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T. G. LeClair

Presentation at the
National Power Conference
San Francisco
September 25, 1961

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JOHN JAY HOPKINS LABORATORY FOR PURE AND APPLIED SCIENCE

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INTRODUCTION

This paper has been listed as "The Status of the High-temperature Gas-cooled Reactor." To me this includes not only the present but some of the history, and a look at the possibilities for the future. The status of the gas-cooled reactor must be kept in perspective with other types of reactors which have been and are being developed throughout the world. Every country has different fuel reserves and different materials available with which to build reactors. It therefore is not surprising that on a world-wide basis, reactors have not developed along the same patterns as in the United States.

In this country, the majority of attention has been given to water-cooled reactors and particularly to light-water-cooled reactors, which require operation with enriched uranium fuel. On the other hand, in the United Kingdom, where the largest kilowatt capacity of power reactors has been constructed, the situation is quite different. That country does not have an abundant supply of enriched nuclear fuel and, for economic as well as political reasons, began the development of the gas-cooled type of reactor. Also, in the early 1950's, Great Britain and other European countries were faced with a serious shortage of fossil fuels and paid a very high price for such fuels. They were therefore driven to rapid development of nuclear power based on the principles which were available to them at the time.

With the foregoing background, Great Britain and later France sought the type of reactor which could be started the earliest and with the assurance that it could be made to operate on a continuous basis. This was the graphite-moderated natural-uranium-fueled reactor of the Calder Hall type. Many of us do not realize the extent to which the development of this type reactor has progressed. Figure 1 shows the total megawatt capacity of various types of reactors which are either under construction

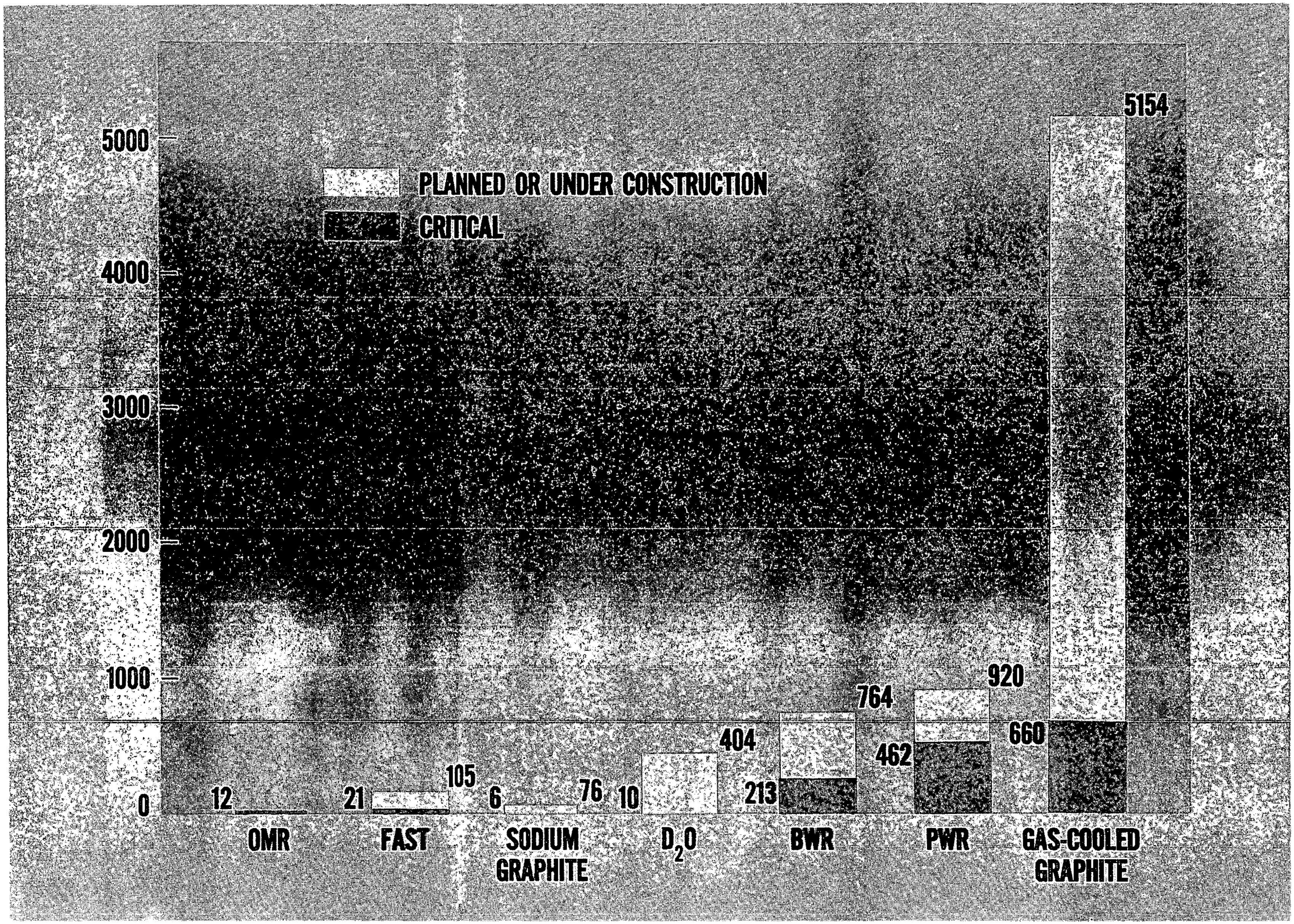

Fig. 1--Bar chart of power reactors' generating capacity (by types)

or in operation throughout the world. This indicates very clearly that the graphite type, including low-temperature and high-temperature, substantially exceeds the capacity of all other types.

The original Calder Hall reactors were designed for relatively low gas temperatures. The turbine generators necessarily were designed for low steam pressures and temperatures and, consequently, have low operating efficiency. This performance was considered satisfactory at the time as coal in Great Britain then cost $20 per ton. More recently, the sharp drop in the price of coal and other fossil fuels in Europe has slowed down the British program but by no means has stopped it. It did, however, bring about considerable incentive to improve nuclear plant efficiency. The route to improved efficiency is better steam conditions, which are obtained from higher gas outlet temperatures. Figure 2 shows the rapid rate at which gas outlet temperatures increased from one project to the next during the last eight to ten years.

Most of the original reactor work was based on fuel made of natural metallic uranium and clad with metal to contain the fission products and reduce corrosion. Metal-clad fuel has appeared to reach the maximum gas outlet temperature which can be tolerated economically. It was this limiting circumstance which led not only the British but the General Atomic organization in this country to consider the use of different cladding for the fuel element.

Both the Dragon Project in England and the HTGR Project in the United States recognized the desirability of attaining temperatures in the core higher than any metals can tolerate. These Projects have adopted the concept of a fuel element clad in graphite. There are no metals to absorb neutrons or to lose their strength at high operating temperatures. The Dragon Project, which is sponsored by the Office of European Economic Cooperation, was started first. This project's small experimental model of 20 Mw of heat output but without an electric generating unit will use graphite for the moderator, helium gas for the coolant, and uranium-thorium-

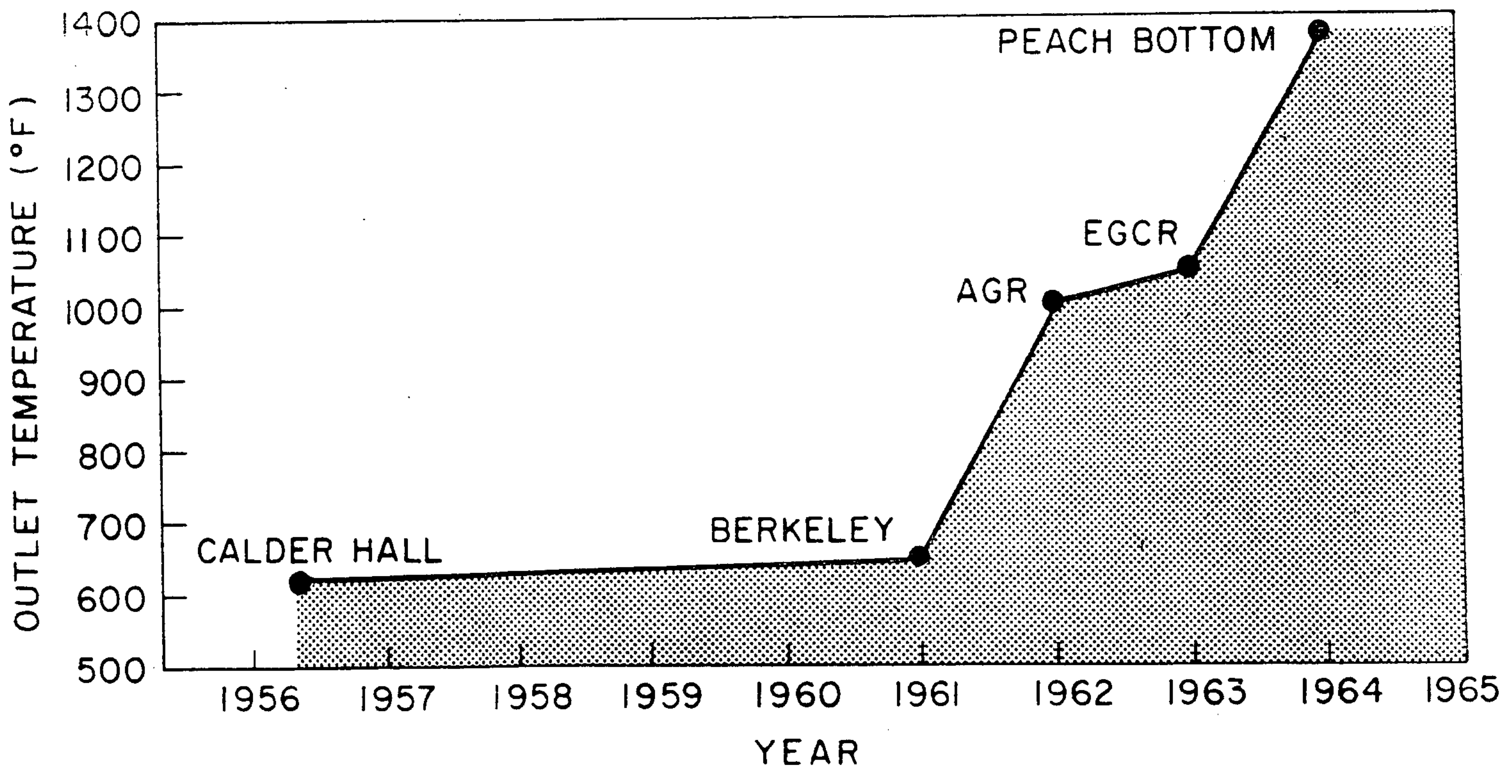

Fig. 2--Improvement in gas outlet temperatures of gas-cooled reactors

carbide for the fuel. The American project--the HTGR--is building a complete gas-cooled prototype power plant that will produce 40 Mw of electrical output. The prototype will serve as an initial experiment; it will also prove that this type of power plant can operate as a generating unit on the system of an electric utility. A technical description of this project follows.

DESCRIPTION OF THE PEACH BOTTOM POWER PLANT

The 40-Mw prototype HTGR plant is being designed to demonstrate those features that in large systems are believed necessary for the attainment of economic power through effective use of modern steam conditions. Table 1 gives a summary of the estimated performance characteristics of this plant. You will notice that it is being designed to produce steam at 1450 psi and 1000°F. These steam conditions are somewhat higher than are warranted by the size of the plant; however, they are used for the prototype plant to demonstrate the operating features intended for larger plants.

Figure 3 shows a simplified flow diagram for the HTGR system. The reactor coolant, which is helium at approximately 350 psi, flows through the reactor core, where it is heated to 1380°F. From the reactor it is directed to the steam generators to generate 366,000 lb/hr of steam. The coolant is returned at 660°F from the steam generators to the reactor by helium circulators. Two identical loops, each including a single steam generator and circulator, are used; each loop contributes half the total plant output. Forced circulation-type steam generators have been selected for the prototype plant. Because of the good heat-transfer properties of helium at 350 psi and because of the high temperatures available, heat-transfer rates in the steam generators are high. Consequently, the surface-area requirement and size of the steam generators are kept to a minimum. Each of the steam generators is approximately 8 ft in diameter by 30 ft in height.

Table 1

TECHNICAL DATA FOR HTGR

Operating Conditions

Coolant	Helium
Pressure, psi	350
Inlet temperature, ^{o}F	660
Outlet temperature, ^{o}F	1380
Maximum fuel-surface temperature (approximate), ^{o}F	2300
Steam temperature, ^{o}F	1000
Steam pressure, psia	1450
Net thermal efficiency, %	34
Reactor thermal output, Mw	115
Net electrical power, Mw	40

Reactor Parameters

Fissile investment (U^{235}), kg	172.2
Thorium investment, kg	1987
Core diameter, ft	9.0
Core length, ft	7.5
Core coolant voidage, %	12.8
Fuel-element diameter, in.	3.5
Number of fuel elements (approximate)	810
Reflector thickness, ft	2
Pressure-vessel diameter (inside), ft	14
Pressure-vessel height, ft	34
Burnup (approximate), Mw-days (heat) per metric ton of U^{235} + Th^{232}	60,000
Core life (at 80% load factor), yr	3

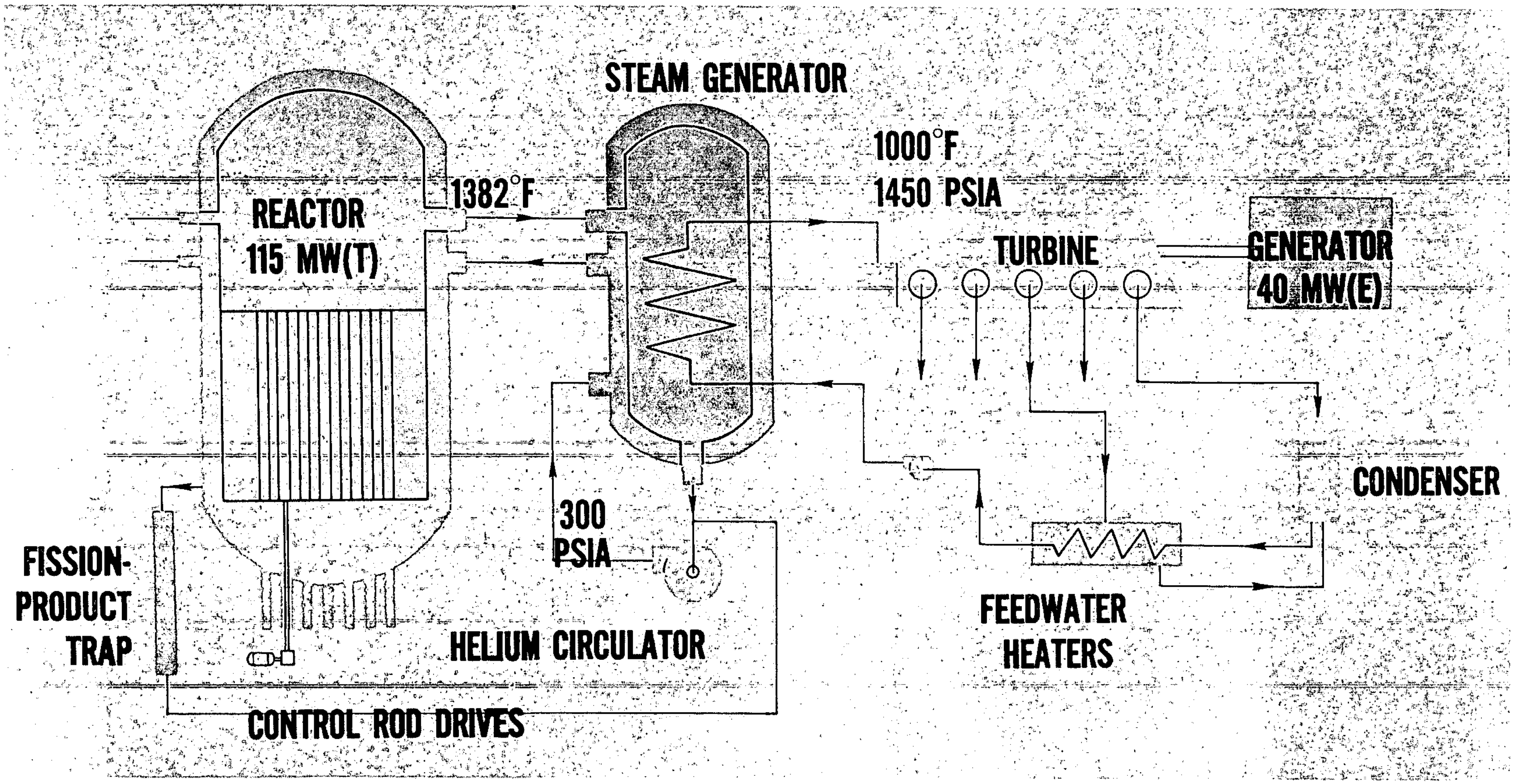

Fig. 3--HTGR flow diagram

The helium circulators are of a single-stage centrifugal design with overhung impellers and a shaft seal to minimize helium leakage. The drive power required for each circulator is about 2500 hp--utilizing about 9% of the electrical output of the plant. The pumping power required for the gas coolant is more than offset by the high thermal efficiency available with the high gas temperatures, so that the over-all net plant efficiency is about 34%.

The construction of the reactor itself is shown in Fig. 4. Helium enters the reactor vessel through the annulus of the concentric pipe and cools the vessel prior to flowing upward through the reactor core. The 660°F inlet gas permits the use of carbon steel as a construction material for the vessel. This inlet gas temperature is high enough to minimize radiation damage to the graphite reflector and to greatly reduce Wigner stored-energy problems. It is also high enough so that the steam cycle can be selected without special consideration being given to pinch-point temperature difference in the steam generator.

The reactor core proper is 9 ft in diameter and 7-1/2 ft high. It is surrounded on all sides by approximately 2 ft of graphite, which serves as the neutron reflector; top and bottom reflectors are integral parts of the fuel element. The core is made up of approximately 800 graphite fuel elements arranged in a closely pitched, equilateral triangular array. The helium coolant flows through the core in the tricusp-shaped passages between the fuel elements. The elements are supported at their bases by a steel support plate and are maintained in alignment at the top by the side reflector, which is designed to provide an elastic restraint for the core. This type of construction eliminates the need for a steel support structure at the hot-coolant outlet face of the core and thus permits the attainment of high gas temperatures.

Owing to the long fuel lifetimes available with HTGR-type fuels, only very infrequent refueling of the reactor is necessary, e.g., approximately once every three years in the case of the prototype plant. Consequently, the reactor can be shut down for refueling and the coolant pressure lowered so that relatively simple fuel-handling equipment can be employed.

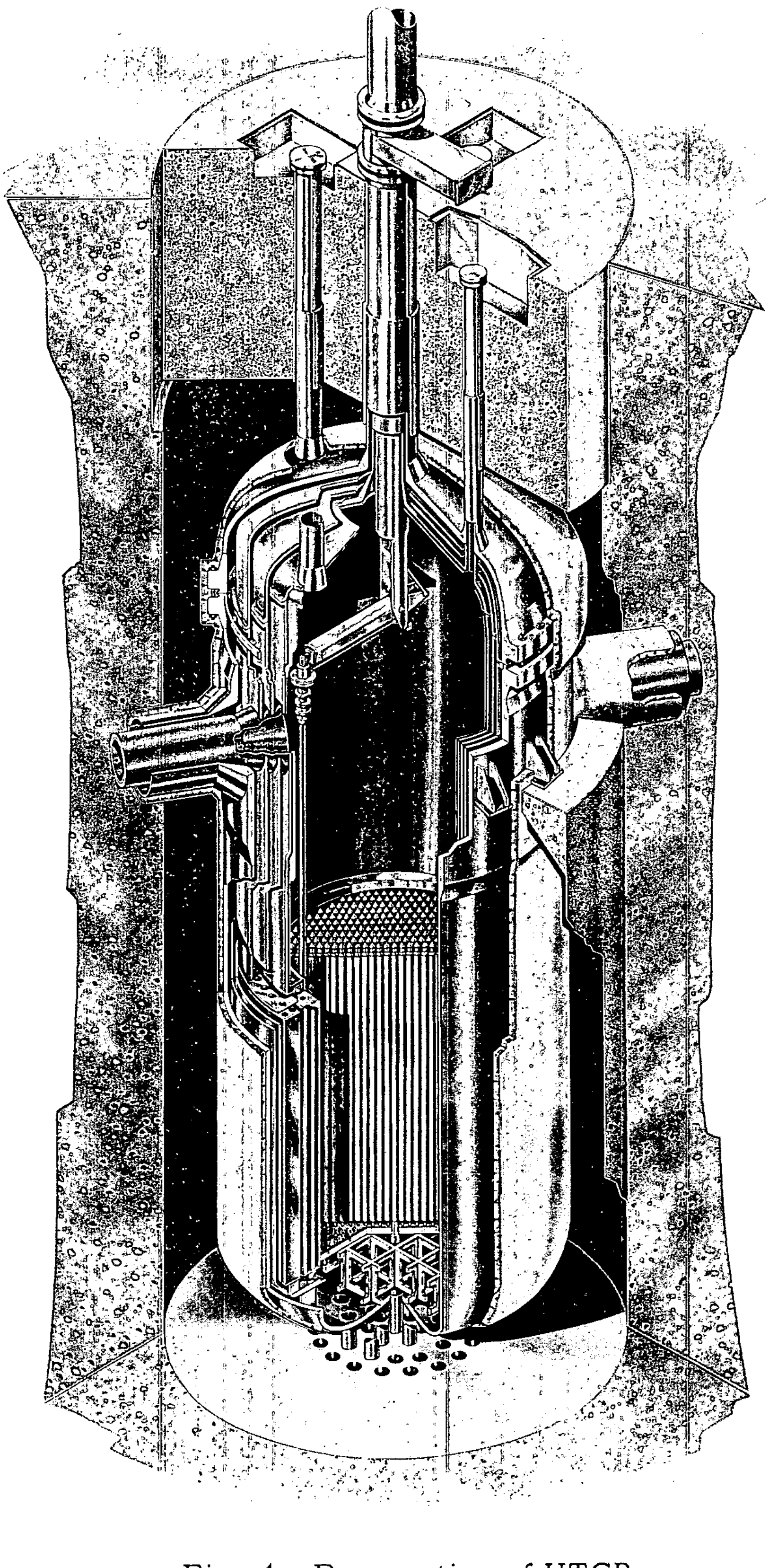

Fig. 4--Perspective of HTGR

The detailed arrangement of the fuel element is shown in Fig. 5. The fuel, consisting of uranium and thorium dicarbide particles about 100 to 200 microns in size, is contained in annular graphite fuel compacts (2-1/2 in. OD). The fuel element is made up of a series of these fuel compacts contained within a graphite sleeve, the sleeve outer diameter being 3-1/2 in. The fuel compacts occupy 7-1/2 ft within the graphite sleeve. The over-all length of the fuel element, including a graphite reflector at each end, is approximately 12 ft. By distributing the fuel particles in a graphite matrix, fission damage is restricted to the immediate vicinity of the fuel particles, leaving most of the matrix undamaged. Thus, very high fuel burnups are achievable without gross deterioration of the fuel structure. For the prototype reactor, the average fuel-element burnup is expected to be approximately 60, 000 Mw-day/ton of uranium plus thorium. Looked at in another way, each fuel element in the core will produce about one million kw-hr of power.

Since at the high temperatures of the HTGR, some fission products have relatively high vapor pressures, it is necessary that the fuel-element design incorporate a system for controlling the release of fission products into the helium coolant. This has been accomplished by the use of graphite sleeves made from highly impermeable graphite. A small purge stream of helium is caused to flow between the sleeve and the fuel compact so that fission products diffusing from the fuel are carried by the purge stream to internal and external trapping systems, thereby minimizing the fission-product contamination of the main helium coolant. The system is being so designed that contamination of the coolant will be limited, making possible essentially direct maintenance of the reactor plant equipment after shutdown.

In addition to the systems described above, several auxiliary systems are necessary to the operation of the HTGR nuclear steam supply. Principal among these are (1) a helium-purification system for maintaining impurities in the helium at a low level so as to prevent mass-transfer of fission products, and (2) a helium-handling system for charging the reactor system

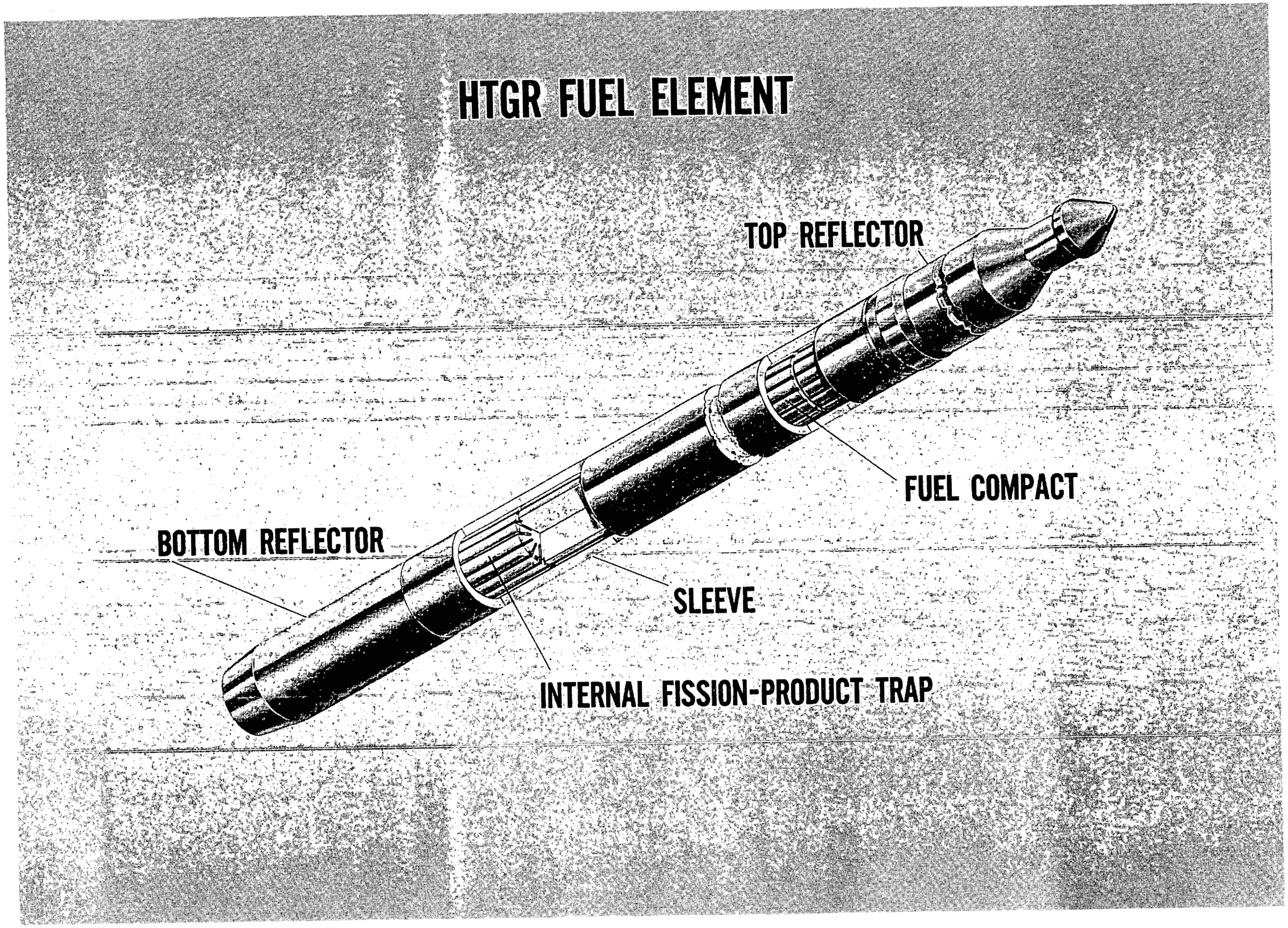

Fig. 5--HTGR fuel element (single wall)

with helium and maintaining the helium coolant at a pressure of 350 psi. The helium-purification system is a bypass type designed to purify the entire helium inventory about once every four hours.

The turbogenerator system is of conventional design. The turbine is a 3600-rpm, tandem, compound double-flow, condensing unit of 40,000 kw net electrical capacity, with 1450-psi 1000°F steam at the throttle. In addition to serving its usual functions, the steam condenser also acts as the heat sink for the main steam flow during startup or emergency shutdown conditions. There are four feedwater heaters.

The over-all plant control system is being designed so that the plant will be load-following. This system is a fairly straightforward adaptation of a conventional power-plant control system. An increase in load causes an increase in the helium circulation rate, which is followed by an increase in reactor power level as a consequence of a prompt negative temperature coefficient of the reactor. Steam temperature is maintained over the load range by automatic regulation of the reactor control rods, which determine the reactor outlet gas temperature and consequently the steam temperature.

The arrangement of the station is shown in Fig. 6. The entire nuclear-steam-supply system is housed within a steel containment shell that is approximately 100 ft in diameter and 130 ft high. Because of the low stored energy in the helium coolant and the absence of any chemical reaction between the helium coolant and the reactor components, the energy released during maximum-accident conditions is low. Consequently, for HTGR-type reactors, the containment structure can be designed for very low pressure rises. For example, in the case of the prototype reactor, the maximum pressure rise is only a fraction of an atmosphere.

The turbogenerator-system construction is of the outdoor type. The turbogenerator auxiliary systems, the reactor control room, and the service systems, as well as offices and shops, are contained within the building adjacent to the containment structure.

The basic design parameters for the Peach Bottom project have been

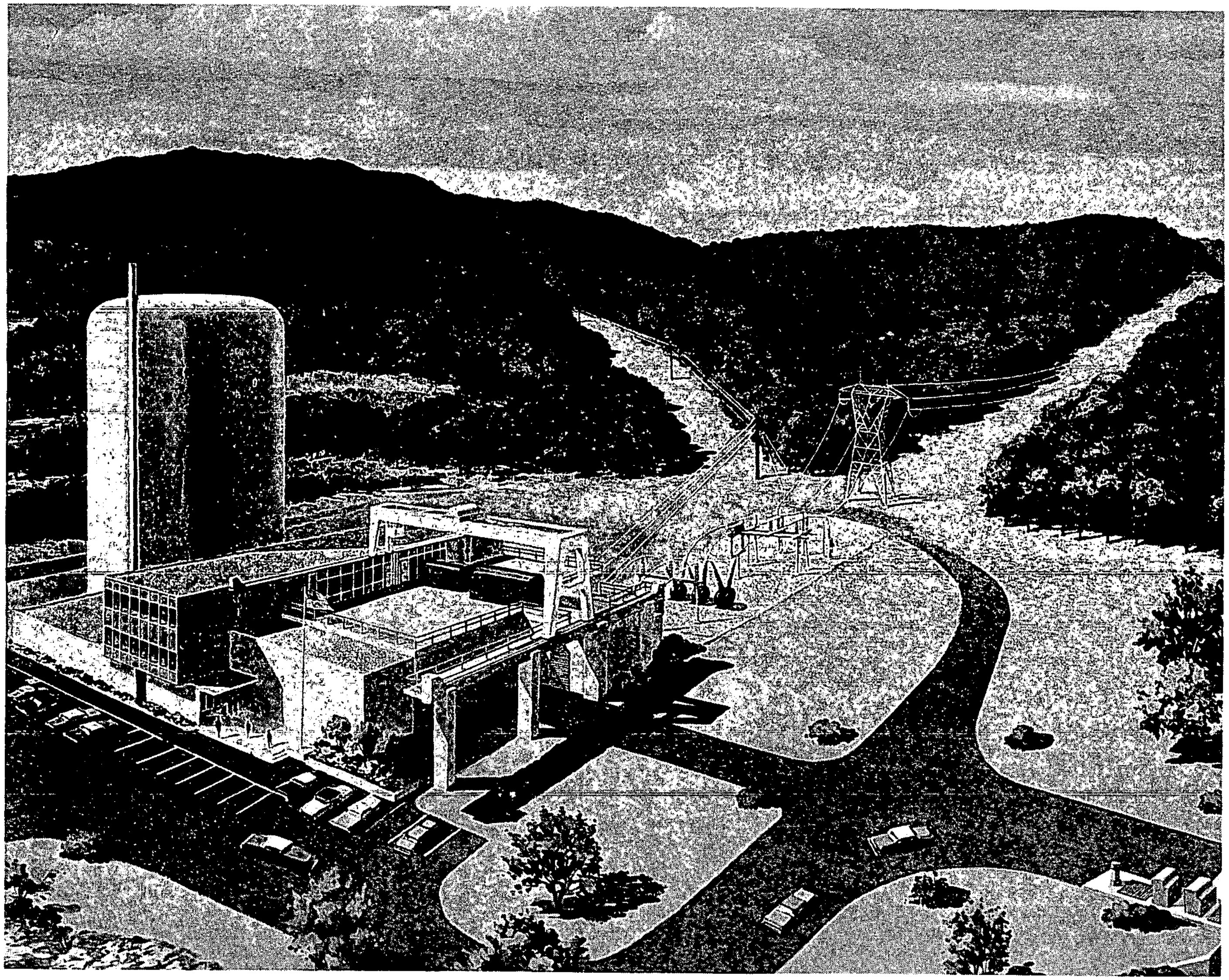

Fig. 6--Arrangement of HTGR Peach Bottom station

established. Also, a very large portion of the total research and development work necessary before beginning construction has been completed. The remaining research and development is primarily to prove out components and operating characteristics with a greater degree of detail. A year ago the U. S. Atomic Energy Commission approved the site for the plant. The application for an AEC construction permit has been amended by extensive information on the details of the design and an analysis of possible emergency conditions. Assuming this information satisfies the regulatory body and a permit is issued on schedule, the construction should be completed in the spring of 1964.

In the meantime, the High Temperature Reactor Development Associates, Inc. (which is comprised of fifty-three utilities companies), the Philadelphia Electric Company, Bechtel Corporation, and General Dynamics have increasing confidence that this type of reactor can be made to be the most efficient--producing steam at the modern pressure and temperature conditions necessary for high efficiency and relatively low power cost. Based on this enthusiasm from the research and development program, there is increasing interest in larger projects to follow and in improvements on the Peach Bottom prototype.

ESADA

Recently a project was started at General Atomic to undertake the preliminary design and development of a larger-sized plant in the range of 300 to 500 Mw of electrical output which would have an over-all efficiency of at least 40%. This program is jointly sponsored by the Empire State Atomic Development Associates, a group consisting of the seven privately owned New York State utilities, and General Atomic. This program is planned to extend over a three-year period and is expected to show the economic feasibility of producing power in New York State with a large HTGR plant. The research and development staff has been assembled at General Atomic and the first tentative layouts completed.

RECENT DEVELOPMENTS

Of the many problems associated with the development of a new reactor concept, the most challenging, historically, are concerned with core materials. Also, in the development of a reactor concept, several milestones mark the progress of the technological advances. The HTGR follows this general pattern, and it is believed that several milestones have recently been attained.

In this type of reactor the fuel, which is dispersed in graphite, is prepared in the form of uranium-thorium dicarbide particles, $(U, Th)C_2$. To aid in retaining fission products within the fuel compacts for long periods during reactor operations and to protect the carbide particles from moisture and thereby provide air-stable fuel compacts, a successful method of coating these particles has been developed. The coating substance is pyrolytic carbon. (A photomicrograph of such a coated particle is shown in Fig. 7).

Another recent giant step in graphite core technology is in the development of fabrication procedures for highly impermeable graphite. It is now possible to produce a 10-ft-long graphite tube that satisfies our present rigid permeability requirements. Furthermore, although the methods used to produce impermeable graphite have been for small quantities, they are simple enough to be adapted to large-volume production.

A third achievement in graphite core technology is the promising results obtained from the irradiation of fuel. The $(U, Th)C_2$ particles coated with pyrolytic carbon and dispersed in a graphite matrix were irradiated in a test reactor. Such fuel compacts closely simulate the required reactor fuel. The test results are given in Table 2. The fuel compacts exhibited good dimensional stability and strength.

In order to provide a complete check of small fuel-irradiation test samples, it was found desirable to establish an in-pile test facility. This test loop is sized to contain a full-diameter fuel element and is designed to operate at full temperature using helium as the coolant. The test loop also

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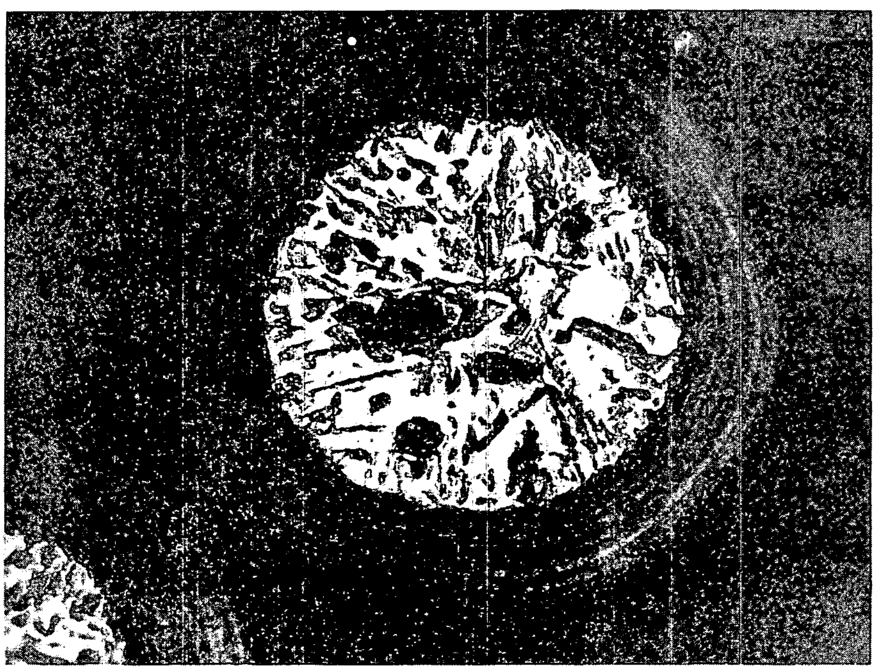

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Fig. 7--Photomicrographs of coated fuel particles

Table 2

DIMENSIONAL AND PHYSICAL STABILITY OF LOW-PERMEABILITY GRAPHITE IN CAPSULE GA-308-2 AFTER IRRADIATION

Container Number	Type of Graphite	Irrad. Temp.* (°F)	Preirradiation Size (in.)		Postirradiation Contraction (%)		Strength to Fracture† (psi)	
			Diam.	Length	Diam.	Length	Preirrad.	Postirrad.
2	RXLP-2	1,275	0.2260	0.2254	0.05	0.03		43,040
	CEY	1,275	0.1243	0.1259	0.03	0.18		-----
Control	RXLP-2						16,200	
	CEY						17,070	
3	CEY	1,400	0.1265	0.1238	0.12	0.09		16,520
	HS-142	1,400	0.2263	0.2271	0.06	0.03		20,120
Control	CEY						17,600	
	HS-142						15,790	
4	HS-4	1,500	0.2252	0.2256	0.02	0.03		23,410
	CEY	1,500	0.1260	0.1252	0.11	0.03		-----
Control	HS-4						15,590	
5	CEY	1,550	0.1260	0.1252	---	---		16,120
	RXLP-2	1,550	0.2259	0.2264	0.06	0.02		15,190
Control	CEY						17,480	
	RXLP-2						15,500	
6	CEY	1,475	0.1260	0.1254	0.10	0.07		16,530
	HS-142	1,475	0.2249	0.2257	0.02	0.03		24,680
Control	HS-142							17,800

*Approximate thermal-neutron exposure was 3×10^{20} nvt.
†Tests were conducted at room temperature.

checks the release rate of fission products. A schematic flow chart of the test loop is shown in Fig. 8. The fuel compact is encased in a thimble located next to the reactor core near the bottom of the reactor pool.

The HTGR Critical Facility, shown in Fig. 9, has been in use since July, 1960. It consists of two identical graphite-fuel assemblies, each of which contains a subcritical mass. When brought together, the assemblies form a cube approximately 6 ft on a side and constitute a critical mass. The critical facility has been designed so that a region of about 16-in. diameter, extending through the center of the cube and perpendicular to the plane of separation, is a replica of the HTGR core lattice. This region is surrounded by a flux-matching driver region.

With this assembly, experiments have been conducted to check the analytical work performed in setting nuclear parameters. We have been heartened to obtain analytical work which agrees very closely with the experimental results. An example of the correlation is shown in Fig. 10. The Doppler contribution to the temperature coefficient of the reactor as calculated has been compared with the experimental data and agreement is within 5%.

SUMMARY AND CONCLUSIONS

The development of the gas-cooled reactor has followed an orderly procedure on its course toward economic power. The first European reactors using natural uranium had successful careers but under conditions where enriched uranium is costly and where coal likewise is a high-priced competitor. It would appear that for these reactors the designs using metal-clad fuel are approaching their practical limit in temperatures and efficiency.

Effort is now concentrated on graphite-clad fuel elements to obtain the high temperatures needed for most modern power-plant steam conditions. The Dragon Project in England and the Peach Bottom Project in Pennsylvania are the first reactors of this type. The Peach Bottom reactor is the first

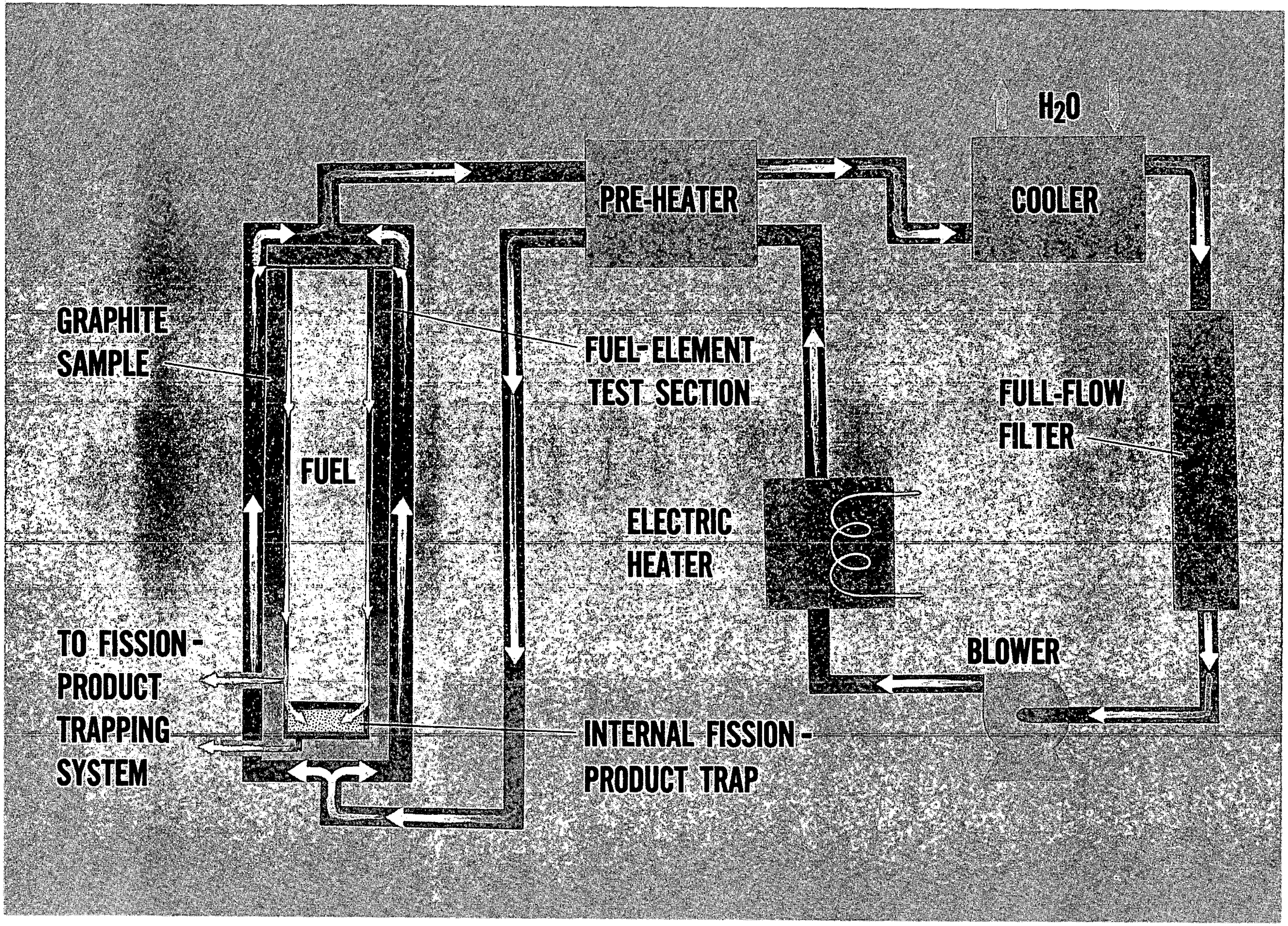

Fig. 8--Schematic of GETR loop

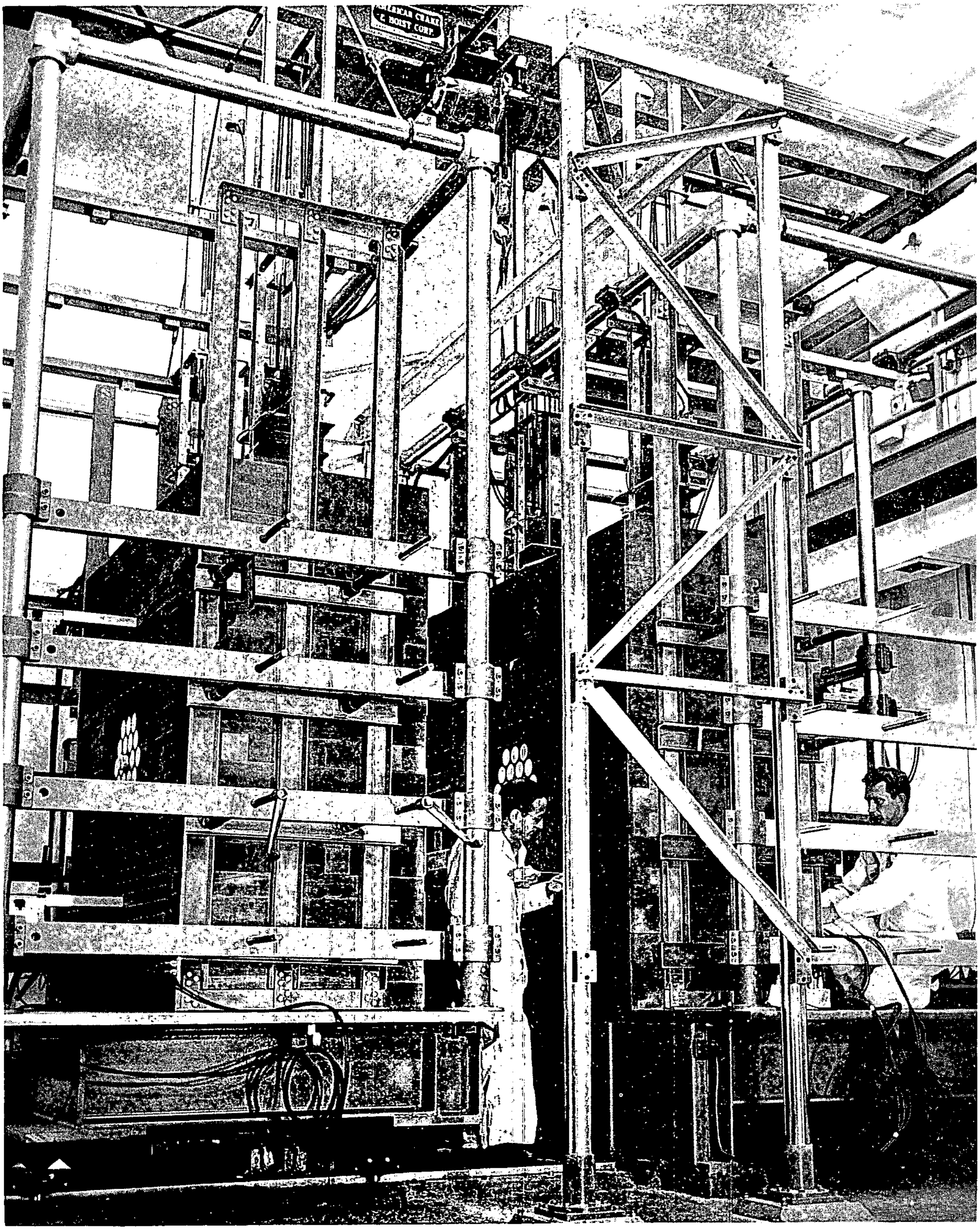

Fig. 9--HTGR critical assembly

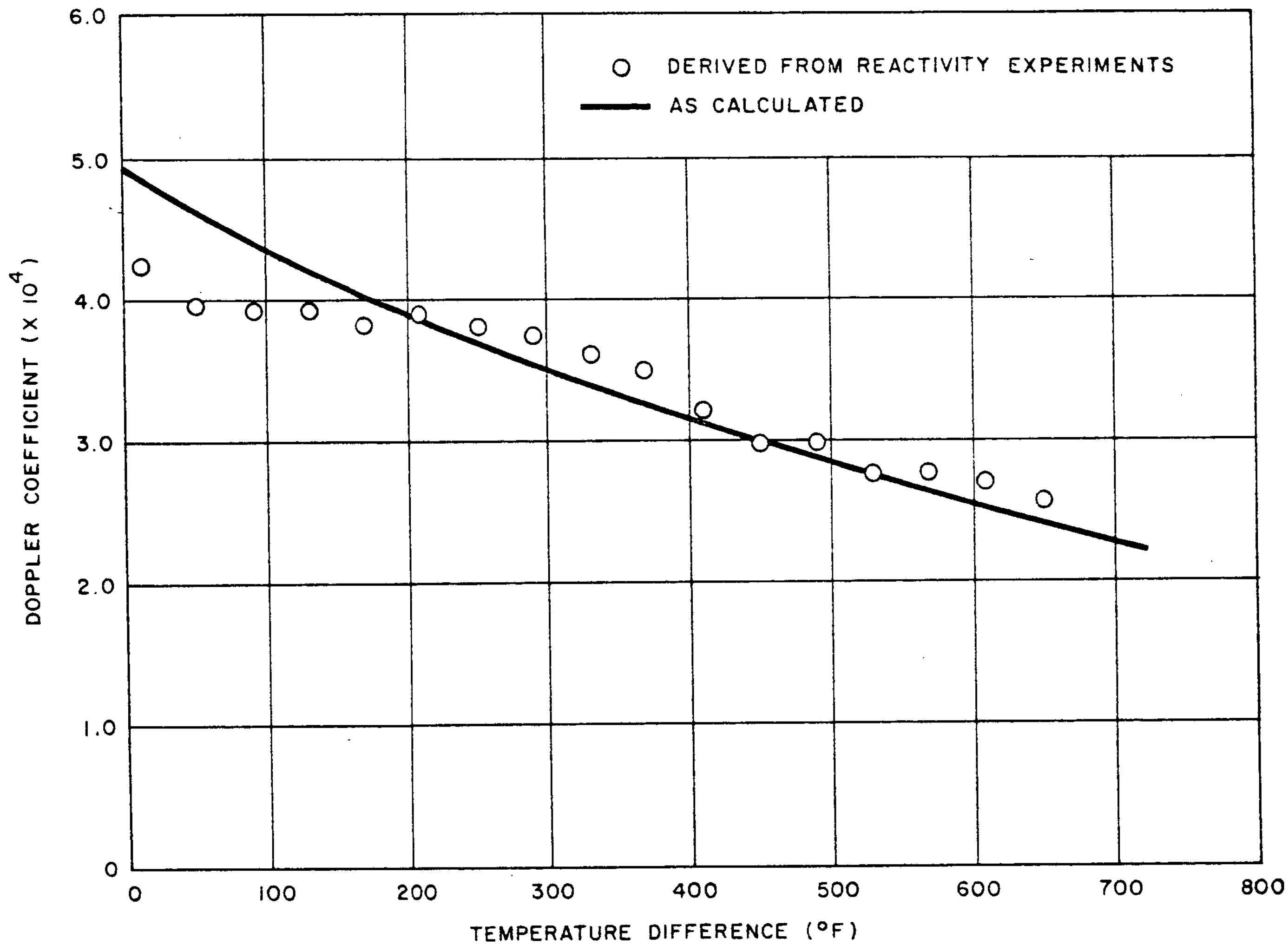

Fig. 10--HTGR Doppler coefficient vs. hot-cold temperature differential

such reactor to have power generation included in the project. This 40 Mw plant, because of its small size and because of its pioneering, has a high capital cost.

Work now proceeding under the ESADA program is on a development and design study directed toward a full-scale plant in the range of 300 to 500 Mw electrical capacity. The expected goal of this program is a plant which can produce electric power at a cost competitive with fossil fuels.

Among the new developments having a potential for improvement is the use of coated fuel particles. Fuel particles coated with pyrolytic carbon or other material can retain fission products until an appreciable fraction of them have decayed. This makes possible the design of a simpler and more economical fission-product trapping system.

Many mechanical developments are being carefully studied. It should be possible to shorten the flow path for the helium coolant and to develop methods of circulating helium which are substantially more economical. It should also be possible to go to higher helium pressures after the helium circulators have been run at the present conservative levels.

Further study should be undertaken to determine whether the plant can ultimately operate on a complete thorium cycle--by using thorium enriched with U^{233}, the life of the fuel elements will be still further extended.

The availability of very high temperature gas as a reactor coolant makes it practicable to have an optimum design of power plant using boiling, superheating, and reheating with only one reactor heat source. Preliminary studies show that in spite of the inherent extremely low fuel cost for the HTGR, reheat will probably be justified by its ability to produce plant efficiencies of about 40% and a consequent reduction in reactor size for a given power output.

In summary, the general category of gas-cooled reactors has had the benefit of more operating experience than any other types of reactors. However, the high-temperature gas-cooled reactor is the newest improvement, with which there has been no operating experience. It is believed that developments based on the operation of the first prototype will result in power plants of a very low over-all power cost.