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SPECIAL DEVELOPMENT
PROBLEMS OF HIGH-TEMPERATURE
GAS-COOLED REACTOR SYSTEMS

by

A. J. Goodjohn and D. B. Trauger

August 21, 1970

**General
Atomic**

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SPECIAL DEVELOPMENT PROBLEMS OF HIGH-TEMPERATURE GAS-COOLED REACTOR SYSTEMS

A. J. Goodjohn and D. B. Trauger*

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INTRODUCTION

The High-Temperature Gas-Cooled Reactor (HTGR) nuclear steam system being offered commercially by Gulf General Atomic has evolved from over a decade of development work in the basic technology of high-temperature gas cooling and in the design of the plant components and the fuel for the required high-temperature service.

The first HTGR plant to be built in the United States was the 40-MW(e) Peach Bottom Atomic Power Station - Unit 1, which has been in successful commercial operation on the system of the Philadelphia Electric Company since June 1967. To date Peach Bottom has produced over 467,000 MW-hr of electrical energy with design steam conditions of 1450 psig and 1000°F and a net plant efficiency of about 35%. The operation of Peach Bottom has confirmed the basic performance features and operating characteristics of the HTGR.

In 1968 Gulf General Atomic, as a part of the USAEC Power Reactor Demonstration Program, began construction of the 330-MW(e) Fort St. Vrain Nuclear Generating Station for the Public Service Company of Colorado. The construction permit for Fort St. Vrain was received in September of 1968. As of August 1, 1970, the overall construction was 72% complete. Initial startup is planned for the third quarter of 1971, with commercial operation scheduled for March of 1972. During the period between the commitment on Peach Bottom and the freezing of the design for Fort St. Vrain a number of advancements in the technology were made, particularly in the areas of prestressed concrete reactor vessels (PCRVs) for enclosing the whole of the primary coolant system and the coated particle fuel. In addition, changes were made in several of the components in order to properly use the advanced pressure vessel and

fuel technology. Fort St. Vrain embodies these features and will serve to demonstrate the most recent developments in high-temperature gas-cooled reactor technology.

The nominal 1100-MW(e) HTGR nuclear steam system incorporates the same basic design features as Fort St. Vrain, and extensions in the technology are minimal in the large plants. Table 1 summarizes the evolution of the HTGR system from Peach Bottom to the 1100-MW(e) HTGR. Except for the larger sizes of the components required for the large HTGR, it is evident that the changes are minor and primarily result from optimization of the larger plant for slightly different operating conditions.

The development programs associated with the components and the fuel for the HTGR system have been extensive. The work has been carried out not only at the Gulf General Atomic facilities in San Diego but also at several other facilities in the country and particularly at the Oak Ridge National Laboratory. It would be impossible in this paper to summarize more than just the major problems and accomplishments in a few of the development programs. The work, in total, is extensively covered in the many topical and progress reports which have been published throughout the course of the work and which are available through USAEC information agencies.

TABLE 1
EVOLUTION OF HTGR NUCLEAR STEAM SYSTEM

	Peach Bottom	Fort St. Vrain	1100-MW(e) HTGR
Net Plant Output, MW(e)	40	330	1100
Reactor Thermal Output, MW(t)	115	842	2804
Net Plant Efficiency, %	34.6	39.2	39.0
Primary Coolant	Helium	Helium	Helium
Primary Coolant Temperature (cold/hot), °F	650/1380	750/1430	630/1413
No. of Fuel Elements	804	1519	3800
Fuel Burnup, MWD/tonne	60,000	100,000	92,000
Reactor Vessel	Steel	Prestressed Concrete	Prestressed Concrete
Control Rods	36 - Bottom Mounted	74 - Top Mounted	146 - Top Mounted
Helium Circulator	2 - Motor-Driven Centrifugal	4 - Steam-Turbine-Driven Axial	6 - Steam-Turbine- Driven Axial
Steam Generators	2 - U-Tube, Drum Type	2 - Once-Through Integral Reheat	6 - Once-Through Integral Reheat
Steam Conditions, psig/°F/°F	1450/1000	2500/1000/1000	2500/955/1000

LARGE HTGR DESCRIPTION

Figure 1 shows an overall view of a large HTGR power plant. The nuclear steam system is located within a conventionally reinforced concrete secondary containment building. The adjacent reactor service building contains the new and spent fuel storage facilities, access for fuel delivery and shipping, some auxiliary systems, and a service facility for maintenance of equipment items. The turbine generator unit and other balance-of-plant equipment are located in a turbine building equivalent in size to a turbine building for a fossil-fired plant.

Figure 2 schematically illustrates the flow diagram for a six-loop nominal 1100-MW(e) HTGR plant. The primary coolant system is wholly contained within a multiple-cavity PCRV. The primary coolant (helium) at a temperature of about 630°F is discharged from the helium circulator at the top of the side-wall cavities and flows through six radial ducts to the top of the core cavity. The helium flows down through the core and exits into the lower plenum at a temperature of about 1400°F . The helium then flows radially to the six PCRV side-wall cavities in which the once-through steam generator units are located.

On the secondary side feedwater at 340°F is converted to superheated steam in the steam generators and is piped to the high-pressure turbine. The steam conditions at the turbine throttle valve are 2400 psig/ 950°F . The steam flows through the high-pressure turbine cylinder and returns to the nuclear steam system. Before entering the reheater portion of the steam generator, the steam provides the power to drive the steam turbine of the helium circulators. This feature of the HTGR is the only major deviation from a conventional fossil-fired plant single reheat steam cycle. The temperature of the steam is increased to 1000°F in the reheater and

Fig. 1. 1100-MW(e) HTGR power plant

K70108

Fig. 2. Schematic flow diagram for 1100-MW(e) HTGR plant

the steam returns to the intermediate-pressure and then to the low-pressure turbine cylinders. The steam is then condensed and the condensate is pumped through four stages of feedwater heaters back to the nuclear steam system.

Figure 3 shows the general arrangement of the major components of the nuclear steam system within the PCRV. The reactor core, reflector, control rods, and internal structures occupy the central cavity, which is about 37 ft in diameter by 44 ft high. The six separate main cooling loops occupy 13-ft-diameter cavities which are spaced symmetrically around the central cavity and are connected to it by insulated radial ducts. The PCRV is a right cylinder, 81 ft high by 96 ft 6 in. in diameter. Process and service connections to the steam generator units are made through penetrations at the bottom of the PCRV, and connections to the circulators are made at the top. Control rod drive assemblies are mounted in the 73 refueling penetrations in the top head, where additional penetrations are also provided for helium purification system components and nuclear instruments. Three additional cavities approximately 7 ft in diameter with penetrations in the top head are provided for the auxiliary circulators and heat exchangers, which are used to satisfy the emergency cooling requirements for the reactor.

The PCRV functions as the primary coolant system boundary. The complete PCRV consists of (1) an internal steel liner which acts as a sealing membrane, (2) a thermal barrier mounted on the inside of the liner and a cooling coil system attached to the outside of the liner whose joint function is to limit the temperature of the concrete, (3) penetrations and closures which provide access to the vessel cavities, and (4) high-strength concrete with bonded reinforcement steel and a prestressing

K80738

Fig. 3. HTGR Nuclear Steam System

system to provide strength. The entire assembly is mounted on a support structure anchored to the foundation pad.

The development of the PCR-V has been one of the major special development areas associated with the HTGR system. This program has involved model testing, testing of various stressing systems, materials development, and the improvement of methods for detailed stress analysis.

The once-through steam generators for the large HTGR are generally larger versions of the Fort St. Vrain modules. Each of the six units is approximately a factor of six larger in weight and heat output per module.

The materials used in the steam generators range from carbon steel to nickel-base alloy. Carbon steel is used in the low-design-temperature items such as the feedwater tubesheet and the feedwater tubes. Various grades of low-alloy steel are used for the closure, the cold reheat tubesheet, portions of the shroud, and the economizer-evaporator tubes and their support structure. Nickel-iron-chrome alloy (Incoloy 800 or equivalent) is used for the superheater tubes, the reheater tubes, and portions of the shroud, and a high-strength nickel-base alloy (Inconel 625 or equivalent) is used for the superheater tubesheet, the hot reheat tubesheet, and the support structures for the superheater and reheater.

The major development program associated with the steam generator units involved material selection and fabrication techniques, together with boiling stability and vibration testing.

The main helium circulators are also generally larger versions of the Fort St. Vrain series steam-turbine-driven axial flow units, with flow and power requirements approximately double those of each of the Fort St. Vrain circulators. The thrust and journal bearings are water-lubricated

hydrostatic bearings with hydrodynamic capability at high speed. A shutoff valve is located at the discharge of each circulator to prevent backflow of helium through a shutdown circulator.

The circulators have also been one of the major special developments associated with the HTGR system. The design is the result of an extensive program progressing from test-rig operation of the water bearings to testing of a prototype machine and finally to the proof testing of the production circulators.

The control rod drive units are installed in the top head refueling penetrations. These units, which are very similar to the Fort St. Vrain units, are essentially electrically powered winches, each of which raises and lowers a pair of control rods by means of flexible steel cables. The control rods are guided into the core by a guide tube assembly located in the upper plenum above the reactor core. Scram occurs by gravity action, with the velocity being controlled by regenerative braking action of the motor.

The development program for the control rod drive units has consisted primarily of life testing of the assembled units under simulated reactor operating conditions and, although not considered special, the testing program has been extensive.

Refueling operations will be carried out with the plant off-line and with the reactor helium inlet temperature and pressure lower than 250°F and atmospheric, respectively. The principal equipment consists of a fuel handling machine for actually removing and replacing fuel and reflector elements in the reactor, fuel transfer casks for transferring canned elements between the fuel handling machine and storage, an auxiliary service cask for removing and installing control rod drive assemblies before and after refueling, reactor isolation valves to

provide shielding and sealing for the refueling penetrations during the refueling operation, and motorized vehicles for transporting the fuel transfer casks between the fuel handling machine and storage.

The refueling cycle is based on a 4-year life; i. e. , one-quarter of the core is replaced each year by new or recycled fuel. The reactor shutdown time required for the refueling operation is estimated to be about 17 days, including the time required for pump-down and cool-down of the reactor and the time required for pump-up and reactor startup.

Refueling operations and equipment for refueling an HTGR have evolved through a continuing development program over the past decade. The refueling concept for a gas-cooled reactor is unique in that operations are carried out remotely with shielded mechanisms through an optically transparent medium. Since the time required to prepare the reactor for refueling has been minimized by the use of remotely controlled mechanisms, the emphasis in the program has been on developing highly reliable machines with computer-programmed operations.

The reactor core and specifically the fuel and fuel elements for the HTGR have been necessarily the key development aspect of the concept.

The reactor core consists of approximately 3800 hexagonal graphite block fuel elements approximately 31 in. long and 14 in. across the flats. These fuel elements are arranged in columns eight blocks high and are surrounded by a graphite reflector approximately 4 ft thick. The core itself is divided into 73 regions, each consisting of a central fuel element column surrounded by up to six columns. Each central fuel element column contains the vertical holes for the associated control rod pair and a hole for the insertion of the reserve shutdown material. Each block, including the control rod column blocks, contains an array of vertical fuel holes and coolant passage holes. The fuel for the reactor consists of

coated uranium and thorium carbide particles which are bonded into rods. The bonded rods are loaded into machined re-entrant fuel holes in the graphite blocks and the upper end of the fuel hole is sealed with a graphite plug before the block is loaded into the reactor.

Conventional nuclear-grade graphite is used throughout the core. The fuel materials in the initial core would be 93% enriched uranium and fertile thorium. In the later recycle mode, U-233 would be included as a feed material, replacing much of the U-235.

FUEL DEVELOPMENT

Much of the versatility of coated particle fuel has been developed specifically for service in the HTGR.

In the Peach Bottom reactor, coated particle fuel was used with the fuel element shown in Figure 4 for the first core. The coating consisted of a layer of laminar pyrolytic carbon. The coated particles were subsequently dispersed in a graphite matrix and hot pressed into compacts. Loop activity was maintained at a low level by a specially impregnated low-permeability graphite sleeve into which the fuel compacts were fitted. A helium bypass purge stream entered each element through a porous plug provided at the top of each element, picked up fission products during passage down the element, and exited through the standoff pins at the bottom of the element to a fission product trapping system outside of the reactor. Integrity of the coating on the fuel particles was not required during reactor operation. Data available at the time the fuel particles for Peach Bottom Core 1 were fabricated indicated that increases in compact dimensions which resulted from fuel compact swelling during irradiation would occur

K11065

Fig. 4. Peach Bottom fuel element

only in the axial dimension due to the structure of the compact, and a sufficient gap was left at the top of each element to accommodate this increased length. Subsequent operation of the first core showed that, due to adherence of the matrix material to the particle coatings and due to the curling characteristics of the laminar particle coatings under irradiation, radial expansion of the compacts also occurred. The radial expansion led to cracking of some of the sleeves; a decision was thus made to remove the first core after 450 equivalent full power days and to insert a second core containing improved coated particles. The reactor continued to operate during this period of time until the scheduled shutdown.

The development of coated particles to provide greater dimensional stability and improved fission product retention was a logical extension of the fuel development program following the specification of the Peach Bottom first core. In preparing such particles, small kernels of thorium and uranium carbide or oxide are coated with highly stable pyrolytically deposited carbon. For some designs, a layer of silicon carbide is also included to more completely retain metallic fission products, particularly cesium and strontium. The particles, examples of which are shown in Figure 5, are bonded into fuel rods, shown in Figure 6, by using carbonaceous materials as binders. The fuel rods are then inserted into the fuel holes of the graphite fuel element shown in Figure 7.

Typically for HTGR applications the particles must retain integrity at temperatures up to 1250°C for fuel burnups to 20 atom percent or greater and for fast-neutron exposure approaching $8 \times 10^{21} \text{ n/cm}^2$ ($E > 0.18 \text{ MeV}$) in lifetimes of 3 years or more. The high-energy neutrons can cause significant damage to the pyrolytic carbon by displacing atoms in the crystal lattice, thus producing distortion. Also the fuel swells during irradiation because the volume of fission products formed exceeds that of the uranium and thorium compounds which are destroyed. This swelling and

M251-94

(a)

(175x)

M275-48

(b)

(200x)

Fig. 5. Irradiated coated particles tested to greater than Fort St. Vrain maximum exposures: (a) TRISO coated particles, irradiated to 8.7×10^{21} n/cm² at 1300°C (max) in capsule P20, 27% FIMA; (b) BISO coated particles, irradiated to 8.4×10^{21} n/cm² at 1300°C (max) in capsule P18, 25% FIMA

K340-39

(~2X)

K340-65

(2X)

(a) $5.0 \times 10^{21} \text{ n/cm}^2$, 1250°C , 18% FIMA

(b) $5.7 \times 10^{21} \text{ n/cm}^2$, 1250°C , 18% FIMA

Fig. 6. HTGR fuel rods after irradiation in GETR capsule F-27

Fig. 7. HTGR fuel particles and element block

the gaseous fission products released from the fuel can produce high internal pressures in the coated particles. The most damaging effect, however, can be caused by the fission product atoms recoiling from the fuel surface at very high energies. The pyrolytic carbon coating contains a sacrificial inner layer which is highly porous to absorb the recoil energy, accommodate fuel swelling, and provide a volume for retaining gaseous fission products. The outer layer of dense isotropic pyrolytic carbon forms a pressure vessel which, by its high ratio of wall thickness to inner volume and its great strength, can accommodate very high pressures.

Although large numbers of coated particles are required, the fluidized bed furnaces in which pyrolytic coating deposition is accomplished can be highly automated. Deposition of the porous layer and the outer pressure vessel layers can be made in a continuous operation. The silicon carbide layer can be deposited with the same equipment. Careful control of the coating gas mixtures, temperatures, and gas flow are utilized to obtain a uniform product. Relatively modest test programs provide excellent statistical assurance that a particle type is adequately developed and satisfactory for the intended service since by their small size, correspondingly large numbers of particles can be tested in irradiation capsules.

Although the fission product retention capability of coated particles was demonstrated some time ago, irradiation of fuel element structures, i. e., the fuel rods and the fuel rods in the graphite element, has only recently been carried to the higher fast-neutron fluence of the commercial HTGR. Early tests showed that the high-energy neutron exposure induced shrinkage of the coatings and bonding material, and that the stresses produced by this differential shrinkage caused some particle coatings to break and the structure to crumble. By increasing

the amount of carbon introduced between the particles and using more stable carbonaceous materials, bonded fuel elements have been produced which remain stable under reactor test conditions exceeding HTGR service requirements for burnup and fast-neutron exposure,^{1,2} as shown in Figures 8 and 9.

Fuel elements using coated particles designed by several projects have been demonstrated to be successful.^{3,4} Core charges manufactured for experimental HTGRs now number four for the Dragon reactor, two for the German AVR, two for Peach Bottom, and one for UHTREX (Ultra High-Temperature Reactor Experiment) in Los Alamos. In addition, fuel is now being manufactured for the Fort St. Vrain Reactor.

PRESTRESSED CONCRETE REACTOR VESSEL DEVELOPMENT

Prestressed concrete has been used widely for non-nuclear vessels and also for reactor vessels in Europe where the experience now exceeds 25 years of total operation of reactors contained in PCRVs. This experience is summarized in Table 2. Several advantages have been noted for the use of concrete vessels, e. g., on-site construction with local labor and materials, shorter construction schedules, and improved safety. Despite the considerable experience with PCRVs, extrapolation to the higher pressures and temperatures of the HTGR and the requirements specific to USAEC licensing of nuclear systems have necessitated extensive development.

Early concrete pressure vessels housed only the reactor core, reflector, and some shielding. Recent designs use the integrated circuit in a manner similar to the large HTGR previously described in this paper, wherein all primary system components are located inside the vessel. Use of the integrated circuit necessitates a close coordination in the

(a) Unirradiated

K83749

(b) Irradiated in HT-4; 1050°C, $8 \times 10^{21} \text{ n/cm}^2$ ($E > 0.18 \text{ MeV}$)

Fig. 8. BISO particles before and after irradiation in fuel rods

M340-148

(a)

(125x)

M340-125

(b)

(75x)

Fig. 9. TRISO particles in fuel rods after irradiation: (a) irradiated to $5.0 \times 10^{21} \text{ n/cm}^2$ at 1250°C , 18% FIMA; (b) irradiated to $5.7 \times 10^{21} \text{ n/cm}^2$ at 1250°C , 18% FIMA

TABLE 2
STATUS OF PCRVS OPERATING OR UNDER CONSTRUCTION

Location	No. of PCRVS	Reactor Type	Working Pressure (psia)	Internal Dimensions (ft) Diam x length	Status March 1970
Marcoule G-2,* G-3* France	2	(2) 30-MW(e) Magnox	215	64x51	Operating since 1958, 1959
Chinon EDF-3,* France	1	480-MW(e) Magnox	400	62x69	Operating since 1966
Oldbury, UK	2	(2) 300-MW(e) Magnox	365	77x60	Operating 1967-68
St. Laurent I France	1	480-MW(e) Magnox	400	62x119	Operating 1969
Wyfla, UK	2	(2) 590-MW(e) Magnox	400	96x96	Construction, vessels finished
St. Laurent II France	1	545-MW(e) Magnox	400	62x119	Construction, vessel finished
Bugey I, France	1	540-MW(e) Magnox	610	56x126	Construction
Vandellos, Spain	1	500-MW(e) Magnox	400	62x119	Construction
Dungeness B, UK	2	(2) 600-MW(e) AGR	450	66x58	Construction
Hinkley Point B, UK	2	(2) 625-MW(e) AGR	600	63x62	Construction
Hunterston B, UK	2	(2) 625-MW(e) AGR	600	62x64	Construction
Hartlepool, UK	2	(2) 625-MW(e) AGR (Multicavity)	600	43x60	Construction
Fort St. Vrain Colorado, USA	1	330-MW(e) HTGR	700	31x75	Construction

*Only the reactor is located inside the PCR V. The entire primary system is inside the PCR V for the other reactors listed.

design and development of internal components and the vessel. Many components are simplified by the elimination of separate heavy pressure vessels and by reduced piping stresses. While the integrated PCR V is complicated by the increased size of the vessel and its penetrations, it eliminates the need for external primary coolant loops and the safety concerns which accompany them.

An important development item has been the thermal barrier which is mounted on the inside of the steel liner. Early designs in the United Kingdom and France used metallic foil or porous concrete, respectively, to protect the liner. For the HTGR, it appears better to use fibrous blankets held by steel plates which are anchored to the PCR V liner. Figure 10 shows such insulation being installed at the Fort St. Vrain reactor. Extensive tests have been run to select the best fibers and to show that degradation will not occur from vibration in the hot, dry helium, from reaction with coolant impurities (in particular, steam from a steam generator tube failure), or from excessive relaxation of mechanical properties at the higher temperatures.

Prestressing systems utilizing a multiplicity of steel cables and anchors have been extensively developed. More recently, designers have preferred cylindrical vessels with unobstructed vertical walls which favor prestressing by the winding of wire on the surface. Development of wire-winding techniques is underway by several manufacturers; one machine employed for this purpose is shown in Figure 11. Wire winding simplifies the vessel wall structure, makes possible greater automation of the prestressing operation, and alleviates embedment congestion in the concrete. Moreover, the development work has shown that sufficient prestress can be applied to the head regions of the vessel by means of circumferential wire winding, thereby removing the need for crosshead tendons. The primary development problems have been to

HT76132

Fig. 10. Installation of fibrous thermal barrier on the liner of the Fort St. Vrain PCRV

HT82934

Fig. 11. Wire-winding machine and prestressing test ring

uniformly tension the wires as they are laid on the vessel and to develop load cells which can monitor the prestressing during the vessel lifetime.

In order to efficiently design a PCR V, it is desirable to utilize comprehensive analytical methods. Three-dimensional elastic analyses and two-dimensional methods which include long-term creep have been developed. These programs are largely based on finite-element methods and are general enough to include the structural action of steel reinforcement, prestressing cables, and the liner. Also nonlinear and ultimate load analysis methods have been developed to account for nonlinearities in materials and loads. These include cracking of concrete and yielding of the liner, reinforcement, and prestressing cables.^{5,6}

In order to provide input data for the analyses and to verify the methods, an extensive program of material studies and model tests has been undertaken by ORNL.⁷ Long-term tests to determine moisture migration and creep as functions of temperature and load are underway. This program features a systematic selection of materials and test conditions.

Prior to the development of analytical methods and the determination of materials properties, it was necessary to utilize model tests for the verification of each PCR V design. Other model tests have been used to study the effect of specific design features on failure modes. Over 100 model tests of various types have been summarized in the literature. One of the most recent test models, shown in Figure 12, is a 1/20 scale model of the large HTGR arrangement. The development of instrumentation for the extensive strain measurements and to determine moisture content has been quite important to the model and materials test programs. The model tests have verified analytical methods and demonstrated that

HT75583

Fig. 12. 1/20 scale model of PCRVR for large HTGR

the behavior of vessels during prestressing, pressure loading, and operation can be predicted. They also have shown that the ultimate failure modes for vessels can be chosen by design to prevent precipitous failures and that very wide margins of safety are inherent when the vessel is properly designed.⁸

HELIUM CIRCULATOR DEVELOPMENT

The decision to develop the series steam-turbine-driven circulator for circulating the primary coolant helium was made early in the overall development program for the integrated type of HTGR design. The use of steam minimized the requirement for electrical power, both on-site for normal operation and off-site for startup and emergency operations. The machines also use water as bearing fluid and lubricant. The development program therefore involved the characteristics of the water-bearing and seal system in addition to the determination of the basic characteristics of the machine itself. In order to satisfy the requirements for redundancy in motive power for emergency conditions, the Fort St. Vrain circulators have an alternate water turbine drive mounted on the same drive shaft.

The development program for this machine was initiated in 1966 and completed early in 1970. The purpose of the development program was to ascertain the suitability of the design throughout the entire possible operational range and to provide detailed data on all of the component parts comprising the machine. The program was carried out in a combination of component tests and on a full-scale prototype machine. Most of the testing was accomplished within a specially constructed test facility located adjacent to the Valmont Power Station of Public Service Company of Colorado near Boulder, Colorado. The test site was selected

early in the development program mainly because of the availability of steam required for testing of the full-scale prototype and production circulator. This facility consisted of a large helium autoclave, shown in Figure 13, in which the circulator was installed in the same manner as in the PCRV. The bottom of the vessel simulated the PCRV penetration liner flange of the reactor installation. The internals of the vessel were arranged to form a closed helium circuit with an ability to vary the flow-pressure drop characteristic so that performance over the entire operating range was possible. In addition, a heat exchanger in the vessel served the function of removing the heat of compression and was used to measure the power.

The first phase of the development program established the dynamic performance of the rotor bearing and seals by means of a bearing test rig. The rig consisted primarily of the prototype rotating assembly with special disks to simulate the mass and inertia of the compressor and turbine wheels.

The aerodynamic performance of the compressor, compressor inlet, and diffuser was then evaluated in air using the water turbine as a power source. The prototype assembly was then tested under simulated reactor conditions with the steam turbine driving the compressor in helium. The operation of the production circulator at simulated design conditions and a final checkout of each production machine prior to installation in the Fort St. Vrain PCRV were then carried out.

The development program uncovered several minor design deficiencies which were corrected as the program progressed through the various phases. In particular, difficulties were encountered with leakage in the seal and buffer parts of the machine, with the ingestion of instrumentation from the test rig into the machine, with higher than desired frictional losses, and with some problems in the braking system.

LC 50107

Fig. 13. Circulator test facility

With all of the problem areas corrected, the complete machines have operated successfully in helium at the design point and over the entire operating range with controls duplicating those in the Fort St. Vrain plant. In addition, successful operational tests were conducted at 50% overspeed conditions corresponding to the design basis accident condition, repeated scram shutdowns with rapid startups on the water turbine, and hot restarts (emergency shutdown with all cooling turned off and restart after equilibrium).⁹

The completed development program has served as a basis for the design of the circulators for the large HTGR described in this paper. As noted, the circulators for the large HTGR will be approximately twice the size of the Fort St. Vrain units in terms of power requirements and mass flow capability. Very few additional development problems are expected with the larger machine.

STEAM GENERATOR DEVELOPMENT

The steam generating units for HTGRs are compact, once-through, single reheat types which are particularly well suited to the integrated type of design. Figure 14 is a schematic diagram of the unit. The main steam bundle is a series arrangement of the economizer, evaporator, and superheater sections. Feedwater enters the unit via a tubesheet located below the primary closure. Orifices are installed at the feedwater tubesheet to provide a stable and evenly distributed flow of steam/water through the bundle. From the tubesheet the tubes are routed to the economizer. At the top of the main steam bundle, the tubes are collected and subheadered. Tube leadouts are routed down the annulus formed between the reheat bundle and the main steam bundle and pass through the primary closure penetration to the superheat tubesheet. The routing of the reheat tube bundle tubes through the tubesheets is similar to that described for

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Fig. 14. Large HTGR steam generator

the main steam bundle. Direct manual access is available for tube-plugging operations from below the tubesheets.

A comprehensive development program on this type of steam generator unit was initiated in 1966 by both Gulf General Atomic and Sulzer Brothers of Switzerland. The areas that have been extensively investigated include: boiling stability, tube bundle and tube lead vibration, materials, gas mixing through the tube bundle, and dry helium effects including wear.

The purpose of the boiling stability tests which were performed in a test facility at Gulf General Atomic was to establish the boiling stability of once-through type steam generators at varying operating load levels. In addition, an analytical code was programmed and evaluated on its ability to analyze the potential stability of the system at various temperature, pressure, and flow conditions. The completely instrumented test loop provided sufficient data to assure that the Fort St. Vrain steam generators, with proper orificing, will operate stably at all power levels throughout the normal operating range.

A number of test programs oriented to investigate flow-induced vibrations and vibrational characteristics of helically-coiled tubes were also undertaken by Gulf General Atomic and Sulzer Brothers. Static natural frequencies of tube leads and helical tubes were studied through the hammer testing of a number of tube configurations, and dynamic tests were conducted on helically-coiled tubes in a helium flow test facility. As the final step in the test program, a major gas flow and vibration evaluation test program is being undertaken by Gulf General Atomic using a full-size Fort St. Vrain steam generator module. This unit has been fully instrumented with thermocouples and strain gages. The test program is being performed to assess the overall heat transfer performance and

vibrational behavior of the complete module. Further testing and evaluation of instrumented steam generators is scheduled in the precritical loop flow tests during the commissioning of the Fort St. Vrain plant.

STEAM GENERATOR MATERIALS DEVELOPMENT

Considerations of performance and economics for the HTGR required that the materials for the steam generator units be chosen specifically for several temperature regions. Thus, test programs have been carried out on a number of materials over the past several years to assure material compatibility with the helium atmosphere within an HTGR system and the superheated steam conditions in the secondary coolant system and to assure material acceptability under long-term, high-temperature conditions while under pressure and thermal stress loadings. On the helium side a high-temperature 18,000-hour metal-coolant compatibility test program was completed, in which a variety of metals was exposed to temperatures of 1100° and 1375° F for long periods of time in pure helium as well as impure helium atmospheres.

One of the materials not fully covered for the Fort St. Vrain units insofar as applicable data for code approval was the Ni-Fe-Cr alloy (Incoloy 800 or equivalent). Therefore, a fatigue test program on this material was run to obtain test data for ASME code approval for use up to 1400° F design conditions. In addition, considerable stress-to-rupture testing was performed to obtain confirmatory 30-year life data at the higher design temperatures for both Cr-Mo and the Ni-Fe-Cr materials.

The inertness and cleanliness of the helium required some unique design solutions related to all moving or sliding parts in the system and specifically to the tube bundle support systems of the steam generators. The formation of an oxide coating on the exposed material surfaces is

inhibited, thereby creating a potential condition of fretting and self-welding. Numerous combinations of materials were run in a series of sliding contact tests at various temperature and helium atmosphere conditions. Based on the results of these tests, various types of hard surfacing were used on mating parts where sliding contact could be expected during plant operation. In addition special sleeve/wedge combinations were developed for the steam generator tube support systems.

On the steam side the regions of welds between both similar and dissimilar metals required investigation since little experience was available for some of the new high-temperature materials. A program by the Oak Ridge National Laboratory and Southern Nuclear Engineering, Inc., has been underway since 1967 to evaluate the behavior of several high-temperature alloys when exposed to commercial-quality superheated steam from the Bartow Power Station of Florida Power and Light Company.¹⁰ Alloy combinations which are being tested at 1100° and 1200°F are shown in Table 3. Tests with these metals have been conducted in both the stressed and unstressed condition and for times up to 12,000 hours. Additional tests are planned for temperatures of 1300° and 1400°F.

In general, all of the materials and the similar and dissimilar metal weldments have performed well except those weldments involving 2-1/4 Cr - 1 Mo low-alloy steel. Suitable filler metals have been identified for each material combination which show no preferential corrosion in the fusion line areas. Tests on U-bend specimens indicate that dissimilar metal weldments are not susceptible to stress corrosion cracking in high-quality commercial steam. A study is presently underway to assess propensities toward stress corrosion cracking in steam containing chloride and oxygen as contaminants. It is planned for the program to include higher temperatures to establish the potential operating range with known margins for reliable long-term operation.

TABLE 3
ALLOY COMBINATIONS TESTED FOR CORROSION RESISTANCE TO STEAM
AT 1100° AND 1200° F

Code No.	Base Metal	Base Metal	Filler Metal
A9	Inconel 625	Inconel 625	Inconel 625
A13	Inconel 625	Incoloy 800	Inconel 625
A12	Inconel 625	Incoloy 800	Inconel 82
A18	IN 102	IN 102	IN 102
A19	IN 102	Incoloy 800	IN 102
A20	IN 102	Incoloy 800	Inconel 82
A5	Hastelloy X	Hastelloy X	Hastelloy X
A6	Hastelloy X	Incoloy 800	Hastelloy X
A8	Hastelloy X	Incoloy 800	Hastelloy W
A7	Hastelloy X	Incoloy 800	Inconel 82
A2	Incoloy 800	Incoloy 800	Inconel 82
A3	Inconel 600	Inconel 600	Inconel 82
A16	2 1/4 Cr-1 Mo	Incoloy 800	Inconel 82
A4	Type 304	Type 304	Type 308

Tube welding techniques also required extensive development due to the compactness of the design and the materials involved. Southwest Research Institute provided the necessary basic welding procedures to perform high-integrity welds for Cr-Mo to Cr-Mo, Ni-Fe-Cr to Ni-Fe-Cr, and Cr-Mo to Ni-Fe-Cr tube joints. Special welding equipment using the pulse-arc power source was developed for both fixed and orbital welding heads. Filler wire type heads were necessary due to the tube wall thicknesses involved. The development of the miniature orbital welding head with filler wire feed was a key in being able to build the compact steam generator units.

In total, the procedures and techniques developed for the Fort St. Vrain steam generator units are readily adaptable for use in the manufacture of the large HTGR steam generating units.

REFUELING EQUIPMENT DEVELOPMENT

The evolution of fuel handling equipment for the HTGR has followed closely the technical advancements in the reactor design. As the reactor size has increased the number of individual elements to be handled has increased, and the time limitations inherent in the specifications for minimizing the refueling times has required the development program to not only emphasize reliable operation of the refueling equipment but also to include more computer-programmed-type automatic operations.

The fuel handling equipment for the 40-MW(e) Peach Bottom plant represents the first family of components for handling HTGR fuel during a reactor shutdown. The 804 fuel elements are about 3-1/2 in. in diameter and 12 ft long, and weigh about 90 lb each. For storage and shipping purposes each element upon removal from the reactor is canned, sealed, and leak-tested before being stored in a water-filled storage pool.

The Peach Bottom refueling system was not designed with a primary objective of high speed operations. As such the controls are manual with a substantial amount of interlocks and limit settings. The operations are performed by trained operators but with no automatic timing or preprogramming involved. An in-core viewing system consisting of an optical system in two modes, depending on the desired viewing area, is provided. The device utilizes its own reactor isolation valve and may be placed on either of two pressure vessel nozzles provided for this purpose.

With the development of the large HTGR, it was realized that certain changes were needed in the area of refueling equipment. The time period for refueling had to be compatible with acceptable annual shutdown periods. Also, the change from bottom-mounted to top-mounted control rod drives required refueling by region rather than considering the entire core as one region. Moreover, the hexagon-shaped short length element resulted in several modifications. Viewing device development followed the path of finding more compact systems, thus permitting them to be built directly into the fuel handling machine. Closed circuit television appeared to be the answer, but required the resolution of radiation and thermal problems.

As a result of these requirements, there are significant differences between the fuel handling equipment concepts for the Peach Bottom and Fort St. Vrain plants. For example, the Fort St. Vrain fuel transfer machine and fuel transfer cask are built as one unit, thus reducing transfer trips from reactor to storage. Controls have been made automatic utilizing computer and tape, thus reducing the human element considerably. Fuel and reflector elements are stored in a dry uncapped condition, which eliminates the need for canning operations. To counteract the effects of multiple vessel penetration fuel handling operations, parallel operations

were introduced, primarily to prepare and to close up a core region prior to and following actual refueling operations. Refueling of the Fort St. Vrain reactor is done, as at Peach Bottom, with the reactor shut down and depressurized.

The Fort St. Vrain fuel transfer machine has been subjected to an extensive testing program. A large portion of the tests have been associated with the development of the automatic control system. The test program, which will be completed this summer, will culminate with tests in purified helium and at elevated temperatures. Proof of operational adequacy will be by demonstrating simulated refueling operations using graphite dummy elements. For the balance of the fuel handling equipment, shop and site checkout tests at ambient conditions have proven quite adequate.

The most recent developments in HTGR refueling technology pertain to the large HTGR. The equipment that will be used is shown in Figure 15.

The fuel design from a handling viewpoint remains identical to that of Fort St. Vrain. However, due to the core size, the 73 refueling regions, and a deeper core, certain basic innovations in the fuel handling system design are required in order to achieve an acceptable refueling time during a reactor shutdown condition.

The large HTGR is also contained in a conventionally reinforced concrete containment building with fuel storage and shipping facilities located outside of the containment shell. Fuel elements are transported in and out of the containment building access opening in 4-element casks on self-propelled vehicles. The fuel handling machine is stored in the adjacent reactor service building between refueling periods for service and operator training.

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Fig. 15. Fuel handling equipment

SUMMARY

This paper has briefly described the major development areas of the HTGR system. Research and development has been carried out in several other areas.¹¹ Although not covered in this paper these areas have been equally important in the overall program which has now led to the ability to offer the HTGR nuclear steam system for the generation of base-load electrical power on a commercial basis.

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