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HELIUM-COOLED HIGH TEMPERATURE REACTORS*

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ABSTRACT

Experience with several helium cooled reactors has been favorable, and two commercial plants are now operating. Both of these units are of the High Temperature Graphite Gas Cooled concept, one in the United States and the other in the Federal Republic of Germany. The initial helium charge for a reactor of the 1000 MW(e) size is modest, ~15,000 Kg.

INTRODUCTION

Helium is a nearly ideal coolant for nuclear reactor power stations. It has both high thermal capacity and conductivity, a low cross-section for neutrons, and is chemically inert. Helium also has no phase change at temperatures of interest, is easily separated from impurities, is optically transparent, and offers convenient leak detection. Thus, helium properly belongs to the nuclear field. However, it has some disadvantages: low density, small molecular size (hence, some propensity to leak from containers), low heat transfer coefficient, and poor lubricating properties.

The effectiveness of helium as a nuclear reactor coolant has been demonstrated in several experimental reactors and in two commercial units (a third is now in the start-up phase). Although many helium-cooled design concepts were considered in the early phase of nuclear energy development, these have focused into a single class of high temperature graphite reactors (HTGRs).

Two basic concepts are employed for the reactor cores. One consists of prismatic fuel elements in an orderly array. The other is comprised of a bed of spherical fuel elements.

HISTORICAL BASE OF EXPERIENCE

Helium was considered as a nuclear power station coolant as early as 1947.[1,2] Despite the many advantages which helium offered, the technology was limited, and a decade passed with little progress. Then, almost suddenly, a surge of interest occurred which led to the establishment of projects for six different helium-cooled reactors. These were Peach Bottom,[3] Dragon,[4] UHTREX,[5] AVR[6], EGCR,[7] and EBOR.[8] The current HTGR commercial units are the Fort St. Vrain

Reactor[9] in the U.S.A. and the Thorium High Temperature Reactor (THTR)[10] in the Federal Republic of Germany. The sponsorship and characteristics of these reactors are shown in Table 1. These projects have been quite varied and, in total, have many accomplishments. To mention a few, they have demonstrated the feasibility of high temperature and high pressure helium systems for nuclear service, produced high thermal efficiencies, operated with low inventory requirements for fissionable materials, demonstrated radioactively clean circuits for convenient maintenance, and, in general, have had excellent availability.

Pebble-bed type reactors are proposed for the future in the Federal Republic of Germany. Electricity generating plants are sized at 300 or 500 MW(e); smaller units are planned for the production of process heat. In the United States, new plants are being designed by a consortium of companies with the cooperation of the Gas-Cooled Reactor Associates (GCRA), which is sponsored by 32 utility companies. These and other concepts are being considered by the U.S. Department of Energy, including a small [100-125 MW(e)] pebble-bed reactor and a slightly larger concept of the prismatic fuel design. A development program also exists in Japan with plans for a 50 MW(t) test reactor.

HTGR BASIC DESIGN FEATURES

The principal features of High Temperature Graphite Reactors are made possible through the use of helium as a coolant and graphite for the high temperature core structures. Another basic feature is the use of pyrolitic carbon-coated ceramic fuel particles which are embedded in a graphitic fuel structure. Tiny microspheres (~350 μm) of uranium carbide and uranium dioxide are coated with multiple layers of pyrolytically deposited carbon and a layer of silicon carbide to contain both the fuel and the radioactive products from fission. An innermost layer of low density pyrolytic carbon provides space for fission product accumulation and for fuel swelling, an impervious layer of high-density pyrolitic carbon serves as a pressure shell, a layer of silicon carbide prevents diffusion of cesium and other fission products which might permeate the inner pyrolytic carbon layer, and an outer protective layer of pyrolitically deposited carbon completes the fuel particle structure.

Table 1. Early helium-cooled nuclear reactors

Designation	Sponsorship	Power Level, MW		Date Construction Started	Date Operational	Date Terminated or Status
		Electrical	Thermal			
Peach Bottom[a]	GA[a]/P.E.[b]/HTRDA[c]	40	115	1962	1967	1974
Dragon	OECD-Dragon[d]	0	20	1961	1964	1976
AVR[e]	AVR GmbH[f]	15	51	1957	1968	Operating
UHTREX[g]	USAEC[h]-LASL[i]	0	3	1962	1967	1969
EGCR[j]	USAEC-TVA[k]	30	88	1959	1965[l]	Cancelled
EBOR[m]	USAEC-GA	0	10	1961	1965[l]	Cancelled
Fort St. Vrain	GA[a]/PSC[n]	330	842	1968	1976	Operating
THTR[o]	HKG[p]	300	750	1971	1985	In startup

[a]General Atomic (now GA Technologies).
[b]Philadelphia Electric Company.
[c]High Temperature Reactor Development Associates.
[d]Organization for Economic Cooperation and Development, European Nuclear Energy Agency Dragon Project.
[e]Arbeitsgemeinschaft Versuchs-reaktor.
[f]Sponsorship includes: Brown Boveri/Krupp, Kernforschungsanlage Julich, and Euratom.
[g]Ultra High Temperature Experiment.
[h]United States Atomic Energy Commission.
[i]Los Alamos National Laboratory.
[j]Experimental Gas-Cooled Reactor.
[k]Tennessee Valley Authority.
[l]Project cancelled before operation.
[m]Experimental Beryllium-Oxide Reactor.
[n]Public Service of Colorado.
[o]Thorium Hoch Temperatur Reaktor.
[p]Hochtemperatur Kernkraftwerk, GmbH.

In the prismatic fuel element, the microspheres are embedded in a partially graphitized matrix in the form of small sticks (1.24 cm dia. by 6.3 cm long). The sticks are inserted into holes in graphite blocks. Helium coolant flows through alternate parallel cylindrical passages. In the spherical graphite fuel element, which is 6 cm in diameter, the microspheres are embedded in a carbon matrix in the inner region of the sphere. The designs of the prismatic and spherical fuel are shown respectively in Figures 1 and 2. To form the reactor core of prismatic elements, individual units are stacked in an array as shown in the sectional view of Figure 3. In contrast, the spheres are essentially poured into the reactor central chamber and cooled directly by helium flowing over their surfaces. In both cases, highly purified graphite is used, which offers excellent neutronic properties.

The large masses of graphite provide high thermal stability with large heat capacity to prevent overheating in the unlikely event of loss of cooling accidents. Since the helium is inert and both the fuel and the graphite can sustain very high temperatures, the margins of safety for this reactor are great. It also has a negative neutronic temperature coefficient such that the fission process is automatically reduced and terminated if a temperature excusion occurs for any reason.

Prestressed concrete pressure vessels are used for both the prismatic and pebble-fueled reactors as shown in Figure 3. These structures have excellent safety and operational features and have been used for both of the large commercial units. (However, steel vessels may be preferable for small reactors and have been used for the several experimental units.) The prestressed concrete pressure vessel maintains a high integrity with respect to failure by virtue of a multiplicity of wires and cables which resist the high internal pressure of the helium coolant (6.2 MPa).

As can be seen in Figure 4, the steam generators and the helium circulators are enclosed in separate but connected pods of the prestressed concrete pressure vessel. Thus, the entire helium circuit is contained within a single enclosure and does not require major external piping. Small water-cooled auxiliary heat transfer systems illustrated in Figure 3 are utilized to remove the fission product decay heat following reactor shutdown in the event of loss of the primary cooling system. The total combination of these

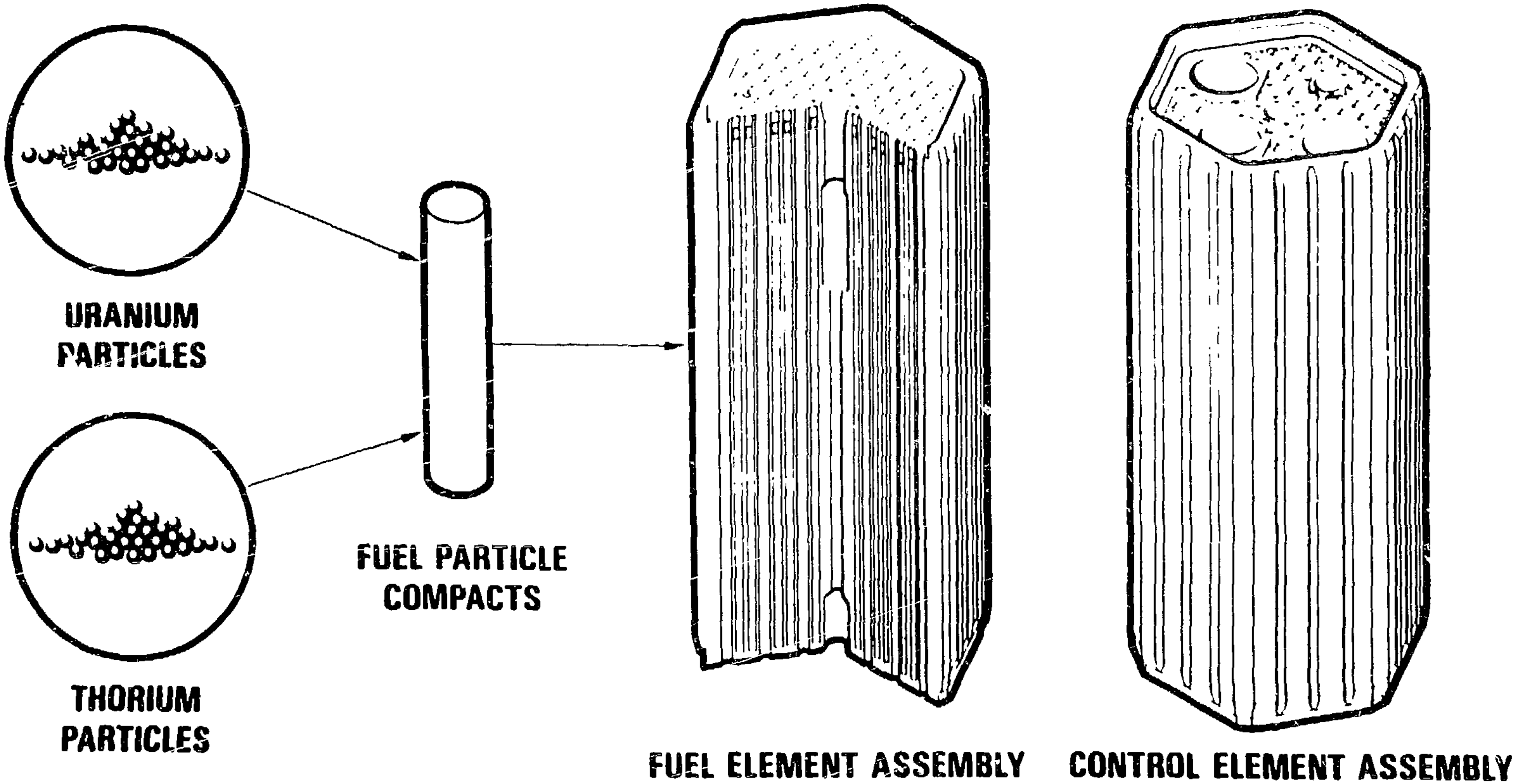

Figure 1. Particles molded into compacts for insertion in integrated HTGR fuel element.

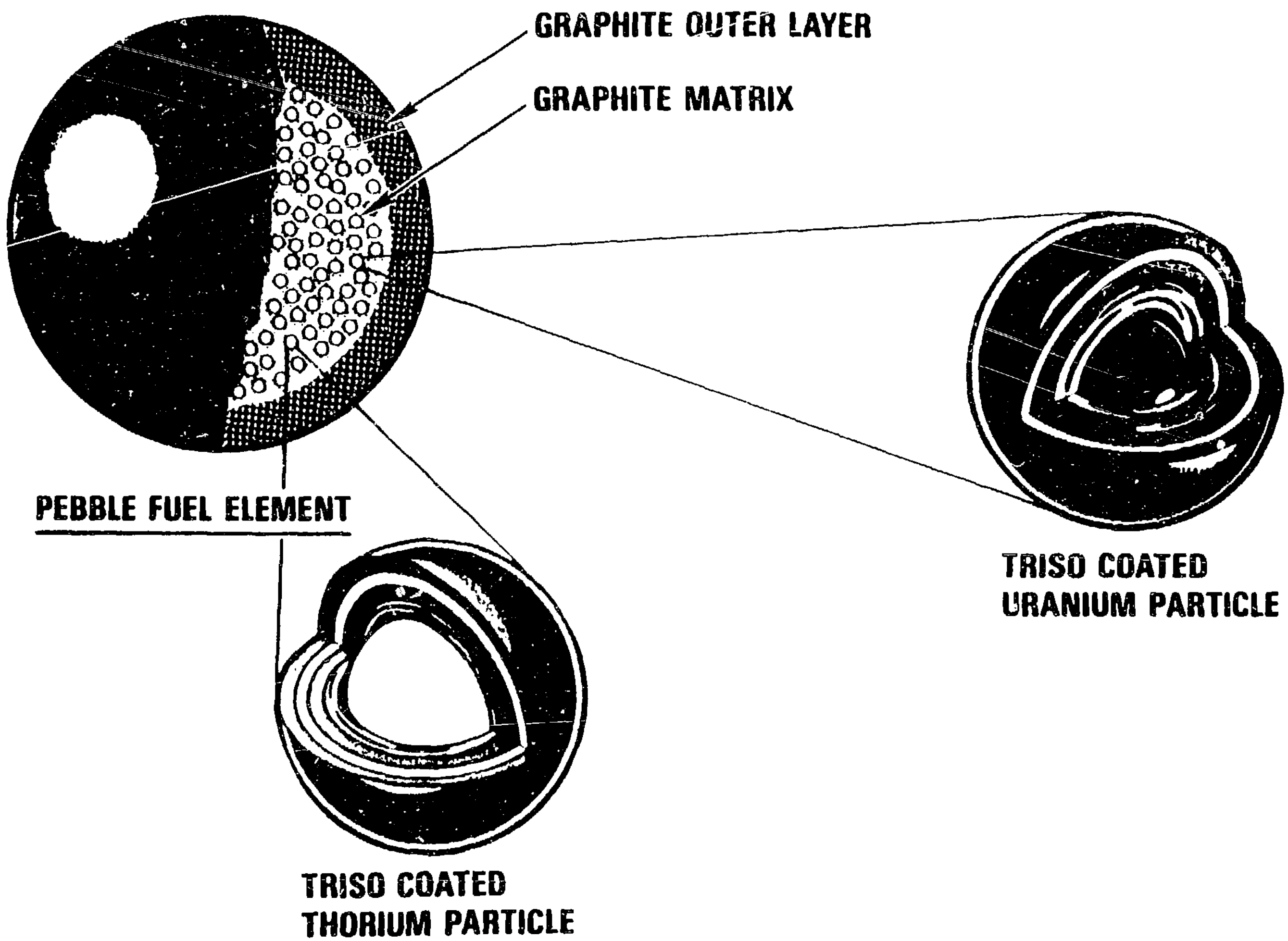

Figure 2. Modular HTGR uses coated fuel particles in pebble form.

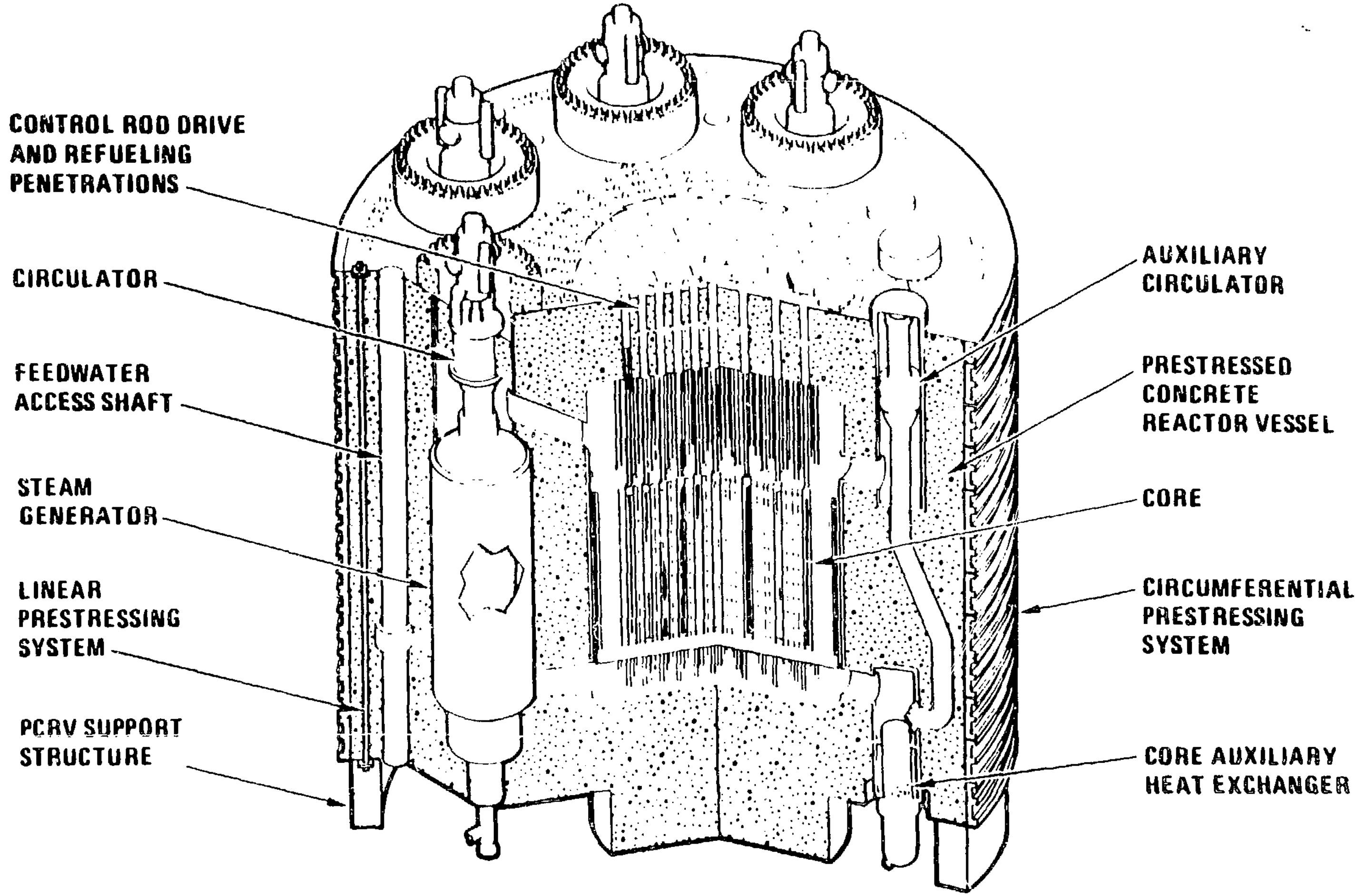

Figure 3. Typical integrated HTGR nuclear steam system configuration.

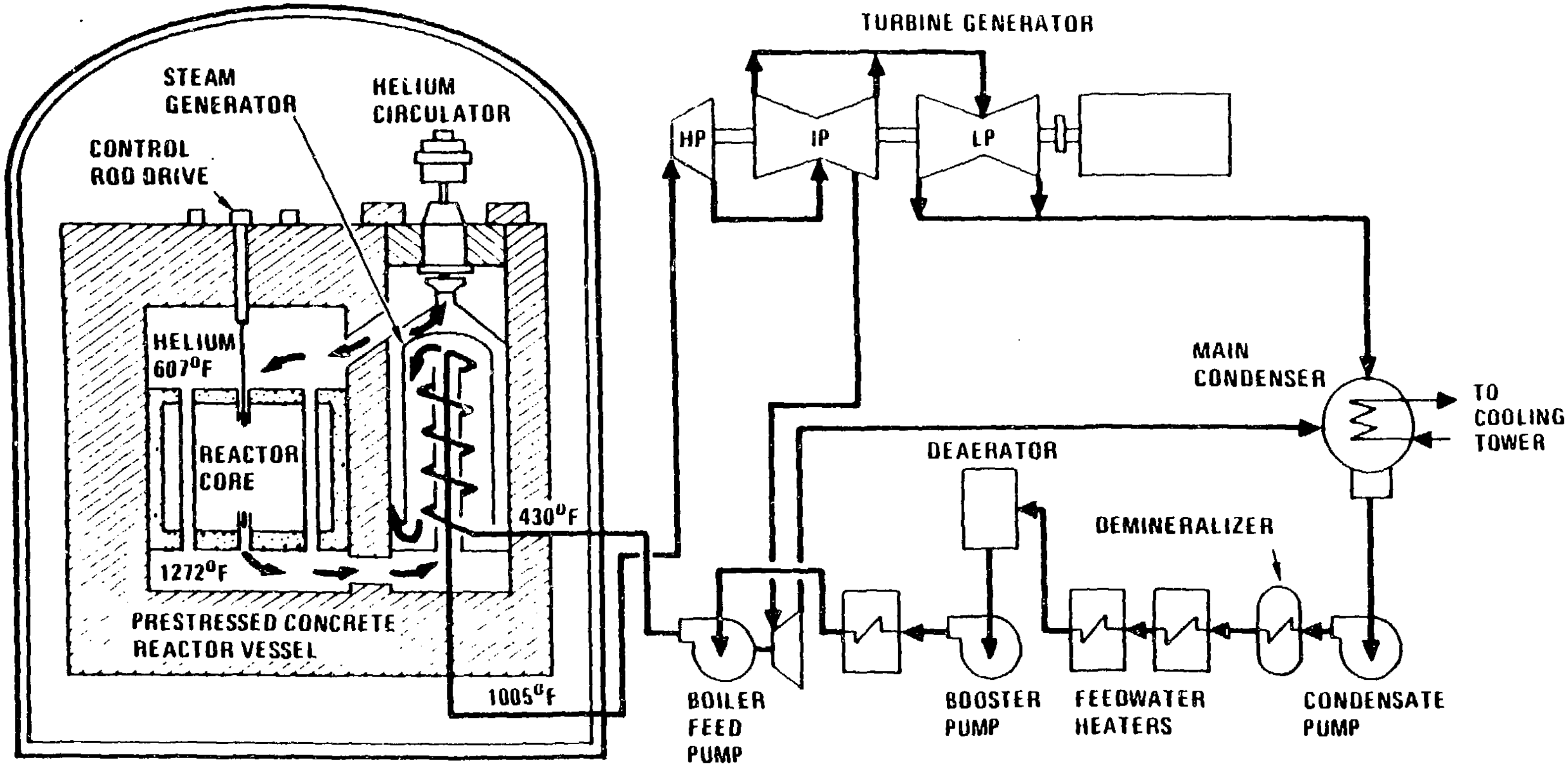

Figure 4. Integrated HTGR plant flow diagram.

features provides a highly secure and well-protected nuclear system with assured cooling for all normal and accidental conditions.

The general configuration of the pebble-bed reactor vessel is similar to that for the prismatic-fueled unit. However, the spherical fuel elements make possible refueling during operation. Fresh or recycled fuel is simply dropped onto the top of the bed. Spent fuel or fuel for recycle is removed element-by-element at the bottom by a selector mechanism similar to that for a gum-ball machine. Pneumatic transport by helium is used to move the spheres either to the top of the reactor for re-entry or to a fuel storage chamber. Selection of fuel to be recycled or discarded is made by a subcritical reactor assembly through which the individual spheres pass and by which the remaining useable fuel can be determined by reactivity measurements. The fueling and refueling transport system is shown schematically in Figure 5.

The refueling of prismatic fuel systems is practical only with the reactor shutdown. A fuel charge machine, which travels on rails, attaches to ports at the top of the reactor, and a remote-handling mechanism removes the fuel from individual stacks and replaces those stacks element-by-element with fresh fuel.

The high integrity of the fuel and the inertness of helium, together with elaborate coolant cleanup systems, facilitate the maintenance of radioactively clean circuits. This makes possible the servicing and maintenance of the fuel and helium handling equipment, the plugging of failed steam generator tubes, and other operations with a minimum of operator and maintenance personnel exposure to radiation. Helium-cooled reactor circuits are cleaner in this respect than those for any other coolant.

HIGH TEMPERATURE GAS-COOLED REACTOR DESIGN

The plant characteristics for a typical large, prismatic-fueled HTGR are shown in Table 2. Well-developed designs exist for these reactors, and commercial orders were placed in the 1970s for several which were subsequently cancelled, primarily because of the smaller projected electrical energy demand in the United States. More recently, smaller units [as low as 100 MW(e)] have been conceptually designed with highly passive safety features. The small sizes contribute both to passivity and to a better match for a

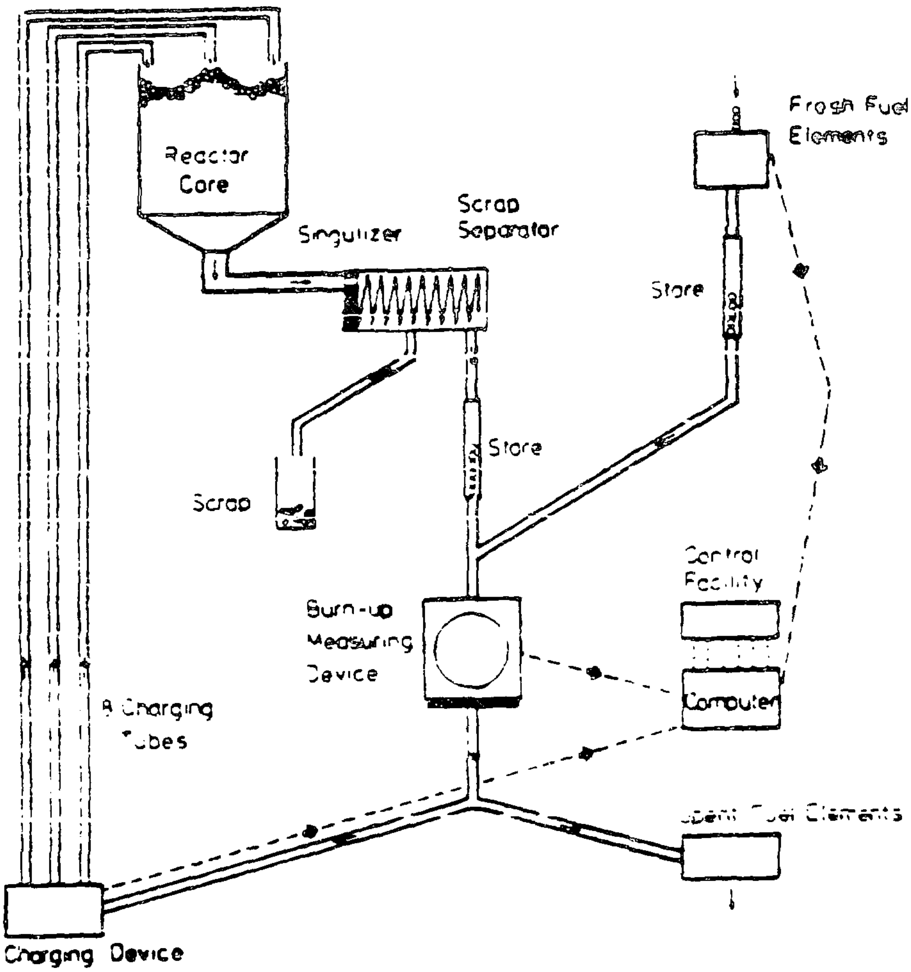

Figure 5. Diagram of fuel-element circuit.

Table 2. HTGR systems have high thermal efficiency, low uranium needs

	SI units		English units
Nuclear system parameters			
Thermal power, MW		2240	
Helium pressure	4.8 MPa		1060 psia
Helium temperature	685/313°C		1266/595°F
Power systems parameters			
Power cycle		Non-reheat	
Turbine steam inlet pressure/temperature	10.9/537°C		2400/1000°F
System parameters			
Net electrical output, MW		860	
Net thermal efficiency, %		39	

small projected baseload demand growth. For large reactors down to 300 MW(e), prestressed concrete pressure vessels have been preferred. In the smaller sizes [100-150 MW(e)], steel pressure vessels are favored.

HELIUM PURIFICATION CIRCUITS

In addition to the components mentioned earlier, auxiliary systems are utilized to charge and discharge the primary helium circuit and to continually purify the helium. Although the initial charge of helium and makeup requirements are provided from conventional pressurized, trailer-mounted cylinders, storage capacity and high pressure compressors must be utilized to depressurize the circuit for maintenance and refueling (for prismatic-fueled reactors) and to repressure it for operation. Helium losses should occur only from leakage, fuel-charge machine operation, and residuals in purged double closures during maintenance operations.

Impurities enter the helium circuit through air and moisture adsorption on graphite fuel (although the fuel is manufactured and stored in helium atmospheres), desorption from structural surfaces following construction and maintenance, and water leakage from steam generators and from helium circulator bearings and seals. The latter source is unique to the Fort St. Vrain Reactor, which has water-lubricated bearings. However, circulators using oil-lubricated bearings also introduce small quantities of hydrocarbons. In addition, a very small fraction of the fuel particles are defective in manufacture and release radionuclides to the circuit. Overall, these sources require elaborate cleanup circuits in duplicate to facilitate simultaneous operation and recharging. A simplified helium purification system is shown in Figure 6. Purified helium is provided at higher than reactor operating pressure to buffer regions between double closures to penetrations and pressure relief valve assemblies. This provision avoids any outleakage which might allow radioactive elements to escape.

The purification system must have capacity for recharging the vessel as well as for normal operation. The latter is a relatively low requirement (less than 10% per hour) which, for a 1170 MW(e) reactor, is approximately 0.1 kg/sec. Impurities in the circuit typically are specified not to exceed 2 ppm each for CO_2, H_2O, and

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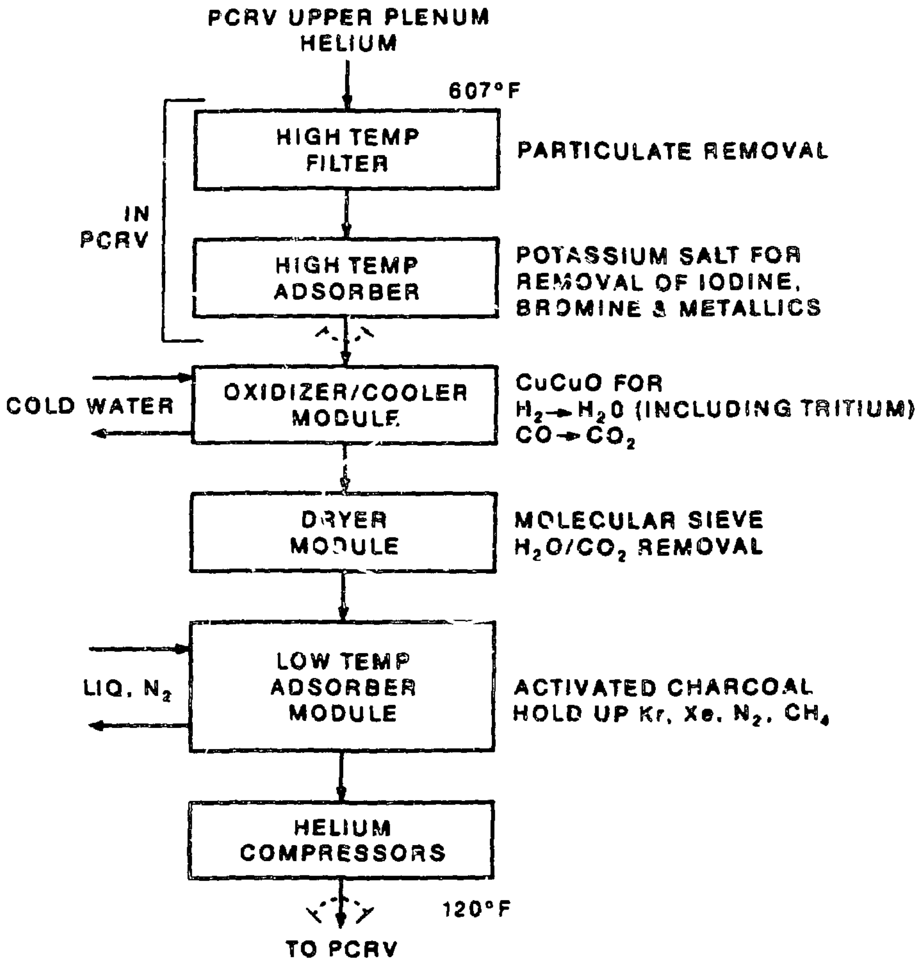

Figure 6. Typical helium purification system.

hydrocarbons; 5 ppm for CO; and 10 ppm for H_2. The purified helium should not exceed 0.5 ppm for the combined amounts of CO, CO_2, and H_2O; 0.5 ppm for H_2; and 0.1 ppm for hydrocarbons. In practice, the molecules are held to considerably lower levels in both helium streams. The purification system also reduces radionuclides to very low levels including the long halflife noble gases, Kr and Xe. This is confirmed by actual experience and results in very low radiation exposure for operating personnel as shown in Figure 7.

Chilled mirror-type dewpoint hygrometers typically are used to continuously monitor moisture content in the helium circuits. Other impurities are determined periodically by taking samples for mass spectrometer analysis. Sampling rakes are installed in the main coolant circuit to obtain representative analyses.

QUANTITIES OF HELIUM REQUIRED FOR HTGRs

Table 3 lists the helium inventories and losses for several HTGRs including the projected large prismatic reactor. Losses have been acceptable for the AVR (which, as an early reactor, has a large number of valves in a complex circuit) over a long operating history. The THTR has only started operation, so those numbers are preliminary. The Fort St. Vrain losses are quite high because of an inaccessible leak in the reactor core support floor, the design of the helium circulators, and other known leaks which are considered uneconomical to repair. With available experience for the design and operation of helium systems, the estimate for losses in the large prismatic reactor are thought to be achievable and probably represent an overestimate. As can be noted, the requirements for makeup of helium losses are an order of magnitude greater than the initial charges. This further indicates the need for effective helium purification systems.

One deterrent to the introduction of high temperature helium-cooled reactors in countries other than the United States has been the unavailability of indigenous sources of helium. Fortunately, the U.S. has large reserves of helium, assuming that they will be conserved and used properly. With the discovery of helium in places other than the U.S., that concern has been somewhat alleviated. It also is necessary to look at the cost of the helium requirements. With helium costing $10 per kg, the initial charge for a 1000 MW(e)

Table 3. Helium inventories, losses, and purification rates

Reactor	AVR	Peach Bottom	Fort St. Vrain	THTR	Large Prismatic
Rated power MW(e)	15	40	330	300	1,160
Primary circuit helium Kg	530	400	2950	8400	15,000
He purification flow Kg/hr	20	92	440	540	1,240
Fraction purified per hr	0.04	0.23	0.14	0.06	0.08
Helium losses Kg/day	2	2	200	10	<10

reactor (or its equivalent in smaller reactors) would be approximately $150,000, and the total unescalated cost for a 40-year reactor lifetime would be less than $1,500,000. Although this is a significant cost, it is a very small fraction of the total for construction and operation of a nuclear plant. If helium were not available from natural gas and were recovered from the atmosphere, the cost is estimated to be approximately 100 times that from natural gas. This probably would render helium cooled reactors economically unattractive.

It is difficult to estimate what commercial future the HTGR may have, but, taking an optimistic estimate of reactors totalling the equivalent of 100 units of 1000 MW(e) each operating for a lifetime of 40 years, the helium requirements would be approximately 15 million kg. If, according to the estimates by Goeller,[11] helium reserves in the United States are 774 million kgs, the helium usage indicated would be a small but significant fraction of the known reserves.

PROSPECTS FOR THE FUTURE

Projecting the future for energy requirements has been very difficult throughout the past decade, and estimates made today probably are no better. However, it is anticipated that new baseload electric generating capacity will be needed soon after the year 2000 if not before.[12] In view of the attractive features of the HTGR and the considerable base of experience now available, it seems possible, if not probable, that the helium-cooled reactor can become a significant supplier of electric power in the United States. The prospects in the Federal Republic of Germany seem even better since their industrial capability for supplying the reactor is well-established through the recent construction of the THTR-300. Also, the German industrial complex is concentrated geographically, which facilitates transport of thermal energy; thus, the HTGR capability for supplying high temperature process heat is a significant incentive for its development and demonstration in the Federal Republic of Germany. Process heat application also appears to be of relatively early interest for Japan. In the long term, the application of heat energy for recovery of oil from tar sands is an attractive but not widely recognized incentive. In the meantime, the environmental and operational advantages of the HTGR in supplying nuclear electric energy provide merit for its application in many locations.

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