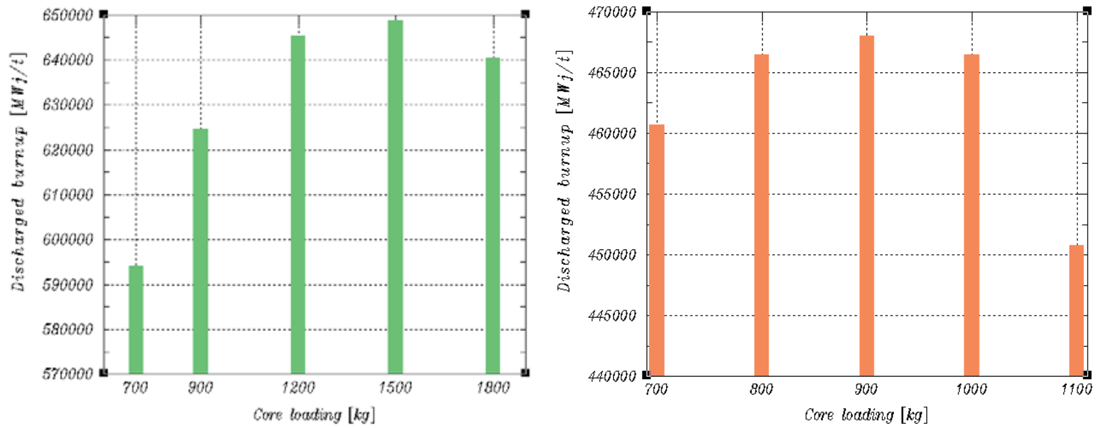

Figure 4.1.1: Dependence of burn-up on core loading.

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 **to maximize the burn-up should minimize both discharged masses of Pu and minor actinides**. In fact, as shown in table 4.1.1, the production of minor actinides raises continuously with the Pu-loading [7].

Consequently, the optimum burn-up obtained from the critical calculations, which leads to an optimum of the plutonium consumption with respect to the fuel loaded, can be explained as follow:

- Despite a smaller initial reactivity, the increase 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 notable 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 of a constant MA-discharge mass increase.

Table 4.1.1: Plutonium and minor actinides balance for 1st and 2nd generation plutonium fuel.

Type of fuel	1st 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 % of metal burnt	8.3	9.2	10.2	11.1	12.0

Type of fuel	2nd 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 % of metal burnt	16.3	18.3	20.5

Finally, as far as the 1st generation plutonium, the temperature coefficient (Doppler and moderator) has also been evaluated on the fuel element geometry (see table 4.1.2). The Doppler coefficient given in the table is an average value between 20 and 900°C. As far as the moderator (graphite) 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 of 1/3rd replacement per cycle 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 HTRs. Final conclusions on this application cannot be drawn yet, as the assembly based calculations do not provide for sufficient accuracy.

Table 4.1.2: Doppler and moderator temperature coefficient for the 1st generation Pu

	Doppler Coefficient [pcm/^{0}C]				
Burnup [GWd/t]	701 kg *without erbium*	900 kg	1200 kg	1500 kg	1800 kg
0	– 2,76	– 3,13	– 3,49	– 3,67	– 3,70
Variable	– 0,98 625 GWd/t	– 1,00 650 GWd/t	– 0,92 650 GWd/t	– 1,04 650 GWd/t	– 1,14 650 GWd/t
	Moderator temperature coefficient [pcm/^{0}C]				
Burnup [GWd/t]	701 kg *without erbium*	900 kg	1200 kg	1500 kg	1800 kg
0	– 2,29	– 2,14	– 1,91	– 1,69	– 1,46
Variable	+ 8,15 625 GWd/t	+ 6,33 650 GWd/t	+ 4,47 650 GWd/t	+ 2,86 650 GWd/t	+ 1,74 650 GWd/t

Prismatic block type HTRs 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 HTR 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/kbarn (a factor 2 with the common requirements) sustained by the fuel particles will be technologically feasible. The use of high burn-up plutonium particles cannot be regarded as proven technology today and an important fuel characterisation program, including irradiation, will be required to demonstrate that a burn-up equal 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 this the previous section. Indeed, the indicated mass balances have not been estimated from 3-D full core calculations and remain to be confirmed. Nevertheless, such a 3-D core calculation is inferred that a core optimisation 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 physics 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 HTR. 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 HTR.

4.2 OPTIMIZATION OF BURNABLE POISON DESIGN

A batch-wise fuel load scheme in HTRs 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 [8].
Such a batch-loading scheme in HTRs combined with BPPs has some attractive properties not offered by the continuous loading scheme. Burn-up calculations have been performed on a standard HTR fuel pebble with a radius of 3-cm containing 9 grams of enriched uranium or 1 gram 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 HTR by varying the radius of the BPP and the number of particles per fuel pebble.

With BPPs mixed in the fuel of an HTR, 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 HTR 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 5,000. 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 reference [8].

5. 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 the 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 [9]:

- 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 neutronics 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 HTR 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 HTR is not optimal for the fast island concept. In fact, since the minimum thickness of the fissile film is imposed by neutronics 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 HTR 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:

- Optimisation 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.

6. PEER REVIEW

The peer review of documents being issued under Task 3.5, as well as the definition of Q.A. requirements necessary for establishing a project framework of reference rules and criteria for carrying out the peer review activity itself. Since late 2002, a peer review document "HTR-N Project Peer Review" [10] has been initiated by Ansaldo by defining the general requirements and objectives of the peer review, as well as a reference list of HTR-N documents designated for reviewing and, possibly, for further investigations through alternative calculation analyses on selected cases of particular interest. The main criteria for peer review aim to verify the consistency of document reports between the different components relevant to the HTR-N Project, such as input data, methods, processes, as well as the reliability and traceability of contents and applications. A set of specific objectives, such as the inherent coherency between problem/purpose definition and scope of analyses, the referential correctness of data sources, the traceability of data and methodologies employed, the editorial completeness of the reports, etc., have been defined in order to enable the review to address recommendations and comments useful for ameliorating both document quality and general comprehension.

Fourteen reports, constituting a total of nine deliverables (table 6.1), have been reviewed, resulting in a number of suggestions and recommendations, which range from simple editorial or formal Q.A. aspects to a few issues concerning referential data sources and criteria/rationales, which should be at the bases of the HTR-N project. Adequate justification of the design grounds may be in fact considered of utmost importance for consolidating the data input on which the neutronics design and calculations of the HTR-N activity have been based and also for assuring the necessary traceability for any future knowledge and requirement for activities to be carried out in the field of HTR technology.

Table 6.1: Reviewed documents.

Item No.	Deliv. No. (Annex I)	Deliverable title (HTR-N contract Annex I)	Activity Task (RWP[1])	Doc. Type [2]	Document Title	Document No. - Revision	Date	Review status [3]	Main Review Result: Comments / Actions
1	3.0.0	Summary Report on Work Package 3	3.0.0	D (vs. S)	Revised Work Program for WP3: Physics Analysis of Different Fuel Cycles	HTR-N-01/6-D-3.0.1 Rev.5	20/6/2002	C	**Review completed on 4/6/2003.**
2	3.1.1	Report on selected input and output parameters and results for reference designs	3.1.4	D (vs. S)	Definition of common output parameters	HTR-N-01/02-D-3.1.5 Rev.1	20/6/2002	C	**Review completed on 4/6/2003**
3			3.1.6	D	Definition of common parameters	HTR-N-02/09-D-3.1.1 Rev.1	10/10/2002	C	**Review completed on 26/6/2003**
4			3.1.3 & 3.1.5	D (vs. S)	Definition of reference coated particles and plutonium compositions (1st and 2nd generation Pu)	HTR-N-02/06-D-3.1.4 Rev.0	25/06/2002	C	**Review completed on 1/7/2003**
5			3.1.1	D (or S ?)	Specification of HTR-MEDUL Core for different fuel cycles	HTR-N-00/10-D-3.1.2 Rev.? (old form used)	17/10/2000	C	**Review completed on 4/7/2003**
6			3.1.2	D (or S ?)	Fuel element physical parameters for Pu fuel cycles in a prismatic block type core	HTR-N-00/12-D-3.1.3 Rev.? (old form used)	26/04/2002	C	**Review completed on 7/7/2003**
7			N1/3.1	S	Plutonium (1st and 2nd generation) cell burnup benchmark specification	HTR-N1-02/06-S-3.1.1 Rev.1	7/10/2002	C	**Review completed on 7/7/2003**
8	3.0.0	Summary Report on Work Package 3	N/3.5.4		Summary Report on Work Package 3 of HTR-N	HTR-N-04/10-D-3.0.0	15/11/2004	C	**Review completed on 30/11/2004**
9	3.5.2	Summary report and inter-comparison of results	N/3.5.4	D	Physics Analyses of HTR Concepts with Different Fuel Cycles - Summary Report and Inter-comparison of Results	HTR-N-04/10-D-3.5.2 Rev.1	25/11/2004	C	**Review completed on 26/11/2004**
10	3.2.0	Summary report on core physics calculations on selected variations on the continuous reload design	N/3.2b.3	D	Studies on the Continuous Reload Pebble Bed HTRs - Summary Report	HTR-N-04/10-D-3.2.0	4/10/2004	C	**Review completed on 30/11/2004**
11	3.3.0	Summary report on core physics calculations on selected variations on the batch-wise reload design	N/3.3b.2	D		merged with HTR-N-04/10-D-3.0.0		C	**see document #8**
12	N1/3.1.1	Report on Pu cell burnup benchmark	N1/3.1	S	CEA's contribution to plutonium cell burnup benchmark problem	HTR-N1-03/04-S-3.1.1 –2, Rev.0	16/04/2003	C	**Review completed on 9/3/2004**
13	N1/3.3.1	Report on strategy study on LWR/HTR symbiosis	N1/	D					**Out of scope of this review**
14	3.3.3	Fuel cycles with burnable poison	N/3.3a.2	D	Design of burnable poison particles to reduce the reactivity swing in batch-wise fuelled High Temperature Reactors	HTR-N-03/05-D-3.3.3 Rev.0	28/05/2003	C	**Review completed on 17/6/2003**
15	N1/3.0.0	Summary report on Work Package 3	N1/	D		--			**Out of scope of this review**

7. CONCLUSIONS

A number of conceptual HTR designs were analyzed with respect to their capability to incinerate plutonium and minor actinides, while maintaining favorable 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 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 several HTR concepts 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.
-

For batch-wise (re-) loaded HTRs the use of burnable poison enables flattening of the reactivity-to-time behavior 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 behavior is quite well possible by varying the diameter of the particles and/or the number of particles per fuel element.

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.

A comparison of the Pu incinerating capacities of the different fueling strategies of the of the pebble bed and batch wise fuelled reactors can be seen in table 7.1, with the Pu-balance as Pu-burned/Pu-charged, showing a slght decrease in capability for the batch wise fuelled reactor because of the neutron consumption by the burnable poison.

Table 7.1: 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 burnup	750 MWd/kg	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 burnup	445 MWd/kg	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

REFERENCES

[1] "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.

[2] J.C. Kuijper et al., "*Definition of common parameters and reference HTR designs for HTR-N[1]*", HTR-N/N1 project report HTR-N-02/09-D-3.1.1, NRG, The Netherlands, September 2002.

[3] J.B.M. de Haas and J.C. Kuijper, *"Feasibility of Burning First and Second Generation Plutonium in Pebble Bed HTRs"*, HTR-N/N1 project report HTR-N-03/11-D-3.2.1, NRG, The Netherlands, November 2003.

[4] H.J. Ruetten and K.A. Haas, *"Incineration of LWR-Plutonium (1.Generation and 2. Generation) in a Modular HTR using Thorium-based Fuel"*, HTR-N/N1 project report HTR-N-02/05-D-3.2.2, FZJ, Germany, May 2002.

[5] W. Bernnat, M. Mattes, J. Keinert, N. Ben Said, *"U based fuel and maximum power size"*, HTR-N/N1 project report HTR-N-04/06-D-3.2.3, IKE, Germany, June 2004.

[6] F. Damian and X. Raepsaet, *"Feasibility of Burning First and Second Generation Plutonium in Prismatic Block HTRs"*, HTR-N/N1 project report HTR-N-04/07-D-3.3.1, CEA, France, July 2004.

[7] F. Damian and X. Raepsaet, *"Feasibility of Burning Minor Actinides in Prismatic Block HTRs"*, HTR-N/N1 project report HTR-N-04/07-D-3.3.2, CEA, France, July 2004.

[8] J.L. Kloosterman, *"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 HTR-N-03/05-D-3.3.3, IRI/TUD, The Netherlands, May 2003.

[9] J. Magill and P. Peerani, *"Improving the incineration of minor actinides in thermal reactors by generation of fast islands with thin fissile films"*, HTR-N/N1 project report HTR-N-01/02-D-3.1.4, JRC-ITU, Germany, March 2001.

[10] A. Negrini, "*HTR-N Project Peer Review*", HTR-N/N1 project report HTR-N-03/07-D-3.5.1, Ansaldo Energia, Italy, Final Version Novembert 2004.