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MASTER

**GAS REACTOR INTERNATIONAL COOPERATIVE
PROGRAM INTERIM REPORT**

Pebble Bed Reactor Fuel Cycle Evaluation

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**General Electric Company
Advanced Reactor Systems Department
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ABSTRACT

Nuclear fuel cycles were evaluated for the Pebble Bed Gas Cooled Reactor under development in the Federal Republic of Germany. The basic fuel cycle specified for the HTR-K and PNP is well qualified and will meet the requirements of these reactors. Twenty alternate fuel cycles are described, including high-conversion cycles, net-breeding cycles, and proliferation-resistant cycles. High-conversion cycles, which have a high probability of being successfully developed, promise a significant improvement in resource utilization. Proliferation-resistant cycles, also with a high probability of successful development, compare very favorably with those for other types of reactors. Most of the advanced cycles could be adapted to first-generation pebble-bed reactors with no significant modifications.

By agreement this report has been reviewed by the cognizant Federal Republic of Germany industrial and laboratory operations and the resulting comments incorporated or noted herein.

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SECTION 1

SUMMARY

1.1 OBJECTIVE

This topical report is concerned with the evaluation of the fuel cycles for pebble bed reactors. It examines the presently available fuel cycles for the HTR-K and PNP reactors as well as advanced fuel cycles applicable to both. It represents an updating of material presented in a previous report to ERDA⁽¹⁾.

1.2 METHODOLOGY

Under contract to DOE, General Electric is engaged in the evaluation of the German Gas Reactor Program. In addition to a review of the published literature, General Electric personnel have visited three German installations to gather first-hand information. These installations are

1. Kernforschungsanlage (KFA) Nuclear Research Institute in Jülich, where much of the basic research in pebble bed reactors is being performed,
2. Gesellschaft für Hochtemperaturreaktor Technik (GHT), a subsidiary of Interatom and KWU, located in Bensberg. GHT is primarily responsible for the PNP process heat plant, and
3. Hochtemperatur-Reaktorbau (HRB), a subsidiary of GA and BBC, located at Mannheim and primarily responsible for the HTR-K electric power plant.

In addition, General Electric has been in contact with the Los Alamos Scientific Laboratory, which has evaluated the KFA nuclear analysis system⁽²⁾, and with Oak Ridge National Laboratory, which has performed some

independent analyses of high-conversion pebble bed systems.

This report is a selective summary of the work of the above groups, as well as of work performed by General Electric.

1.3 REFERENCE FUEL CYCLE

Table 1-1 shows the reference fuel cycles and reactor parameters for the HTR-K, HHT, PNP, and THTR plants. Note that, despite the plant differences, the fuel cycle is identical for all. This is extremely important, since the development of the THTR, scheduled for operation in the early 1980's, is applicable to all these pebble bed power plants. The fuel element has been tested extensively in the AVR reactor and is qualified for the THTR.

1.4 CONCLUSIONS

GHT does not agree that the fuel system meets all requirements. They specifically mention fission product release. Other sources support our position.

- The basic fuel system which is specified for the HTR-K and PNP is well qualified and will meet the requirements of these reactors. The only additional testing underway is for the purpose of evaluating the limits of fuel performance under conditions which simulate possible accident conditions, and which map the ultimate potential of the fuel.
- Advanced fuel systems, which promise a significant improvement in resource utilization, have a high probability of being successfully developed. Existing reactors could use these cycles with no significant modification.
- Proliferation-resistant fuel cycles can be developed, with a high degree of probable success, which should be as good or better than similar cycles in other types of reactors (LWRs or LMFBRS).
- The development of fuel cycles which permit net breeding ratios up to 1.05 are possible with extensions of proven technology, given the incentive in terms of economics and resource conservation.

- Concepts have been presented which, by means of new technology, could reach breeding ratios as high as 1.1. There are many uncertainties requiring further investigation, including increased core pressure drop and potentially higher control rod insertion forces. Economic evaluation is also needed to see if sufficient incentive for development exists.

TABLE 1-1

REFERENCE FUEL CYCLES AND REACTOR PARAMETERS

PARAMETER	UNITS	REACTOR			
		HTR-K	HHT	PNP	THTR
Power Conversion Method	--	Steam Electric	Gas Turbine Electric	Process Heat	Steam Electric
Reactor Power	MWth	3000			750
Power Density	MW/m ³	5.5			6
Core Height	m	5.5			5.1
Core Diameter	m	11.2			5.6
Helium Inlet Temperature	°C	260	450	300	250
Helium Outlet Temperature	°C	700	850	950	750
Cycle	--	OTTO			MEDUL
Fuel Element Type	--	Ball, 60 mm O.D.			
Max. Fuel Temperature	°C	850	950	1050	1050
Max. Fuel Surface Temperature	°C	830	930	1030	950
Burnup	MWd/kg	100			
Fuel Particle Concept	--	BISO, Mixed Oxide *			
U-235 Enrichment	%	93			
Heavy Metal Loading	g/ball	11.2			

*GHT comments that the similarity extends only as far as being all HEU-Th. Other sources support the data in the table.

SECTION 2

TECHNICAL STATUS OF PEBBLE BED FUEL

The basic pebble bed fuel system consists of coated fuel particles in an all-ceramic graphite ball. Much experience at a number of installations, in several countries, has been achieved. An article by Wolf et al.⁽³⁾ provides a good summary of the fuel status in 1975, and the reader wishing greater detail should consult it. A General Electric evaluation⁽⁴⁾ was published in late 1974, containing data on fabrication processes as well as material on the fuel performance and development needs.

Present research continues in Germany, in the United States (ORNL and GA), and to a lesser extent in other European countries.

This section of the report describes the mechanisms which affect the performance of pebble bed fuel. The performance of the fuel particles within the element is first described, followed by a description of and discussion on the performance of the ball-shaped elements themselves.

2.1 FUEL PARTICLES

2.1.1 DESCRIPTION

Within each pebble, the fuel particles are dispersed in a graphite matrix. Each particle consists of a nearly-spherical fuel kernel surrounded by a coating. This coating forms the principal barrier to fission product release. The coating, which is applied in a fluidized bed, consists of several layers, each having a specific function in the retention of fission products. Two types of coatings are presently in use: BISO coatings, which are multiple layers of pyrolytic carbon (PyC), and TRISO coatings, which are similar except that a layer of silicon carbide is interspersed between the two high-density PyC layers.

Coating layers and their functions are as follows:

1. Pyrolytic Carbon Buffer Layer

The buffer layer is a relatively low-density ($\sim 1 \text{ g/cm}^3$) layer of pyrolytic carbon, ranging from 25 to 85 μm in thickness. The function of this layer is to absorb fission recoils, and, with its porosity, to provide the volume to accommodate fission gases and other fission products as well as kernel swelling.

2. Inner Pyrolytic Carbon Layer

This layer is of high-density ($\sim 1.8 \text{ g/cm}^3$) and 30 to 40 μm thick. It serves as a pressure vessel for retaining fission gases and other gases. It also serves as a barrier to gross diffusion of kernel material (fissile, fertile, or fission products) which may affect the integrity of the next layer (SiC in TRISO particles).

3. Silicon Carbide Layer (TRISO Coating Only)

This layer is high-density ($\sim 3.2 \text{ g/cm}^3$) SiC of 25 to 40 μm thickness. It is the main barrier to metallic fission products which diffuse more slowly through SiC than through pyrocarbon. In addition, because of its stability under fast neutron irradiation, the SiC ~~imparts dimensional stability to the fuel particle.~~

4. Outer Pyrolytic Carbon Layer

The outer high-density ($\sim 1.8 \text{ g/cm}^3$) pyrocarbon layer is 30 to 80 μm thick. Because it contracts initially under neutron irradiation, this outer layer puts the SiC into compression during operation and therefore adds to the overall ability of the coating to withstand gas pressure. This layer also protects the SiC layer during molding of the fuel element.

Cross-sectional views of BISO- and TRISO-coated particles are shown in Figure 2-1.

2.1.2 PERFORMANCE

Because the coating on the fuel particle is the principal barrier to fission product release, any failure of that coating releases fission products to the graphite matrix and eventually to the helium gas coolant.

Figure 2-1. BISO and TRISO Fuel Particles.

The mechanisms by which fission product release occurs and the establishment of performance limits have been extensively studied. The primary mechanisms of release from the particle itself are

- Release due to coating failure resulting from internal gas pressure (from fission gases and other gases)
- Release due to coating failure resulting from stresses generated from the effects of fast neutron doses
- Release due to coating failure resulting from chemical interaction between kernel and coating
- Release due to fission product diffusion through the coating. (This is not accompanied by mechanical failure of the coating.)

2.1.2.1 Internal Gas Pressure

Gas pressure arises from the accumulation of fission gases. Approximately one-third of the fission products are the rare gases, krypton (Kr) and xenon (Xe), which, because of their chemical inertness, remain in the gaseous state. The fraction of the fission gas atoms which reach open pores and escape from the kernel gives rise to an internal gas pressure. ~~Fission gas that is retained within the UO_2 may nucleate in bubbles and~~ cause swelling.

In oxide fuel kernels, fissioning of each fuel atom releases two oxygen atoms in addition to two fission-product atoms. Although oxides are formed by many of the fission products, a significant number of excess free oxygen atoms remains. These atoms react with the pyrolytic carbon coatings to form CO and thus contribute to the gas pressure. Gas pressures can be predicted from fuel kernel size, heavy metal loading, burnup, available free volume, and temperature, and, therefore, the coating thickness can be designed accordingly.

2.1.2.2 Neutron Fluence

Dimensional stability of polycrystalline graphite under irradiation is highly dependent on the degree of orientation as shown in Figure 2-2. Therefore, the pyrolytic coatings must be nearly isotropic to

Figure 2-2. Dimensional Change of Graphite of Different Orientations vs Fast Neutron Fluence(8).

avoid the development of stresses under irradiation which could lead to coating failure. It is also important that initial anisotropy be low because anisotropy increases with fast neutron fluence.

Proper selection of coating thicknesses, densities, and structure (isotropy) results in coatings capable of withstanding fast neutron fluences in excess of those anticipated in the reference designs.

2.1.2.3 Chemical Interaction

Attack of the SiC layer of TRISO-coated carbide fuel particles by the rare earth fission products, La, Ce, Pr, and Nd, has been observed, but this is less of a problem with oxide fuels.

The primary chemical failure mode is migration of the kernel partially through the pyrolytic coating layers (amoeba effect).^{*} Kernel migration can occur to the extent of weakening the structural coatings and increasing the probability of coating failure due to gas pressure and/or neutron induced stresses. Although the kernel appears to migrate up the temperature gradient, it is the buffer layer carbon that moves down the temperature gradient, thus causing the intact kernel to move into the buffer layer. This phenomenon occurs in both carbide and oxide fuels. The rate controlling process in UC_2 fuels is believed to be thermal diffusion of carbon in the UC_2 phase⁽⁵⁾. In oxide fuel, the reaction is thought to be $CO_2 + C \rightarrow 2CO$ at the hot side of the kernel, CO diffusion to the cooler side, then $2CO \rightarrow CO_2 + C(\downarrow)$, and CO_2 diffusion back to the hot side. Alternatively, the CO_2 could react with the fuel kernel to form interstitial oxygen ions which could then diffuse through the kernel to its hot side. The rate of oxide kernel migration is therefore a function of

- O_2 , i.e., CO/CO_2 pressure (a function of burnup),
- kernel temperature, and
- ΔT across the kernel.

Conflicting statements have been made concerning the relative migration rates of oxides and carbides. All carbides have been stated to

^{*}At planned PNP operating temperatures, no amoeba effect is expected. This discussion is intended to place this effect in proper perspective.

migrate faster than all oxides with the rate for ThC_2 greater than that for UC_2 ⁽⁶⁾. A comparison of the data of Ref. 7, however, shows that UO_2 migrates faster than UC_2 below 1500°F. At this time insufficient experimental evidence is available to determine whether amoeba motion is greater in UO_2 or UC_2 fuel particles.

For the pebble bed fuel element, the effect of amoeba migration is eliminated in the OTTO cycle. The reasons for this are discussed below.

The temperature of a fuel element is highest at its center, whereas the temperature gradient is greatest at the fueled zone-shell interface. Since kernel migration depends on both temperature and temperature gradient, amoeba motion will therefore vary across the fuel element. Kernel displacement as a function of radius in the ball is shown for both the conventional and zoned element design (See Section 2-2) in Figure 2-3*. Although Figure 2-3 accurately shows the radial position of maximum amoeba displacement for both fuel element designs, the shown magnitudes of the displacements cannot be compared, since they were calculated for different reactor operating conditions. The apparent agreement at the fueled zone surface should be considered to be fortuitous. Figure 2-3 does, however, show that the maximum displacement does not occur at the point of either maximum fuel temperature or maximum temperature gradient.

As a fuel pebble moves through the reactor core in the OTTO cycle, the temperature gradient across the pebble reaches a maximum value early in its life and then continuously diminishes, whereas pebble temperature rises to a maximum at the point of discharge (Figure 2-4). The rate of amoeba displacement, therefore, varies with the vertical position of the pebble in the core, as is also shown in Figure 2-4. As seen in the same figure, the maximum rate of amoeba displacement occurs at a point in the OTTO core where neither the fuel temperature nor the temperature gradient are at their maximum values. At high core power densities the fuel temperature reaches a maximum and drops slightly at the end of the residence time. The position of maximum rate of amoeba displacement will then shift upward from that of Figure 2-4. Note that the exit gas temperature shown is 1200°C, 250°C greater than the PNP requirements.

*Note that this example is for a much higher exit gas temperature than required for the PNP application.

NOTE THE RELATIVE MAGNITUDES OF THE TWO FUEL FORMS CAN NOT BE COMPARED BUT THE POSITION OF THE MAXIMA ARE CORRECT

Figure 2-3. Incremental Amoeba Displacement of a UO_2 Particle as a Function of Element Radius for a Conventional and Zoned Fuel Element.

Figure 2-4. Cumulative Amoeba Attack at the Radius of Maximum Displacement of a UO_2 Particle Along the Core's Most Demanding Column

Figures 2-3 and 2-4 imply that the rate of kernel displacement is a function of the kernel position in the pebble as well as a function of the radial and axial position of the pebble in the reactor. Kernel displacements are usually determined for the worst zone in a fuel element, and these displacements are summed for the worst radial zone of the reactor as the pebble passed through it. Clearly these displacements are maximum values which apply to only a very small fraction of the total fuel in the reactor.

The limit for kernel migration beyond which the integrity of the particle for fission product retention might be affected has been set at the outer edge of the buffer layer (the buffer layer of the reference fuel particle is 80 μm in thickness). As demonstrated by Figure 2-4, serious amoeba motion is clearly absent at a power density of 5 MW/m^3 .

The above calculations are all based on peak nominal temperatures. Systematic errors (γ heating, pebble rod rearrangements by control rod insertion, and injection of fresh fuel pebbles into the core) can raise the temperature of about 5% of the total fuel by as much as 22°C. This modest increase will not significantly modify the above conclusions. Statistical temperature variations of ± 30 to $\pm 70^\circ\text{C}$ are also possible. These variations, however, are statistical and only apply to a small fraction of the fuel. Since the worst case amoeba displacements only apply to a small fraction of the fuel, the increased mobility of fuel subjected to both temperature statistical variations and damaging amoeba attack, is insignificantly small, (and completely absent for the PNP temperature conditions).

2.1.2.4 Fission Product Release

Fuel particle coatings do not need to fail mechanically for fission products to be released. Although some fission products are released from defective coatings or coatings broken during fuel element fabrication, there are two other means for fission product release as well. Some of the released fission products come from the heavy metal contamination of the matrix outside the particles, which occurs inadvertently during manufacture. And even intact coatings release some fission products. These move through the coating by the diffusion process, and some have sufficient mobility to traverse the entire coating

thickness. Fission products that do traverse the entire thickness of the coating must diffuse through the graphite matrix and shell, and evaporate from the pebble surface in order to reach the helium coolant.

The release of fission products such as Cs, I, and Ag restricts the repair and maintenance of components of the primary heat transport circuits. Estimates of the plate-out of fission products in the colder parts of the AVR circuits suggest for Cs-137 an upper release limit of about 10 Ci/year (i.e., a R/B* ratio of 2×10^{-6}). (21)

Fission products are divided into three main categories with respect to their effect on HTR safety:

1. Inert gases (Kr and Xe)
2. Electronegative elements (principally I), and,

Figure 2-5. Correlation of Fission Gas Release and Heavy Metal Contamination of the Matrix.
(From E. Balthesen and H. Ragoss, HTRB-TN-19/74, Int. Conference on Phy. Met. of Reactor Fuel Elements, Berkeley, September, 1973).

*R/B = rate of release/rate of birth for nuclides in radioactive equilibrium.

3. Electropositive elements (principally the metals Cs, Ag, and Sr).

The primary barrier to the inert gases is the inner high-density pyrocarbon coating. Kr and Xe have very low diffusion coefficients in pyrocarbon (on the order of 10^{-15} cm²/s at 1250°C), and are therefore released only from broken fuel particles and from heavy metal contamination of the matrix outside the fuel particles (Figure 2-5).

At HTR core temperatures, iodine behaves like a rare gas of equivalent half-life. Much of the iodine that does reach the helium coolant is deposited on metal or graphite surfaces in the low temperature parts of the circuit.

Electropositive elements (e.g., Cs, Ag, Sr) have significant mobility in pyrolytic carbon coatings as shown by the data in Table 2-1, where fractional release of Sr-90 from BISO- and TRISO-coated fuel particles is given for three temperature ranges. The table also shows that TRISO coatings retain Sr-90 much better than BISO coatings at the higher temperatures. The ability of the SiC layer of TRISO coatings to better retain metallic fission products allows TRISO-coated particles to be operated at higher temperatures than BISO-coated particles for a given fission product release rate. (Note that TRISO particles are not needed for PNP).

TABLE 2-1

FRACTIONAL RELEASE OF Sr-90 THROUGH BISO AND TRISO PARTICLE COATINGS

Particle Type		Fuel Temperature Range, °C		
Name	Retaining Layer	900-1000	1040-1150	1160-1230
BISO	P _y C	4.6×10^{-6}	1.7×10^{-4}	1.4×10^{-2}
TRISO	P _y C/SiC/P _y C	4.3×10^{-6}	3.0×10^{-6}	3.3×10^{-6}

Fuel burnup 3.5% FIMA, 280 days irradiation (Ref. 8).

Calculations for Cs-137 release in an OTTO-cycle pebble bed reactor were performed at KFA for a process heat reactor for different gas outlet temperatures (770°, 870°, 970°, 1070°, 1170°C) and for different fuel designs. For these calculations, the reactor core was divided into six vertical loading regions of equal volumes. The Cs-137 release was calculated in each region as a function of the power and temperature

history of a fuel element residing in this region. Since there is a temperature gradient within a pebble moving through the core, the pebble was divided into three temperature zones. In each zone the Cs release was calculated for a representative coated particle.

The diffusion coefficients and the release model were tested against irradiation experiments performed in reactors in Studsvik and Jülich. Encapsulated pebbles were irradiated, and the measured Cs concentration profiles were shown to agree with those calculated from the above diffusion coefficients. The calculated release rate also agreed with the release rate from the AVR as measured in the VAMPYR experiments. The calculated Cs-137 release rates from the above fuel types are shown in Figure 2-6 as a function of matrix contamination and exit gas temperature for a 3000 MWth reactor operating in an OTTO-cycle mode.

For a 100 Ci/year Cs release in a 3000 MWth reactor, curve 3 of Figure 2-6 shows that the THTR fuel pebble with its BISO-propylene-coated, mixed oxide fuel may be used up to mean gas outlet temperature of 970°C with an initial U contamination of the matrix of no more than 2.5×10^{-5} *. Since the present contamination is about 2.5×10^{-4} , however, Figure 2-6 shows that the existing THTR fuel would release about 100 Ci/year of Cs-137. A uranium contamination of 2.5×10^{-5} has been reached in small fabrication batches, but development (presently underway) is necessary to reach this level in large scale fuel production. Little improvement beyond the latter value of contamination can be expected since the uranium level in the natural graphite used in the fuel element matrix is about 10^{-6} , i.e., curve 6 of Figure 2-6 is the lowest release possible.

The following conclusions may be drawn from the above:

1. Up to a He outlet temperature of 950°C the uranium contamination of the fuel matrix (all of the pebbles except the coated particles) determines the Cs release.
2. The well developed THTR fuel pebble with coated particle coatings deposited from propylene (BISO) is acceptable up

*This is $\frac{\text{Weight of U in the graphite}}{\text{Total weight of U in the fuel}}$

CYCLE OTTO
 POWER 3000 MWt
 BURN UP 100 000 MWd/T
 GAS INLET TEMPERATURE 250°C
 FUEL RESIDENCE TIME 800 DAYS
 POWER DENSITY 9 MWm³

CURVE NO	SYMBOL	FUEL	MATRIX CONTAMINATION	FRACTION OF FAILED PARTICLES	FUEL ELEMENT
1	————	FEED-TRISO BREED-BISO	2.5×10^{-4}	10^{-3}	CONVENTIONAL
2	-----	MIXED BISO	2.5×10^{-4}	NONE	CONVENTIONAL
3a	- · - · - ·	MIXED BISO	2.5×10^{-5}	NONE	CONVENTIONAL
3b	-----	MIXED BISO	2.5×10^{-5}	NONE	ZONED
4	·····	FEED TRISO BREED-TRISO	2.5×10^{-5}	10^{-4}	CONVENTIONAL
5	—●—●—	MIXED TRISO	2.5×10^{-5}	NONE	CONVENTIONAL
6	-----	MIXED-TRISO	2.5×10^{-6}	NONE	CONVENTIONAL

Figure 2-6. Release of Cs-137 as a Function of Gas Outlet Temperature.

to a He outlet temperature of 970°C if the uranium level in the fuel matrix is $\leq 2.5 \times 10^{-5}$.

3. TRISO-coated particles in the THTR fuel element form will give acceptable fission product release up to at least 1170°C if the matrix contamination is $\leq 2.5 \times 10^{-5}$, and the irradiation-induced, pyrolytic coating failure fraction is $\sim 10^{-4}$.
4. BISO fuel will give acceptable release rates for the HTR-K conditions without further development (gas outlet temperature of 700°C).

2.1.3 PERFORMANCE POTENTIAL AND LIMITATIONS

GHT feels
this is a
somewhat
simplified
discussion

The primary factors which influence the performance of coated fuel particles are operating temperature, fast neutron fluence, fuel burnup, and time spent in a particular regime zone of the reactor. As indicated earlier, fast neutrons cause dimensional changes in graphite materials and, therefore, result in the buildup of stresses in the particle coatings. Coating stresses are also caused by gas pressures resulting from the accumulation of fission gases; these stresses are thus a function of burnup and temperature. Because gas pressure, neutron-induced dimensional changes, graphite strengths, and creep rates are all temperature dependent, limiting values of temperature, burnup, and neutron fluence are interrelated and cannot be stated independently of one another.

In order to define the operating potentials of existing coated-particle fuels with respect to the reference pebble bed reactor, temperature limitations associated with the various failure modes are given, assuming that the maximum fast fluence is $4.65 \times 10^{21} \text{ n/cm}^2$ ($E > 0.1 \text{ MeV}$) and the maximum burn up is 13.4% FIMA. Exit gas temperature limitations associated with the various failure modes for these conditions are summarized in Table 2-2.

The maximum gas temperature of OTTO cycle reactors when BISO-coated fuel particles are used is limited to 970°C by release of fission products. When TRISO-coated particles are employed, the gas temperature can be greater than 1170°C with acceptable fission product release, providing that heavy metal contamination of the matrix is less than 2.5×10^{-5} .

TABLE 2-2

OUTLET GAS TEMPERATURE LIMITS FOR
OTTO CYCLE REFERENCE SYSTEM

Failure Mode Particle Type	Fission Gas Pressure	Irradiation Cracking	Kernel Migration	Metallic Fission Product Release
BISO	>1500°C	>1350°C	>1170°C	970°C
TRISO	>1500°C	>1350°C	>1170°C	>1170°C

Based on calculated amounts of kernel migration that are predicted to occur in UO_2 fuel particles in the reference reactor, all three exit gas temperatures (HTR-K, HHT, PNP) can be obtained without consequential damage to the coatings by amoeba attack. Limiting temperatures due to gas pressure and neutron fluence are also above the maximum temperatures of the reference designs.

2 2 FUEL ELEMENTS

Figure 2-7 shows schematically two types of fuel elements for pebble bed reactors. The conventional element consists of a fueled zone which is made up of a mixture of flake graphite, coke phenolic resin, and fuel particles. The outer shell is made of the same mixture. When the entire element is pressed and fired, the resin is cracked and the entire element coked.

For the THTR, and also as specified fuel for the HTR-K and PNP reactors, the coated particles take up approximately 9-10% of the fueled zone volume, giving a heavy metal loading of approximately 11.2 g per ball.

Conventional
Element

Zoned
Element

60 mm (Ball Diameter)

Figure 2-7. Pebble Bed Fuel Element Design.

The zoned ball is a variation which may be used to reduce the peak fuel temperature. Note that, in the example shown in Figure 2-7, the unfueled center section is only 22% of the volume within the 5 mm outer shell. This means that the volume fraction of coated particles in the fueled zone does not increase significantly.

From the standpoint of fission product release, the barriers are basically those of the particles. Although the matrix graphite of the elements does impede release, the performance of the element with respect to fission product release is determined by the particles. This is important since work on coated fuel particles is being performed not only for the pebble bed reactor concept, but also for the HTGR.

The advantages of the ball fuel element become apparent when it is realized that complete element testing in available reactors is possible.

Additional development is being performed in the following areas:

- Improvement of the matrix to produce more reproducible results and better radiation resistance
- Reduction in heavy metal contamination of the matrix to reduce fission product release from fission outside the coated particles
- Investigations of the limits of cold press procedure with respect to contamination, particle breakage, and element mechanical strength (cold pressing is the present procedure).
- Development of hot-pressed elements in order to increase the heavy metal loadings for advanced design.

2.3 OPERATIONAL EXPERIENCE

Fuel system operating experience can be grouped under two broad headings,

- Testing for the THTR reactor, using the AVR and other materials testing reactors,
- Testing of elements for advanced concepts such as the HTR-K and PNP, and including high-conversion systems and proliferation-resistant systems.

The testing for THTR fuel is aimed at fully qualifying the fuel system for THTR conditions. This work is essentially complete. Verbal communications have indicated that the manufacturer, HOBEG, will warrant their fuel for THTR conditions.

Testing for advanced systems has been aimed at qualifying THTR fuel for the OTTO cycle in the HTR-K and PNP reactors, as well as more advanced systems. A key parameter for the advanced fuel systems is the heavy metal loading per ball. High-conversion systems will require loadings of 20-25 g/ball, while the near and net breeders will require up to 45 g/ball. Table 2-3 shows the present status.

TABLE 2-3

PBR FUEL ELEMENT DEVELOPMENT STATUS

FUEL DESCRIPTION	HEAVY METAL LOADING (GRAMS/BALL)	VOLUME FRACTION OF COATED PARTICLES IN FUEL MATRIX	COMMENTS
AVR Fuel	6	~0.1	Off-the-shelf, fully qualified. 3.5 years with 950°C exit gas.
THTR Fuel	11	~0.10	Developed, tested in the AVR. Qualified for use.
Developed Fuel	16 → 25	0.12-0.21	Developed, under test in AVR. Not fully qualified.
Projected Fuel	30 → 45	0.25-0.40	Needs manufacturing development, which is under way.

- Full scale fuel elements well qualified
 - Macro configuration constant
- Coated particles are the same for all
 - Burnups to 160,000 MWd/t or greater

2.3.1 TESTING TO QUALIFY THTR FUEL

Many thousands of fuel elements have been tested in the AVR reactor. Reference 9 by GHT reports some of these findings, as does reference 3. The comments below are summarized from reference 9.

2.3.1.1 Dimensional Stability

The dimensional changes of the integral fuel element correspond qualitatively to those of the pure matrix material at the same temperature and under the same irradiation conditions. The maximum (linear) shrinkage of fuel elements for maximum THTR irradiation conditions of fast neutron fluences and temperature has been found to be about 1.7%. The dimensional changes of the complete fuel element result from a combination of dimensional changes in the isotropic inner zone and the anisotropic fuel-free shell.

2.3.1.2 Mechanical Strength

High minimum values of crushing strength were specified for the THTR in order to withstand mechanical interaction with the control rods. Following development work, it was ascertained that this specification could be met. From extensive investigations with irradiated materials, including the AVR elements, no systematic changes of crushing strength resulting from irradiation could be detected. Furthermore, no dependence upon particle type or number of elements present has been observed. There is some indication of a slight increase in crushing strength from irradiation, especially for recently produced elements.

2.3.1.3 Oxidation Resistance

The corrosion behavior of the semi-isostatically moulded fuel elements can be influenced by the final temperature of the HTT.* In early stages of development, it was often difficult to achieve the specified values. For the current production of elements for the first core of THTR, the maximum HTT temperature has been raised to 1950°C (3542°F) which yields oxidation resistance values better than those specified and

* High Temperature Treatment

tolerable heavy metal contamination of the matrix (outside the particles). Oxidation measurements, especially with fuel elements irradiated in AVR and Dragon reactors, have shown that there is practically no change from pre-irradiation behaviour.

2.3.1.4 Behavior in the AVR (Peeling Effect)

In the spring of 1972, a "peeling" effect, characterized by 0.1- to 0.3-mm-thick surface layers peeling from fuel elements, was observed in AVR fuel. Although not detrimental to the overall performance of the elements, this damage caused concern. Since that time, extensive investigations showed that this was caused by corrosion effects due to air ingress. It was observed in only one batch of fuel and has not reoccurred. It is not considered a problem.

In general, the spherical fuel element as an integral unit has a high state of development and testing with regard to the stringent requirements of the THTR.

2.3.2 EXPERIENCE WITH ADVANCED SYSTEMS

Work on elements with higher fuel loadings is continuing. Elements of the cold-pressed variety have been tested with loadings up to 25 g/ball. Manufacturing development on hot (warm)-pressed balls is underway. It is believed that the 45 g per ball are entirely feasible.

Testing of THTR balls to PNP conditions continues in the AVR, and no barriers to their use of 950°C in the OTTO cycle are apparent. Balls with low enrichment have also been tested in the AVR.

No barriers are apparent, but continued development is required.

SECTION 3

POTENTIAL FUEL CYCLES FOR THE PEBBLE BED REACTOR

This section contains detailed descriptions of various cycles studied for the PBR. The majority of the work reported was performed at KFA. The results of work at HRB and GHT, while differing in their design emphasis, are in essential agreement with respect to the nuclear calculations.

Table 3-1 shows a summary of the cycles discussed in this section. The first, the M093 or TOT, uses the proven THTR fuel described above in Section 2. This is the cycle selected for the HTR-K, HHT, and PNP reactors. The others are potential cycles designed to meet future economic or political conditions.

3.1 LOW-ENRICHMENT CYCLES

3.1.1 URANIUM CYCLES

Low-enrichment cycles are those using a mixture of U-235 and U-238 in the coated particle as an oxide. The enrichment needed is of the order of 8-12% and depends on the desired burnup and on the carbon-to-heavy-metal ratio (N_C/N_{HM}). Early work at KFA on these cycles, using the OTTO fuel management scheme, is reported in reference 10. Four cases have been identified as being of special interest, three reported in reference 12 and one reported in reference 16. Table 3-2 summarizes these cases, and Figure 3-1 shows the trends of ore requirements versus burnup.

The first three cases show the effect of burnup on ore requirements, economics, and conversion ratio. The fourth case shows that a lower power density provides superior ore requirements, due to the better conversion ratio and lower parasitic capture. The influence on the conversion rate is not only due to the increasing absorption in the fission products, which varies as shown in Table 3-3, but also to the increase in axial leakage. This increase in axial leakage is caused by the presence of the varying amounts of fission products and secondary

TABLE 3-1

PEBBLE BED REACTOR FUEL CYCLES

Designation	Description	Fuel Type	Loading (gm HM/Ball)	Moderation Ratio (N_c/N_{HM})	Burn-up (MWd/kg)	Conversion Ratio
M093 (TOT)	Reference Cycle (THTR fuel)	U-Th Mixed Oxide	11.2	325	100	0.60
1113	Low Enrichment Uranium Cycle (9 MW/m ³)	U Mixed Oxide	10.4	363	70	0.58
1013	Low Enrichment Uranium Cycle (9 MW/m ³)	U Mixed Oxide	10.4	363	100	0.55
1213	Low Enrichment Uranium Cycle (9 MW/m ³)	U Mixed Oxide	10.4	363	130	0.52
UOT	Low Enrichment (5 MW/m ³)	U Mixed Oxide	10.2	357	100	0.58
M020	20% U-Th Fuel	U-Th Mixed Oxide	8.1	458	100	0.58
SB20	Like M020, except separate fertile Balls	U Oxide Th Oxide	6.0/16.5	458	101	0.56
4011	Once Through Cycle (9 MW/m ³)	U-Th Mixed Oxide	~15	~244	101	0.58
4021	Like 4011 (5 MW/m ³)	U-Th Mixed Oxide	~15	~244	102	0.62
SB93	Like M093 with Separate fertile ball	U Oxide Th Oxide	6.0/20.1	355	100	0.58
MOR	Recycling Reactor	U-Th Oxide	11.2	325	100	0.57
RSE	More Complex Recycling	U-Th Oxide	1.7/15.0	242/2110	100	0.62
SFB	Even More Complex Recycling	U-Th Oxide	2.0 - 20.0 (4 streams)	180 - 1879	100	0.65
PB	Prebreeder	U-Th Oxide	1.4 - 32.4	198	23.2	0.74
321	Variation of MOR/RSE	U-Th Oxide	16.5	100	93.0	0.71
NB	Near Breeder	U-Th Oxide	32.4	110	24.0	0.97
110/20	Net Breeder with Radial Blanket	U-Th Oxide	32	110	20	1.01
110/10	Net Breeder with Radial Blanket	U-Th Oxide	32	110	10	1.04
80/20	Net Breeder with Radial Blanket	U-Th Oxide	44	80	20	1.03
80/10	Net Breeder with Radial Blanket	U-Th Oxide	44	80	10	1.05
PBTBR	Net Breeder with Decoupled fissile and fertile fuel flow	U-Th Oxide	32	110	24	1.1

TABLE 3-2

LOW ENRICHMENT FUEL CYCLES
(KFA DATA)

Parameter	Units	Case			
		1113	1013	1213	UOT
<u>Overall Reactor</u>					
Power Level	M _{Wth}	3000			
Conversion Efficiency for Electric Power	-	0.40			
Core Height	m	5.0			5.5
Core Diameter	m	9.22			11.78
Power Density	MWth/m ³	9.0			5.0
Core Outlet Temperature	°C	985			
Pressure Level	b	~40			
<u>Fuel</u>					
Type	Conv/ Shell	shell			
Coated Particle Dia.	μm	800			
Avg. Moderator Ratio	N _C /N _{HM}	363			357
Avg. Heavy Metal Loading	g/ball	10.38			10.2
Initial Enrichment	% N _f /N _{HM}	7.16	9.71	12.35	8.62
<u>Overall Results</u>					
Average Burnup	MWd/kg	69.8	100.0	130.0	100.0
Average Fuel Residence Time	F.P. days	439	627	815	1122
Conversion Ratio	-	0.580	0.549	0.521	0.575
Fast Dose to Balls >10 ⁵ ev	10 ²¹ / cm ²	2.89	4.14	5.31	4.08
Ore Requirements (0.25% Tails)	kg/ GWde	621	594	583	521
Separative Work	kg SWU/ GWde	441	446	453	384

TABLE 3-2 (Cont'd.)

LOW ENRICHMENT FUEL CYCLES

Parameter	Units	Case			
		1113	1013	1213	UOT
<u>Mass Balance</u>					
Charged					
Heavy Metal	kg/GWde	35.43	24.80	19.08	24.64
U-235	"	2.52	2.38	2.32	2.10
U-238	"	32.91	22.42	16.76	22.54
Discharged					
Heavy Metal	kg/GWde	32.77	22.09	16.36	21.77
U-235	"	0.65	0.48	0.39	0.34
Pu-239+Pu-241	"	0.35	0.27	0.21	0.23
Net Balance (w/o re-processing)					
U-235	kg/GWde	-1.86	-1.90	-1.94	-1.76
Pu-234 & Pu-241		0.35	0.27	0.21	0.23
<u>Inventory-Fissile</u>					
U-235	kg/GWe	481	547	604	762
Pu-239 & Pu-241	"	137	162	177	260

Figure 3-1. Ore Requirements for LEU Fuel Cycles versus Burnup

plutonium isotopes in the lower core. The presence of these products causes a shift in the neutron flux profile towards the upper part of the core. The effect of the increasing neutron losses with higher burnup is a corresponding decrease in the fertile material absorption rates, thus lowering the conversion rate.

The utilization of the in-core bred fissile plutonium isotopes is independent of conversion ratio. Table 3-3 shows the fraction of bred fissile plutonium isotopes which are burnt in the reactor directly. The highest burnup case discharges only 11% of the fissile plutonium, which therefore either helps the reprocessing problem, or, in a non-reprocessing environment, helps the disposal problem, from both the economic and long-term storage standpoints.

TABLE 3-3

EFFECT OF BURNUP ON CONVERSION RATIO
(9 MW/m³ Power Density)

Burnup	Mwd/kg	69.8	100.0	130.0
Absorption in Fission Products	% per Source Neutron	7.18	6.26	9.23
Axial Leakage	% per Source Neutron	4.93	5.42	5.82
Conversion Ratio	-	0.580	0.549	0.521
Fraction of Bred Fissile Burnt in Reactor	%	84	87	89

KFA Data

This combination of increasing plutonium utilization and decreasing conversion ratio explains why there is only a small increase in net U-235 consumption in the higher burnup cases. Thus, because of the higher burnup, the uranium ore requirement decreases.

A decrease in power density improves both the conversion ratio and the net balance of U-235, so that the ore requirements decrease. In the case of the 100.0 MWd/kg, an improvement of 12% is realized in going from 9 MW/m³ to 5 MW/m³. From Figure 3-1, it is clear that a cycle operating at 5 MW/m³, and with a burnup approaching 150 MWd/kg, will have an ore requirement approaching 500 kg/GWde.

Variations of the low-enrichment cycles in the area of possible recycle are limited to the following schemes:

1. Throwaway (or store-away) cycles in which no reprocessing is performed, and the fuel is stored either permanently or temporarily in the same fuel elements as when discharged.
2. Limited reprocessing in which the plutonium is never separated, and reprocessing is used solely as a means of improving the long-term storage safety and/or economics.
3. Uranium recycle in which the uranium which is discharged is either reenriched or mixed with moderately or highly enriched fuel to regain the necessary 8-12% enrichment. For realistic calculations it will be necessary to account for the buildup of U-236, which is a nuclear poison.
4. Plutonium recycle in which the plutonium is also fed back into the system. This case has not been analyzed, but may not be economically attractive because of the lower nuclear worth of plutonium in a PBTBR spectrum compared to other reactor types.

Economically, cases 1 and 2 will represent a net loss to the fuel cycle cost since no credit is available, and some charges are incurred. In case 3, if the U-236 is equated to U-238, and if the plutonium is sold, there may not be a lower fuel cycle cost, but certainly less ore will be required. Case 4 is the same as case 3, economically, if the cost/value of plutonium is based on nuclear equivalence in a PBR.

3.1.2 20% URANIUM-THORIUM CYCLES

Recent work at KFA⁽¹⁸⁾ has shown that there are several fuel cycles which are especially suited for use in a nonproliferation environment. They use uranium enriched to 20% U-235 (or some acceptable value) plus thorium. The concept limits the fissile fraction of the feed uranium to a "safe" value, and in addition, allows the use of the conversion of thorium to U-233 to improve ore conservation. Table 3-4 shows the parameters of two such cycles, along with the M093 (TOT) cycle for reference.

In the first cycle (M020), the fuel form is mixed oxide particles with both UO_2 and ThO_2 in the same particle. There is no possibility, short of reenrichment, of separating U-235 or U-233 into weapons grade material in any part of the fuel cycle.

In the second cycle (SB20), the thorium is kept in separate balls. Thus the U-233 can be separated for use in near breeders or in net breeders at a future time. It is intended that the spent fuel be stored until re-cycle becomes economically and politically acceptable.

The advantages of these cycles are that, if nonproliferation rules dictate, a cycle distinctly better than the low enriched cycles (in terms of resource conservation) and only slightly worse than the nonrecycle, highly enriched cycles, can be used in the pebble bed reactor.

3.2 HIGHLY ENRICHED CYCLES

Those fuel cycles which make use of highly enriched uranium are discussed below under four headings. First are cycles designed primarily for use without recycle of the spent fuel, generally optimized for fuel cycle cost and ore conservation. They are characterized by moderate conversion ratios. Second is a series of fuel cycles which use reprocessing, recycle, and advanced fuel elements to reach progressively higher conversion ratios. Third is a cycle using high purity U-233 to achieve a very high conversion ratio, near 0.97. Fourth are cycles which permit net breeding by means of various conceptual schemes.

TABLE 3-4

20% URANIUM-THORIUM FUEL CYCLE

(KFA DATA)

Parameters	Units	Case		
		M020	SB20	TOT (M093)
<u>Overall Reactor</u>				
Power Level	MWth	3000		
Conversion Efficiency for Electric Power	%	40		
Core Height	m	5.5		
Core Diameter	m	11.78		
Power Density	MWth/m ³	5		
Core Outlet Temperature	°C	985		
Pressure Level	b	~40		
<u>Fuel</u>				
Type	Conv/ Shell	Conv.		
Coated Particle Diameter	μm	400		
Average Moderator Ratio	N _C /N _{HM}	458		325
Average Heavy Metal Loading	g/ball	8.07	0.6/16.54	11.2
Initial Enrichment	% N _F /N _{HM}	-		7.16
<u>Overall Results</u>				
Average Burnup	MWd/kg	100	101	100.0
Average Fuel Residence Time	F.P. Days	872		1217
Conversion Ratio	-	0.58	0.56	0.597
Fast Dose to Ball, E>10 ⁵ ev	10 ²¹ / Cm ²	-	-	4.25
Ore Requirements (0.25% Tails)	kg/GWde	493	499	462
Separative Work	kg SWU/ GWde	405	410	418

TABLE 3-4 (Cont'd.)

20% URANIUM-THORIUM FUEL CYCLE

(KFA DATA)

Parameters	Units	Case		
		M020	SB20	TOT (M093)
<u>Mass Balances</u>				
Charged				
Heavy Metal	kg/GWde	25.04	25.21	24.97
U-235		1.95	1.97	1.81
U-238		7.90	8.00	0.14
Th-232		15.19	15.23	23.02
Discharged				
Heavy Metal		22.33	22.35	22.15
U-233 + Pa-233		0.33	0.30	0.47
U-235		0.14	0.12	0.08
Pu-239 and Pu-241		0.054	0.050	-
<u>Net Balance</u>				
U-233	kg/GWde	0.33	0.30	0.47
U-235		-1.81	-1.85	-1.73
Inventories - Fissile	kg/GWe			
U-233		218	201	461
U-235		432	399	453
Pu-239 and Pu-241		57	53	-

3.2.1 LOW CONVERSION RATIO CYCLES

Table 3-5 shows four fuel cycles using highly enriched U-235, and designed primarily for use in a no-recycle environment. The first three cases use mixed oxide fuel particles. The first two differ in the power density, and show the improved nuclear performance of the lower power density in terms of ore requirements and conversion ratio. The third case (nearly identical to the reference THTR cycle) shows the slightly lower conversion ratio due to the lower fuel loading and higher carbon-to-heavy metal ratio.

The fourth cycle is nuclearly almost identical to the third. The only difference is that separate elements are used for pure thorium and fissile fuel. Eighty percent of the thorium is put in balls with no U-235. Thus 79% of the bred U-233 is discharged in balls which can be reprocessed separately to provide relatively pure U-233 for near breeders and net breeders. It is assumed that the spent fuel balls would be stored until the U-233 was needed, rather than immediately reprocessed.

TABLE 3-5

HIGH ENRICHMENT, NO RECYCLE FUEL CYCLES

(KFA DATA)

Parameters	Units	Case			
		4011	4021	TOT (M093)	SB93
<u>Overall Reactor</u>					
Power Level	MWth	3000			
Conversion Efficiency for Electric Power	%	40			
Core Height	m	5.00	6.0	5.5	
Core Diameter	m	9.22	11.28	11.78	
Power Density	MWth/m ³	9		5	
Core Outlet Temperature	°C	985			
Pressure Level	b	~40			
<u>Fuel</u>					
Type	Conv/ Shell	Conv.			
Coated Particle Diameter	μm	600		400	400
Average Moderator Ratio	N _C /N _{HM}	~244		325	355
Average Heavy Metal Loading	g/ball	~15.0		11.2	60/20.1
Initial Enrichment	% N _f /N _{HM}	7.93	7.12	7.16	~ 7.3
<u>Overall Results</u>					
Average Burnup	MWd/kg	100.7	101.5	100.0	100.0
Average Fuel Residence Time	F.P. Days	907	1633	1217	1117
Conversion Ratio	-	0.583	0.625	0.597	0.58
Fast Dose to Ball, E>10 ⁵ ev	10 ²¹ / cm ²	5.62	5.68	4.25	-
Ore Requirements (0.25% Tails)	kg/FWde	506	448	462	462
Separative Work	kg/SWU/ GWde	460	408	418	418

TABLE 3-5 (Cont'd.)

HIGH ENRICHMENT, NO RECYCLE FUEL CYCLES

(KFA DATA)

Parameters	Units	Case			
		4011	4021	TOT (M093)	SB93
<u>Mass Balance</u>					
Charged					
Heavy Metal	kg/GWde	24.65	24.45	24.97	24.85
U-235		1.98	1.76	1.81	1.81
U-238		0.15	0.13	0.14	0.14
Th-232		22.52	22.56	23.02	22.90
Discharged					
Heavy Metal		21.97	21.77	22.15	22.14
U-233 + Pa-233		0.52	0.48	0.47	0.42
U-235		0.14	0.10	0.08	0.08
<u>Net Balance</u>					
U-233	kg/GWde	0.496	0.478	0.47	0.42
U-235		-1.836	-1.662	-1.73	-1.73
<u>Inventories - Fissile</u>	kg/Gwe				
U-233		365	677	461	381
U-235		452	584	453	432
Pu-239 and Pu-241		2.5	4.2	2.4	1.9

3.2.2 HIGH CONVERSION RATIO CYCLES

Reactor cycles, in which U-233 is bred from thorium and recycled, are of interest for three basic reasons,

- 1) The net ore requirements are significantly lower than nonrecycle cases.
- 2) The fuel cycle cost may be comparable with other, less ore conserving cycles, especially when fuel prices rise faster than the general inflation rate.
- 3) The high conversion ratios possible with advanced fuel elements permit a transition to a prebreeder* reactor.

The key elements of any thorium recycle reactor are the reprocessing of the fuel to remove fission products and spent thorium, and a refabrication of fuel elements containing U-233. Highly enriched uranium is added to make up the fissile deficit for these nonbreeding reactors.

Over the last four years, many variations of those cycles which use thorium have been studied at KFA. This section describes this work, and presents cycles which show the logical progression up to a near breeder with a conversion ratio of approximately 0.97. Four areas are varied to achieve this improvement. These are

- 1) Control of the U-236 buildup (which is a nuclear poison) from the U-235 fissile material,
- 2) Increase in the fuel loading per ball, which increases the conversion ratio,
- 3) Decrease in the average burnup which minimizes losses to fission products, and
- 4) Use of bred U-233 in a separate reactor (No U-235 or U-236).

Table 3-6 summarizes the results of five fuel cycle cases. Figure 3-2 shows three of the cycles in diagram form, while Figure 3-3 shows the near-breeder cycle in diagram form. Table 3-7 shows the results of the near breeder cycle.

*A prebreeder is a reactor operated on the U-235/thorium cycle and used to generate U-233 for a near breeder or breeder.

TABLE 3-6

HIGHLY ENRICHED FUEL CYCLES — WITH U-233 RECYCLE (KFA DATA)

		CASE											
Parameter	Ball Type	MOR		RSE		SFB				PB			321 (MOR)
		1	2	1	2	1	2	3	4	1	2	3	1
<u>Overall Reactor</u>	<u>Units</u>												
Power Level	MWth							3000					
Conversion Efficiency for Electric Power	%							40					
Core Height	m							5.5					
Core Diameter	m							11.78					
Power Density	MWth/m ³							5					
Outlet Gas Temperature	°C							985					
Pressure Level	b							~40					
<u>Fuel Description</u>													
Type Fabrication	Hot/Cold	Cold	Hot	Cold	Hot	Hot	Cold	Cold	Hot	Hot	Cold	Cold	Hot
Type Matrix	Conv/Shell	Conventional											Conv
Coated Particle Diameter	μm	400		200		400	200	400		200	400		400
Average Moderator Ratio	N _C /N _{NHM}	325		242	2110	410	1879	180		2100	735	110	220
Average Heavy Metal Loading	g/ball	11.2		15.0	1.7	8.9	2.0	20.1		1.8	5.0	32.4	16.5
Initial Enrichment	% N _F /N _{NHM}	7.4	9.0	4.9	57.7	18.0	93.0	0.0	65.9	43.1	24.0	0.0	-
Fractional Supply	%	72	28	72	28	2	33	44	21	15	36	49	100
<u>Overall Results</u>													
Average Burnup	MWd/kg			100							23.2		93.6
Average Fuel Residence Time	F.P.Days	1217		1209		1499				451			1666
Conversion Ratio	-	0.57		0.62		0.65				0.74			0.71
Fast Dose to Ball, E>10 ⁵ eV	10 ²¹ /cm ²	4.25		4.26		5.23				1.56			5.83
Ore Requirement (0.25% Tails)	kg/GWde	342		301		283				561			<280
Separative Work	kgSWU/GWde	311		274		257				510			<254

MOR: Mixed Oxide Recycling
 RSE: Recycle in Separate Elements
 SFB: Separate Feed and Breed Elements

PB: Prebreeder
 321(MOR): Advanced MOR

TABLE 3-6 (Cont'd.)

	Net Mass Balances (kg/GWde)				
	MOR	RSE	SFB	PB	321
<u>Charged</u>					
Th-232	21.55	22.74	22.79	103.45	24.850
U	3.36	2.23	2.28	4.38	1.705
U-233	0.49	0.49	0.47	0.12	0.574
U-235	1.49	1.24	1.22	3.58	1.037
U-236	0.75	0.19	0.22	0.29	0.018
U-238	0.63	0.31	0.37	0.39	0.076
<u>Discharged</u>					
Th-232	20.09	21.01	21.06	101.14	22.924
U	2.00	1.41	1.29	4.06	0.944
U-233/Pa-233	0.50	0.51	0.48	1.54	0.578
U-235	0.16	0.10	0.12	1.35	0.122
U-236	0.79	0.43	0.35	0.65	0.186
U-238	0.55	0.37	0.34	0.52	0.058
<u>Net</u>					
Th-232	-1.46	-1.73	-1.73	-2.32	-1.926
U	-1.36	-0.82	-0.99	-0.32	-0.761
U-233	0.014	0.02	0.01	1.42	0.004
U-235	-1.33	-1.14	-1.10	-2.23	-0.915
U-236	-0.04	0.24	0.13	0.36	0.168
U-238	-0.08	0.06	-0.03	0.13	-0.018
<u>Inventory (kg/GWe)</u>					
U-233	585	572	708	358	942
U-235	504	365	500	1018	510
U-236	998	438	525	231	281

FUEL CYCLE

	M O R	R S E	S F B
	MIXED OXIDE RECYCLING	RECYCLING SEPARATE ELEMENTS	SEPARATED FEED-BREED

Figure 3-2. PBR Recycling Variants (KFA DATA)

FUEL ELEMENT TYPE

FUEL

URANIUM

THORIUM

FABRICATION

GR/BALL

GR/BALL

FRACTIONAL BALL SUPPLY

REACTOR

AV. MODERATION RATIO

N_C/N_{HM}

FUEL RESIDENCE

DAYS

CONVERSION RATIO

AV. BURNUP

MWD/T

INVENTORY U-233 / U-235 KG/GW_{TH}

SUPPLY U-233 / U-235 KG/GWO

DISCHARGE U-233 / U-235 KG/GWO

BALANCE U-233 / U-235 KG/GWO

BREAK DOWN:

MODERATION RATIO

N_C/N_{HM}

BURNUP

MWD/T

MAXIMUM POWER

KW/BALL

INVENTORY U-233 / U-235 KG/GW_{TH}

SUPPLY U-233 / U-235 GR/GWO

DISCHARGE U-233 / U-235 GR/GWO

BALANCE U-233 / U-235 GR/GWO

STORAGE U233/U234/U235/U236/U238

Note: One prebreeder plant supplies U-233 for 10.5 near breeder plants.

Figure 3-3. Prebreeder/Near Breeder (KFA DATA)

TABLE 3-7

NEAR-BREEDER SYSTEM
(KFA DATA)

PARAMETER	UNITS	PREBREEDER				NEAR BREEDER		
		Fuel Type			AVG TOT	Fuel Type		AVG TOT
		1	2	3		1	2	
<u>OVERALL REACTOR</u>								
Power Level	MWth					3000		
Conversion Efficiency	-					0.40		
Core Height	m					5.5		
Core Diameter	m					11.78		
Power Density	MWth/m ³					5.0		
Core Outlet Temperature	°C					985		
Pressure Level	b					~40		
<u>FUEL DESCRIPTION</u>								
Type Fabrication	Hot/Cold	H	C	C	--	C	H	--
Type Ball	Conv/Shell	Conventional						
Coated Particle Diameter	μm	200	400		--	400		--
Average Moderator Ratio	N _C /N _{NHM}	2100	735	110	198	110		
Average Heavy Metal Loading	g/ball	1.8	5.0	32.4	17.9	32.4		
Initial Enrichment	½ N _F /N _{NHM}	43.1	24.0	0.0		0.0	10.6	
Fractional Supply	½	15	36	49	100	68	32	100
<u>OVERALL RESULTS</u>								
Average Burnup	MWd/kg	23.2				24.0		
Fuel Residence Time	F.P. Days	451				833		
Conversion Ratio	-	.739				0.972		
Fast Dose to Balls E>10 ⁵ eV	10 ²¹ /cm ²	1.56				2.92		
Ore Requirements (0.25% Tails)	kg/GWde	561				Nil		
Separative Work	kg SWU/GWde	510				Nil		
Net Ore Requirements	kg/GWde					48.8		
Separative Work	kg SWU/GWde					44.3		
(at equilibrium with one prebreeder per 10.5 near breeder)								

The first cycle, the MOR (Mixed Oxide Recycling), uses the recovered uranium, including U-233, U-235, U-236, and U-238, to mix with fresh thorium in 28 percent of the feed balls. The remaining 72 percent of the balls are made of fresh, highly enriched uranium and thorium. The U-236 builds up continuously, and decreases the conversion ratio from approximately 0.59 in the early years of operation to approximately 0.57 later in the life of the reactor. This particular cycle has a conservative fuel loading of 11.2 g heavy metal per ball, like the THTR fuel, and a single (THOREX) reprocessing plant.

The second cycle, the RSE (Recycle in Separate Elements), achieves the removal of the U-236 by means of separately recycled fuel balls which contain no thorium, and which are discarded after one pass through the reactor. This achieves the removal of the U-236 at a cost of 3.7 percent of the daily fissile uranium rate. In addition, the fuel loading of the fresh balls is increased to 15 g heavy metal per ball, and both PUREX and THOREX reprocessing plants are required. The conversion ratio reaches 0.62 over the cycle life.

The third cycle, the SFB (Separate Feed and Breed Elements), completely separates the feed thorium, and its bred U-233, from the fissile feed elements with enriched uranium. The thorium portion of the cycle is operated like the MOR cycle. Fresh thorium is added to the bred U-233, which is then recycled. The uranium portion of the cycle operates like the RSE cycle in that the uranium is processed once for recycle and then discarded to remove U-236 from the cycle. Only 0.2 percent of the initial loaded U-235 is spent in removing the U-236. A conversion ratio of 0.65 is achieved. The ore requirement is only 61 percent of the thorium throwaway cycle described in the previous section. The fuel loading has been increased to 20 g heavy metal per ball in the recycle circuit, as well as requiring both PUREX and THOREX reprocessing.

In the above three cycles, two independent changes were being made—first the change in cycle type, and second, a change in heavy metal loading. A separate case, the fifth in Table 3-6, shows the effect of increasing the heavy metal loading of the MOR and/or RSE cycles from their average of 11.2 g/ball to 16.5 g/ball. The conversion ratio increases from approximately 0.6 to approximately 0.7,

with a similar improvement in the ore requirements. Obviously, the ability to increase the heavy metal loading provides a significant improvement in performance.

This is also shown by the PB (Prebreeder), the fourth case in Table 3-6. This system, designed for use with the NB (Near Breeder), has a conversion ratio of 0.74 and is operated like the SFB. It achieves this high value by going to 32.4 g heavy metal per ball, and by decreasing the burnup to 23.2 MWd/kg. The bred U-233 is removed rather than recycled, and used as feed material for the near breeder.

In summary, the high conversion ratios are reached by using the same standard reactor plant with improved fuel loading and additional reprocessing complexity.

3.2.3 NEAR-BREEDER CYCLES

The near-breeder cycle, shown in Table 3-7 and diagrammed in Figure 3-3, achieves a conversion ratio of 0.97. In addition to using a high heavy metal loading of 32.4 g/ball, the burnup has been reduced to approximately 24 MWd/kg. This case is discussed more fully in references 16 and 17.

It requires approximately 8.5 years of prebreeder operation to generate enough U-233 for one near breeder (including enough for 1.5 years' disloading at 0.7 capacity factor). At equilibrium, only one prebreeder is required to supply the U-233 makeup for 10.5 near breeders.

This cycle has the virtue of using the same standard plant as the previously described cycles. The gains are due solely to the fuel, fuel management, and reprocessing complexity. The changeovers required between cycles can be made on line.

3.2.4 NET BREEDER CYCLES

Net breeder cycles have been proposed both by General Electric⁽¹⁾ and by KFA⁽²⁰⁾. Table 3-8 shows the effect of making changes in reactor and fuel cycles to reach high conversion ratios and to achieve net breeding.

TABLE 3-8

INCREASED CONVERSION RATIO FOR PEBBLE BED REACTORS

FUEL CYCLE	CONDITIONS	CONVERSION RATIO	REMARKS
1. Normal Converter Reactor	5 MW/m ³ 100,000 MWd/tHM 11.2 g HM/ball $N_C/N_{HM} \sim 325$	0.6	<ul style="list-style-type: none"> no problem reference cycle (THTR fuel)
2. Recycle Reactor	5 MW/m ³ 93600 MWd/tHM 16.5 g HM/ball $N_C/N_{HM} = 200$	0.71	<ul style="list-style-type: none"> no problem increase heavy metal loading (new THTR fuel)
3. Recycle Reactor	5 MW/m ³ ~100,000 MWd/tHM ~30 g HM/ball $N_C/N_{HM} \approx 120$	~0.8	<ul style="list-style-type: none"> further increase in heavy metal separate feed-breed balls
4. Near Breeder	5 MW/m ³ 24000 MWd/tHM U-233 fuel 32.4 g HM/ball $N_C/N_{HM} = 110$	0.97	<ul style="list-style-type: none"> increase further heavy metal loading U-233 fuel needed
5. Near Breeder	2.5 MW/m ³ 24000 MWd/tHM U-233 fuel 35 g HM/ball $N_C/N_{HM} = 110$	1.0	<ul style="list-style-type: none"> increase further heavy metal loading U-233 fuel needed reduce power density
6. Breeder	5 MW/m ³ 10000 MWd/tHM U-233 fuel 35 g HM/ball $N_C/N_{HM} \sim 80$	1.015	<ul style="list-style-type: none"> reduce burnup increase further heavy metal loading
7. Breeder	4.5 MW/m ³ 10000 MWd/tHM U-233 fuel 45 g HM/ball radial blanket $N_C/N_{HM} \sim 80$	1.05	<ul style="list-style-type: none"> add radial blanket look at cold streaks
8. Breeder	5 MW/m ³ 10000 MWd/tHM U-233 fuel 45 g HM/ball radial blanket top thorium blanket $N_C/N_{HM} \sim 80$	1.06	<ul style="list-style-type: none"> add top blanket remove top blanket periodically
9. Breeder	5 MW/m ³ 24000 MWd/tHM 32.4 g HM/ball $N_C/N_{HM} \sim 110$ Decoupled fertile throughput	1.1	<ul style="list-style-type: none"> new technology for core (balls, flow of balls, control rods) high pressure drop, lower efficiency, greater blower power

Figure 3-4. Conversion Ratio versus Burnup (KFA DATA)

Early work at KFA⁽¹⁴⁾ has shown that conversion ratios greater than 1.0 could be achieved by means of increased heavy metal loading and low burnup, using a reactor fueled solely by U-233 from prebreeders. More recent work⁽²⁰⁾ has increased the conversion (breeding) ratio by means of a radial blanket of thorium-containing balls. Figure 3-4 shows these results. (These are also cases 1-7 in Table 3-8.)

For the lower set of cases, a cross plot, Figure 3-5, shows the conversion ratio versus carbon-to-heavy-metal ratio for constant burnup. It appears that heavy metal loadings of about 45 g/ball are optimal.

Figure 3-5. Conversion Ratio versus Moderation Ratio

The upper curves show the more recent results⁽²⁰⁾. The data is also shown in Table 3-9. For all of these cases, a modest breeding gain is achieved, at the cost of rapid recycling and large fissile inventories. Fuel cycle costs will depend strongly on the treatment of inventory costs and the equivalent value of the U-233. The development of high heavy metal fuel loadings is believed to be feasible (see Section 2.2).

Case 8 in Table 3-8 is a conceptual design in which the upper reflector is replaced with a thorium-containing blanket. This concept has the disadvantage of requiring periodic removal of the upper blanket for reprocessing, and hence represents a break in the basic concept and design of the pebble bed reactor.

TABLE 3-9

NET-BREEDER FUEL CYCLES

(KFA DATA)

Parameters	Units	Case			
		1 (110/20)	2 (110/10)	3 (80/20)	4 (80/10)
Moderation Ratio	N_C/N_{HM}	110		80	
Heavy Metal Loading	g/ball	32		44	
Average Power-Density (Including blanket)	MW/m ³	4.5			
Uranium Enrichment	% Fissile	92			
Average Burnup (Including Blanket)	GWd/Mg _{HM}	20	10	20	10
Fuel Residence Time	Full Power Days	765	382	1052	526
Breeding Ratio	-	1.014	1.038	1.026	1.052
Core Inventory-Fissile Uranium	kg/GWe	2792	2666	4044	3827
Cycle Inventory - Fissile (80% Load Factor - 1 year out-of-pile)	kg/GWe	3915	4819	5218	6059
Fissile Discharge	kg/GWde	3.85	7.38	4.02	7.64
Breeding Gain @ 1% Reprocessing Losses	g/GWde	2	40	32	78

The concept described by General Electric in reference 1 differs from the KFA concept in that two sizes of fuel elements are used. Most of the fertile thorium is placed in small balls which can flow through the interstices of the fissile elements at a different rate of speed.

The original nuclear analysis, since confirmed at KFA and ORNL, was based on a perturbation analysis of the KFA near breeder.

The indicated neutron loss reduction from the PBNBR* is accomplished as follows:

- Increase the pebble bed H/D ratio from .48 to a value approaching 1.0.
 - Add interstitial breeder fertile balls to the normal size pebble bed. This will reduce the unfilled core volume from 39 percent to approximately 25 percent, give a very high fertile loading, and blanket the region of the radial and bottom graphite reflectors. The small interstitial balls are also loaded to cover and blanket the top of pebble bed.
 - Decouple the throughput of the interstitial breeder balls from the throughput rate of the larger seed balls. Breeder balls ~~with a diameter 0.154 times that of the seed balls will freely~~ pass through the bed of large seed balls at a rate determined by a dual feed discharge valve system.
-
- Uranium-233 breeding is accomplished by storing irradiated breeder balls in a hold-up vessel to allow the decay of protactinium-233. The breeder balls are stored for 40 to 120 days before being returned to the original or a new core. The breeder balls are not reprocessed until maximum burnup is achieved. This allows breeding at fuel costs which may be equal to or less than near-breeder fuel costs.
 - A steep xenon concentration gradient along the axis of the reactor is developed as breeder balls pass through the core and attain a steady-state xenon concentration. The lower region of the core will be xenon poisoned. The depth of the poison region for constant power operation will be determined by the

*Pebble Bed Near Breeder Reactor

throughput rate of the breeder balls. Therefore, the reactor can be controlled by adjusting the level of xenon poisoning in the core by regulating the throughput rate of the breeder balls. This method of control will incur no additional neutron losses.

The most significant reduction is in the leakage and control losses. These are decreased from 9 percent to 1 percent by means of the combined effects of the presence of the interstitial pebbles (which decreases the migration length and provides fertile blankets on all sides of the reactor), the improved surface-to-volume ratio of the core, and the removal of control rods during normal operations from the original high-flux upper reflector.

The projected loss reductions will give a breeding ratio of approximately 1.1, although development and design changes might be expected to reduce this somewhat during the reactor development period.

Table 3-10 shows a summary of the results of this perturbation analysis of the PBTBR, based on a calculated breeding ratio of 1.095.

TABLE 3-10

RESULTS OF PBTBR STUDY

Breeding Ratio	$\equiv \frac{\text{Fissile Fuel Generation Rate}}{\text{Fissile Fuel Consumption Rate}}$	BR = 1.095
Reactor Fissile Yield	$\equiv \frac{\text{Rate of Reactor Fissile Fuel Discharge}}{\text{Initial Inventory Reactor Fissile Fuel}} - \frac{\text{Rate of Reactor Fissile Fuel Feed}}{\text{Reactor Fissile Fuel}}$	$Y_{R\infty} = 0.035 \text{ (YR}^{-1}\text{)}$
Fuel Cycle Yield (Including processing, fabrication, shipping, and reactor fissile fuel inventories)	$\equiv \frac{\text{Rate of Fuel Cycle Fissile Fuel Discharge}}{\text{Initial Fuel Cycle Fissile Fuel Inventory}} - \frac{\text{Rate of Fuel Cycle Fissile Fuel Feed}}{\text{Fuel Cycle Fissile Fuel Inventory}}$	$Y_{FC} = 0.030 \text{ (YR}^{-1}\text{)}$
Fuel Cycle Compound Doubling Time	$\equiv \text{Time to Double the Fissile Fuel in the Cycle}$	$\tau_{\text{CDT}} = 23.1 \text{ (YRS)}$

Note that the fuel cycle yield is 0.030 year^{-1} , giving a compound doubling time of 23 years.

It is recognized that this is a conceptual design only. It also represents a departure from the current technological development of the pebble bed reactor. Much work would be required to remove uncertainties which exist. These include

- Effects of the expected larger pressure drop across the core
 - Experimental ΔP evaluation
 - Effect of ΔP on pumping power
- Effects of potentially larger control rod insertion forces
- Economic evaluation, including the effect of the large inventories of U-233 which may be needed.

In summary, there are ways to achieve breeding in the pebble bed reactor. The best choice will depend on external incentives and the perception of economics versus ore supplies.

SECTION 4

RESOURCE REQUIREMENT COMPARISONS

Table 4-1 and Figure 4-1 show the required uranium resource for some of the fuel cycles of interest. Points of interest include the very low total ore required for the near-breeder system, as well as the good performance of the low-enrichment cycle with respect to the light water reactors. It should be noted that the "Initial Core" requirement is not a strictly accurate value, since a reactor with on-line refueling builds up to equilibrium over a period of many months. This is especially true for the prebreeder/near breeder where it requires a number of years to achieve equilibrium. Figure 4-2 shows the buildup to equilibrium for a group of 11.5 reactors which switch over, one by one, as enough U-233 is available. Although more ore is required initially, within 25 years the near-breeding systems require the same ore as the recycling HTR, and has used only about 56 percent as much after 60 years.

TABLE 4-1

RESULTING ORE REQUIREMENTS FOR VARIOUS FUEL CYCLES (KFA DATA)

Case	U ₃ O ₈ Requirements ⁽¹⁾ Mg/GWe ⁽³⁾		
	Initial Load	Annual Reload	Lifetime (30 Years)
Low Enrichment Uranium (UOT)	257	133	4247
High Enrichment Thorium (4021)	323	114	3743
Mixed Oxide Recycle (MOR)	274	87	2884
Recycle Separate Elements (RSE)	236	77	2546
Separate Feed Breed (SFB)	304	72	2464
Prebreeder (PB)	346	143	4636
Prebreeder/Near-Breeder (Ratio of 1 PB per 10.5 NB)	700 ⁽²⁾	12.5	1075
PWR (No Recycle)	~417	180	5817
BWR (No Recycle)	~430	182	5890

(1) At 70 percent capacity factor and 0.25% tails.

(2) Equivalent initial ore estimate.

(3) 1 Mg = 1 metric tonne.

Figure 4-1. U_3O_8 Requirement (Load Factor = 0.8. Tail = 0.25%)

An important point has been recently noted. The near breeder, as well as all net breeders, requires high quality U-233 from some sort of prebreeder. The prebreeder described in Section 3.2.2 and 3.2.3 is a low-burnup, highly efficient U-233 "factory". Economically, this reactor may be difficult to justify.* However, the SB93 (Section 3.2.0) and the SB20 (Section 3.1.2) are both prebreeders in that they produce high quality U-233 as a byproduct. Table 4-2 shows the comparison of these three reactors as U-233 producers. Both the SB20 and SB93 cycles appear to have competitive fuel cycle costs with other once-through cases, so that, even though they only produce a little over 20% of the PB U-233, it may be more attractive to build up a stockpile of U-233 with the SB20 or SB93 cycle rather than use the PB.

*See Topical Report No. 1, Cost Benefit Study

Figure 4-2. Ore Requirements for Near-Breeder System for Constant Power Generation. (KFA Data)

TABLE 4-2

PEBBLE BED PRE-BREEDERS

Reactor Designation		SB2U	SB93	PB
Burnup	MWD/kg	101	100	23.2
Fuel Residence Time	F.P. Days	872	1117	451
Conversion Ratio	-	0.56	0.58	0.74
Ore Requirements (0.25% Tails)	kg/GWde	499	462	561
Separative Work	kg SWU/GWde	410	418	510
U-233 Production (Net)	kg/GWde	0.30	0.33	1.42
Ratio of U-235 Input to U-233 Discharge (Net)	-	0.15	0.18	0.64
Fissile Inventory: U-235	kg/GWe	399	432	1018
: U-233	kg/GWe	201	381	358
: Pu	kg/GWe	53	2	-

SECTION 5

FUEL CYCLE CLOSURE

At the present time, the PBR fuel cycle is in much the same situation as the other fuel cycles, such as the LWRs and the LMFBR. That is, much of the technology is developed, but little or no commercial capacity exists. Hence cost estimates and full scale development problems are not well known. In addition, problems of the ultimate disposition of long-lived fission products and the diversions of fissile material for weapons use seriously cloud the issue.

Fuel from the AVR is presently being stored, except for that amount being used for development of reprocessing facilities. Except for the volume involved, the PBR element is well suited for long term storage because of the combination of all-ceramic construction, a mechanically strong structure, and reasonably low specific activity. For cycles which do not involve reprocessing, or for these which defer reprocessing, few short-term storage problems exist. For long-term storage (ultimate disposal), the problems of PBR fuel are much the same as for other reactor fuel cycles.

5.1 REPROCESSING

Reprocessing the fuel involves two basic procedures,

- Head-end processing. This involves breaking up the fuel balls, removing the carbon, and removing the coatings from the fuel kernels.
- Chemical reprocessing. This involves separating the heavy metals from the fission products, and further separation of the various elements of the heavy metals.

The first one, head-end processing, consists of first crushing the elements in a manner which avoids too much particle damage. Tests

have been performed at KFA to characterize this process. After crushing, the carbon is burned off in a fluidized bed. Various techniques are being studied for removal of radioactive gases. One potential problem is that of C-14, produced primarily by activation of nitrogen either in, or adsorbed to, the graphite. The extent of this problem and possible solutions are under investigation.

For elements containing more than one kind of particle, for example, TRISO uranium particles and BISO thorium particles, separation after combustion by either sieving, flotation, or air sifting is being investigated. Note that the BISO particles can be processed directly after separation while added steps are required to remove the SiC coating and inner graphite from TRISO particles.

After the above separations, the THOREX and/or PUREX process, which are well developed, will be used to continue the processing.

Two facilities at KFA are being designed and built. The JUPITER plant (Jülich pilot plant for thorium element reprocessing) aims to demonstrate the technical feasibility of hot HTR reprocessing. This plant will process up to 2 kg heavy metal per day from spent AVR fuel.

The VENUS plant (combustion of HTR fuel elements containing natural ~~uranium fuel and separation of particles~~) will be a cold (nonirradiated) simulation of the head-end process, up to and including the separation of different particles. Its size is based on processing the equivalent fuel from 5000 MWe of reactors.

5.2 REFABRICATION

GHT understands that the SATURN plant will not be operated.

Development of remote fabrication is underway at KFA. The SATURN plant (Semi-Technical plant for Thorium-Uranium refabrication) is a shielded facility for the remote refabrication of coated particles. It is expected to be in operation in the early 1980's. Complete fuel elements will not be fabricated.

It is expected that remote fabrication of ball elements may not be necessary due to the low quantity of fuel per element. This remains to be determined.

Obviously the problems of remote fabrication have not been solved yet, but work is going on, and no insurmountable barriers appear to exist.

SECTION 6

NONPROLIFERATION AND SAFEGUARD CONSIDERATIONS

This section discusses the considerations of nonproliferation and safeguards on the implementation of pebble bed fuel cycles in the U.S. and abroad.

6.1 U.S. NONPROLIFERATION POLICY

The possibility of manufacturing nuclear weapons from nuclear fuel has long been a concern. This concern with respect to national governments motivated the International Atomic Energy Agency, an arm of the U.N., to draft the nuclear Nonproliferation Treaty (NPT) in 1968. To date over 95 nations have signed the NPT. By so doing, they have pledged that they will not use nuclear power for the purposes of making military weapons. In exchange for this pledge, the treaty puts no controls on the peaceful uses of nuclear power.

The possibility of terrorists making weapons from stolen nuclear fuel was emphasized by Willrich and Taylor in 1974. This concern grew rapidly, resulting in strengthened safeguard measures being applied at U.S. nuclear facilities.

With nuclear fuel reprocessing and the accompanying separation of plutonium getting closer to fruition in 1975 and 1976, the concern over nuclear weapons proliferation became intensified. This higher level of concern was reflected in a major policy statement issued by President Ford in November, 1976. The new policy questioned the assumption that reprocessing is necessary and related the decision to use plutonium directly to efforts to minimize weapons proliferation.

The energy plan issued by President Carter in April, 1977 voiced the concern in stronger terms, calling for indefinite moratoria on both nuclear fuel reprocessing and the use of plutonium. These dual moratoria

and the rationales behind them have a major effect on the direction of the nuclear power program in the U.S. and on the emphasis placed on it. They also have a direct impact on the prospects for gas-cooled reactors and the fuel cycles used in them.

The moratoria on reprocessing and plutonium fuel are based on the principle that the separation of fissionable material in weapons quantities by either isotopic enrichment or chemical recovery from spent fuel would be difficult to achieve in a relatively short time unless the facilities for doing so are already available through nuclear power programs. This principle leads to the conclusion that nuclear fuel in general use throughout the world must have the following two properties:

- Uranium content that never exceeds 10 to 20 percent fissile isotopes (U-233 or U-235), and
- Minimum content of plutonium or other chemically separable weapons material only in irradiated (spent) fuel.

The policy further concludes that efforts to limit nuclear fuel in this way will be substantially strengthened by a unilateral application of the same limits to fuel used within the U.S.

It is generally recognized that the nonproliferation criteria listed above, and included in the U.S. energy program, greatly reduce the ~~energy resource available from nuclear power.~~ Uranium resources in the U.S. and the world appear to be severely limited. Since plutonium is the only fissile material currently available to augment the natural supplies of U-235, an extended moratorium on plutonium use can delay the wide application of nuclear power for many decades. Thorium can be used to produce U-233 and extend the uranium resource, but it is seriously questioned in some quarters whether significant quantities of U-233 can be produced before the natural supplies of U-235 are depleted.

The U.S. Department of Energy (DOE) has a sizeable program of evaluation of alternative reactor and fuel cycle concepts to identify those which will achieve higher utilization of nuclear fuels while still being consistent with nonproliferation objectives. Gas-cooled reactors display some advantages in this respect. With greater ease of conversion

to other fuel cycles and a history of thorium-bearing fuels, gas reactors may allow earlier introduction of U-233 fuel. Even without recycle, the improved neutronics of U-233 over U-235 results in better fuel economy if thorium is used in place of some of the U-238.

The U.S. energy program recognizes further that both reprocessing and the use of plutonium fuel eventually may be achieved with low proliferation risks if they are carried out in internationally controlled, highly secured centers. A potential scenario for the use of such centers has been specified by DOE. These centers would contain fuel recycle facilities as well as possible plutonium "burning" breeder reactors. Fuel shipped from the center to outlying reactors would contain low enriched uranium (<20%) either in all-uranium fuel or mixed with thorium (the so-called "denatured" fuel).

6.2 FUEL CYCLE OPTIONS CONSISTENT WITH NON-PROLIFERATION CRITERIA

A number of different fuel cycle options for gas-cooled reactors are considered in this study. The constraints imposed by U.S. nonproliferation policies will affect these options differently with respect to future use and possible dates of introduction. Table 6-1 lists five basic fuel cycles, the nonproliferation constraints that will apply with current U.S. policy, and the possible dates of introduction. Cycles 2 through 5 incorporate thorium fuel in order to increase the utilization of uranium in this type of reactor.

Only the first two fuel cycle types in Table 6-1 meet the currently applicable nonproliferation criteria of no reprocessing. Reactors operating on these cycles could conceivably be built anywhere in the world with low proliferation risk, providing the necessary enrichment services are provided by the U.S. or other nuclear weapons states, and the spent fuel is put in long-term storage.

The third and fourth fuel cycles both require recycling of fissile uranium into "denatured" fuel. The fuel may have virtually unrestricted use but will require, under the nonproliferation policy, the establishment of one or more internationally controlled and secured fuel recycle centers. Since a number of international and institutional problems must

TABLE 6-1

NON-PROLIFERATION CONSTRAINTS APPLYING TO VARIOUS
GAS-COOLED REACTOR FUEL CYCLES

Type of Fuel Cycle	Fissile Material		Fertile Material	Non-Proliferation Constraints	Possible Introduction Date
1. Low-Enriched Uranium, Once Thru (UOT)	U-235 (~ 8%)		U-238 (~ 92%)	Unrestricted Use, Providing There Is No Reprocessing Capability Available	Before 1990
2. Low-Enriched Uranium/Thorium Once Thru (MO20)	U-235 (~ 8%)		U-238 (32%) Th (60%)	Unrestricted Use, Providing There Is No Reprocessing Capability Available*	Before 1990
3. Denatured, Recycled U-235 and U-233/Thorium	U-235 (~ 6%) U-233 (~ 2%)		U-238 (32%) Th (60%)	Unrestricted Use, But Requires Secured Fuel Recycle Center	1995 - 2000
4. Denatured, Recycled U-233/Thorium	U-233 (~ 8%)		U-238 (48%) Th (44%)	Unrestricted Use, But Requires Secured Fuel Recycle Center With U-233 Breeder	Post 2000
5. High-Enriched Uranium/Thorium (MO93, TOT Plus High Conversion Near Breeders and Net Breeders)	U-235 (~ 8%)		U-238 (0.5%) Th (92%)	Use Limited To Secured Centers	1995 - 2000

*Because of the low amount and poor quality of plutonium generated, this cycle may be acceptable anywhere, even if recycling is available.

be overcome before these centers can be built, these cycles do not look possible until after 1995. In fuel cycle 3, the source of the recycled fissile material is the U-235 and U-233 contained in the spent fuel from gas-cooled reactors. Thus, the secured centers are only for fuel re-processing and refabrication. In case 4, it is assumed that the secured centers also have prebreeder or breeder reactors which use plutonium or enriched U-235 or P.B. for fuel and produce additional U-233.

Reactors operating on fuel cycle 5 must be located in secured centers because of the proliferation risk of high-enriched uranium. This restriction could act to severely limit the potential reactor capacity of this cycle. Again, the expected difficulty in establishing international secured centers extends the introduction date of this cycle beyond 1995, with the possible exception of demonstration programs. The use of recycled U-233 in place of all or part of the high enriched uranium in fuel cycle 5 is a good possibility, but the requirement that reactors using this fuel be placed in secured centers still applies.

6.3 EFFECT OF NONPROLIFERATION CONTROLS ON FUEL COSTS

Limiting the free use of nuclear fuel to reduce weapons proliferation risk will not be done without incurring some additional costs. The gas-cooled reactor fuel cycles that meet current U.S. policy requirements (1 and 2 in Table 6-1) are not those expected to give the lowest fuel cycle costs. This is particularly true in the long term as those cycles tend to deplete U-235 reserves more rapidly, and will be more sensitive to increasing uranium costs as the reserves are depleted.

In cases 3 through 5, the safeguard measures and additional transportation requirements associated with nonproliferation constraints will also increase fuel cycle costs. Neither the required safeguard measures nor the sizes and locations of fuel recycle centers have been established well enough to obtain a good estimate of these additional costs. However, judging from safeguard requirements already in effect in the U.S. at plutonium facilities, it is expected that nonproliferation measures and transportation will add a significant fraction to the fuel cycle costs reported in this study. These increases will not apply uniformly to all fuel cycle options nor uniquely to gas-cooled reactors. For ex-

ample, the first two types of fuel cycles in Table 6-1 will be less sensitive to these adders because they are independent of recycle and the need for secured centers.

An additional cost may also be incurred in pebble bed reactors if separate thorium fuel elements (the SB20 cycle in particular) are found to give too great a proliferation risk. The U-233 produced in these balls can be separated into weapons material by chemical processing. The risk of diversion introduced by these separate balls will depend upon the rate that fission products build up in them and the gamma ray source levels that accompany U-233 production. A requirement to mix thorium with U-238 (denaturing of U-233) could result in an increase in fuel costs for the pebble bed reactor and probably prevent the development of a net breeder.

The concept of locating all fuel recycle facilities and reactors that use highly enriched uranium or plutonium fuel in secured centers raises the possibility of control of trade and supply. Ideally, such centers would be owned and operated by a large number of nations. In practice, the difficulties in overcoming international and institutional obstacles may lead to a relatively small number of nations in control of fuel recycle. In such a situation, prices for fuel processing could be higher than would be the case with no proliferation constraints. As with the other effects on fuel costs, this effect cannot be estimated at this time, but it does remain a distinct possibility.

In summary, the constraints imposed by the current U.S. weapons nonproliferation policy are expected to result in both increases in fuel cycle costs for gas-cooled reactors and in delays in commercialization of certain fuel cycles until future nonproliferation regulations are put into force. Uncertainty in the requirements makes it nearly impossible to estimate the magnitude of the cost increases at this time.

6.4 FUTURE TRENDS IN NON-PROLIFERATION POLICY

The above discussion of the effect of weapons nonproliferation constraints on the choice of fuel cycle and fuel cycle costs is based entirely on currently stated U.S. policy. With a continually changing world nuclear power scene, the possibility is quite great that signifi-

cant changes will occur in U.S. policy over the next few years. This is particularly true with respect to how it affects the domestic use of nuclear power.

The effectiveness of the unilateral moratorium on reprocessing in the U.S. will be measured by whether other supplier nations postpone fuel reprocessing until international nonproliferation safeguard measures are established. There are indications that the U.S. policy has had some delaying effect in the other nations. However, the economic pressure for obtaining maximum efficiency from nuclear fuels is much greater in the other nations than in the U.S., where there are large reserves of uranium and coal. It is likely that some agreement will be reached in the next few years whereby fuel reprocessing will occur in the supplier nations subject to international controls. (Actually, agreement has already been reached to operate some newly completed reprocessing facilities). As other nations begin to commercially reprocess fuel, the U.S. may well follow suit.

Such an occurrence would open the recycle fuel options (types 3 and 4 in Table 6-1) for domestic use in the U.S. The fully enriched fuel cycles, type 5, would likely also be permitted in that event.

Other changes in U.S. nonproliferation policy, including some of major importance, can also occur in the future. Uncertainties in future trends in the policies of the U.S. and other nations make it difficult to anticipate the impact that nonproliferation efforts will have on commercial gas-cooled reactors. With the current U.S. policy, substantial constraints and cost impacts would be expected. Future changes may result in very different impacts by the time gas-cooled reactors are built on a commercial basis.

SECTION 7

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