GENERAL ATOMIC

GA-A13653

DEVELOPMENT OF COMPONENTS FOR THE GAS-COOLED FAST BREEDER REACTOR PROGRAM*

by

J. B. Dee, T. Macken and L. Meyer

This is a preprint of a paper to be presented at the IAEA/OECD Nuclear Energy Agency International Symposium, Julich, Germany on October 13-17, 1975.

*This work was supported by the U.S. Energy Research and Development Administration under Contract E(04-3)-167, Project Agreement 23 and the GCFR Utility Program.

General Atomic Projects 3228 and 2102   Date Published: September 24, 1975

DEVELOPMENT OF COMPONENTS FOR THE GAS-COOLED
FAST BREEDER REACTOR PROGRAM*

by

J. B. Dee, T. Macken and L. Meyer
General Atomic Company
San Diego, Calif., USA and Zurich, Switzerland

ABSTRACT

The Gas-Cooled Fast Breeder Reactor (GCFR) component development program is based on an extension of High Temperature Gas-Cooled Reactor (HTGR) component technology; therefore the GCFR development program is addressed primarily to those components that differ in design and requirements from the HTGR. The principal differences in primary system components are due to the increase in helium coolant pressure level, which benefits system size and efficiency in the GCFR, and differences in the reactor internals and fuel handling systems, due to the use of the compact metal-clad core.

The purpose of this paper is to present an overview of the principal component design differences, when compared with those of typical HTGRs, and the consequent influences on the GCFR component development programs. Development program plans are discussed and include those for the prestressed concrete reactor vessel (PCRV),the main helium circulator and

*Work supported by the U.S. Energy Research and Development Administration under Contract E(04-3)-167, Project Agreement 23 and the GCFR Utility Program.

its supporting systems, the steam generators, the reactor thermal shielding, and the fuel handling system. Facility requirements to support these development programs are also discussed. Studies to date show that GCFR component development continues to appear to be incremental in nature, and the required tests are adaptations of related HTGR test programs.

1. INTRODUCTION

The Gas-Cooled Fast Breeder Reactor (GCFR) is of interest today because it offers excellent breeding performance with mixed oxide fuel, combined with the many practical advantages of gas-cooled reactor technology.[1,2] A breeding ratio of 1.4 can be provided by a core fabricated from today's materials,[3] with helium system components of similar size and lower temperature requirements than those to be employed in the current large HTGR programs.

Reactor plant equipment in general, and large components in particular, for the 300-MW(e) GCFR Demonstration Plant are very similar to those for the large HTGRs, and consequently the development program is addressed principally to the differences in design and requirements between the two types of reactor. These differences derive from the increase in helium coolant pressure level, which benefits component size and system efficiency, and from differences in the reactor internals and the fuel handling system, which derive from the use of the compact metal-clad core.

The principal plant components for the NSSS are illustrated in Fig. 1. The reactor vessel is a prestressed concrete reactor vessel (PCRV) containing separate cavities for the reactor core, each of three main loops, and each of three auxiliary cooling loops. Main and auxiliary circulators are located in the corresponding loop cavities.

Generally, the primary coolant circuit equipment is very similar to that of the HTGR, particularly the PCRV, steam generators, helium circulators, auxiliary heat exchangers, and associated service systems. Equipment that is less related to HTGR equipment includes the fuel and fuel-related systems such as the reactor internals (core support and reactor shielding), the control rod drives, core element locking mechanisms, and the fuel handling system.

Development programs for this equipment have been planned and will include the principal tests described in the following sections of this paper.

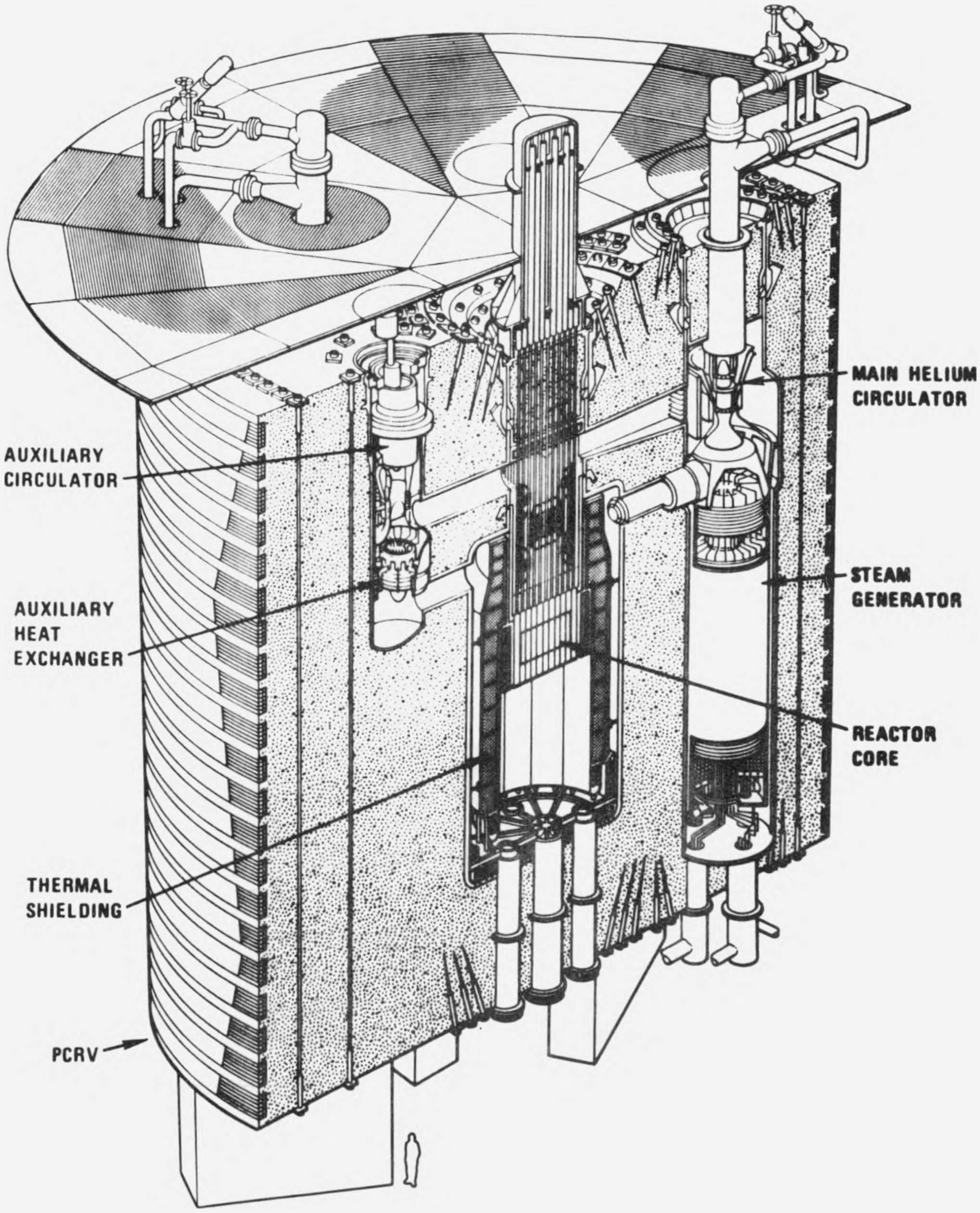

Fig. 1 300-MW(e) GCFR nuclear steam supply system

2. PRESTRESSED CONCRETE REACTOR VESSEL

The multicavity PCRV configuration is similar to the 1160-MW(e) HTGR and the Hartlepool AGR, with the exception of location of both hot and cold cross ducts near the top of the vessel in the GCFR (to simplify separation of reactor coolant inlet and outlet regions in the reactor cavity), and the addition of a large vessel penetration and closure over the reactor cavity, which allows the grid plate to be installed in a single piece and shortens the construction time required to install the reactor internals. Control rod drive penetrations in the closure are factory assembled.

Comparisons of some of the more significant PCRV characteristics of the 300-MW(e) GCFR Demonstration Plant with those of the Fort St. Vrain HTGR, the 1160-MW(e) HTGR, and the British Hartlepool AGR vessels, are shown in Table I. The table also includes characteristics of the Scandinavian model because it is a high-pressure PCRV of significant size. It should be noted that because of the small size of the reactor, the main cavity for the GCFR is relatively smaller than for an HTGR of the same power; however, the vessel walls and ligaments are thicker because of the higher internal pressure.

The PCRV will be designed in accordance with the requirements of the ASME Pressure Vessel Code, Section III, Division 2. Concrete, prestressing steel, and rebar steel stress allowables in the GCFR are maintained at the same ASME code level as for the HTGR vessels, and although the internal pressure is higher for the GCFR design, no fundamentally new or different development is required for the concrete, liner, penetrations, prestressing, or rebar.

Special features are incorporated into the design of the vessel penetrations, closures, and seals to insure that there can be no failure modes that would lead to an excessively rapid depressurization of the primary coolant circuit. These design features include redundant closure holddown

Table I

COMPARISON OF PCRV CHARACTERISTICS

	330-MW(e) Fort St. Vrain HTGR	1160-MW(e) HTGR	622-MW(e) Hartlepool AGR	Scandinavian Model	300-MW(e) Demonstration Plant
PCRV type	Single Cavity	Multicavity	Multicavity	Single Cavity	Multicavity
Vessel external diameter x length, m	14.94 x 32.31	29.41 x 27.80	25.91 x 29.26	4.27 x 6.40	25.60 x 24.54
Vessel external diameter x length, ft	(49 x 106)	(96.5 x 91.2)	(85 x 96)	(14 x 21)	(84 x 80.5)
Reactor cavity diameter, m (ft)	9.45 (31)	11.28 (37)	13.11 (43)	2.07 (6.8)	6.25 (20.5)
Steam generator cavity diameter, m (ft)	---	4.33 (14.2)	2.74 (9)	---	3.51 (11.5)
Operating pressure, MPa (psig)	4.744 (688)	4.895 (710)	3.930 (570)	6.998 (1015)	8.894 (1290)
Core outlet gas temperature, °C (°F)	772 (1422)	741 (1366)	649 (1200)	---	550 (1022)

provisions and flow restrictors at the penetrations to back up primary system seals and secondary system nozzles.

Thermal barrier requirements are considerably less demanding than for HTGR because GCFR core outlet temperatures will be highly uniform with appreciably lower peaks. Some increase in thermal conductivity is expected due to the higher helium density, and some increase in pressure drop along the flow path will accompany the design for high coolant pumping power.

The development program plan for the PCRV is intended to provide the necessary design verification and support for meeting code requirements and for licensing reviews through a test program which is similar to that followed for the HTGR. The principal tasks are listed below and are described in the sections that follow.

- PCRV Structural Overpressure Response Test
- PCRV Closure Response Tests
- Closure Primary Holddown System Test
- Flow Restrictor Tests
- Thermal Barrier Tests
    - Conductivity
    - Depressurization
    - Vibration
    - Duct

2.1. Structural Overpressure Response

The PCRV overpressure response test program meets the requirements of the ASME Pressure Vessel Code and is intended to demonstrate the failure mode and ultimate design margin of the vessel. This test program will include an overpressure test of a 1/14 or larger scale model of the vessel, which will be hydraulically pressure tested progressively through and beyond the elastic range up to failure. Areas of particular interest include behavior of the vessel in the vicinity of the multiple cross ducts, in the top head around the reactor cavity closure, and in the region of the bottom head, which contains a number of fuel handling penetrations. The large scale model (1/14 or larger), which will closely simulate the PCRV for the demonstration plant, will be preceded by a smaller scale scoping model (approximately 1/20 scale) that will be used to determine the fundamental failure mode and ultimate design capacity. This latter test work is being undertaken by Instituto de Energia Atomica in Sao Paulo, Brazil.

2.2. Main Closures

The design adequacy of the reactor and steam generator cavity closures will be demonstrated by a series of model tests. Two 1/15 scale scoping models of each closure configuration will be tested, to be followed by two 1/4 scale models, which will be hydraulically tested to verify the ultimate capacity and failure mode. The reactor cavity closure contains a large number of small-diameter penetrations for the control rod drives and the fuel element latching mechanisms.

The steam generator cavity closure contains a single large penetration in which the circulator is mounted. These tests will demonstrate the primary (normal) loading conditions on each closure design and will also demonstrate the behavior of the closure in the accident mode when it is held in place by the backup holddown system.

The first phase of this work is being carried out at ORNL under an ERDA program. Other developmental items associated with the closure will include demonstration testing of the primary holddown mechanism.

2.3. Closure Seals

Half-scale tests will be made on the primary seal between the closure and the PCRV liner. The test consists of pressure cycling the seal for simulated start-of-life conditions and for simulated end-of-life conditions assuming that creep effects have caused the main cavity to distort relative to the closure.

2.4. Flow Restrictors

Large closures are provided with flow restrictor devices to limit the rate of depressurization in the event of loss of the primary seal. A number of alternative flow restrictor designs will be evaluated, and one or more will be tested. Nozzles for steam or feedwater penetrations are also fitted with flow restrictor devices. These devices are considerably smaller than, and designed on a different principle from, those for the large closures; development tests are not required.

2.5. Thermal Barrier

The thermal barrier is installed inside the PCRV and acts in conjunction with the reactor thermal shield and the liner cooling system to maintain the concrete temperature within specified limits. The thermal barrier restricts heat flow from the primary coolant to the liner surface. The barrier consists of fiber insulation and a retaining cover plate that is not gas-tight but is designed to permit coolant pressure equalization. Failure of the thermal barrier under either normal or accident conditions could impose undesirable thermal loads on the PCRV liner and its cooling system. Hence, it is important to assure the integrity of the thermal barrier.

Factors to consider in the verification of the adequacy of the thermal barrier design include: resiliency, friction and wear, thermal cycling, attachment-fixture fatigue, vibration, moisture absorption, depressurization, conductivity, stress and permeation flow. Although most of these factors have been investigated over a sufficient range to cover the design conditions of the GCFR, there are three major problems to be resolved under GCFR operating conditions.

First, the thermal conductance under the pressure and helium flow conditions of the GCFR requires verification. The data that influence the conductance are primarily the internal helium pressure, the permeability of the installed thermal barrier, and whether or not natural or forced convection can develop within the thermal barrier. The permeability is dependent on the permeability of the insulating material itself and also on the specific design of the seal plates and attachment fixtures of the barrier to the cavity liner. This is particularly true in the main cross (hot) duct; therefore, a large-scale or a full-scale model test will be conducted to verify the performance and to determine the mechanical and thermal behavior under accident conditions.

Two additional problem areas to be investigated are (1) the flow-induced vibration effects, and (2) the adequacy of the thermal-barrier attachment system for withstanding postulated GCFR depressurization rates.

3. HELIUM CIRCULATOR

The helium circulator employs a short, stiff, high-speed rotor with a single axial flow compressor stage at one end and a similar but smaller steam turbine stage at the other end (see Fig. 2). The water-lubricated bearings and seals are located between the two discs. This configuration is sufficiently compact to permit installation and removal of the rotor and bearing assembly through a comparatively small central penetration in the large closure to the steam generator cavity in the PCRV.

Steam connections to the circulator steam turbine are through vertical concentric pipes installed in the closure central penetrations.

The circulator drive turbines are connected in series with the power-producing main steam turbine, and each takes the full flow of steam leaving the superheater section of the steam generator at 19.99 MPa (2900 psi). After expansion through the circulator turbine, the steam is resuperheated; then it flows to the main turbine. This thermal cycle differs from the

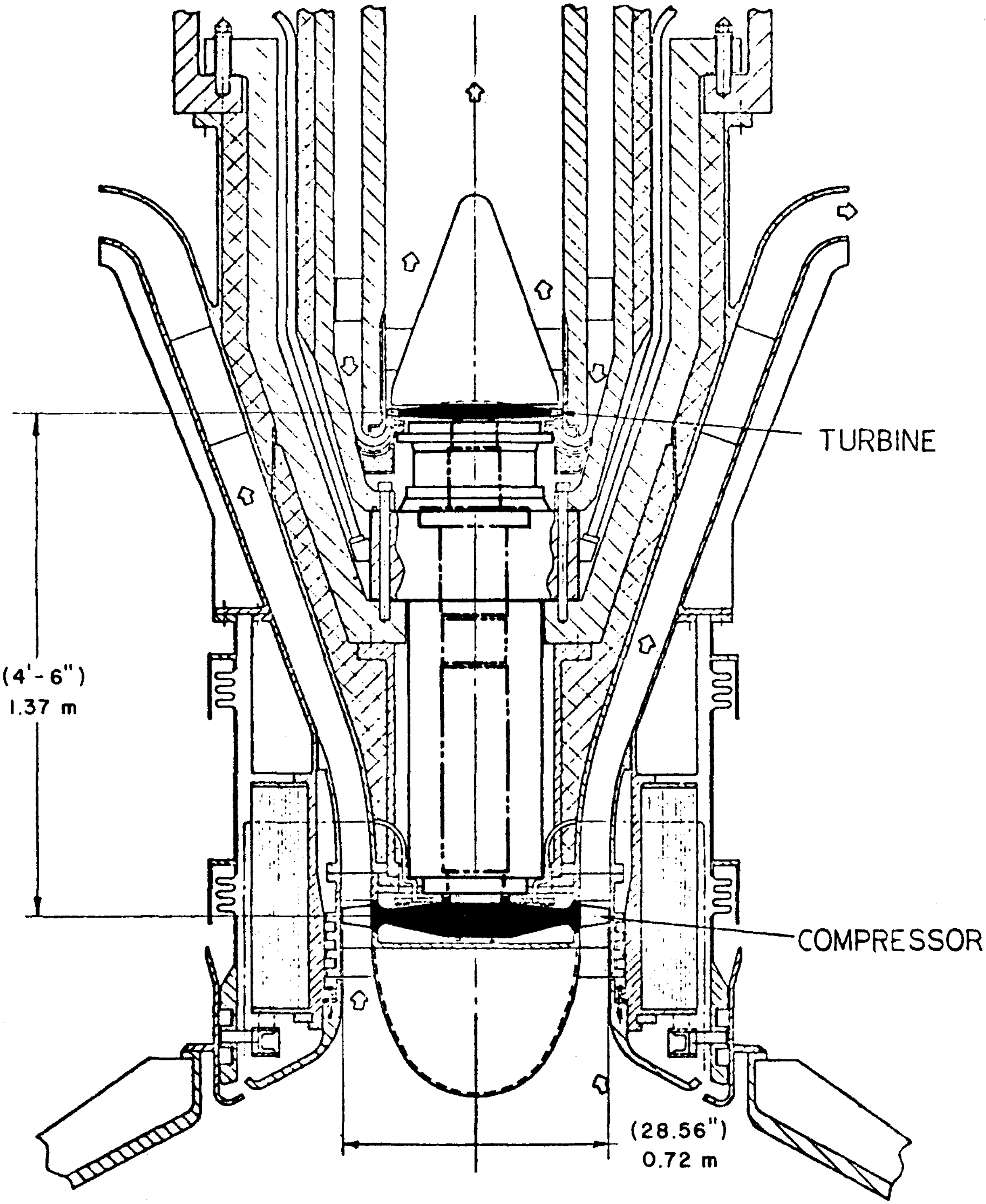

Fig. 2 300-MW(e) GCFR helium circulator

HTGR cycle in which the circulator turbine is driven by steam from the main turbine at 6.22 MPa (902 psia) before it is reheated for return to the main turbine. A self-actuating loop isolation check valve is included in each main loop to prevent backflow through the loop when the circulator is shut down.

Water-lubricated bearings and controlled leakage seals are used for the main circulators. This type of seal provides for hydrostatic operation for maximum life. Water is chosen for the lubricant in preference to oil because of its greater compatibility with helium and steam and the consequent simplification of the shaft sealing systems.

The tangential flow of water into the journal bearings causes rotation of the rotor when the bearing water is turned on. When the circulator is to be shut down for a prolonged period, a solid face seal is actuated on the helium side, and a brake is used to keep the rotor from turning.

Bearing water is circulated through the bearings, filters, and a bearing-water cooler by a steam-turbine-driven pump. These items, as well as the means for supplying and purifying the buffer helium, are incorporated in the main circulator service system.

The circulator turbine is controlled by the coordinated actions of the main power turbine control valve and a throttle valve.

The differences between the GCFR and HTGR circulator design and operation are a consequence of the increased pumping power requirements and the need for continuous operation even during reactor shutdown of the GCFR.

The principal differences influencing development include the use of high-temperature, high-pressure steam for a compact turbine drive, and a compressor stage design for higher speed (and stress levels) to produce the greater helium circuit pressure rise required. These factors affect the design of the shaft bearings and seals as well as the circulator service system and control system.

In the sequential development program outlined in Fig. 3, preliminary designs of the bearing and seal system, the service system, and the control system provide the basis for a combined bearing and seal test that dynamically simulates the rotor assembly and bearing interactions. The results of this development will be applied to the final design of the circulator. Tests will be conducted in parallel to verify diffuser performance. Subsequently, the first circulator will be thoroughly tested and qualified in a major test facility to demonstrate reliability of performance for all anticipated modes of operation including reactor shutdown and simulated helium circuit depressurization. These tests will precede completion of the second and third circulators, and any design modifications required will be incorporated before these machines are tested. The development program also includes testing of the main loop isolation valve. These tests are described in more detail below.

3.1. Bearing and Shaft Seal Development

The design principles have been well established, and the GCFR circulator will be the third design in the series (i.e., preceded by the circulator for the Fort St. Vrain and Delmarva HTGRs). The seals and bearings for the main circulator shaft and the bearing-water drain scavenging system for the circulator will be verified experimentally to ensure that their operation is satisfactorily reliable under normal, transient, and accident conditions.

The verification will be done using a full-scale dynamic model of the circulator rotor with its bearings and shaft seals and utilizing a service system representing the essential features of the service system for the plant. This will permit the bearing and seal system performances to be demonstrated before the final design of the complete circulator is committed. These tests will include verification of bearing stiffness, shaft critical speeds, operation and performance of the turbine end seals,

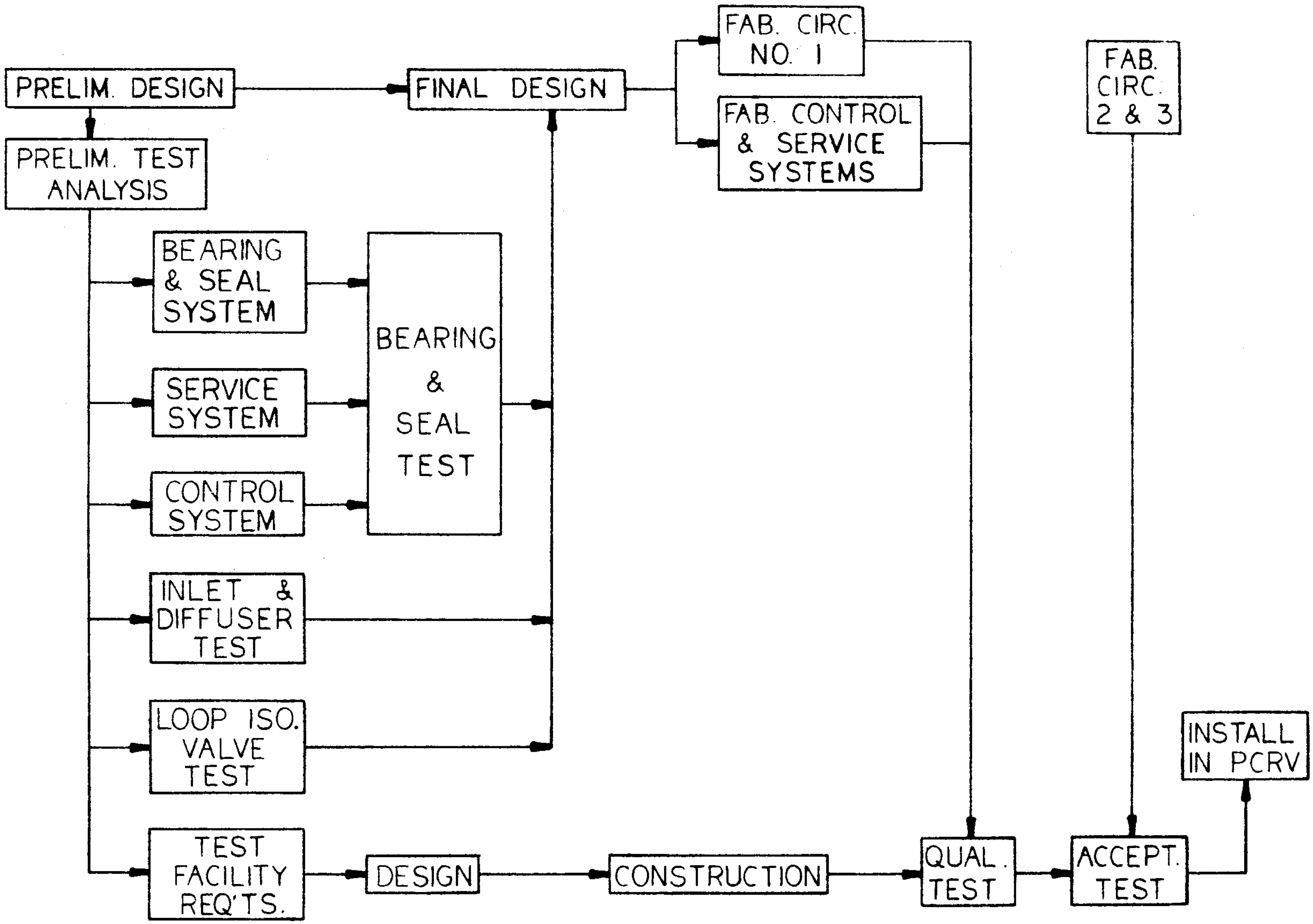

Fig. 3 Main circulator development

and performance of the condenser system, which condenses steam that leaks past the seal into the water drain cavity. The simulated service and control system will be used to investigate bearing and seal operation under normal and accident conditions. This procedure will allow these major circulator components to be thoroughly tested prior to completion of the first circulators for the plant and its subsequent qualification test.

3.2. Main Circulator Inlet Duct and Diffuser Flow Tests

The purpose of this test is to verify the performance of the main circulator inlet, diffuser, and outlet plenum for normal flow and also for backflow conditions with and without windmilling. The pressure rise of the compressor is approximately 0.41 MPa (60 psi) and diffuser losses are approximately 0.04 MPa (6 psi), yielding a net pressure rise of 0.37 MPa (54 psi).

The test will include a 1/4-scale model of the circulator inlet, a simulated circulator compressor, the diffuser, and the circulator outlet plenum. The model will use air at ambient temperature and pressure as the flow medium. The air may be circulated either by a scaled circulator compressor or by an external air flow system.

3.3. Main Loop Isolation Valve Tests

The loop isolation valve is installed in the inlet to the cross duct into the reactor core inlet plenum in each of the three main helium loops. If a circulator fails, the valve, which is normally open, is automatically closed to prevent backflow of helium through the failed circulator.

The valve will be in a helium environment at a temperature of about 315°C (600°F) and will be subjected to flow variations throughout its operating life. The valve is required to close and reopen with a high degree of reliability without external mechanical assistance in the event of loop shutdown and restart.

A full-scale model of the valve will be tested in air to verify its operation and leakage rate. Subsequently, cyclical testing of the valve bearings in a helium environment will be conducted to verify the operability, performance, and reliability under simulated operating conditions.

3.4. Main-Circulator Development and Qualification Tests

Proof testing following the extensive component and subsystem testing is planned for qualifying the full-scale GCFR circulator, complete with its service system, by testing it under conditions representative of normal operation, anticipated transients, and design-basis accidents. To accomplish this most important phase of the development program, a completed circulator unit and service system will be tested under simulated service conditions in helium and using steam to drive the turbine. The qualification test program for the lead circulator will require construction of a special test facility to match the pressure, temperature, and flow requirements.

The circulator tests made in a helium environment will simulate operating temperature conditions. The test schedule will include circulator seal system transient tests, shutdown seal and shaft brake tests, blade vibration and acoustic tests, overspeed and depressurization tests, hot restarts, and circulator acceptance performance and endurance tests. Speed, temperature, and pressure transients will subject the unit to steady-state and transient temperatures and stresses at least as severe as those to be encountered in reactor operation. A circulator service system providing bearing water, buffer helium, drain system, and other supporting services will be installed as part of the test facility and will reproduce the actual system, including components that will support the circulator in the reactor. Thus, the performance and behavior of the service system will be fully tested in conjunction with the circulator.

3.5. Helium Circulator Test Facility

Requirements for the helium circulator test facility are currently being formulated, and the relative costs and merits of full or part-power testing are being compared because of the significant cost associated with the relatively high power requirements for delivering the high pressure steam for driving the turbine. Table II shows a comparison of the 300-MW(e) Fort St. Vrain, the 1160-MW(e) HTGR, and the 300-MW(e) GCFR circulator characteristics.

To establish a basis for the circulator test facility power requirements, circulator operating conditions were identified for all loading conditions for critical components. Most operating conditions could be demonstrated at less than full power although demonstration of the full design performance and maximum loading associated with aerodynamic loading, compressor and turbine blade bending, vibration, and design basis depressurization accident transient loading conditions would require operation at full power.

Facility concepts have been studied and are feasible at full and part power. Two approaches were evaluated: a closed steam loop system concept using steam compressors to generate and circulate steam to and from the circulator turbine (see Fig. 4), and an open steam loop concept using high pressure steam from an existing fossil-fired steam generator. Both facility concepts require a helium compressor loop and a circulator steam turbine loop. The helium compressor loop is contained in a pressure vessel (to reproduce reactor conditions) provided with a means for restricting the helium flow to simulate the reactor pressure drop. A heat exchanger is required to reject the heat generated by the compressor work (Fig. 5).

The facility also houses helium circulator service modules that include bearing water and buffer helium systems in addition to provisions for supply and storage of helium and nitrogen. Approximately 500 hours of testing are projected for qualification testing of the first unit, and 200 hours for acceptance testing of each of the two circulators.

Table II

COMPARISON OF HELIUM CIRCULATOR CHARACTERISTICS

(100% Power)

	Fort St. Vrain 330-MW(e) HTGR	770-MW(e) and 1160-MW(e) HTGRs	300-MW(e) Demonstration Plant GCFR
Speed, rpm	9550	6750	11,700
Circulator power, kW (hp)	3878 (5201)	10,812 (14,500)	15,660 (21,000)
Helium flow, kg/sec (lb/sec)	110 (242)	236 (520)	234 (516)
Compressor diameter			
Hub, m (in.)	0.46 (18)	0.78 (30.75)	0.57 (22.2)
Tip, m (in.)	0.69 (27.1)	1.07 (41.0)	0.72 (28.5)
Tip speed, m/sec (ft/sec)	336 (1101)	388 (1273)	445 (1460)
Helium inlet pressure, MPa (psia)	4.73 (686)	4.85 (704)	8.62 (1250)
Helium outlet pressure, MPa (psia)	4.82 (700)	5.0 (725)	8.90 (1304)
Compressor pressure rise, MPa (psi)	0.10 (14)	0.14 (20.7)	0.37 (54)
Helium inlet temperature, °C (°F)	395 (742)	311 (592)	310 (590)
Steam inlet temperature, °C (°F)	392 (738)	375 (707)	469 (876)
Steam inlet pressure, MPa (psia)	5.90 (853)	6.22 (902)	20.0 (2900)
Steam flow, kg/sec (lb/sec)	70 (155)	160 (352)	111 (244)

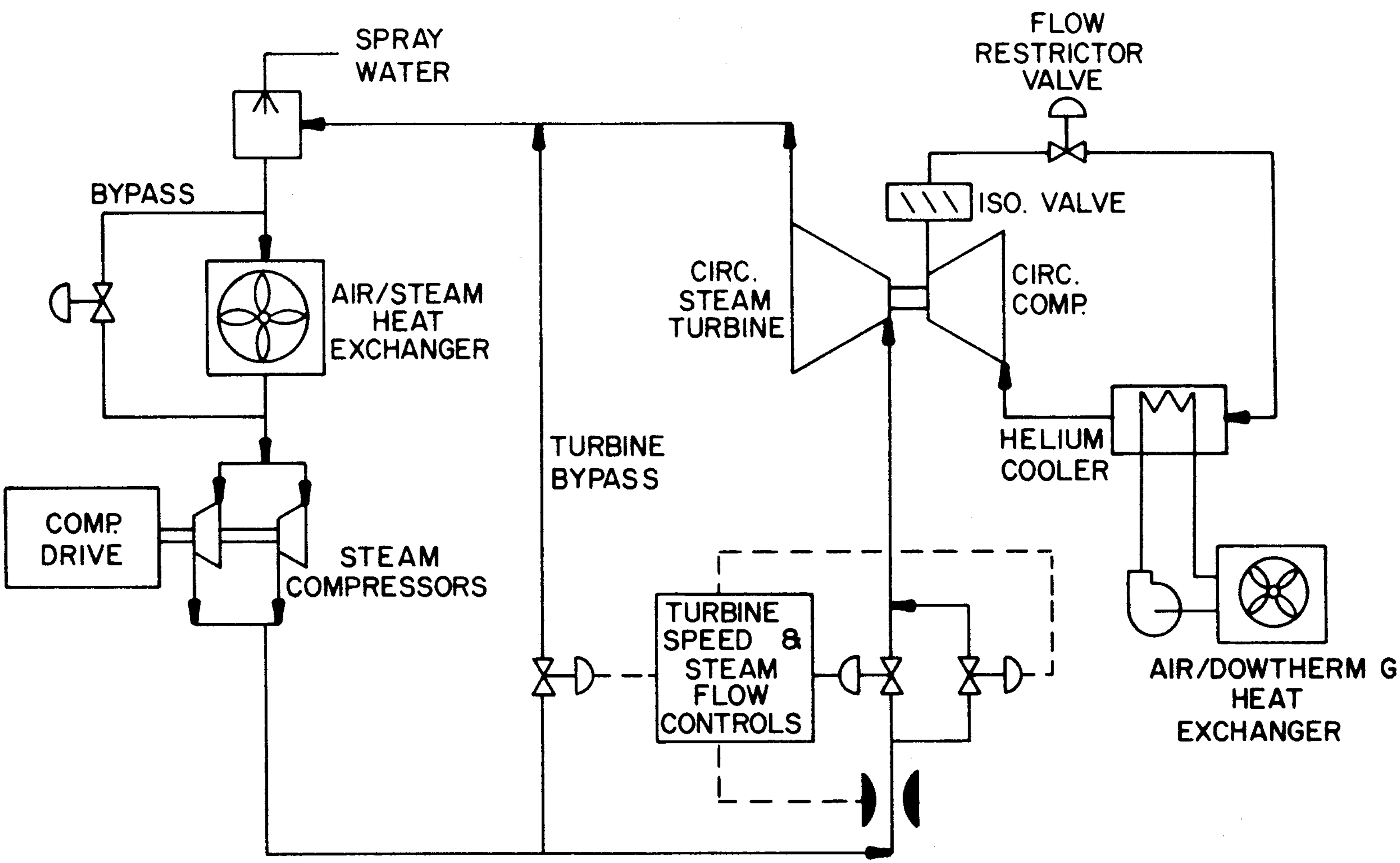

Fig. 4 Closed steam loop driven by parallel compressors

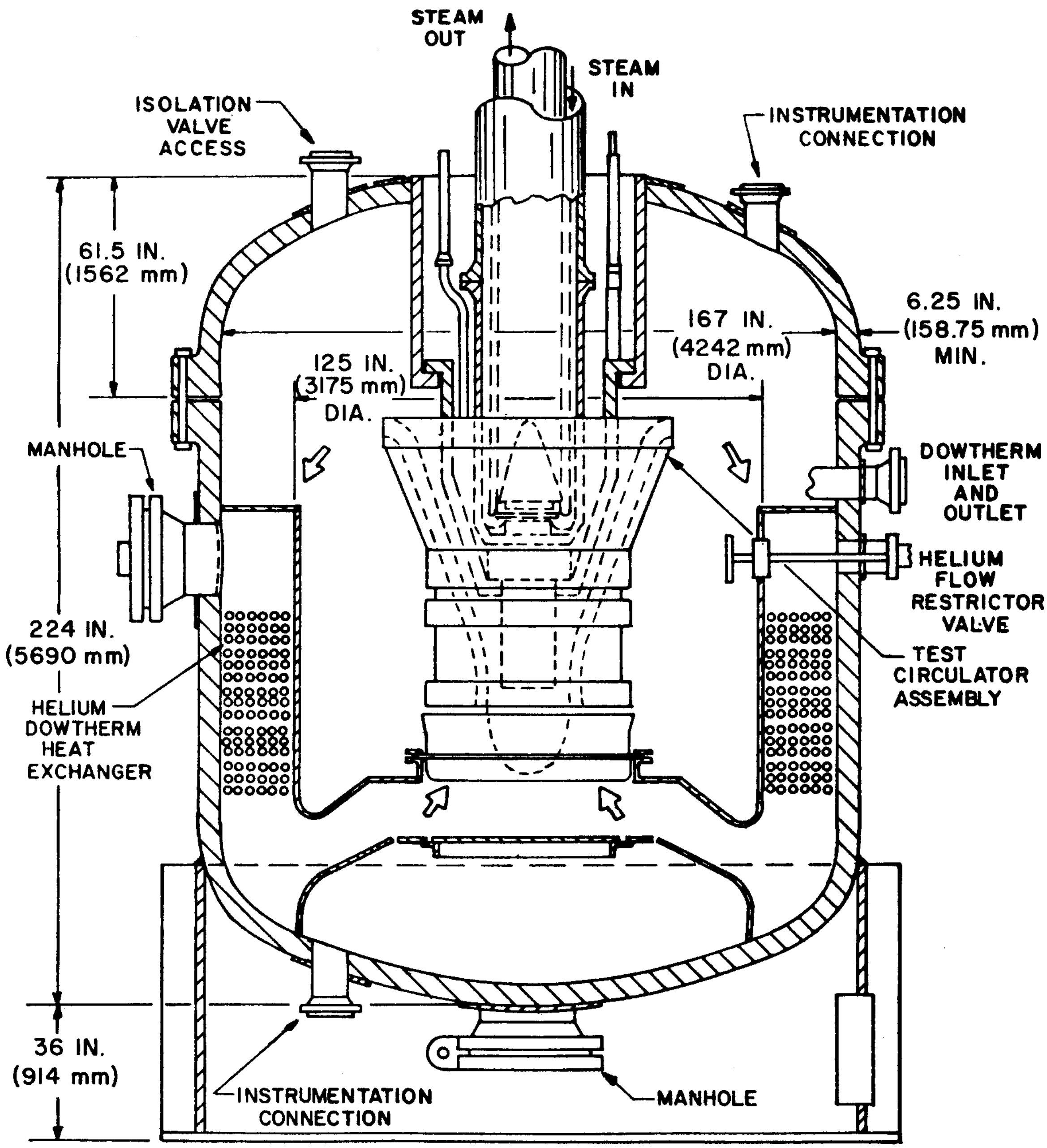

Fig. 5 Helium circulator test vessel

4. STEAM GENERATOR

In applying HTGR primary coolant technology to the GCFR, design differences arise as a consequence of the higher coolant pressure, higher circulator power requirements, and more stringent demands for uninterrupted forced cooling during shutdown and decay heat removal cooling modes. These differences, as shown in the comparison in Table III, affect the design of the steam generator as well as other components.

The steam generator is made of helical tube bundles arranged for counterflow of helium and steam. The high temperature helium from the reactor core flows downward through the steam generator and transfers heat first to the resuperheater section and then to the main steam bundle. The cooled helium is pressurized by the helium circulator and then returns to the reactor core to complete the cycle. A schematic diagram of the steam generator is shown in Fig. 6.

Superheated steam leaves the steam generator and flows through the circulator control valve to the circulator drive turbine. After expanding, the steam is piped to the resuperheater where it is heated before going to the plant turbine. After expanding in the turbine, the steam passes to the condenser where it is condensed and returned to the feedwater supply system.

The steam generators will be designed for continuous operation between 25% and 100% load, and for conditions below 25% during reactor shutdown when the main loops are used for decay heat removal. For startup and long-term decay heat removal operation, steam for driving the circulator turbines will come from auxiliary boilers.

The design of the steam generator is based on a 30-year plant life with a load factor of 80%, resulting in 210,000 hours of equivalent full-load operation. The helical boiler tube bundles are .9 to 3 m (3 to 10 ft) in diameter and 9.5 m (31 ft) high, and are supported by six vertical

Table III

COMPARISON OF GCFR AND HTGR STEAM GENERATORS

	Peach Bottom HTGR	Fort St. Vrain HTGR	Large HTGR	GCFR Demonstration Plant
Plant size, MW(e)	40	330	770 or 1160	300
Thermal megawatts per module	57.5	70.2	506	291
Helium conditions:				
Pressure, MPa (psia)	2.34 (340)	4.76 (690)	4.76 (690)	8.65 (1255)
Hot gas temperature, °C (°F)	738 (1360)	775 (1427)	741 (1366)	550 (1022)
Steam generator:				
Steam exit pressure, MPa (psia)	10 (1450)	17.3 (2512)	17.3 (2512)	20 (2900)
Steam exit temperature, °C (°F)	538 (1000)	538 (1000)	513 (955)	469 (876)
Resuperheater:[a]				
Steam pressure, MPa (psia)	---	4.14 (600)	4.65 (675)	8.86 (1285)
Steam exit temperature, °C (°F)	---	538 (1000)	539 (1002)	498 (928)

[a]Reheater for Fort St. Vrain and Large HTGR.

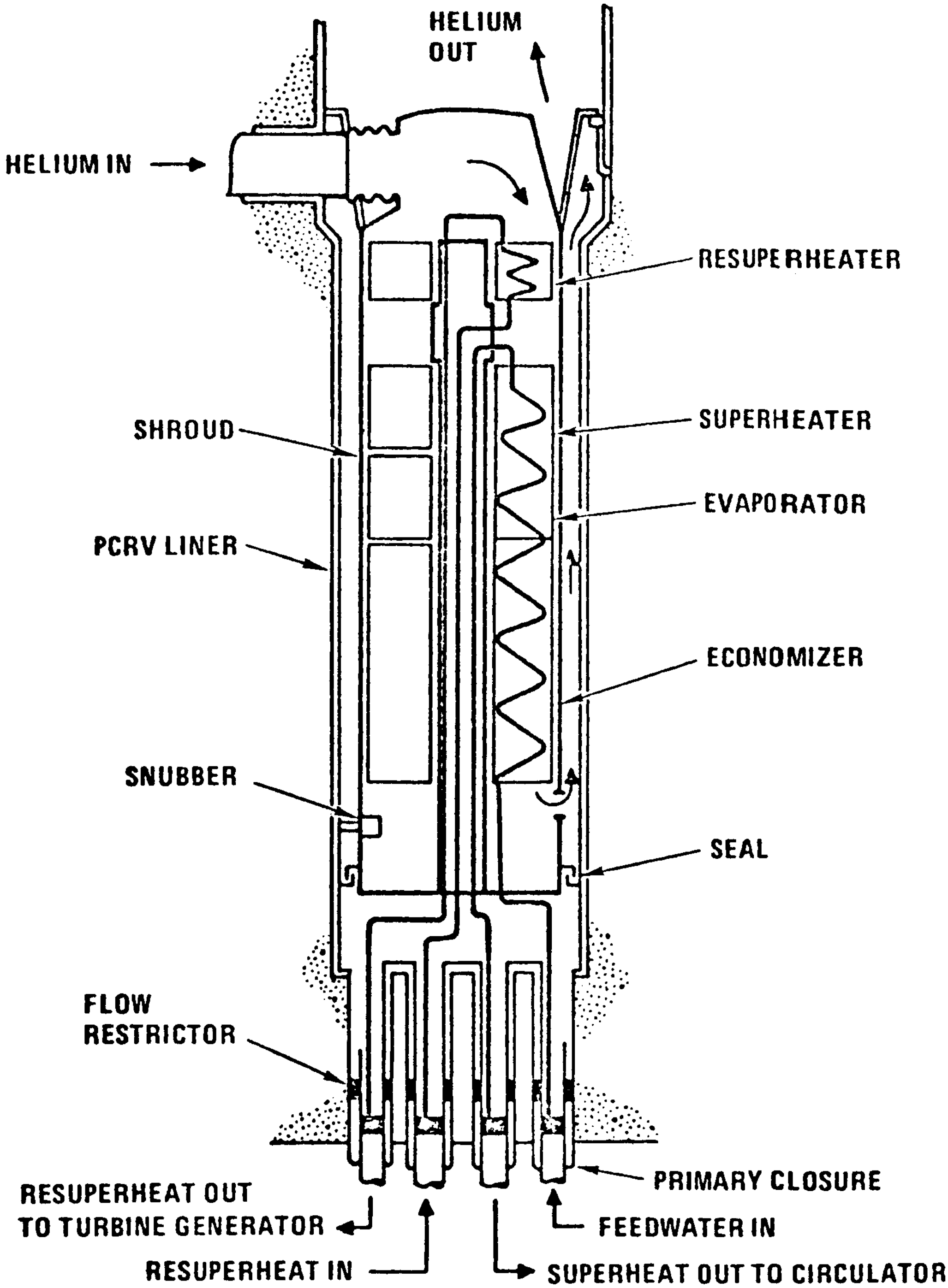

Fig. 6 Steam generator diagram

perforated plates, radially oriented. Because the GCFR steam generators are similar to those for the large HTGR in size and general construction and operate at reduced peak gas temperatures, much of the existing design and performance data is applicable. Consequently, the development program required is modest and investigates the principal operational difference: boiling stability at very low steam flow rates utilized for driving the circulators for as much as 30 minutes after shut down. Other tests are conventional and are directed toward verification of specific design configurations for controlling helium flow distribution and tube support to prevent flow-induced vibrations. The boiler and superheater tubes for the GCFR Demonstration Plant are planned to be made of Incoloy-800, and only a small supporting materials effort is needed. The specific principal development tasks planned are outlined below.

4.1. Low Flow Boiling Stability and Flow Distribution Test

When the reactor is shut down, the feedwater flow is simultaneously reduced to conserve thermal energy stored in the steam generator. This action also prevents thermal shock from possible overcooling. The stored energy subsequently provides sufficient power to drive the helium circulators to remove the reactor core afterheat.

In order to accomplish this it is necessary to demonstrate that the steam generator can operate satisfactorily at very low flows and do so without experiencing boiling instability and flow distribution problems.

The low flow boiling stability and flow distribution test is planned to demonstrate the ability of the steam generator to perform in a stable manner over the flow range of 25% through 2%. The test section will consist of approximately 12 full-length tubes connected in parallel, and will be heated by pressurized helium at conditions representative of design conditions.

4.2. Helium Flow Distribution Test

The purpose of this test is to verify that the helium flow distribution and pressure drop at critical sections in the steam generator module are satisfactory and compatible with thermal performance requirements. This test will be conducted using air at atmospheric temperature and pressure to simulate the helium flow conditions.

4.3. Vibration and Tube Wear Protection Test

The purpose of this test is to verify that vibratory stresses due to aerodynamic, acoustic, and seismic excitation are within acceptable limits, and that adequate wear protection is provided at points of contact between components.

4.4. Materials Evaluation

Based on HTGR steam generator technology and GCFR requirements, materials planned for tubing and support structures will be evaluated under specified conditions.

5. OTHER EQUIPMENT

5.1. Shielding

The reactor shield assembly protects the PCRV from radiation-induced heating and damage to materials. It will be structurally resistant to seismic forces and will include penetrations to accommodate the flow of the helium reactor coolant. It will be made of graphite enclosed in steel with boron added for suppression of gamma-ray production.

The development program will principally address:

1. Inlet and outlet helium penetration design for minimizing pressure drop losses while providing adequate radiation attenuation,

2. Effectiveness of suppression of radiation heating of the PCRV,

3. Resistance of the shielding materials to the radiation and thermal environment for the plant lifetime, and

4. Structural integrity under the various anticipated conditions of service.

5.2. Core Support

Core elements in the GCFR are firmly mounted at their cold end to the grid plate at the top of the reactor cavity, which provides a single-ended, cantilevered attachment resulting in a negative power coefficient of reactivity and other considerable safety and operational benefits. The grid dimensions are sized for stiffness, and consequently this component is not stress-limited. A test program has been initiated to verify analytical models used for computing core element motions caused by coolant flow-induced deflection of the thick core support grid, and for measuring natural frequencies, damping, and coherence of core element assembly motions caused by simulated seismic excitation.

5.3. Fuel Handling

The fuel handling equipment incorporates features analogous to those of several other systems. The most important items of equipment for possible development are (1) an in-vessel fuel transfer machine, which moves core elements to a lifting machine (elevator) for lowering the element from the PCRV, and (2) a fuel transfer cask that carries the spent fuel to a water-filled pool for storage until shipment. Developmental testing of the equipment will ensure (1) reliability of forced convection cooling, and (2) the security of grappling at all times the fuel is in transit. Additionally, the plant design includes a service facility that has a partial core mockup for checkout and servicing of the fuel handling equipment between refueling periods and, of course, the service facility is planned to be used for operator training to assure proficiency.

5.4. Control Drives and Locking Mechanisms

The core element locking mechanisms are conceptually simple devices, designed to bind the core elements to the grid plate with long shafts so that they can be actuated from above the PCRV. They will be adequately tested in a simulation of their operational environment to assure reliable operation. The control drive mechanisms incorporate a locking mechanism in addition to a compact ball-screw drive shaft powered by a direct-current stepping motor. A spring-assisted gravity trip is triggered by interrupting electric current to a holding magnet, and an inertial damper (rotary) is included in the design. Shutdown drives are related, but they are simpler and employ a higher-speed battery-operated drive motor, and the high-speed trip features are omitted. A full conventional control drive development program is planned for these drives with environmental testing.

6. CONCLUSION

The development program for testing GCFR components is limited by establishing as a design basis the maximum use of existing technology and experience. The primary coolant system components are sufficiently related to their HTGR counterparts to justify high confidence in meeting reliability criteria for first-of-a-kind components based on incremental technology and a modest test program. The single exception is the planned helium circulator test facility (HCTF) to be used for qualification testing of the higher-rated circulator, including demonstration of performance during simulated primary loop depressurization. Development of other components and systems, such as shielding, core support, and fuel handling and control rod drives, have similar counterparts in other reactor programs, and thorough testing under simulated conditions will be conducted.

References

1. SIMON, R. H., DEE, J. B., MORITA, W. I., Gas-Cooled Fast Breeder Reactor Demonstration Plant, Am. Nucl. Soc. Topical Mtg on Gas-Cooled Reactors: HTGR and GCFBR (Proc. Conf. Gatlinberg, Tenn., 1974), CONF 740501.

2. HUNTER, R. S., COMPAS, W. J., SIMON, R. H., GCFR Demonstration Plant Design and Program Progress, General Atomic Report GA-A13355 (1975).

3. OPPENHEIM, C., CERBONE, R. J., International Benchmark Calculation on a Typical 1000-MW(e) Gas-Cooled Fast Breeder Reactor, Am. Nucl. Soc. Topical Mtg on Advanced Reactors; Physics, Design, and Economics (Proc. Conf. Atlanta, Ga. (1974) 523.