Cooled
Fast Breeder Reactor Designs

Part 1—The 300-MW(e) GCFR Demonstration Plant

Various studies of gas-cooled fast breeder reactor (GCFR) systems with both steam and gas turbine cycles have been performed in Europe and the U. S. for about 10 years. Recently, Gulf General Atomic designed a 300-MW(e) demonstration plant under the sponsorship of a group of U. S. utilities and performed safety studies for this system. Here, the authors discuss this plant with its indirect steam cycle and safety features. Next month, Part 2 of this article will be devoted to recent performance studies of large—1000-MW(e)—GCFR plants.

J. B. DEE and G. B. MELESE-d'HOSPITAL[1]
Gulf General Atomic Co., San Diego, Calif.

THERE are approximately 50 gas-cooled power reactors now in operation or under construction around the world, mostly in Europe. The main types are Magnox reactors, using natural uranium metal rods with Magnox cladding, and advanced gas-cooled reactors (AGR) with stainless-steel-clad enriched-uranium-oxide rods. Both types are thermal reactors moderated by graphite and cooled by carbon dioxide; they produce about 15,000 MW(e), which is a large fraction of the world's nuclear electric generating capacity. The next version of the thermal gas-cooled reactor is the high-temperature gas-cooled reactor (HTGR) developed in the U. S. and in Europe; it is also moderated by graphite but it is cooled by helium, and the fuel consists of enriched uranium and thorium carbides in the form of coated particles. This leads to a high exit temperature (~800 C) and a high net cycle efficiency (~40 percent). The first U. S. prototype, the 40-MW(e) HTGR at Peach Bottom, Pa., has been producing power since 1967, and two smaller reactors are operating in England and in West Germany.

A 330-MW(e) HTGR is now under construction at Fort St. Vrain, Colo., and should be in commercial operation in 1972; like all the latest gas-cooled power reactors, the whole nuclear steam supply system is contained within a prestressed concrete reactor vessel (PCRV). Two other HTGRs are also planned for construction in the near future, one in England and the other in Germany, and two 1160-MW(e) HTGRs have been ordered in the U. S.

Fuel utilization in such advanced converters as the HTGR is nearly twice as good as that for "burners," such as light-water reactors (BWR or PWR), but still better utilization of the uranium resources could be obtained with fast breeder reactors where natural or depleted uranium is converted into more fissionable plutonium than is consumed in the reactor itself. About 50 percent of the U-238 in a fast breeder reactor (FBR) could be converted into fissile plutonium in a

---

[1] Mem. ASME.

Based on a paper contributed by the ASME Nuclear Engineering Division. This work was supported in part by the member utility companies participating with Gulf General Atomic in the GCFR development program.

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

Although the first electric power ever produced by a nuclear reactor came out of the first fast breeder [0.2 MW(e) in EBR-I in 1951], it will have been 20 years before the first sizable fast breeder demonstration plant will be operative (the BN 350 in the USSR). Two other demonstration plants are scheduled to start up in 1972 and 1973 in England and France, respectively. The first demonstration plant in the U. S. will probably not be operating before 1978. All of these fast reactors are cooled by a liquid metal.

Four coolants have been considered for fast breeder reactors: liquid metals (e.g., Na or NaK), steam, helium, and carbon dioxide. Sodium has several advantages as a fast-reactor coolant, such as good heat-transfer characteristics at low pressure and high temperature and good emergency cooling characteristics, but it is an opaque fluid that can boil or freeze, is active chemically, and becomes radioactive in the reactor. Therefore, a great deal of effort is needed to develop reliable components such as steam generators. An intermediate liquid-metal heat-transfer circuit is required to avoid the possibility of steam entering the primary circuit and reacting with the radioactive sodium. The metallurgical and safety problems that would arise from the use of steam as a fast-reactor coolant are much less severe with helium and carbon dioxide. Helium is chemically inert, does not become radioactive, does not change phase, is transparent, and does not degrade the neutron spectrum, thus leading to a high conversion ratio and a negligible void reactivity coefficient. Heat-transfer characteristics of helium under typical fast-reactor operating conditions are not much different from those of sodium [1],[2] especially since the surface heat-transfer coefficient can be significantly increased (≥2) by artificial roughening of the fuel-rod surface. Although pressurization is required (70 to 85 atm), the fact that the whole primary system is totally enclosed within a PCRV makes a rapid depressurization accident highly improbable. The combination of a pressurized secondary containment and several independent main and auxiliary cooling loops helps to alleviate emergency cooling problems since natural convection in helium is usually insufficient [2]. Carbon dioxide has properties similar to those of helium but it could create corrosion problems.

Several types of gas-cooled fast breeder reactors (GCFR) have been proposed in the past decade but only two are being seriously considered: conservative designs using stainless-steel-clad, mixed plutonium and uranium oxide fuel rods cooled by helium, with an indirect steam cycle; and advanced designs with vanadium-clad rods or ceramic-clad, mixed plutonium and uranium carbide-coated particle fuel, with a direct helium gas turbine cycle. Most of the efforts spent on design development in Europe and in the U. S. have been on the first type of GCFR, which is based on LMFBR fuel and physics development and on HTGR technology, such as the PCRV, circulator, and steam generator. This deliberate choice should lead to development of a GCFR within a time scale comparable to that of the LMFBR, while maintaining a capability for even further substantive improvements, such as higher-temperature cladding, carbide fuel, and direct cycle. As an example of commonality with LMFBR fuel, the GCFR fuel rods are collectively vented to a manifold so as to equilibrate the pressure on either side of the cladding, thus removing the effect of high helium coolant pressure.

300-MW(e) GCFR Demonstration Plant Design

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

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

Conservative design bases have been used throughout, and a breeding ratio of 1.33, or 1.5 with 3 rows of radial blanket, is obtained under these conditions. This is largely due to the desirable properties of helium as a fast-reactor coolant. The helium has a small neutronic interaction, thereby leading to a good neutron economy and avoiding any possible reactivity effects; furthermore, the coolant does not become radioactive. Because the design assumptions are conservative, there is considerable performance growth potential inherent in the GCFR concept.

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

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

---

[2] Numbers in brackets designate References at end of article.