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A HISTORY OF AND PROSPECTS FOR GAS-COOLED REACTORS IN THE UNITED STATES

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by
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This is the United States input to a report on gas-cooled reactors being prepared by the International Atomic Energy Agency.

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ABSTRACT

This is the United States input to a report on gas-cooled reactors that is being prepared by the International Working Group on Gas-Cooled Reactors of the International Atomic Energy Agency. This report covers gas-cooled reactors in the United States starting with the conceptual work at Oak Ridge, Tennessee, immediately after World War II.

The Fort St. Vrain Nuclear Generating Station, which is the only operating High-Temperature Gas-Cooled Reactor (HTGR) in the United States and which recently achieved 100% power, is described in some detail.

The report covers the use of helium-cooled reactors for electricity generation, cogeneration of steam for industrial processes and for electric power, and process heat. Three types of reactors are described for electricity generation: the steam cycle HTGR, which is based on Fort St. Vrain but scaled up in size and with significant changes in layout and components; the gas turbine HTGR, which utilizes a direct cycle; and the Gas-Cooled Fast Breeder Reactor (GCFR), which is also based on Fort St. Vrain and which not only would produce electricity but would have the potential to achieve compound system doubling times of fissionable material of under ten years.

The HTGR is particularly well suited for the cogeneration of steam and electric power because it can deliver steam at temperatures up to 540°C and 17 MPa pressure. The cogeneration plant, which has the same nuclear steam supply system as the steam cycle HTGR, is described along with a number of possible industrial applications including petrochemical plants, in situ recovery of heavy oil, and synthetic fuel processes.

The process heat HTGR is being designed in two versions: an 850°C core outlet temperature with an intermediate helium loop and a 950°C core outlet temperature with a reformer and steam generator in the primary circuit. The high-grade heat from the HTGR can be used to produce hydrogen and synthetic fuel from coal and other fossil sources of carbon or for thermochemical watersplitting processes to produce hydrogen without carbon.

The technical performance, fuel cycles, safety characteristics, and environmental impact of the HTGR and GCFR are also discussed in the report.

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1. INTRODUCTION

In the spring of 1981, the International Working Group on Gas-Cooled Reactors of the International Atomic Energy Agency (IAEA) decided to prepare a report on gas-cooled reactors for publication in the Agency's Technical Report Series. This report, entitled "Status of and Prospects for Gas-Cooled Reactors," will be available to all 110 Agency Member States to inform them about the state-of-the-art of gas-cooled reactor power plants and reactor concepts for electricity generation, process heat application, and fissile fuel breeding.

This document is the U.S. contribution to the IAEA report. It describes the long and varied history of gas-cooled reactors in the United States, extending from studies at Oak Ridge National Laboratory immediately following World War II to the present steam cycle/cogeneration and high-temperature process heat reactors currently in various stages of design and development.

2. HISTORICAL DEVELOPMENT OF GAS-COOLED REACTORS

2.1. EARLY U.S. GAS-COOLED REACTORS

Gas-cooled reactors were the subject of studies carried out at Oak Ridge National Laboratory immediately after World War II. As part of this program, in 1946 Farrington Daniels proposed a semihomogeneous high-temperature, helium-cooled reactor for electric power production, and some conceptual design work was done on this reactor (Ref. 2-1). A summary of early helium-cooled, high-temperature reactor designs, including direct cycle plants in which the heated gas was used directly in gas turbines, was prepared by J. R. Johnson in 1957 (Ref. 2-2).

In the 1950s and early 1960s, a number of open cycle air-cooled and hydrogen-cooled reactors were built for aircraft and aerospace applications under the Aircraft Nuclear Propulsion (ANP), ROVER, and PLUTO programs (see Table 2-1) (Ref. 2-3). However, all of these programs were abandoned--the aircraft nuclear propulsion program because of the success of non-nuclear systems in meeting increasingly more stringent requirements and the aerospace applications because of the treaty limiting atmospheric testing of nuclear fission.

The U.S. Army built two small nuclear plants using nitrogen-cooled reactors with a closed cycle gas turbine, intending to develop a mobile

TABLE 2-1

SUMMARY OF GAS-COOLED AIRCRAFT AND AEROSPACE REACTORS

Program	Reactor	Thermal Power (MW)	Operation Date	Moderator	Coolant Temperature (°C)	Power Conversion
			Air-Cooled Reactors			
ANP	HTRE-1 and -2	17.5	Dec. 1955	Water	724	Turbojet
ANP	HTRE-3	32.4	Oct. 1958	ZrH	780	Turbojet
PLUTO	TORY-11-A-1	155	Dec. 1960	BeO	1080	Ramjet
PLUTO	TORY-11-C	150	1963	BeO	--	Ramjet
			Hydrogen-Cooled Reactors			
ROVER	KIWI-A	--	July 1959	Graphite	1650	Rocket
ROVER	KIWI-A-PRIME	--	July 1960	Graphite	1650	Rocket
ROVER	KIWI-A3	--	Oct. 1960	Graphite	1650	Rocket
ROVER	KIWI-B-1A	--	Dec. 1961	Graphite	1650	Rocket
ROVER	KIWI-B-1B	--	Sept. 1962	Graphite	1650	Rocket
ROVER	KIWI-B4A	--	Nov. 1962	Graphite	1650	Rocket

power plant. Nitrogen was used because it is cheaper than helium and less corrosive than carbon dioxide at elevated temperatures. The Gas Cooled Reactor Experiment (GCRE-1), designed for a thermal output of 2.2 MW, started operation in 1960 at the National Reactor Testing Station (NRTS) (Ref. 2-4). The heat removed by 1.2-MPa* nitrogen (with 0.5% O_2 to prevent nitriding of the materials) was transferred to a secondary circuit and dumped to the atmosphere. The Mobile Low Power Reactor (ML-1), similar in design to GCRE-1, had a closed cycle gas turbine and represented a real operational prototype (Refs. 2-5, 2-6). With 3.3 MW of thermal power, the net electrical output was 330 kW. The reactor went critical in 1961 at the NRTS in Idaho and achieved power in 1962-63. Lack of a suitable mission resulted in termination of the program.

The Maritime Gas Cooled Reactor (MGCR), with a 14.9-MW, helium-cooled closed cycle gas turbine propulsion plant, was designed in 1960 by General Atomic and Electric Boat, both divisions of General Dynamics (Ref. 2-7). The Experimental Beryllium Oxide Reactor (EBOR) was designed and built as a prototype for the MGCR to test the high-temperature behavior of BeO as a reactor moderator, together with helium cooling (Ref. 2-8). The selection of the EBOR fuel element was based in part on its inherent simplicity, readily predictable thermal performance, and satisfactory behavior in case of a loss of coolant accident. For EBOR, helium at 7.6 MPa entered the reactor at 400°C and exited at 690°C in 36 channels, each containing 18 fuel rods.

*1 MPa = 145 psi.

The fuel rods contained a mixture of 62.5%-enriched UO_2 and BeO within a 0.5-mm-thick Hastelloy cladding. These 8.5-mm diameter, 2550-mm-long rods, spaced by helical wires, operated at a maximum surface (hot spot) temperature of 860°C, equalized in each fuel element by proper orificing. Construction of this 10-MW(t) reactor was begun at the NRTS in 1964, but because of budget restrictions it was mothballed in 1967 after completion but prior to fuel loading.

During the mid-1950s, the Congress of the United States observed the expanding British program on gas-cooled reactors and urged the U.S. Atomic Energy Commission to look into this type of coolant (Ref. 2-9). The result was the Experimental Gas-Cooled Reactor (EGCR), a 22-MW(e) plant that was similar in concept to the British Advanced Gas-Cooled Reactor but used helium rather than carbon dioxide as the coolant. The key design data for the EGCR are given in Table 2-2 (Ref. 2-10). The EGCR program was terminated in 1966, after most of the construction had been completed, on the basis that it was not competitive with Light Water Reactors (LWRs).

The temperature, power density, and fuel burnup limitations of graphite-moderated reactors using metal-clad fuel led to consideration of helium-cooled reactors with all-ceramic cores. The High-Temperature Gas-Cooled Reactor (HTGR) was conceived by Peter Fortescue in 1954. The use of helium coolant eliminates the problem of corrosion and possible mass transfer of carbon that is encountered in reactors cooled by carbon dioxide, particularly at high temperatures. Helium is a gas (as is CO_2) under all

TABLE 2-2

EXPERIMENTAL GAS-COOLED REACTOR DESIGN DATA

Reactor Type	Helium cooled, graphite moderated
Heat Output	85 MW(t)
Electrical Output	22.3-MW(e) net
	29.5-MW(e) gross
Location	Oak Ridge, Tennessee, USA
Reactor Designer	Oak Ridge National Laboratory
Coolant	
Inlet Temperature	265°C
Outlet Temperature	566°C
Pressure	2.1 MPa
Steam	
Temperature	482°C
Pressure	8.9 MPa
Flow Rate	33.2 kg/s
Pressure Vessel	
Diameter	6.1 m
Height	14.0 m
Core	
Diameter	4.4 m
Height	4.9 m
Average Power Density	1.97 MW/m^3
Fuel Elements	
Type	7-rod clusters
Number	1416
Fuel	UO_2
Enrichment	2.5%
Cladding	Stainless steel

conceivable reactor conditions, and this provides a significant safety advantage (no voids or two-phase flow).

2.2. PEACH BOTTOM ATOMIC POWER STATION - UNIT 1

In the late 1950s, General Atomic Company proposed that a 40-MW(e) HTGR be built, and this was done at Peach Bottom, Pennsylvania, on the system of Philadelphia Electric Company with financial support from 52 other electric utility companies (High Temperature Reactor Development Associates) (Ref. 2-11). It operated from 1967 to 1974 under the U.S. Atomic Energy Commission's Power Reactor Demonstration Program.

The nuclear steam supply (NSS) system, which was supplied by General Atomic, achieved an availability of 88%, excluding planned shutdowns for research and development programs, during the 7 yr of plant operation. The plant produced 538°C steam at 10.0 MPa and had a net efficiency of 34.6%. The helium temperature was 728°C at the reactor outlet.

The primary coolant system consisted of the reactor vessel and two coolant loops, each containing a steam generator and helium circulator as shown in Fig. 2-1. The hot helium from the core circulated through concentric pipes to the steam generators, where it was cooled to 343°C. The cold gas was then returned by the circulators through the annulus of the concentric pipes to the reactor vessel. Baffling and thermal shielding channeled the cold gas inside the reactor vessel to cool the shell and internals before returning the helium to the core. The steam generators were of the

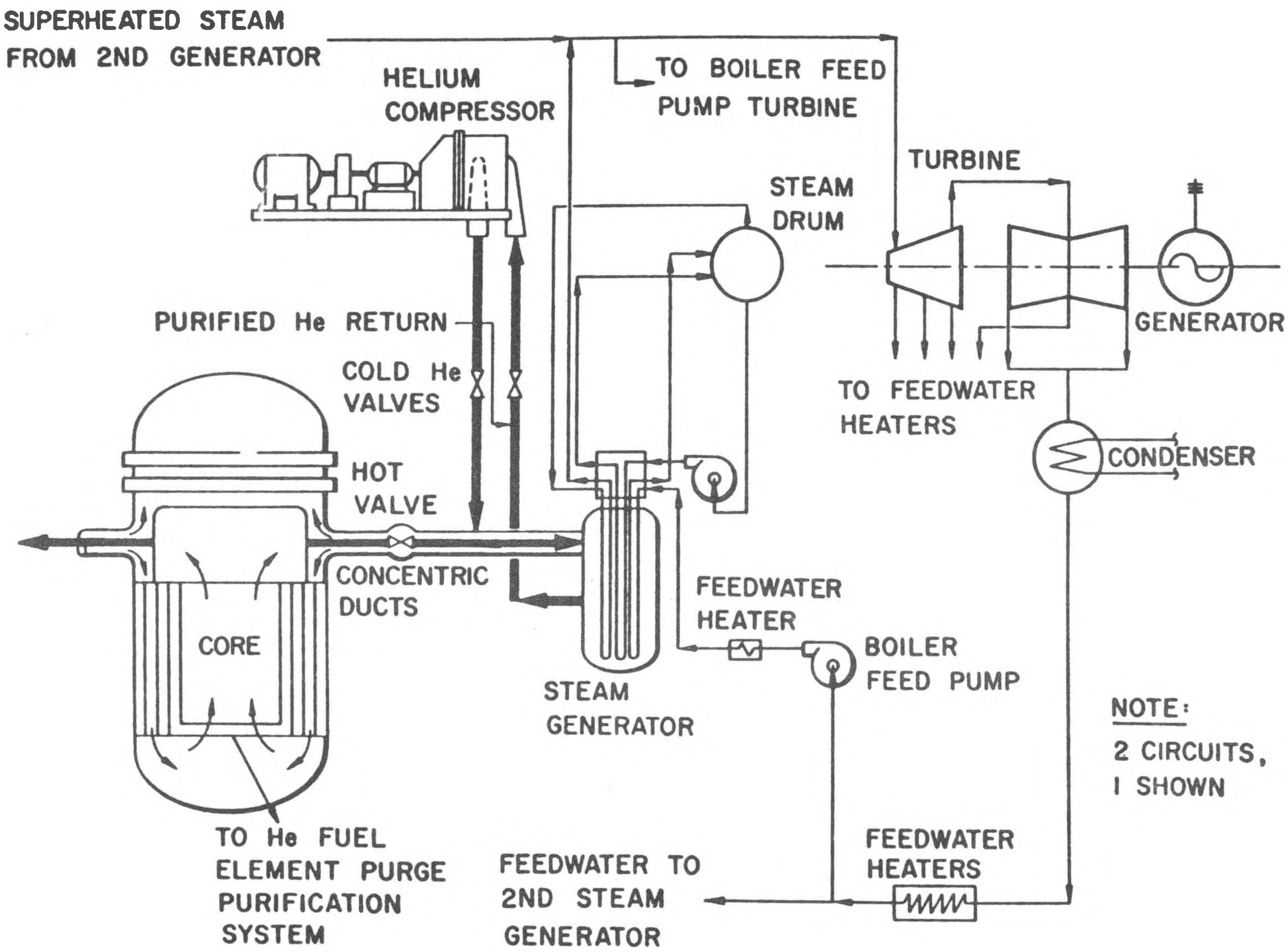

Fig. 2-1. Simplified process flow diagram for the main coolant system of the Peach Bottom HTGR

vertical shell and tube type with banks of U-tubes for the economizer, evaporator, and superheater sections contained in a single carbon steel shell. During the entire life of the plant, not a single leak occurred in any tube of either of the steam generators.

The fuel for the Peach Bottom reactor consisted of pyrolytic-carbon- and silicon-carbide-coated uranium-thorium dicarbide fuel particles that were dispersed in graphite compacts and encased in a graphite sleeve to form a fuel element (Fig. 2-2). There were 804 fuel elements, 89 mm in diameter and 3660 mm long, which were oriented vertically within the reactor vessel. Helium at a pressure of about 2.4 MPa flowed through the tricusp-shaped coolant channels between the fuel elements to remove the heat produced in the core. A small helium stream passed down through each fuel element and purged any fission products leaking from the fuel particles to the helium purification system. This system removed fission products from the fuel element purge stream by adsorption on charcoal beds operating at low temperature. Sufficient delay time was provided to permit the decay of almost all fission products with the exception of Kr-85. An oxidizer, a dehydrator, and a liquid-nitrogen-cooled charcoal trap were provided to remove moisture, chemical impurities, and the Kr-85.

During Peach Bottom operation, plant personnel received very low radiation levels. The average radiation exposure was 10 mR/yr per individual (Ref. 2-12).

The main plant parameters are shown in Table 2-3.

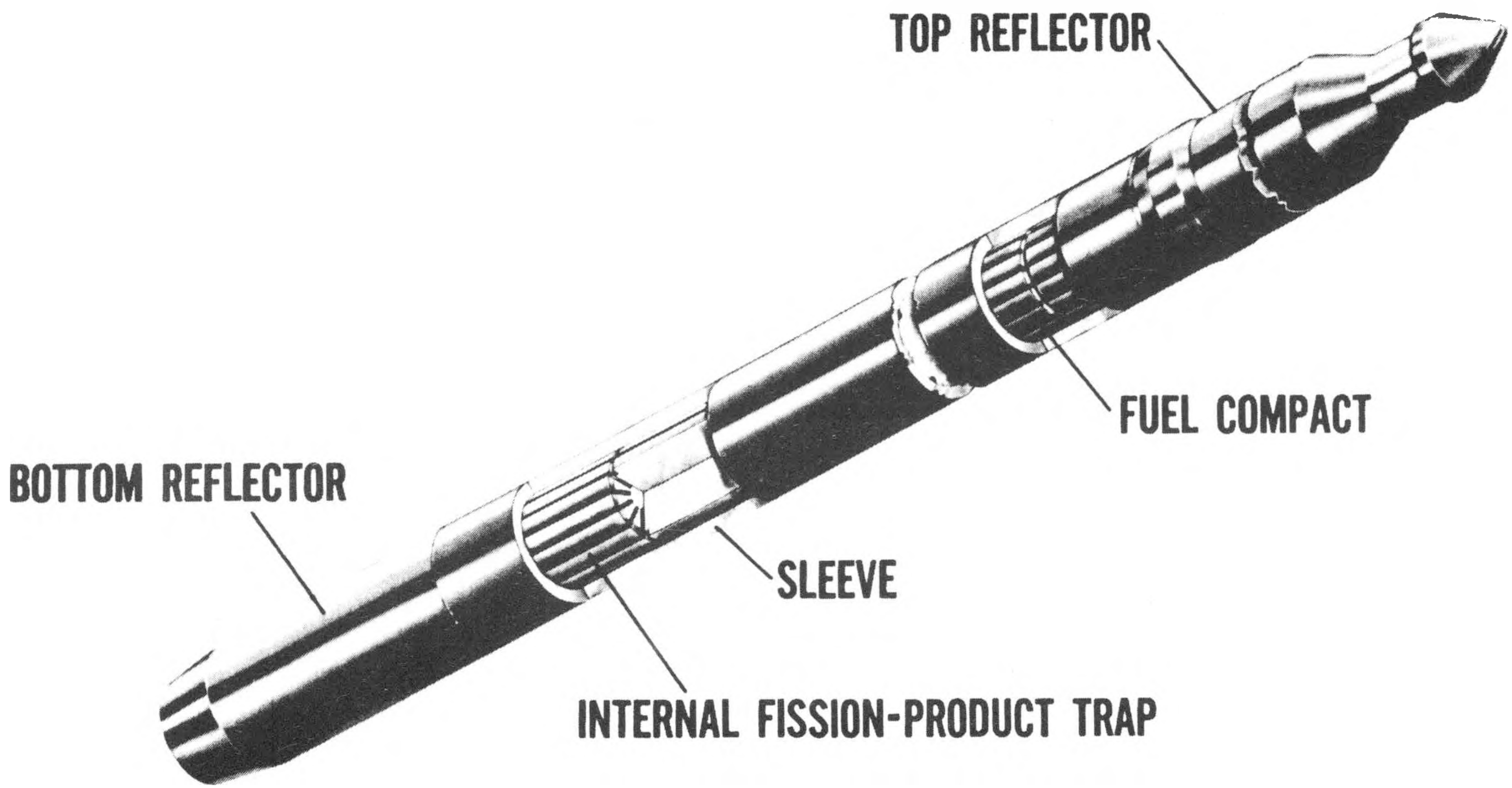

Fig. 2-2. Peach Bottom fuel element

TABLE 2-3

PEACH BOTTOM ATOMIC POWER STATION PLANT CHARACTERISTICS[a]

Coolant	Helium
Pressure	2.4 MPa
Core Inlet Temperature	344°C
Core Outlet Temperature	728°C
Steam Temperature	538°C
Steam Pressure	10.0 MPa
Net Thermal Efficiency	34.6%
Reactor Thermal Output	115 MW(t)
Net Electrical Power	40 MW(e)
Pressure Vessel	
Diameter	4.3 m
Height	10.8 m
Core	
Effective Diameter	2800 mm
Active Height	2300 mm
Fuel Element Diameter	89 mm
Fuel Element Length	3660 mm
Number of Fuel Elements	804
Reflector Thickness	610 mm
Control Rods	
Normal Operating Rods	36
Shutdown Rods	19
Fuel Life at Full Power	900 days

[a]Ref. 2-13.

2.3. FORT ST. VRAIN NUCLEAR GENERATING STATION

During the construction and start-up of the Peach Bottom reactor, design and development were proceeding on larger HTGRs. In 1965 a contract was signed by General Atomic, the Public Service Company of Colorado, and the U.S. Atomic Energy Commission for a 330-MW(e) HTGR nuclear generating station to be built at Fort St. Vrain, about 35 miles north of Denver, Colorado (Ref. 2-14). This plant, which was part of the third phase of the U.S. Atomic Energy Commission's Power Reactor Demonstration Program, was issued a construction permit in September 1968 and construction started immediately thereafter. New features included in this HTGR plant, compared with the Peach Bottom station, are a prestressed concrete reactor vessel (PCRV) enclosing the whole primary system, once-through modular steam generators with integral superheaters and reheaters, steam-driven axial flow helium circulators, and hexagonal graphite fuel elements incorporating improved coated fuel particles. Also, the reactor is located within a confinement building rather than a conventional secondary reactor containment. The reactor arrangement is shown in Fig. 2-3.

Site construction was started in September 1968. Fuel was loaded in late 1973, and initial criticality was achieved in early 1974. Power generation commenced in late 1976, but the plant was limited to 70% power because of core region gas outlet temperature fluctuations. Mechanical restraints placed in existing holes on top of the top reflector tied the core regions together and stopped the fluctuations. The U.S. Nuclear Regulatory

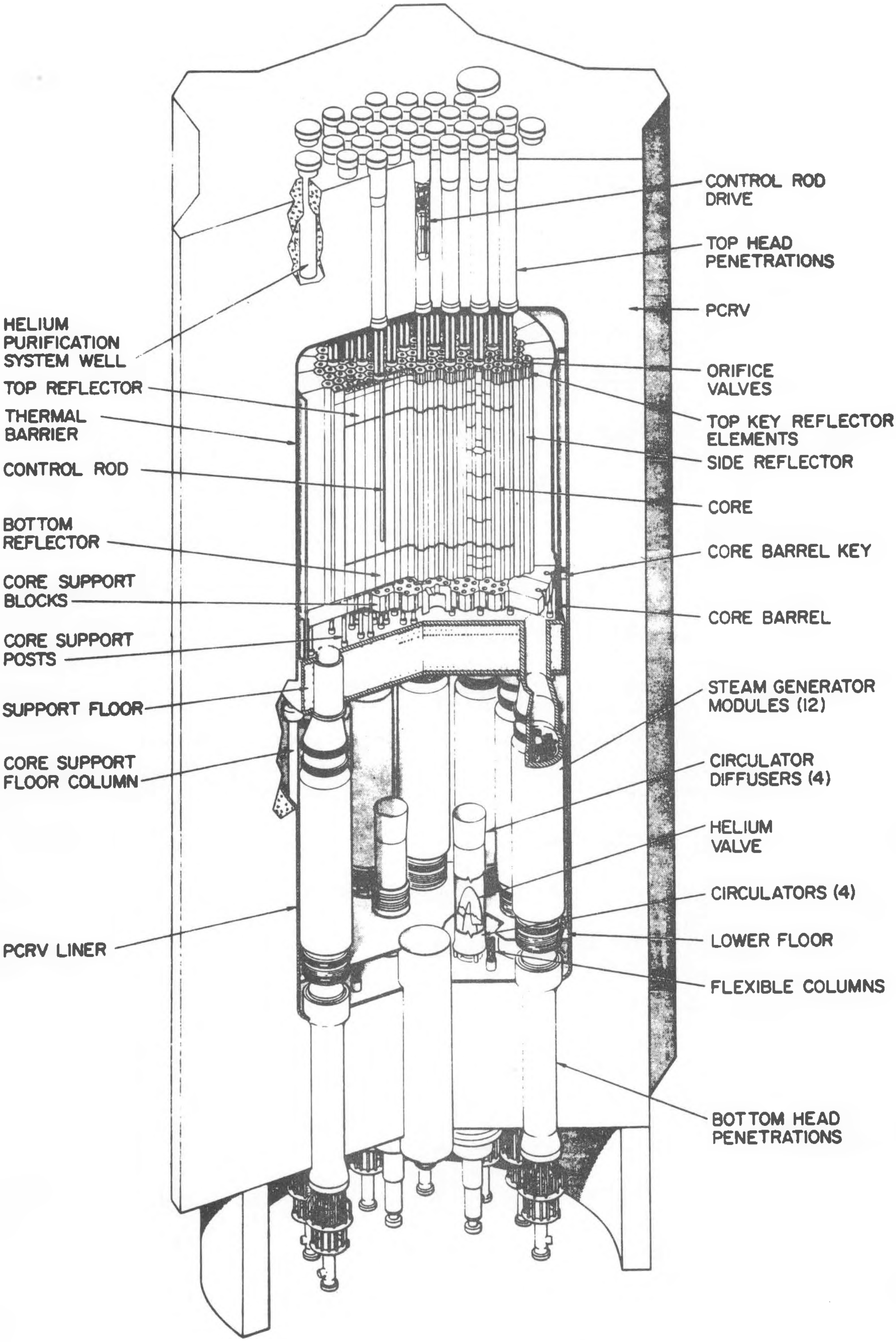

Fig. 2-3. Fort St. Vrain NSS system

Commission then gave permission to test the reactor at higher power levels, and 100% power was achieved in November 1981.

2.4. LARGE HIGH-TEMPERATURE GAS-COOLED REACTORS

During the period from 1971 to 1973, U.S. utility companies placed orders with General Atomic for ten HTGRs: four 770-MW(e) and six 1160-MW(e) units. The designs were based on the Fort St. Vrain reactor except for the following: the PCRV was changed to a multicavity type to improve seismic characteristics and to allow for easier replacement of steam generators; much of the auxiliary equipment was modularized; and the fissile and fertile fuel was separated into two types of particles.

This initial commercial success fell victim to the wholesale cancellations and deferrals of nuclear contracts that followed the Arab oil embargo. As the newest vendor, General Atomic was particularly affected and, realizing that it was no longer feasible to commercialize the HTGR using private funds, withdrew from the market. A cooperative development effort involving the Federal government, utilities, and potential suppliers was launched in 1975.

In 1978 the utility supporters of the HTGR formed a nonprofit corporation, Gas-Cooled Reactor Associates, to develop the HTGR for the production of electricity and other industrial and commercial applications. This group now acts as coordinators of the United States program. In 1979 the various participants agreed to focus the national program on a 2240-MW(t) plant for electricity or cogeneration.

2.5. ULTRA HIGH TEMPERATURE REACTOR EXPERIMENT (UHTREX)

Interest in the HTGR for high-temperature process heat applications originated in the early 1960s. The U.S. Atomic Energy Commission sponsored the design, construction, and operation of the 3-MW(t) UHTREX, which was built by the Los Alamos Scientific Laboratory and operated there between 1966 and 1970 (Refs. 2-15, 2-16). The purpose of this experimental reactor was to provide very-high-temperature helium (1300°C) at 3.4 MPa and to test various ceramic fuels in a graphite-moderated core. A cutaway view of the reactor is shown in Fig. 2-4. The fuel elements consisted of extruded hollow cylinders made of a mixture of graphite and 93%-enriched pyrolytic-carbon-coated (Triplex) UC_2 particles.

The reactor was designed with the following unique features:

1. An annular core that was rotatable to facilitate on-line refueling.

2. Articulated control rods that could always be inserted directly into the core.

3. The possibility of operation with a contaminated circuit and on-line coolant cleanup.

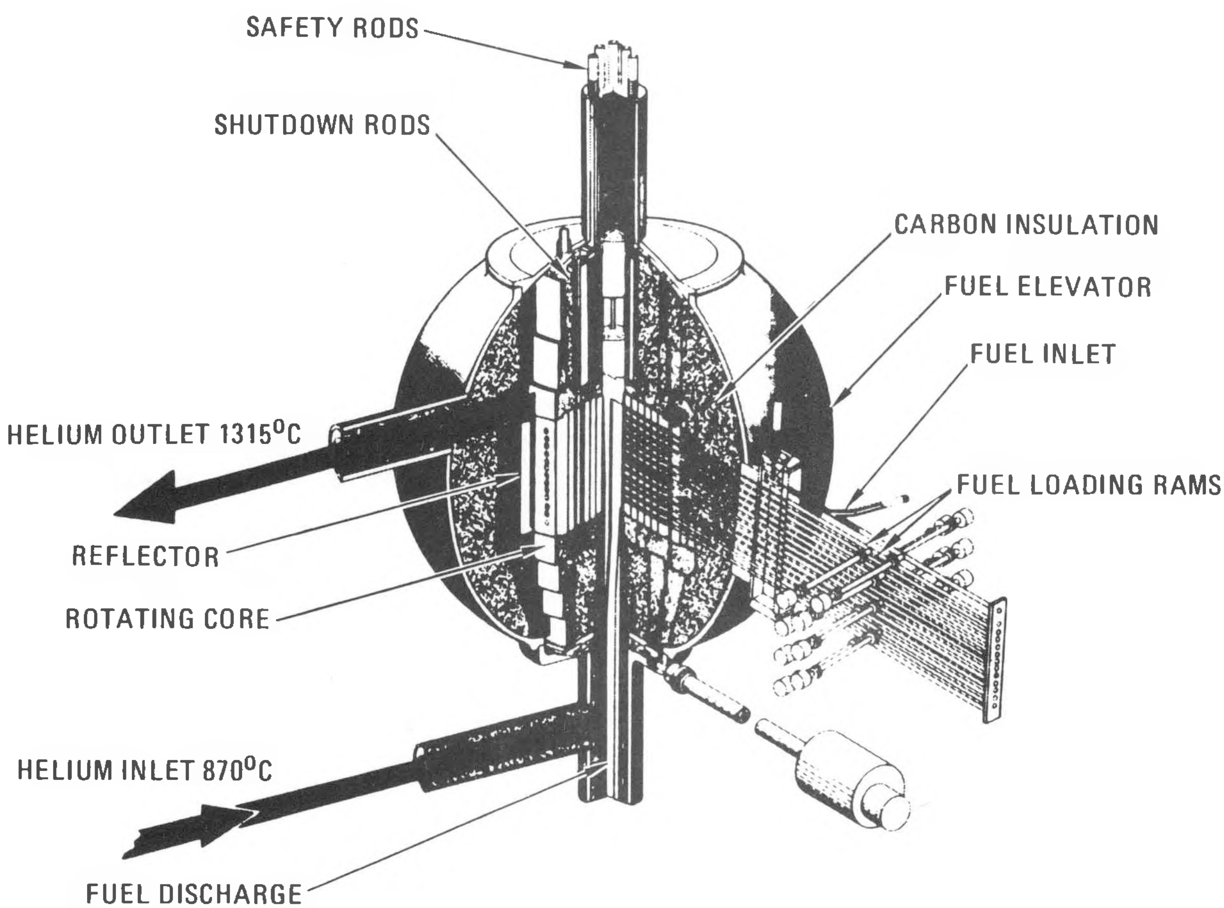

Fig. 2-4. Ultra High Temperature Reactor Experiment (UHTREX)

Helium entered the core at its axis at a temperature of 870°C, flowed through the fuel channels and exited at 1315°C, entered a recuperator from which it emerged at 760°C, and flowed through a helium heat exchanger where it was cooled to 315°C. It was then pumped back into the recuperator by a gas-bearing 48.5-kW centrifugal blower. The secondary helium loop operating at a maximum temperature of 540°C dumped the heat through an air-cooled heat exchanger. The reactor vessel was 3.75 m in diameter.

This reactor experiment showed the feasibility of producing 1300°C helium, which is much hotter than the temperatures needed for electricity-producing HTGRs (700° to 800°C) and even for high-temperature process heat reactors (800° to 1000°C). The UHTREX was shut down in 1970.

The gas-cooled reactors that have been built in the United States for terrestrial or maritime use are described in Table 2-4.

2.6. GAS-COOLED FAST BREEDER REACTOR (GCFR)

If nuclear fission is to make a significant long-term contribution to world energy needs, breeder reactors must be developed. They have been recognized as necessary for full utilization of fission power since the inception of nuclear energy. The capability of nuclear reactors to breed more fissile fuel than they consume was predicted very early in the reactor program, and the concept had achieved substantial technical acceptance by the mid-1940s. If successful, they could supply energy to the world for millenia.

TABLE 2-4

GAS-COOLED REACTORS CONSTRUCTED FOR TERRESTRIAL OR MARITIME USE IN THE UNITED STATES

Reactor	MW(t)	MW(e)	Operation Dates	Coolant	Moderator	Fuel	Cladding	Purpose
GCRE-1	2.2	--	1960 - 1961	Nitrogen	Water	UO_2 pellets	Hastelloy X	Mobile military power (prototype)
ML-1	3.3	0.33	1962 - 1963	Nitrogen	Water	UO_2 pellets	Hastelloy X	Mobile military power
EBOR	10	--	--	Helium	Beryllium oxide	UO_2-B_eO pellets	Hastelloy X	Maritime propulsion (prototype)
UHTREX	3	--	1966 - 1970	Helium	Graphite	UC_2 particles	Pyrolytic carbon coatings	High-temperature helium for process heat
EGCR	85	22.3	--	Helium	Graphite	UO_2 pellets	Stainless steel	Electric power
Peach Bottom Unit 1	115	40	1967 - 1974	Helium	Graphite	$(U\text{-}Th)C_2$ particles	Pyrolytic carbon and silicon carbide coatings	Electric power
Fort St. Vrain	842	330	1977 →	Helium	Graphite	$(U\text{-}Th)C_2$ and ThC_2 particles	Pyrolytic carbon and silicon carbide coatings	Electric power

In general, high-gain breeders are "fast" reactors; that is, they utilize, without moderation or slowing down, the high-velocity neutrons that are emitted when an atom fissions. Since the first fast reactors had small high-power-density cores with little room for cooling, Liquid Metal Fast Breeder Reactors (LMFBRs) were the type of breeders that received the earliest and greatest attention in the United States as well as in most other countries. Starting in the early 1960s, the possibility of using high-pressure helium as a coolant in a fast breeder reactor was investigated (Ref. 2-17). The use of helium as a coolant has several advantages: (1) helium is transparent to neutrons, leading to a very high breeding ratio; (2) helium is chemically inert, allowing the use of direct transfer of heat from the reactor coolant to water in a steam generator; (3) helium is a gas under all reactor conditions, and therefore some of the problems associated with phase changes are eliminated; and (4) helium is visually transparent, which eases some of the operating and maintenance problems.

Work on GCFR design and development has proceeded in parallel over the past 20 yr. The former has led to the conceptual design of the NSS system for a 350-MW(e) demonstration plant (Ref. 2-18). This design features an upflow core, pressure-equalized and vented fuel rods, a multicavity PCRV, electric motor drives for the main helium circulators, and extensive provisions for residual heat removal. There are three main and three auxiliary loops of diverse design, and a pony motor on the main loop provides a backup safety class system. In addition, the core auxiliary cooling system can operate with natural circulation. The main characteristics of this plant are given in Table 2-5. A cutaway of the NSS system is shown in Fig. 2-5.

TABLE 2-5

GCFR PLANT PARAMETERS

Overall Plant	
Thermal Power	1090 MW(t)
Net Electric Power	360 MW(e)
Net Plant Efficiency	33%
Number of Main Loops	3
Circulator Power	10.7 MW/loop
Reactor	
Fuel Material	$(P,U)O_2$
Fuel Rod Diameter	8.0 mm
Blanket Material	UO_2
Maximum Cladding Temperature	750°C
Reactor Inlet Temperature	298°C
Reactor Outlet Temperaure	524°C
Helium Pressure	10.5 MPa
Reload Interval	1/3 of core/yr
Breeding Ratio	1.35
Power Conversion System	
Turbine Inlet Pressure	10.0 MPa
Turbine Inlet Temperature	482°C

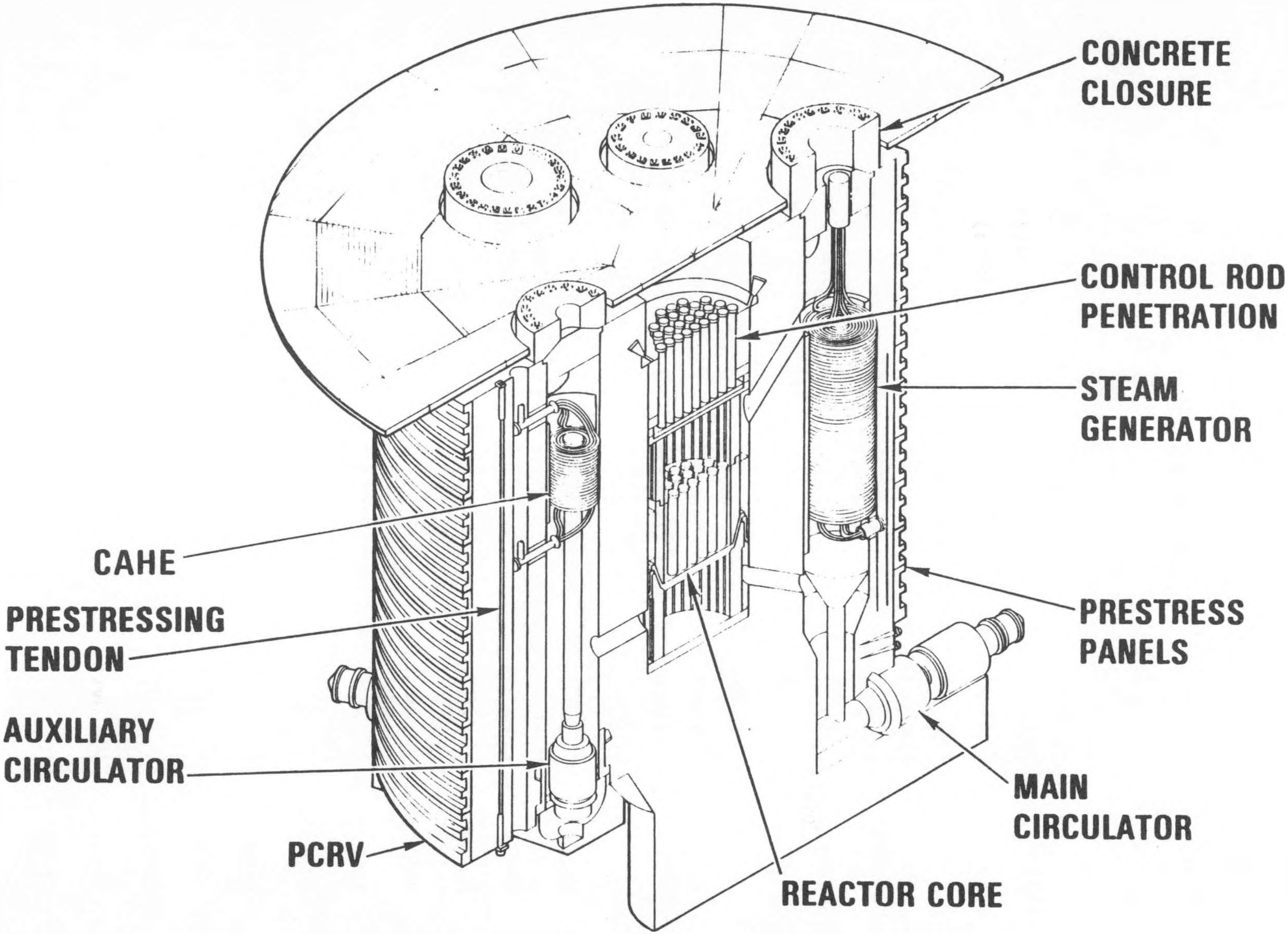

Fig. 2-5. GCFR demonstration plant NSS system

In parallel with the design effort, extensive research and development was carried out in the United States in cooperation with Switzerland and the Federal Republic of Germany. This work covered a series of critical experiments to verify physics parameters, neutron streaming tests, an extensive program of thermal-hydraulic studies on core rods and subassemblies, and an irradiation program emphasizing the differences between the LMFBR and GCFR fuel rods.

Because of critical funding problems, U.S. government support for the program was terminated and all development work in the United States was stopped at the end of September 1981.

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3. TECHNICAL STATUS OF THE APPLICATION OF GAS-COOLED REACTORS

3.1. ELECTRICITY GENERATION

3.1.1. High-Temperature Reactors

3.1.1.1. Incentives for Deployment. The HTGR-SC is a second-generation nuclear power system that offers potential advantages to the electric utility industry as well as the United States as a whole. From the utility perspective, the key interest factors are:

1. A high-efficiency power cycle and a high-conversion fuel cycle, which combine to offer a competitive power plant alternative.

2. Inherent safety/operational characteristics that offer the potential for improved public acceptance and improved protection of the owner's capital investment.

3. Potential for being easily sited owing mainly to low radiological releases and reduced cooling water requirements.

4. Potential for lower operating and maintenance costs due to reduced levels of radioactive contamination, system simplicity, and early

design attention to operation, inspection, and maintenance functions.

From the public sector perspective, the added key interest factors are:

1. The relatively low front-end public investment required to make this option available. In light of the uncertainties with current nuclear systems and the limitations and environmental constraints associated with coal, the pay-off for this investment may be profound.

2. Minimal environmental impact plus conservation of key resources, i.e., water, fossil fuels, and uranium. In addition, the HTGR efficiently exploits thorium resources and hence enhances fuel supply diversity.

The HTGR-SC system is a developed concept that applies to the well-established market for electrical generation. Based upon a substantial accumulation of effort both in the United States and Europe, the current design also reflects a detailed assessment by utility interests. While the orders for new generation plants have been depressed in recent years, it is essential that such orders resume to support continued economic growth and to replace gas- and oil-fired power plants as well as the older, inefficient nuclear plants. Market studies consistently project a steady growth in generation capacity that is expected to be met by coal and nuclear. Estimates have been developed that indicate approximately one-fourth of the electricity generation market in the United States is located in regions that will face limitations of water for power plant cooling purposes beyond the turn of the century. The size of the electrical generation market and the expected increasing siting constraints, such as cooling water limitations, combine to support the deployment of the HTGR-SC as a power plant alternative.

Product costs have been estimated and projected for an equilibrium HTGR-SC, an LWR, and a coal plant. The HTGR-SC plant is competitive with the LWR, with site-specific factors being the dominant basis for plant evaluation. Relative to coal, the evaluation of product costs is even more site-dependent owing to the coal transportation costs. However, for most regions of the United States, nuclear power costs are expected to command an increasing advantage over coal.

3.1.1.2. Fort St. Vrain. The Fort St. Vrain 330-MW(e) HTGR nuclear power plant, owned and operated by the Public Service Company of Colorado, U.S.A., was described briefly in Section 2. The plant had generated 2,849,000 MW-h of electricity by the end of 1981. Following the achievement of 100% power during a test run in November 1981, it is hoped that the limitation of continuous operation at 70% power level that has prevailed for the past several years will be lifted. The Fort St. Vrain plant is described below in more detail.

As shown in Fig. 2-3, the primary circuit is wholly contained within the PCRV: the core and reflector in the top cavity and the steam generator and circulators in the lower cavity. The PCRV (Ref. 3-1) acts both as pressure vessel and biological shield. The internal dimensions are 9.5 m in diameter and 23 m high, while the exterior surface is approximately a hexagonal prism, 15 m across flats and 42 m high. The top head of the vessel has 37 refueling penetrations, which also house the control rod drives. The bottom head has 12 penetrations for the steam generator modules, four for the helium circulators, plus a large central opening for access. All PCRV penetrations are provided with two independent closures: the PCRV inner cavity and the primary closures act as primary containment for the reactor; the PCRV itself and the secondary closures act as the secondary containment. A 19-mm-thick carbon steel liner anchored to the concrete provides a helium-tight membrane. Two independent systems of water-cooled tubes welded to the concrete side of the liner and a thermal barrier on the reactor side of the liner control the temperatures in both the liner and concrete and limit heat losses. Sets of horizontal and vertical tendons located in steel tubes in the concrete are used to place the concrete structure in compression. These redundant tension members, removed from both radiation and elevated-temperature fields, can be monitored and even replaced if ever needed.

The helium coolant at about 4.8 MPa and 405°C flows downward through the reactor core, where it is heated to an average temperature of 775°C. It is then directed to the steam generators located below the core support floor to produce reheated steam (3.9 MPa/538°C) and superheated steam (16.5 MPa/538°C). The helium, at 405°C, then flows through the four circulators, which discharge into a common plenum below the core support floor. All of the flow passes upward around the core support floor and the barrel to the core inlet plenum above the reactor before returning to the core. The coolant flow out of the reactor divides equally into two loops, each consisting of a six-module steam generator and two helium circulators.

The Fort St. Vrain reactor core (Ref. 3-2) is composed of 1482 hexagonal graphite fuel elements arranged in 247 columns and grouped in 37 refueling regions. The coolant flow in each region is controlled by an orifice valve at the top of the core. Each year during shutdown about one-sixth of the regions are refueled through PCRV penetrations centrally located over each region. One pair of B_4C control rods operated by electric drives and cable drums is provided for each refueling region; as reserve shutdown, B_4C spheres can fall by gravity into the reactor core. The active core, 5.9 m in diameter and 4.75 m high, is surrounded by a graphite reflector 1.2 m thick with one row of replaceable hexagonal graphite blocks. The top graphite reflector is about 0.8 m thick, and the bottom reflector is 1.2 m thick. Each hexagonal fuel element is made up of a graphite prism, 360 mm across flats and 793 mm high with 108 15.9 mm diameter coolant channels and 210 12.7-mm-diameter fuel channels, and a central hole for handling (Fig. 3-1). Six fuel elements are stacked on top of each other to form a column and are aligned with three graphite dowels each. The central fuel element of each refueling region has three larger holes, two 102 mm in diameter for control rods and one 95 mm in diameter for the reserve shutdown spheres. The fuel holes are filled with rods made of coated fuel particles bonded together by a graphite matrix. Two types of coated particles are

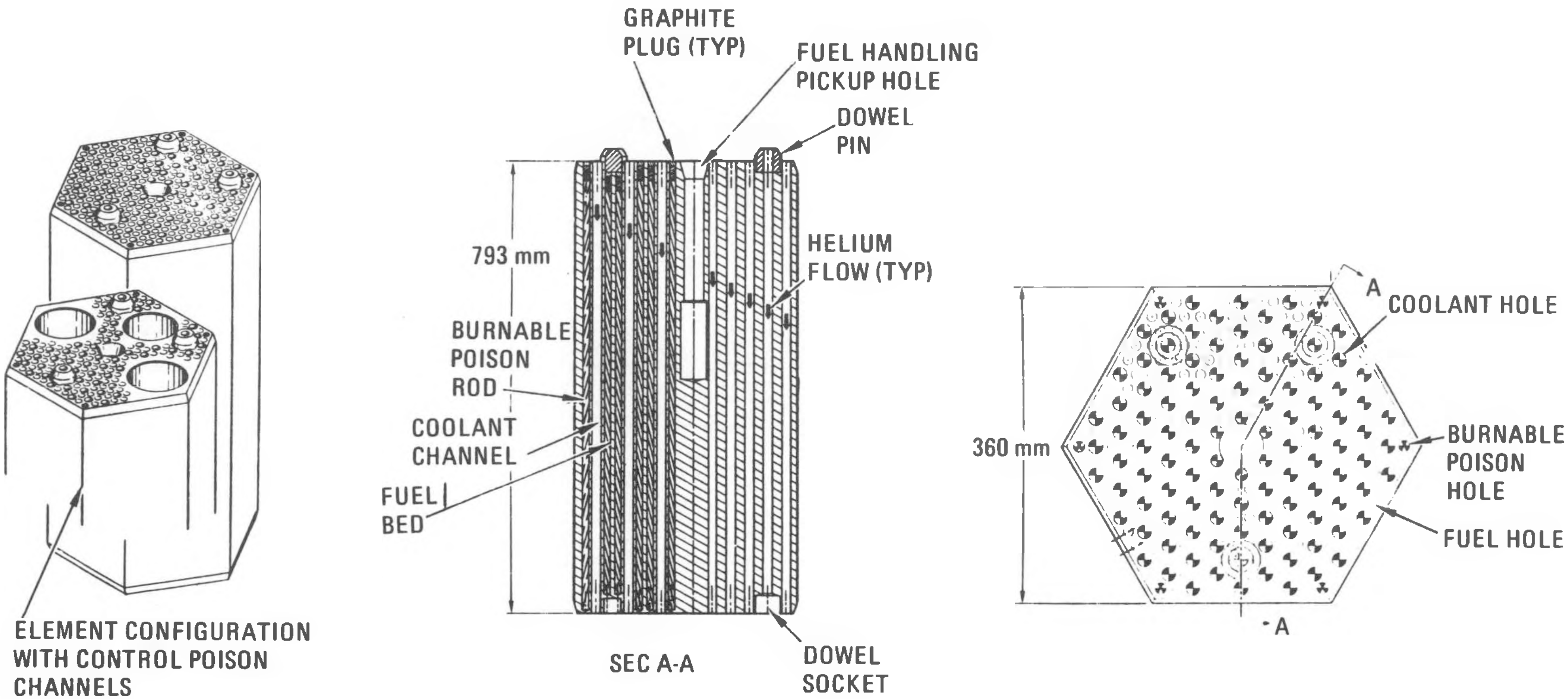

Fig. 3-1. Fort St. Vrain fuel element

used: 400-μm-diameter ThC_2 kernels and 200-μm-diameter $(Th,U)C_2$ kernels. Both particles have about 120-μm TRISO coatings, consisting of a first buffer layer of low-density pyrolytic carbon, an inner high-density isotropic pyrolytic carbon layer, an intermediate SiC barrier, and an outer isotropic pyrolytic carbon layer. These coatings are designed to retain fission products and withstand the effects of irradiation and of fuel burnup up to 20% fission per initial heavy metal atom in the fissile particles. The maximum fuel temperature is 1260°C, the maximum burnup is 200 MWd/kg, and the maximum fast fluence ($E > 0.18$ MeV) is 8×10^{21} n/cm^2. The initial fuel loading consists of 770 kg of uranium (in the form of 93% enriched in U-235), and 160,000 kg of thorium. Approximately 75% of the fissions in the fuel at end of life will take place in U-233 bred from Th-232. U-235 and thorium atoms constitute only about 1% of the total atoms in the fuel, and this dilution ensures both structural integrity of the graphite elements and a high degree of safety because of the large heat capacity of the graphite.

Figure 3-2 shows the flow diagram for the Fort St. Vrain power plant. Feedwater at 206°C is transformed into superheated steam at 538°C in the once-through steam generator and flows to the high-pressure turbine. Cold reheat steam then flows to the helium circulator turbines before being reheated in the reheater section of the steam generator from where it emerges at 538°C to return to the intermediate-pressure turbine. The turbine-generator is a conventional 3600-rpm, tandem-compound, two-casing condensing turbine driving a synchronous generator. The rated exhaust pressure from the steam turbine is 8.4 kPa, and the condenser water is cooled in an induced-draft wet cooling tower. The steam generators are designed to

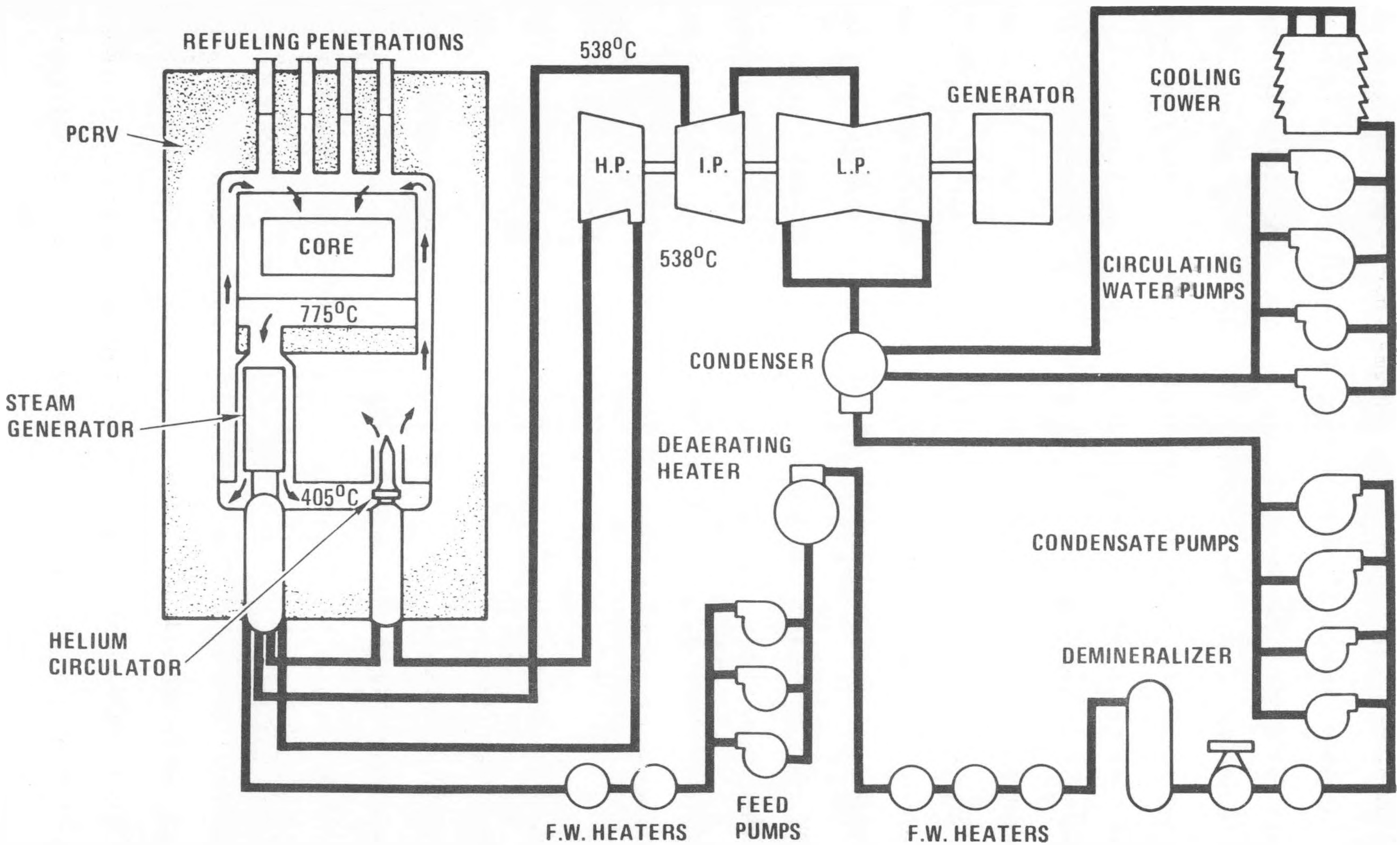

Fig. 3-2. Simplified Fort St. Vrain flow diagram

operate continuously at full load or part load (down to 25%) and also to remove decay heat during reactor shutdown (Ref. 3-3). Hot helium from the reactor enters the top of the steam generator module and flows downward through reheater, superheater, and evaporator-economizer helically coiled tube bundles in succession. The superheater is designed for co-current flow of steam and helium, while the reheater is designed for countercurrent flow. All tubes are drainable, and all water and steam piping go through bottom penetrations in the PCRV.

Each of the four helium circulators consists of a single-stage axial flow compressor, a single-stage steam turbine main drive, and a single-stage water turbine auxiliary drive, all mounted on a single vertical shaft, with water lubrication (Ref. 3-4). The steam turbine drives are normally supplied with cold reheat steam from the main turbine at 5.8 MPa and 342°C, although auxiliary steam could also be used. High-pressure water can be provided in emergencies to the auxiliary Pelton wheel drive. The normal speed of the circulator is 9550 rpm at a helium flow rate of 110 kg/s with a pressure rise of 0.1 MPa corresponding to a power of about 4.5 MW. A helium purification system maintains a low impurity level in the primary loop and provides purified helium to various subsystems, such as the helium circulator seals.

Although the Fort St. Vrain plant has encountered problems of a type that would be expected in a first-of-a-kind design, it has successfully demonstrated the basic performance of the HTGR concept. The high thermal efficiency inherent in the HTGR design has been demonstrated with an efficiency of 38.5% at the 100% power level.

The graphite-moderated prismatic block core components were found to be in excellent condition following the detailed inspection during refueling. The once-through steam generator modules have continuously operated at rated steam conditions of 538°C at both the main and reheat steam outlets. Only one leak in a steam generator module was experienced, and it was readily detected and quickly plugged. Fission product release from fuel has been even lower than the very low initial predictions. Measured circulating coolant activity levels to date are about 1% of those permitted by the technical specifications.

The four helium circulators, driven by axial flow steam turbines utilizing cold reheat steam, have performed within approximately 2% of predicted values. These circulators turn on hydrodynamic bearings that have also performed successfully. The circulator auxiliary system, which provides pressurized water for these bearings and pressurized helium for the buffer seals on the circulator shaft, has been a source of some problems. Upsets in the auxiliary system have caused water to travel up the shaft through the seals and into the primary system. Moisture removal problems in the buffer helium system have also resulted in moisture injection into the primary system. Consequently, a number of circulator auxiliary system modifications have been made, and additional improvements are currently under study.

Of particular note is the very low personnel radiation exposures that have occurred [<0.1 man-rem/MW(e)-yr (Ref. 3-5)], which confirm the excellent results obtained at the Peach Bottom plant.

Many lessons have been learned from the design and operation of the Fort St. Vrain reactor, and a significant effort has gone into applying them to future HTGRs.

3.1.1.3. HTGR - Steam Cycles. In the past few years, HTGRs for electricity production have been through a number of design iterations, and two paths have emerged: (1) the conventional steam cycle and (2) the use of a gas turbine in a direct cycle. In 1980 it was concluded that an extensive development effort was necessary to establish a technically viable gas turbine HTGR plant and that further design innovation was necessary to identify plant features for improved economics. Accordingly, the steam cycle HTGR (HTGR-SC) was designated as the lead plant and the gas turbine HTGR (HTGR-GT) was classed as a long-term option. At the present time, the steam cycle/cogeneration option has been chosen as the lead plant. This plant is discussed in Section 3.2 and features the same NSS as the HTGR-SC.

NSS System

The current design of the NSS system for the HTGR-SC (Ref. 3-6) is similar to earlier General Atomic designs of large HTGRs but differs in two ways in addition to the changes from the Fort St. Vrain design mentioned in Section 2:

1. The gas reheaters are removed from the NSS.

2. The steam turbine drives for the main helium circulator are replaced by electric motor drives.

The 2240-MW(t) [900-MW(e)] HTGR-SC plant serves as a reference design for base load electricity generation. In this system, high-quality steam is generated at the elevated conditions of 17.2 MPa/541°C, with the primary helium gas conditions maintained in the low range of HTGR capability at less than 700°C. The plant features a modular four-loop primary coolant system that can be scaled up or down by varying the number of loops from two to six while maintaining the basic NSS system configuration.

The HTGR core is cooled with pressurized helium, moderated and reflected with graphite, and fueled with a mixture of uranium and thorium. It is constructed of prismatic hexagonal graphite blocks with vertical holes for coolant channels, fuel rods, and control rods. The entire reactor core, together with other major primary system components, is contained in a multicavity PCRV. Helium coolant flows from electric-motor-driven circulators through the core, through the steam generators (each located in separate cavities in the PCRV wall), and back to the circulators. Superheated steam produced in the once-through steam generators is expanded through a tandem compound turbine-generator. Exhaust steam is condensed in a water-cooled condenser, and waste heat is rejected to the atmosphere in a wet cooling tower.

In addition to the primary coolant loops, three core auxiliary cooling system (CACS) loops are provided. Each consists of a gas/water heat exchanger with auxiliary electric-motor-driven circulators located in cavities in the PCRV wall. Should the main loops not be available, coolant gas would be circulated from the reactor core through the heat exchangers, where heat would be transferred to an auxiliary cooling water system for rejection from cooling towers to the atmosphere.

The PCRV and auxiliary systems are housed inside a conventional secondary containment building that is a steel-lined, reinforced-concrete cylindrical structure. Typically, balance-of-plant systems and equipment are arranged and housed in separate buildings according to function and service. Spent fuel storage for 1-1/3 cores is provided, with railroad access for shipping and receiving provided onsite.

The components and systems described for the NSS system are shown in an isometric view of the PCRV in Fig. 3-3. Figure 3-4 shows a simplified schematic diagram of the primary and secondary coolant systems. The major parameters for the HTGR-SC are given in Table 3-1.

The NSS operates with a core outlet temperature of 686°C and a primary coolant pressure of 7.24 MPa. The NSS comprises the following five major systems and several support systems:

1. The PCRV system.
2. The reactor internals system.

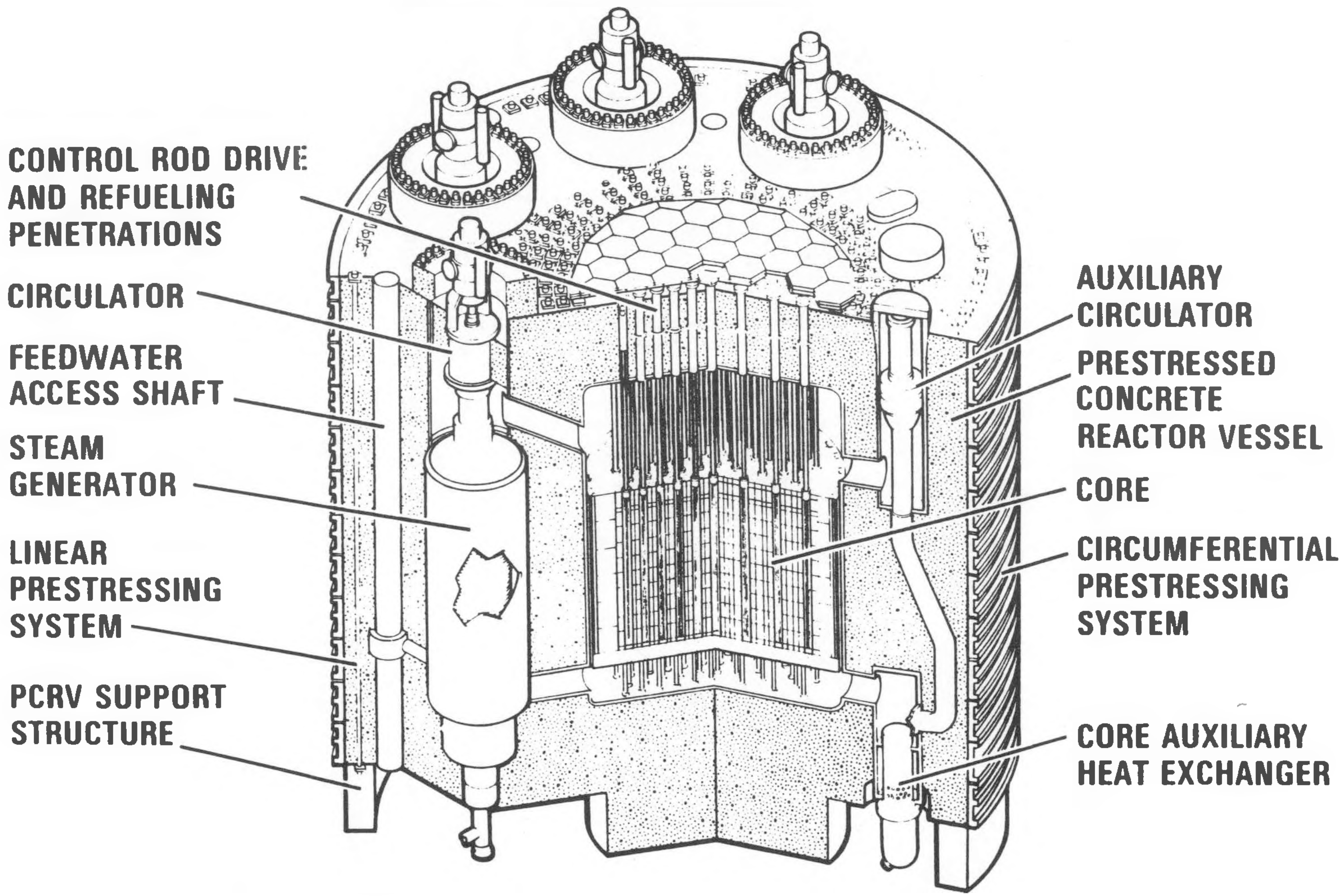

Fig. 3-3. HTGR nuclear steam supply system

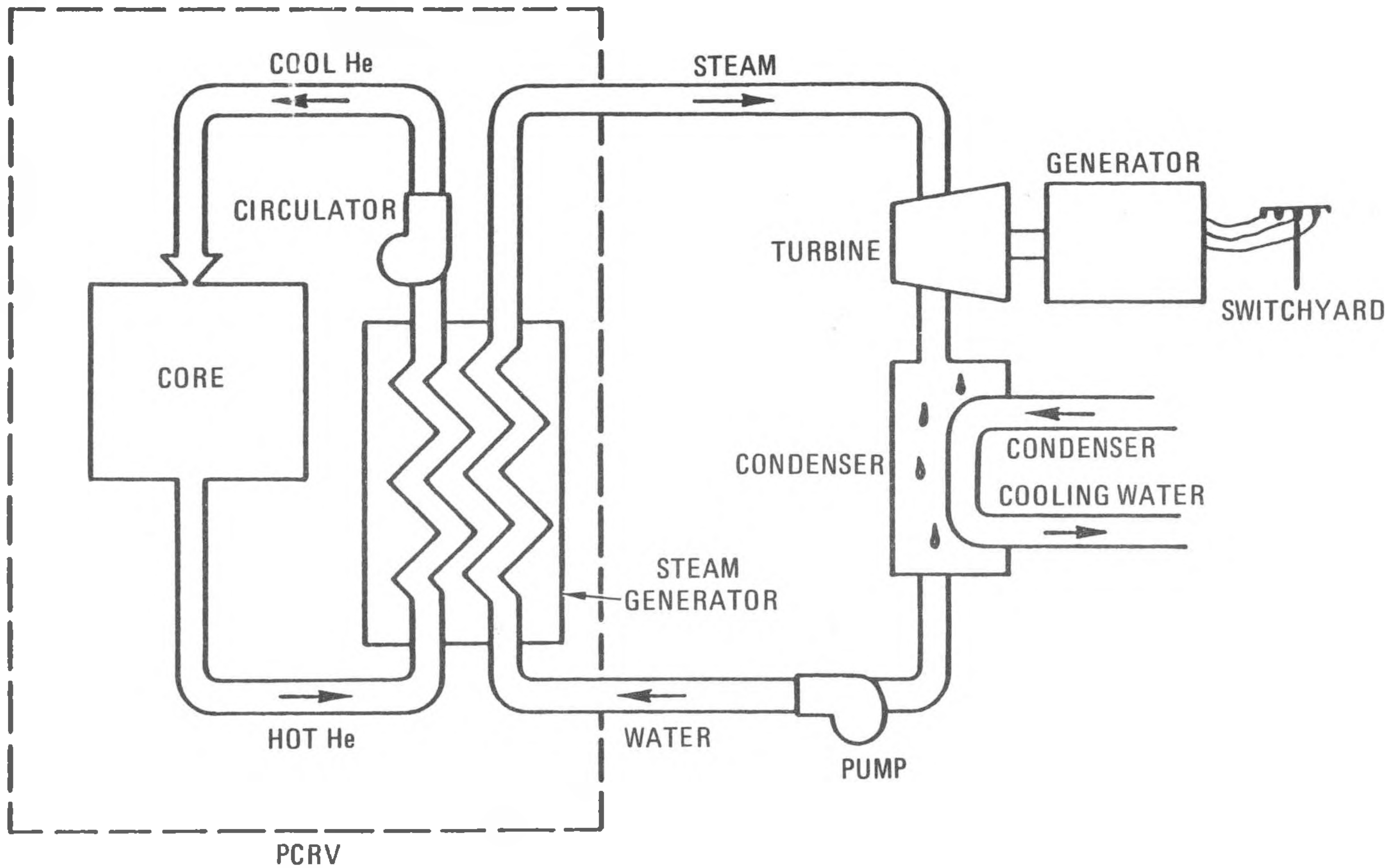

Fig. 3-4. HTGR-SC/C primary and secondary coolant systems

TABLE 3-1

HTGR-SC MAJOR PARAMETERS

Nominal Reactor Power	2240 MW(t)
Number of Primary Coolant Loops	4
Nominal Electric Output (net)	858 MW(e)
Nominal Net Station Efficiency	38.3%
Plant Layout	Single unit
Availability Factor	90%
Fuel Cycle	LEU[a]/Th stow-away
Helium Pressure	7.24 MPa
Core Outlet Temperature	686°C
Core Power Density	7.15 MW/m^3
Fuel Lifetime	4 yr
Refueling Cycle Times	1 yr
Total Steam Flow	925 kg/s
Steam Outlet Conditions	17.3 MPa/541°C

[a] Low-enriched uranium.

3. The reactor core system.

4. The primary coolant system.

5. The CACS.

PCRV System. This system comprises the PCRV structure; the cavity liners, penetrations, and closures; the thermal barrier; and the pressure relief system. The overall function of the PCRV system is to provide the primary coolant pressure boundary, to house the components of the NSS, and to provide a biological shield around the reactor.

The PCRV is a multicavity structure of prestressed concrete characterized by a central core cavity and peripheral cavities that house the primary cooling system components and the CACS components. For the 2240-MW(t) plant, the PCRV is 31 m in diameter and 29 m high. The high-strength concrete is prestressed circumferentially by wound strand cables and vertically by linear strand tendons. This prestressing is designed to induce sufficient precompression in the concrete to resist the primary and secondary loads during the 40-yr life of the plant.

The steel liners and the closures at the penetrations form the continuous, gas-tight boundary of the PCRV. The liner and penetration anchors transmit loads from the internal equipment supports to the concrete structure. The liners are cooled by circulating water in tubes attached to the liners at their interfaces with the concrete. Thus, the heat that penetrates the thermal barrier, as well as that generated by ionizing radiation within the steel and concrete, is removed.

The thermal barrier minimizes heat losses from the primary coolant and maintains the liner and concrete temperatures within acceptable limits. Different types of thermal barrier are used in the various cavities and zones within the PCRV, depending primarily upon the local gas temperatures. Typically, the thermal barrier consists of layers of fibrous insulation material held against the liner by metal cover plates. These plates are anchored to the liner via attachment fixtures, which are designed to minimize the thermal conduction to the liners.

The pressure relief system limits the pressure of the primary coolant in the PCRV to a specified safe value and also limits the rate of pressure relief flow from the PCRV.

Reactor Internals System. This system comprises five major components: the core support floor structure, the core lateral restraint, the permanent side reflector, the core peripheral seal, and the upper plenum structures of the in-vessel refueling system.

The core support floor consists of graphite core support blocks resting on graphite posts. These posts, in turn, are supported on graphite seats atop ceramic bases on the floor of the lower plenum. The major function of the core support floor is to provide vertical support for the reactor core. Each core support block contains gas flow passages and a plenum in which the primary coolant from several core columns is mixed before passing out into the lower core cavity plenum.

The core lateral restraint comprises metal support assemblies located in a regular array between the permanent side reflector and the PCRV liner. The primary function of this component is to provide lateral support for the reactor core, the support floor structure, and the reflectors. It also acts as a neutron side shield.

The permanent side reflector consists of columns of stacked graphite blocks that form a border around the hexagonal reflector columns of the reactor core. The primary function of this component is to reflect neutrons back into the core. It also attenuates the flux of neutrons to the surrounding components of the PCRV liner and structure and couples the core to the core lateral restraint to maintain the integrity of the core array.

The core peripheral seal is formed by graphite prisms of triangular cross section that fit in the annular space between the core support floor and the thermal barrier. A sloping shelf in the outer face of each peripheral core support floor block provides the inner seat for the seal. The outer seat is provided by a metal structure supported by cantilever beams attached to the liner and enclosed in the thermal barrier. The primary function of the seal is to restrict the flow of primary coolant that would otherwise bypass the reactor core by flowing between the permanent side reflector and the liner.

The upper plenum structures of the in-vessel refueling system are steel structures, supported above the top of the core from extensions of the refueling penetrations, which carry the refueling conveyor mechanisms within the upper part of the core cavity. During refueling, spent fuel blocks are raised to the level of this structure, placed on the conveyors, and transported laterally to the elevator in the PCRV side wall.

Reactor Core System. The function of this system is to generate nuclear heat for the HTGR plant. The system comprises the fuel elements, the reflector elements, the top layer/plenum elements, and the start-up neutron sources.

The fuel elements are graphite blocks of hexagonal cross section, with only minor differences from those in Fort St. Vrain, which are as shown in Fig. 3-1. The graphite has the dual function of containing the fuel and moderating the neutrons in the core. Coolant holes drilled axially through the block allow the passage of helium to remove the heat of fission. Other axial passages contain the fuel particles embedded in graphite rods. Some fuel elements also have large axial passages to allow the insertion of control rods and reserve shutdown devices.

The fuel particles consist of fissile and fertile kernels. The fissile kernels are uranium oxy-carbide surrounded by a buffer layer of low-density pyrolytic carbon, a layer of silicon carbide to help contain fission products, and an outer layer of high-density pyrolytic carbon that adds strength to the coating. This coating system is called TRISO coating. The fertile kernels are thorium oxide surrounded by a coating similar to that on the fissile kernel.

The reflector elements are graphite blocks similar in size and shape to the fuel elements. They all have coolant holes and some of them have holes for control rods and reserve shutdown devices, but they do not contain fuel.

The fuel elements and the top and bottom reflector elements are arranged in columns resting upon the core support blocks, as shown in Fig. 3-5. Each support block in the central region of the active core holds a central element that has control rod passages and six fuel elements without control rod passages. This cluster is called a refueling region. Each refueling region has its own flow control valve, which is used for adjustment of the coolant outlet temperature. All the elements in a single refueling region are loaded and unloaded at the same time.

The fueled regions are surrounded radially by two rows of replaceable reflector elements and then by permanent side reflector columns. Some of the refueling regions have fewer than seven columns in order to fit the core into the circular plan.

The top layer/plenum elements include steel components that provide plena for distributing the coolant flow from the control valves to the individual columns.

The start-up neutron sources consist of Cf-252 in a suitable container. These are inserted in the fuel elements to provide a source of neutrons of

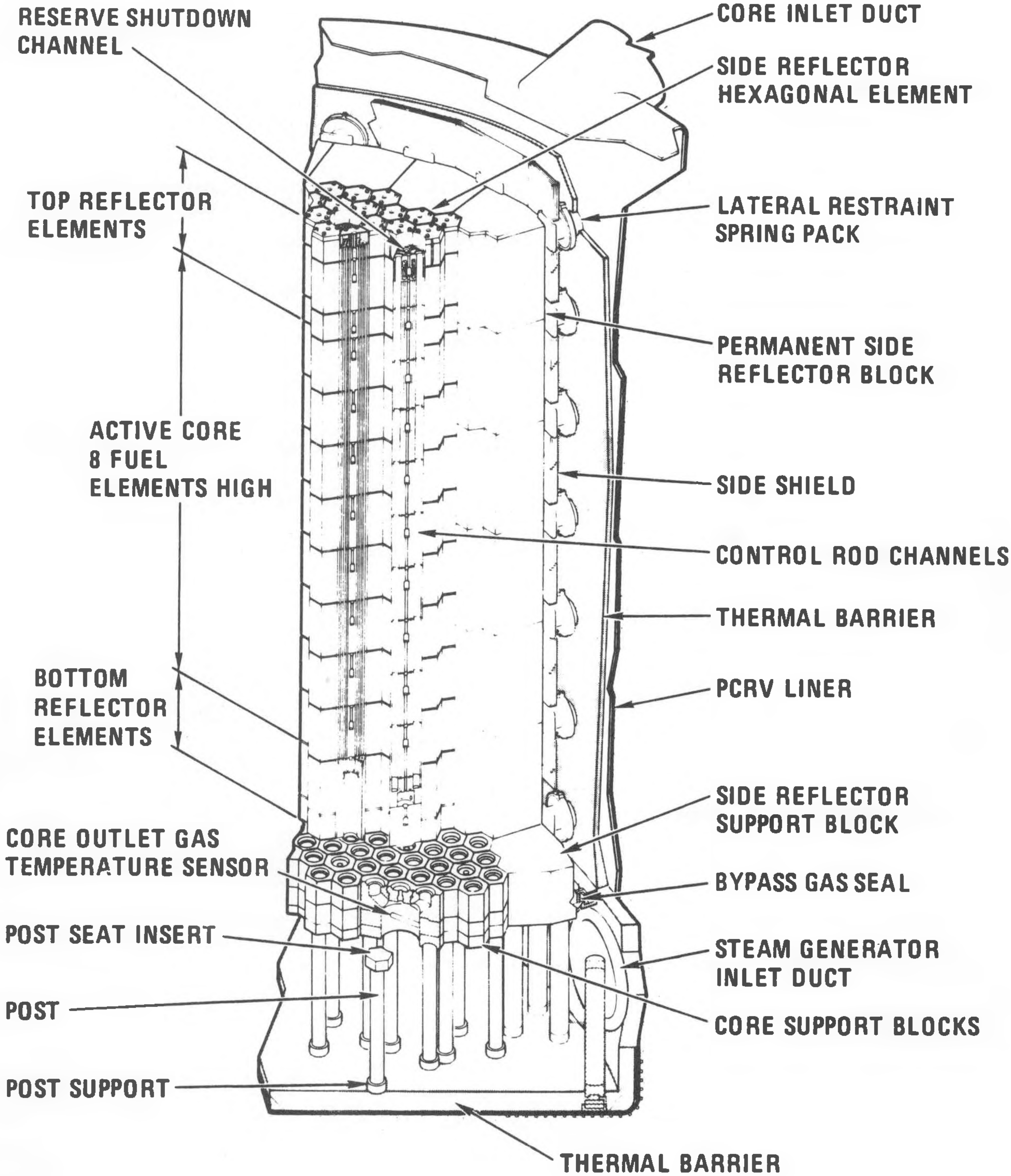

Fig. 3-5. HTGR core and support arrangement

sufficient strength to ensure a safe, controlled approach to criticality during reactor start-up.

The fuel cycle is based on a graded loading scheme in which a certain fraction of the core, known as a "fuel segment," is refueled on an annual basis. For this design a 4-yr cycle has been chosen in which four approximately equal segments are refueled, one per year. The refueling scheme is chosen so that the regions in each segment are symmetrically distributed throughout the core.

Table 3-2 gives the major reactor core system design parameters.

Primary Coolant System. The function of this system is to remove heat from the reactor core during normal plant operation and to transfer that heat to the secondary coolant, which is water and steam. This system comprises parallel forced circulation loops, each loop containing one steam generator, one circulator with its drive motor, and one loop shutoff valve. These components, except for the drive motors, are located in PCRV cavities peripheral to the reactor core cavity (see Fig. 3-3.) These peripheral cavities are connected to the core cavity by upper and lower cross ducts.

The steam generators are of the once-through type, consisting of a helical coil in the economizer-evaporator-superheater (EES) region made of 462 2-1/4 Cr-Mo tubes and a straight-tube superheater (STSH) made of 462 Incoloy 800 tubes located in the center of the helical coil (see Fig. 3-6).

TABLE 3-2

HTGR-SC BASIC CORE PARAMETERS

Nominal Core Power	2240 MW(t)
Nominal Core Power Density	7.15 MW/m^3
Number of Fuel Blocks per Column	8
Number of Fuel Columns	439
Number of Control Rod Pairs	67
Number of Power Rods	61
Number of Reserve Shutdown Passages	55
Core Volume	312 m^3
Fuel Cycle	
Initial Fuel Cycle	LEU/Th
Refueling Cycle [(a)]	4 yr
Fissile Material/Particle Coating	UCO/TRISO
Fertile Material/Particle Coating	ThO_2/TRISO
Fuel Enrichment	19.9% U-235

[(a)] 25% reload anually.

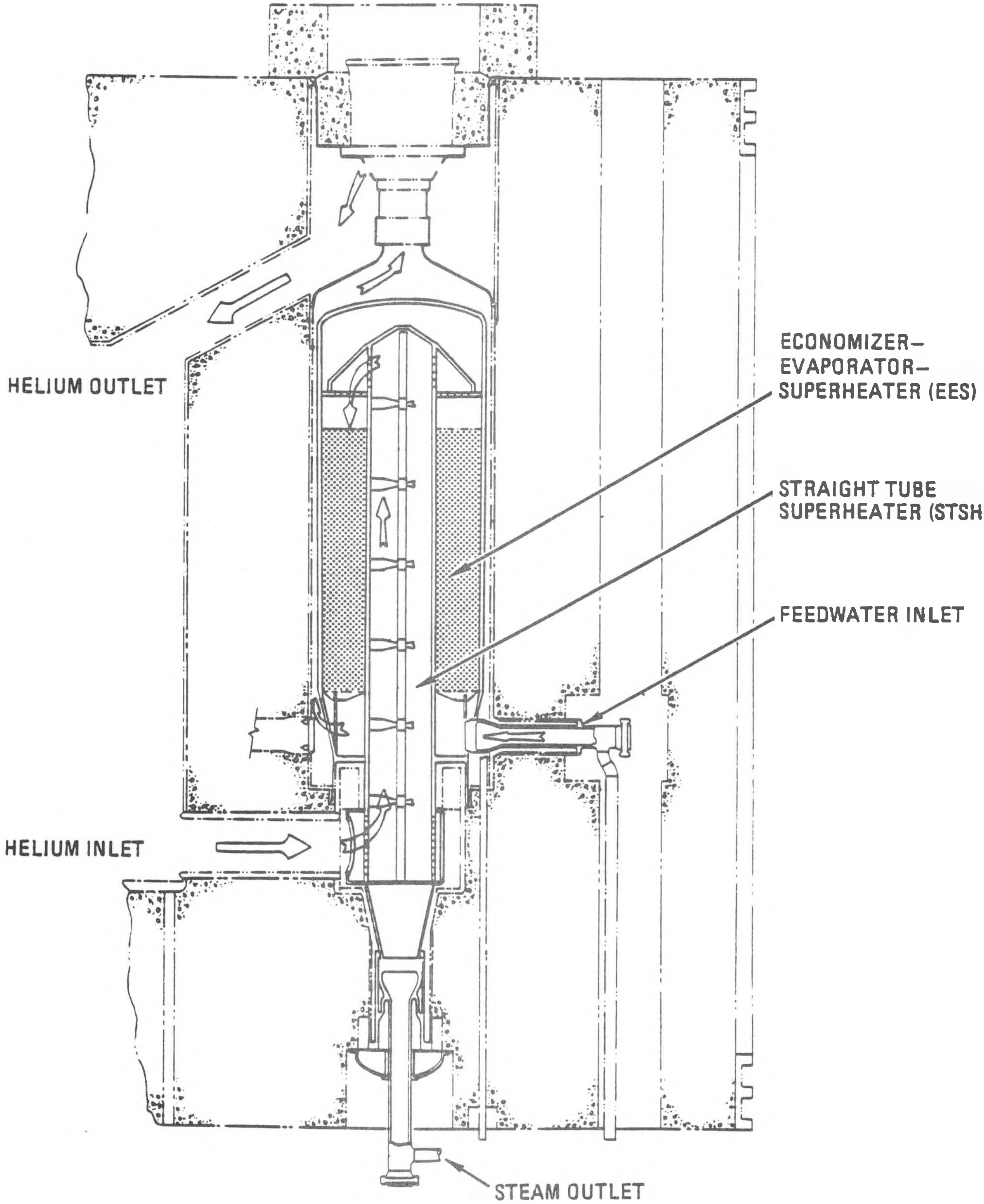

Fig. 3-6. HTGR steam generator cross section

The module is supported by a skirt (thermal sleeve) that extends upward from the liner at an elevation just above the lower cross duct.

Feedwater is supplied to each module through a side penetration. The superheated steam leaves the module at the bottom of the steam generator cavities. The hot helium emerging from a lower cross duct is directed upward through the STSH. It is then turned and directed downward through the EES bundle. After emerging from the EES, the helium turns upward and flows in the annular region between the steam generator module and the cavity thermal barrier. This helium then flows into the entrance of the circulator.

Each main circulator consists of a single-stage centrifugal compressor, a synchronous electric motor drive, and a loop shutoff valve (flapper valve) (see Fig. 3-7). The motor is mounted on top of the PCRV outside of the pressure boundary. The shutoff valve restricts the reverse flow of coolant through its primary loop in case the circulator should fail to provide sufficient pressure while the other loops, or the CACS loops, are operating.

Core Auxiliary Cooling System. The function of the CACS is to provide an independent means of cooling the reactor core if the primary cooling system should become nonoperable. It consists of three parallel independent cooling loops, each loop comprising a heat exchanger, a circulator with its drive motor, a loop shutoff valve, and the motor controls. These components, with the exception of the motor and controls, are located in a PCRV cavity peripheral to the central core cavity.

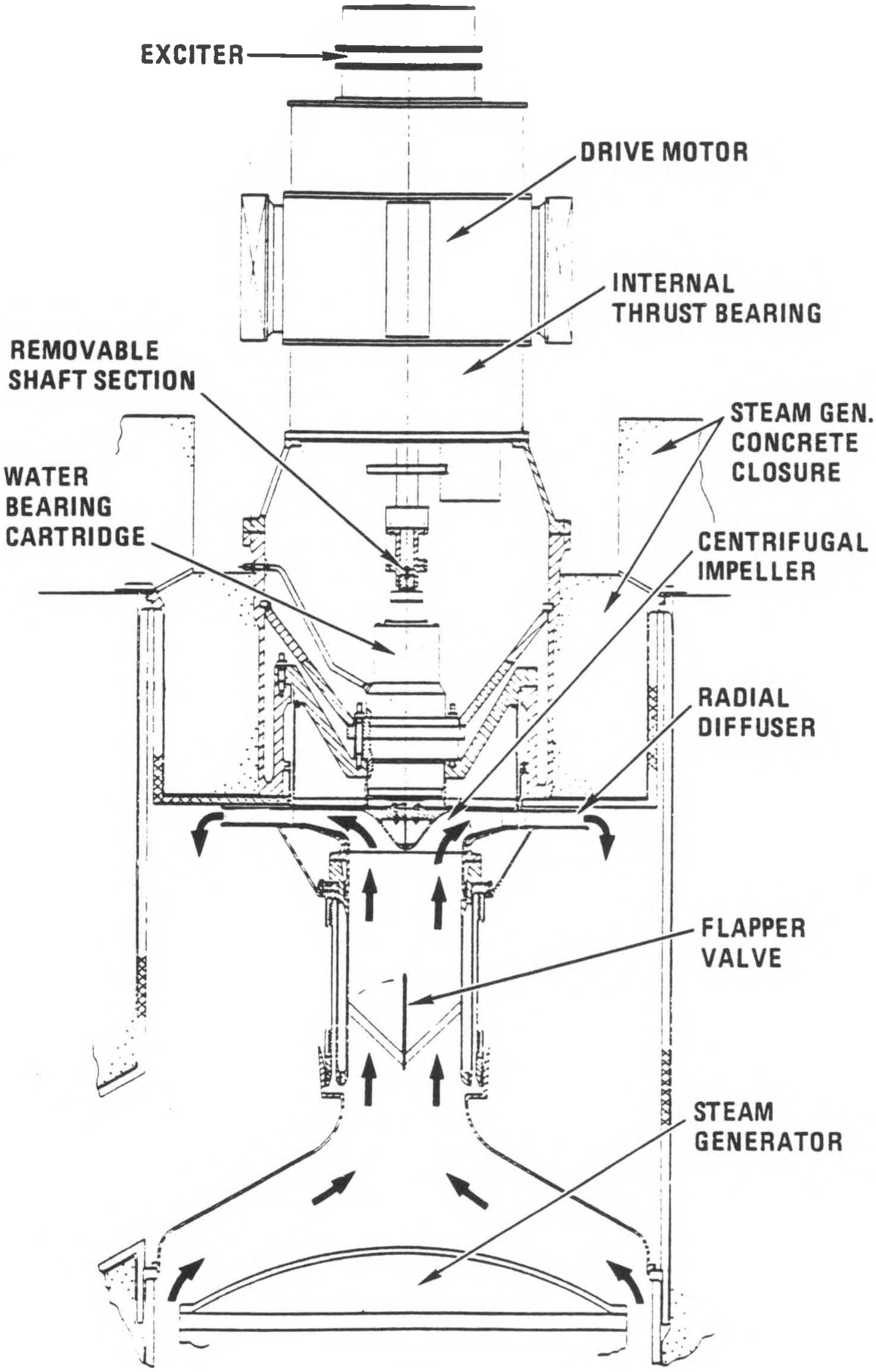

Fig. 3-7. HTGR main circulator installation

The CACS has the capability of maintaining the temperatures of all components in the PCRV within safe limits with the system either pressurized or depressurized. This is accomplished by forced circulation of the primary coolant through the auxiliary loops and through the reactor core. The heat is transferred through the core auxiliary heat exchanger (CAHE) to the core auxiliary cooling water system (CACWS). This water is cooled by air-blast heat exchangers, and thus the heat is ultimately rejected to the atmosphere.

Each CACS loop is capable of removing 100% of the residual and decay heat from the core following a reactor trip from 102% normal rated power, provided that the helium in the PCRV is pressurized. Two out of the three loops are required for heat removal if the PCRV is depressurized.

Balance of Plant (BOP)

The selected site arrangement for the 900-MW(e) HTGR-SC is shown in Fig. 3-8. The arrangement achieves a high degree of close-coupling between major structures while allowing for minor increases in the size of each structure. The reactor service building is located directly adjacent to the reactor containment building to minimize travel distance for refueling equipment and the length of piping runs. The control, auxiliary, and diesel generator building is located near both the reactor containment building and the reactor service building to minimize cable runs. The turbine building is located on the side of the reactor containment building opposite to the reactor service building. This orientation precludes any interaction

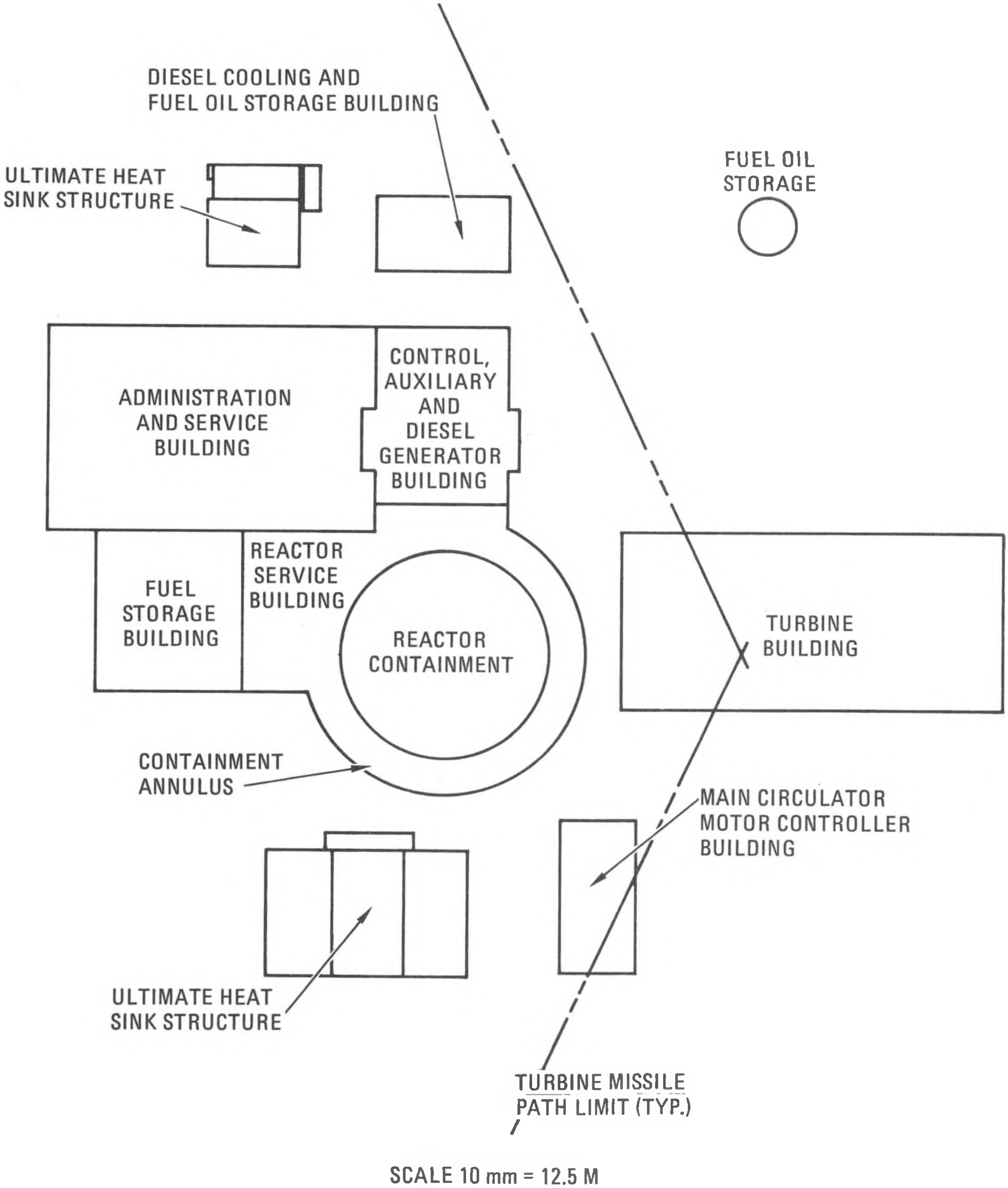

Fig. 3-8. HTGR-SC/C site arrangement

between Seismic-Category-I structures and positioned turbine missiles, as illustrated in Fig. 3-8.

The reactor containment building incorporates the containment annulus building, the containment penetration building, and the reactor service building on a common mat. The containment annulus building contains all the major safeguards equipment for the reactor plant cooling water system, the CACWS, and the auxiliary circulator motor cooling water system, with the exception of the CACWS air-blast heat exchangers. The air-blast heat exchangers and the nuclear service water system equipment are located in the ultimate heat sink structures. The containment annulus building also houses related heating, ventilating, and air conditioning (HVAC) equipment and the piping penetrations for the safeguards cooling system as well as the secondary helium system. The containment penetration building houses some equipment for the radioactive waste management system and includes two separate cable penetration areas, which link the control, auxiliary, and diesel generator building and the reactor containment building.

The major systems comprising the balance of reactor plant include the safeguards cooling system, the reactor plant cooling water system, and the reactor plant instrumentation and control system. These systems are discussed below.

The safeguards cooling system consists of the CACS, which is part of the reactor cooling system, the CACWS, and the auxiliary circulator motor cooling water system.

The reactor plant cooling water system has an essential subsystem consisting of two 100% redundant trains that provide cooling water to the PCRV cooling coils, the moisture-monitoring equipment, and the auxiliary circulator motor cooling water system. The reactor plant cooling water system also has a single-train, non-essential subsystem that is not safety related and that provides cooling water to non-safety-related equipment and to a separate non-essential cooling coil in each auxiliary circulator motor to remove parasitic heat losses when the circulator is not operating.

The reactor plant instrumentation and control system is designed to ensure that the reactor can be safely and efficiently operated and that in the event of an abnormal or accident condition, it can be shut down and maintained in a safe-shutdown condition. The system consists of automatic and manually initiated protection systems for safety under accident conditions; safety-related display systems required during normal, upset, emergency, and faulted conditions; a computer-based data acquisition and display system; and regulating systems used for normal operation of the unit.

The instruments and controls are located in the main control room, which provides remote operation of the plant. In the event that access to the main control room is lost, equipment is provided outside the main control room to shut the reactor down and maintain it in a safe-shutdown condition.

The energy conversion system consists of a tandem-compound, six-flow turbine with 790-mm last-stage blading without moisture separation or reheat and a two-pole, hydrogen-cooled generator and rotating exciter with a synchronous speed of 3600 rpm. The turbine-generator is calculated to deliver 913,200 kW(e) gross output with throttle steam conditions of 16.8 MPa/538°C/925 kg/s flow and 8.5 kPa exhaust pressure while operating in a regenerative feedwater heating cycle having steam-driven boiler feed pumps and six stages of feedwater heating. Turbine-generator accessories include the lubricating oil supply and purification system, hydraulic oil system, stator cooling water system, gland sealing system, gas storage system, and associated instrument and control systems.

3.1.1.4. HTGR-GT. During the 1970s, design studies, assessments, and evaluations were carried out on an advanced, direct-cycle HTGR option. The motivation for this effort included (1) further exploitation of increased reactor outlet temperature (above the current 700°C value) to achieve high efficiency, (2) opportunity to take full advantage of the favorable Brayton cycle characteristics related to economic dry-cooling capability, (3) potential of very high efficiency when operated in a combined-cycle mode, and (4) projected plant simplification with attendant economic benefits.

As with all advanced technology endeavors, several iterations were required to establish a reference 800-MW(e) plant concept. As the design studies progressed, it became increasingly apparent that introducing a high-energy helium turbomachine into the reactor primary system resulted in complexity and problems compared with the simpler steam-cycle plant nuclear heat source. The various problems manifested themselves not only in technical and economic areas, but also with regard to plant operability, safety, and licensing.

Perhaps the salient feature of the HTGR-GT concept that attracted utility interest in the early 1970s, both in the United States and Europe, was the perceived siting flexibility associated with dry cooling. For sites with limited water availability, however, the concept of wet/dry cooling was introduced in the late 1970s, and in this regard, the Brayton cycle did not benefit nearly as much as other nuclear systems (based on the steam Rankine cycle) from performance and economic standpoints. As had been found earlier in Europe for fossil-fired systems, the key to acceptance of the closed-cycle gas turbine (CCGT) is the utilization of the high-grade waste heat. Essentially, foreseen institutional problems prevented economic credit from being taken for the waste heat from the HTGR-GT plant.

The aforementioned studies led to the following conclusions in 1980: (1) extensive development was necessary to establish a technically viable HTGR-GT plant to satisfy demanding safety and licensing criteria, (2) new cycle studies and design innovation were necessary to identify features for improved economics, and (3) the study findings were not regarded as being consistent with the goal of having a commercial-size HTGR plant operational in the mid-1990s. Accordingly, the HTGR-SC was designated as the lead plant and the HTGR-GT was classed as a long-term, advanced technology, follow-on HTGR plant option (Ref. 3-7).

The Brayton cycle can take full advantage of future higher reactor outlet temperatures (above the current 700°C for the steam cycle), and in this regard, deployment of the HTGR-GT can be allied with advanced HTGRs being studied for high-temperature process heat applications. With the necessary development, a dry-cooled gas turbine plant with a reactor outlet temperature of approximately 950°C, operating in a combined cycle mode (with an efficiency of over 50%) or cogeneration mode (power plus process steam production), could be realized in the early decades of the 21st century.

Overall Gas Turbine Plant Features

In late 1979, efforts were directed toward designing a plant based on a 2000-MW(t) reactor rating. The design that evolved (Ref. 3-8) is briefly outlined below.

The reactor turbine system consists of primary coolant loops, the reactor core, and auxiliary cooling loops. The reference plant concept embodies two primary coolant loops rated at 400 MW(e) each and consists of a turbomachine, a recuperator, a precooler, and control valves. As shown on the reactor turbine system arrangement in Fig. 3-9, all of the primary system equipment is installed in the PCRV to give an integrated plant configuration.

The two turbomachines are installed in horizontal cavities toward the base of the PCRV. The helium turbomachine is rated at 400 MW(e) and consists of a simple and rugged arrangement with a single shaft and direct

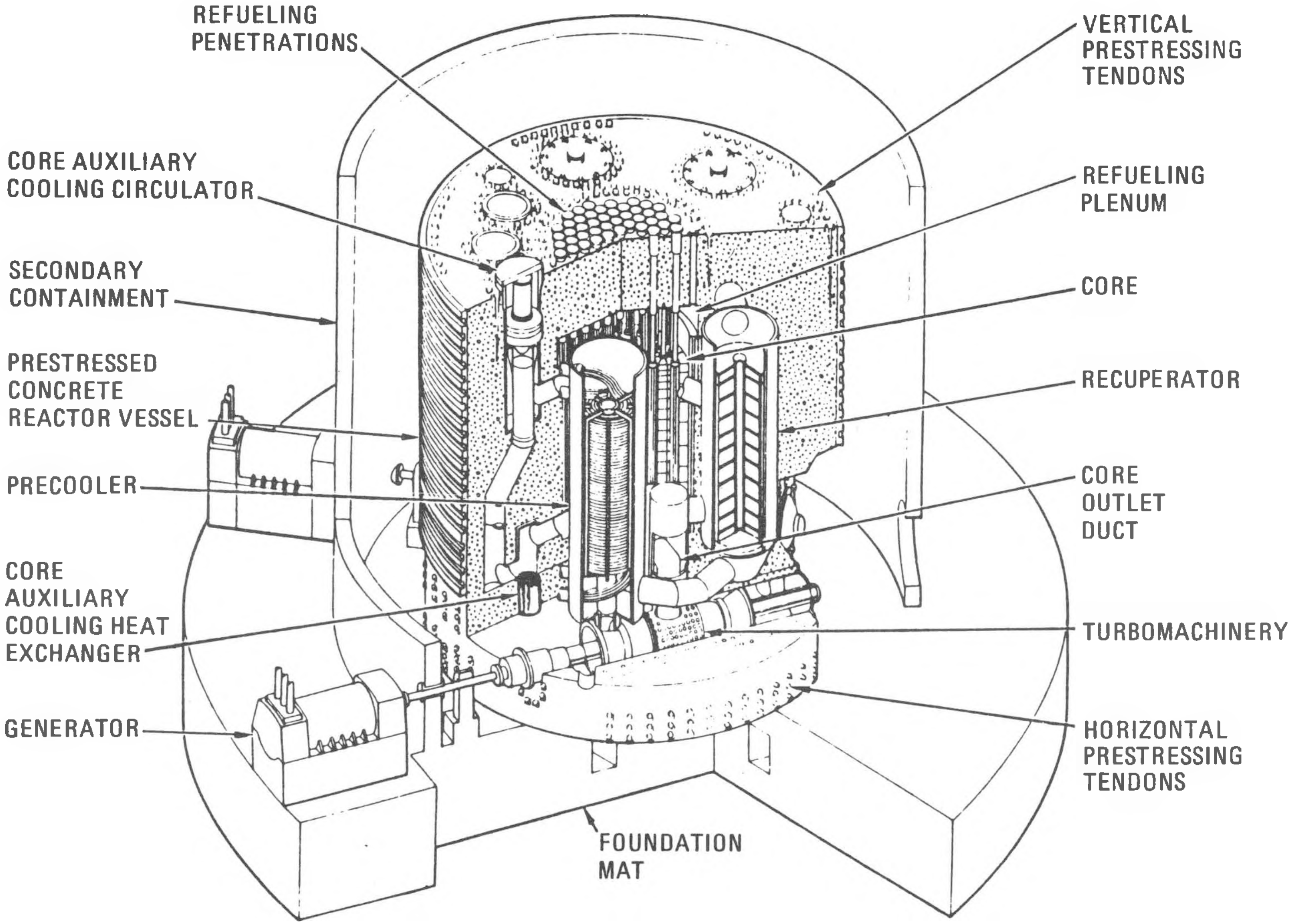

Fig. 3-9. Integrated 2000-MW(t) HTGR-GT reference plant reactor turbine system arrangement based on two primary coolant loops

drive to the generator. The turbomachine drive to the generator is from the compressor end of the turbomachine, and the thrust bearing is located external to the PCRV to facilitate inspection and maintenance.

The heat exchangers are installed in vertical cavities in the PCRV, as shown in Fig. 3-9. The recuperator (which contributes significantly to the high cycle efficiency) is a helium-to-helium heat exchanger of high effectiveness. The reference design is based on a straight-tube, axial counterflow configuration and has provision for in-service inspection and plugging of individual tubes. The precooler (which is used to remove reject heat from the cycle) is a helium-to-water heat exchanger of very high thermal effectiveness. The reference design has a helical bundle of cross-counterflow geometry. Both of these heat exchangers are regarded as state-of-the-art technology, and with modest metal temperatures and pressure differentials (compared with modern steam generators), they utilize code-approved lower-grade alloys of reduced cost.

The CACS consists of three separate and independent cooling loops composed of auxiliary circulators, CAHEs, and ultimate heat sinks rejecting heat to the atmosphere. The system can remove the core residual and decay heat for cooldown following loss of helium circulation in the primary coolant loops with the reactor in a shutdown condition and either pressurized or depressurized.

The plant arrangement uses horizontal electric generators and turbomachines with access and removal through grade-level penetrations in the containment. The reference design has hydrogen-cooled generators located outside of the containment building and coupled to the turbomachine via a shaft penetration through the building. Table 3-3 summarizes the major features of the plant.

Gas Turbine Plant Performance

System optimization and design evaluations were performed during the conceptual design stage to identify (1) an optimized set of parameters for further design development, (2) sensitive areas where additional design definition or improvements would be effective, and (3) where margins can most effectively be applied to achieve the desired probability of performance success at minimum cost. Figure 3-10 shows the parameters selected from the optimization study with a reactor outlet temperature of 850°C. A maximum system pressure of 8.4 MPa was selected from the sensitivity studies. This pressure level, which gives high gas density and compact turbomachinery, heat exchangers, and ducts, is only a modest extension of PCRV structural design practice.

With a reactor outlet temperature of 850°C, a compressor pressure ratio of 2.6 was selected, which together with the recuperator effectiveness of 0.9 resulted in a plant efficiency of 40% for the nonintercooled cycle with dry cooling. Figure 3-11 shows an array that relates the important cycle parameters to plant efficiency. The plot shows projected operating regimes for various CCGT plants. It is clear from the figure that the HTGR-GT plant offers significant potential for high levels of efficiency.

TABLE 3-3

CHARACTERISTICS OF HTGR-GT

Reactor Outlet Temperature	850°C
Reactor Inlet Temperature	494°C
Reactor Core Thermal Rating	2000 MW(t)
Core Power Density	6.6 MW/m^3
Core Type	HTGR prismatic fuel elements
Fuel Type	LEU/Th
Fuel Lifetime	3 yr
Primary System Fluid	Helium
Maximum Primary System Pressure	8.4 MPa
PCRV Diameter	32.6 m
PCRV Height	35.4 m

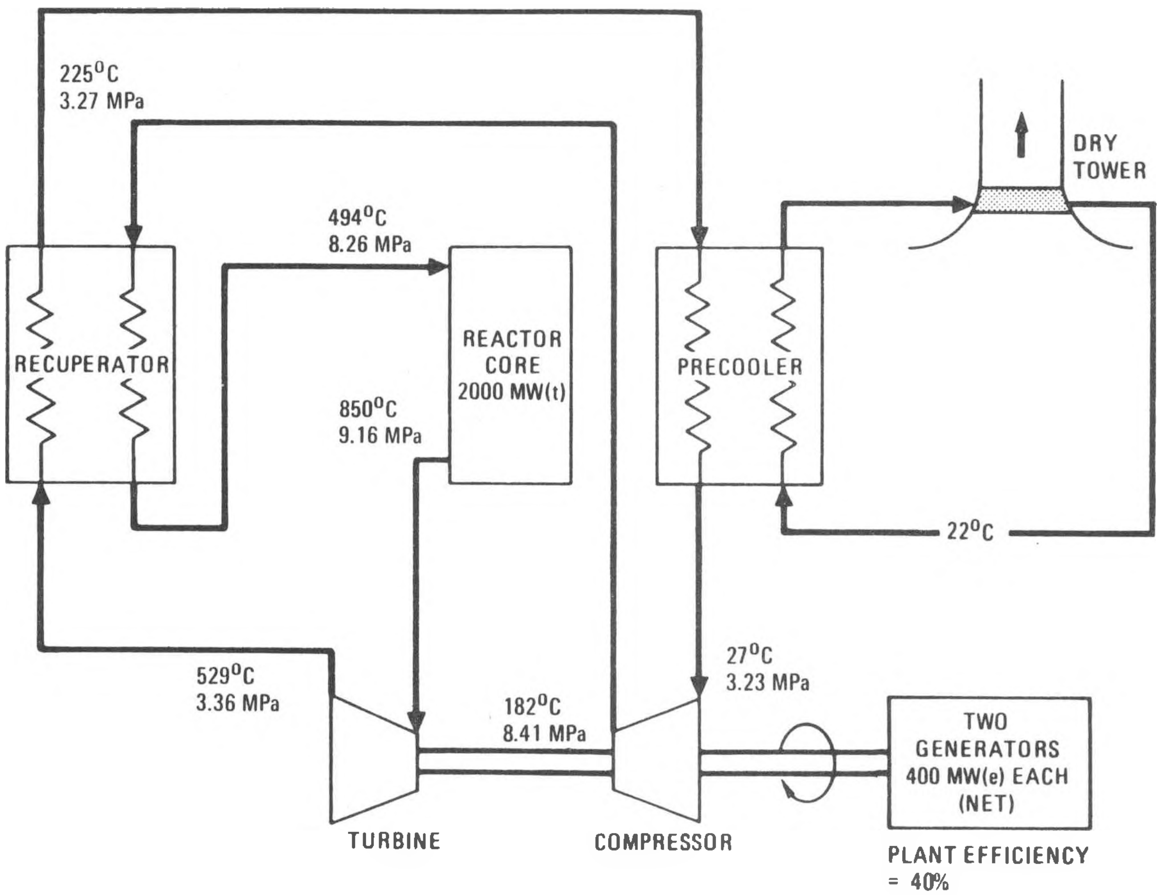

Fig. 3-10. Flow path diagram showing salient parameters for HTGR-GT reference plant

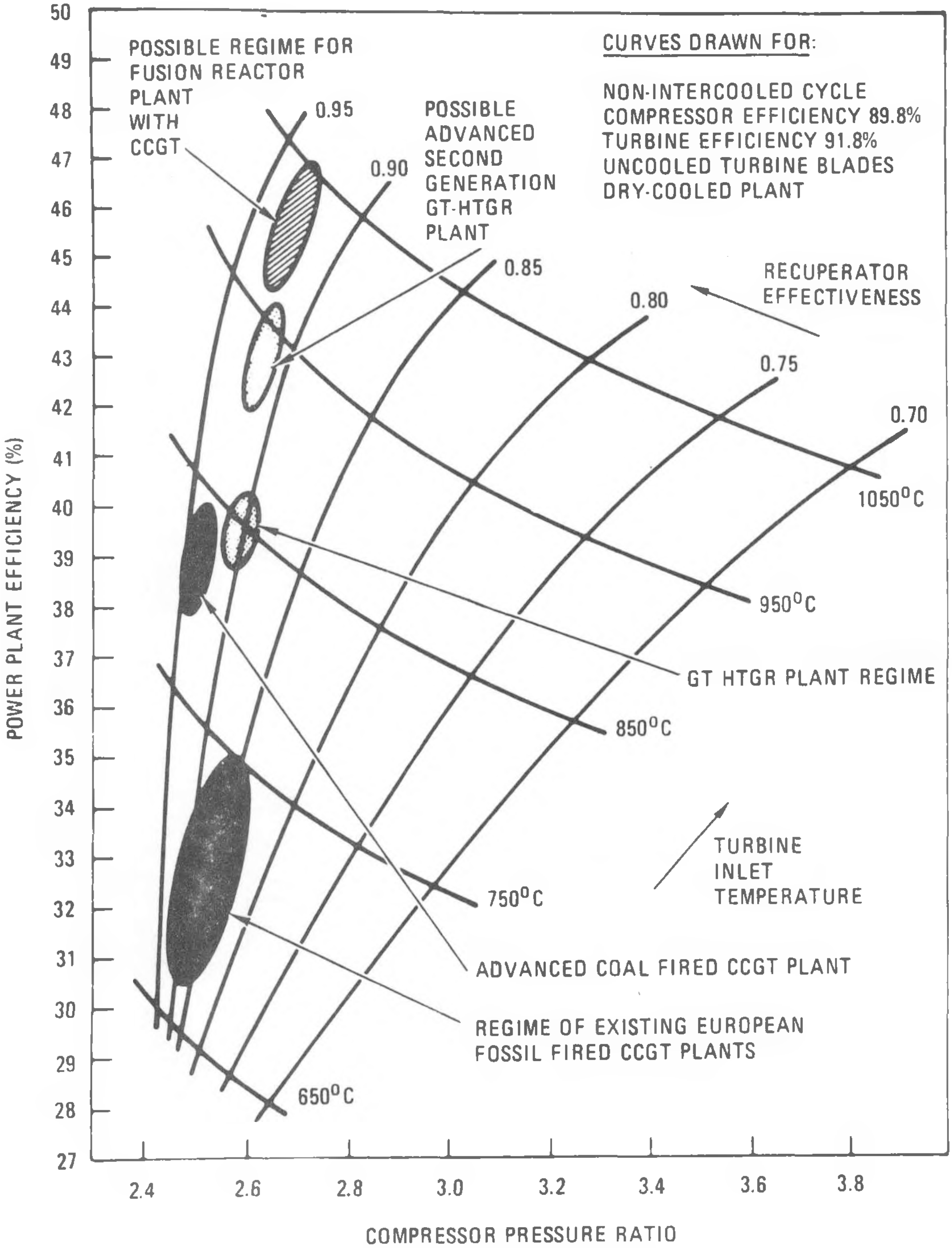

Fig. 3-11. Representative cycle parameter array for nonintercooled HTGR-GT plant

HTGR-GT Plant Technical Issues

In all new advanced technology endeavors, several iterations are usually necessary to resolve technical issues to satisfy all of the established criteria. In the case of the gas turbine plant, a formidable technology transfer from the demonstrated steam cycle was realized. However, the incorporation of the helium turbine prime-mover in the reactor primary system raised issues that required new thinking and the generation of new design criteria. The following technical issues are essentially those related to having the helium turbine in the reactor system and do not include generic HTGR issues:

1. Helium turbomachine integrity.
2. Turbomachine structural integrity features.
3. Reactor system pressure transients.
4. Lubrication oil ingress.
5. Seal design.
6. Primary system acoustic level.
7. Materials considerations.
8. Maintenance considerations.

Plant Economics

At the onset of gas turbine plant studies, economic projections were made that indicated significant cost savings over Rankine cycle systems were possible by combining (1) high conversion efficiency, (2) plant simplicity, and (3) economic dry cooling. Even with efforts to optimize cycle parameters and the adoption of alternative plant concepts, increasing design effort made it apparent that the earlier projection of economic benefits could not be realized for a dry-cooled plant with a reactor outlet temperature of 850°C. In addition to the higher-than-projected plant capital cost, the costs of design, development, and demonstration of a new system were concluded to be large relative to the limited incentives of the reference design.

The gas turbine plant was recognized to have potential beyond the reference plant concept. Specifically, designs were examined for higher temperatures and for using the high-grade reject heat. The aforementioned economic studies did not take credit for the high-grade reject heat. In the fossil-fired closed-cycle gas turbines operating in Europe, a significant portion of the owner/operator revenue comes from thermal energy production. Waste heat utilization is the key to HTGR-GT acceptance in the United States, and this unique capability needs to be stressed. Utilization of the high-grade reject heat for district heating, process steam production, or desalination must be part of a national program to conserve resources. Such a program would require a period of decades to be implemented. Establishing users and operations involves long-term planning and, in many cases, overcoming institutional barriers. The heat rejection characteristics of the HTGR-GT are well suited to a coming period of energy consciousness and eventual energy shortage.

Accompanying the aforementioned projected gains with an advanced HTGR-GT will be increased technical risk, which must be addressed through development or experience. In 1980, such a system was concluded to be more prudently evolved as a follow-on to a less demanding concept (the HTGR-SC) and in conjunction with or following development of a process heat HTGR plant. Accordingly, given the evolving economic and institutional framework that exists in the energy supply industries, the HTGR-GT was concluded to be a longer-term goal of the HTGR program with reduced priority in the near term relative to predecessor options.

3.1.2. Gas-Cooled Fast Breeder Reactor (GCFR)

3.1.2.1. Incentives for Deployment. The GCFR has the potential to achieve breeding ratios of over 1.5 and doubling times of less than 10 yr. It also shares the advantages of helium with HTGRs, namely: a single phase, transparent, chemically and neutronically inert coolant. In addition, its economics appear attractive, although the design has not been tested in a full licensing review or by construction of a plant. The breeder reactor can make a substantial contribution to the energy resources of the world and to world economic and political stability. The probability of success in making this essentially unlimited source of energy available would be enhanced if the development of the GCFR concept were to be continued. Since the total costs of developing the GCFR are relatively uncertain compared with costs for the more fully developed LMFBR, at the present time the entire U.S. Government program is devoted to the LMFBR.

The 350-MW(e) plant described briefly in Section 2 and in more detail below was intended to demonstrate the components and systems of a GCFR on a sufficient scale to permit the next plant to be a prototype of a commercial plant (Ref. 3-9). The rationale behind the decision to build a 350-MW(e) plant as the first unit was based on the following principle, which was established by General Atomic in conjunction with the electric utility sponsors of the program in 1969: The GCFR program would make maximum use of the helium technology being developed under the HTGR program, particularly the experience from the Fort St. Vrain plant and the fuel and physics technology of the LMFBR program (Ref. 3-10). The wedding of these two technologies would allow development of the GCFR at much lower cost than starting with a completely new design.

In 1976 the utility supporters, who represented over 30% of the electric generating capacity of the United States and several European utility companies as well, formed a non-profit corporation, Helium Breeder Associates, the purpose of which was to support and guide the development of the GCFR. They believed that combining the HTGR and LMFBR technologies with a development program specific to the GCFR would permit construction of a demonstration plant as the first step in the program without excessive risk.

3.1.2.2. Demonstration Plant Conceptual Designs. The design of the 350-MW(e) demonstration plant proceeded through a number of iterations in the period from 1969 through 1980 and resulted in the concept shown in Fig. 2-5 (Ref. 3-11). Typical plant parameters are given in Table 3-4.

TABLE 3-4

GCFR DEMONSTRATION PLANT PARAMETERS

	Initial Operation	Upgraded Operation
Maximum Cladding Temperature (°C)	700	750
Electric Power (nominal) (MW)	300	350
Reactor Pressure (MPa)	10.5	10.5
Reactor Outlet Temperature (°C)	496	524
Reactor Inlet Temperature (°C)	293	293
Circulator Power (MW)	10.7	10.7
Steam Throttle Temperature (°C)	466	482
Steam Throttle Pressure (MPa)	9.3	10.0

The entire primary coolant system is contained within the PCRV, which is a multicavity pressure vessel reinforced with steel rods and prestressed by a system of longitudinal tendons and circumferential wire wrappings. The reactor core is located in the central cavity; peripheral cavities surrounding the central cavity contain heat exchanger and helium circulation equipment. The peripheral cavities are interconnected with the central cavity by cross ducts. All PCRV interior surfaces are lined with a leak-tight steel liner that contains the primary coolant. This, in turn, is lined with thermal insulation to protect the PCRV from the high temperatures of the helium coolant. The limited quantity of heat that passes through the thermal insulation is removed by a cooling water system consisting of tubes welded to the outside surface of the liner.

The reactor coolant system consists of three main loops, each with an independent steam generator, a horizontally mounted electric-motor-driven circulator, and a gravity-closing isolation valve. Three core auxiliary cooling loops are also provided, each having a vertically mounted electrically driven circulator, a heat exchanger, and a gravity-opening isolation valve.

The general configuration of the reactor core is illustrated in Fig. 3-12. The active core region consists of 169 hexagonal assemblies; 150 of these are fuel assemblies, 15 are control assemblies, and 4 are secondary shutdown assemblies. The central core region is surrounded by a radial blanket region of fertile material. The radial blanket consists of 162 hexagonal blanket assemblies arranged in three concentric rows around

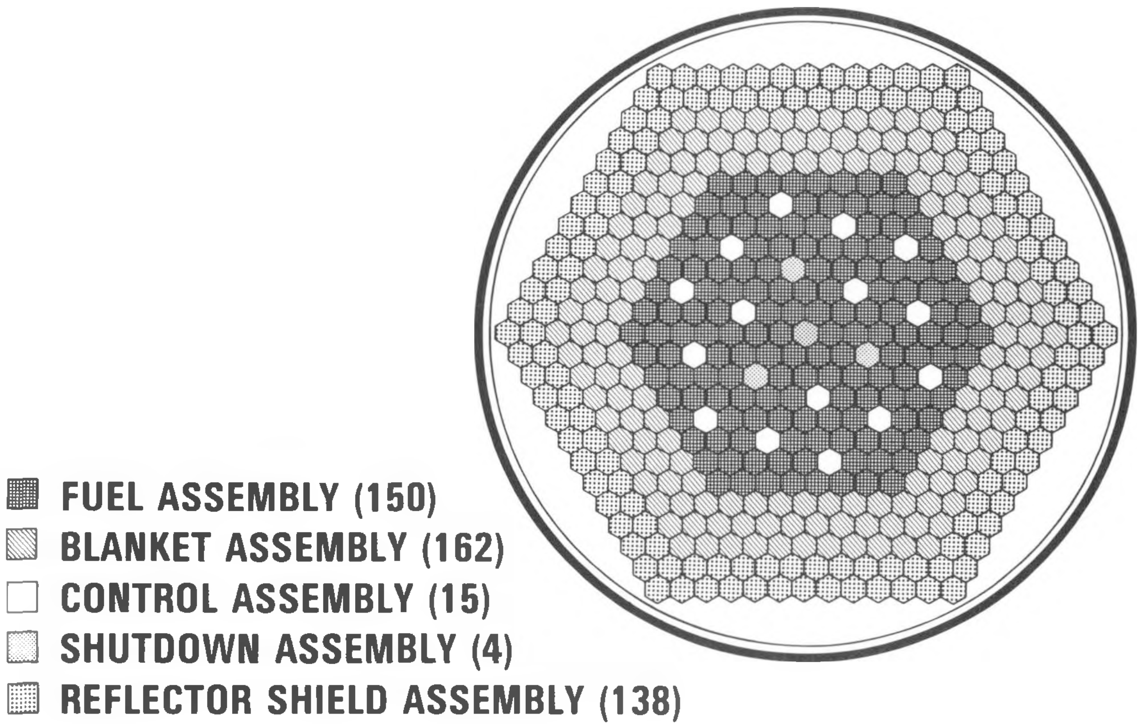

Fig. 3-12. GCFR core plan view

the active core. Fertile blanket material is also included in the fuel assemblies above and below the active core (upper and lower axial blankets). The radial blanket is surrounded by 138 hexagonal reflector/shield assemblies arranged in two concentric rows to protect the core restraint.

The control and shutdown rod assemblies are located in the centers of regions that consist of a central control (or shutdown) assembly surrounded by six fuel assemblies. Removal and replacement of assemblies during refueling are accomplished through the penetrations in the closure plug provided for the drive mechanisms. Each seven-assembly region is serviced by one of the penetrations. In the regions outboard of the control and shutdown assembly locations, additional penetrations are provided in the closure plug to service the blanket and reflector/shield assemblies.

The core assemblies are axially supported at their lower ends by the bottom-mounted grid support plate (Fig. 3-13). The assemblies are supported laterally by a core restraint mounted on the grid plate, which consists of a cylindrical support barrel and core formers surrounding the core. The core restraint is designed to provide lateral support at two elevations: (1) at the top of the active core and (2) at the top of the core assemblies. This core restraint system has been adopted from the technology developed in the LMFBR program and is designed to minimize reactivity insertion due to core distortions.

The GCFR fuel assembly design is similar to designs employed in LMFBR programs, particularly the fuel rod design. The fuel rod design employs the

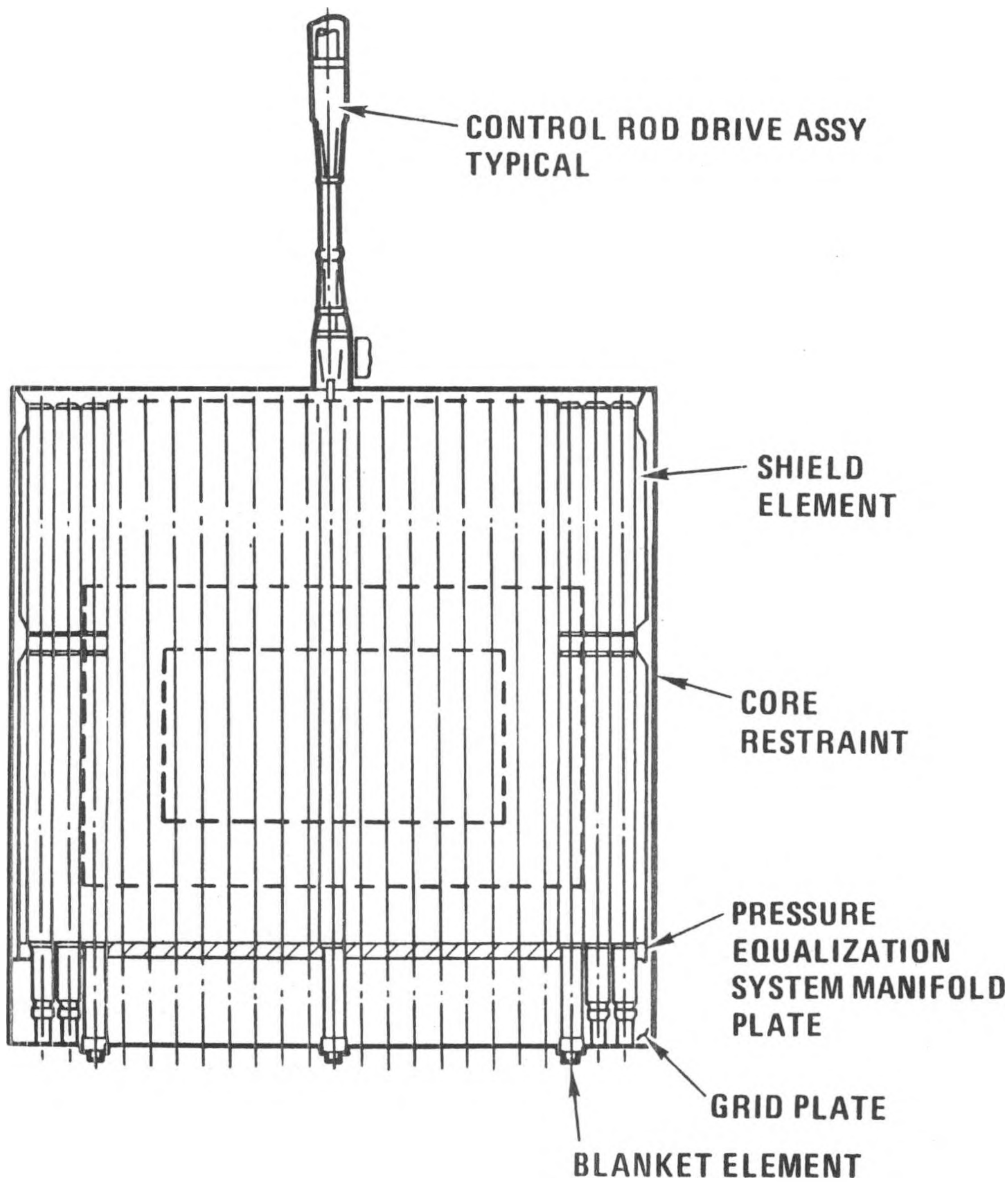

Fig. 3-13. GCFR core elevation view

same materials and has similar geometry and operating conditions as those being developed in the international LMFBR programs. The unique characteristics incorporated in the GCFR fuel assembly design include the following:

1. Roughened fuel rod cladding to enhance heat transfer to the helium coolant.

2. Pressure-equalized and vented fuel rods, which essentially eliminate any pressure-induced stresses on the cladding from either the primary coolant system pressure or from fission gases generated within the rods.

3. A large fuel rod pitch-to-diameter ratio relative to that commonly employed in the LMFBR designs.

The fuel assembly (see Fig. 3-14) is hexagonally shaped, 201 mm across flats, and 5000 mm long. A cylindrical inlet nozzle, 172 mm in diameter by approximately 675 mm long, is located at the bottom end between the cylindrical and hexagonal shapes. An exit end nozzle is provided on the top end of the assembly. The exit nozzle is used for handling the assembly and contains a fixed-area replaceable orifice. The orifice is designed to be changed at refueling outages.

Contained within the fuel assembly is a 2870-mm-long fuel bundle consisting of 265 fuel rods. The fuel rods are 8 mm in diameter and are located on an 11.5-mm triangular pitch. The rods are fastened to a grid

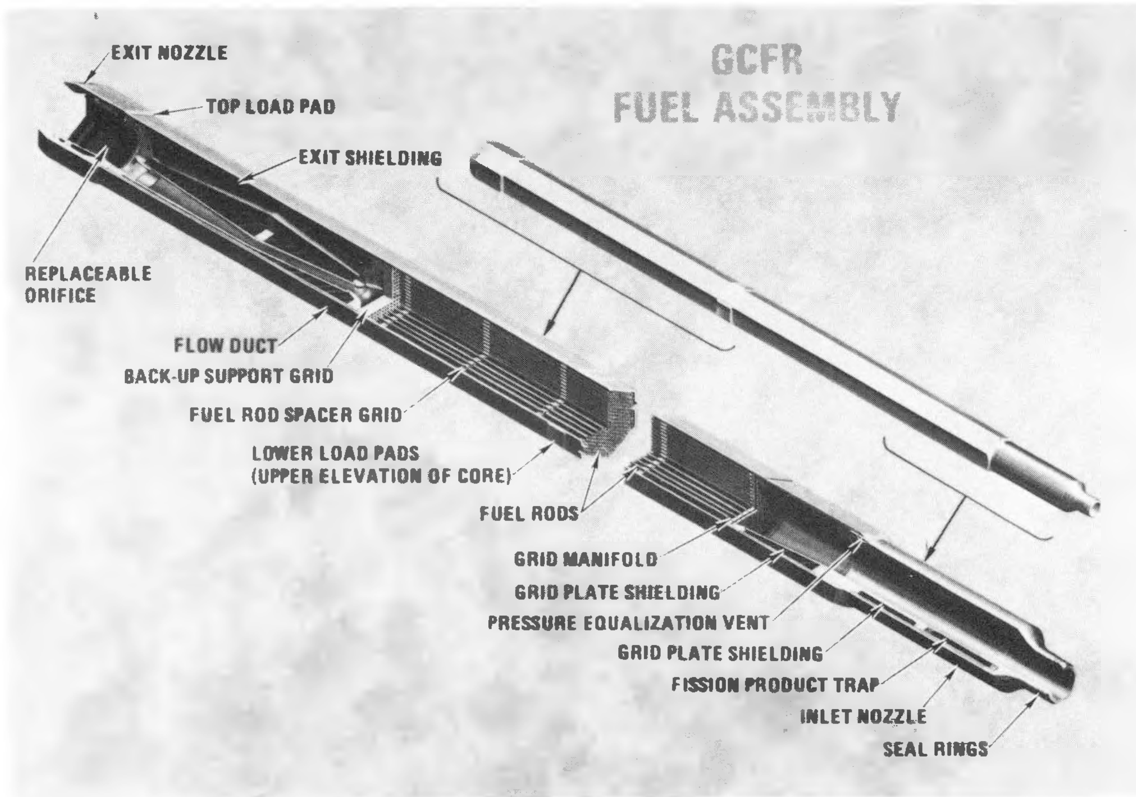

Fig. 3-14. GCFR fuel assembly

manifold at their lower ends and are laterally spaced by 12 spacer grids positioned at selected axial locations along the length of the rods. The spacer grids are retained in position by six hanger rods. The hanger rods are structural members containing no fuel material. A hexagonal flow duct is provided around the rod bundle for channeling the coolant flow through the rod bundle.

Each fuel rod (Fig. 3-15) consists of a cladding tube, a fuel stack, upper and lower axial blankets, a hold-down spring, a fission product trap, and end plugs. The cladding tube is roughened with trapezoidally shaped ribs over the fuel region to enhance heat transfer to the helium coolant. The fuel region consists of a 1200-mm-long stack of mixed oxide $(U,Pu)O_2$ cylindrical pellets with a fissile plutonium enrichment of approximately 20%. The upper and lower axial blankets are located above and below the fuel region. Each axial blanket consists of a 600-mm-long stack of depleted UO_2 cylindrical pellets.

The blanket assembly design is essentially the same as the fuel assembly, except that the blanket rod bundle consists of a smaller number of larger rods. Other than the rod bundle, the radial blanket assembly is designed to use the same major components as the fuel assembly. Use of either depleted UO_2 or ThO_2 as blanket material is foreseen.

Each of the three main helium circulators (Fig. 3-16) is driven by a 13.9-MW electric motor horizontally located outside the PCRV. Helium leaving the steam generator enters the circulator and is accelerated by a

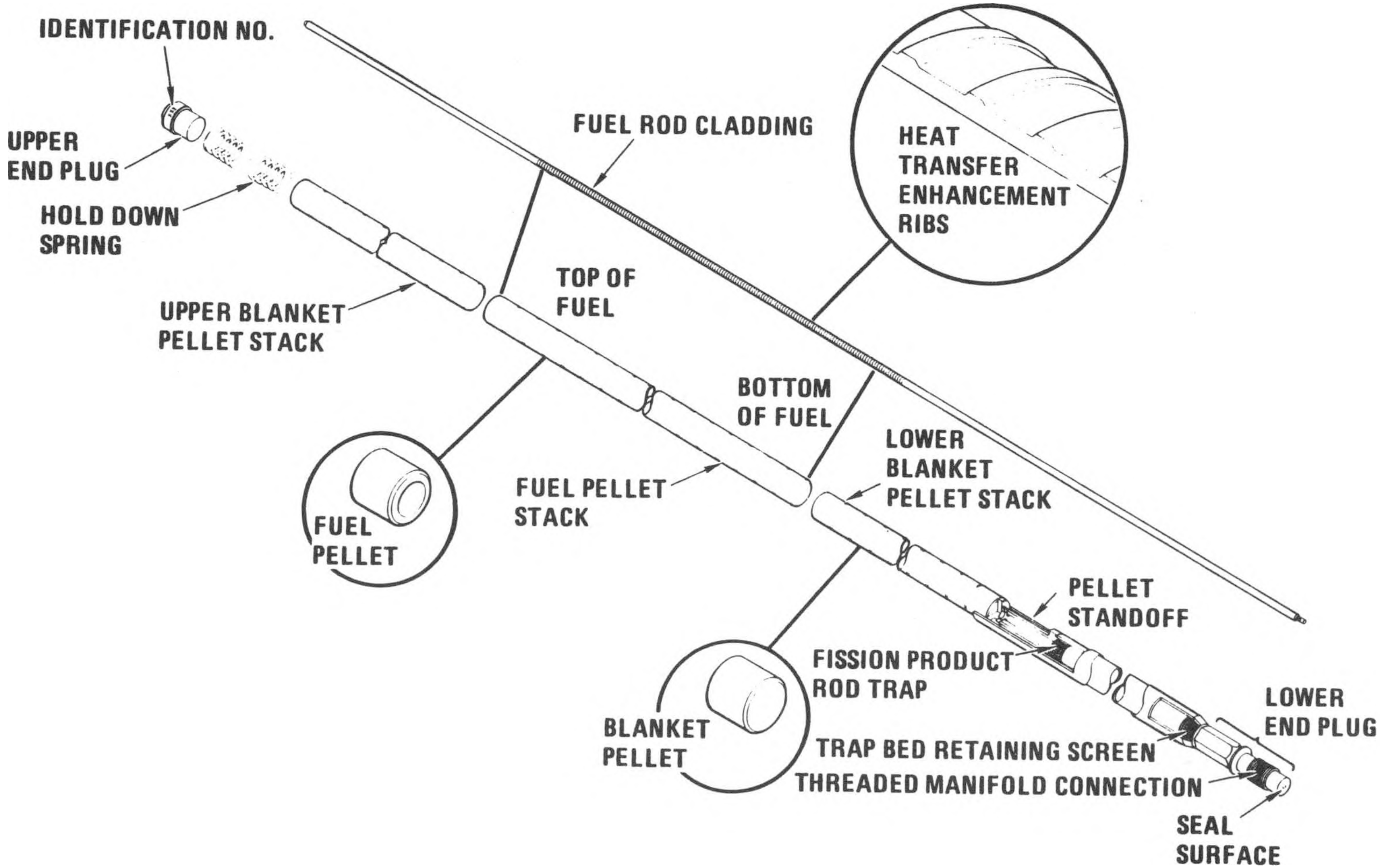

Fig. 3-15. GCFR fuel rod assembly

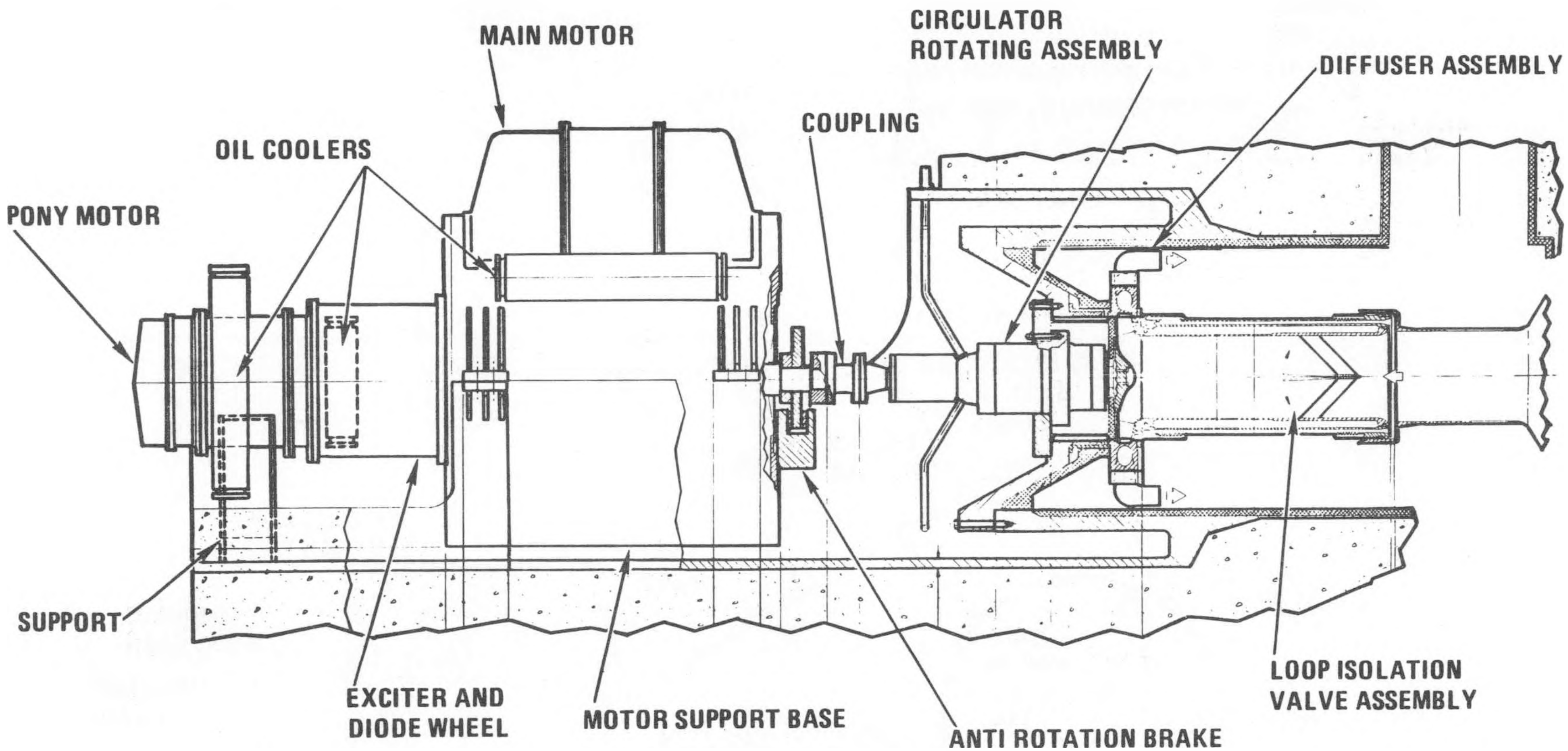

Fig. 3-16. GCFR main circulator horizontal installation

centrifugal impeller, passes into a series of pipe diffusers, and discharges into a plenum surrounding the circulator. Helium from this plenum then flows through the cold duct to the core inlet plenum.

Water-lubricated journal bearings and controlled leakage seals similar to those in the HTGR are used in the main circulators. This type of seal provides for hydrostatic operation desirable for maximum life. Water is used for the lubricant in preference to oil because of its simple shaft sealing systems. Any leakage would not contaminate the primary cooling system. A conventional oil-lubricated thrust bearing has been located in each electric motor.

The use of synchronous electric motors to drive the single-stage radial flow compressors simplifies testing of the circulator and preoperational testing in the reactor by permitting use of readily available electric power. Independently powered pony motors provide circulation for decay heat removal.

The steam generators (Fig. 3-17) in the GCFR demonstration plant are based upon design features similar to those already operating in the Fort St. Vrain plant and are expected to provide highly reliable operation. Since helium rather than water is the primary coolant in the GCFR, erosion, corrosion, and denting problems currently being experienced in other steam generator designs are expected to be minimized.

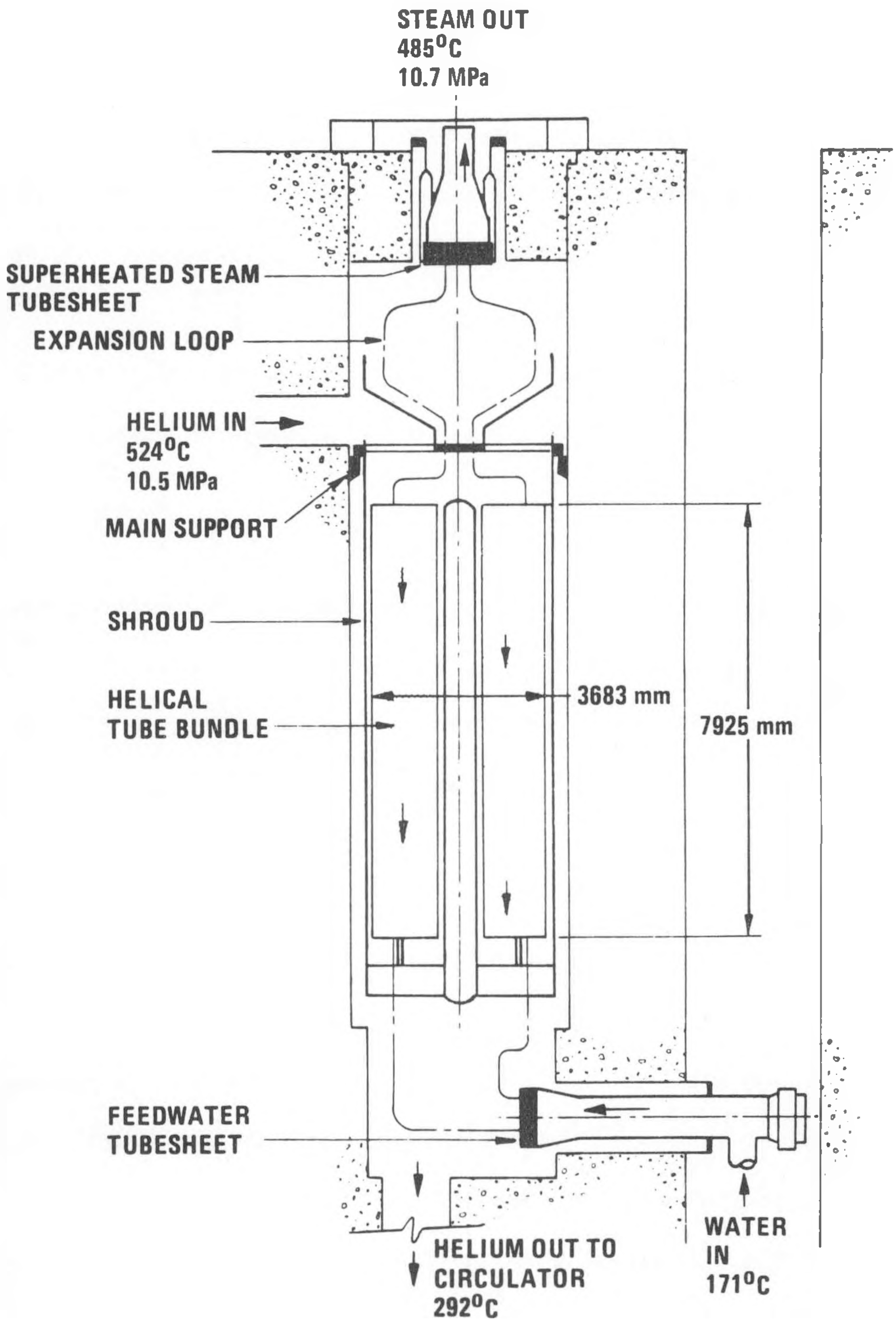

Fig. 3-17. GCFR steam generator

The GCFR steam generators are located in cavities in the PCRV. Hot helium enters at the top of the steam generator, flows downward on the shell side of the tube bundle, gives up heat to water and steam, and exits at the bottom. The water enters through a feedwater tubesheet located at the bottom, flows upward inside the tubes, and exits at the top through a superheater tubesheet.

The reactor core cooling systems are designed against numerous, sometimes conflicting, criteria. The normal power conversion system must have high availability, ease of control and maintenance, and reasonable thermal efficiency. Components for this system should be as nearly "off-the-shelf" as possible.

When the reactor is scrammed, safety grade residual heat removal (RHR) systems are available to back up the normal power conversion system. The criteria for these safety systems are (Ref. 3-12):

1. Two redundant, independent, diverse safety-grade cooling systems that are capable of core cooling for all initiating events except those with a very low probability. For low-probability events, such as rapid depressurization accidents, only one system is required.

2. A reliability goal, considering common mode failures, of 10^{-6}/yr probability of exceeding the U.S. Code of Federal Regulations radiation dose guidelines.

Implementation of these criteria has resulted in the overall cooling system design shown schematically in Fig. 3-18. Normally, core cooling is provided by the main loop cooling system (MLCS) composed of the main circulator, steam generator, turbine, condenser, boiler feedpumps, and feedwater heaters (not shown). Water from the cooling towers is circulated to take heat away from the condenser. The MLCS is designed to be powered either from outside line power or from house power, whichever is available. Thus, for a loss of outside power lines, the system is ramped rapidly back to about 25% thermal power, the turbine generator covers house loads, and steam is bypassed through a desuperheater and flash tank for house steam requirements with the excess going to the condenser, thereby maintaining core cooling on the MLCS. For a turbine trip, the plant power is ramped back, steam is similarly bypassed providing for steam requirements, and electric power needs are taken from the grid. During refueling and maintenance outages, MLCS cooling is somewhat more vulnerable since house power is not available.

If for any reason the MLCS is not available, the cooling function automatically transfers to the shutdown cooling system (SCS). The SCS uses the main circulator driven by a pony motor, the steam generator, and a separate heat dump system composed of an atmospheric water storage tank with integral heat exchanger/condenser and a makeup water supply. Coastdown of the main circulator provides time for activating the emergency power supply for the pony motors for cases where both the turbine and off-site line power are lost. Water flow from the SCS starts immediately by natural circulation upon opening of one valve. The atmospheric pressure water storage tanks

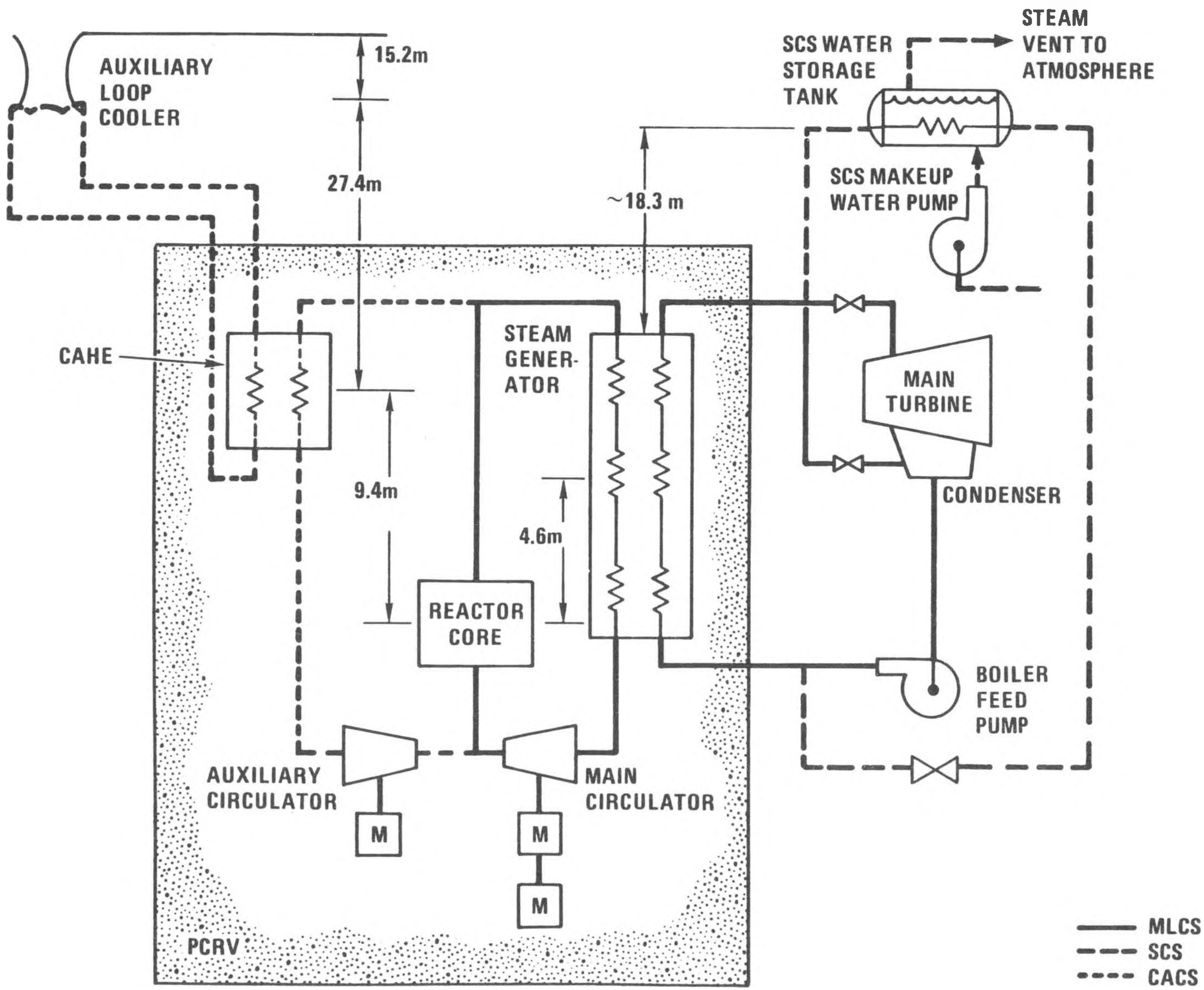

Fig. 3-18. Schematic of GCFR core cooling system

have 20 to 30 min of inventory before the makeup water supply must be established.

If for any reason MLCS/SCS cooling is lost, then the cooling function automatically transfers to the CACS in the forced circulation mode. The auxiliary loop isolation valves and auxiliary loop cooler louvers open and the auxiliary circulator provides helium flow.

If for any reason forced circulation on the CACS is lost, natural circulation provides adequate passive core cooling. The CACS gives up its heat in the CAHE to a pressurized water system that can also operate under natural convection. The heat is dumped to the atmosphere through a natural draft cooling tower.

Figure 3-19 shows the results of a calculation of the maximum fuel cladding temperature that would occur when the core, after scram, is cooled only by natural circulation.

The various residual heat removal systems for the GCFR are summarized in Table 3-5. It is believed that with these systems and the use of helium, the GCFR can be adequately cooled under all conditions.

The above discussion relates to pressurized events. For depressurization accidents, the SCS is bypassed. Both the main loops and the CACS are designed to cover the approximately 2 x 10^4 mm^2 design basis depressurization accident event. During a design basis depressurization accident, MLCS

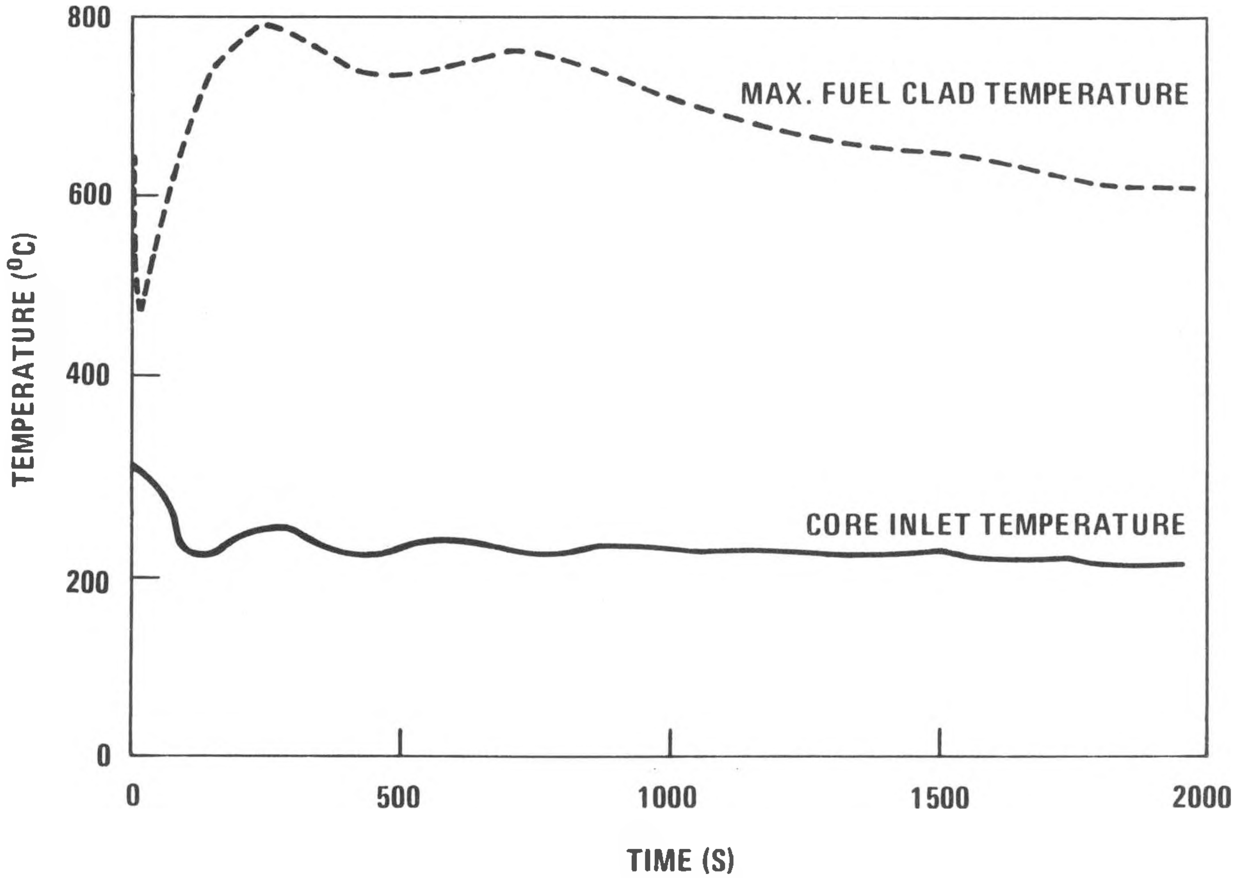

Fig. 3-19. GCFR core cooling by natural circulation only

TABLE 3-5

GCFR RESIDUAL HEAT REMOVAL SYSTEMS

Main Loop Shutdown Cooling System (Non-Safety Class)

Shutdown Cooling Systems (Safety Class)

Core Auxiliary Cooling System (Safety Class)

Natural Circulation (Safety Class)

cooling would normally continue with power from the outside line and steam supply from the flash tanks. If for any reason the MLCS is not available, then the CACS automatically takes over the cooling duty.

For depressurized refueling and maintenance events, the core cooling reliability criteria have led to a requirement to be able to repressurize the primary coolant system to 1 MPa. This pressure level permits adequate, passive core cooling by natural circulation on the CACS. The refueling machine and cask, as well as the circulator casks, were being designed to hold this pressure.

3.1.2.3. Development Program. In parallel with the GCFR design effort, extensive research and development was carried out in the United States in cooperation with Switzerland and the Federal Republic of Germany. A series of program plans for each of the major systems was prepared, initially in 1972 and updated in the period 1975 to 1980. The documents were prepared by General Atomic but were reviewed by the others working on the GCFR. The analytical, design, and experimental programs were all complementary, with duplication only where necessary to assure agreement in methodology.

For example, in 1974 a benchmark physics calculation was carried out by seven organizations in different countries with remarkably good agreement (Ref. 3-13). Likewise, a series of benchmark calculations was done and several international meetings held to obtain agreement on heat transfer computer codes.

Among the major accomplishments of the research and development program were the following:

1. A series of experiments were done to validate physics parameters on the critical assembly ZPR-9 (Fig. 3-20) at Argonne National Laboratory (Ref. 3-14).

   The critical experiments confirmed the very high breeding ratio that could be anticipated with the GCFR. Another experiment that has important safety implications is the effect of steam flooding on reactivity. Figure 3-21 shows the results, which are in excellent agreement with the calculations. Note that the reactivity effect is calculated to be negative even when the inventory of one of the three steam generators is in the core.

   The experiments and calculations were done for room temperature. Calculations for reactor operating conditions give more negative reactivity.

2. Using the Tower Shield Facility (TSF) at Oak Ridge National Laboratory (Fig. 3-22), experiments were done to determine the extent of streaming out the ends of fuel bundles (Fig. 3-23) so that appropriate shielding of the grid plate could be provided (Ref. 3-15). In 1979 a radial blanket and shield experiment was performed at the same facility to verify the radiation transport methods and nuclear data used for the design of the radial blanket

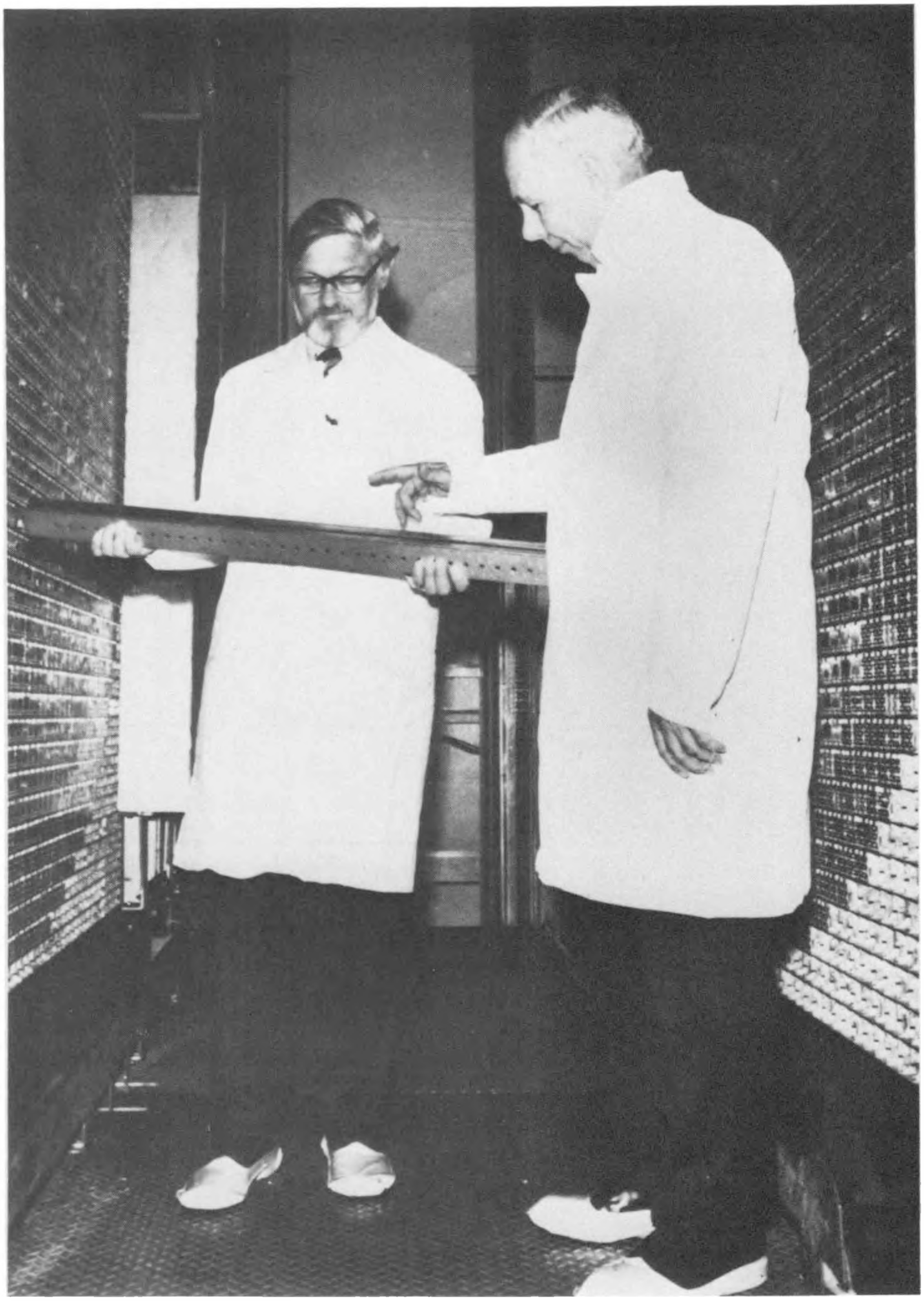

Fig. 3-20. Argonne National Laboratory ZPR-9

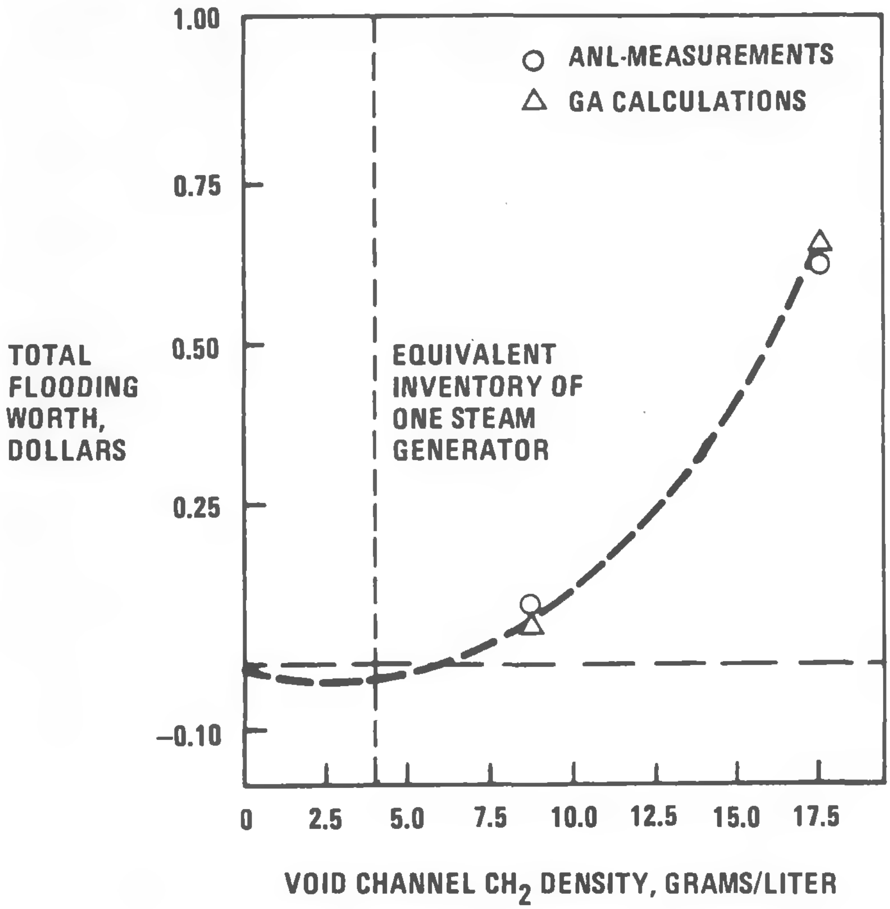

Fig. 3-21. Reactivity worth of simulated steam flooding in GCFR critical assembly with eight B_4C rods in a 1350-liter core

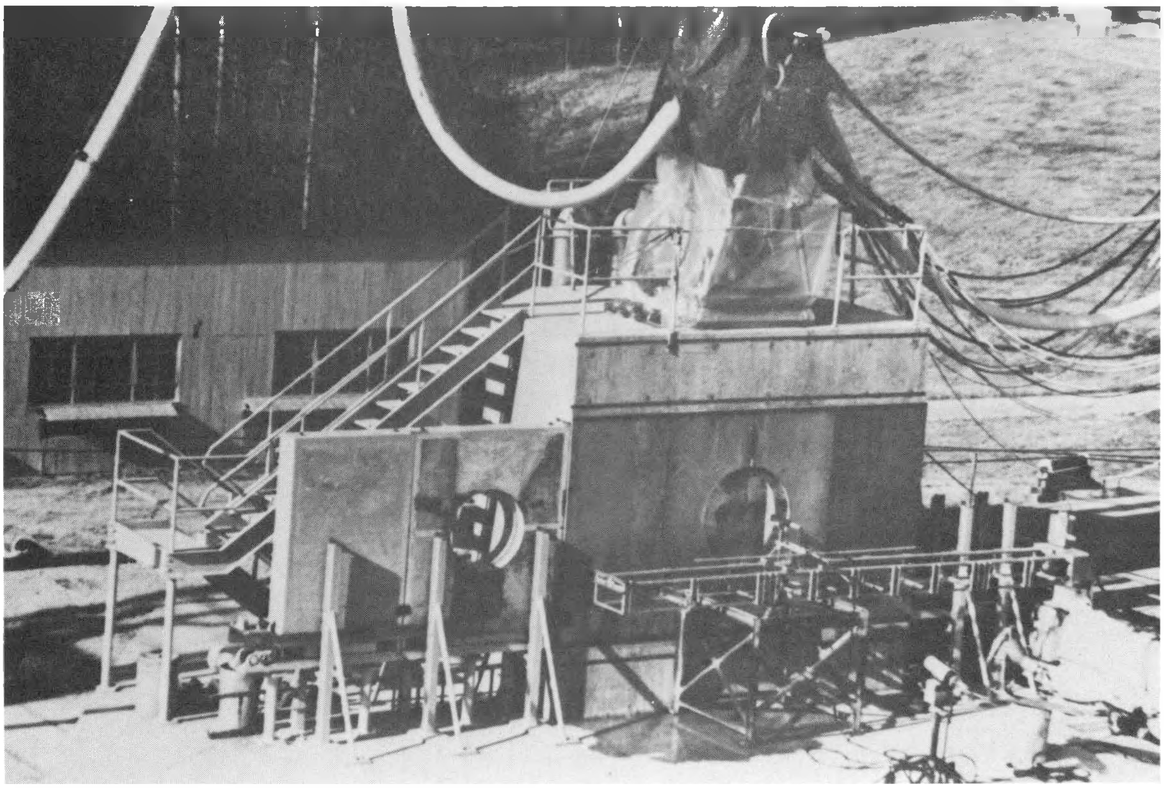

Fig. 3-22. Oak Ridge National Laboratory Tower Shielding Facility

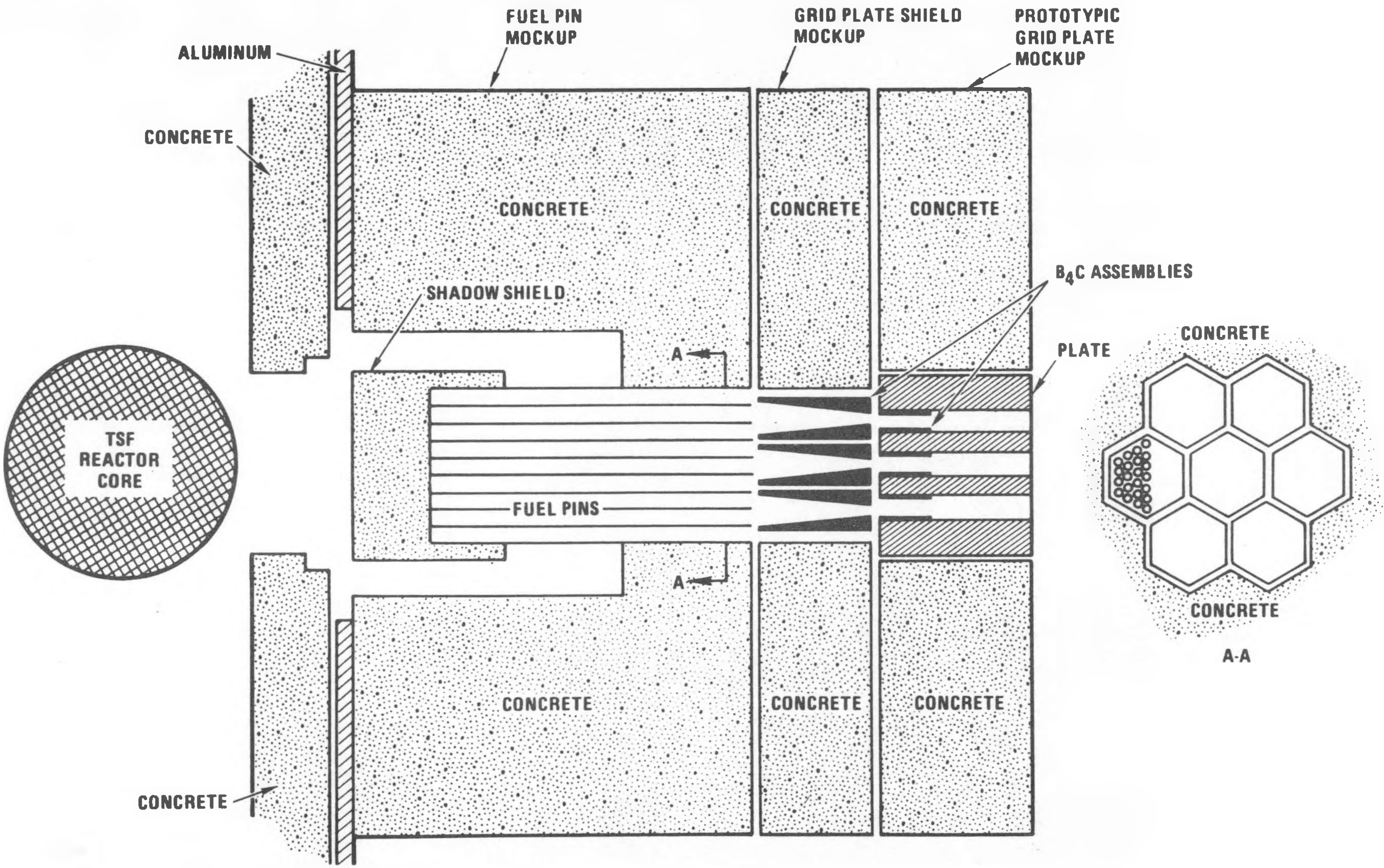

Fig. 3-23. Grid plate shield experiment with prototypical grid plate

and shield for the proposed 300-MW(e) GCFR. The completed post-analysis showed that a capability exists for accurately predicting (a) the neutron transmission through thick UO_2 and ThO_2 blankets and laminated shields containing steel, graphite, and boronated graphite, (b) the gamma ray heating rates within the configuration, and (c) the neutron spectra beyond the configurations.

3. An extensive program of thermal-hydraulic experimental studies was performed for the GCFR core assemblies (Ref. 3-16). The experiments consist of basic studies to obtain correlations and bundle experiments that provide input for code validation and design verification. These studies have been performed at European laboratories, U.S. national laboratories, universities in the United States, and General Atomic. The program included:

   a. Single rod experiments.
   b. Spacer tests.
   c. Effect of grid spacers on pressure drop.
   d. Effect of grid spacers on heat transfer
   e. Effect of wire-wrap spacers.
   f. Mixing tests.
   g. Effect of heat transfer enhancement rib shape.

   Simultaneously, computer code development to predict accurately the pressure and temperature within the GCFR fuel assemblies under a wide range of operating conditions was proceeding.

Special mention should be made of the British work on roughened rods for the Advanced Gas-Cooled Reactors, which provided the initial understanding of the thermal-hydraulic performance; the work at the Swiss Federal Institute for Reactor Research, which progressed to 37 rod bundles of electrically heated roughened rods (Ref. 3-17); and the work at Kernforschungszentrum Karlsruhe, which was the first to demonstrate the capability of high-pressure helium to remove the high power density of a fast reactor core at anticipated reactor temperatures (Ref. 3-18).

The next step was to have been tests in a high-pressure, high-temperature helium loop at Oak Ridge National Laboratory. The Core Flow Test Loop could handle bundles of up to 91 electrically heated fuel rod simulators operating at maximum GCFR linear heat rates (38.5 kW/m) under steady-state, transient, and accident conditions. The construction of the Core Flow Test Loop, which has been converted for use on the HTGR program, has been completed.

4. An irradiation program was conducted that emphasized the differences between the LMFBR and GCFR fuel rods and formed a bridge between the two so that the extensive fuel irradiation base of the former could be used to support the latter. The work progressed on two paths: development of the pressure equalization system and fast flux testing. The former proceeded through two increasingly sophisticated sweep gas capsules that were irrradiated in the Oak Ridge Research Reactor (Ref. 3-19).

The next step was a series of tests of bundles of 12 rods in the helium loop in the BR-2 at Mol, Belgium (Ref. 3-20). These tests were very successful and were a combined effort of Kernforschungs-anlage Julich and Kraftwerk Union in Germany, the Centre d' Etude de l' Energie Nucleaire Belgium, and General Atomic, with the primary responsibility and credit belonging to KFA Jülich.

The fast flux testing is continuing in the Experimental Breeder Reactor 2 in Idaho, U.S.A. Individual fuel rods and bundles have been tested in sodium up to approximately 780°C maximum hot spot cladding temperature and to burnups up to 13.2 atom %. Of particular interest were studies of fission product movements within the rods. Figure 3-24 shows a gamma scan of the zirconium and cesium fission products.

Other experimental programs have included:

a. PCRV model closure tests.
b. Materials tests in impure helium.
c. Cladding mechanical properties tests.
d. Friction and wear tests.
e. Steel melting and relocation tests.
f. Natural circulation tests.

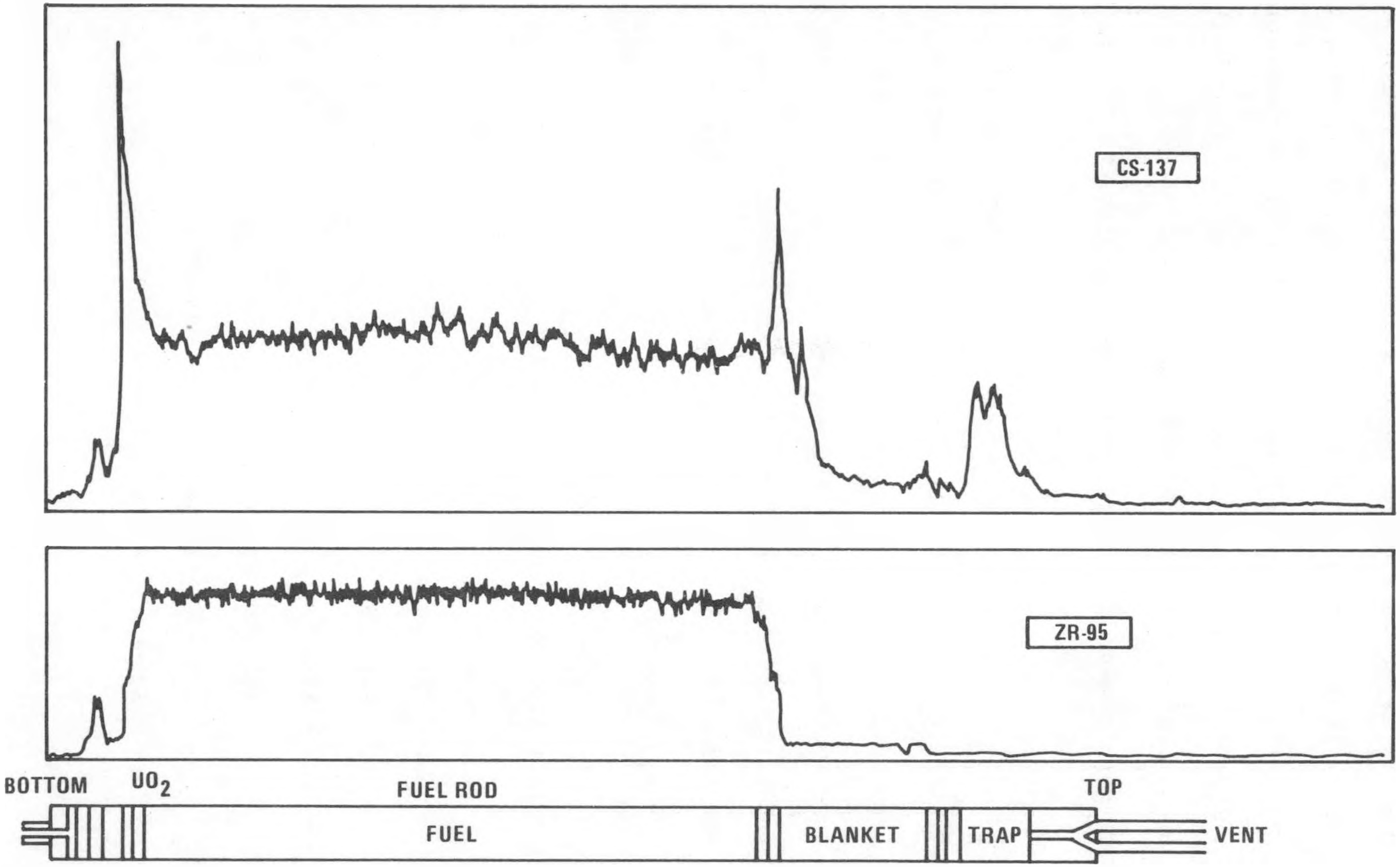

Fig. 3-24. Gamma scan of zirconium and cesium fission products in irradiated GCFR fuel rods

The results of the GCFR experiments were used in an extensive analytical program that was applied to the design of the reactor. At the time the conceptual design of the demonstration plant was completed, it was believed that there were no unresolvable technical problems, although a great deal of development work remained to be done and the design and analysis needed to be reviewed by the regulatory authorities.

3.1.2.4. Economics. Studies carried out by United Engineers and Constructors (Ref. 3-21), an architect/engineer company, for the U.S. Department of Energy indicate that the large GCFR will have an energy cost less than that of any other nuclear reactor power plant system. The results are given in Table 3-6.

3.2. INDUSTRIAL COGENERATION APPLICATIONS OF THE HTGR-SC*

3.2.1. Introduction

Many industries depend totally on the energy of oil and natural gas. Increasing costs and rapid depletion of these fuels necessitate the use of alternative energy sources for industrial power requirements. In addition, alternative energy sources curb oil imports. The only long-term, large-scale alternatives to fulfill these needs are coal and nuclear power.

*Ref. 3-22.

TABLE 3-6

TOTAL ENERGY COST

NORMALIZED COST UPDATE SUMMARY ($ 1978)

Mode	MW(t)	MW(e)	Mills/kW-hr 2001 Start-Up
BWR[a]	3800	1264	19.1
HTGR	3800	1504	16.8
PWR[b]	3800	1268	19.0
PHWR[c]	3800	1162	21.7
GCFR	3800	1440	15.8
LMFBR	3800	1390	18.1

(a)Boiling water reactor.
(b)Pressurized water reactor.
(c)Pressurized heavy water reactor.

Most energy-intensive industrial processes require considerable steam and electric power. Cogeneration of electric and steam power is an advantageous blending of two well-developed technologies: (1) electric power generation with its high-temperature, high-pressure inlet steam and condensing cycle and (2) industrial steam characterized by production at moderate pressures and temperatures with productive use of the heat of condensation. Cogeneration schemes have a high thermal efficiency and result in increased electric power generation.

The HTGR is particularly well suited for the cogeneration option,* since it can deliver steam at temperatures up to 540°C and 17 MPa. Use of a cogeneration plant results in economic benefits and considerable gas and oil fuel savings. In addition, the inherent safety features of the HTGR (Ref. 3-23) make it easier to site plants adjacent to industrial areas, permitting economical transmission of steam.

The cogenerating HTGR plant can be owned and operated by either an industry or a utility, or by a combination of both. Recent U.S. energy regulations encourage industries to own and operate cogenerating plants and have provided incentives. Several closely located small-and medium-size industries, each with moderate energy requirements, can form a consortium and use a centralized HTGR plant for obtaining steam and electric power.

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*This option has been chosen as the basis for the lead plant in the United States.

Excess energy can be sold to a local utility as electric power at avoided costs per current U.S. federal guidelines.*

3.2.2. Energy Demands

There is a large potential market for new energy options capable of serving the needs of industry. Increased energy demands projected for the year 2020 must be met both for growth in existing industrial processes and for large new processes, such as the production of synthetic fuel (synfuel).

Existing process plants primarily use oil and natural gas fuels. A study by General Energy Associates (GEA) (Ref. 3-24) indicated that 77% (500 GW/yr) of the process energy requirements are at temperatures of 540°C or less and that approximately 46% of the process energy requirements are in the form of steam. Another study by Dow Chemical Company (Ref. 3-25) focused on process steam and electric energy requirements of specific plants in selected geographic areas in the United States. The study identified 119 locations with a combined industrial steam requirement of at least 63 kg/s located within a 3.2-km-radius circle and 19 locations with a combined industrial steam requirement of at least 504 kg/s located within a 16-km-radius circle. These potential sites are shown in Fig. 3-25.

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*Cost of electric power that a utility avoids by not operating their highest-cost plant.

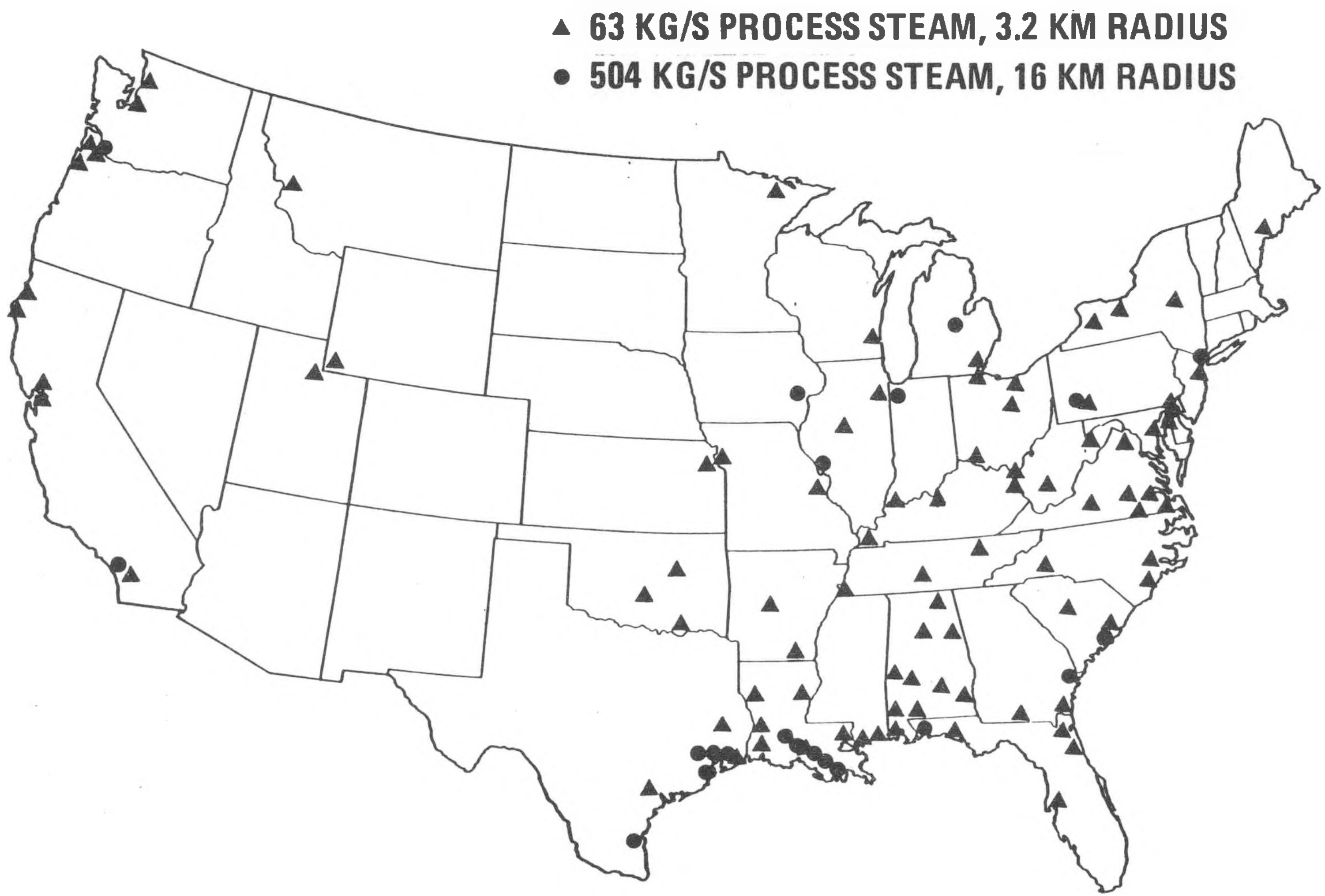

Fig. 3-25. Potential sites for steam grids for industry in the United States

General Atomic recently performed a study estimating the potential process steam market based on the Dow data and assuming a centralized heat source. Using a growth rate of 1.8%/yr based on a forecast by the Energy Information Administration (Ref. 3-26), the study showed by the year 2000 a total of 52 sites could use steam equivalent to the output of an 1170-MW(t) cogeneration plant, 10 sites could use 2 x 1170 MW(t) [or a 2240-MW(t) plant], and 11 sites could use 3 x 1170 MW(t) [or a 3360-MW(t) plant]. These numbers are upper limits because site-specific factors would reduce the usable number.

The synfuel industry is a large potential market area for the HTGR. According to a Pace Engineering forecast, total synfuel production (including shale oil) is expected to rise from the crude oil equivalent of 11,000 m^3/d in 1985 to almost 1.6 Mm^3/d in 2020. Process energy requirements for these synfuel plants will be substantial [estimated at 700 to 1200 MW(t) per 8,000 m^3/d plant] and may provide a significant market for HTGR units.

3.2.3. Plant Description

The HTGR plant operating in a cogeneration mode is identified as the HTGR-Steam Cycle/Cogeneration (HTGR-SC/C) plant. The NSS system is exactly the same as that of the HTGR-SC plant described earlier and shown in Fig. 3-3. It produces steam at the same elevated conditions of 17.2 MPa and 540°C. Flexibility in the process steam temperature and pressure conditions is provided by several possible combinations of turbine arrangements (topping, condensing). A steam cycle diagram for the reference multipurpose plant is shown in Fig. 3-26. The superheated steam from the NSS system is first partially expanded in a back-pressure turbine to the conditions required for process steam. The flow is then divided and the bulk of the steam is directed to the process users. The remaining steam is cascaded through a second back-pressure turbine-generator, and the extraction and exhaust steam is used for feedwater heating. Considerable flexibility is obtained with the operation of the HTGR from cogeneration mode to non-cogeneration mode by adding a low-pressure condensing turbine, as shown in the dotted portion of the figure. This portion of the plant could be built initially if the process steam load is projected to occur well after plant construction, or could be added later if the initial process steam load should drop off.

3.2.4. HTGR Applications

Table 3-7 shows various categories of energy-intensive industries/operations in the United States and their energy requirements. Most of the applications require substantial process steam at high-pressure superheated conditions. The availability of information on the energy requirements for each application was a key factor in determining its merit for consideration. In some cases, the commercial plant energy data were obtained by appropriate scaling of data from pilot or test plants. These were included for consideration when the energy requirements appeared promising.

A review of the energy requirements of the applications surveyed suggests that although the HTGR is not the only nuclear energy source

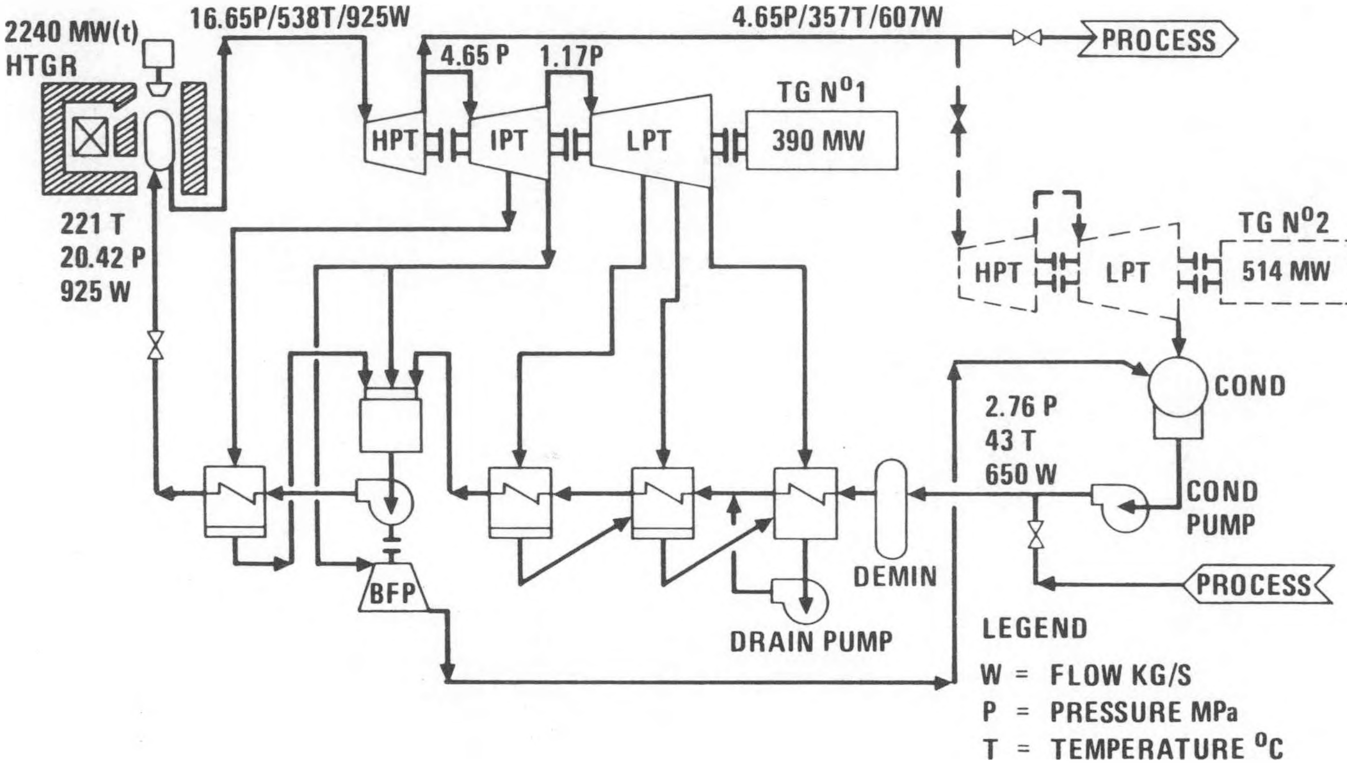

Fig. 3-26. Cycle diagram for 2240-MW(t) HTGR-SC/C application, maximum cogeneration and maximum electricity modes

TABLE 3-7

HTGR APPLICATIONS SURVEY - UNITED STATES

	Commerical Plant Size	Process Steam Requirements [MW(t)]	Electrical Power Requirements [MW(e)]
Existing Industries			
Petrochemical Plants (Geismar, LA)	--	2745	582
Refinery Plant	62,000 m^3/d	1929	111
Aluminum Mill (Grammercey, LA)	720 Gg/yr	317	94
Steel Mill	6500 Gg/yr	296	240
Petroleum Products Recovery			
In-situ Heavy Oil Recovery	5,600 m^3/d	993	1
In-situ Oil Sands Recovery	7,300 m^3/d	1090	83
In-situ Shale Oil Recovery	5,200 m^3/d	1090	--
Synfuel Processes			
Catalytic Coal Gasification - High-Btu gas (e.g., Exxon process)	6,800 m^3/d	1116	187
Coal Liquids - H-Coal	12,900 m^3/d	755	251
Coal Liquids - Donor Solvent (Exxon)	9,500 m^3/d	577[a]	260[a]
Syngas (slagging Lurgi) - methanol add-on	11,400 m^3/d	371	105

[a]Preliminary.

capable of supplying industrial high-temperature steam requirements, it is the most likely one. It has the capability of supplying higher temperatures than any other nuclear heat source. The LWR primary steam is limited to 288°C. The LMFBR and the GCFR are both capable of producing steam at temperatures higher than 450°C; however, the cost effectiveness of these applications of fast breeder reactors does not appear as favorable as that of the HTGR.

The applications shown in Table 3-7 presently use or propose to use oil, gas, and coal as the source of thermal energy. The petroleum products recovery operations use 30% to 50% of recovered product (crude oil). Synfuel processes will use mostly coal for thermal power, supplemented by use of produced fuels. For many of these applications, electrical power is purchased. Brief descriptions of the applications surveyed, as outlined in the table, are presented below.

The petrochemical complex (Geismar, Louisiana) application demonstrates the use of HTGR plants at sites where several energy-intensive industrial plants are concentrated in the area suitable for process steam distribution from a centralized source. Three 1170-MW(t) HTGR plants at a single site [or one 3360-MW(t) HTGR-SC/C plant] could supply process steam and electrical power to four user plants located within 4.8 km of the HTGR site. In addition to providing all of the process steam and electrical power required by the user plants, a surplus of electrical power is available for export or other uses. Process steam requirements vary from 2 MPa/265°C to 7 MPa/409°C, but most is required at 5 MPa/360°C (Ref. 3-27).

In the refinery complex application, the use of the HTGR plant to supply steam and electrical power to a 62,000 m^3/d refinery complex was studied. Two 1170-MW(t) HTGR plants are required to supply electrical power and steam at several pressure/temperature conditions. An excess of electrical power (~216 MW) is available for other uses internal or external to the plant. Process steam supplied to the refinery ranges from 9.5 MPa/452°C to 1.6 MPa/248°C.

In the aluminum mill application, a Grammercey, Louisiana, plant that produces bauxite was studied. For the 720 Gg per year alumina plant considered, the 1170-MW(t) HTGR supplies all of the process steam and electrical power requirements of the alumina plant and, in addition, produces a surplus of process steam, which can be used for other process users or for electrical power production. With the integration of an 1170-MW(t) HTGR unit in a commercial alumina plant, the excess electrical power available [~233 MW(e)] could be used for the electrolysis of alumina in a combined plant.

The application of an 1170-MW(t) HTGR plant to a 6500 Gg (liquid steel) per year typical commercial steel mill was studied. For this size plant, the HTGR could supply all of the steel mill thermal energy and electrical power requirements and produce a surplus of energy for other users. This surplus could be in the form of either process steam (~395 MW) or electrical power (~115 MW). Steam is provided to the steel mill at 5 MPa/365°C.

The petroleum product recovery schemes studied include heavy oil, oil (tar) sands, and shale oil by the in-situ injection method. In-situ heavy oil recovery by steam injection has been commercially developed, whereas oil sands and shale oil are at the developmental stage in the United States. Energy requirements for the latter two recovery schemes were obtained by scale-up of the pilot plant data. Depending on the oil field development by the user, a good potential exists for the use of several 1170-MW(t) HTGR plants in all three schemes. Steam requirements vary from 4 MPa/250°C in heavy oil fields to 11 MPa/540°C in the in-situ shale oil recovery projects. The HTGR plant can supply all of the process steam and electric power requirements for the petroleum product recovery schemes and, in most cases, has excess energy available. This excess energy can provide for on-site crude oil up-grading. The HTGR plant, as an alternative energy source, eliminates the present practice of burning recovered crude oil and produced fuels in the boilers.

A study was carried out for the application of the HTGR as an energy source for the Exxon catalytic coal gasification process, which is currently under development. For a 13 Gg per day coal processing plant, the HTGR plant could supply all of the plant energy requirements, except for a small portion (~10%) needed for feed preheating at very high temperatures (>540°C).

The coal liquids processes studied included the H-Coal and Donor Solvent (Exxon) processes. Commercial plants are planned to process about 27 Gg of coal per day. Energy requirements for commercial operation were obtained by scale-up of either pilot plant or demonstration plant data. Process steam conditions range from 0.86 MPa/230°C to 11 MPa/510°C. A considerable amount of steam is required for heating the coal slurry feed. One 1170-MW(t) HTGR plant can supply all of the process steam and most of the electrical power requirements of a 27 Gg/d coal liquefaction plant. Steam (thermal) power requirements constitute about 70% and electrical power (not thermal equivalent) about 30% of the total power requirements for the coal liquefaction processes studied.

An additional coal liquids process was also studied in which the syngas produced by a slagging Lurgi process is further treated to produce methanol. An 1170-MW(t) HTGR plant can supply all of the steam and electrical power requirements for such a plant processing about 18 Gg/d of coal. A surplus of energy [~240 MW(e)] is also available.

3.2.4.1. HTGR-SC/C Comparative Assessment. There are three main areas for comparison of HTGR-SC/C plants with other energy alternatives: economic, environmental, and conservation.

Table 3-8 shows an economic comparison of the HTGR-SC/C with a coal-fired cogeneration plant of comparable size and a No. 2 oil-fired plant (non-cogeneration) (both units configured for multipurpose applications). These estimates are based on the ground rules in Ref. 3-6. The comparison indicates that the HTGR-SC/C has a significant (37%) steam cost advantage over the reference coal plant when both operate at 70% capacity factor. At an 85% capacity factor, the HTGR advantage is 67%. For large plant sizes this advantage is considerably greater. The advantage in comparison with small oil-or gas-fired units is substantially larger (500+%).

TABLE 3-8

COST COMPARISON FOR HTGR-SC/C AND COAL- AND OIL-FIRED PLANTS

($ Millions)

	HTGR Multipurpose	Coal Multipurpose	Existing No. 2 Oil
Heat Input to Cycle (MW)	1170	1230	1000
Heat Output in Process Steam (MW)	1000	1000	1000
Net Electrical Power Output (MW)	150	157	Insignificant
Total Capital Cost (1995 $)	1517	972	Insignificant
Annual Costs (mills/MJ[a]) (1995 - 30-yr levelized)			
Fixed charges, fuel plus operation and maintenance	18.6	23.1	57.5
Credit for electric power	<7.1>	<7.4>	--
Total annual costs	11.5	15.7	57.5
Ratio (normalized to HTGR)	--	1.37	5.00

[a] One mill = $0.001

Another key economic consideration is the distance of energy transmission and the associated cost. Remote siting of the HTGR from the user area may be required in some situations and would decrease the energy cost differential due to the higher energy transmission cost. Space limitations or environmental restraints in the vicinity of a process plant may also require the remote siting of a coal-fired plant.

Although significantly larger environmental issues exist for a coal-fired plant than for an HTGR, quantitative data for comparison with an HTGR are yet to be developed. The environmental advantages of the HTGR will become more apparent when data are developed on CO_2 effects, ash and sludge disposal, land use, atmospheric pollution, and other aspects.

Other cogeneration applications offer similar possibilities for energy savings and for displacement of large volumes of fossil fuels. The petroleum recovery applications (heavy oil or oil sands) in particular require between 35% and 50% of produced oil as process fuel for steam production. Use of an HTGR would save this valuable oil for more beneficial use.

3.2.4.2. Key Issues. Several key issues must be resolved to deploy nuclear cogeneration projects, including project schedule length, backup power, siting, and institutional matters. New coal plants in the United States have long schedules (6 to 8 yr) for deployment, and nuclear plants require even longer schedules. These long schedules present difficulties, particularly since the industrial owner(s)/user(s) finds it difficult to project energy needs over such long periods. For a utility-owned cogeneration plant, uncertainties would arise regarding future load requirements. The HTGR-SC/C plant can greatly mitigate concerns in this area, since the plant can be adapted to operate in a full cogeneration mode, at an intermediate point, or at the full electricity generation mode.

Backup power provisions (for steam and electric power) are essential for cogeneration applications. These requirements can be met in several ways: by additional nuclear or fossil units that normally produce electric power but can be switched to the cogeneration mode, by use of existing fossil units, and possibly by energy storage systems such as the sensible energy transport system (SETS) using heat transfer salt. In addition, a larger-than-required HTGR unit can be deployed with part of the capacity devoted to cogeneration and the remaining capacity devoted to electric power production using a condensing cycle and backup when needed.

There are several options regarding siting of HTGR-SC/C plants. In general, close-in siting is preferred, and the unique safety features of the HTGR should be of benefit in this respect. Distant siting would be at sites within practical steam transmission distance, and remote siting would involve long-distance energy transport via a heat transfer salt or similar system. A high-temperature version of the HTGR can utilize a chemical energy transport system that could also be used for remote siting. Distant or remote siting will add to the delivered cost of process steam. The practical range of applicability of these systems will be determined by comparison with other energy alternatives.

Institutional issues are a major consideration in the use of HTGR cogeneration plants. The institutional barriers for small nonutility-owned units have been substantially reduced and incentives defined by recent U.S. legislation and rulemaking. Some of this work also applies to large cogeneration plants, but additional institutional changes are needed to remove impediments and provide incentives for deployment of large cogeneration projects owned and operated by electric utilities.

3.2.4.3. Conclusions. The following conclusions were reached from the HTGR applications study:

1. Most of the applications require considerably more steam power (~80%) than electric power (~20%), and a significant part of the steam is required at high-pressure superheated conditions. The HTGR was found to be a good technical fit to most processes and is the only nuclear source that could provide the required high-temperature steam, since light water reactor plants are limited to a maximum of 288°C.

2. The HTGR plant can fulfill the needs of several energy-intensive industrial processes/operations and can conserve natural gas and oil now used in industrial, petroleum product recovery, and synthetic fuel production processes.

3. Based on delivered steam thermal energy to an application, the economic advantage of the HTGR over its nearest competitor, coal,

is approximately 35% to 50%. Oil is nearly four times more expensive than coal and nuclear energy.

4. The HTGR plant has environmental advantages over a coal-fired plant, and the expensive environmental control devices required for a coal- or oil-fired plant are not necessary.

5. Site/process-specific application with user participation is essential to deal with key issues to implement necessary institutional changes.

6. The multipurpose design used in the HTGR configuration facilitates adaptation of the basic HTGR for a wide range of applications.

3.3. HIGH-TEMPERATURE GAS-COOLED REACTOR FOR PROCESS HEAT APPLICATIONS*

3.3.1. Introduction

The high-temperature heat available from the HTGR makes it suitable for many process applications. The high-grade heat can be used to produce hydrogen and synthetic fuel, with coal, lignite, residual oil, or oil shale as the carbon source (Ref. 3-29). It can also serve as the heat source for

*Ref. 3-28.

thermochemical water-splitting processes to produce hydrogen without carbon (Ref. 3-30).

The process heat HTGR (HTGR-PH) is being designed in two versions: an 850°C core outlet temperature with an intermediate helium loop and a 950°C core outlet temperature with a reformer and steam generator in the primary circuit.

With the many possibilities of application and the differing HTGR configurations under consideration, the following general deployment strategy has evolved. The chronology starts at steam cycle plants that are based upon the proven technology of the Peach Bottom and Fort St. Vrain plants. Core outlet temperatures are in the range of 700° to 750°C. Higher core outlet temperatures of 850°C follow with application to process heat and the gas turbine. These conditions are somewhat beyond current technology in terms of material behavior for commercial life components. Beyond this range, the very-high-temperature reactor (VHTR) at a temperature of 950°C can be considered. At each of these levels, new processes are presented for potential coupling to the nuclear heat source.

3.3.2. Sensible Energy Transport and Storage (SETS)

The advantages of steam production and cogeneration can be extended by the use of a heat transport and storage medium such as a sensible energy heat transfer salt (HTS) (Ref. 3-31). The salt can be used to store energy from a constant output heat source to meet cyclic loads. It can also be used to transport energy for distances beyond the practical range of steam transmission on the order of up to 40 km.

The SETS system uses a core outlet temperature of 750°C, about equal to the Fort St. Vrain steam cycle plant, which is adequate to heat the salt (a mixture of sodium nitrate and potassium nitrate) to its maximum practical temperature of 565°C. An intermediate helium loop is used to isolate the salt from the primary coolant and the core. It may be possible to eliminate this additional loop, but at present the effects of salt ingress on the primary circuit are not fully known. A flow diagram using this system is shown in Fig. 3-27.

3.3.3. 850°C Indirect Cycle HTGR-PH Plant

In this variant of the process heat plant, the reactor thermal energy is transferred to the process plant via a secondary helium transfer loop, hence the indirect cycle designation. For the indirect cycle concept, as for other HTGRs, the entire primary coolant system is contained in a PCRV, which provides the necessary biological shielding in addition to the pressure containment function. An isometric representation of the indirect cycle arrangement is shown in Fig. 3-28.

The HTGR-PH plant design envisages a nuclear-chemical process whose product is hydrogen (or a mixture of hydrogen and carbon monoxide) generated by steam reforming of a light hydrocarbon mixture. The basic flow diagram for steady-state operating conditions is shown in Fig. 3-29. Within the

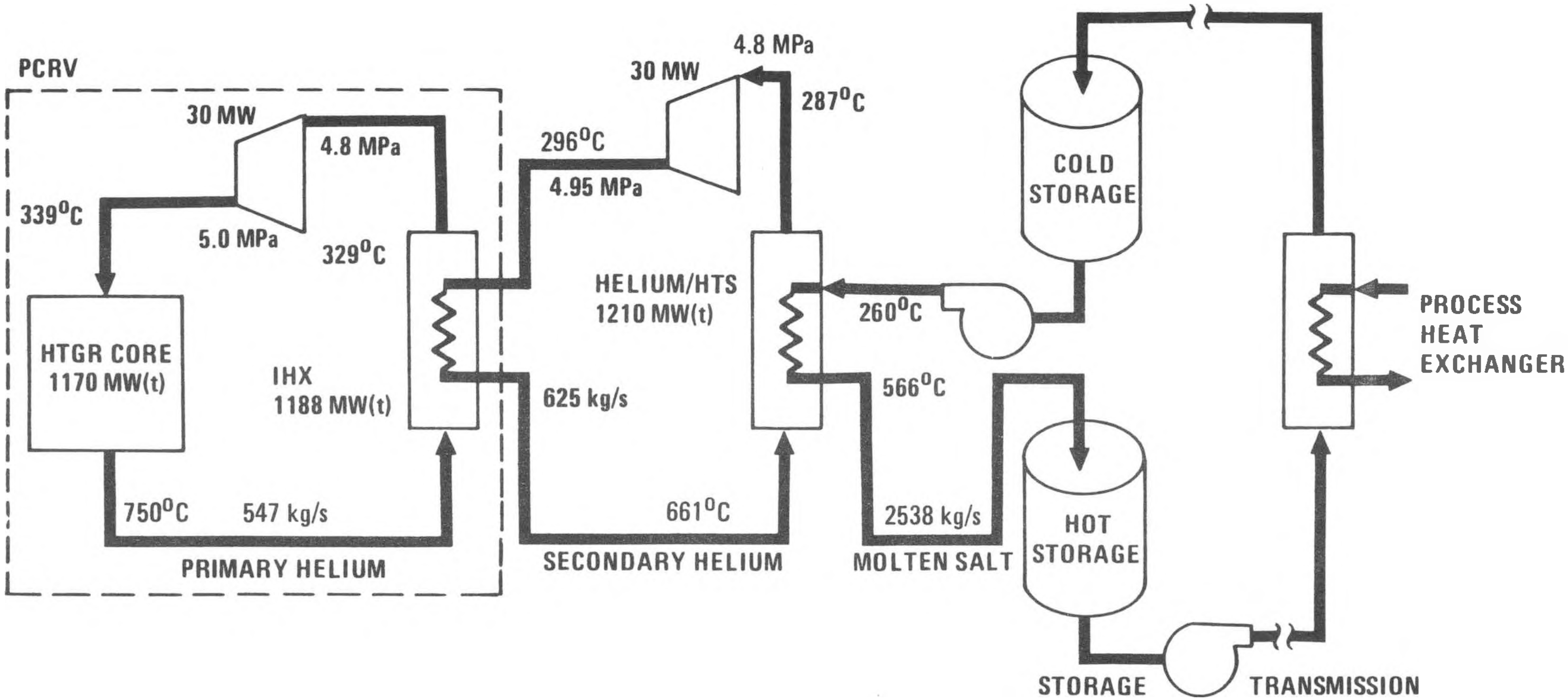

Fig. 3-27. Indirect cycle HTGR-SETS plant

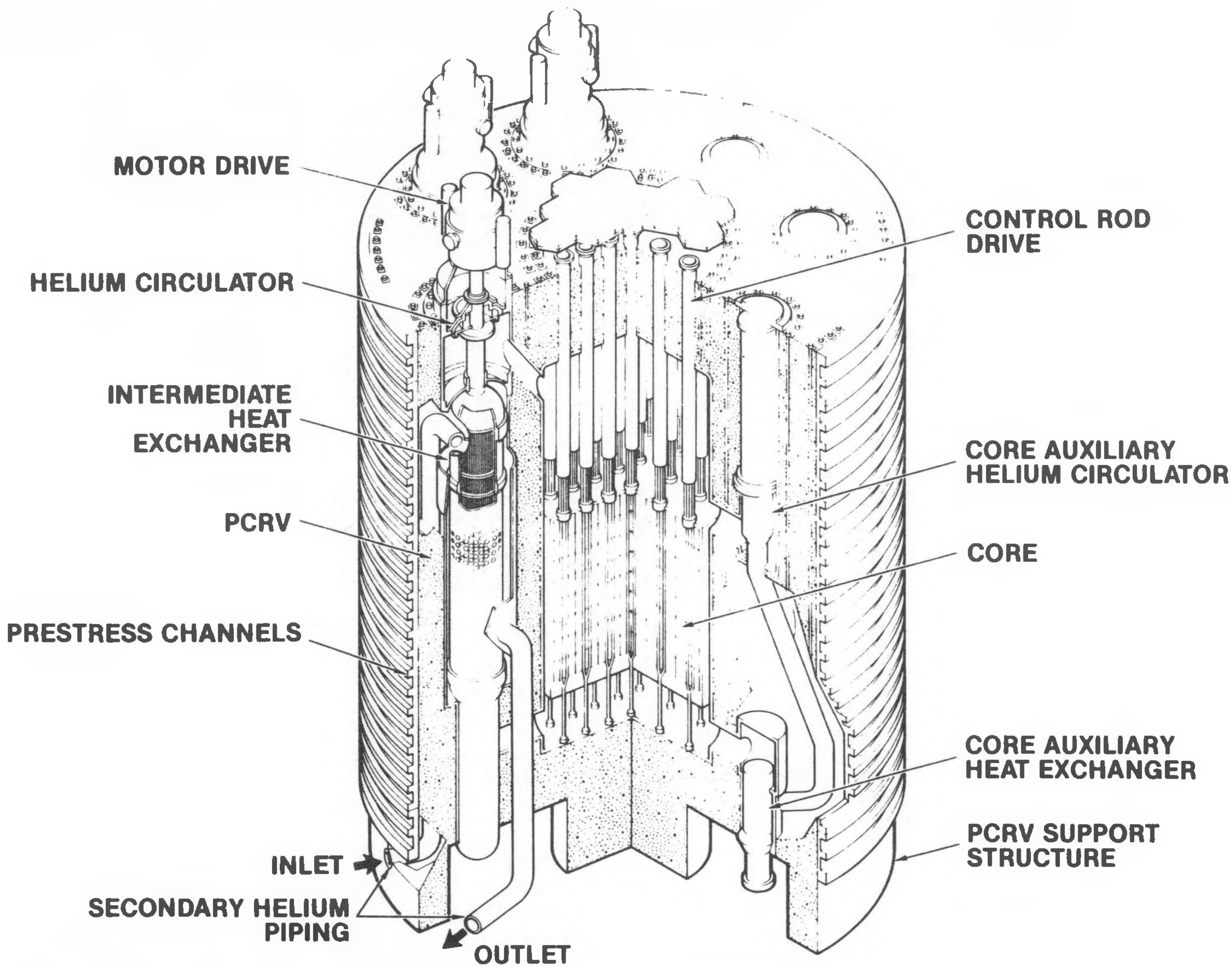

Fig. 3-28. Isometric view of PCRV for 850°C indirect cycle HTGR-PH plant

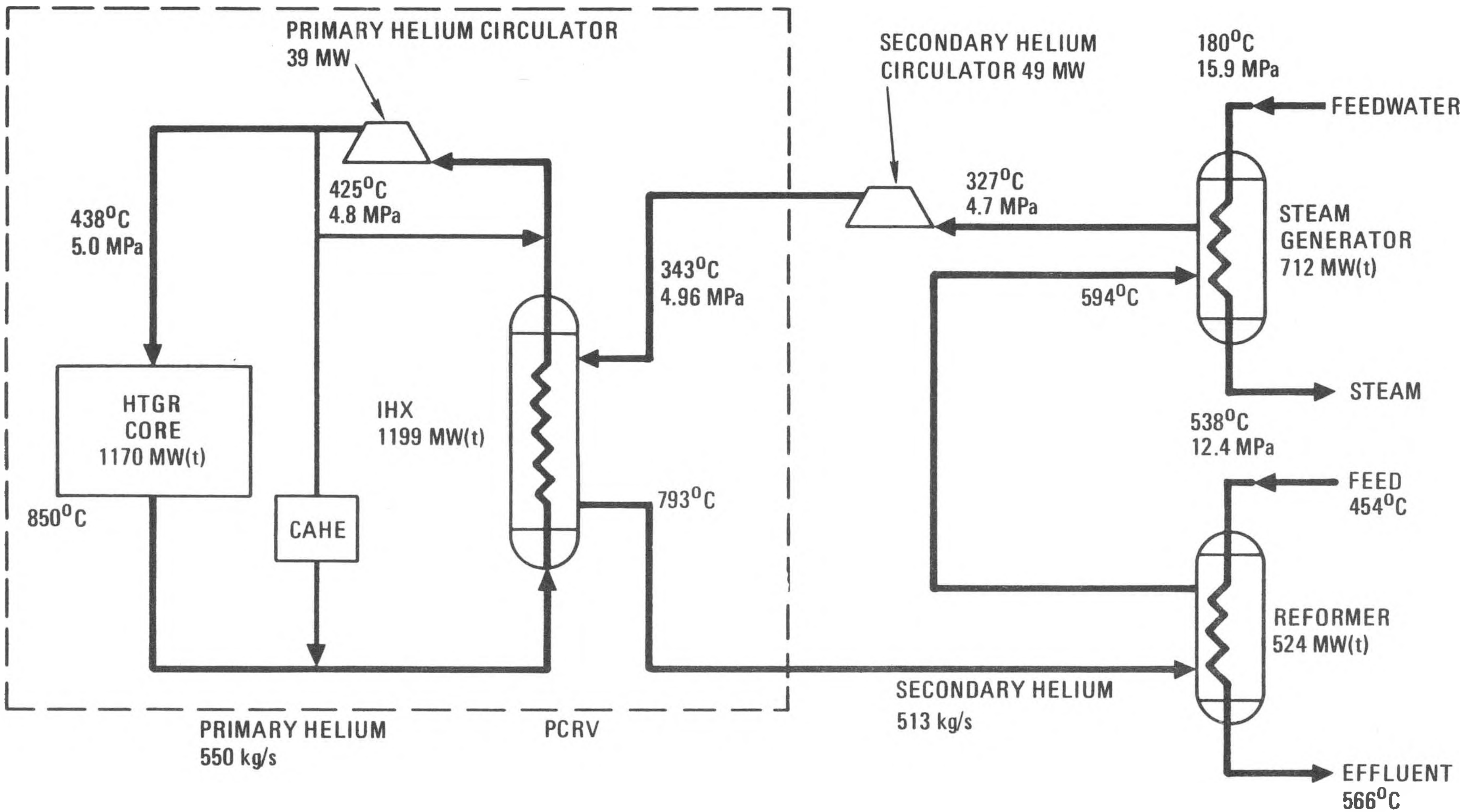

Fig. 3-29. Flow diagram for 850°C indirect cycle HTGR-PH plant

primary circuit the thermal energy from the 1170-MW(t) reactor core is removed by four independent loops. Each loop includes an intermediate heat exchanger (IHX), an electric-motor-driven helium circulator, and related instrumentation and controls. The primary helium temperature conditions (particularly the core outlet temperature of 850°C) were established to meet the requirements of the chemical process while staying within the structural limits of the Inconel 617 material used in the IHX. The pressure level of 5 MPa is consistent with approaches followed in the Fort St. Vrain design. The pressure level of the secondary helium is maintained only slightly above that of the primary system (thus preventing leakage of reactor helium into the secondary circuit) in order to minimize the long-term loading on the IHX, in which the combination of high temperature and material limitations requires a near-pressure-balanced operation for structural integrity.

The four secondary helium loops each consist of an IHX, a reformer, a steam generator, a secondary helium circulator, and related piping, valves, and instrumentation. During normal operation secondary helium is heated to 793°C in the IHX and routed outside the PCRV to the reformer and then to the steam generator, which extract the heat necessary for the process and auxiliary power generation. The secondary loop equipment (eight reformers, four steam generators, and four circulators) is installed in four prestressed concrete pressure vessels (PCPVs). The PCPV closely resembles the PCRV with regard to structural and construction features.

3.3.4. 950°C Direct Cycle HTGR-PH Plant

In this variant of the process heat plant, the reactor thermal energy is transferred directly to the process in the primary system. As in the direct cycle case, the nuclear heat is used for the reforming process and to generate high-quality steam in sufficient quantities to satisfy both the process and electrical generation needs for operation of the nuclear plant and the reforming process.

The basic flow diagram for steady-state operating conditions is shown in Fig. 3-30. To investigate the maximum potential of the VHTR, the direct cycle reforming concept being discussed has a core outlet temperature of 950°C. This temperature increase from the 850°C indirect cycle necessitates the use of more advanced technology in the nuclear heat source, which implies a longer schedule for realization of the first commercial size plant.

In the direct cycle plant all of the nuclear heat source equipment is installed in the PCRV. In the plant discussed, four reformers, two steam generators, and four circulators are used for the 1170-MW(t) variant. From Fig. 3-30 it can be seen that the reactor outlet gas flows directly to the convectively heated reformer, transferring 650 MW(t) to the process gas. After leaving the reformer, the primary helium flows through the steam generator and, via the circulator, is transported back to the reactor inlet at a temperature of 500°C. The maximum primary system pressure of 4.8 MPa is

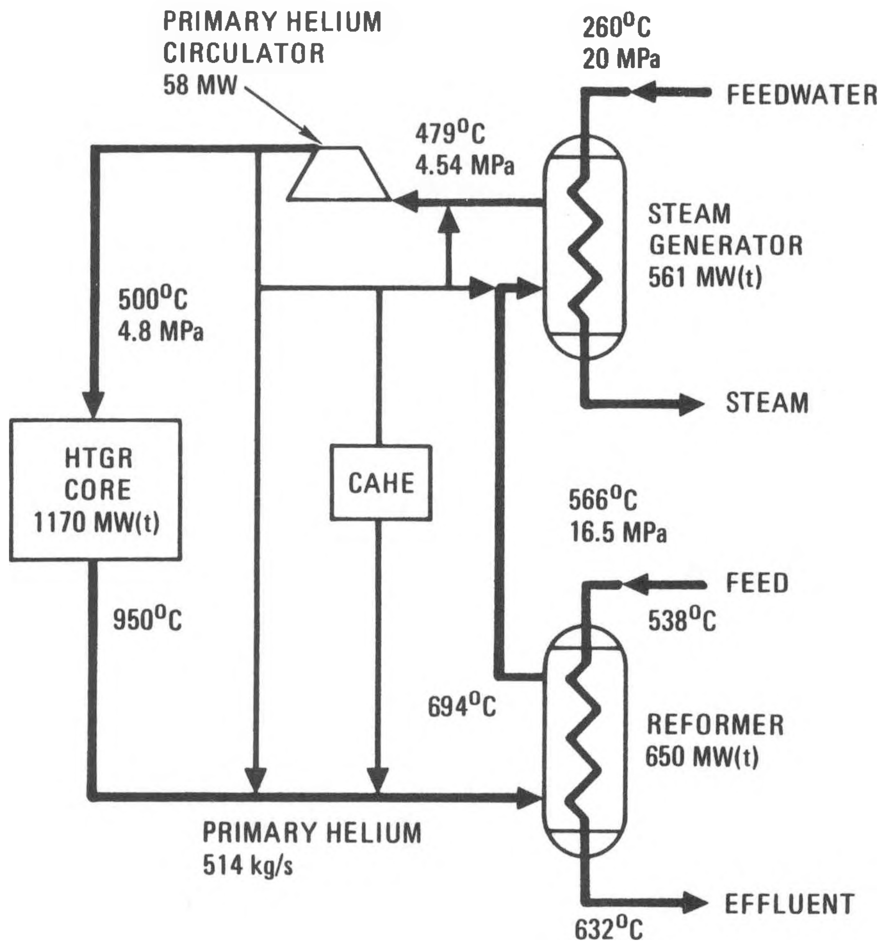

Fig. 3-30. Flow diagram for 950°C direct cycle HTGR-PH plant

slightly less than the indirect cycle case, and selection was based on process plant considerations (particularly the reforming operation).

While the technology transfer from the HTGR-SC plant to the advanced plants will be significant, it is recognized that for the higher-temperature systems embodying advanced technology features, considerable development effort must be expended (i.e., materials characterization, component development, etc.). For the advanced high-temperature systems, work is ongoing to identify the most appropriate plant concepts and configurations (including technical feasibility, cost effectiveness assessment, safety and licensability, benefits, risks, etc.) for the various applications.

3.3.5. Major Component Descriptions

All of the HTGR systems discussed are integrated configurations in which the nucear heat source equipment and entire reactor helium inventory are contained within the PCRV. Brief descriptions of the changes that would have to be made in the major components for the 850° and 950°C HTGR applications are given below.

The PCRV would not be affected, but for the 950°C system, extensive use will be made of ceramics and carbon-carbon fiber composite materials for the thermal barrier.

The basic structural material of the reactor core is nuclear-grade graphite machined in the form of hexagonal blocks. These blocks also serve as the moderator and heat transfer medium between fuel and coolant. In the case of the systems up to 850°C, the fuel elements are based on a 10-row block with a fuel cycle of 4 yr. For the 950°C case, an 11-row block was selected and a fuel lifetime of 3 yr was established. The 3-yr fuel cycle, with annual refueling of one-third of the core, should reduce the effects of fuel age difference in the core by minimizing peak fuel temperatures, neutron exposure, and consequent fission product release. Design requirements for yielding low fuel particle failure fractions, low fission product release, and low graphite block stress levels can therefore be satisfactorily met for the very-high-temperature systems.

The primary helium circulator requirements (in terms of pressure use and power) for the 700°, 750°, and 850°C systems are very similar. Indeed, as the programs progress, circulator commonality is regarded as being of paramount importance. The technology base for the circulator design lies with the steam cycle plant.

In the case of the 950°C direct cycle process heat plant, the higher system pressure loss (resulting from the incorporation of additional equipment in the primary system) can no longer be met by a single-stage centrifugal compressor. Recent studies have shown that a four-stage axial compressor could supply the necessary pressure rise. The technology base for this unit would be the Fort St. Vrain circulator with its single-stage axial flow compressor.

The function of the IHX is to transfer thermal energy from a primary coolant loop to a secondary loop and, in addition, to provide a barrier for egress of fission products (circulating within the reactor primary coolant) into the secondary helium loop. This component has been designed as a straight tube heat exchanger with the primary helium flowing in a single pass inside the tubes and the secondary helium flowing across the bundle in a multipass cross-counterflow configuration. As shown in Fig. 3-31, the IHX is located entirely in the PCRV and is welded at the lower end to a liner extension support. The upper end of the unit is attached to a primary/secondary gas boundary dome via a bellows/seal assembly, which compensates for IHX axial thermal expansion.

Although normal operation is in a near-pressure-balanced condition, the design basis is predicated on the loss of the secondary loop pressure. The current selection of tubing material for the 850°C plant design is Inconel-617. An extensive high-temperature materials program is in progress to obtain high-temperature data for the potential materials in this application. The material selection and design of the IHX represent a substantial technical development effort.

For the HTGR-SETS indirect cycle plant, the salt heater is a key component. Its function is to transfer into the molten salt loop the heat that is deposited in the intermediate helium loop by the IHX and by the secondary circulator pumping load. This unit is a straight tube assembly that is baffled to produce cross-counterflow heat transfer. The intermediate helium is routed inside the tubes, and the salt, which is at a lower temperature

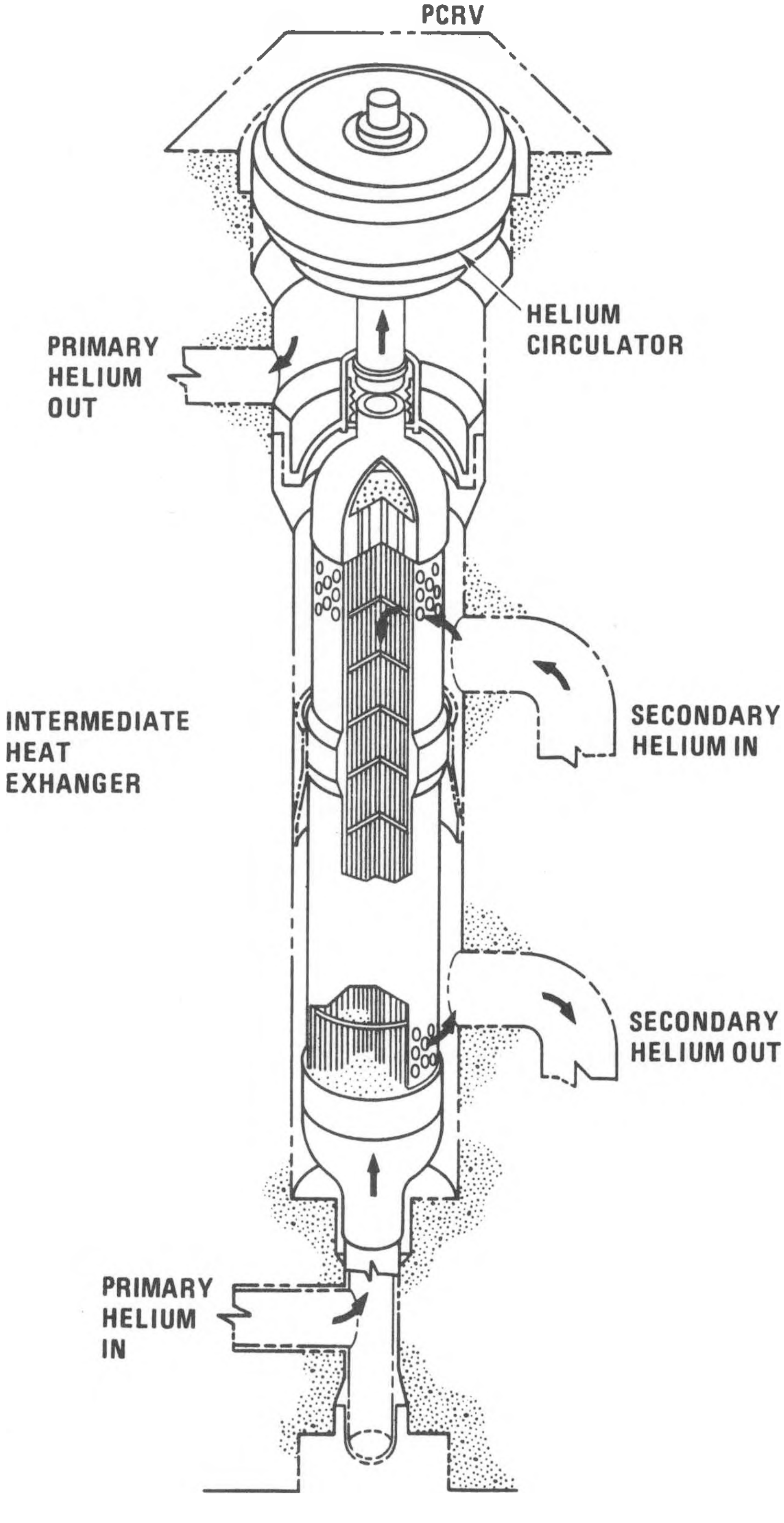

Fig. 3-31. HTGR-PH intermediate heat exchanger concept

and pressure, is the shell-side fluid. Provisionally, the unit is designed as a horizontal installation outside the PCRV, but it may be changed to a vertical orientation if justified by balance-of-plant considerations.

3.3.6. Reactor Applications

A large number of process heat applications can be envisioned for a nuclear reactor with a core outlet temperature of 850°C or above. Among these are coal gasification, thermochemical watersplitting, methane reforming, oil shale recovery, direct reduction of iron ore, and energy distribution and storage.

3.3.6.1. Coal Gasification. The incremental potential of an HTGR-PH coal gasification system may be illustrated through a typical example that was explored during 1981. In this example, the HTGR facility was configured to provide both process energy (in the form of direct heat and steam) and electrical energy for the Exxon catalytic coal gasification (ECCG) process (Ref. 3-32). The product of the ECCG process is methane (called substitute natural gas or SNG).

The ECCG process uses alkali metal salts as catalysts mixed directly with the feed coal to promote low-temperature gasification. These catalysts also increase the rate of steam gasification, reduce agglomeration of caking coals, and promote the achievement of gas compositions closely approaching gas phase methanation equilibrium. The process uses a fluidized bed gasification system that operates in a well-mixed mode approaching isothermality, the fluidizing gas being steam and recycle hydrogen and carbon monoxide.

Figure 3-32 shows the ECCG process heat requirements that have potential for HTGR coupling. Up to 997 MW(t) of energy at temperatures ranging from 245° to 855°C could be coupled to a plant designed to gasify 10.9 Gg/d of coal. This process also consumes electrical power in the amount of 190 MW(e). Four process steps show potential for coupling: (1) the gas preheat furnace for the gasifier, (2) the coal/catalytic drier (second drying stage), (3) the raw coal drier (first stage drier), and (4) offsite boilers supplying process steam.

The use of HTGR-derived heat to replace combustion in the ECCG process results in significant savings to the environment and of coal and product gas including:

- 2670 Mg/d of coal not burned.
- 210 Mg/d of product methane not burned.
- 240 Mg/d of ash not generated.
- 6270 Mg/d of carbon dioxide not emitted.

The technical and economic implications of high-temperature direct heat applications such as the above continue to be assessed in the U.S. HTGR

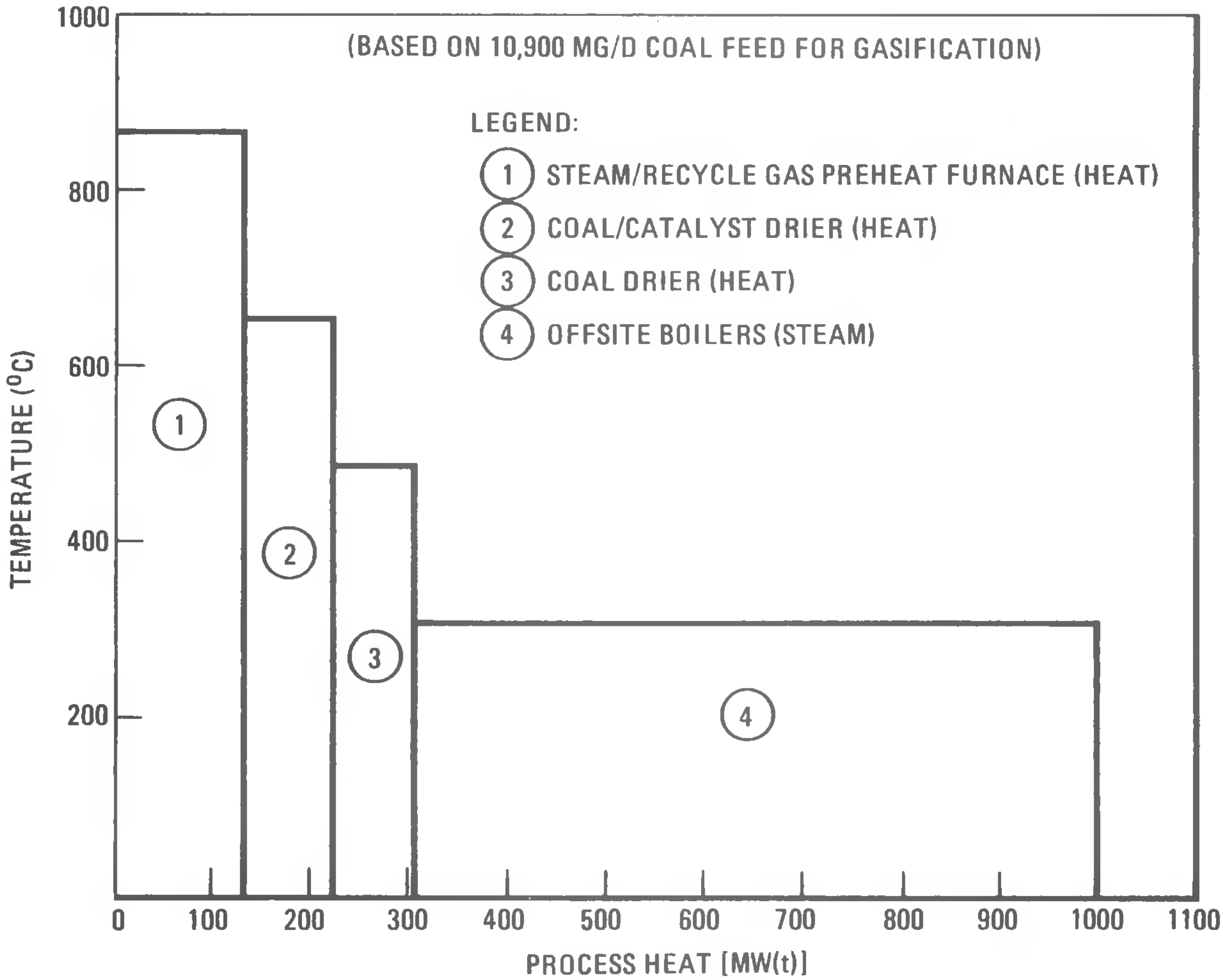

Fig. 3-32. Process heat requirements as a function of temperature

program. Future emphasis wil be placed upon reducing the relatively high capital costs that appear to be associated with such systems. Unless such cost reductions can be realized, the incremental capabilities of high-temperature direct heat systems may not be warranted except in specialized circumstances.

3.3.6.2. Hydrogen Production by Closed Loop Thermochemical Process. Extensive development work is proceeding in the United States and abroad on hydrogen processes that utilize a series of chemical steps to separate water into hydrogen and oxygen by thermal means. Most of these processes need a high-temperature heat input step, which often is the decomposition of sulfuric acid. The sulfur-iodine process for thermochemical watersplitting is under development at General Atomic (Refs. 3-33, 3-34). A bench-scale unit demonstrating all the major steps in this process has been constructed and operated.

Watersplitting methods, when coupled to a nuclear heat source, provide a way of more than doubling synfuel production from a fixed quantity of fossil feedstock. These savings occur because none of the fossil source is used to provide heat for the process and no carbon dioxide is produced in the hydrogen production process itself. In the longer term, hydrogen is a likely candidate for replacement of current liquid and gaseous fuels.

3.3.6.3. Steam-Methane Reforming and the Thermochemical Pipeline. The HTGR is unique among nuclear energy systems in that it can operate at temperatures high enough (850° to 1000°C) for efficient steam reforming of methane.

The high-temperature helium coolant is used to drive the reformer to produce hydrogen in the form of syngas and thereby chemically store the nuclear heat from the HTGR. The syngas product can be transported long distances to dispersed process heat users. Using the reverse (methanation) reaction, a closed loop energy system, or thermochemical pipeline (TCP), can be formed to deliver nuclear energy to small dispersed industrial process heat users with methanators added at the user sites. Water and methane are returned from the methanator plants to the High Temperature Gas-Cooled Reactor-Reformer (HTGR-R) plant. This TCP concept is depicted in Fig. 3-33. Furthermore, for an open loop reforming system, the hydrogen in the syngas can be used as a feedstock or as a fuel for a variety of dispersed applications such as production of coal-derived liquids, ammonia, and methanol and the processing of steel. Studies indicate that implementation of the HTGR-R in these types of applications could both increase the supply of and substitute for fluid fuels and thus have a major impact on all global energy systems.

The benefits of such a system have been evaluated and show that the TCP energy system concept has the potential to compete with nuclear electricity and with fossil energy systems such as SNG and local fluidized bed coal combustors for one- and two-shift process heat operations. Energy delivery cost projections show that at distances of approximately 50 km or greater, the TCP may be the lowest-cost system for delivery of energy including direct transmission of nuclear-generated steam. This relationship continues for distances as great as 320 km.

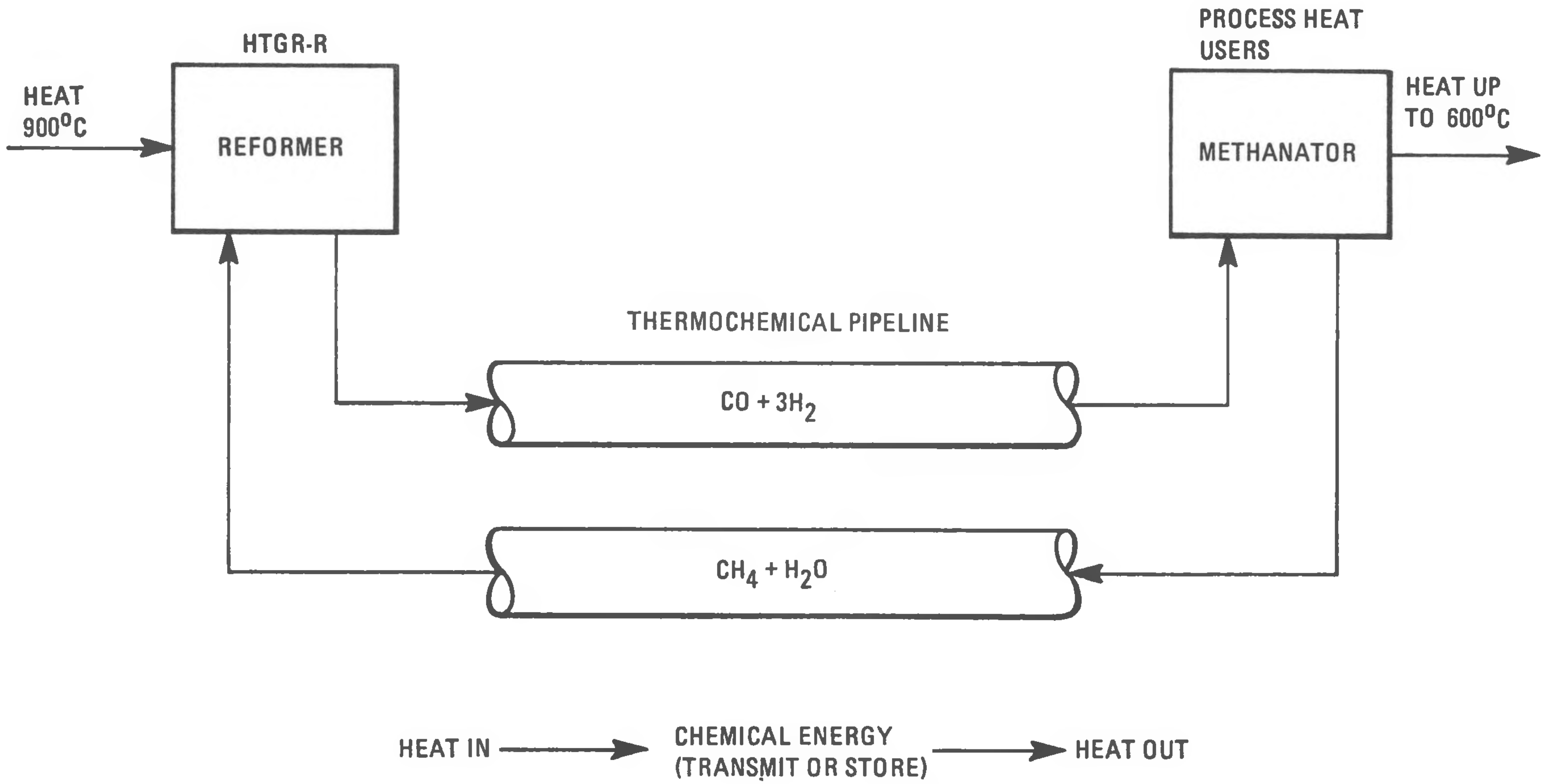

Fig. 3-33. Thermochemical pipeline concept

A second method of energy transport is the use of a sensible heat salt system that is heated to 540°C at the reactor end and transported to the user. The cooled salt is returned to the reactor. This system has the potential for relatively inexpensive energy storage and may be useful for electric power peaking applications (Ref. 3-35). A third system would be a direct steam line to the users if distances are not too long (Ref. 3-31).

The reforming version of the HTGR is a developmental advanced system, and major tradeoffs must be made to select the optimal HTGR-R plant design. Principal issues under consideration are high or low reactor core outlet temperatures and direct or indirect cycle reactor plant configurations.

Since reforming activity decreases below 600°C and the rate of reforming increases with elevated temperature, there is an incentive to use a higher core outlet temperature to achieve better plant performance. However, the higher temperatures usually require more expensive materials and additional technology development. The selection of the direct or indirect steam reforming configuration depends upon economical and safety/licensing considerations. For the indirect cycle configuration, an IHX is located within the PCRV and secondary helium is piped to the reformers and the steam generators located outside the containment building. The direct cycle configuration eliminates the secondary helium loops, with both the reformers and steam generators located within the PCRV.

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4. SPECIAL FEATURES OF GAS-COOLED REACTORS

This section of the report will be devoted to other aspects of GCRs, such as their technical performance, fuel cycles, safety, and environmental impact.

4.1. TECHNICAL PERFORMANCE OF FORT ST. VRAIN HTGR*

The Fort St. Vrain Nuclear Generating Station, one of only two HTGRs operating in the world today, reached 100% power on November 6, 1981. The operators conducted full-power tests until November 9, when the plant was shut down for scheduled maintenance and modification of the helium circulator buffer system. All HTGR systems and components performed at or near design conditions, and there is every indication that the plant can continue to operate at full load.

As mentioned in Section 2, construction of this 330-MW(e) HTGR started in September 1968 at a site 35 miles north of Denver, Colorado. A full-power operating license was subsequently granted by the U.S. Nuclear Regulatory Commission (NRC) in December 1973. Since that time the plant has been undergoing start-up testing, modification, or operation for power generation. It has generated over 2700 GW-hr of electric energy since December 1976.

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*Ref. 4-1.

Fort St. Vrain embodies several advanced features that are unique in U.S. reactor construction: (1) a PCRV containing the entire primary coolant system; (2) an all-ceramic core of hexagonal graphite block fuel elements and reflectors; (3) once-through steam generators producing 538°C main steam at a pressure of 16.9 MPa and 538°C reheat; and (4) steam-turbine-driven, axial helium circulators. Primarily because of the prototype nature of many of its components and systems, the start-up and initial operation of Fort St. Vrain proved difficult at times, and some delays were encountered in bringing the plant to full power. Most of the problems causing major delays were corrected with appropriate modifications to the original systems and components.

One unexpected phenomenon, core outlet temperature fluctuations, occurred during the plant's rise above 50% power. Small temperature changes over a period of 5 to 20 min were observed on instruments reading core outlet, steam generator module inlet, and main steam temperatures. The most probable explanation is small movements of fuel elements and reflector columns, which were probably induced by a combination of non-uniform fuel column temperatures and pressure differences in the gaps between blocks and columns. It was primarily because of this temperature fluctuation phenomenon that the NRC maintained a 70% reactor power limit on Fort St. Vrain. After much analysis, modeling, and testing, region constraint devices were added to the top of the core in 1979 to restrict these fuel element movements and thus reduce or eliminate the temperature fluctuations. The core power limit was lifted for testing purposes, and during operation up to 100% power, the fluctuation phenomenon was no longer observed. At higher power levels (i.e., 83%-86%), however, the core does experience a slight redistribution, resulting in small changes in the core outlet coolant temperature. This redistribution is well within permissible operating parameters.

Testing and evaluation of this demonstration plant's systems have been an ongoing process since initial construction. Inspection and evaluation of the reactor components indicate that the graphite fuel blocks have retained their structural integrity through many plant evolutions, and the fuel blocks were found to be in excellent condition when examined during two refueling operations. The fuel particle fission product barriers (multiple carbon and silicon carbide coatings) have maintained circulating radioactivity in the primary loop at about 1% of the permitted level.

The PCRV, which houses the entire nuclear system, has been problem-free since the plant began producing power in 1976.

As expected with most demonstration plants, there have been a number of shutdowns to repair or modify equipment. Most of 1975 was devoted to rerouting electric cables found to be in non-compliance with safety separation and segregation criteria. This work was accompanied by the design and installation of upgraded fire detection and protection measures newly required by the U.S. Nuclear Regulatory Commission (NRC) in the aftermath of the cable fire in the Tennessee Valley Authority's nuclear plant at Brown's Ferry.

The first-of-a-kind four main helium circulators have required several major modifications, primarily to their auxiliaries. Three circulators were removed and replaced because of failure of a shutdown seal or of a small pressure pipe for actuating the seal. Major plant delays were required during preoperational and low-power physics testing to resolve metallurgical and cavitation problems associated with the water-driven Pelton wheel backup circulator drive. A subsequent NRC-required circulator inspection in the spring of 1979 did not disclose any additional difficulties. At 100% power, the circulators achieved more than 100% of their design flow of 1.5 Gg/hr of helium.

A steam generator tube leak occurred in November 1977, shortly after the plant reached 60% power. This incident demonstrated the capability of the reactor moisture-monitoring equipment to detect small amounts of moisture. However, this small leak did not require automatic protective system action. After an orderly plant shutdown, the leak was located and the failed tube was plugged. No access was required to the PCRV, and the plant was back at power in 44 days with no radiation exposure to workers. This has been the only steam generator leak through February 1982.

The reactor was shut down for two regularly scheduled refueling outages from February through April 1979 and May through July 1981. Fort St. Vrain refueling requires replacing six or seven fuel regions (~1/6 of the core) with fresh fuel every 200 effective full-power days for the first two cycles and every 300 effective full-power days thereafter. During each of the refueling shutdowns, one helium circulator was replaced and the main steam turbine was partially disassembled for inspection and preventive maintenance activities. The fuel handling equipment developed especially for Fort

St. Vrain experienced minor operational and maintenance difficulties but on the whole performed well in handling over 800 fuel and reflector blocks.

The inherent safety of the HTGR was brought into focus on a number of occasions, primarily during the rise-to-power testing program when all forced circulation of the helium coolant was lost. The operators returned circulation to the core in all cases within 20 min with no fuel degradation. The high heat capacity of the graphite makes core temperatures change very slowly under accident conditions. If forced circulation to the core is terminated at 100% power, operators have up to 5 hr to recover it. After 2 hr there would be concern about other components in the reactor system, and the core would then be cooled by radiative heat removal using the PCRV liner cooling system following depressurization of the primary coolant.

Following the full-power run, the plant was shut down for a 4-month scheduled outage to modify the buffer helium system associated with the four main helium circulators. Interaction among the circulators through the buffer system has been a cause of circulator trips, resulting in lowered plant availability and capacity factor. It is expected that separation of the buffer helium systems will improve plant performance.

Fort St. Vrain is fulfilling its role of demonstrating the basic performance and safety of the HTGR concept. The 38.5% thermal efficiency reached at full power is the highest of any nuclear power plant in the United States. The achievement of full power at Fort St. Vrain is an important milestone in development of HTGR technology and confirms that this advanced nuclear power system can play an important role in the world's future energy supply. The lessons learned during the extended rise to power have pointed the way to improved designs now being detailed for the next large HTGR.

4.2. FUEL UTILIZATION

4.2.1. Low-Enriched Uranium Cycle for the HTGR*

All HTGRs built or designed to date utilize a uranium-thorium fuel cycle (HEU/Th) in which fully enriched uranium (93% U-235) is the initial fuel and thorium is the fertile material. The U-233 produced from the thorium is recycled in subsequent loadings to reduce U-235 makeup requirements. This fuel cycle will be discussed later in this section.

The HEU/Th cycle has the best economics and resource utilization of the various fuel cycles. However, the high-enriched uranium (HEU) used to fuel the thorium cycle is a concern in view of potential nuclear weapons proliferation, and this concern has prompted the consideration of alternative fuels in the HTGR. Generally speaking, these alternative cycles use uranium at 20%-or-lower U-235 enrichment (designated LEU for ~10% and LEU/Th for ~20% enriched, respectively) and are designed to minimize the availability of the fissile material for diversion at any point in the cycle.

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*Ref. 4-2.

Studies show that the flexibility afforded by the HTGR coated-particle fuel design allows a variety of alternative cycles, each having special advantages and attractions under different circumstances. Moreover, these alternative cycles can all use the same fuel block, core layout, control scheme, and basic fuel zoning concept. They can also be designed to fit within the HEU core performance envelope so that the effect on plant operation is minimized. This means that HTGR conversion from one fuel cycle to another can, in principle, be effected on a normal refueling schedule by reloading each segment with fuel blocks containing the alternate fuel until a complete changeover is accomplished. Plant operation before, during, and after the conversion remains within the design envelope; i.e., the power, efficiency, and reactivity control margins are unchanged. This point has been confirmed with detailed design studies on conversion of the Fort St. Vrain HTGR core from HEU to 20%-enriched uranium-thorium fuel (LEU/Th).

The HTGR core is powered by fuel rods composed of small coated particles mixed and bonded together in a graphite matrix. These rods are inserted into hexagonal graphite blocks, several thousand rods to a block, to form a fuel element. The particles are essentially kernels of fuel coated with high-strength carbon. The kernels can be oxides or carbides of uranium, plutonium, or thorium, and the uranium can be of any enrichment. The HEU/Th cycle uses two kinds of fuel particles: (1) 93%-enriched UO_2 or UC_2 particles and (2) ThO_2 particles. The fuel is illustrated in Fig. 4-1.

The basic fuel cycle alternatives to the HEU cycle in an HTGR are shown in Fig. 4-2. Essentially, these include the LEU cycle at approximately 10%

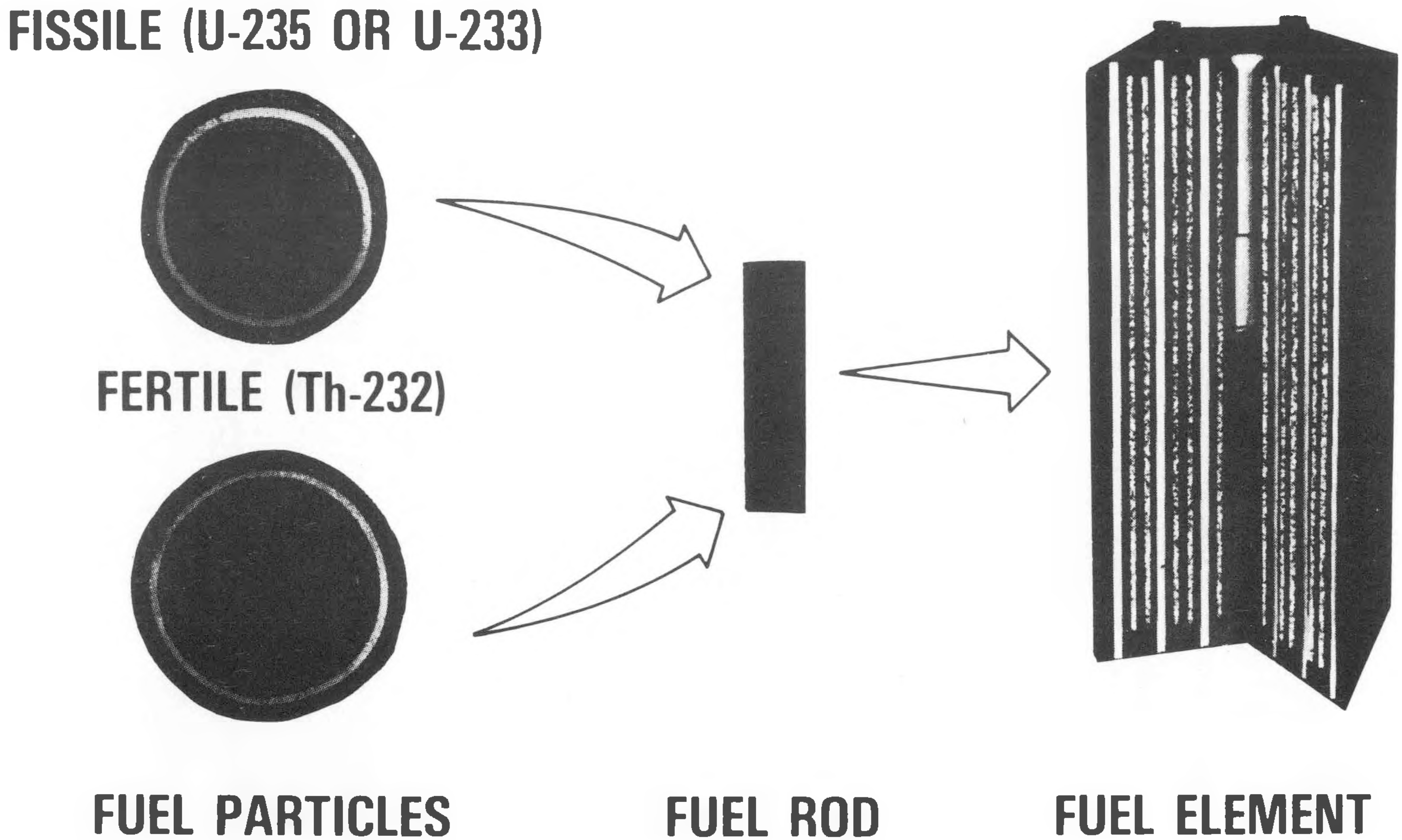

Fig. 4-1. HTGR fuel components

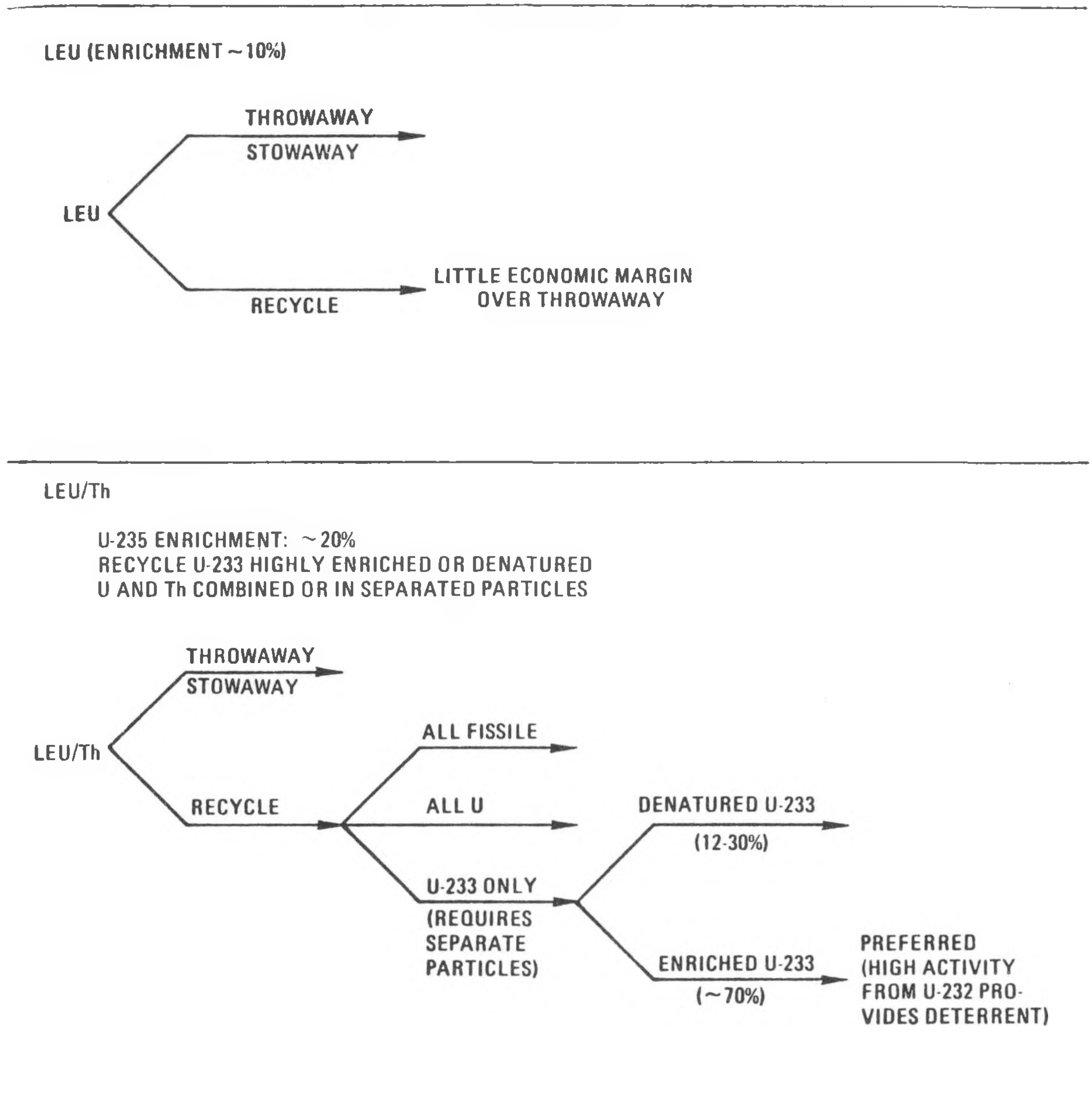

Fig. 4-2. Nonproliferation fuel cycles for HTGRs

enrichment, rather than the 3% used in LWRs, and LEU/Th cycles at just under 20% enrichment. Variations on these cycles involve once-through cycles (throwaway or stowaway) or recycle of all or part of the discharged fissile material. The bred material, mainly U-233, can either be denatured in situ by combining thorium and uranium in a single particle or produced in separate thorium-only particles where the high gamma activity of the associated bred U-232 provides a natural deterrent to illegal diversion.

From the standpoint of core neutronics and resource utilization, HEU/Th is the most efficient fuel cycle. High conversion ratios are possible with the graphite moderator because of the low parasitic neutron absorption. The bred fuel, U-233, is the best thermal neutron fuel, with a neutron-produced-per-neutron capture about 20% greater than Pu-239. Furthermore, U-233 has nuclear properties much closer to U-235 than does Pu-239. This means that burnup changes are minimized and fuel zoning and power shaping are simplified. However, in view of the licensing and potential political concerns associated with the handling of highly enriched uranium, in addition to the development costs required, the decision has been made to use LEU/Th in the once-through-stowaway cycle in planning for the lead HTGR plant.

The LEU/Th systems are the second most efficient cycles for HTGR use (depending on the availability of U-233 recycle), and LEU or uranium/plutonium fuel cycles are the least efficient. Figure 4-3 graphically compares the U_3O_8 requirements versus plutonium discharge rates for an LWR and various HTGR fuel cycles. The plutonium discharge rates are included

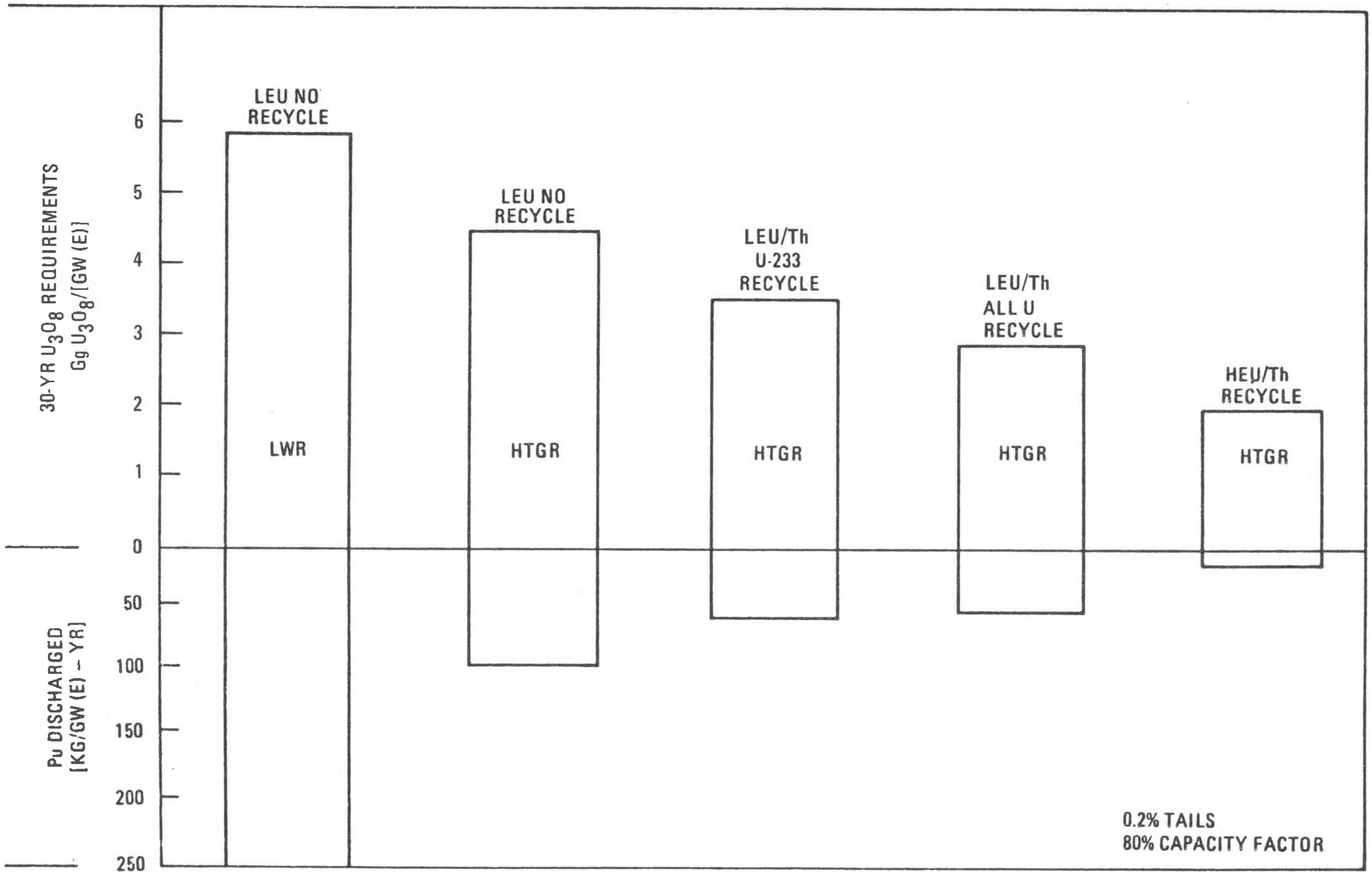

Fig. 4-3. U_3O_8 requirements and plutonium discharge rates

because of proliferation concerns. These rates roughly demonstrate the problems inherent in making a given cycle diversion-proof.

4.2.2. Uranium/Thorium Cycle

Since the most attractive fuel cycle for the HTGR is the thorium/uranium cycle, it is the only fuel cycle on which comprehensive development work has been done (primarily at Oak Ridge National Laboratory and General Atomic). The technology can, however, be readily applied to the LEU cycle or to plutonium utilization. Developments in the United States of process technology and equipment for the thorium/uranium fuel cycle are described below (Ref. 4-3).

4.2.2.1. Fuel Cycle Operation. Figure 4-4 shows the principal operations and nuclear material flow in the Th-233/U fuel cycle for the HTGR.

The Th-233/U fuel cycle is complicated by the fact that U-232 is also generated in HTGRs, and the U-232 has a relatively short half-life, decaying to Th-228 and, in a series of short-lived intermediates, to stable Pb-208. The most significant of the intermediates are Rn-220, which, being a gas, can be transported through filters, and Tl-208 and Bi-212, both of which emit energetic gamma rays. The gamma activity associated with these intermediates must be accommodated in the refabrication facility, since it is not possible to separate the U-233 chemically from U-232. Although the decay

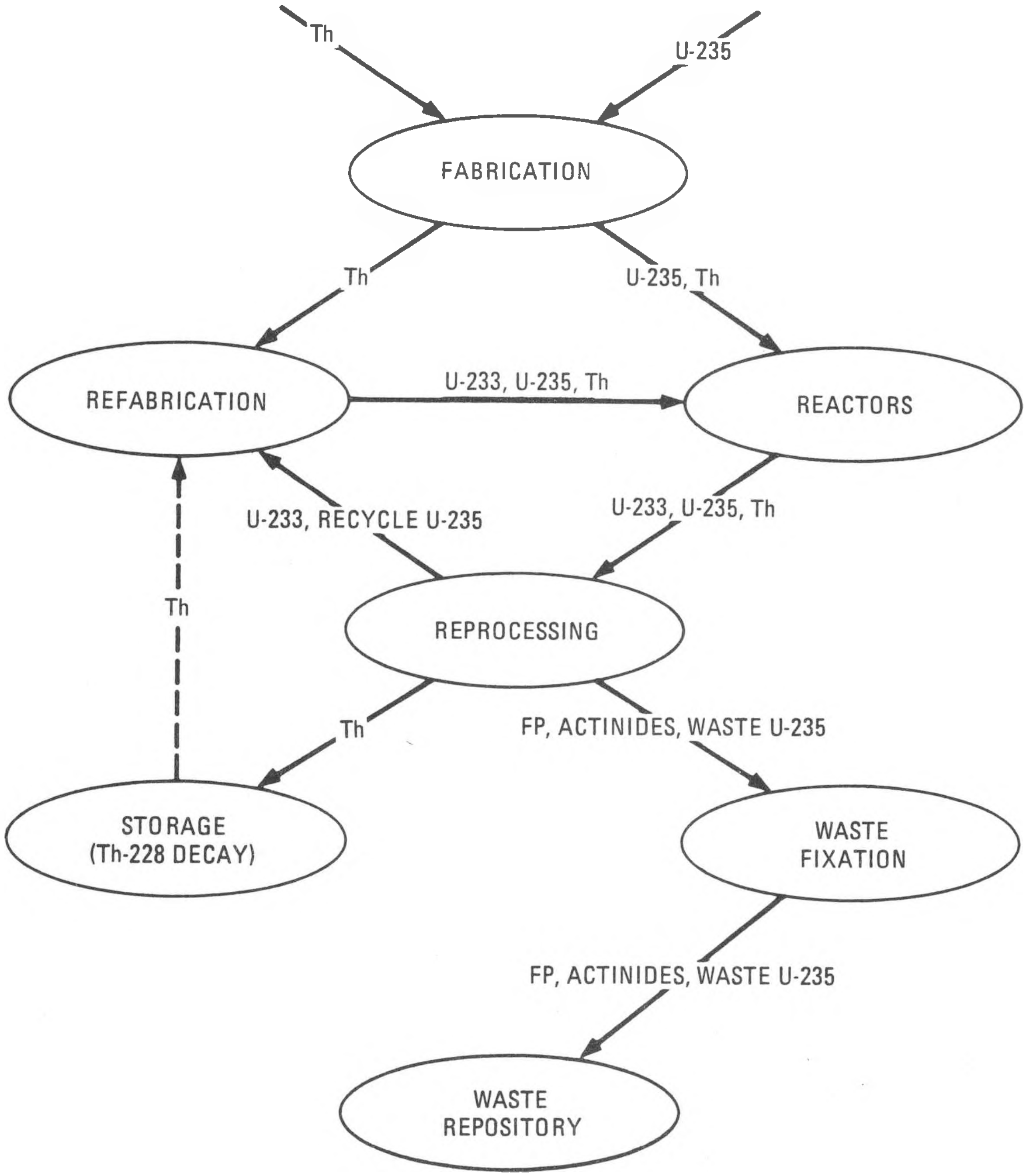

Fig. 4-4. Operations and material flow in the HTGR U-233/Th fuel cycle

chain can be broken by use of an ion exchange cleanup, the activity grows back sufficiently fast so that shielded facilities are required for refabrication. In reprocessing, the problems associated with U-232 decay product activity are obscured by the activity of the fission products that must be handled.

Other activities of key importance in the fuel cycle are shipping, fabrication of fresh fuel, and finally waste isolation in a repository. After fixation of the waste, the problems in waste isolation are very similar to those encountered in the fuel cycles for other reactors.

4.2.2.2. Process and Facility Requirements. Any fuel cycle development program must be directed at solving the technical problems associated with a commercial-size fuel recycle facility. It appears that some small advantages of scale can be realized in facilities having a range of capacity from 20,000 to 50,000 prismatic fuel elements per year, based on reprocessing load. Figure 4-5 summarizes the principal flows to and from such a central HTGR recycle facility. Such a facility could provide the recycle needs for approximately 20-GW(e) installed HTGR capacity. In the facility shown in Fig. 4-5, 20,000 prismatic fuel elements per year are reprocessed and, including the assumption of once-through recycle of U-235, 10,000 fuel elements per year are refabricated. Means would need to be provided for consolidation and isolation of the various waste streams. Spherical fuel elements could be handled by only slight modification of the reprocessing flowsheet and by changing the fuel element fabrication equipment.

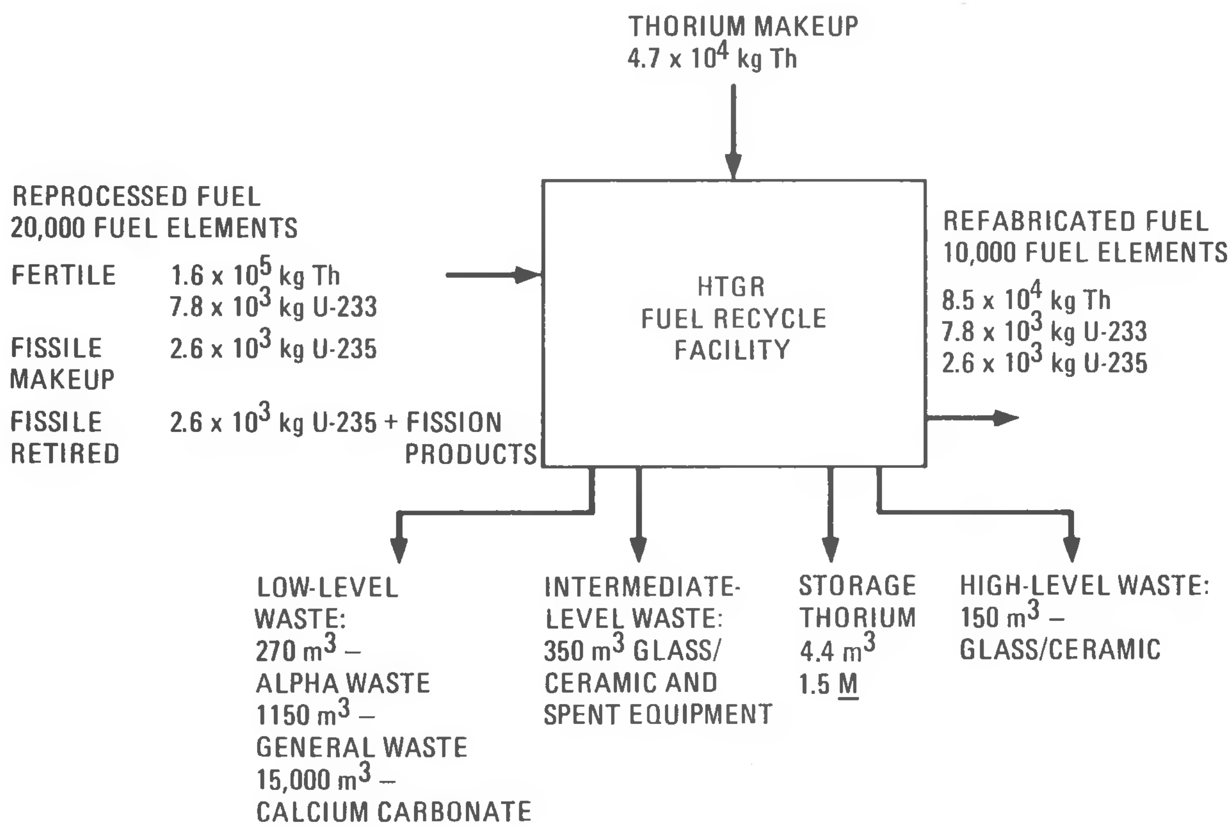

Fig. 4-5. Nominal annual material flow for a central HTGR fuel recycle facility

In reprocessing, spent fuel elements are removed from storage at the facility and are first prepared for burning by crushing (primary burner feed preparation) or by milling. Primary burning eliminates most of the moderator and the outer coatings of particles. Particles are then classified by size to separate U-235 from the thorium and U-233. Particles having silicon carbide coatings and intended for recycle are then crushed and burned in a secondary burner. The uranium-thorium oxides are then dissolved and processed by solvent extraction. Large-scale operations are involved in the reprocessing flowsheet to handle the off-gases and the liquid and solid waste.

In refabrication (which must be done remotely because of the U-232), U-233 is introduced from reprocessing storage. It is decontaminated by ion exchange if necessary, made into high-density UCO particles by a sol-gel technique, and then coated with various layers of pyrolytic carbon and silicon carbide. Following particle coating, the fissile particles are mixed with coated fertile particles prepared in direct manual contact facilities and fed to the fuel rod fabrication step, where the particles are bonded together with a carbonaceous matrix. The fuel rods are placed into a premachined graphite fuel block, and the complete assembly is then cured in place at high temperature. Substantial operations in the refabrication flowsheet are involved with scrap treatment and waste treatment for off-gases, liquids, and solids.

The operations for fresh fuel (U-235 and ThO_2) fabrication are similar to those for refabrication, except that they are performed in contact facilities.

4.2.3. GCFR Fuel Cycles

At the present time, all of the plutonium generated in LWRs in the United States is stored as spent fuel. If nuclear power is to fulfill its potential as a long-time source of energy, this fissionable material must be used as efficiently as possible. The GCFR will do exactly that because of its very high breeding ratio and potential doubling times of under 10 yr. The breeding ratio stems from the high average neutron spectrum in the GCFR, which results from the essentially negligible moderation of the fission neutrons by the helium coolant.

The two fuel cycles that have been explored most extensively for the GCFR both use a core of mixed plutonium and depleted uranium oxides. They differ in the use of depleted uranium oxide as the fertile material in the blankets in one case and thorium oxide in the other. The latter case will be discussed briefly in Section 4.2.4.

A scoping study, using present PuO_2/UO_2 fuel, was performed to determine the effect of primary system pressure and helium circulator power on breeding ratio, reactor inventory, and compound system doubling time (CSDT) (Ref. 4-4). The reactors studied all had the common characteristics listed in Table 4-1. The results of the study are shown in Fig. 4-6. Both

TABLE 4-1

COMMON CHARACTERISTICS OF GCFR PLANTS IN SCOPING STUDY

Power	3600 MW(t)
Fuel Cladding and Core Structural Material	316 stainless steel
Load Factor	0.75
Out-of-Pile Time	1 yr
Reprocessing Loss	2%
Burnup (Max.)	100 MWd/kg
Fuel Rod Diameter	8 mm
Core Outlet Temperature	566°C
Maximum Mid-Clad Temperature	750°C

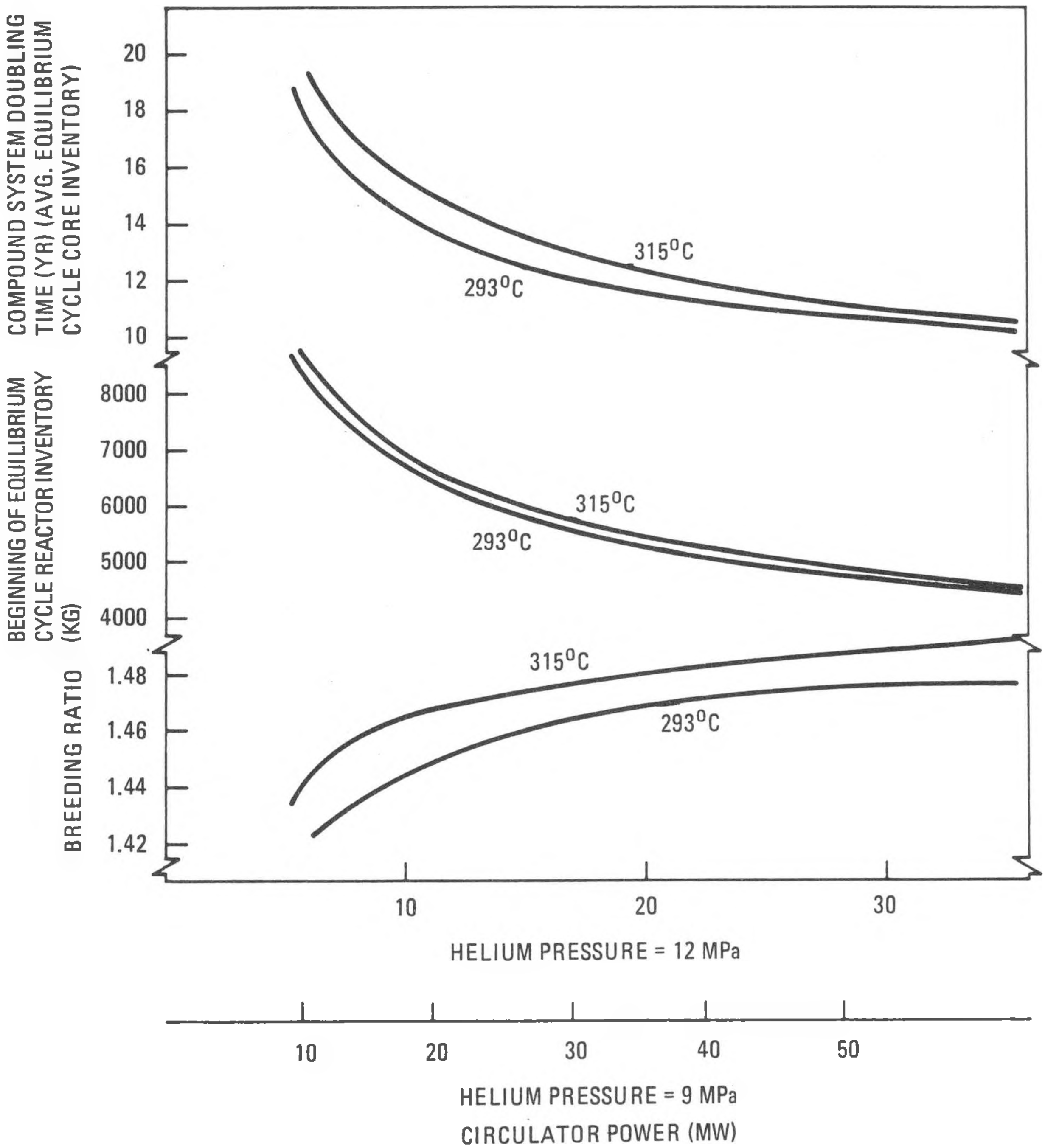

Fig. 4-6. Reactor core performance versus circulator power at different system pressures and inlet temperatures

system pressures can be achieved within present technology and structural codes. The data base for circulator driver motor power levels above 15 MW is limited, which would probably reduce plant reliability and increase the technological risk in early plants with larger motors.

Another scoping study compared plants using present oxide fuel and cladding and more advanced oxide and carbide fuel with improved cladding. Table 4-2 summarizes the core design parameters and Table 4-3 the results of the study.

All of the plants have the following common characteristics: 3600-MW(t) reactor power and 33% thermal efficiency, six loops, Pu/U-238 cores and U-238 blankets, three rows of blanket assemblies and 600-mm-long axial blankets, 8-mm-diameter fuel rods, 750°C maximum mid-clad temperature, and 19 dedicated control rod assemblies.

Comparatively low-power helium circulators (16.4 MW nominal) reduce the technological risk associated with the early (present oxide fuel) plants. The fuel clad and assembly duct material is 316 stainless steel.

The advanced oxide and advanced carbide core designs were developed based on increased primary pressure (12 MPa) and high circulator power (33.6 MW nominal) and utilization of the advanced alloy HT-9 for the clad and duct material. The increase in pressure and circulator power results in compact fuel rod lattice and high fuel volume fractions of about 0.37 and 0.34 in the advanced oxide and advanced carbide designs, respectively. The

TABLE 4-2

SUMMARY OF CORE DESIGN PARAMETERS FOR GCFR SCOPING STUDY

	Fuel		
	Present Oxide	Advanced Oxide	Advanced Carbide
Core			
Reactor Thermal Power [MW(t)]	3600	3600	3600
Reactor Electric Power [MW(e)]	1200	1200	1200
Primary System Pressure (MPa)	10	12	12
Peak Linear Rating (kW/m)	35.1	49.2	65.6
Core Inlet Temperature (°C)	311	308	307
Core Outlet Temperature (°C)	564	562	534
Number of Fuel Assemblies	450	378	312
Core Height (mm)	1210	1063	964
Core Diameter (cylinder) (mm)	4626	3868	3694
Core ΔP (kPa)	3.73	9.17	9.17
Fuel Cycle Length (full-power days)	364	251	231
Fuel Assembly			
Number of Rods per Assembly	264	271	271
Assembly Pitch (mm)	203	185	193
Interassembly Gap Spacing (mm)	12.9	6.5	6.5
Spacer Separation (mm)	200	300	300
Volume Fractions Inside Middle of Assembly Gap			
Fuel	0.300	0.372	0.341
Structure	0.146	0.144	0.134
Helium	0.554	0.483	0.510
Fuel Rod			
Clad Material	316 SS	HT-9	HT-9
Outside Diameter (mm)	8.0	8.0	8.0
Pitch (mm)	11.1	10.3	10.8
Smear Density (volume fraction inside clad)	0.859	0.900	0.900

TABLE 4-3

NUCLEAR PERFORMANCE FOR SEVERAL GCFR FUELS

	Fuel		
	Present Oxide	Advanced Oxide	Advanced Carbide
Fissile Inventory (kg)	6074	4285	3875
Cycle Length (0.75 load factor) (yr)	1.33	0.92	0.84
Fissile Gain/Cycle (kg)	496	489	550
Breeding Ratio	1.38	1.57	1.75
Compound System Doubling Time (yr)	15.9	8.3	6.2

amount of structure in the advanced cores is minimized by using thin (3-mm) fuel assembly ducts and increasing the separation of fuel rod spacer grids to 300 mm.

Even with present oxide fuel and an open core lattice (low fuel volume fraction), this plant attains a CSDT of 16.0 yr. The doubling time could be lowered by about 1.5 yr by replacing the 316 stainless steel clad and duct with HT-9 and increasing the primary system pressure by 1 MPa to 11 MPa.

The benefits of the high fuel volume fraction attainable with increased circulator power and primary system pressure are evidenced by the performance of both of the advanced systems. The performance of the advanced carbide system is somewhat better than that of the advanced oxide system. The higher heavy metal density attainable in the carbide fuel results in a harder spectrum yielding a larger neutron production term. The carbide system neutron production terms are about 5% to 7% higher than those of the oxide system. Fertile atom fission and neutron leakage is increased, and absorption in structure and fission products, etc., is reduced. The breeding ratio is 1.75 and the CSDT is 6.2 yr for the Pu/U core with U-238 blankets. This is a remarkably high breeding ratio and low doubling time and augurs well for the symbiotic systems discussed below.

Since the GCFR fuel and blanket assemblies are similar to those for the LMFBR, the rest of the fuel cycle (fabrication, reprocessing, refabrication, waste handling, and storage) is essentially the same as for the latter concept.

4.2.4. Symbiotic Systems

An alternative fuel cycle for the GCFR is the use of thorium in the fertile blankets, which generates U-233. The U-233 is an excellent fuel for thermal spectrum reactors, particularly advanced converters (ACRs) such as the HTGR. This symbiosis between ACRs and FBRs has been studied at General Atomic for several years (Refs. 4-5, 4-6). Recent analyses have taken into account the large uncertainty in nuclear projections that exist today (Ref. 4-7). Although several growth projections have been studied, only the low projection will be discussed here.* This projection is based on the U.S. Department of Energy 1979 Low Growth Projection extrapolated from the year 2000 to 2040 with a constant value subsequently. The value reaches 235 GW(e) in the year 2000 and 290 GW(e) in 2040. Added nuclear capacity for non-electric heat applications is omitted to simplify the computer model, although these applications are likely to be important during the time period considered.

The current LWR once-through cycle is assumed to be the only cycle until 1990. The high-burnup LWR is introduced in 1990, the ACR in 2001, and the FBR in 2006.

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* Present projections are even lower. This would affect both the date and the rate of introduction of the ACRs and breeders. However, the concept of the symbiotic relationship and its long-term benefits is still valid if it is assumed that there is an appreciable cost penalty for FBRs over ACRs under equilibrium conditions.

The improvement to the LWR once-through fuel cycle is limited to the use of high-burnup fuel (51 MWd/kg) that is retrofittable by 1990. The improved fuel leads to a 13% reduction in annual uranium requirements.

Advanced converter reactors are available for operation by the year 2001 on the low-enriched U-235 cycle. The ACR may be the HTGR or an improved Advanced Converter LWR. After 2010, highly enriched U-233 is acceptable to fuel the ACRs.

Fast breeder reactors, liquid metal (LMFBR) or gas-cooled (GCFR), are available for operation in secured areas by 2006. The FBRs are introduced gradually, with the rate of introduction dependent on the particular nuclear demand model used. Eventually, all breeders in the system would be self-sufficient in plutonium (Pu/U core-Th blankets); at equilibrium the system can produce U-233 for the ACRs at the ratio of one FBR to three ACRs.

All the reactors are assumed to operate at 70% capacity factor. Uranium tails enrichment is 0.15%.

Spent fuel reprocessing is assumed to start in 1991, with 50% of available stored spent fuel reprocessed by 1995, 75% in 2000, and 100% in 2005. The recovered bred fuel is recycled within 2 yr of the time discharged, after 1991.

The value of uranium yellow cake (U_3O_8) is assumed to be \$99/kg during the period 1980 to 2000, then to increase linearly to \$161/kg in 2010,

$264/kg in 2020, and $441/kg in 2040. The value of uranium enrichment is assumed to be $100/separative work unit over the entire period.

A series of calculations was performed to determine the number of reactors of each design type that could be installed while meeting two important goals:

1. Minimize the annual production requirements for U_3O_8.
2. Minimize the total cost of generating electricity.

For the DOE low demand case (Fig. 4-7), a sharp reduction in the growth rate occurs after the year 2000. The LWRs, which dominate the system until about 2020, comprise 30% of the total reactors installed (during the total period evaluated). This implies that a fairly large amount of plutonium is available for the breeders. Hence, a greater portion of the breeders installed could be designed with a U/Pu core and thorium blankets which would keep the number of FBRs small and at the same time maintain a large U-233 inventory to allow the introduction of a substantial number of U-233-consuming ACRs.

Eventually only FBRs that are self-sufficient in plutonium and ACRs operating on the Th/U-233 cycle in a ratio of about 4 to 11 are built in this low-growth scenario. This scenario could continue for many centuries. To improve this ratio of FBRs to ACRs, a high breeding ratio is necessary. The breeders in this calculation have a capital cost of 1.25 times that of LWRs. If the capital cost could be reduced, the breeders would be

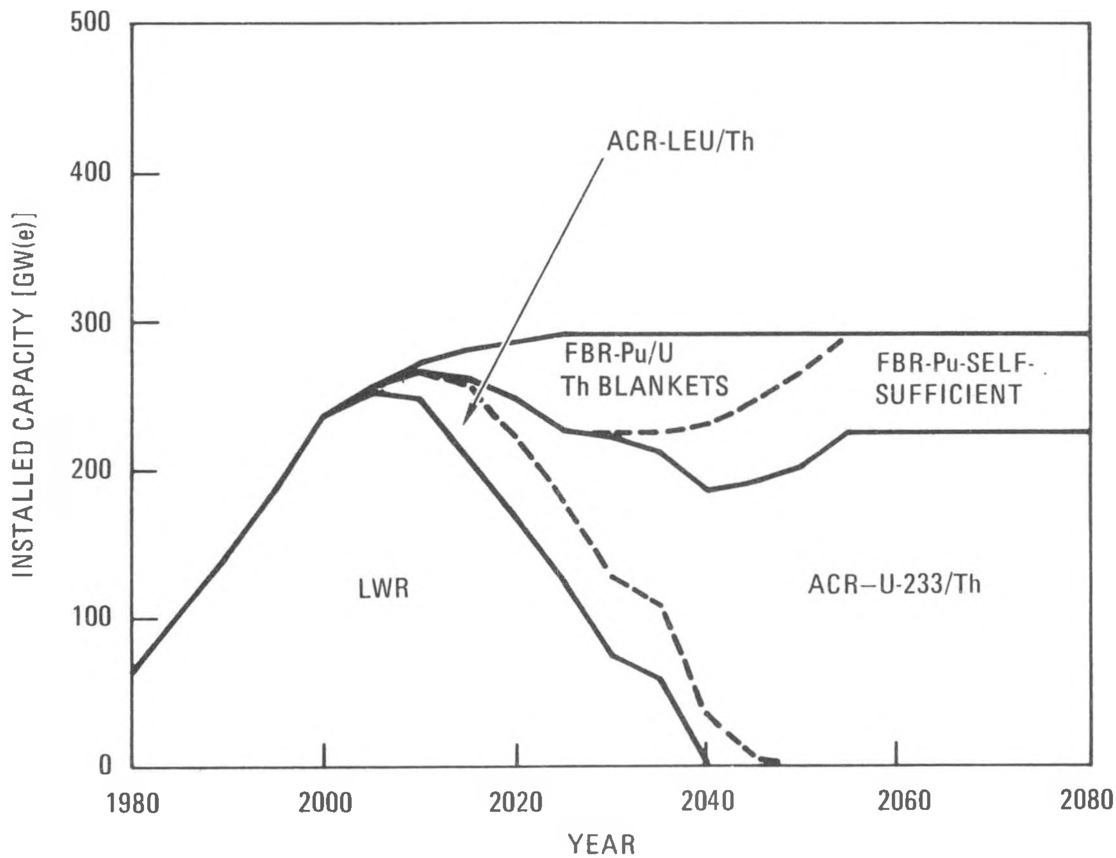

Fig. 4-7. Possible reactor scenario based on DOE 1979 low growth projection

introduced earlier and the ultimate fraction of breeders in the total would be larger. Improving the breeding ratio of the FBRs would allow a larger number of ACRs to be supplied with U-233 from a single FBR.

Using advanced oxide fuel, the GCFR with thorium blankets would have a breeding ratio of 1.54 (versus 1.57 for U-238) and a CSDT of 9.2 yr (versus 8.3 yr for U-238).

4.3. SAFETY CHARACTERISTICS

4.3.1. HTGR

The HTGR concept has inherent and passive safety features (Ref. 4-8) that make gas-cooled reactors a low risk to the public and to the operating personnel. These features are the first lines of defense in maintaining the integrity of the barriers against the release of radioactivity. An additional characteristic of critical importance is that the consequences of HTGR accidents develop slowly, allowing time for the operators to take deliberate and planned actions.

Inherent features that contribute to overall HTGR safety are (1) coated-particle ceramic fuel that releases fission products slowly even under extreme temperatures, (2) a graphite moderator/reflector that responds slowly to transient temperature while maintaining structural integrity at very high temperatures, and (3) helium coolant that remains a single-phase gas under all conditions and is neutronically transparent and chemically inert. A passive feature of safety significance is the massive PCRV, which contains a multiplicity of load-bearing tendons for structural integrity and provides effective radiation shielding and retention of fission products. Engineered features that contribute to low probability of serious accidents and low consequences, should accidents occur, include (1) the triply redundant CACS, which is used when main loop cooling is unavailable, (2) the reserve shutdown system, which can bring the reactor to cold shutdown in the absence of control rod insertion, and (3) the plant protection system, which initiates various safety actions, such as reactor trip, steam generator isolation and dump, and CACS start-up. Table 4-4 summarizes the key safety features inherent to the HTGR concept; these features are described below.

4.3.1.1. Coated-Particle Ceramic Fuel. HTGR fuel consists of enriched-uranium oxycarbide particles and thorium-oxide particles coated to retain fission products. The spheroidal fuel particles are coated with multiple layers of pyrolytic carbon that act as tiny independent pressure vessels that contain the fission products. Also, the fuel particles have a coating of silicon carbide, which provides additional containment of metallic fission products. These particles remain intact and retain fission products up to about 2000°C. Forced core cooling would have to be totally interrupted for approximately 3 hr before any fuel damage would occur and for approximately 20 hr before 50% of the core radioactivity would be released. Thus, time is provided for fission product decay and for the operator to take mitigating actions.

TABLE 4-4
KEY INHERENT AND PASSIVE SAFETY CHARACTERISTICS OF THE HTGR

Inherent or Passive Feature	Relevant Properties	Safety Significance
Reactor Core	High heat capacity	Slow transient response
	Low power density	Allows adequate time for remedial measures, both within and external to plant
	Strong negative temperature coefficient	Fast-acting shutdown system not required
	Graphite cannot melt, but may locally sublime	Structural integrity of core maintained for days following loss of cooling
Fuel	Coated particle ceramic fuel	Slow release of volatile nuclides under no-cooling conditions
Helium Coolant	Single-phase gas	No boiling, bubbbles, liquid level, or pump cavitation problem; no added coolant inventory needed for core cooling, only forced circulation
	Neutronically transparent	Negligible reactivity effects
	Chemically inert	No chemical fuel cladding-coolant interactions
	Low stored energy	Reduced containment damage potential
PCRV	Multiplicity of tendons	Failure of individual structural members inconsequential
	Tendons shielded by concrete	Neutron embrittlement and subsequent fracture eliminated
	Concrete under compression	Cracks self-sealing, do not propagate
	Massive structure	Effective retention of radioactivity; retains great fraction of heat escaping core

4.3.1.2. Graphite Moderator and Reflector. The core and reflector structure is composed of graphite, a material that sublimes at about 3800°C and retains good strength to above 2500°C. The structure for a 2240-MW(t) HTGR weighs almost 1.4 Gg, and its associated heat capacity, high-temperature capability, and low power density ensure that reactor temperature transients proceed slowly. The slow thermal response provides a "forgiving" reactor, since the behavior of the system is more predictable and more time is available to prevent transients from progressing into major accidents. Time is available to repair equipment, adjust the system, or take other corrective action.

In the event of a permanent loss of core cooling capability, the graphite acts as a crucible by maintaining its structural form for weeks with only some local sublimation and with no possibility of melt-down. Volatile radionuclides are released gradually over a period of days, while nonvolatile nuclides are retained. In the majority of accident scenarios, the integrity of the PCRV and containment is maintained indefinitely. In the even lower-frequency accidents resulting in PCRV concrete degradation, containment integrity is maintained for days.

4.3.1.3. Helium Primary Coolant. The primary coolant of the HTGR is helium, a noncondensable gas that obeys a simple linear temperature-pressure relationship. Helium is chemically inert, has low stored-energy content, and remains in the gaseous phase under all conceivable operating conditions. It cannot chemically react with the core, vessel, or containment structures in a destructive way and would cause only a mild pressure transient in the containment as a result of a blow-down. It does not contribute to or affect the nuclear chain reaction. Since heat can be removed from the reactor core with a mixture of gases, even at atmospheric pressure, maintaining an inventory of pure helium in the reactor vessel to prevent overheating of the fuel is not necessary. Hence, coolant injection systems are not needed.

4.3.1.4. PCRV. A large number of load-bearing steel tendons running axially through and circumferentially around the vessel provide the primary safety advantage of the PCRV. The independence and redundancy of these tendons provide a barrier to fault propagation within the vessel. The tendons are shielded from the effects of irradiation by the concrete. The steel liner functions as a non-load-bearing seal that is always held in compression by the surrounding prestressed concrete. This design feature greatly limits possible fault propagation in the liner. In addition, the liner cooling system furnishes an additional heat sink for emergency core cooling.

4.3.1.5. Fort St. Vrain Experience. The safety of the HTGR concept has been demonstrated at Fort St. Vrain (Ref. 4-9). As mentioned in Section 4.1, on a number of occasions all forced circulation of the helium coolant was lost. The operators returned circulation to the core in all cases within 20 min with no fuel degradation. In fact, a loss of forced circulation situation, if corrected within 1 hr, will not result in exceeding any component design limit temperatures. The long time afforded for status evaluation in response to abnormal conditions is demonstrated by the fact that the technical advisors required by the Nuclear Regulatory Commission following the Three Mile Island accident to be onsite for all water-cooled reactors are permitted to be on 1-hr call at Fort St. Vrain.

4.3.2. GCFR

The GCFR shares the safety advantages of helium with the HTGR, i.e., a single-phase coolant that has low stored energy content and is chemically inert. However, being a fast reactor, the GCFR has no moderator and thus only a small heat capacity in the core. In addition, it has a power density of 140 MW/m^3 as compared with 7 MW/m^3 for an HTGR. Thus, not only must large blower power be expended to cool the core during normal operation, but the time to take action during a transient is greatly reduced. Fast reactors, in common with LWRs, use metal cladding on the fuel rods. This means that the temperatures are limited by the properties of the cladding in contrast to the high-temperature capability of the all-ceramic HTGR core.

The single-phase nature of the coolant, the large quantity of graphite, the all-ceramic core, and the massive PCRV are the pillars of the safety advantages of the HTGR. In case of failure of forced circulation, the HTGR has up to an hour before any action needs to be taken to prevent damage to the core or any primary system components.

To achieve a high level of safety in the GCFR, the design has, of necessity, followed a different direction, namely assurance of coolant circulation. Redundant and diverse cooling systems have always been part of the design. In addition, with upward flow of helium through the core, there is the possibility of natural convection cooling to remove residual heat following reactor scram. Carrying this theme further, provision has been made for natural circulation cooling on the secondary side of the heat exchangers and also in the tertiary (natural draft cooling tower) system, which dumps the heat to the atmosphere. The GCFR then becomes, as long as the helium is pressurized, a "walk-away" system after the reactor scrams; i.e., the operators could let the plant continue cooling by itself.

To obtain some thermal lag, a tank with sufficient capacity to boil water for 20 to 30 min was designed into the shutdown cooling system (see Fig. 3-18). Upon shutdown of the main turbine-generator, the steam from the steam generator is routed through a heat exchanger submerged in the water tank. This tank is located 18 m above the steam generators so that it works under natural circulation. The water, which is boiled off and vented to the atmosphere, can be replenished by a make-up pump or, as a backup, via the fire water system.

Taken all together, these various forced and natural circulation systems provide a high degree of reliability, and cooling will always be provided to the core. There is, of course, the question of residual heat removal when the reactor is depressurized for maintenance or refueling. At atmospheric pressure, the core cannot be cooled adequately with natural convection. Therefore, provision has been made for rapid repressurization of the primary cooling system within the PCRV to a pressure sufficient for natural circulation to be effective (about 1 MPa) if the number of forced circulation systems available for cooling drops below a predetermined level.

4.4. ENVIRONMENTAL IMPACT

4.4.1. HTGR

4.4.1.1. Resource Requirements. Because of its higher conversion ratio, the HTGR operating on the HEU/Th fuel cycle uses less uranium than an LWR. As indicated earlier in this section, the HEU/Th fuel cycle is the optimum one for the HTGR. The lifetime fuel requirements of three HTGR fuel cycles are given in Table 4-5 (Ref. 4-10). In the HEU/Th cycle, 2070 Mg/GW(e) of U_3O_8 would be used. In addition to the uranium oxide, HTGRs require about 400 Mg/GW(e) of ThO_2 over their 30-yr lifetime.

4.4.1.2. Heat Rejection. At an efficiency of over 39%, HTGRs are equivalent to modern fossil-fired units in thermal efficiency and hence in rejection of heat to the ambient atmosphere.

The use of direct cycle helium turbines with the HTGR has the potential to reduce the requirements for cooling water because dry cooling towers can be used in place of a cooling pond, wet towers, or once-through systems. An advanced direct cycle system would utilize an organic "bottoming" cycle. If this system can be developed, thermal efficiencies approaching 50% can be realized. The reduced need for cooling water in a direct cycle HTGR may potentially be of considerable environmental (and resource) significance. Some studies have shown that obtaining adequate supplies of cooling water for evaporative use by power plants in the United States will be a problem by the period 1990 to 2000. The potential capability of the HTGR to

TABLE 4-5

FUEL REQUIREMENTS FOR SELECTED HTGR CYCLES[a]

	LEU/Th Once-Through	LEU/Th Recycle	HEU/Th Recycle
Feed-Uranium Enrichment (%)	20	20	93
Core Conversion Ratio	0.54	0.60	0.75
Average Exposure upon Discharge (MWd/kg)	130	96	60
Initial Fissile Inventory [kg/GW(e)]	1350	1350	1870
Annual Net Fissile Makeup [kg/GW(e)]	575	400	205
U_3O_8 Requirements [Mg/GW(e)]			
Initial Core	309	297	461
Equilibrium Annual Makeup	131	91	49
30-yr Total	4095	3015	2070
Enrichment Service Requirements, 10^3 SWU[b]/GW(e)			
Initial Core	309	297	509
Equilibrium Annual Makeup	131	93	54
30-yr Total	4100	3090	2285

[a] Data assume 75% capacity factor, 0.2% U-235 tails assay, 39.6% thermal efficiency.

[b] Separative work units.

alleviate this problem is significant. In addition, the potential siting flexibility in not being dependent on large supplies of cooling water would be an important advantage.

4.4.1.3. Radiological Impact. From the standpoint of radiological impact, the HTGR requires less uranium mining than LWRs. The use of thorium introduces a problem similar to that encountered during uranium mining (Ref. 4-11). Natural thorium has only a single isotope, Th-232, which decays with a half-life of 14 billion years. Included in the decay chain is Rn-220, which is gaseous and diffuses into mine atmospheres, although its short half-life of 55 s indicates that this radon isotope has less chance for escape to the atmosphere than the Rn-222 (3.8-day half-life) in the U-238 series. The daughters of Rn-220 are also relatively short-lived and include some alpha particle emitters, which give the principal dose to the lungs.

Tritium is produced in the HTGR by ternary fission and is mostly retained in the fuel particles and the graphite matrix. Tritium is also produced by neutron activation reactions with helium (He-3) and with lithium and nitrogen impurities. These last two are the main source of tritium in the coolant. It is removed from the coolant by the hydrogen getter (absorber) unit in the helium purification system, through which a fraction of the helium passes instead of going through the reactor. The tritium release to the environment from a large HTGR is expected to be less than 5 Ci/yr.

The krypton and xenon isotopes will also be removed from the coolant in the helium purification system by using a low-temperature absorber delay bed. Rather than being released after regeneration of the low-temperature delay bed (mostly long-lived Kr-85 since the other radionuclides have decayed), the gas can be bottled for offsite disposal, or it can be returned to the primary coolant system where it will be removed again by the low-temperature absorber (Ref. 4-12). By this means, the lifetime inventory of krypton can be retained, with the result that less than 10 Ci of Kr-85 are expected to be released to the environment annually at a large HTGR. Figure 4-8 is a diagram of the helium purification system.

Liquid wastes will be accumulated at the HTGR from decontamination operations or as a result of equipment failure such as a steam generator tube leak. Under normal conditions the liquid radioactive waste is expected to be about 10 Ci/yr. Treatment systems using devices such as those used in LWRs will keep liquid releases as low as practicable.

Compared with the LWR, the HTGR produces a somewhat lower volume of solid wastes to be shipped to a Federal repository or commercial burial grounds. This is mainly owing to the higher thermal efficiency of the reactor, the elimination of cladding hulls, and the relatively high amount of thermal energy extracted per unit mass of fuel material in the HTGR. The burial ground area needed to store the solid wastes from an HTGR has been estimated to be about 70% of the area needed for a similarly sized pressurized water reactor (PWR).

Experience with Fort St. Vrain has confirmed the low levels of radioactive release anticipated. For example, even though significant plant

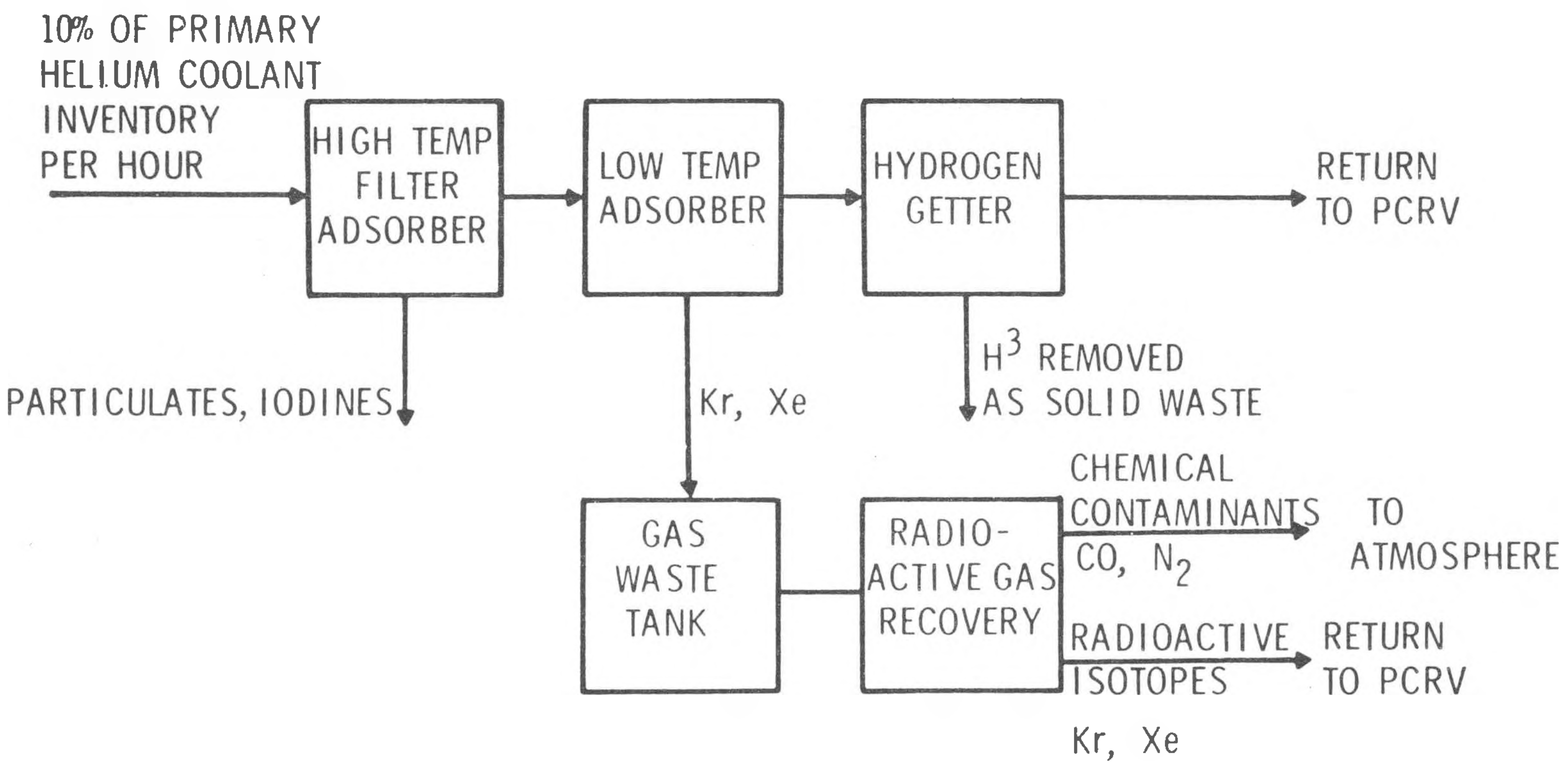

Fig. 4-8. HTGR helium purification system

operation has occurred since 1976, Fort St. Vrain has yet to make its first shipment of low-level radioactive solid waste (Ref. 4-13). The combined total activity of this accumulated waste radioassayed to date is less than 1 Ci. The noble gas airborne effluent released from Fort St. Vrain in 1978 totaled 332 Ci, whereas the average BWR and PWR releases in 1977 were 120,000 and 6600 Ci, respectively (Ref. 4-14).

4.4.1.4. Radiation Exposure. Fort St. Vrain provides the best evidence of one of the major advantages of the HTGR: low radiation exposure of the plant personnel. Table 4-6 gives 1978 and 1980 personnel exposure data for Fort St. Vrain. This extremely low personnel exposure level is even more significant in view of the fact that Fort St. Vrain does not have a containment and virtually unlimited access is provided into the reactor building for all operation, maintenance, and construction personnel.

4.4.2. GCFR

4.4.2.1. Resource Requirements. The GCFR has a core of plutonium and depleted uranium and fertile blankets of depleted uranium or natural thorium. By the time that breeders are required in significant numbers, there will be adequate quantities of plutonium available in the spent fuel from LWRs to start up many breeders. Gigagram quantities of depleted uranium already are in storage (Ref. 4-16) from diffusion plant operations over the past four decades; hence, no uranium mining would probably be required for centuries.

TABLE 4-6

FORT ST. VRAIN (FSV) HTGR PERSONNEL EXPOSURE DATA[a]

<u>Data for 1978</u>

Average Number Personnel Monitored	930
Average Number Personnel with Measurable Doses	34
Percent Personnel with Measurable Doses	3.7
Average Collective Dose (person-rem)	0.87
Average Person-Rem Exposure per GW(e) (γ)	11.5

<u>Data for 1980</u>

Average Person-Rem	1.32
Average Person-Rem Exposure per GW(e) (γ)	17

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[a] Ref. 4-15.

If a symbiotic system with ACRs on the Th/U-233 cycle were to develop, then thorium mining and processing would be required. About 100 Mg/GW(e) of ThO_2 would be required during the 30-yr life of the plant (Ref. 4-4).

4.4.2.2. Heat Rejection. Both GCFRs and LMFBRs use metal-clad fuel, and hence the maximum coolant temperatures are limited by the properties of the cladding. As a result, the GCFR has a thermal efficiency of about 33% as compared with the 39% in an HTGR. Since it is envisioned that ultimately one GCFR would supply fissionable material for several HTGRs in a symbiotic system, the number of GCFRs would be relatively small and the additional heat rejection would not be a significant penalty.

4.4.2.3. Radiological Impact. In terms of generation of gaseous, liquid, and solid radioactive wastes, the GCFR is similar to the HTGR. The only difference is that the GCFR fuel rods are vented to a helium purification system. As a result, larger quantities of tritium and the noble gases krypton and xenon are removed by the helium purification system than in an HTGR. The noble gases (mostly Kr-85) will be bottled and eventually shipped offsite.

4.4.2.4. Radiation Exposure. Even though no GCFR has been built or operated, it is anticipated that the radiation exposure of the plant personnel will be at least as low as for the HTGR and perhaps even lower. This is because not only does the GCFR share the advantages of the inert helium coolant and the integrated primary system within the PCRV, but the primary coolant will normally carry a lower level of radioactivity because of the venting of the clad fuel to the helium purification system. Evidence of this is the cleanliness of the helium coolant from the radioactivity standpoint in the in-pile loop in the BR-2 reactor at Mol, Belgium. This loop contains a 12-rod bundle of GCFR fuel rods that vent to a prototypical helium purification system. If this excellent performance could be extrapolated to a reactor plant, the GCFR would have the cleanest primary coolant of any reactor concept.

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