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The High Temperature Reactor and its Fuel Cycle Options

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THE HIGH TEMPERATURE REACTOR AND ITS FUEL CYCLE OPTIONS

Prepared for INFCE WG-8

Contents

1. Introduction

The peaceful use of nuclear energy has been promoted by the Government of the Federal Republic of Germany since 1956. Today nuclear energy is being used commercially on the basis of the light water reactor technology. In this field efforts are now concentrated on the closing of the LWR fuel cycle. In addition, two advanced reactor systems are being developed: the High Temperature Reactor (HTR) and the sodium-cooled Fast Breeder Reactor (FBR). Prototype power plants of 300 MWe each are under construction at present.

The Light Water Reactor (LWR) operating with low enriched uranium (LEU) constitutes a widely established mature system which has proven its potential for economic energy production in large-scale commercial operation. By mid 1978 approximately 7 000 MWe of LWR capacity were installed in the Federal Republic of Germany and further capacity is under construction or planned for the future.

According to Germany's nuclear energy policy and the corresponding regulation provisions for a closed fuel cycle including waste management are a prerequisite to grant new nuclear power plant construction permits as well as for operating licenses for units in operation or under construction. This back end of the fuel cycle concept takes care of safety, security and environmental requirements especially important in a densely populated country like Germany as well as enables a considerable improvement in resource utilization. The corresponding provisions are also required for prototype and demonstration plants of new reactor systems.

The fuel cycle for the new reactor systems under development in the Federal Republic of Germany will be developed and be implemented according to the standards set by the LWR fuel concept. However, in the phase of introduction of advanced reactor systems interim storage may be necessary for some years.

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The German HTR development started as early as 1957, mainly concentrating on the pebble-bed reactor type and the thorium fuel cycle. Due to the use of ceramic materials in the reactor core and the use of helium as a coolant the HTR has achieved outlet temperatures of the coolant up to 950° C. The reason for the intensive Government support for this reactor line is mainly due to the following aspects:

- application of process heat for industrial processes
- high efficiency electricity generation and the possible use of a direct cycle, dry cooling towers combined with the utilization of waste heat for district heating
- reduced consumption of uranium from the use of thorium with relatively high conversion ratios due to recycle of U233.

This paper is intended to present the status of the HTR system in the Federal Republic of Germany as well as the consecutive steps and the probable cost of further development. The considerations are based on a recycling Th/highly enriched uranium (HEU) fuel cycle which has been chosen as the main line of the German HTR R&D efforts. Alternative fuel cycles such as medium-enriched uranium (MEU) and low-enriched uranium (LEU) are discussed as well.

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2. Aims of the HTR Development in the Federal Republic of Germany

In the Federal Republic of Germany the economic interest in the further development of the HTR is based on its two possible applications:

- process heat generation
- electricity generation

The generation of heat at a high temperature level is of special importance for the Federal Republic of Germany in terms of energy policy becauce this energy could be used efficiently for the gasification of coal. Coal is the only fossil energy source available in large amounts in the Federal Republic of Germany. The use of HTR process heat in coal gasification processes is meant to conserve the coal reserves by 4o % and to reduce the cost of the product as compared to fossil heating. Research and development in this field are aiming at

- efficient conversion of energy
- new applications of coal by substituting oil and natural gas

It is organized in the project PNP (Nuclear Process Heat Prototype Plant), a joint effort of all major industrial companies involved, at present, lead by the nuclear research center at Juelich. The main objective of this project is the establishment of a concept for a 5oo MWth pebble bed prototype plant consisting of two loops for the gasification of hard coal and lignite, respectively.

The significance of the HTR for electricity generation is based on a number of arguments. None of them for itself is as convincing as e.g. the fuel savings of the fast breeder or the process

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heat potential of the HTR. In combination, however, they are a sufficient incentive for the additional development work on the electricity generating HTR.
The advantages of a HTR power plant are

- high thermal efficiency and, consequently, a relatively low waste heat release to the environment due to the high cooling gas temperature

- the potential for the application of a direct cycle helium turbine (HHT) allowing for

  . the use of dry cooling with negligible efficiency losses,
  . the use of waste heat for district heating without reducing the electric power generated and
  . a further increase in the efficiency of up to 44 %

- reduced uranium requirements

- flexible fuel cycle in principle.

At the moment the activities are focused on the completion of the 3oo MWe- protoype power station. Further R&D-activities will be concentrated on a direct cycle reactor, the HHT project (High Temperature Reactor with Helium Turbine). The next step in this development should be a 67o MWe HHT-plant. However, the realisation of such a project requires active engagement of utilities to take into account market introduction aspects as early as possible.

The establishment of a complete HTR - fuel cycle requires the development of the following steps:

- fabrication of fresh fuel elements
- use of the fuel in the reactor
- transport of spent fuel elements

- interim storage of spent fuel elements
- reprocessing and waste treatment
- final waste storage
- refabrication of fuel elements from recycle fuel.

For the time being, fabrication facilities for fresh fuel elements are available in the Federal Republic of Germany on the scale required for the THTR 3oo. Experience in long term irradiation, transport and interim storage was gained with the AVR fuel elements. Reprocessing, waste solidification and refabrication were experimentally tested in hot cell runs on a laboratory scale. A pilot plant for reprocessing experiments (max. capacity 2oo kg/a) is under construction.

3. The Pebble Bed Reactor System

3.1 System Layout

The heat source of a HTR power plant is a graphite-moderated core with helium coolant in a downward flow. The pebble bed reactor is fuelled continuously on load with spherical fuel elements moving slowly from the top to the bottom of the core. Fig. 1 shows the prototype THTR 300 (300 MWe) which is now under construction. Depending on the application of the HTR as a steam cycle or a direct cycle plant for electricity generation or as a process heat reactor, the helium outlet temperature ranges between 750 and 950° C.

In HTR steam cycle plants the helium, heated to approximately 750° C, transfers its heat to a water/steam circuit as it flows upwards in the steam generators. The high coolant temperature enables the use of modern dry-steam turbines and results in plant efficiencies of 38 - 39 % with conventional wet cooling towers.

All components of the primary circuit are integrated in a prestressed concrete reactor vessel, PCRV. The vessel cavities are equipped with a gas-tight water-cooled steel liner. Pressure compensation is ensured by the concrete, prestressed by cables. A pressure-tight concrete reactor containment building encloses the PCRV and all systems holding radioactive gas under pressure. The reactor containment building provides protection against external impact and, in the event of loss of coolant, prevents the release of radioactivity to the environment.

A direct-cycle plant with coupling of the reactor to a gas turbine in the primary circuit, exploits fully the high gas temperature potentials of a HTR. The absence of a secondary circuit and the relatively small dimensions of the gas turbine enable a compact design integrating all components of the gas circuit in the PCRV. In this vessel the heat-exchanging

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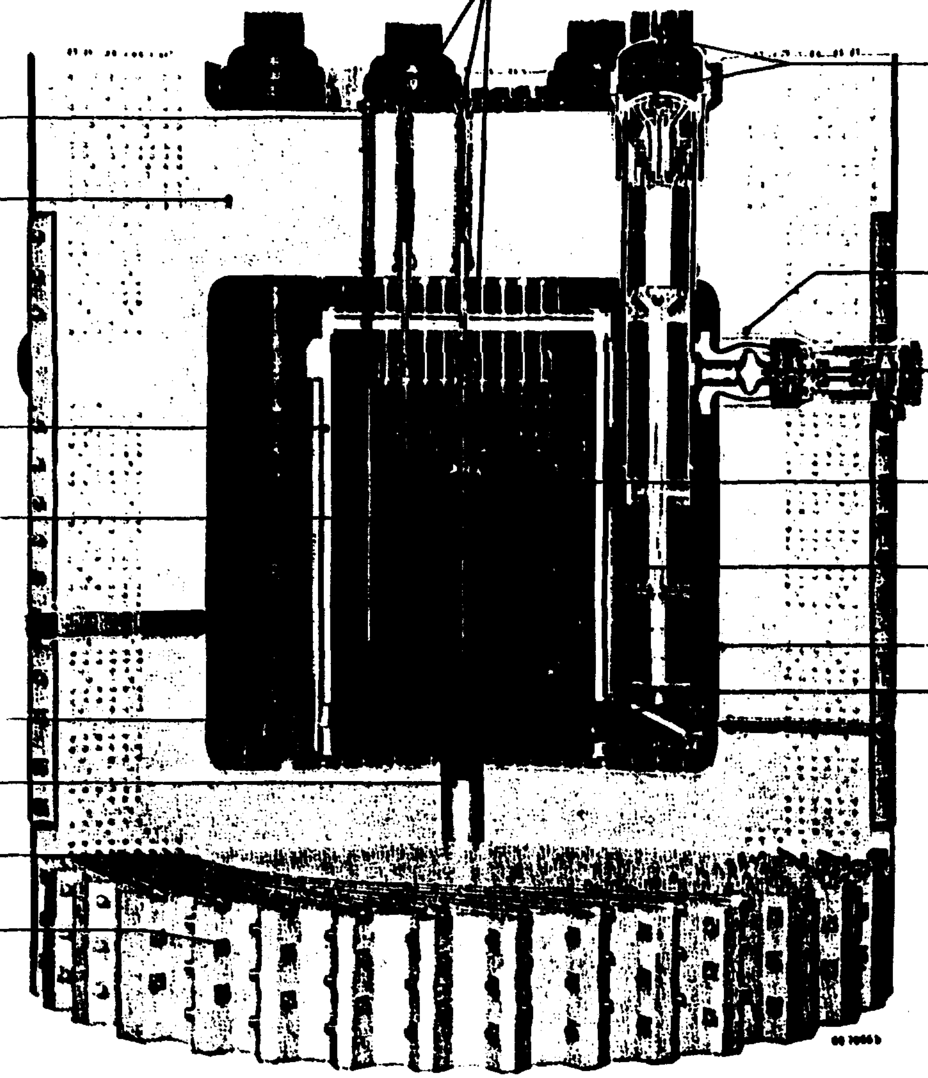

FIG. 3.1/1:
PROTOTYPE REACTOR THTR 300
(300 MWe)

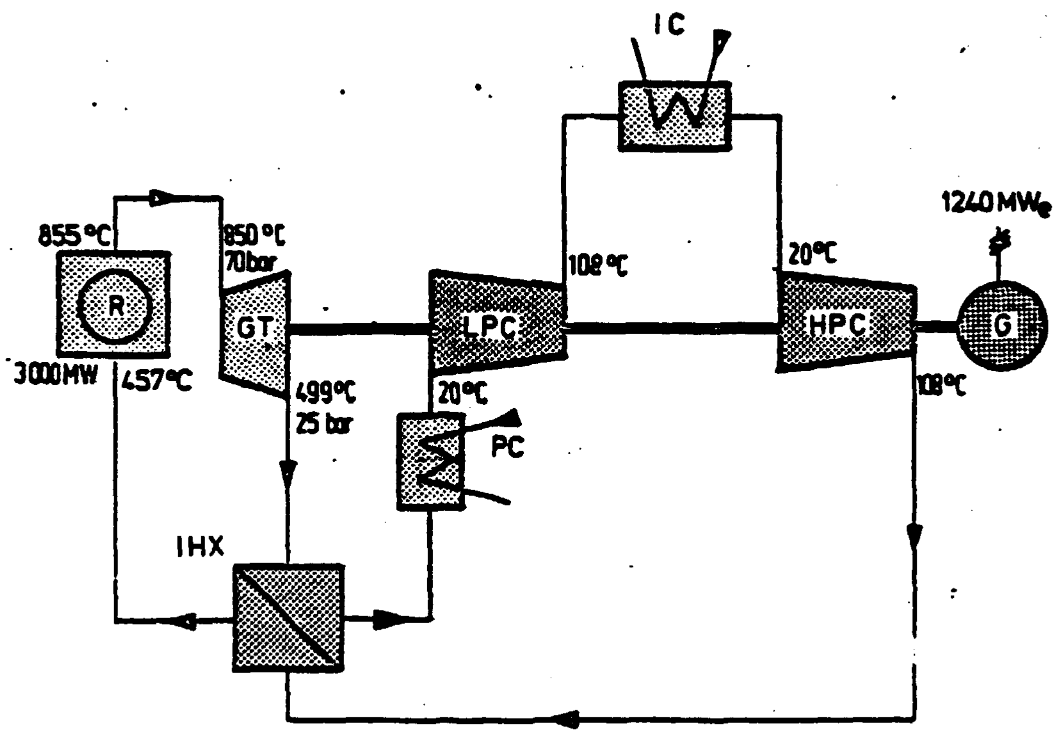

Figure 2: Principle layout of thermodynamic cycle of HTR gas turbine plant with a net efficiency of 41,4%

R	Reactor core	PC	Precooler
GT	Gas turbine	IC	Intermediate cooler
LPC	Low pressure compressor	IHX	Intermediate heat exchanger
HPC	High pressure compressor	G	Generator

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components are located in vertical cavities around the reactor core. The turboset is arranged in a horizontal cavity below the reactor core. The Figure 2 shows the main data of the thermodynamic layout. They apply to a 3 000 MWth plant which was chosen for the fuel cycle calculations in INFCE.

The waste heat in direct-cycle plants ranges between a temperature of 150° C and 20° C and can be rejected to the atmosphere in a dry-cooling tower. Only a relatively low cost penalty is expected compared to wet-cooling. A direct-cycle plant with dry-cooling and a gas temperature of 850° C has a plant efficiency of approximately 41 %. Due to its high temperature level, part of the waste heat may be utilized for district heating. It should be noted that the direct cycle plant allows combined heat and power generation without loss in electricity output.
Other design characteristics already mentioned for the steam cycle plant apply also to the direct-cycle plant, in particular the concrete reactor containment building and the independent auxiliary loops for the removal of decay heat. The direct-cycle plant possesses, however, one particular feature: the steel liner of the PCRV may be designed without a thermal barrier on its inner surface. This would considerably improve the accessibility of the vessel for inspection, maintenance and repair.

As a process heat reactor the HTR system can basically supply heat directly at temperatures up to 1 000° C for utilization in chemical processes. This opens a wide field of application in the energy conversion and chemical industries. The largest prospects for near-term realization lie in the gasification of hard coal and lignite. In a joint venture German coal producers and reactor manufacturers are developing a prototype HTR for nuclear coal gasification.
The prototype plant concept consists of a pebble-bed reactor with a power of 500 MWth. Via a He/He heat exchanger system one part of the reactor heat is fed into a plant for combined hard coal

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gasification (combination of hydrogasification and steam gasification), while the remaining part is fed into a lignite hydrogasification plant. The prototype also serves as a test-bed for parts of the long-distance energy supply concept. A mixture of methane and steam is split into carbon monoxide and hydrogen at the reactor using high-temperature heat. The recombination of the gases after pipeline transport to the consumer districts results in a recuperation of that heat.

The reference design for a station with an electrical output of 1240 MW (Fig. 2) is taken as the baseline in the fuel cycle studies for INFCE. The main reactor data (Table 1) used for the physics calculation correspond to the requirements, in particular with respect to the gas outlet temperature, for process heat application. The present concept for a direct cycle power plant foresees a hot gas temperature of 850° C and a slightly higher core power density of 5.5 MW/m³. In spite of these slight inconsistencies in some data pertaining to the electricity generating system and the process heat system respectively the main physics results are deemed representative for both systems. The principle conclusions with respect to fuel cycle parameters are valid for the various HTR applications.

THERMAL POWER	3 000 MW
POWER DENSITY	5 MW/m³
CORE INLET TEMPERATURE	250° C
CORE OUTLET TEMPERATURE	985° C
MAX. FUEL TEMPERATURE	1050° C
CORE DIAMETER	11.8 M
CORE HEIGHT (TOP OF BALL LAYER)	5.5 M
FUEL BALL DIAMETER	6.0 CM

TABLE 3.1/1: MAIN REACTOR DATA OF A PROCESS HEAT PLANT FOR PHYSICS CALCULATIONS

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3.2 Reactor Core

The design of the HTR core is in principle independent of its applications and the same applies to the whole fuel cycle. Two types of fuel elements have been developed for the HTR: spherical fuel elements of 6 cm diameters as used in the AVR and the THTR-300, and hexagonal blocks of 80 cm height and 36 cm width across flats as used in the Fort St. Vrain reactor in the U.S.

In the Federal Republic of Germany only the HTR concept with spherical fuel elements, is considered for future HTR projects. The reactor is fuelled continuously during power operation by inserting the fuel balls at the top of the core. The elements move slowly downward through the pebble bed by gravity and are extracted at the lower end. The discharge rate is adjusted to control the residence time in the core and simultaneously the fuel burn-up. Due to the continuous reload and discharge of fuel elements excess reactivity to compensate burn-up effects is not required in the pebble bed reactor. Hence, the insertion of absorber rods during power operation is necessary only for control purposes; and for shutdown. The absorber rods are inserted directly into the pebble bed, i.e. without guide tubes. Additional rods for control purposes may be provided in channels in the side reflector.

A particular feature of the present pebble bed core design is its fuel management scheme. The fuel elements pass only once through the core before being removed. This fuel management scheme is called "OTTO" (once-through-then-out). The term OTTO describes the fuel management in the reactor only and should not be confused with the once-through fuel cycle. According to the OTTO scheme only fresh fuel elements are loaded on the top of the core. Consequently the power production is considerably higher in the upper core layers. The downward flow of the coolant gas results in an effective heat transfer in the high power region without a concomitant rise in the fuel temperature. The temperature of the depleted fuel balls in the lower core regions is not markedly higher than the gas outlet temperature (Figure 3). The OTTO type reactor is well suited to achieve high gas outlet temperatures with relatively low fuel temperatures.

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The pebble bed fuel management operation is characterized by

- continuous refuelling and therefore minimum burnable poison for long term reactivity control,
- simple one pass ball flow (OTTO cycle) with a minimum of fuel element manipulations, and at the same time achieving high gas outlet temperatures,
- possibility to separate feed and breed fuel in different fuel elements and so simpler reprocessing and refabrication,
- possible use of graphite balls to adjust the moderation ratio.

Most of the proliferation aspects are determined by the characteristics of the fuel cycles. But for the pebble bed reactor system itself there are some aspects resulting from the fuel handling operations (OTTO) and equipment in the reactor which should be mentioned. There is only one possibility of loading the fuel elements to the core through a controlled loading channel and of extracting them from the core through a controlled discharge channel. There is no possibility to manipulate or divert fuel elements between these two points. Another feature bears on the possible abuse of the reactor by loading fuel elements with an abnormal composition to produce special nuclear materials. In a pebble bed reactor such a fuel element could not be identified during its residence in the core and special measurements would have to be performed to separate it out from the bulk of elements discharged. And finally, it is impossible to withdraw individual fuel elements earlier or later than determined by the normal pebble bed flow pattern in order to obtain fuel which, due to different irradiation time would have special isotopic compositions.

The safeguards systems take these individual aspects into consideration and appropriate systems are under development at the reactor.

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A number of favourable safety characteristics of the HTR system are practically independent of the choice of the fuel cycle and of the envisaged design and application of the plant. These characteristics result rather from HTR-specific plant features, particularly from the design of the reactor core.

In the event of a hypothetical failure of the safety systems, in spite of their multiple redundancy, a number of inherent system characteristics give the HTR a good safety behaviour. These characteristics are: the high heat capacity of the large graphite masses in the primary circuit, the low power density in the reactor core, the great safety margin between the fuel element operating temperatures and the failure temperature of the coated particles, and the negative temperature coefficient of reactivity existing under all operating conditions.

The above safety considerations refer to a thorium/high-enriched uranium core with a moderately high conversion ratio as in the reference cycle HRM.* It is not expected that the alternative cycles will change appreciably the safety and accident behaviour of the HTR. In the case of the high-conversion PB/NB* system some aspects such as

- the lower burn-up decreasing the fission gas pressure in the c.p. and lowering the probability of failure,
- the larger fraction of resonance absorption in Th in the harder neutron spectrum enhancing the effect of the negative Doppler coefficient of reactivity,

are likely to improve further the good safety characteristics.

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*) For definition of the various fuel cycles refer to chapter 4.

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3.3 Fuel Element

The actual fuel is a mixture of uranium and thorium oxide, which is encased in graphite as shown in Fig. 4. The use of graphite at high temperatures needs a noncorrosive coolant. Helium was found to be the most favourable of the various possibilities. The actual fuel element is a coated particle, which consists of an approx. 0.4 mm uranium/thorium-oxide kernel surrounded by a buffer zone of low density graphite. This zone is enclosed by a graphite coating, which is impermeable for fission products. The main function of the buffer zone is an elimination of mechanical contact between the fuel oxide kernel and the graphite coating. The coated particles thus obtained have a diameter of less than 1 mm. About 40 000 are pressed together with graphite powder to form a fuel ball, which has a diameter of 60 mm The outer zone of the graphite ball does not contain fuel. Depending on the composition of the oxide kernels, fuel cycles with low-enriched uranium and with uranium/thorium of varying degrees of enrichment can be realized with this same fuel element. More details are given in the following chapter on the fuel cycles.

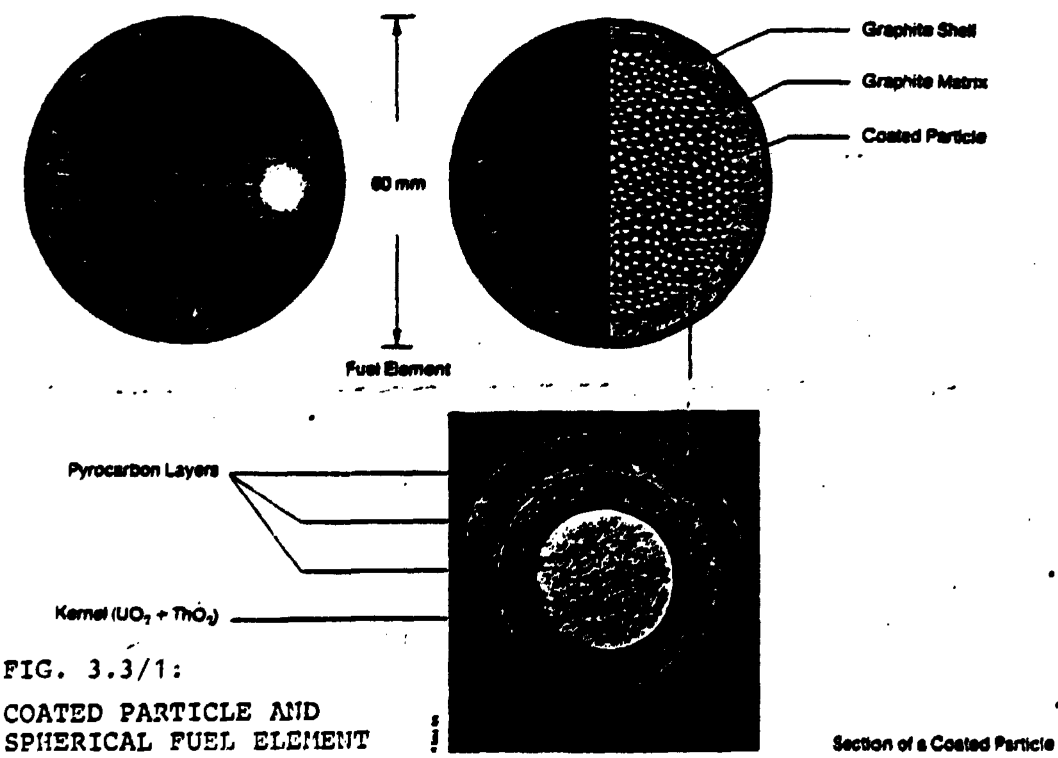

FIG. 3.3/1:
COATED PARTICLE AND SPHERICAL FUEL ELEMENT (FUEL BALL)

3.4 Technology Status of the HTR-System and R&D Requirements

Worldwide three experimental high temperature reactors, each of which has been successfully operated for some 10 years, have provided a broad basis of experience. However, the technology has not yet been demonstrated in a large scale commercial power plant. In the Federal Republic of Germany the experience is related to the operation of AVR and the construction of prototype THTR (Table 2).

Nuclear Power Plant	AVR	THTR
Overall Plant:		
Electric Power Net Output, MW	15	300
Recooling Mode	wet	dry
Reactor Core:		
Thermal Power, MW	46	750
Mean Power Density, MW/m³	2.6	6
Height/Diameter, m	2.47/3	6/5.6
Number of Fuel Elements	98000	675000
Helium Pressure, bar	10.9	39
Absorber Rods In-core/Reflector	-/4	42/36
Steam Generator:		
Number	1	6
Helium Inlet Temperature, °C	950	750
Superheated Steam Temperature, °C	505	550/535
Reactor Pressure Vessel:	Steel	Prestressed Concrete
Outer Diameter, m	5.8	24.8
External Height, m	24.9	25.5

TABLE 3.4/1: AVR AND THTR DESIGN DATA

The AVR is an experimental reactor with 15 MW electrical output and conceived as a large-scale experiment to

- demonstrate the feasibility of a HTR with a pebble bed core,
- perform functional testing of main components in particular the fuel element under HTR-specific conditions, and

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- demonstrate the safe and reliable operation of a HTR as a power plant.

The construction of the AVR was started in August 1961, and on December 17, 1967 electricity was first supplied into the public grid. The experimental reactor has demonstrated an extremely successful operation. The plant availability of the AVR averaged up to the end of 1976 reached 78 %. The specific coolant gas activity is approximately 0.9 Ci/MW-thermal which is very low in view of the 950° C coolant gas outlet temperature. Maximum burn-up values of about 185 000 MWd/t have been achieved. On average only one out of 10 000 circulated spheres was damaged.

The Thorium High Temperature Reactor (THTR) now under construction in Hamm/Uentrop is a prototype reactor with a steam cycle giving a net electrical power of 300 MW. The construction of the THTR is central to the further development of the HTR line.

The following aims are pursued by the prototype THTR:

- Gain experience with design, construction, and operation of a HTR-system in the 300 MWe - scale and its specific components. Proof of feasibility, reduction of first-of-its-kind uncertainties.
- Gain experience with licensing procedures for a nuclear power plant of this new reactor line.
- Demonstrate the anticipated HTR advantages under conditions of normal operation.

Contractual delivery time started on February 1, 1972 with a delivery time of 61 months. Commissioning of the plant is scheduled for 1981. A delay of several years has occurred mainly due to changes in the licensing requirements after the construction started.

Dry cooling has already been incorporated in the prototype THTR 300 which will demonstrate the operating characteristics of a large dry cooling tower equivalent to an electric plant with an output of 500 MWe. The cooling tower construction was completed in 1977.

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The further devélopment and planning activities in the HTR sector are based on the following essential boundary conditions:

- The nuclear heat generation system for the HHT (High Temperature Reactor with Helium Turbine) and PNP (Prototype Nuclear Process Heat) projects is a pebble bed core based on the experience gained with AVR and THTR 300.
- The underlying concept for power generation will be a direct cycle plant. The next development step will be a 670 MWe demonstration power plant.
- The nuclear process heat plant will be designed for lignite hydrogasification and hard coal gasification according to the combination process. It will have a power of 5oo MWth.
- With regard to the schedule, the construction of the power generation demonstration plant has priority over the process heat plant.

The HTR direct cycle demonstration plant will incorporate following design features:

- pebble bed core HTR in a prestressed concrete pressure vessel with a power of 1 600 MWth corresponding to approx. 670 MWe
- helium temperature at the gas turbine inlet: 850° C
- coolant pressure in the core: 70 bar
- integrated construction with single-shaft turbo set and dry cooling.

Because of the substantial support which is expected to come from the utilities during the commercialization phase the design has to be finally approved by the future operator of the plant.

According to the project partners the Nuclear Process Heat Prototype Plant (PNP) will have the following design characteristics:

- pebble bed core HTR in a prestressed concrete pressure vessel with a power of 500 MWth
- coolant outlet temperature at the core: 950° C

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- coolant pressure in the core: 40 bar
- integrated construction with heat exchangers in cavities in the prestressed concrete vessel
- two cycles for the gasification of hard coal and lignite with an additional system for the demonstration of nuclear long-distance energy

In any case successful commissioning and satisfactory operation of the THTR 300 prototype are prerequisite for the construction of these demonstration plants. The necessary support of future operators provided the construction could be started in the mid-80s. Successful operation of these demonstration plants could lead to the commercialization of the HTR system.

4. Fuel Cycles of the HTR System

Reactor Physics Considerations

Small neutron losses in the graphite moderator and the absence of neutron absorbing structural materials in the core of the High Temperature Reactor imply a good neutron economy. The potential of high conversion ratios is best exploited with recycling of the fissile material contained in discharged fuel.

Theoretically the performance of high converting systems may be extended to a conversion ratio of up to unity for the thorium/U233 fuel cycle. Due to various technical and economic constraints the present reference cycle has not aimed at utilizing this potential to the full but rather to couple a reasonably low uranium consumption (conversion ratios in the range of 74% for the HRM reference case considered here) with favourable economics. In principle various combinations of fissile and fertile materials can be used in the HTR.
However, in a fast neutron spectrum the neutron yield of U/Pu fuels is higher whereas in a thermal energy spectrum Th/U fuels are more favorable. This is caused in particular by the lower capture to fission probability in the thorium chain. The homogenized fuel distribution with a low resonance shielding increases the epithermal capture in U238 and Th232. The resonance absorption is more predominant in U238 and requires the compensating effect of increased moderation to maintain criticality. Typically HTRs with thorium fuel have a lower moderation ratio than with low enriched uranium fuel.

The fission cross sections for Pu 239 and Pu 241 are twice as high as for U233 leading to a better in-situ utilization of bred fissionable nuclides in the U/Pu cycle. In consequence the fissile content in discharged fuel is higher from the thorium cycle and the incentive to recover that material and recycle it is stronger. Closing the fuel cycle is of greater importance for thorium than for uranium fuel. In this operating mode the high fuel utilization of the HTR can be exploited fully. Immediate and sizeable benefits accrue from normal converter operation and with appropriate

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fuel development core layouts with higher thorium loading and shortened burn-up may increase the conversion ratio to 0.95 and above.

The coated particles, c.p., in themselves fully functioning fuel elements, are dispersed in the graphite fuel body. Changing the coated particle fraction in the graphite is a possible way to vary the moderation ratio in the core without altering the dimensions of the fuel elements. The helium coolant does not absorb or slow down neutrons so that thermo-dynamic considerations do not impact on the reactor physics. All this add to the possibilities of the designer to optimize the core for a given fuel cycle with negligible effect on the basic structural and thermodynamic design of the HTR core.

As a consequence of these features the spherical fuel element forms the basic module for all core designs and a transition between fuel cycles is in principle possible. The "switch-over" can be performed during power operations when due regard to control requirements and power distribution in the transition period is taken.

Fuel Technology of the HTR

The coated particle, the spherical fuel element and the fuel management in the reactor have already been mentioned in section 3. In addition to the above it must be emphasized that the fission product release is sufficiently prevented by the two layers of pyrolitic carbon (BISO coating) at moderate temperatures (T_{Helium} = 750° C) as to be used in power generating plants employing the steam cycle. The retention of the fission products in the particle is further increased by an additional interlayer of silicon carbide (TRISO coating). On the other hand reprocessing of TRISO coated particles is more difficult because SiC is not removed in the reprocessing head end by burning. In principle particles may be fabricated which contain either fissile material (feed particles) or fertile material (breed particles). The different kinds of particles may then either be put into the same spherical fuel element or into different ones leading in this case to feed elements and

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breed elements. The thorium-containing breed particle or breed element offers the possibility to be separated from the residual uranium and to obtain pure U233 in the course of reprocessing. This is needed for HTR systems with extremely high conversion ratios. Most experience in Germany is at present based on the mixed oxide particle $(Th,U)O_2$ (biso coated) which has been extensively used in the AVR. It is further used in the THTR 300 fuel element and was chosen for the reference fuel element HRM in this paper. With the only exception of a feed-breed separation in the high converter system, the conclusions of this paper refer to mixed oxide particles exclusively.

The production of fresh elements is technically established with fuel on the HEU basis for commercial operation. The first core of the prototype THTR 300 has been manufactured after this type of fuel element had previously undergone irradiation tests to fulfill the licensing requirements. Aspects of other fuel compositions are given in the following section on Fuel Cycle Technology Status and R&D Requirements. With fuel on the HEU basis the heavy metal loading in the element is a restricting factor with present fabrication technology which achieves a loading in the range of 15 to 20 g HM per fuel ball. This enables the layout of effective converter cycles. In the longterm high conversion system may be introduced requiring HM loadings of roughly 30 g. The appropriate technology is being developed and the first test batches have been manufactured.

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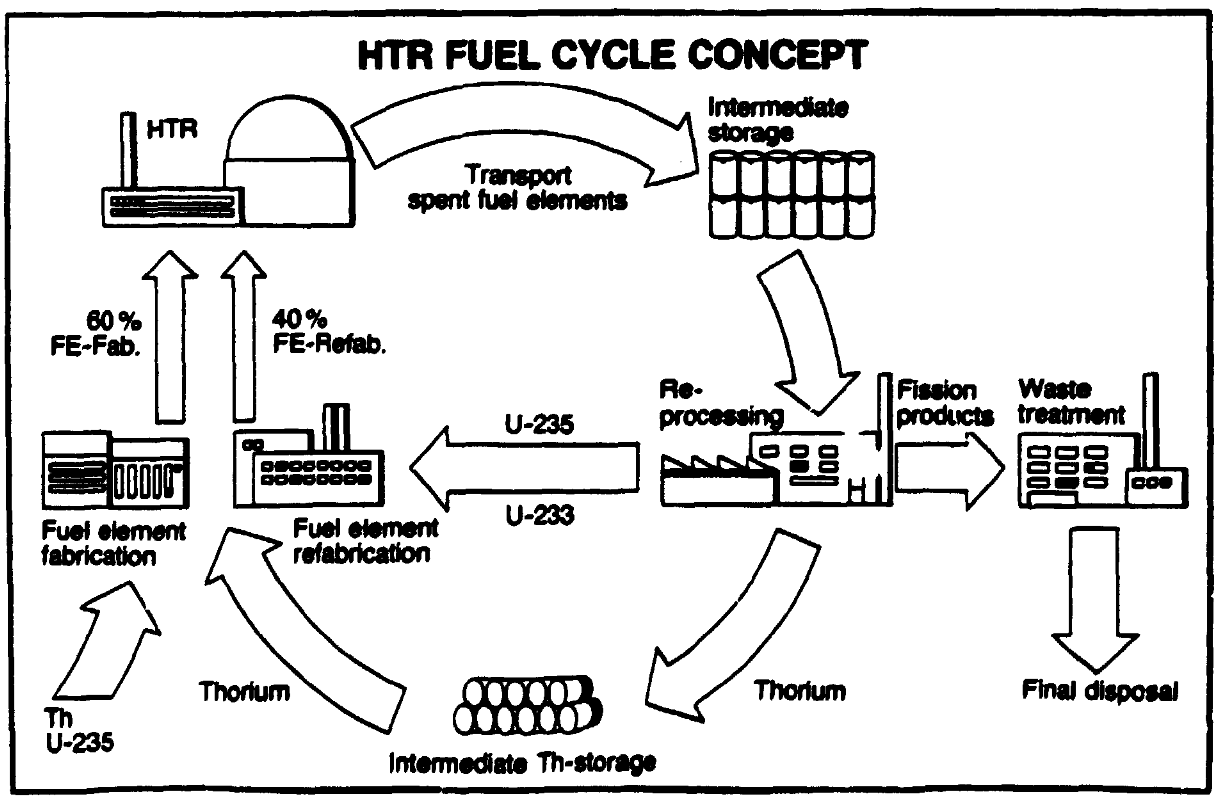
4.1 Closed Fuel Cycles

Recovery of fissile material from spent fuel and recycling of this material in reactors improves resource utilization. These improvements are dependent on the physical properties of the fuel used.

Reprocessing of HTR fuel consists of the following steps: Head-end removal of the graphite moderator and the coatings, dissolution of heavy metal, chemical separation of fission products and fuel material, waste management. In a closed fuel cycle this process is followed by the fabrication of new fuel elements where the recovered material is used.

Because of the physical properties of the HTR described above the closed thorium-cycle has advantageous features especially in connection with HEU. This is why this line was in the focus of development activities right from the beginning. The following graph shows the concept of the closed thorium-cycle:



In this paper the consideration of the closed HTR fuel cycles concentrates mainly on the Thorium/HEU cycle as this combines substantial uranium savings with favourable economics. Even in the thorium cycle the scope for alternative schemes is large and the recycling of bred fissile material must be discussed in the context of technical and economic consideration such as

- the frequency of recycling and the fraction of refabricated fuel elements
- the build-up of the parasitic absorber U236, originating from the U235 feed fuel and accumulating at high burn-ups.

The recycle strategies may deploy different fuel element types in the refuelling charges and the reprocessing handles either one common fuel stream or a separation of the various particle types into two or more streams. The separation of feed and breed material may take place on the fuel element level or with a segregation scheme with different coated particles in one and the same fuel element. The aim of separation is to isolate the residual feed fuel with its high U236 content from the fresh feed stream containing U233 or U235. The residual feed uranium may either be disposed of after one irradiation period or refabricated into special fuel for additional irradiation. The discarded U236 will always be accompanied by losses of fissile U235. In contrast the "eternal recycling" requires an increase in the fissile inventory to compensate for the accumulation of U236 in the core. The latter scheme with mixed oxide recycle implies U235 "losses" similar to the ones in the segregated cycles. In any case the separation in the reprocessing stage appears more complicated. The U236 problem will arise also for other thermal reactors and with the uranium cycle.

The Closed Reference Cycle and Advanced Converter Systems using HEU

The following fuel cycle calculations were performed on the basis of the 3000 MW_{th} base case described in Section 3.1 of this paper. The reference converter cycle investigated here is based on the mixed oxide concept. Similar performance data can be achieved with a feed/breed coated particle type fuel element.

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			Year of Operation				
			2	6 +)	13	20	25
Inventory	U235	kg/GWe	1,310	900	710	730	770
	U233	kg/GWe	790	1,200	1,495	1,550	1,550
	Fissile	kg/GWe	2,100	2,100	2,205	2,280	2,320
	Th232	kg/GWe	54,070	52,300	51,220	51,200	51,030
	U236	kg/GWe	255	426	635	663	752
Loading	U235-F	kg/GWe·a	510	183	201	223	237
	U235-R	kg/GWe·a	–	113	60	58	67
	U233	kg/GWe·a	–	165	245	281	291
Disloading	U235	kg/GWe·a	95	53	50	61	70
	U233	kg/GWe·a	112	181	225	258	280

+) Start of recycling; U235-F: fresh make-up, U235-R: recycled

TAB. 4.1/1: INVENTORY AND LOADING FOR THE HTR REFERENCE CYCLE DURING LIFETIME. ANNUAL OPERATION 7000 H.

In the light of enhanced emphasis on resource conservation the closed HTR reference fuel cycle HRM, High-enriched Recycling in Mixed oxide elements, shows an improvement in the uranium utilization over previous cycles. This is achieved with a higher thorium loading in the core, 22 g HM per fuel element and a reduction of the mean burnup from 100 to 80 MWd/kg HM. A heavy metal loading of 22 g seems to be possible in the light of the values of 15 - 20 g already achieved. All discharged uranium is recycled together with thorium in a mixed oxide particle. Therefore a U236 build-up in the core must be compensated for by a slight increase in the fissile inventory (Table 4.1/1). On the other hand the out-of-pile fuel management, and in particular the reprocessing with one stream only, is greatly simplified using mixed oxid as compared to fuel elements with separated feed/breed particles. As a point to note, the higher the conversion ratio the lower is the required U235 make-up fuel and consequently the production of U236. Over the lifetime the conversion ratio averages 0.74. Due to the difficulty of defining an equilibrium state for the HRM cycle the subsequent discussion will refer to typical lifetime average values as exemplified by the midlife loading and disloading vectors, year 13, in Table 4.1/2.

		HRM	PB	NB
Fuel residence time	days	1945	451	833
Moderation ratio	C/HM	160	198	110
Average feed fissile/HM	%	5,8	3,4	3,4
Average burnup	MWD/kg	80	23	24
Conversion ratio	-	0,74	0,74	0,97
Fissile inventory	kg/GWe	2200	1330	2800
Fissile reload	kg/GWe·a	635	1309	1281
Discharge U233+U235	kg/GWe·a	385	1015	1249
Removed U233+U235	kg/GWe·a	-	485	-

TABLE 4.1/2: CLOSED FUEL CYCLE PARAMETERS FOR NEAR EQUILIBRIUM CONDITIONS AND FULL-POWER-YEARS

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In the long term and with successful fuel development work a conversion ratio of near to 1. may be attained with a further increase in thorium loading and shortened burnup. Recent calculations indicate that this may be achieved by the following means:

1) Using U-233 - as pure as possible - as feed fissile material because of its high η-value
2) Employing a high heavy metal load in the fuel elements (32 g)
3) Employing a relatively low heavy metal burnup of 20 MWd/kg
4) Loading fuel elements containing only ThO_2-coated particles into a zone of 30 cm thickness adjacent to the side reflector. These serve as a radial blanket.

This, however, requires U233 for the start-up core and as make-up in subsequent reloadings. In a hard neutron spectrum U235 is much inferior to U233. The initial system inventory of U233 may come either from normal HTR converters or specially designed pre-breeders with separation of feed and breed fuel streams in the reprocessing. Conceivably, fast breeder reactors with Th-blanket could prove to be a suitable combination with HTRs.

The scheme proposed for a HTR high converting system is illustrated in Figure 4.1/1. Roughly half of the fuel balls in the prebreeder PB, containing thorium only are reprocessed separately and the recovered U233 from these elements is reserved for the nearbreeder NB. In the PB the uranium from the fresh mixed oxide elements is recycled once in special "feed" elements before being retired.

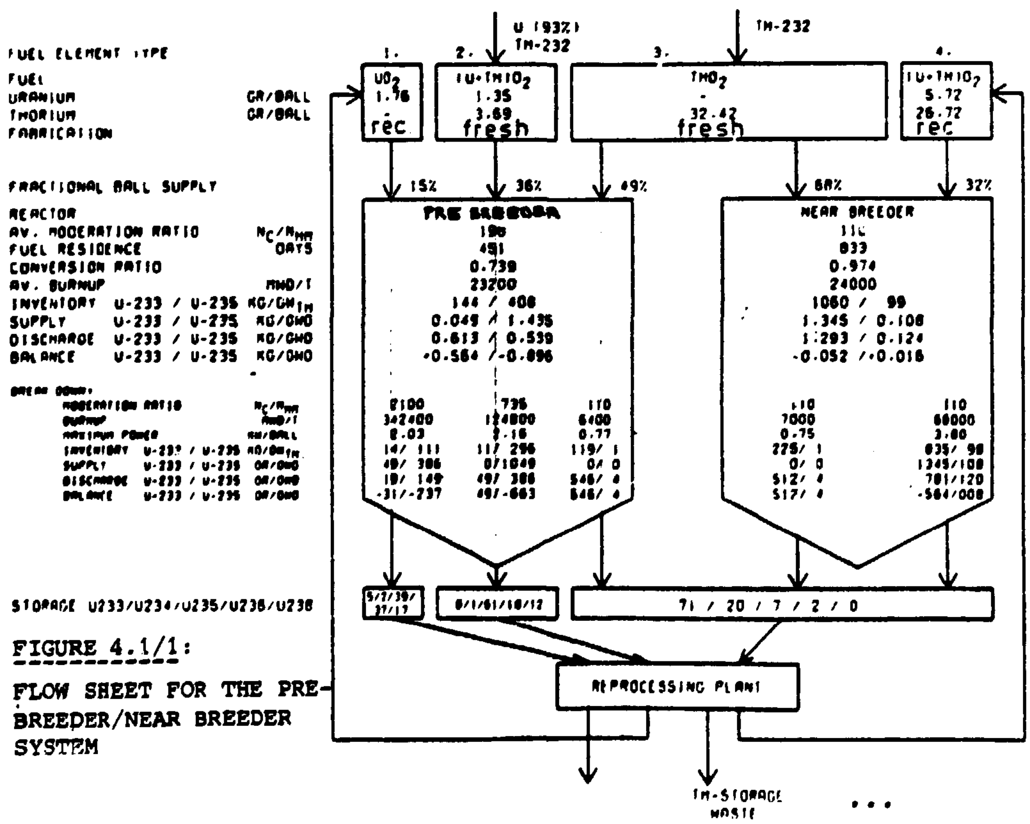

FIGURE 4.1/1:

FLOW SHEET FOR THE PREBREEDER/NEAR BREEDER SYSTEM

The fuel for the high converting system,NB, consists to 2/3 of breed elements with thorium only and to 1/3 of mixed-oxide feed elements with recycled bred uranium and U233 make-up from the PB. In this manner the NB calculated here achieves a conversion ratio of 0.97 at a mean burn-up of 24 MWd/kg HM. In equilibrium operation one PB produces enough U233 to serve 8 or 11 NB according to reprocessing losses of 2 or 1 % respectively.

The main problem with the combined PB/NB system lies in building up the initial fissile inventory.
The PB suggested here would have to operate 12 to 14 years to accumulate sufficient U233 to start up a NB and supply the reloads for the first three years. Thereafter the NB is virtually self-sufficient in fuel. When the first NB is closed down the system inventory becomes available for the start-up of the next in line of NBs. The effective uranium ore savings will depend very much on the reactor strategy and the growth rate of the energy system. The final choice of fuel cycle for HTR converters will be on the balance of arguments with respect to resource conservation and economics.

Closed Alternative Cycles Using MEU and LEU

The use of denatured uranium, 20 % enriched, in combination with thorium has not yet been analyzed in much detail for the closed cycle. A preliminary qualitative understanding may be gained from the results for the once-through operating mode.

Because this modification was suggested for reasons of proliferation resistance physically separable particle types may not be used. All fuel components have to be incorporated into the particles as mixed oxide. Due to the presence of U238 plutonium is formed in the neutron flux of the reactor. This can be separated from uranium by chemical methods. Th-232 is converted into Pa-233 which decays into U-233 with a half life of 27.4 d. Because of its relatively long half life also Pa-233 can be separated chemically if reprocessing is started immediately after the removal of the fuel from the reactor. Recovery of Pa-233 would, however, involve an unusually short cooling time and highly radioactive operations. Even under optimum conversion conditions which may not be attained with MEU the HTR needs additional fissile material in the closed fuel cycle.

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This fuel make-up normally has to be higher enriched material, because otherwise this cycle would drift towards a pure U/Pu cycle and this is not the original objective of the use of "denatured uranium". In a stationary cycle of this type the danger of proliferation has only been shifted to other parts of the fuel cycle and a more detailed investigation would have to clarify whether it is reduced considering the total system.

It has to be emphasized that in reprocessing the difficulties of the THOREX process are combined with those of the PUREX process.

The combination of Pu-feed with thorium is a promising alternative as the incineration of plutonium is coupled with the production of valuable U233. The important point in the economic evaluation is that the fissile plutonium substitutes expensive high-enriched uranium, i.e. the equivalence value of plutonium is high.

It has already been mentioned that the closed U/Pu cycle in the HTR is not as advantageous as the Th/U-233 cycle with regard to neutron reactor physics. In this case, however, the reprocessing technique of the LWR fuel could be adapted with the exception of the head end. The typical enrichment of the LEU fuel would be in the order of 7 to 11%.

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4.2 Once-Through Fuel Cycles

Open fuel cycles are economically only acceptable if high burn-up values are achieved in order to use the fuel as extensively as possible. Related to the power produced high burn-up values lead additionally to reduced fuel fabrication costs. Handling of spent fuel elements and storage over long periods as well as the final disposal of the spent fuel elements must yet be considered as a general problem with respect to environment, safety and cost. The burn-up is mainly limited by the change of reactivity due to the loss of fissile inventory and the neutron absorption by the fission products. Additional points to be mentioned are the long term behaviour of the fuel element under high neutron fluxes, high temperature and other conditions such as mechanical stress.
The practical experience gained with the AVR has shown that the spherical fuel element based on coated particle design can stand very high burn-up values (185 000 MWd/t HM).

In spite of the fact that various combinations of fissile and fertile material can be used in the reactor it should be mentioned that the main work of development has concentrated on thorium/high-enriched uranium fuel as the reference cycle for HTRs. In addition core studies and fuel testing have been performed for the low enriched uranium cycle. No experimental work has been done on denatured uranium fuel cycles. However, manufacture and irradiation experience of the mixed oxide HEU particles can be applied to a great extent.
The fuel cycles defined in Table 4.2/1 are based on mixed oxide coated particles. The parameters were chosen mainly in the view of low fuel cycle costs. The cycles are compared for the same basic core layout and with a discharge burn-up of 100,000 MWd/t. The core design is optimized with two fuelling zones, an inner and outer one to achieve radial power flattening and balanced burn-up. The values quoted are core averages. The same spherical fuel element is used for all cycles differing only in the type and volume loading of coated particles.

The reference cycle for the following considerations with thorium/high enriched uranium HEU employs a mixed oxide particle and a heavy metal loading per ball almost identical to the THTR fuel element. The THTR fuel element has a total weight of 205 g. It contains 1,03 g uranium (93% enrichment), 10,2 g thorium and 192 g carbon.

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The choice to use the THTR element is based on the fact that for once-through operation and high burnups the effect of variations in the moderation ratio is small.

		HEU	MEU	LEU
FUEL RESIDENCE TIME	DAYS	1217	872	1122
MODERATION RATIO	C/HM	325	458	366
AVERAGE FEED FISSILE/HM	%	7.2	7.8	8.5
AVERAGE BURNUP	MWD/KG	100	100	100
CONVERSION RATIO	-	0.60	0.58	0.58
FISSILE INVENTORY	KG/GW(E)	886	683	988
RELOAD U235	KG/GWA(E)	638	688	741
DISCHARGE U233 + U235	KG/GWA(E)	194	167	120
DISCHARGE Pu239 + PU241	KG/GWA(E)	1	19	81

Table 4.2/1: EQUILIBRIUM FUEL CYCLE PARAMETERS

The MEU cycle is a variant of the mixed oxide thorium cycle with the enrichment of feed uranium reduced from 93 to 20%. This cycle has been suggested in order to reduce the possibility to separate pure sensitive material in the course of reprocessing. The calculated MEU cycle employs a fuel element with a heavy metal loading between that of the present THTR and AVR elements, and containing coated particles with a U/Th ratio of approximately 2/3. The choice is extrapolated from the broad field of development work and irradiation tests with fuels of various enrichment degrees and thorium contents. A specific demonstration test will have to be performed.

The low enriched uranium/plutonium cycle (LEU) in the HTR has an average feed enrichment over the inner and outer core zones of 8.53% and the composition of discharged plutonium contains 44% of non-fissile Pu isotopes. The low enriched cycle can rely on extensive fuel development in particular in the UK, in the OECD Dragon Project and to some extent in the Federal Republic of Germany. The irradiation behaviour of the AVR-LEU fuel element was not satisfactory due to the choice of untypical parameters. Development of a reliable LEU fuel element for the AVR was in the past not treated with great urgency as it was not the main line (HEU) of R & D work. The LEU cycle for the pebble bed reactor is now intended to employ a fuel similar to a Dragon type coated particle

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and with a heavy metal loading somewhat lower than for the THTR. In spite of the positive experience in the UK, licensing requirements in this country may require modified irradiation and performance tests. Typically the U/Pu cycle in the HTR has a higher moderation ratio than the Th/U cycle to exploit the in-situ utilization of bred plutonium.

Details of the mass balance for the alternative fuel cycles HEU, MEU and LEU are given in Table 4.2/2. These figures also are relevant with regard to the proliferation aspects of the various fuel cycles.

	HEU	MEU	LEU
Inventory at Equilibrium . kg/GWe			
Pa233	67	37	-
U233	445	211	-
U235	438	417	736
Np239	0	2	6
Pu239	2	38	177
Pu241	1	17	74
Fissile	886	683	988
Loading: .... kg/GWe.a			
Th232	8119	5357	-
U235	638	688	741
U238	49	2787	7946
Uranium Enrichment	93 %	20 %	8,5 %
Discharge: .... kg/GWe.a			
Th232	7465	5000	-
U233+Pa233	165	117	-
U234	49	29	-
U235	30	50	120
U236	94	101	103
U238	38	2531	7368
Pu239+Np239	0.4	12	54
Pu240	0.3	12	34
Pu241	0.2	7	27
Pu242	0.7	17	30
U-fissile/U-total	0.52	0.06	0.02
Pu-fissile/Pu-total	0.36	0.39	0.56

Table 4.2/2: MASS BALANCE FOR ONCE-THROUGH CYCLES

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	HEU	MEU	LEU
U235 Depletion Ratio %	97	93	84
U233 In-situ Utilization %	74	67	-
Pu239+Pu241 In-situ Utilization %	-	95	89

TABLE 4.2/3: UTILIZATION OF FISSILE MATERIAL

In particular, the amount and composition of fissile material in the fuel elements has to be considered. In any case the diversion of fissile material from HTR fuel elements is complicated by the low densities of fissile material in the fuel elements. Consequently a very large number of fuel elements would have to be processed.

In a once-through operating mode the discharged fissile material is discarded. To achieve a good fuel economy a low core inventory together with a high depletion ratio of U235 feed and in-situ utilization of bred U233 and fissile plutonium becomes mandatory. The depletion ratio of feed material, defined as the ratio of U235 consumed to U235 supplied, increases with burn-up and a comparison between the alternative HTR cycles is shown in Table 4.2/3. The in-situ utilization of bred material is also a function of burn-up and to a lesser degree of the neutron spectrum.

In spite of these favourable aspects the problems of spent fuel management must not be ignored in the case of the once-through fuel cycle option.

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4.3 Resource Utilization

The discussion on fuel economy for closed cycles must distinguish between the gross demand accumulating over the operating lifetime of the plant and the net consumption. Fissile material is required for the inventory of the start-up core and for reloads to bridge the gap until discharged bred material is ready for recycling. The in-pile and out-of-pile inventory becomes available after the reactor has been shut down and may be used in a subsequent station. The gross demand is taken to mean the amount of uranium the utility will have to procure to operate the reactor. The net consumption takes account of the credit for the system inventories. In a way the net consumption reflects the impact the system has had on the natural resource uranium. To provide a comprehensive picture the whole operating lifetime time of 30 years at a 0.7 load factor has been included, i.e. for 21 full power years, and averaged to give specific figures per GWe·a. The tails assay is taken to 0.20 %. Uranium losses in conversion and fabrication amount to 1 %.
The calculation of the natural uranium credit for bred fissile material not recycled in the same reactor is based on the equivalence values

$$1 \text{ g U233} = 1.25 \text{ g U (93\% U235)} \mathrel{\hat{=}} 229 \text{ g } U_{nat}$$
$$1 \text{ g Pu-fis} = 0.9 \text{ g U (93\% U235)} \mathrel{\hat{=}} 165 \text{ g } U_{nat}$$

The results, Table 4.3/1 show that with a good converter like the HRM with recycle the gross demand is lowered by 40 % compared to that of once-through operation. Accounting for the fissile inventories of the system the net consumption figure of the closed cycle may improve the savings to 64 %. Nearly 3 times as much electricity is produced per t Unat consumed. Also the separative work requirements are lower for the closed cycles.

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System	Gross Demand		Net Consumption	
	tUnat/GWe·a	tSWU/GWe·a	tUnat/GWe·a	tSWU/GWe·a
Closed HRM	80	104	48	62
Once Through	134	174	134	174

TAB. 4.3/1: AVERAGE REQUIREMENTS FOR NATURAL URANIUM AND SEPARATIVE WORK PER GWE AND FULL-POWER-YEAR *FOR CLOSED AND OPEN THORIUM CYCLES.*

The introduction of a high-converting system of the near-breeder-type (NB) requires the availability of U233 and for this purpose pre-breeders (PB) must be operated. The NB has only an indirect demand for natural uranium so that the consumption of the PB will be apportioned to the number of NBs it can serve. The net uranium consumption of a PB shows Table 4.3/2.

The U233 output over the lifetime of the pre-breeder is sufficient to supply the net consumption of a system of 8.4 NB and hence to reduce the demand on the natural resource uranium by a factor of 7 to 9 compared to HTR once-through cycles. But it must be stressed that these favourable results apply only to a large system of PBs and NBs in equilibrium.

	Uranium Requirements to $U_{nat}/GW_e \cdot a$	Separative Work Requirements $tSWU/GW_e \cdot a$
PB gross consumption	202	262
PB net consumption	184	248
1 PB + 8.4 NB net consumption in equilibrium	20	25

TAB. 4.3/2: FUEL UTILIZATION IN HIGH CONVERTING HTRS

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During the introductional phase of the PB/NB systems the uranium demand is governed by the number of PBs and the time needed to build up the U233 inventory of NBs.

The amount of thorium needed for the HTR converters is typically 5 to 10 times less than the corresponding natural uranium demand. In high conversion cycles the loading of fertile material is increased and the thorium demand may rise to half that of uranium. The possibility of reusing the thorium after an appropriate decay period has not been considered here. Obviously, also for thorium cycles a resource problem is more likely to arise with uranium than with thorium.

In the discussion of fuel economy for once-through fuel cycles as with closed cycles the full lifetime of the reactor is chosen as the basis for comparison. The specific fissile inventory per net electric output is relatively small in the HTR and so are the amounts of discharged fissile materials. These characteristics are reflected in the requirements for uranium and separative work.

The gross natural uranium demand, i.e. without credit for recoverable fissile isotopes in the spent fuel, is given in Table 4.3/3.

System	Gross Demand	
	t Unat/GWe.a	t SWU/GWe.a
HTR - Th/HEU	134	174
HTR - Th/MEU	142	168
HTR - LEU	152	161

TAB. 4.3/3: AVERAGE REQUIREMENTS FOR NATURAL URANIUM AND SEPARATIVE WORK PER GWe AND FULL-POWER-YEAR AS AVERAGE OVER 21 FULL POWER YEARS FOR ONCE-THROUGH FUEL UTILIZATION

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For once-through cycles it can be concluded that the Th/HEU is most effective in decreasing the demand on natural uranium. The consumption of thorium is relatively small and accounts for 6 to 7 % of the uranium requirement. The separative work requirement is comparable to that of present thermal reactors on a once-through fuel cycle.

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4.4. Fuel Cost Evaluations

The following cost estimates apply to a reactor system which - provided the successful running of the prototype plant - will result in the operation of a power generating demonstration reactor by the end of this century. The market introduction of the HTR system will not take place before the next century. Closing the fuel cycle will also need a longer period of time.
It cannot be claimed that the following cost estimates represent a precise forecast of the costs of these systems. The estimates are rather intended to indicate relative cost differences between the various fuel cycle options.
The calculations were performed for German conditions with 1977 data. Prices commonly quoted in US$ have been transferred at an exchange ratio of 1 $ = 2.50 DM.

The reference design with a thermal output of 3000 MW and a gas turbine in a direct cycle is the basis for the cost evaluations. The start-up year in the calculation has fictiously been chosen to 1977. All costs during construction and operation are subject to inflation.

The uranium ore has been assumed to cost 30 $/lb$U_3O_8$ per 1.1.1977 and to escalate above the general inflation rate of 5 %/a with 7.5 %/a to reflect a real increase (2.4 %/a) in value due to a rising demand.

The following fuel cycle data are assumed:

Uranium ore	30 $/lb$U_3O_8$
Thorium	20 $/kg Th
Enrichment	110 $/SWU
Escalation	5 % p.a.
Interest & Discount rate	8 % p.a.
Tax on Fissile	2 % p.a.

The value of recycled material has been determined either from its specific reactor physics utilization in the same reactor or from credit calculation. In case a credit is given this is based on equivalent values for fissile plutonium and U233 when used in thermal reactors.

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The total out-of-pile time of fuel from discharge to reloading of refabricated elements is at present estimated to 2 1/4 years.

The fabrication costs valid for 1977 of fresh fuel have been extrapolated to commercial conditions from the experience gained from the manufacturing campaigns of more than 1 Mio elements for the AVR and THTR stations. Additional cost penalty for refabrication of hot recycle fuel must be assumed on the basis of rough estimations:

FRESH TH C.P.	200 DM/Kg HM
FRESH TH/U MOX C.P.	355 DM/Kg HM
FRESH HEU FEED C.P.	2550 DM/Kg HM
BALL PRESSING INCL: GRAPHITE	5.80 DM/FE
DUMMY BALL	3.54 DM/FE
TOTAL FABRICATION WITH 11 GHM/FE	885 DM/Kg HM
REFABRICATION PENALTY C.P.	+300 %
REFABRICATION PENALTY BALL	+100 %

Reprocessing and waste disposal are technologies for which commercial experience is still lacking. A recent German evaluation of a large reprocessing plant for LWR fuel gives indications of the corresponding HTR costs. The following values are related to the actual THTR-fuel element with 11 g heavy metal (HM) per ball and 100 000 MWd/T

SHIPPING	175 DM/Kg HM
HEAD-END	740 "
SOLVENT EXTRACTION & PRODUCT CONVERSION	600 "
OFF-GAS AND WASTE TREATMENT WASTE SHIPPING AND STORAGE	1070 "
TOTAL REPROCESSING	2585 DM/kg HM
SPENT FUEL STORAGE	2300 DM/kg HM

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The plant costs are the single most important factor in the economic comparison. The present estimates for a mature HTR system without "first-of-its-kind" expenditures expect the costs to be slightly higher than those of large LWRs, in particular when no credit is given for the dry cooling of the HTR power plant and for the potential to deliver some 400 MW heat for district heating. The following results of the cost estimates for HEU fuel cycles are related to the 1240 MWe reference plant. The choice of ~~the start-up year (1977)~~ is of no great importance to the findings. The uranium price is assumed to be 30 \$/lb U_3O_8 in 1977. Obviously, the future uranium ore price is central to the evaluation of high-converting systems.

Fuel cycle	Th/HEU once through	HRM (High-enriched Recycled in Mixed oxide elements)	PB (pre-breeder)	NB (near breeder)
FUEL INCL. CREDIT	0,96	0,59	0,51	0,51
FABRICATION	0,11	0,18	0,35	0,39
REPROCESSING AND DISPOSAL	-	0,23	0,57	0,46
SPENT FUEL STORAGE	0,25	-	-	-
TOTAL FUEL CYCLE COSTS	1,32	1,00	1,43	1,36

Fuel Cycle Costs (DPf/KWh) for thorium cycles in the first year of operation (1977)

The comparative calculations show that a fuel cycle with a moderately high conversion ratio of approx. 0.75, like the reference cycle HRM, commands superior economics over a wide range of assumptions. Compared to once-through fuel cycle operation this implies a cost saving of some 30 %.

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The high converting system PB/NB is 37 % more expensive than the reference HRM cycle. For the HTR such a cycle will only be considered if in future the saving of uranium ore and, possibly, to counteract a rapid increase in the uranium price will be the predominant aspect.

In general the fuel handling costs, fabrication, reprocessing, etc., effect the fuel cycle costs of the HTR to a lesser degree than expected from the specific costs per kg HM. But this is due to the high specific electricity generation from the fuel in this type of reactor with its high burnup and good thermal efficiency.

Obviously, for closed fuel cycles the costs are generally subject to greater uncertainties than for once-through cycles. This is mainly due to uncertainties regarding the costs of reprocessing and remote handling in the fuel element fabrication. It must be emphasized, that the cost assumptions for the different fuel cycles have been made as consistent as possible so that at least the relative comparison of fuel cycle costs should be valid. A few HTR-specific processes are, however, calculated with large uncertainty margins due to lack of commercial experience. The head-end stage in the reprocessing of HTR fuel is thought to be of particular cost significance. Although due consideration was given to contingencies in the basic estimates it is interesting to note that even an additional 100% in the head-end costs would increase the total contribution of reprocessing and disposal by only 30% (page 38). As a consequence the fuel cycle cost for the reference cycle would rise by 6%.
A similar argument is valid for the refabrication costs.

For the once-through-fuel-cycles the results are:

	LEU	MEU	HEU
FUEL WITH FINANCING AND TAX	0.93	0.93	0.96
FABRICATION	0.12	0.13	0.11
SPENT FUEL STORAGE	0.27	0.34	0.25
TOTAL FCC	1.32	1.40	1.32

They were calculated under the same assumptions as the above HEU fuel cycle costs and are given in DPf/KWh for the start-up year (1977).

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For the once-through cases, the results show that the Th/HEU and LEU fuel cycles have the most favourable fuel cycle costs although the differences are not large.

The main uncertainty in these cost estimates relating to once-through fuel operation are the expenses incurred from spent fuel storage. As mentioned above it is interesting to note that in spite of the costs of handling HTR fuel in the fabrication and storage stages are relatively high per unit of heavy metal the resulting effect on electricity costs is modest.

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4.5 Fuel Cycle Technology Status and R&D Requirements

The complete HTR fuel cycle consists of the following stages

- fabrication of fresh fuel elements
- deployment of the fuel in the reactor
- transport of spent fuel elements
- interim storage of spent fuel elements
- reprocessing and waste treatment
- final waste storage
- fabrication of coated particles and fuel elements from recycled fuel (refabrication)

Of these operations all but reprocessing, waste disposal and refabrication have been successfully developed during the past 2 decades mainly with fuel on the HEU basis. The once-through fuel cycles in the HTR with common fuel must therefore be regarded as technically available, except the problems arising from spent fuel storage. About 1 million spherical fuel elements for AVR and THTR containing about 25 billion coated particles were fabricated. A number of individual fuel elements, containing especially mixed oxide fuel, mixed carbide fuel, low enriched UO_2 fuel, separate feed UC_2 and breed ThO_2 particles were tested under normal and transient conditions in test reactors (e.g. R2-Studsvik). In addition, large numbers of fuel elements have been inserted in the AVR reactor. Coatings of pyrolytic carbon (BISO) and carbon plus SiC (TRISO) have been developed to provide good fission product retention. In the AVR, a large number of fuel elements has reached a burn-up of already 185 000 MWd/t HM without failure. A compilation of the various fuel types loaded into the core of the AVR test reactor is given in the following Table:

Number of Fuel Elements	Type of Fuel	Content (g/ball) Th	Content (g/ball) U-235	Enrichment %
88 000	$(U/Th)C_2$	5	1	93
73 000	$(U/Th)O_2$	5	1	93
20 000	$(U/Th)O_2$	10	1	93
12 000	UC_2		1	93
	ThO_2	5		
3 000	UO_2		1	93
	ThO_2	10		
2 400	UO_2	-	1,4	7

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As indicated by the enrichment value in the last column the uranium used was in most cases of the HEU type. Mostly the fuel elements contained mixed oxide particles, e.g. $(U/Th)O_2$, sometimes separated feed breed particles, e.g. UC_2, THO_2, were used.

Low-enriched uranium fuel particles could rely on established fabrication techniques, however, an irradiation and demonstration programme would be required to introduce the LEU cycle in the pebble bed reactor. But the utilization of LEU in the HTR can be considered as feasible taking into account the experience gained with the DRAGON project.

Little work has been done on denatured uranium thorium fuel. A full long-term R+D programme would be necessary to establish reliable MEU fuel elements. A minimum of 5 years have to be scheduled for this programme under optimum conditions.

The quality of the BISO coating is completely sufficient for HTR power plants with steam cycle. For HHT and process heat reactors, however, TRISO coated fuel will probably be necessary for fission product retention to enable the maintenance of reformer tubes in a large process heat plant and of the gas turbine in a direct cycle HTR. Work on advanced particle concepts and on fuel elements with high heavy metal loading is under way. With the AVR and THTR experience and the current R+D program it is ensured that satisfactory Th/HEU fuel elements can be specified and fabricated also for direct cycle HTR plants and process heat HTRs.

As part of a long term programme fuel elements with a high volume fraction of coated particles (more than 20g HM/fuel ball) are being developed. In particular high conversion systems require a somewhat different fabrication technology to achieve the higher heavy metal loading per element. A number of small batches have been produced and samples tested.

After discharge from the reactor the fuel is stored on site. Until reprocessing is established the spent fuel can after 2 years be taken to an intermediate long term storage facility. The technical concept and the preliminary plan for an intermediate surface storage bunker for spent fuel elements from AVR and THTR has been established. The capacity of the storage facility is dimensioned

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to hold approximately 2.5 Million elements equivalent to 15 years of operation. The decay heat of the fuel elements will be dissipated by natural convection with air. A preliminary safety report has been submitted together with the first conceptional design. From some technical points of view the fuel elements may be taken to final storage without further treatment. The coatings and the graphite matrix prevent fission product leakage and protect the kernels effectively against chemical attack.

The reprocessing of thorium/uranium fuel consists of the three basic steps; head-end with removal of the graphite moderator, chemical separation of fission products and heavy metals, waste treatment and disposal. The head-end demands particular attention and is a separate development needed for all HTRs fuel cycles. The THOREX process is chemically analogous to the already well developed PUREX process and is supported by considerable research experience. The problems with waste treatment and disposal are regarded similar to the ones arising in the uranium/plutonium cycle for LWRs and FBRs.

Experience has been gained in cold and hot laboratory and in pilot plant scale reprocessing experiments with spherical HTR fuel elements. These tests were intended for the development of the Thorex flowsheet for high burn-up fuel. The following topics were investigated.

- burn-leach head-end process for the removal of graphite and pyrocarbon
- aspects and variation of the dissolver
- modified flowsheet for solvent extraction to be applicable to high burn-up fuel
- treatment for burner off-gas and dissolver off-gas
- tail-end treatment for the heavy metal products
- special systems for process control and nuclear materials accounting.

Following successful cold and hot laboratory scale testing of all reprocessing steps at Jülich, a reprocessing pilot plant with a capacity of 2 kg heavy metals per day, JUPITER (JUelich PIlot facility for Thorium Element Reprocessing) is now being constructed. The entrance cell and the head-end cell with the hammer mill and fluidized-bed burner are complete; installation of the chemical processing cell, which is the most complex part, is due to follow. For the MEU cycle the flow scheme for the combined reprocessing of thorium/uranium/plutonium mixed oxide fuel is at present not available.

In the case of LEU fuel elements, the same head-end has to be developed as for HEU and MEU fuel elements. The successive step of chemical separation can be based on the well-established Purex process. Due to the possibly higher burnup and the higher enrichment of the HTR fuel elements, the flow scheme must be expected to be somewhat different from that used in LWR reprocessing. Since thorium is not used in the fuel, the recycled uranium would not contain U-232 and could be handled like fresh uranium.

The waste treatment for the reprocessing pilot plant JUPITER has already been developed. Small plants FIPS I (FIssion Product Solidification) and PAMELA with throughputs of approx. 1 kg glass/h have been put into hot operation. Active waste was transformed to borosilicate blocks in FIPS I and to glass pearls and glass blocks in PAMELA for further embedding into a metal matrix. The experimental programme was finished end of 1976 and investigations of the componets (corrosion effects) are in progress. Improved facilities with higher throughputs and for continuous operation are under construction.

After successful operation of JUPITER a prototype facility for reprocessing and refabrication would be designed and built in the 1990's. The time scale for the likely market introduction of the HTR requires commercial recycling to be available after 2000. The R&D programmes in the fuel cycle can meet this date.

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A characteristic contaminant of the Th/U-233 fuel cycle is U-232. While U-232 itself does not emit troublesome radiation, some of its daughter products mainly Tl-208 do. U-232 cannot be separated from the residual uranium by chemical methods. In contrast to natural uranium and to uranium separated from the U/Pu cycle this type of uranium causes a hard -radiation. In the case of large technical throughputs therefore all steps needed for the refabrication of fuel elements must be performed remotely. R&D work in this field was directed to the remote fabrication of fuel kernels material. Investigation has been concentrated on reliable kernel fabrication according to modified flow sheets adaptable to remote operation and maintenance. Remote refabrication is needed for any reprocessed uranium from thorium-bearing fuel, including HEU and MEU as well. Only the use of LEU which is free from thorium would avoid this complication.

Accountancy, containment and surveillance are the basic tools to prevent diversion of special material from the fuel cycle activities. For the thorium cycle procedures and equipment are being developed taking into account the specific radiation characteristics of these cycles. The procedures developed are based on active neutron interrogation. Though they are still in a formative stage they are not disturbed by high intensity γ-radiation (100 Ci fission products) and can be expected to evolve further with increasing experience.

5. Summary and Conclusions

This paper is an outline of the characteristic properties of the pebble-bed HTR system on which the German development activities are based.

These properties are in particular:

- the potential for high temperature process heat generation
- power generation with high thermal efficiency possibly in combination with district heating and dry cooling towers
- reduced uranium demand by the use of thorium as a fertile material and utilization of the bred U-233.

Although experiences gained with the experimental reactor AVR have been positive, the construction of demonstration plants requires as a prerequisite the successful commissioning and operation of the prototype power plant THTR 3oo.

The HTR-system can, in principle, be adapted to a large variety of fuel cycle strategies.
In the context of this presentation,based on the German HTR-R&D-program, the following HTR-fuel cycle concept have been considered:

- closed fuel cycle using

  . highly enriched uranium and thorium
  . medium enriched uranium (app. 2o %) and thorium
  . low enriched uranium and plutonium

- once through fuel cycle

  . highly enriched uranium and thorium
  . medium enriched uranium (app. 2o %) and thorium
  . low enriched uranium and plutonium

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As far as possible data have been given for each of these fuel cycle options which may be relevant to questions on resource utilization, economics, environmental protection and non-proliferation.

With regard to resource utilization and economics the closed Th/HEU fuel cycle is the most advantageous of all possible fuel cycle options.

The closed U/Pu (LEU) fuel cycle is not expected to offer the same advantages, although a considerable part of it could be based on the experience gained already with U/Pu reprocessing.

The closed MEU fuel cycle requires special attention in the layout of the fuel management because higher enriched material is normally used as fuel make-up. Pu cannot be avoided either. The recycling technology is difficult as it combines the Thorex and the Purex process and practical experience is lacking in this field. The development costs for this cycle are regarded to be by far the most expensive.

Among the open cycles the Th/HEU cycle is also the best one with regard to resource utilization and economics. The open MEU and LEU cycles offer a number of advantages with respect to non-proliferation aspects; because no HEU is present in the fresh fuel. With regard to resource utilization the MEU cycle offers certain advantages as compared to the LEU cycle.

The status of the fuel cycle strategies with regard to the solution of the waste treatment problem is different:

- The closed (HEU) cycle, being the reference case of the German HTR-program, is the basis for the ongoing extensive fuel cycle R&D efforts which have lead meanwhile to facilities and experiences at the laboratory scale.
- The closed LEU cycle could be based to a large extent on existing Uranium/Plutonium fuel cycle technology.
- For all open cycles at present only intermediate storage of the spent fuel is considered, which does not solve the long term waste disposal problem.

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Regarding resources utilization, the closing of the fuel cycle has by far a greater impact than the variation of enrichment grades within the respective fuel cycle strategies.

A similarly clear statement of the impact of the various fuel cycle choices on the proliferation resistance cannot be deduced. All fuel cycles have different characteristics in this respect, since fissile material of different nature and different concentrations appears at different stages of the fuel cycle. Any judgement on the proliferation risks will therefore largely depend on the weight attributed to the various characteristics.

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Note: This paper is based on technical working papers from the Kernforschungsanlage Juelich (KFA) and Hochtemperatur-Reaktorbau (HRB). These papers use the following

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