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THE GAS COOLED HIGH TEMPERATURE REACTOR: PERSPECTIVES, PROBLEMS AND PROGRAMMES

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1. The HTR and its potential for electricity and nuclear process heat production

For nearly 2o years, extensive r+d programs have been carried out on He-cooled high-temperature reactors in several countries, in particular in the FRG and in the USA, but also in the UK, France, and Switzerland. As result of the longstanding efforts, satisfactory technical solutions have been found for many of the basic problems of this reactor system. Three small experimental plants - Peach Bottom in the USA, DRAGON in the UK and AVR in the FRG - have been operated successfully over extended periods of time, the AVR still being in operation.

Based on this experience, in the early seventies the commercial market introduction of the steam cycle high temperature reactor, HTGR, as a competing system to the LWR seemed possible. The construction of demonstration power plants, FSV and THTR-3oo, of about 3oo MWe size each was started in the US and in Germany, and commercial size plants were designed. At that time the GAC in the US began its efforts to market the HTGR and a total of five orders for ten reactor plants of 77o to 116o MWe size were received by mid 1974 Also in the FRG and in Switzerland orders for 11oo MWe stations were expected in 1975.

In 1975 and early 1976, however, all these orders in the US were cancelled and the expected orders in Europe were not realized. The cancellations were due primarily to a general slowdown in electric power demand coupled with the increased costs of these plants and a shortage of financing for the utilities. It was recognized that a market introduction of the HTGR on purely commercial basis was not possible, and that a substantial governmental support would be required to make a second, successful approach. Spending of large amounts of public money, however, needs a strong justification. Reviews were therefore intitiated in both countries to evaluate the market potential of HTR systems and to make a cost/benefit analysis. The reviews, though not yet completed, reveal that this reactor system offers major potential advantages as a source of electricity and/or of nuclear process heat for the following reasons:

1. The HTR-system shows a high nuclear fuel conversion efficiency, permitting a better utilization of uranium and in particular of thorium reserves. It also shows a great flexibility in the mode of fuelling allowing an adaption to various boundary conditions such as availability or non-availability of highly enriched uranium, existence or non-existence of a closed fuel cycle etc.

2. The HTR-system offers a high degree of inherent safety and thus a good potential for adoption to very strict safety standards.

3. The HTR-system permits high-efficiency electricity generation using either the indirect steam cycle or the direct cycle with a closed He turbine. Dry air cooling can be employed without major economic penalties, especially in case of the direct He cycle.

4. The HTR-system has the unique and proven potential to supply Helium of 95o - 1ooo^{o}C, and of even higher temperatures if the necessary materials will be available. This permits the direct use of nuclear heat for the production of gaseous or liquid secondary fuels from coal and other fossil fuels. On a more extended time scale thermochemical water splitting seems a realistic goal opening an enormous new market for nuclear energy.

If the HTR is to proceed to commercialization, new industrial arrangements will have to be made and the private sector will have to be supported by the public sector in sharing the costs and risks associated with the development and introduction of this technology. Due to the advanced state of technological development, the time, man-power, and budget requirements for the market introduction including further development and provision of the necessary infrastructure of the whole system will be limited and within reach of a group of major industrialized countries.

2. Status of HTR technology

2.1. Alternative design concepts

The technology of HTR for electricity production has been described in various publications and will therefore not be repeated here in any detail [1 - 1o]. Table 1 shows the main features of the experimental and demonstration reactors DRAGON, Peach Bottom, AVR, Fort St. Vrain and THTR-3oo. While both FSV and THTR-3oo use prestressed concrete pressure vessels to contain the reactor itself and the components of the primary system, the core design differs: The FSV core consists of large prismatic fuel elements of hexagonal cross section, the THTR-3oo core consists of a bed of spherical fuel elements of 6o mm diameter. Both types of fuel elements contain the fissile and fertile materials in the form of coated particles consisting of a kernel of carbides or oxides of uranium or thorium or of a uranium-thorium mixture (about o.2 to o.6 mm in diameter) surrounded by an effective fission product barrier, comprising several different pyrolytic carbon (PyC) layers (BISO) and, where necessary, an additional SiC-layer (TRISO) [11]. For reasons of good heat transfer, mechanical integrity and safety under reactor operating conditions, the coated particles are homogeneously dispersed in a graphite matrix both in the spherical and the block-type fuel elements. The irradiation performance of the coated particles and of whole spherical fuel elements has been shown to be satisfactory. Experience with whole block type fuel elements is now being gathered in the FSV reactor.

The main differences in the design of the pebble bed and prismatic core resp. are pointed out in fig. 1. The main differences are the loading and unloading systems - influencing also the design of the bottom reflector - and the control rod system. The spherical fuel elements are continuously added through pipes from the top and withdrawn through special holes at the bottom of the core (on-load refuelling). The block type fuel elements are loaded and unloaded from the top by a fuel handling machine (off-load refuelling). The control rods in both cases are moved into the core from the top. Stronger forces have to be applied to the control rods in case of the pebble bed core,however, because the rods are driven directly into the bed of spherical fuel elements. Control rod guide tubes are not possible as they would perturb the flowing behaviour of the pebble bed. Large scale testing has proven the feasibility of both designs.

The prismatic core has the advantages of a well defined geometry, the accessibility to each individual fuel element and a more conventional control rod system. The pebble bed core has the advantages of on-load refuelling, of greater flexibility in the mode of operation with changing types of fuel, and the potential of reaching higher outlet temperatures with a fixed maximum fuel temperature. The latter feature is of special importance to nuclear process heat applications. It is expected that the power costs of HTRs of both designs will be essentially the same. Commercial size plants based on the prismatic fuel concept were designed by GAC and HRB in the early 197oth (HTGR, HTR-116o), larger size pebble bed reactors are now under design in the FRG both for electricity production and process heat application (see table 1).

2.2. Operation experience of experimental reactors

The small 2o MWth DRAGON reactor experiment of OECD at Winfrith/UK has in its 1o years of operation heavily contributed to the basic technology of high temperature reactors until it was shut down in autumn of 1975. Coated particle development, long-time irradiation testing of fuel elements, testing of high temperature materials and components were performed and at the same time an overall availability of 78 % was achieved.

The Peach Bottom No. 1 Nuclear Generating Station (4o MWe) was the first US HTR developed to produce electricity. It operated successfully for seven years from 1967 to 1974 at a reactor outlet temperature of 1382^{o}F (75o^{o}C). While operating the plant had an overall lifetime availability of 88 % and a gross plant capacity factor of 74 %. The plant is now being decommissioned and examination of the fuel and plant components are scheduled to be completed in 1977.

The AVR reactor at Jülich reached full power of 15 MWe in May 1967. In the 1o years since it has been used as a test bed for various types of spherical fuel elements (low enriched uranium, U/Th, mixed oxide , carbide fuel, BISO and TRISO coating) and has demonstrated the excellent fission product retention capability of these fuel types. After seven years of operation at 75o^{o}C Helium outlet temperature the temperature was raised to 95o^{o}C in 1974. Even at this extreme temperature the fission products in the cooling gas remained at a very low level showing the feasibility of the HTR for both direct cycle operation and nuclear process heat.The overall time availability of the AVR until end of 1976 was 78 %. By end of 1976 it had produced 85o million kWh of electricity.

2.3. FSV

Construction of the FSV reactor started in 1968/69. The plant was built by GAC for Public Service of Colorado in a remarkably short time. The initial series of startup tests (the Zero Power series) was completed satisfactorily in late 1974, and confirmed the readiness of the reactor systems to proceed into higher power nuclear testing. The predicted core reactivity and reactivity distribution were confirmed. During this test series there were some problems which delayed testing such as malfunctions of the control rod drives, leakage of hydraulic systems, helium circulator, seal problems and the sticking together of fuel elements. In general these were not problems of fundamental deficiencies in the technology or HTGR concept but were associated with first-of-a-kind design, manufacturing, and operating problems common to many new systems.

A major startup problem occured in August of 1974 when cracks were discoverd in the Pelton wheel of one of the helium circulators. The Pelton wheel is a water turbine drive for the circulator which is used for emergency cooling of the core or for cooling during refuelling. The cracked wheels made of cast INCO 718 were replaced on all four circulators with higher strength forged INCO 718 Pelton wheels. This replacement was completed by the end of 1974.

At the beginning of the next series of startup tests (the power escalation series) in January 1975, a sudden, large increase in the moisture content of the helium was detected indicating that a large quantity of water had leaked into the reactor vessel. About 15.ooo l of water were removed from the reactor during the next month and a half. The cause of this large leakage was determined to be leakage through a closed water supply valve to a Pelton wheel and blockage of the drain from the Pelton cavity which forced open a static shutdown seal in the circulator which was provided to prevent leakage. Modifications were made to prevent recurrence of this problem.

After completion of moisture removal operations and repair of some moisture damage to the control rod drives, the startup tests were resumed in April. Testing proceeded up to the 2 percent power level. Results were satisfactory except that four vessel top head liner cooling tubes, which provide cooling to the control and orifice assembly penetration liners, had higher than normal water temperatures. The cause is believed to be hot helium flowing between the control and orifice assembly shroud and the penetration liner, and into the assembly through access holes and control rod holes. All the assemblies are being modified to restrict this helium flow.

In May an audit of the essential, safety-related electrical cables by the Nuclear Regulatory Commission (NRC) and subsequently by the plant owner revealed that a substantial number of cables did not meet the safety design criteria for separation and protection. Necessary cable modifications were completed in December and startup resumed again in January 1976.

Startup testing continued in 1976 and in June 28 % power operation was achieved when circulator and steam valve problems were encountered. After overcoming these problems electric power was produced for the first time in December, 1976. It is expected that the plant can proceed to high power testing in early 1977 and reach commercial operation sometime in 1977.

2.4. THTR-3oo

Construction of the German demonstration plant THTR-3oo started at Schmehausen near Hamm close to the Ruhr district early in 1972. The plant is being build by the consortium HRB, BBC, NUKEM for a group of utilities, HKG, headed by VEW of Dortmund [12] . Completion of the plant was originally expected in 1977 but major design changes, mainly caused by added safety requirements following new regulations in the LWR area, were experienced during the construction so that start of operation is now expected only in 198o. Among others, those added requirements concerned after-heat removal, earth quake protection, and protection against air plane crashes. With the plant already in an advanced state of construction it was - and still is - very difficult to make the appropriate changes and/or to proof the capability of the design to cope with the added requirements.

By end of 1976 all buildings - including the huge dry air cooling tower - were essentially complete. Also the PCRV has been completed, the tension cables, the liner as well as the insulation of the hot gas ducts of the primary circuit have been mounted. All essential components (main blowers, turbogenerators, feed water pumps etc.) are ready for shipment at various manufacturers or are in the manufacturing process (steam generators, control rods) according to the foreseen time schedule. The production of the fuel elements for the first core loading was started in 1973 and completed in early 1977.

2.5. Advanced HTRs for electricity production

Development work aiming towards the direct coupling of a gas turbine to a HTR was initiated around 197o in the USA by GAC. At about the same time the HHT project was started by KFA, BBC, NUKEM and HRB in Germany und by EIR and some industrial groups in Switzerland [13].

The advantages of a direct cycle HTR plant are, firstly, the full utilisation of the gas outlet temperatures achievable in the reactor associated with improved economy and, secondly, the wide variation range of the lower process temperature, resulting in greater flexibility in the design of the cooling tower system and in wider possibilities of utilisation of the waste heat. The integration of the reactor coolant cycle with the turbine cycle reduces the number of components. The very compact helium turbine makes it possible to enclose the entire energy conversion system into the burst proof PCRV.

A comprehensive r+d program was carried out in the FRG in parallel to the planning work for a large size HHT plant. Main areas of the development work were:

- materials for hot gas ducts and turbine blades
- behaviour of fission products in the primary circuit
- gas duct system (insulation, fluid mechanics performance, vibrations etc).
- closed He turbine
- PCRV
- heat exchangers
- improved fuel elements

The r+d work performed in the frame of the HHT project has proven the feasibility of the concept and has added remarkably to the general HTR technology. An important testing facility for the closed He turbine was built at Jülich, and is presently undergoing startup testing. At this HHV facility tests at mass flow rate up to 2oo kg/sec and very high temperatures (85o to 1ooo ^{o}C) are possible.

The design work for a direct cycle HTR plant was based on the assumption that its realization would be preceeded by a series of steam cycle HTR plants. It therefore followed closely the overall design of the HTGR 116o of GAC, leading to a common HHT-GAC 1o8o MWe plant concept with 3 loops [14]. In 1975 it was recognized by the German-Swiss project, however, that the costs and also the thermal efficiency of this concept failed to yield convincing improvements as compared to the HTGR. In view of this fact and of the changed overall situation of the HTR no detailed design work was started on the HHT. Studies directed towards improved concepts showed that an integrated design with only 1 loop at 85o ^{o}C - 95o ^{o}C would be technically feasible and economically attractive. Design work at GAC was continued with an improved 3 loop concept. Both groups are cooperating closely and comparing their respective results.

Work on direct cycle HTR now has been reduced both in the US and in Europe pending the decisions to be taken in 1977/78 on the future HTR concept.

In 1976 work was reactivated in the FRG on the concept of a commercial size steam cycle HTR with spherical fuel elements (HTR-K). Requests from the utilities for improved inspectability, maintainability and final decommissioning led to a reevaluation of non-integrated designs and to the investigation of the prestressed cast iron vessel (PCIV) concept with hot liner. Out of a number of concepts a design with a central PCIV containing the core surrounded by several directly connected smaller PCIV's for the blowers and heat exchangers was selected for a more detailed design study at HRB. This concept as well as the HTGR, the HHT and a pebble bed design with PCRV will be included in the selection process to be started towards the end of 1977 in the FRG.

2.6. The development of HTR for process heat application

The FRG - as other industrialized countries - is confronted with an increasing dependence on imported primary energy, especially oil and natural gas, putting a burden on the balance of payment and leading to risks in the continuity of supply. While nuclear energy based on the LWR now is starting to substitute oil and gas in the electricity market, the HTR offers the possibility to open also the large heat market to nuclear energy. In the FRG in the medium term the gasification of lignite and hard coal, using nuclear heat, is seen as a viable way to cover an increasing part of the energy demand.

In cooperation of coal industry (Bergbau-Forschung GmbH, BF, and Rheinische Braunkohlenwerke AG, RBW), reactor manufacturers (Gesellschaft für Hochtemperaturreaktor-Technik mbH, GHT, and Hochtemperatur-Reaktorbau GmbH, HRB) and the Kernforschungsanlage Jülich GmbH (KFA) a Project Nuclear Process Heat (PNP) was formally established in 1975, aiming at the development and construction of a demonstration plant for nuclear coal gasification using a HTR as heat source.

In the first phase of the project, which was completed by end of 1976, the concepts and design parameters of a commercial size coal gasification plant were fixed (see table 1) and the design of a smaller demonstration plant of 75o MW thermal capacity was initiated. The reactor design is based on the pebble bed concept with OTTO loading: the spherical fuel elements enter the reactor at the top and then flow once through the reactor. In this way a uniform burn-up of the fuel elements is achieved and neutron flux-, power-, and temperature distributions are such that a helium outlet temperature of 95o ^{o}C can be obtained with a maximum temperature of the coated particles of only 1o5o ^{o}C - 11oo ^{o}C.

A comprehensive r+d program is under way in the FRG including the development of the nuclear system and its components as well as the development of the gasification plants. Major parts of the r+d program concern high temperature materials testing, tritium and hydrogen permeation through metallic walls, experimental verification of fuel element flow and of shut down rod behaviour in a core model, and component development and testing. Two rigs of about loo kg C/h each for the gasification of hard coal and lignite by steam and by hydrogen respectively are in operation at BF and RBW. The heat transfer from the primary helium circuit in a fluidized bed for steam gasification occurs via an intermediate helium circuit. The development of an IHX, therefore, is a precondition for the application of this gasification process. The gasification with hydrogen is carried out in such a way that the nuclear heat is used for the production of hydrogen by steam reforming of methane at high temperature in a tubular reformer furnace and for various auxiliary processes at lower temperature.

The amount of hydrocarbons produced by nuclear gasification is nearly twice as high-based on the amount of coal employed - as compared to conventional processes. Electricity can be produced as a by-product, and the amount of CO_2 released to the environment is markedly reduced. An economic evaluation of the market potential confirmed the expectation that nuclear coal gasification can lead to lower gas costs than does autothermal gasification of coal, and that after the market introduction, i.e. towards the end of the century, the products of nuclear gasification can be competitive in the FRG with oil and natural gas [15]. There will be also a market for liquid products such as methanol and for synthetis gas (CO, H_2) in the chemical and steel industry respectively. The market introduction of nuclear process heat in the FRG would be strongly supported by the existing infrastructure: the national natural gas pipe system and major regional synthesis gas pipe systems.

The HTR in connection with the reformer furnace can also be directly used for energy supply by the so-called chemical heat pipe. In the steam reforming of methane by nuclear heat, chemical energy is brought into a closed gas circuit. After cooling the resulting gas mixture, which consists of carbon monoxide and hydrogen, is transported to the consumers. The utilization of the transported heat occurs through the synthesis of methane from the gas mixture, during which the applied reactor heat is released again. The methane so produced is led back to the nuclear plant in a closed circuit. The methanation process in which the reactor heat is released occurs at a temperature of apprimately 5oo to 6oo ^{o}C, so that there is a broad spectrum of applications e.g. for electricity production, for household heating and for process steam in industrial firms. A larger number of consumers situated at distances up to loo km from a large nuclear plant can be supplied. This concept is now being developed by KFA and RBW. A small test facility, EVA I, has already demonstrated the steam reforming process of methane for several years and allows an optimization of the process parameters and catalysts. It will soon be coupled to a methanation facility, ADAM I. An electrically heated demonstration plant of lo MW capacity, EVA II/ADAM II, has been ordered and will be operational in 1979 at KFA.

In the FRG the gas market and the market for nuclear distant heat has been estimated to be in the order of 1o plants of 3ooo MWth nuclear capacity by the year 2ooo and in the order of 5o such plants by the year 2o25. It is expected that further development of this technology then will allow the thermochemical splitting of water for which a very big market would be available in many countries.

2.7. HTR safety analysis

One of the attractive features of the high temperature reactor is its high degree of inherent nuclear safety. This results from the use of a gaseous (one-phase) coolant, the high heat capacity of the graphite core, the low primary circuit activity level, the negative temperature coefficient of reactivity, the large temperature difference between operating and dangerous conditions, and the characteristics of the PCRV.

GAC has performed a preliminary probabilistic risk analysis of its HTGR 116o [16] showing that indeed the initiating events considered in the study - which among others include loss of coolant pressure and water ingress into the primary circuit - do not lead to very serious consequences because the accident proceeds on a slow time scale (due to the properties of the fuel elements) and because there is no mechanical failure of the containment. As a consequence, there was no prompt emission of fission products out of the containment, and no lethalities would occur even at accident probabilities as low as $1o^{-9}$ per year.

The AIPA-Study has been reviewed and commented bei NRC and will be improved and revised in the future. It will also form the basis of a probabilistic risk analysis now being started in the FRG as part of its HTR safety program. If the results of GAC can be confirmed, the HTR indeed can be considered a very low risk nuclear power system.

3. HTR fuel cycle considerations

In principle, the HTR can be operated with a variety of fuel cycles including the low enriched uranium cycle and the thorium cycle. Work in the US and in the FRG has been devoted primarily towards the development of the thorium fuel cycle. Interest in the thorium fuel cycle exists primarily because that cycle provides improved fuel utilization. However, at the same time, the thorium cycle uses highly enriched uranium and requires the development of an economic fuel recycle technology. The status of thorium fuel cycle technology is presented in several other papers to this conference.

A large number of studies have been conducted comparing the uranium and thorium fuel cycles in HTR's. These studies generally show that under the design and economic conditions employed, use of the thorium fuel cycle yields fuel cycle costs which are slightly below those for the uranium cycle, on the basis that economic fuel recycle plants are available. The difference in economic performance between the two fuel cycles is primarily due to the different fuel utilization characteristics of the cycles. The thorium cycle generally has a fuel conversion ratio higher than the uranium cycle (with recycle in both cases), leading to less net fuel feed requirements for the thorium cycle. Thus, fuel utilization differences between the two cycles favor the thorium cycle since the fuel-makeup need dominate the 3o-year fuelling requirements of a given reactor.

Teuchert et al.[17] have shown that for the pebble bed HTR a wide choice is available for the loading of the fuel elements and for the reactor fuelling scheme, reaching from low enriched uranium fuel without recycling to mixed uranium-thorium oxide fuel, to separate feed and breed elements and finally to the pre-breeder/near-breeder system. The feed-breed cycle variant allows to prebreed U-233 for the near breeder variant, which achieves a conversion ratio of o,97 and thus leads to a very efficient use of the fissile and fertile materials.

Fig. 2 shows the demand for uranium ore and for separative work for various reactors as a function of burnup. In the calculation, reprocessing losses of 1 % were assumed, leading perhaps to somewhat optimistic figures. Within the existing uncertainties in the cost assumptions, fuel cycle costs are found to be comparable for all HTR cycles considered including the near breeder. From the point of view of fuel conservation one should therefore at the long run aim at the pre-breeder/near-breeder cycle. With the pebble bed reactor, it is possible to start with one fuelling scheme and go without engineering changes or economic penalties later on gradually to another scheme. For instance, a first HTR could be optimized for U/Th fuel without recycling leading to a design burnup of loo GWd/to (point A). The spent fuel can be stored for an extended period of time (in principle for very long times, because of the excellent fission product retention of the coated particle fuel) until the fuel cycle will have been closed. Then the fuelling scheme can be changed to U/Th recycle (point B) and, after sufficient U-233 has been accumulated to the near breeder system (point C).

If, for any reason, no highly enriched uranium would be available in the initial phase of reactor operation, without immediate economic penalty the operation point D can be chosen. In this case relatively little fissile material is left in the discharged fuel elements. Recycling of this type of spent fuel would not be economically worthwhile, and throw-away-cycle therefore might be considered as an alternative. Even at this mode of operation, the U ore consumption would be as low as in a CANDU reactor and appreciable lower than in a LWR without Pu recycling.

One can conclude, that for a limited period of time HTR's can well be operated without recycling, if this was considered a means of alleviating the market introduction of this new reactor system. The risk to the utilities could be further reduced, if direct long-time storage of spent HTR fuel elements could be accepted. From a technical point of view, the final storage of spent coated particle fuel embedded in a graphite matrix seems to be feasible. At the salt mine ASSE in the FRG a special charge of looooo spherical fuel elements from the AVR reactor is now beeing disposed in a large scale experiment.

In any case, the thorium fuel cycle for HTR can be closed without time pressure. A demonstration recycle facility of a service capacity of about 5 - lo GWe should be operational by the year 2ooo. US and FRG plans foresee the erection of a common demonstration facility within this time frame.

4. Outlook

It has been realized that the commercialization of high temperature reactors by the private sector is not feasible, and that international cooperation will assist the further development and market introduction.

The possibility of international cooperative development programs on gas cooled reactors is currently being explored to share costs and benefits, and to reduce duplication of effort. In parallel with the present commercialization study being undertaken to determine a long term strategy for both thermal and fast reactors in the US and with similar studies in the FRG negotiations are underway between the US and FRG to establish an agreement for a cooperative program in the field of gas cooled reactor concepts and technology. Under the "Umbrella" agreement which was signed early in 1977 provisions have been made for cooperative programs in the areas of Gas Cooled Fast Reactors, Gas Cooled Thermal Reactors for electricity production and/or process heat and HTR Fuel Recycling. Joint US/FRG-planning, in anticipation of the execution of this "Umbrella" agreement, has been underway since June 1976. - In February 1976 the French and FRG-governments agreed to undertake joint efforts to further develop andvanced reactors. Technical discussions in 1976 led to a definition of areas for cooperation, and the possibility of France joining the US-FRG "Umbrella" agreement on GCR cooperation is being seriously considered. Other European countries, especially Switzerland, have indicated their interest to join into a future cooperative development program. In Japan, HTR development is oriented towards a very high temperature reactor, VHTR, now being designed under the leadership of JAERI. Links are being established between the HTR programs of FRG and Japan.

Thus, a broad basis for the further development and commercialization of the HTR has been established in 1976 giving confidence in the future of this system. Remaining impediments still confronting the commercialization of the HTR are:

- Achievement of full power operation of Fort St. Vrain Reactor in the US and completion of THTR-3oo in the FRG
- Selection of a unique design concept for future HTR power plants and process heat plants
- Establishing of common safety standards and regulations for licensing of large HTR's for electricity and process heat
- Development and qualification of high temperature materials for long-term operation in helium for advanced HTR's, especially for direct cycle and process heat applications
- Development of the fuel recycle technology
- Development and testing of scaled-up reactor and process plant components.

Of particular importance will be to establish new,possibly international, industrial consortia on the side of reactor suppliers and of joint utility ventures on the side of customers with starting assistance of the governments. The ongoing commercialization studies in the US and FRG will help to characterize the domestic and international markets for each of the GCR technologies and their interrelationsships,and to define the conditions which are necessary for the GCR concepts to enter their respective markets allowing to establish a commercialization strategy. It is expected that the way for the further development and commercialization of HTR's can be clearly defined in 1978.

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Table 1: High Temperature Reactor Characteristics

	DRAGON	PEACH BOTTOM	AVR	THTR	FORT ST. VRAIN	HTR-K xx)	HTR-1160	PNP-3000 xx)
Power MW(th)/MW(e)	20/-	115/40	46/15	750/300	837/330	3000/1170	3000/1160	3000/ -
Fuel Elements	Cylindrical	Cylindrical	Spherical	Spherical	Hexagonal	Spherical	Hexagonal	Spherical
He-Temperature Inlet/Outlet (^{o}C)	350/750	377/750	270/950	270/750	400/785	300/750	320/730	350/950
He-Pressure (bar)	20	25	11	40	48	50	50	50
Core Power Density (MW/m^3)	14	6,3	2,3	6	6,3	5,5	8,7	5,5
Fuel Coating	TRISO PyC, SiC	BISO-BISO PyC	BISO PyC	BISO PyC	TRISO-TRISO PyC, SiC	BISO/TRISO x) PyC, SiC	TRISO-BISO PyC, SiC	TRISO PyC, SiC
Fuel Kernel Composition	Thorium-, Uranium-Dicarbides	Thorium-, Uranium-Dicarbides	Thorium-, Uranium-Mixed Oxide	Thorium-, Uranium-Mixed Oxide	Thorium-, Uranium-Dicarbides	Thorium-, Uranium-Mixed Oxide/ U-Carbide, Th-Oxide x)	Uranium-Carbide, Thorium-Oxide	Thorium-, Uranium-Mixed Oxide/ U-Carbide, Th-Oxide x)
Reactor Vessel	Steel	Steel	Steel	PCRV	PCRV	PCRV/PCIV x)	PCRV	PCRV/PCIV x)
Start/end of power operation	1966/1975	1967/1974	1968	(1980)	1977			

x) not yet decided

xx) preliminary design

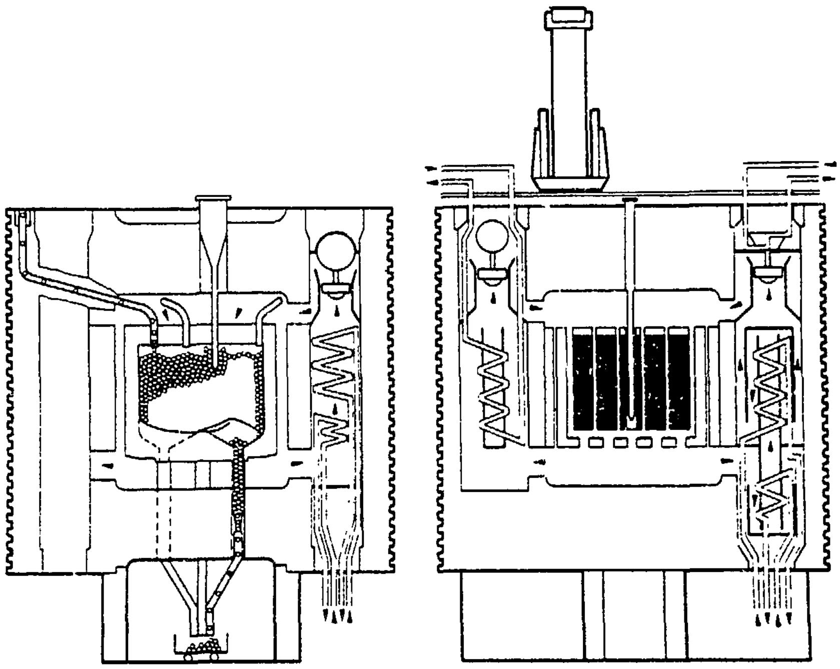

Fig. 1: Main systematic differences of typical designs with pebbles and prismatic fuel (taken from ref. [3]).

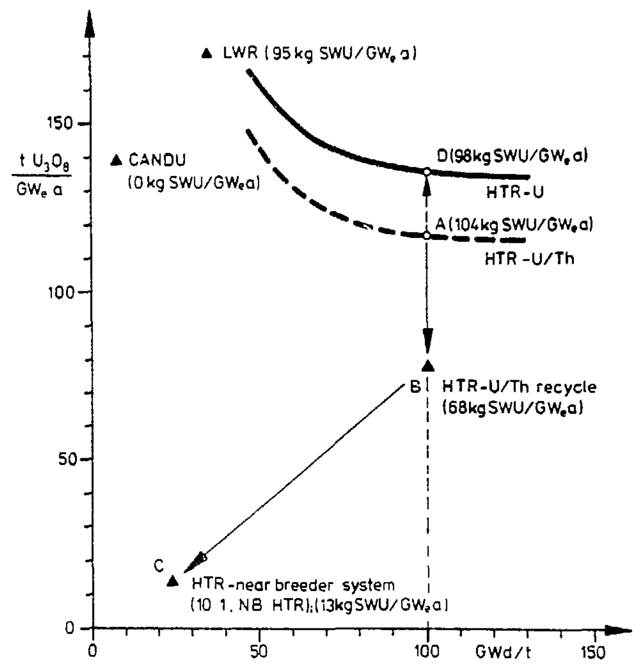

Fig. 2: Annual uranium ore requirements of various reactors as a function of burnup, and corresponding separating work requirements. Assumptions: load factor o.7; tail enrichment o.25%; recycling losses 1%.

Acronyms

1	-	ADAM	- Methanation Facilities
2	-	AVR	- Arbeitsgemeinschaft Versuchsreaktor
3	-	BBC	- Brown, Boveri & Cie AG. Mannheim
4	-	CANDU	- Canadian Heavy Water Reactor
5	-	EIR	- Eidgenössisches Institut für Reaktorforschung, Würenlingen, Switzerland
6	-	EVA	- Einzelspaltrohr-Versuchsanlage
7	-	FRG	- Federal Republic of Germany
8	-	FSV	- Fort St. Vrain
9	-	GAC	- General Atomic Company, San Diego
1o	-	GCR	- Gas-cooled Reactor
11	-	HHT	- HTR with He-Turbine
12	-	HHV	- Hochtemperatur-Helium-Versuchsanlage
13	-	HKG	- Hochtemperatur-Kernkraftwerk GmbH, Hamm-Uentrop
14	-	HRB	- Hochtemperatur-Reaktorbau GmbH, Mannheim
15	-	HTR	- High Temperature Reactor
16	-	IHX	- Intermediate Heat Exchanger
17	-	JAERI	- Japan Atomic Energy Research Institute
18	-	KFA	- Kernforschungsanlage Jülich GmbH, Jülich
19	-	LWR	- Light Water Reactor
2o	-	NUKEM	- Nukem GmbH, Wolfgang near Hanau
21	-	OECD	- Organization for Economic Co-operation and Development, Paris
22	-	OTTO	- Once Through Then Out
23	-	PCIV PCRV	- Prestressed Cast Iron Reactor Vessel - Prestressed Concrete Reactor Vessel
24	-	THTR	- Thorium High Temperature Reactor
25	-	UK	- United Kingdom
26	-	VEW	- Vereinigte Elektrizitätswerke Westfalen AG