

fuel and physics development and on HTGR technology such as the PCRV, circulator, and steam generator. This deliberate choice should lead to development of a GCFR within a time scale comparable to that of the LMFBR, while maintaining a capability for even further substantive improvements, such as higher temperature cladding, carbide fuel, and direct cycle. As an example of commonality with LMFBR fuel, the GCFR fuel rods are collectively vented to a manifold so as to equalize the pressure on either side of the cladding, thus removing the effect of high helium coolant pressure.

300-MW(e) GCFR Demonstration Plant Design

The principal design objective of the GCFR demonstration plant is to demonstrate reactor performance and operational characteristics typical of large commercial plants. The nominal power level of 300 MW(e) was chosen to demonstrate performance of full-scale components such as fuel elements, helium circulators, and steam generators, and also to demonstrate the neutron and fuel-cycle characteristics under conditions of irradiation that correspond to those of a large commercial GCFR power plant.

The design is based on the maximum utilization of fuel technology under current development in the U.S. and in Europe on the LMFBR program, and on the continuing development of the advanced HTGR technology that forms the basis of the 40-MW(e) prototype HTGR at Peach Bottom and the 330-MW(e) HTGR Fort St. Vrain power plant.

Conservative design bases have been used throughout, and a breeding ratio of 1.33, or 1.6 with 3 tons of radial blanket is obtained under these conditions. This is largely due to the desirable properties of helium as a fast-reactor coolant. The helium has a small neutron interaction cross-section leading to a good neutron economy and avoiding any possible reactivity effects. Furthermore, the coolant does not become radioactive. Because the design assumptions are conservative, there is considerable performance growth potential inherent in the GCFR concept.

The reactor, the helium primary coolant system, and the steam generator are enclosed in a PCRV located in a reactor building that functions as a secondary containment structure and also contains the fuel-handling area and the reactor plant process and service systems. The fuel storage pool is in a fuel service building adjacent to the reactor building and is connected to it through a leading port. The steel-lined PCRV is provided after completion of the concrete construction by a system of longitudinal and circumferential steel tendons.

Containment of the entire primary system is a fundamental aspect of the GCFR design, and PCRV is a fundamental aspect of the GCFR design, which makes a rapid loss of coolant through depressurization, caused either by failure of primary coolant ducts or by vessel failure, not credible. This characteristic limits loss of coolant safety and design problems to the penetration closure. For these flow-restriction means are designed into each large penetration, strictly independent of the primary closure, to limit the maximum rate of depressurization into the secondary containment.

30-year period. The incentive for development of fast breeders is not only the need to utilize existing depleted uranium and enriched plutonium stock piles and to conserve existing uranium resources, but also to achieve high cycle efficiency (~40 percent) and low fuel cycle costs (<1 mill/kWh).

Although the first electric power ever produced by a nuclear reactor came out of the first fast breeder reactor (EBR-I in 1951), it will have been 20 years before the first sizable fast breeder demonstration plant will be operative (the BN 350 in the USSR). Two other demonstration plants are scheduled to start up in 1973 and 1975 in England and France, respectively. The first demonstration plant in the U.S. will probably not be operating before 1978. All of these fast reactors are cooled by a liquid metal.

Four coolants have been considered for fast breeder reactors: liquid metals (e.g., Na or NaK), sodium, helium, and carbon dioxide. Sodium has several advantages as a fast-reactor coolant, such as good heat-transfer characteristics at low pressure and high temperature and good emergency cooling characteristics, but it is an opaque fluid that can fall or freeze, is chemically active, and becomes radioactive in the reactor. Therefore, a great deal of effort is needed to develop reliable components such as steam generators.

An intermediate liquid-metal heat-transfer circuit is required to avoid the possibility of steam entering the primary circuit and reacting with the radioactive sodium. The metallurgical and safety problems that would arise from the use of sodium as a fast-reactor coolant are much less severe with helium and carbon dioxide. Helium is chemically inert, does not become radioactive, does not change phase, is transparent, and does not degrade the neutron spectrum, thus leading to a high conversion ratio and a negligible void reactivity coefficient. Heat-transfer characteristics of helium under typical fast-reactor operating conditions are not much different from those of sodium (1), especially since the surface heat-transfer coefficient can be significantly increased (2) by artificial roughening of the fuel-rod surface. Although pressurization is required (70 to 80 atm), the fact that the whole primary system is totally enclosed within a PCRV makes a rapid depressurization accident highly improbable. The combination of a pressurized secondary containment and several independent main and auxiliary cooling loops helps to alleviate emergency cooling problems since natural convection in helium is usually insufficient (3). Carbon dioxide has properties similar to those of helium but it could create corrosion problems.

Several types of gas-cooled fast breeder reactors (GCFR) have been proposed in the past decade but only two are being seriously considered: conservative designs using stainless-steel-clad mixed plutonium and uranium oxide fuel rods cooled by helium, with an indirect steam cycle; and advanced designs with uranium carbide-coated particle fuel, with a direct helium gas turbine cycle. Most of the effort spent on design development in Europe and in the U.S. has been on the first type of GCFR, which is based on LMFBR technology.

The primary coolant system contains three main loops, each with independent boilers and circulators, Fig. 1 (right), and three auxiliary loops, Fig. 1 (left), each with its own circulator and heat-removal system. The auxiliary loops are used for long-term shutdown cooling and as backup for the main loops. The steam generators and their associated circulators are housed in vertical cavities in the walls of the prestressed concrete vessel surrounding the reactor core. The helium coolant, at a pressure of about 1250 psia, flows downward through the core where it is heated to a temperature of 1007 F. The flow is also downward across the tube banks of the helical-coiled once-through steam generators to accommodate the use of upflow boiling in the generators.

The reactor outlet gas flows up through a central hole in the tube bundles, down through the resuperheater and steam generator, and up again around the boiler shells to top-mounted circulators, from which it is discharged to the reactor top plenum at a temperature of 593 F. The three main coolant circulators each have a single-stage axial blower driven by a series steam turbine in the high-pressure steam line. Thus, a mechanically simple and very compact power source provides the necessary large circulator power (22,300 hp each), making each main loop as self-contained and independent of the others as possible. In addition, after-heat initially provides power for its own removal. Circulation for auxiliary cooling is provided by centrifugal circulators, each driven by a 500-hp electric motor.

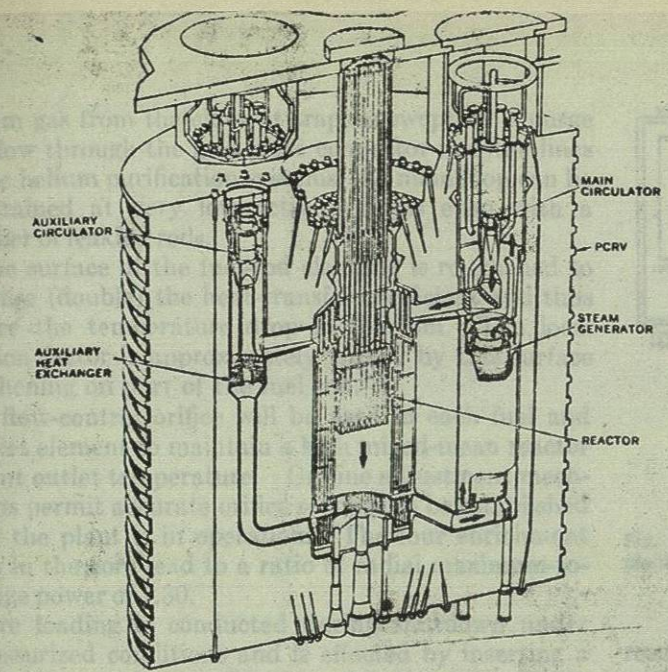


Fig. 1 300-MW(e) GCFR demonstration plant.

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The reactor assembly contains 118 hexagonal fuel and 93 blanket elements. The elements, which are 10 ft in length and 6 1/4 in. across flats, are supported from a top-mounted grid plate. They are clamped to the grid plate solely at their cold ends. Irradiation-induced metal swelling will be accommodated in the design of the core by the provision of a 0.25-in. gap between adjacent elements in the active core region. Bowing due to differential swelling will be minimized by rotation of elements at partial refueling intervals. The fuel-element-to-grid-plate clamps and variable

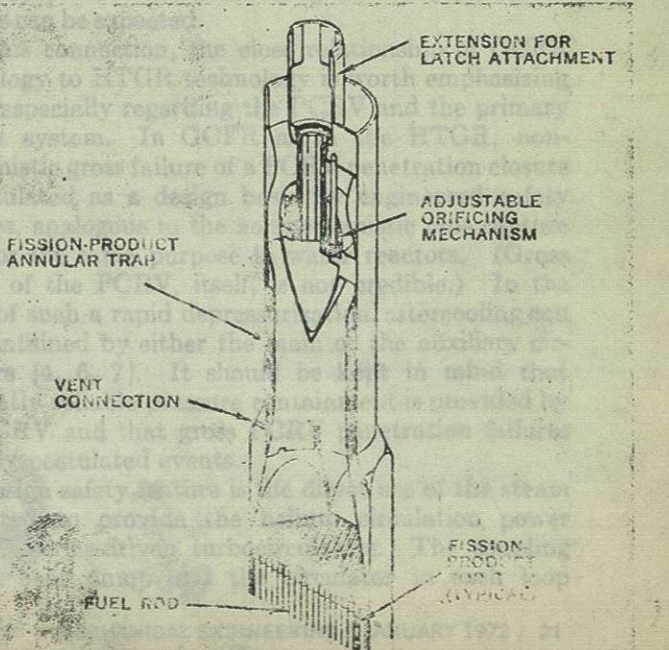
orifices of each element are actuated by external mechanisms that have drive shafts above each element through the top access plug.

Each standard fuel element contains 271 fuel rods. The fuel rods consist of annular (Pu-U)O₂ pellets within a 316 stainless steel cladding about 20 mil thick. Upper and lower axial blankets are contained in the ends of the fuel rods and consist of depleted UO₂ pellets. The blanket elements each have 127 rods of larger diameter that contain depleted UO₂ pellets.

The fuel-rod design conditions include a maximum temperature of 700 C (1292 F) at mid-thickness of the fuel cladding (including hot-spot factors), a cladding thickness ratio of 1.15, and a "smeared" fuel density within the cladding of 80 percent of theoretical density. The maximum design burnup was chosen to be 100,000 Mwd/tonne, and the maximum linear rating (with 10 percent overpower) is 13.8 kw/ft. These design parameters were selected after evaluation of existing irradiation data and are within the range now planned to be tested in the AEC's Fast Flux Test Facility (FFTF) and in the LMFBR demonstration plant programs.

The fuel rods are vented to equalize internal gas pressure to that of the reactor coolant outside the rod. This eliminates the need for a rod designed to prevent cladding creep collapse and will provide a demonstrated basis for reducing cladding thickness in later designs in order to increase plutonium production through better conversion ratio. Radiation monitors on the vent lines leading to the helium purification system provide means for detecting any leaks in the fuel rods. The fuel elements contain charcoal-filled fission-product delay traps in each of the fuel rods and also a single second-stage trap in the inlet end of the fuel element, as shown in Fig. 2. These traps are well cooled and delay the passage of the volatile and gaseous fission products long enough to minimize subsequent heat release. The arrangement permits adequate trapping for the life of the element even with several leaking rods in an element. Helium entering such a leak has, of course, no effect on reactor operation, and since the flow of vent

Fig. 2 Section through GCFR fuel-element inlet.



orifices of each element are actuated by external mechanisms that have drive shafts above each element through the top access hole.

Each standard fuel element contains 371 fuel rods. The fuel rods consist of annular (U-U)O₂ pellets within a 316 stainless steel cladding about 30 mil thick. Upper and lower axial blankets are contained in the ends of the fuel rods and consist of depleted UO₂ pellets. The blanket elements each have 127 rods of larger diameter that contain depleted UO₂ pellets.

The fuel-rod design conditions include a maximum temperature of 700 C (1302 F) at mid-length of the fuel cladding (including hot-spot factors), a cladding thickness ratio of 1.15, and a "matured" fuel density within the cladding of 80 percent of theoretical density. The maximum design burnup was chosen to be 100,000 Mwd/tonne, and the maximum linear rating (with 10 percent overpower) is 13.8 kw/Ft. These design parameters were selected after evaluation of existing irradiation data and are within the range now planned to be tested in the AEC's Fast Flux Test Facility (FFTF) and in the LAMPRE demonstration plant program.

The fuel rods are vented to equalize internal gas pressure to that of the reactor coolant outside the rod. This eliminates the need for a rod designed to prevent cladding creep collapse and will provide a demonstrated basis for reducing cladding thickness in later designs in order to increase plutonium production through better conversion ratio. Radiation monitors on the vent lines leading to the helium purification system provide means for detecting any leaks in the fuel rods. The fuel elements contain charcoal-filled radon-product delay traps in each of the fuel rods and a single second-stage trap in the inlet end of the fuel element as shown in Fig. 2. These traps are well cooled and delay the passage of the volatile and gaseous radon products long enough to minimize subsequent heat release. The management permits adequate trapping for the life of the element even with several leaking rods in an element. Helium entering such a leak has, of course, no effect on reactor operation, and since the flow of vent

The primary coolant system contains three main loops, each with independent pumps and circulators. Fig. 1 (right) and three auxiliary loops, Fig. 1 (left). The auxiliary loops are used for long-term shutdown cooling and as backup for the main loops. The steam generators and their associated circulators are housed in vertical cavities in the walls of the pressurized concrete vessel surrounding the reactor core. The helium coolant, at a pressure of about 1200 psia, flows downward through the core where it is heated to a temperature of 1007 F. The flow is also downward across the tube banks of the helium-cooled once-through steam generators to accommodate the use of upflow boiling in the generators.

The reactor outlet flows up through a central hole in the tube banks, down through the recuperator heater and steam generator, and up again around the boiler shells to top-mounted circulators from which it is discharged to the reactor top plenum at a temperature of 593 F.

The three main coolant circulators each drive a single-stage axial blower driven by a gas turbine turbine in the high-pressure main line. This is a mechanically simple and very compact power source providing the necessary large circulator power (31,100 hp each), making each main loop as self-contained and independent of the others as possible. In addition, the heat initially provides power for its own removal. Circulation for auxiliary cooling is provided by centrifugal circulators, each driven by a 500-hp electric motor.

The reactor assembly contains 18 hexagonal fuel and 93 blanket elements. The elements, which are 10 ft in length and 6 ft in cross diameter, are supported from a top-mounted grid plate. They are clamped to the grid plate solely at their cold ends. Irradiation-induced metal swelling will be accommodated in the design of the core by the provision of a 0.35-in. gap between adjacent elements in the active core region. Flowing due to differential swelling will be minimized by rotation of elements at intervals of 180 degrees during the fuel-element-to-grid-plate change and vertical

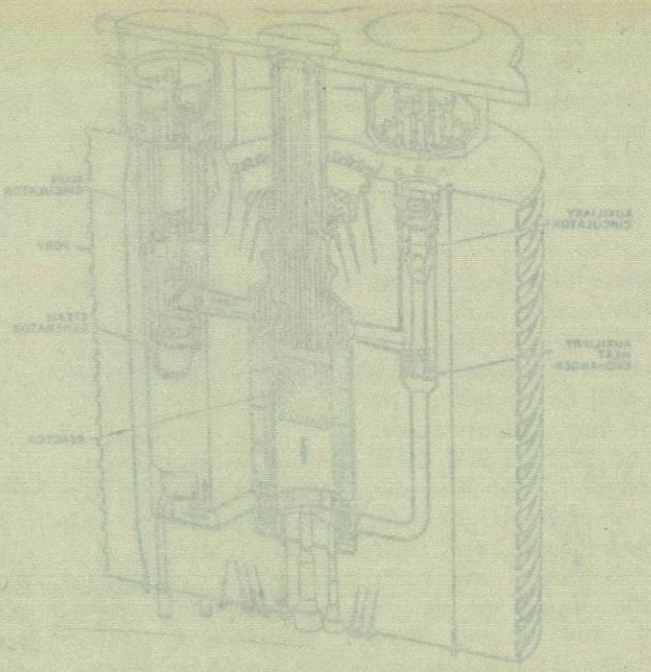


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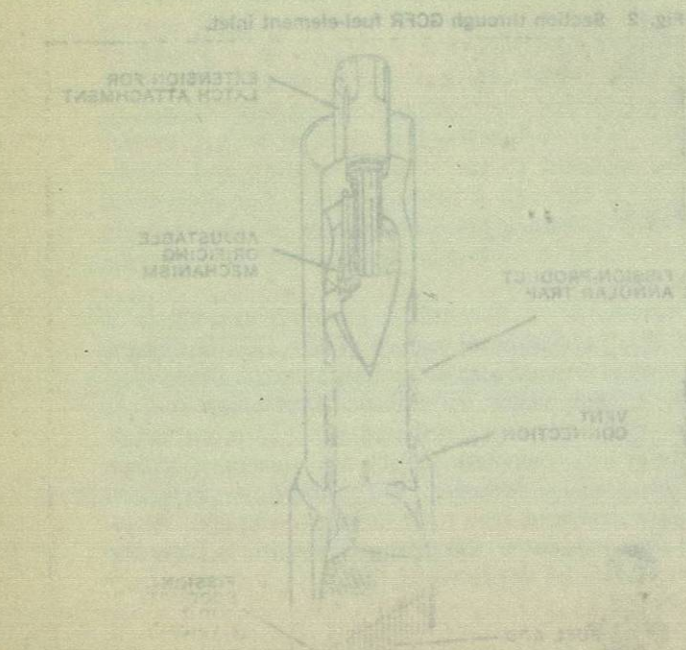


Fig. 2 Section through GCFR fuel element.

system gas from the element traps is swept by a purge gas flow through the grid plate connector into the lines to the helium purification systems, the main loop can be maintained at very low activity levels even with a number of leaking rods.

The surface of the fuel-rod cladding is roughened to increase (double) the heat-transfer coefficient and thus reduce the temperature drop in the film. The local friction factor is approximately tripled by this surface roughening on part of the fuel rod.

A flow-control orifice will be used in each fuel and blanket element to maintain a high mixed-mean reactor coolant outlet temperature. On-line adjustment mechanisms permit accurate orifice settings to be established while the plant is in operation. The four enrichment zones in the core lead to a ratio of radial maximum-to-average power of 1.30.

Core loading is conducted during shutdown under depressurized conditions and is effected by inserting a fuel-transfer machine through the bottom of the PCRV. This machine lowers and traverses fuel in the vacant space below the core to a single exit port leading to a transporting cask beneath the vessel structure [3]. Partial core reloading will occur at approximately annual intervals, one-third of the core being changed every year.

Reactivity control is by 27 rods in the control fuel elements, which have central channels to accommodate the rods. The control-rod drives are located above the reactor. Normal operation of the reactor, requiring a total reactivity swing of \$17, including a minimum \$3 shutdown margin at all times, is provided by 21 control rods, each of which is limited for safety reasons to \$0.85 worth. The six shutdown rods, each having a value of \$1.60, form a backup system capable of independently shutting down the reactor from any anticipated operating conditions.

Protection of the PCRV liner and ducts from neutron irradiation is provided by thermal shielding. Around the core this shielding takes the form of a replaceable inner layer of steel blocks surrounded by an annular region consisting of steel cylinders containing graphite. Cooling of the radial shielding is by a small bypass from the inlet helium.

The concrete plugs above the steam generators incorporate large central holes for circulator removal and smaller surrounding holes for steam pipes. Steam generator tube plugging can be done externally; the main penetration closure is removed only for complete removal of the steam generators.

The GCFR steam cycle is noteworthy in that resuperheaters are used following the circulator turbines. This, in effect, confers most of the advantages of normal reheat and provides steam dry enough to avoid the necessity for moisture separation in the main turbine.

Fig. 3 shows a simplified heat-balance diagram for the demonstration plant. In each main loop, hot helium (at 1007 F) out of the reactor first reheats the steam in a resuperheater, after which the helium flows into the superheater, evaporator, and economizer sections of the steam generator. It then passes through a helium circulator before it is returned to the reactor at 593 F (311 C). The main steam flow goes through a blower turbine, is returned to the steam generator to be

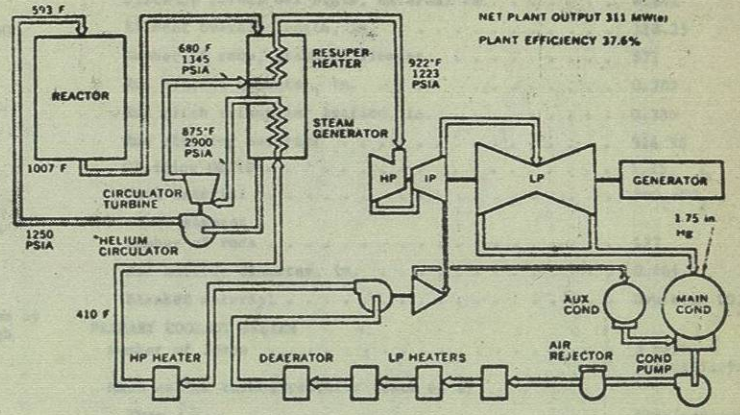


Fig. 3 Simplified heat-balance diagram for GCFR demonstration plant.

resuperheated to 925 F, and then goes to the main turbine. The net cycle efficiency of 37.6 percent leads to a net electric power of 311 MW(e) for 875 F, 2900-psi steam conditions at the superheater outlet. The steam conditions at the main turbine throttle are 922 F and 1223 psia. Further design data on the demonstration plant are given in Table 1.

Safety Considerations

A major effort has been placed on safety studies [4] and these have continued to confirm the advantages of helium as a reactor coolant. There are no possible change-of-phase problems nor are there any cladding-coolant, fuel-coolant, or steam-coolant reactions to design for; engineering for overall system safety is, therefore, eased.

An important design safety feature of the GCFR is the enclosure of the entire primary coolant system in the PCRV, thereby eliminating primary coolant ducts. Because of the conservative design bases, the highly redundant prestressing system, and the predictable, noncatastrophic failure modes, PCRVs are considered by many to have desirable safety features [5]. As design, construction, and operational experience accumulates with the 20 PCRVs both in this country and in Europe, wider understanding and acceptance of PCRVs can be expected.

In this connection, the close relationship of GCFR technology to HTGR technology is worth emphasizing again, especially regarding the PCRV and the primary coolant system. In GCFR as in the HTGR, non-mechanistic gross failure of a PCRV penetration closure is postulated as a design basis for engineered safety features, analogous to the nonmechanistic pipe rupture used for the same purpose in water reactors. (Gross failure of the PCRV, itself, is not credible.) In the event of such a rapid depressurization, aftercooling can be maintained by either the main or the auxiliary circulators [4, 6, 7]. It should be kept in mind that inherently reliable pressure containment is provided by the PCRV and that gross PCRV penetration failures are only postulated events.

A design safety feature is the direct use of the steam generator to provide the helium circulation power through series-driven turbocirculators. The coupling of the heat dump and the circulator in each loop

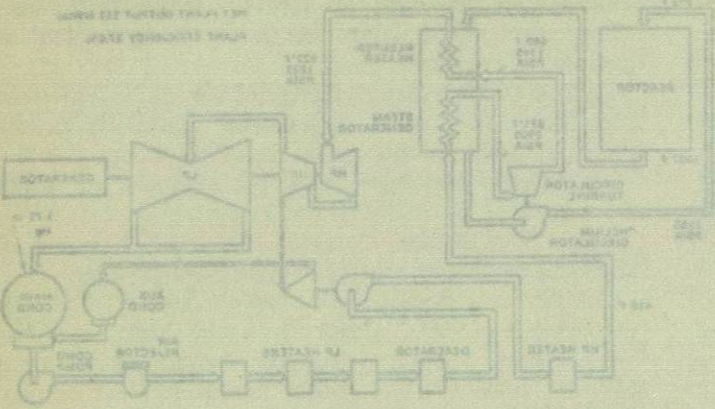


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reheated to 925 F, and then goes to the main turbine. The net cycle efficiency of 37.6 percent leads to a net electric power of 311 MW (e) for 875 F, 2000-psi steam conditions at the superheater outlet. The steam conditions at the main turbine throats are 825 F and 1325 psi. Further design data on the demonstration plant are given in Table 1.

Safety Considerations

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An important design safety feature of the GCFR is the enclosure of the entire primary-coolant system in the PORV, thereby eliminating primary coolant ducts. Because of the conservative design basis, the highly redundant pressure system, and the predictable noncatastrophic failure modes, PORVs are considered by many to have desirable safety features [4]. As design construction and operational experience accumulates with the 30 PORVs both in this country and in Europe, wider understanding and acceptance of PORVs can be expected.

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A design safety feature is the direct use of the steam generator to provide the helium circulation power through a helium turbine. The coupling between the steam generator and the helium turbine is direct, and the helium turbine is returned to the steam generator to be reheated.

system gas from the element traps is swept by a purge gas flow through the grid plate connector into the lines to the helium purification system, the main loop can be maintained at very low activity levels even with a number of leaking rods.

The surface of the fuel-rod cladding is roughened to increase (double) the heat-transfer coefficient and thus reduce the temperature drop in the film. The local friction factor is approximately tripled by the surface roughening on part of the fuel rod.

A flow-control orifice will be used in each fuel and blanket element to maintain a high mixed-mean reactor coolant outlet temperature. On-line adjustment mechanisms permit accurate orifice settings to be established while the plant is in operation. The four enrichment zones in the core lead to a ratio of radial maximum-to-average power of 1.30.

Core loading is conducted during shutdown under depressurized conditions and is effected by inserting a fuel-transfer machine through the bottom of the PORV. This machine lowers and traverses fuel in the vacant space below the core to a single exit port leading to a transporting cask beneath the vessel structure [5]. Partial core reloading will occur at approximately annual intervals, one-third of the core being changed every year.

Reactivity control is by 37 rods in the control fuel elements, which have central elements to accommodate the rods. The control-rod drives are located above the reactor. Normal operation of the reactor, requiring a total reactivity swing of \$1\beta\$, including a minimum 88 shutdown margin at all times, is provided by 31 control rods, each of which is limited for safety reasons to a 60.5 worth. The six shutdown rods, each having a value of \$1.00\beta\$, form a backup system capable of independently shutting down the reactor from any anticipated operating conditions.

Protection of the PORV from dusts from neutron irradiation is provided by thermal shielding. Around the core this shielding takes the form of a regionally inner layer of steel blocks surrounded by an outer region consisting of steel cylinders containing graphite. Cooling of the radial shielding is by a mixed flow from the inlet helium.

The concrete plugs above the steam generator incorporate large central holes for convoluted removal and smaller surrounding holes for steam pipes. Steam generator tube plugging can be done externally, the main penetration closure is removed only for complete removal of the steam generator.

The GCFR steam cycle is noteworthy in that recuperative heat exchangers are used following the auxiliary turbine. This in effect, contains most of the advantages of normal steam and provides steam dry enough to avoid the need for moisture separation in the main turbine.

Fig. 3 shows a simplified heat-balance diagram for the demonstration plant. In each main loop, the steam is returned to the reactor first reheats the steam in a superheater, after which the steam flows into the reheater, evaporator, and condenser sections of the steam generator. It then passes through a helium turbine generator. The main steam flow goes through the steam generator, after which the steam is returned to the steam generator to be reheated.

TABLE 1 300-MW(e) GCFR Demonstration Plant Data Summary

GENERAL		Fuel element	
Average breeding ratio	1.33	Distance across hex flats, external in.	6.642
Maximum fuel burnup, MWd/Te heavy metal	100,000	Element overall length, in.	118.25
Net electrical power, MW(e)	311	Number of rods, standard element	271
Plant efficiency, %	37.6	Rod outside diameter, in.	0.282
Steam conditions at main turbine		Rod pitch triangular lattice, in.	0.386
Throttle pressure, psia	1223	Rod cladding material	316 SS
Throttle temperature, °F	922	Cladding OD/ID	1.15
Condenser pressure, in Hg, absolute	1.75	Fuel material	PuO ₂ -UO ₂
Reactor coolant	Helium	Blanket element	
Reactor coolant pressure, psia	1250	Number of rods	127
Reactor vessel and primary containment	PCR/V	Rod outside diameter, in.	0.464
PCR/V dimensions, ft	84 diam by 17 high	Blanket material	Depleted UO ₂
REACTOR		PRIMARY COOLANT SYSTEM	
Reactor geometry		Number of loops	3 main, 3 auxiliary
Core height, in.	39.2	Main helium turbocirculator (each of 3)	
Core length-to-diameter ratio	0.5	Type	Single-stage axial
Axial blanket length, each end, in.	17.7	Drive	Steam turbine
Reactor subassemblies		Pressure rise, psi	60
Standard fuel elements	91	Brake horsepower (per circulator)	22,300
Control fuel elements	27	Steam generators (each of 3)	
Radial blanket elements	93	Type	Helical once-through
Core volume fractions, %		Heat duty, Btu/hr	8.45 x 10 ⁶
Fuel	30.1	Surface area, ft ²	33,400
Helium coolant	44.6	Feedwater temperature, °F	412
Cladding	10.0	Steam outlet temperature, °F	875
Structure	6.0	Steam pressure, psi	2900
Gaps (box interspace, control-rod channel)	9.3	Reheater	
Reactor heat transfer		Type	Helical
Helium temperatures		Heat duty, Btu/hr	1.47 x 10 ⁸
Reactor inlet, °F (°C)	593 (312)	Surface area, ft ²	3600
Mixed mean outlet, °F (°C)	1007 (541)	Steam temperature out, °F	925
Average power density, kW/liter of core	238	Auxiliary heat exchanger (each of 3)	
Maximum linear rating (10% overpower), kW/ft	13.8	Type	Helical, water cooled
Hot-spot cladding temperature, °F (°C)	1290 (700)	Heat duty, Btu/hr	56.4 x 10 ⁶
Radial maximum-to-average power	1.30	Surface area, ft ²	1180
Axial maximum-to-average power ratio	1.20	Auxiliary circulator (each of 3)	
Rod surface roughening		Type	Single-stage, centrifugal
Fraction of active core length roughened, %	75	Drive	Electric motor
Roughening heat-transfer multiplier	2	Brake horsepower (per circulator)	500
Roughening friction-factor multiplier	3	TURBINE GENERATOR	
Maximum heat flux, Btu/(hr)(ft ²)	520,000	Type	TC6F-23
Core and axial blanket power fraction, %	95.55	Speed, rpm	3600
Radial blanket power fraction, %	4.45	Gross electrical output, MW	316
Nuclear characteristics (midcycle)		SECONDARY CONTAINMENT	
Fissile core loading (Pu), kg	1320	Type	Reinforced concrete
Average fast neutron flux (E > 0.1 MeV), n/cm ² -sec	2.2 x 10 ¹⁷	Inside diameter, ft	114
Reactor rating, MW(t)/kg fissile	0.605	Height, ft	176
Doppler constant, TdK/dT (T in °K)	-0.0032	Atmosphere	Atm
Fuel lifetime, full-power days	750	Equilibrium pressure, atm, absolute	2
Partial refueling cycle, yr	1		

increases the reliability of cooling. The GCFR turbocirculators are similar in concept and in many details to the HTGR turbocirculators.

The primary system is designed to operate with a limited amount of steam leakage and the effect on reactivity is negative.

The use of pressure-equalized fuel rods also has important safety benefits. Most important is the elimination of fuel failure modes due to cladding collapse from high external pressure (at start of irradiation) or due to cladding deformation or rupture from internal fission-gas pressure (later during irradiation).

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