International status of HTGRs

by Juergen Kupitz and John B. Dee

For more than 25 years, development programmes on the high-temperature gas-cooled reactor (HTGR) have been conducted in a number of Member States.

Today, the technical feasibility and the advantages of this reactor concept for electricity generation and process-heat applications have been demonstrated by experimental and demonstration reactors in the Federal Republic of Germany, the United Kingdom, and the United States, as Table 1 shows. While HTGRs have not yet been commercially deployed on a large scale, their potential to provide, besides electricity, high-temperature process steam and process heat for various industrial applications has been, together with high safety margins, the continuing incentive for further development.

The concept of the HTGR evolved from the Magnox reactors (with carbon-dioxide-cooled, graphite-moderated, natural uranium-magnesium alloy cladded fuel element) that were developed and built in France, Japan, Italy, Spain, and the UK. These early Magnox reactors had steel pressure vessels which were much larger than those used for light-water reactors (LWRs). Later on, pre-stressed concrete reactor vessels (PCRVs) developed in France were used that could house the entire primary system in a single monolithic pressure vessel. PCRVs were subsequently adopted for all gas-cooled reactors built in France, the Federal Republic of Germany, the UK, and the USA.

Mssrs Kupitz and Dee are staff members in the Agency's Division of Nuclear Power.

Temperature limits otherwise imposed by fuel-coolant interactions were virtually eliminated by the replacement of carbon dioxide by helium as coolant and the change to coated-particle fuel embedded in a graphite matrix in the form of cylinders, pebbles (spheres) or blocks without metal cladding. The objective of developing the HTGR in this way was to provide a nuclear heat source capable of providing temperatures comparable to those used in fossil-fuelled electricity generating plants.

Consequently, superheated high-pressure steam is produced that permits the use of compact high-speed turbine-generators typical of conventional power plants. This high-temperature capability of the HTGR also allows for generating high-temperature process heat for a number of industrial purposes.

Fuel cycle characteristics for this reactor system also are exceptional since it routinely operates at high burn-ups of more than 100 000 megawatt-days per tonne of heavy metal. Consequently, low fuel-cycle costs are achievable with once-through fuel cycles, and exceptional performance in terms of reduced demands for nuclear fuel resources can be obtained in the future through the use of recycled thorium-based fuel.

Main design features: built-in safety

HTGRs are typically characterized by the following main design features: a PCRV, graphite moderator and reflector, helium as coolant, coated particle fuel, and low power density.

Table 1. Main design data of HTGRs

	Dragon (UK)	Peach Bottom (USA)	AVR (Germany, F.R.)	FSV (USA)	THTR (Germany, F.R.)
Start/end of power generation	1966/1975	1967/1974	1968	1976	1985
Power MWth/MWe	20/–	115/40	46/15	837/330	750/300
Fuel element	Cylinders	Cylinders	Spheres (pebble bed)	Hexagonal blocks	Spheres (pebble bed)
Helium temperature outlet (°C)	750	750	950	785	750
Helium pressure (bar)	20	25	11	48	40
Fuel composition	Thorium, uranium carbides	Thorium, uranium carbides	Thorium, uranium oxides	Thorium, uranium carbides	Thorium, uranium oxides
Reactor vessel	Steel	Steel	Steel	PCRV	PCRV

The entire primary coolant system of HTGRs is contained within a PCRV – a multi-cavity pressure vessel made of steel-lined reinforced concrete that is pre-stressed by a system of longitudinal tendons and circumferential wire wrappings. The independence and redundancy of these tendons constitute a major safety advantage of the PCRV. The concrete also functions as the reactor shield. PCRVs are designed to contain high-pressure coolant in much larger sizes than are feasible with steel vessels at the same pressure.

The core and reflector structure is composed of graphite, a material that has a high heat capacity and that sublimes at about 3600°C and retains adequate structural strength to above 2500°C. This contributes to large safety margins between normal operating temperatures and damage limits.

The helium primary coolant is a non-condensable, chemically inert gas that does not contribute to or affect the nuclear chain reaction. Pressurized helium has low stored-energy content and remains in the gaseous phase under all conceivable operating conditions.

HTGR fuel is composed of small spherical particles of uranium and thorium oxides or carbides about 0.2 to 0.6 millimetres in diameter. Each particle is encased in an outer coating of pyrolitic carbon and by several inner layers of ceramic material for fission-product retentions with high-temperature stability. These coated particles are homogeneously dispersed in a graphite matrix that is subsequently compressed and sintered in the form of spheres for use in spherical fuel elements, or in the form of rods that are inserted into the fuel channels of a multi-hole graphite block.

These particles remain intact and retain virtually all fission products at temperatures up to about 2000°C. They do not melt at a given temperature threshold but fail gradually and statistically under accident conditions, so a sudden release of fission products cannot occur.

The power density of an HTGR is about one order of magnitude less than that of an LWR and contributes in a major way to the high inherent safety of this type of reactor. Together with the high heat capacity of graphite in the core and reflector (a 2240 MWe HTGR contains more than 1.3 million kilograms of graphite), it is ensured that reactor temperature transients in response to disturbances proceed very gradually. The slow thermal response provides for a forgiving reactor since the behaviour of the system is more predictable and more time is available to prevent transients from progressing into major accidents. Time is available to adjust the system or to take other corrective action.

HTGR applications wide-ranging

On the basis of HTGR operating experience and the significant potential of the technology, development programmes in the Federal Republic of Germany, Japan, the USA, and the USSR now aim at commercial introduction of the system for electricity and process-steam generation (co-generation) and/or preparatory work for process-heat applications. Reactor design work, extensive research and development (R&D), process studies and development work, site investigations, and economic assessment studies are being performed with emphasis on different topics in the four countries. The main application objectives are:

- Electricity generation by HTGR steam-cycle systems and HTGR with an integrated direct-cycle gas turbine
- Process steam (and electricity in co-generation) for organic chemistry production processes and petro-chemistry, extraction and processing of crude oil, and substitute natural gas (SNG) production by Lurgi/Exxon process coal gasification
- Process-heat application
- Synthesis gas production for SNG production by hydrogasification of coal, ammonia production, iron ore reduction, and methanol synthesis
- SNG production by steam gasification of coal
- Cracking of heavy oil residues
- Long-distance energy transport by gases
- Hydrogen production by thermochemical splitting of water and high-temperature electrolysis of water.

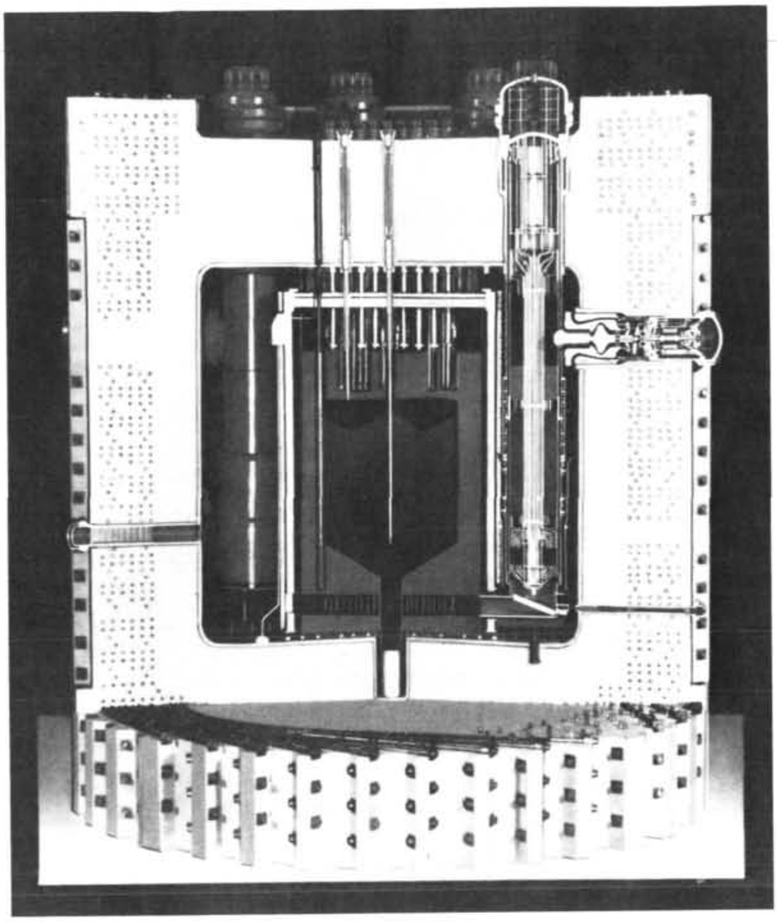
Plants operating or being built

HTGRs in operation or under construction today are all electricity generating plants. These are:

AVR reactor. The HTGR programme in the Federal Republic of Germany started with the construction of the small pebble-bed prototype, the Arbeitsgemeinschaft Versuchsreaktor (AVR). Plant construction at the site of Jülich was started in 1960 and operation began in 1967. The reactor core consists of 94 000 spherical fuel elements each of 60 millimetres diameter. Fuel charge and discharge is performed continuously under operation. The reference fuel elements contain one gram of uranium-235 (93% enriched) and five grams of thorium-232 in particles with pyrocarbon layers. The power density is 2.6 MW per cubic metre.

In spite of interrupted operations for conducting various test programmes, average availability over the first 11 years of operation was 77%. The best availability of one year's operation was 92% in 1976. The fuel elements have exceeded burnup of more than 190 000 MW days per tonne.

The high-temperature potential of HTGRs was successfully demonstrated in the AVR by raising the helium outlet temperature from its original value of 750°C to 850°C in 1972 and to 950°C in 1974. During the whole operation time of the plant, the radiation exposure of the operating and maintenance personnel has been very low.



Cross-section diagram of the pre-stressed concrete reactor vessel of the thorium high-temperature reactor (THTR-300) prototype nuclear power plant, which houses the entire primary circuit system. Thickness of the cylindrical walls measures about 4.5 metres, that of the top and bottom slabs, just over 5 metres. The fuel is housed in the pebble-bed reactor core (centre of diagram), in which technicians (above) conducted tests following the insertion of the absorber rods after first criticality. (Photos courtesy of Hochtemperatur-Reactorbau Gmbh.).

For the further utilization of AVR, the Kernforschungsanlage (KFA) Jülich proposed in 1983 a modification of the plant for high-temperature process-heat system demonstration, principally in connection with conversion of coal into high-grade chemical products for use as fuels or other chemical carriers, or as chemical feedstocks. Such processes offer various benefits including improvements in environmental protection with more efficient utilization of coal resources.

THTR-300. Based on the AVR experience a 300-MWe pebble-bed demonstration power plant, THTR-300, was designed in the late 1960s. Construction began near Hamm in the Federal Republic of Germany in 1972. Completion of the plant was originally expected in 1977–78, but major delays were encountered as a result of changes in licensing requirements. After reaching an advanced stage of construction, additional potential problems – such as impacts by missiles from outside objects, vertical earthquake vibration, and spontaneous cracking of pipes and tanks – had to be taken into account. Now the reactor system is almost completed.

Initial criticality was reached on 13 September 1983, with the loading of 198 180 spherical fuel elements. After power tests during the first half of 1985, the reactor is scheduled for handover to utilities in October 1985.

Fort St. Vrain (FSV). During the construction and start-up of the Peach Bottom HTGR, design and development proceeded in the United States on larger HTGRs resulting in a 330-MWe HTGR nuclear generating station at Fort St. Vrain.

Site construction was started in September 1968 and power generation commenced in late 1976. However, the plant was limited to 70% power because of core-region, gas-outlet temperature fluctuations that were subsequently attributed to small movements of some fuel block columns. Subsequently, mechanical restraints were installed on the top reflector and the fluctuations ceased. The US Nuclear Regulatory Commission then gave permission to test the reactor at higher power levels, and 100% power was achieved in November 1981.

The excellent fuel performance and the resulting low radiation exposure to plant personnel documented during the refuelling stage in 1979 are significant. Radiation dose rates on primary system components continue to be low enough to allow substantial direct contact maintenance. These low dose rates are clearly attributable to the impermeability of the ceramic-coated fuel particles, the radiologically inert helium coolant, and the PCRV structure.

Fort St. Vrain also has demonstrated high-performance and high-temperature capabilities, having achieved 785°C primary loop and 400°C steam temperatures in operation to date. The reactor has achieved 38.5% thermal efficiency at full power and to date has generated 4.3 terawatt-hours of electricity.

Development programmes in Member States

Japan. The Japanese programme began early in the last decade with preparatory design, research and development by the Japan Atomic Energy Research Institute (JAERI) for an experimental very high-temperature reactor (VHTR). The VHTR will be a 50-MW (thermal) plant, equipped with a steel pressure vessel with low-enriched uranium-coated fuel particle in graphite blocks, two primary cooling circuits with intermediate heat exchangers, and heat-removal components as a first step. The coolant outlet temperature will be 950°C.

The start of construction is expected about 1990. Recent programme review resulted in a broadening of the Japanese HTGR programme beyond direct steel-making towards consideration of (1) electricity and process-steam production (co-generation); (2) synthesis gas as a chemical product on the basis of oil residue; (3) coal gasification or liquefaction; and (4) hydrogen production by themochemical water-splitting.

USSR. In the USSR an extensive HTGR programme was launched for the development and introduction of the HTGR into the heat market.

The reference design is a pebble-bed reactor with a low-enriched fuel cycle. All important reactor components are undergoing extensive theoretical and experimental studies. Large test facilities are used for gas dynamics studies, heat exchange, neutron physics characteristics, material qualification, interaction between rods and core, and fuel testing. The fuel manufacturing process has been developed successfully and irradiation testing has been performed.

On the basis of these studies the design for a prototype plant has been completed. This reactor, the 50-MWe VGR-50, is designed both to generate electricity and to serve as a 300 to 400 kilowatt gamma-radiation facility for chemical processes by means of a continuous flow of fuel elements from the core to the irradiation facility and back to the core. Other plant characteristics include a helium coolant pressure of 40 bar with an outlet temperature of 800°C. The primary circuit has four heat-exchanger loops and is contained in a steel vessel.

The decision to begin construction is expected soon after completion of the design and R&D work. Preparations for component production are under way.

Simultaneously, significant progress has been made on design work for a demonstration plant, called VG-400, as a source for process heat and electricity generation. This is a 1000-MW(thermal) pebble-bed reactor with a pod boiler PCRV, once-through fuel scheme, 950°C gas outlet temperature, four heat-exchange loops and electric blowers. The reactor will be equipped with helium/helium heat exchangers and steam reformers in the secondary circuits for ammonia production by steam catalytic conversion of methane.

Federal Republic of Germany. In the Federal Republic of Germany there are several candidates as possible follow-up plants to the THTR-300 including the HTR-500 (500 MWe) and the Modular Reactor System (approximately 200 MW(thermal) per module).

The structural elements or component concepts of the THTR-500 are being adopted for the HTR-500 in their greater part and a number of basic primary circuit elements have been standardized. They include components of the PCRV, steam generators, circulators, control and shutdown mechanisms, ceramic blocks of the core structure, and fuel elements. Main design data are summarized in accompanying Table 2.

Table 2. Main design data of the HTR-500

Power MWth/MWe	1250/500
Helium pressure (bar)	50
Helium outlet temperature (°C)	700
Steam pressure (bar)	180
Temperature (°C)	525

The consequent use of HTR-specific safety characteristics as well as an optimization of the design of the components and circuits lead to the result that the HTR-500 – in spite of its lower power level as compared to the typical 1230-MWe pressurized-water reactor of the present design – may have only slightly higher specific plant costs and nearly the same electricity generating costs in concurrence with substantially lower capital costs.

When applying the HTR-500 for electricity and process-steam generation, the HTR also can supply high-pressure and high-temperature steam. Furthermore, the HTR-500 presents clear cost advantages compared to hard-coal-fired power plants.

The basis of the HTR-Module concept is the combination of small 200-MW(thermal) standardized plant units, the so-called factory-built modules, to form plants of a wide range of thermal ratings up to 1600-MW for electricity production, co-generation of steam and electricity, or process-heat applications.

The design features of the HTR-Module Reactor System are based upon well-established technologies, the operational experience of light-water reactors, and of the AVR (with special attention to a simple safety technology), high availability, and wide flexibility for application. An HTR-Module consists of the following system elements:

One of the 1482 graphite fuel blocks being lowered into the core of the 330-MWe Fort St. Vrain nuclear plant in Colorado, USA, the first commercial-size reactor using the advanced HTGR system. Since power generation began in 1976, the plant has generated more than 4 terawatt-hours of electricity. (Photo courtesy of GA Technologies Inc.).

The pebble-bed reactor core produces a thermal output of about 200 MW. Depending on the application, it has a helium temperature of 700°C for electricity generation or 950°C for process-heat production at a pressure of 40 to 50 bar and an average power density of 3 MW per cubic metre.

The reactor and helium circuits are contained in one vessel and the steam generator in the other connected by a single coaxial gas dust. The arrangement of the reactor and the heat-transfer components in separate steel vessels facilitates their accessibility for inspection, repair, and disassembly.

The HTR-Module's limited size, low power density, and slim core design (3 metre diameter × 9.6 metre height) yield an extraordinary degree of passive safety. In response even to fast changes of the primary circuit heat balance, the temperature-transients of the HTR are extremely slow, owing to the high heat-capacity of core and other graphite structures. Thus, even on the assumption of extreme hypothetical accidents, ample time remains for engineered safeguards to react or for taking emergency measures to prevent or limit any potential damage to reactor components.

One very important field of application of HTR-Modules is for supply of process steam to medium or large industrial complexes. The HTR is the only type of nuclear reactor system that can generate all of the steam conditions usually needed in the process industries.

The possibility of the combined generation of process steam and electrical power contributes towards diminishing the dependence upon the external grid. Together with the flexible power-rating, the expected high availability and environmental advantages of the HTR-Module, nuclear steam generators soon will be available for application niches formerly filled by fossil-fuelled steam generators.

The small HTR-100 reactor concept is also based on an adaptation of the concept of the AVR-15 MWe experimental power plant design.

The HTR-100 also is designed as a unique heat source for electricity generation, co-generation of steam and electricity, and process-heat applications. It is characterized by the integrated arrangement of all primary system components in a single steel reactor pressure vessel, upward helium flow through the core to the heat exchanger located above the core, and control and shutdown of the core by reflector rods and small absorber spheres in channels of the reflector buttresses.

For electricity production or co-generation, the thermal power of 256 MW is transferred by helium at 70 bar and 700°C to the secondary circuit by three loops of circulators and steam generators that produce superheated steam at 190 bar and 530°C. Decay-heat removal is also ensured by these triply redundant loops at a high safety level. As a backup, decay-heat removal also could be effected by radiation and conduction of the heat from the reactor vessel to its cooling system located within the safety containment.

United States. In the USA, a major programme activity has been design and test activities for supporting an HTGR lead plant project, which would include the demonstration of full commercial-size HTGR systems and components including features incorporating technology evolution and Fort St. Vrain operation experience. Other objectives include establishment of the licensing criteria for follow-on commercial HTGR systems and for defining plant operational characteristics for co-generation.

The US programme also includes conceptual design of the advanced reactor systems for high-temperature process-heat applications and the investigation of waste treatment and spent-fuel storage for near-term closure of the HTGR fuel cycle, continued development of the technology for eventual reprocessing of HTGR-type uranium/thorium fuel elements, and investigations on the technical and economic merits of small modular reactors for special applications.

At the last Conference on the HTGR in San Diego, USA, August 1984, it was reported that the US HTGR programme is now passing a reorientation phase. A major portion of the programme will continue as a base technology programme. The 2240-MW(thermal) lead plant project, as it has been designed until 1984, will be taken as a baseline reference.

The US HTGR programme is now focusing on plant sizes of about 1000 to 1300 MW(thermal), representing the most likely power range for a next order of the utilities. The programme is structured to evaluate integrated and multi-module plant concepts with emphasis on enhanced safety and environmental protection. The programme goal is to select one integrated and one multi-module concept by the end of fiscal year 1984.

Benefits extensive

The HTGR is a reactor system of the second generation with many promising features; its predicted benefits have been confirmed.

The system is able to provide steam for electricity generation, high-quality process steam, and high-temperature process heat. For electricity generation, the technology was ready to be realized in the early 1970s.

Although the system always had to compete with established LWR technologies, it approached market introduction in the USA and Europe in a period of rapid expansion of nuclear energy in the early 1970s. However, during the worldwide stagnation of nuclear energy in the second half of the decade there was no chance for commercial introduction of a new system additionally burdened with normal first-of-its-kind costs.

Simultaneously, under the pressure of the continuous oil and gas price escalation and the increasing uncertainties about the long-term availability of fossil fuel for many countries, it was recognized that the HTGR is a unique tool to save or replace fossil energy in the heat market. That market now consumes an overwhelming portion of oil and gas not accessible for other nuclear technologies.

The HTGR can have an important function in the energy market. Coal, as long as it is burned, oil in shale and sand, heavy oil deposits and oil residues mostly require high-temperature conversion processes to provide clean, environmentally acceptable, easy-to-use and yet economical energy carriers. However, conventional processes require the burning of almost as much additional fuel as is converted. Consequently, nuclear conversion could nearly halve the coal consumption for a given demand.

Even more, the HTGR can provide the necessary process heat in a clean, safe, fuel conserving, and economical manner without burdening the environment.

Further details of HTGR programmes in IAEA Member States are available in *Status of and Prospects for Gas-Cooled Reactors*, IAEA Technical Reports Series No. 235, IAEA (1984).