HIGH-TEMPERATURE GAS-COOLED REACTOR DEVELOPMENT AND ITS MECHANICAL-STRUCTURAL REQUIREMENTS AND PROBLEMS

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The first chapter of this paper includes a survey on the development of high-temperature reactors and its present state. The second chapter deals with future development potentialities and possible trends. Beside the application of high-temperature reactors for electricity generation the important question of direct utilization of nuclear heat is raised in this connection. The third chapter gives a short survey on the structure-mechanical problems of high-temperature reactors.

Preface

The development of high-temperature gas-cooled reactors is highly advanced. The first generation of experimental power plants or experimental reactors such as the Peach Bottom Power Station of 40 MW (e), the AVR Power Station of 15 MW (e), and the DRAGON Reactor Experiment have been successfully in operation for several years. The generation of prototype power plants is under construction, the Fort St. Vrain Reactor of 330 MW (e) in Colorado/USA is nearly finished; the reactor is to become critical within this year. The construction of the THTR power station of 300 MW (e) was initiated near Uentrop in Northrhine-Westphalia; Fig. 1 shows the present state of its construction. Meanwhile a letter of intent was issued by Philadelphia Electric to Gulf General Atomic for the construction of a twin plant of 2 × 1160 MW (e); this means that the high-temperature gas-cooled reactor now entered the phase of its introduction to the energy market.

1. Survey on the present state of high temperature reactor development

1.1. *Development history*

The development of high-temperature reactors had been started almost at the same time in the United States, Great Britain and the Federal Republic of Germany at the end of the fifties. The British development was based on the experience of gas-cooled reactors, but was performed to a large extent by the international DRAGON Reactor Project. HTR development in USA at first was largely influenced by the British concept, but was then rapidly adapted to industrial purposes. By the end of the fifties, the Federal Republic of Germany had no relevant know-how at all, and even contacts with England and the United States had not been very close in this respect. This might ultimately be the reason why, for the realization of the German HTR concept, the basically different concept of the pebble-bed reactor was adopted. The aim of development was the same in all cases, viz. to increase the temperature of the coolant gas as much as possible by a completely ceramic core and by the choice of helium as the coolant, so that, contrary to the technology of light-water reactors already highly advanced at that time, modern hot-steam turbines could be used.

At first, the basic problem was that no ceramic material being impermeable for fission products could made available for fuel coating. Therefore, one had to expect a highly contaminated coolant circuit. Experiments to render graphite impermeable for fission products by impregnation proved unsuccessful. When designing the first three high-temperature reactors DRAGON, Peach Bottom, and AVR, it had been taken into consideration by relevant technical measures that fuel elements release fission products to a large extent. In

Fig. 1. THTR-power reactor – present state of construction.

the case of DRAGON and Peach Bottom, the contamination was limited to a small gas volume by means of a purch system, to which each fuel element is connected, whereas in the case of the AVR pebble-bed reactor, a double containment of the total primary circuit provides for sufficient safety for the handling of a highly contaminated coolant-gas circuit.

By the successful development of the coated-particle technique any engineering measures for the further development of high-temperature reactors were no longer necessary, and it created the basis of a continued development to economical power plants. Another essential fact in the history of HTR 's was the quick progress of technology in the field of prestressed concrete pressure vessels. Only the application of prestressed concrete pressure vessels permits the construction of high-capacity gas-cooled nuclear power plants.

The following outline focuses on a few items which from the authors' point of view are very important for HTR development and, at the same time, compare with each other the development of HTR 's in Britain, the USA, and the Federal Republic of Germany. As a result, it will be proven that in spite of the diversity of initial positions, the development in the said countries is very similar after all irrespective of certain differences in individual problems. It might be concluded from this fact that the development of high-temperature reactors has reached a certain stage of maturity.

1.2. *Coated particles*

All projects have in common the application of coated particles. Pyrolytically deposited carbon is well suited as a coating material for fuel (see fig. 2), and is nearly impermeable for radioactive noble gases up to high temperatures. It is true that, due to its mechanical properties and its dimensional variations and neutrons irradiations, pyrolytic carbon can only be utilized for coating of very small particles, i.e. 300 to 800 μm; however, they can be produced in fluidized beds at very low cost and with high reliability and extremely low rates of failures. Coated particle producers at present use to guarantee a release rate of 10^{-5} for noble gas produced by fission.

The hold-back ability of pyrolytic carbon for metallic fission products like cesium, barium, and strontium is not so good as for noble gases. These elements have a relatively high diffusion coefficient in pyrolytic carbon; however, they are in part highly adsorbed on the graphite of the matrix in which the coated particles are embedded. Nevertheless, intermediate layers of silicon-carbide were developed for safety reasons, since silicon-carbide in the interesting temperature range of 1100 – 1300°C has a hold-back ability for metallic fission products being increased by 2 or 3 orders of magnitude. Whether, in spite of higher cost, a mixed coating of pyrolytic carbon and silicon-carbide, or a pure coating of pyrolytic carbon will finally be chosen, will be dependent on the exact knowledge of fission product behaviour in the graphite matrix and in the coolant cycle, especially under exident conditions. Surface contaminations of fuel cycle components from fission products

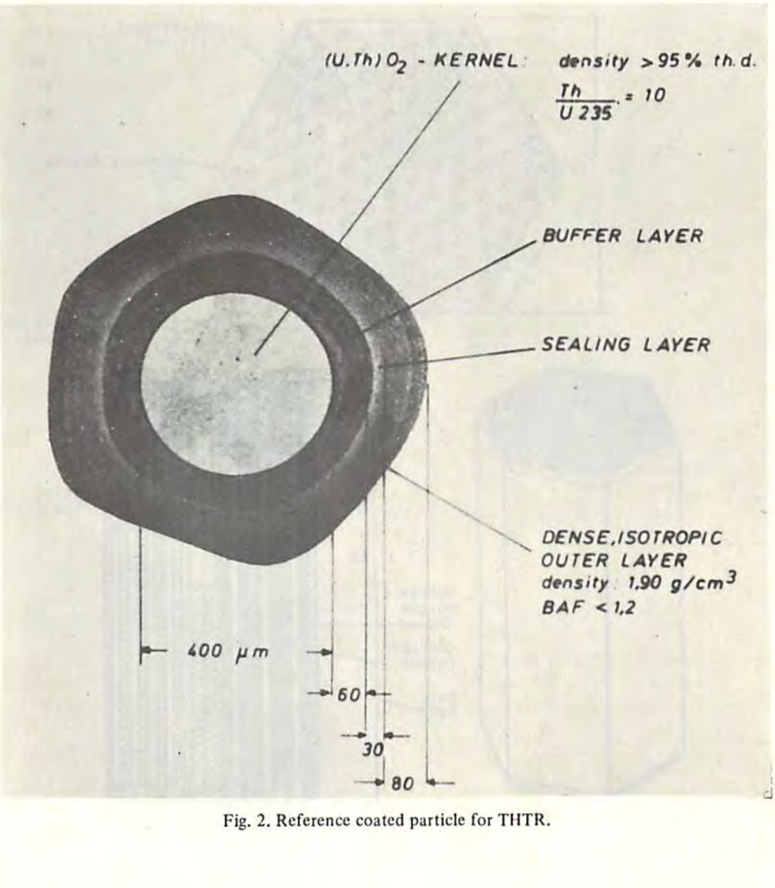

Fig. 2. Reference coated particle for THTR.

were measured in Peach Bottom, DRAGON, and AVR; the results showed an extremely low contamination rate.

The small dimensions which are necessary for the coating of fuel particles bear an essential advantage: the finely subdivided fuel permits to reach very high burn-ups without any damage. Burn-ups of more than 12% fima were reached not only during irradiation tests but also during AVR reactor operation, and that without an increase of activity in the coolant cycle being indicative of any damage to coated particles. High-temperature reactors therefore bear the very interesting possibility to optimize the fuel cycle only due to neutron physics, i.e. reactivity consideration, without limitation by damage to fuel material. This leads to economically very attractive fuel cycles.

The future development of coated particles will involve further improvement and cost reduction of production methods as well as an improved hold-back ability for solid fission products and an increase of maximum permissible operating temperatures. At present, maximum temperatures of 1250–1300°C are being guaranteed by producers. However, unobjectionable results have already been proven by irradiation tests even with temperatures up to 1450°C.

1.3. *Fuel elements*

In all HTR concepts, coated particles are embedded in a graphite matrix. The matrix itself is protected by a fuel-free graphite layer against possible mechanical

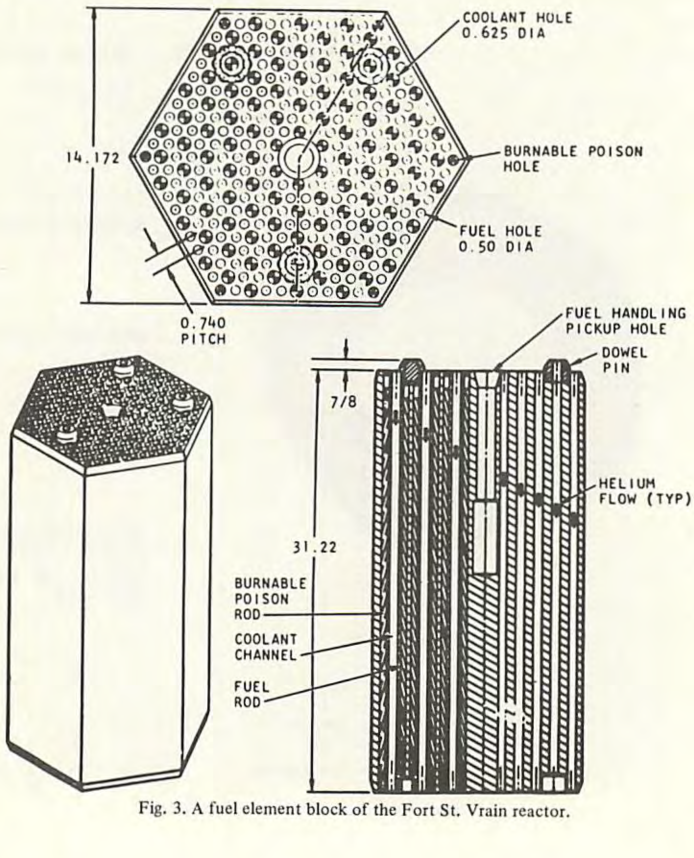

Fig. 3. A fuel element block of the Fort St. Vrain reactor.

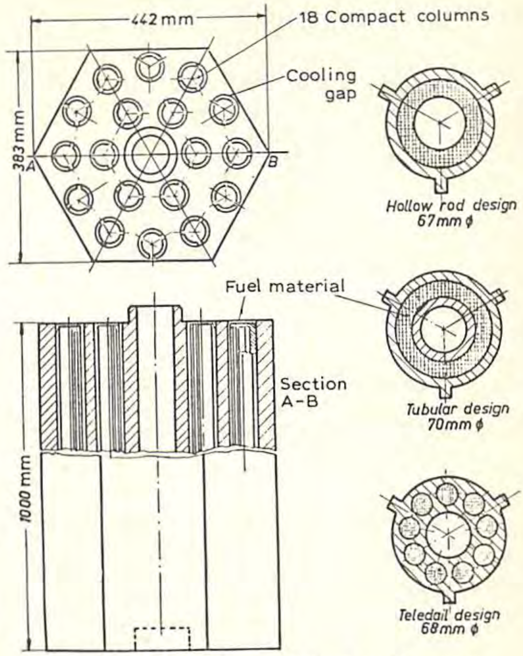

Fig. 4. A fuel element block of the British HTR-concept.

damages and corrosive attacks by helium impurities. Major differences between the individual projects are only to be found in connection with the shape of fuel elements. Prismatic fuel elements in Britain and USA changed from rod element to blocks. Fig. 3 shows the Gulf General Atomic model being applied for the Fort St. Vrain reactor. It consists of a hexagonal graphite block with a key width of about 40 cm and a length of about 80 cm. Fuel material is placed in numerous axial bores in the graphite block. Coolant gas is flowing in separated parallel bores. The coated particles in the fuel element channels of the blocks are connected with each other by a graphite binder. In the beginning of binder material development, a shrinking of the binder under neutron irradiation led to a destruction of coated particles. These difficulties have been overcome meanwhile, and fuel element prodcution for the first core of Fort St. Vrain power station is nearly completed.

Fig. 4 shows the British model with a graphite block being of the same hexagonal form and with almost equal dimensions. Fuel particles are concentrated to an essentially lower amount of axial borings in the graphite block. However, in contrast to the Gulf model, the coolant gas streams directly through the fuel element channels. The coated particles are pressed in a graphite matrix to cylinder rings, which are introduced into two coaxil graphite tubes. The coolant-gas streams around the inner and outer walls of the graphite tubes. The first fuel elements of this design with representative specifications are being tested in the DRAGON reactor since last year.

The Gulf model takes more into consideration the requirements of the uranium/thorium cycle, in which fuel is nearly homogeneously distributed over the core, whereas the English model is primarily optimized towards the requirements of the low-enriched uranium cycle. Fuel material is more heterogeneously arranged.

NUKEM is presently performing a development which in future might become interesting for both models. Starting from the special moulding technique developed for spherical fuel elements, experiments are being carried out with a view to surround the hollow cylinders of the coated particle matrix as well with a fuel-free graphite layer, in accordance with the British model. It seems even possible to form complete hexagonal blocks with coated fuel and corresponding coolant channels by means of a similar technical method. It is the aim of this development to achieve higher power densities as well as lower fabrication cost.

Contrary to British and American projects, the German AVR and THTR reactor projects make use of spherical fuel elements (fig. 5). The 6 cm diameter originally fixed for HTR fuel elements has been maintained. Repeated optimization calculations show a very flat optium which is not very far away from the used value of 6 cm. There is no reason to deviate from this diameter, not even for big power plants of 1000 MW (e) and more. These constant geometric dimensions are of course very advantageous for fuel element development and, particularly, for fuel element testing, which, as everybody knows, has always been a bottleneck in time and money. Several production procedures were developed, and fuel elements were examined by irrediation tests as well as in the AVR reactor. In doing so, isostatic moulding procedure proved to be superior both in technical and economical respect, so that THTR fuel element production is being prepared in accordance with

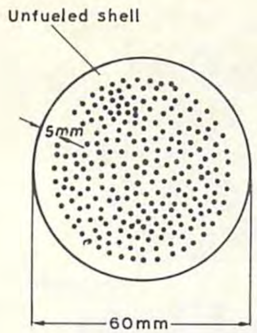

Fig. 5. Spherical fuel element of the THTR reactor. Figures are: heavy metal loading: 0.96 g ^{235}U, 9.62 g Th; matrix density: 1.7 g/cm^3; burn up: mean 12% fima, max. 14% fima; fast dose ($E > 0.1$ MeV): mean 4.8×10^{21} n/cm^2, max. 6.4×10^{21} n/cm^2; max fuel temperatures: a) ball surface 1050°C: – b) coated particle 1150°C.

this procedure. Nearly 50 000 spherical fuel elements have already been produced by NUKEM, out of which about 40 000 are already working in the AVR reactor. Part of these fuel elements meanwhile reached a burn-up of 10% fima. This corresponds nearly to THTR specification. Irradiation experiments in test reactors were successfully performed with a full burn-up specified for THTR and, specifically, on full neutron dose.

The development of spherical fuel elements is practically finished. Additional work is being carried out with the objective to gain further knowledge by irradiation tests and, thus, more certainty about durability and limits of sphere load capacity, to achieve and improvement and cost reduction of production methods, and to provide a certain increase of fuel content for the individual spheres which might become necessary in connection with specific fuel cycles.

The operating experience of many years with high-temperature reactors like Peach Bottom, DRAGON, and AVR proved an excellent hold-back ability for fission products of HTR fuel elements. Peach Bottom and DRAGON, as mentioned above, have a sweep-gas system for fuel elements. and the total coolant-gas activity was therefore limited by a few Curie. As shown in fig. 6, coolant-gas activity in the AVR reactor decreased during operating time, in spite of the fuel element burn-up having much increased in the meantime. The gradual decrease of contamination is substantially due to the fact that subsequent batches of fuel element production have less fissile impurities at the surface of coated particles and graphite matrix. It is obvious that the coolant-gas contamination is caused almost exclusively by these external impurities of the fuel elements.

Also the surface contamination of components of the primary circiut proved to be so insignificant for all of the three reactors that dismantling and repair work could be carried out without special safety measures. On the whole, pyrolytic carbon in combination with the surrounding graphite layers of the matrix proved to

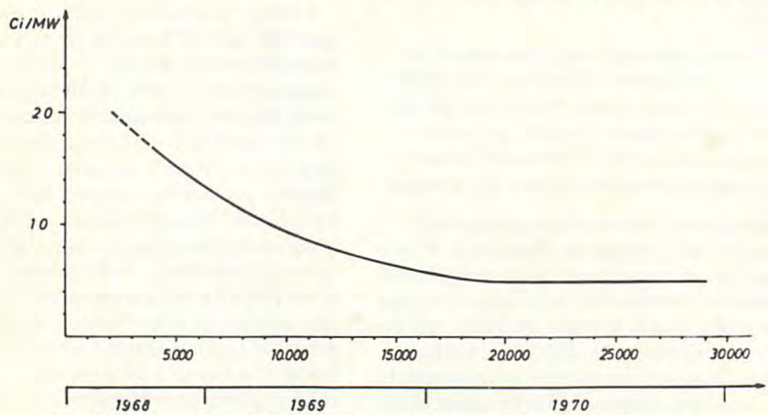

Fig. 6. Primary coolant gas activity during reactor operation of the AVR nuclear power plant.

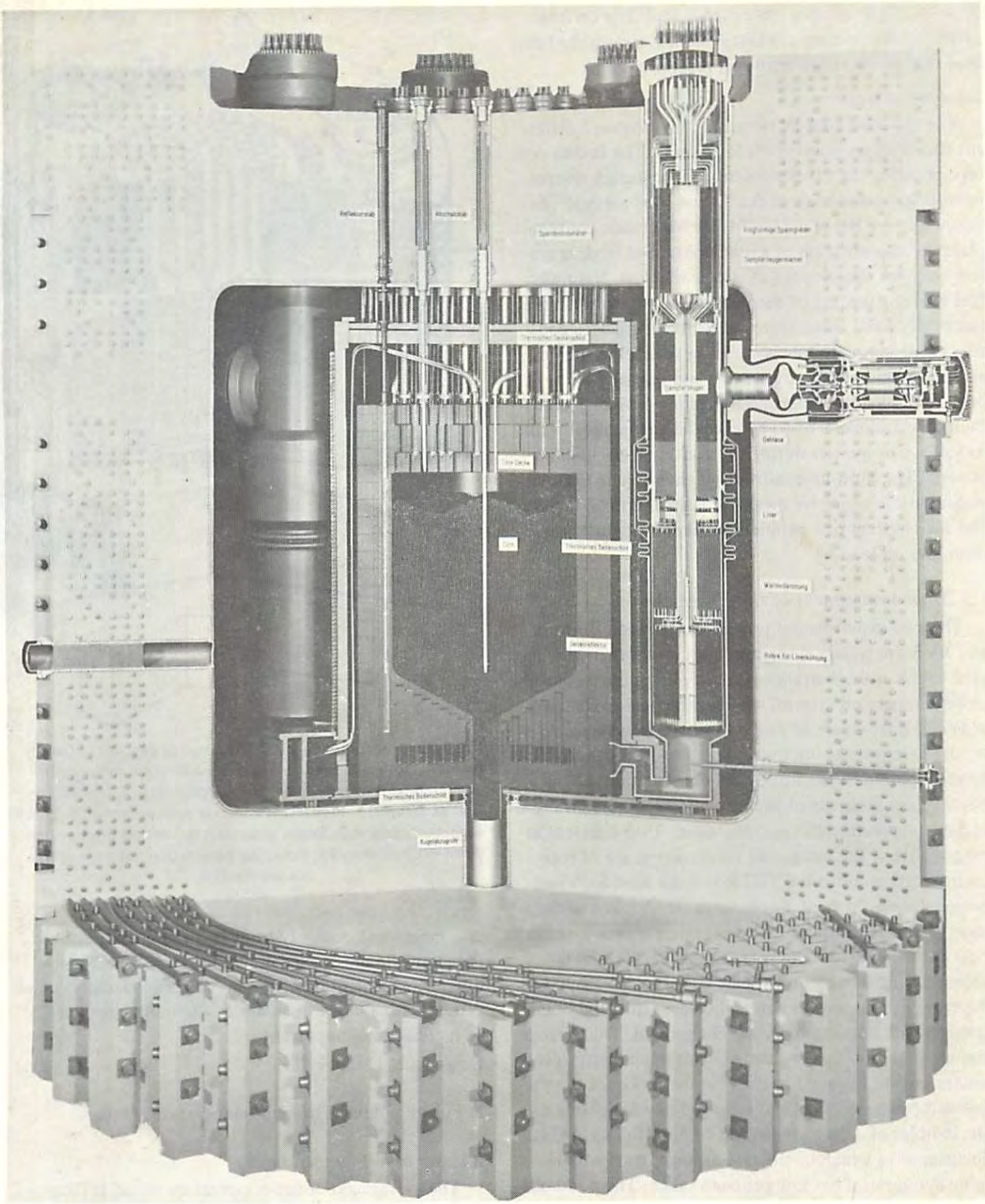

Fig. 7. Cross section of a model of the THTR reactor.

be an excellent coating material for fuel. The contamination of the primary coolant system is essentially lower than that of other conventional reactor systems.

1.4. *Feed system*

The different fuel element concepts required different fuel feed systems for HTR projects. The British concept provides for a fuel element change during reactor operating, on the basis of the "stand-pipe method" developed for AGR, while Gulf General Atomic contemplate the introduction of a load-controlled feeding system only for the purpose of future reactors. Naturally, fuel element feeding of the pebble-bed reactor differs essentially from those systems. Its feeding device consists of an adequate tube system, in which the spheres are conveyed partially by gravity and partially by a pneumatic system. Appropriate testing devices enable each spherical element to be checked automatically for its various properties during the continual circulation process. The burn-up condition of each sphere is also measured each time by means of a small critical facility. The feed-system of a pebble-bed reactor operates continuously on loading.

1.5. *Total arrangement of the power plant*

The two experimental power plants Peach Bottom and AVR are widely divergent from each other with regard to this overall arrangement. AVR applies a completely integrated system, whereas the heat exchangers of Peach Bottom are arranged in separated vessels. After the prestressed conctrete vessel was introduced, all projects preferred the integrated construction, i.e. all essential components of the primary circuit are arranged within the prestressed concrete vessel. Two different arrangements of the integrated construction are of topical interest: on both the THTR and the Fort St. Vrain power station the heat exchangers are installed in the inner chamber of the prestressed concrete vessel, i.e. in Fort St. Vrain below the core, and in THTR – as is shown by fig. 7 – by an annular arrangement around the core. For big power plants, however, the pod-boiler arrangement is increasingly gaining ground. This arrangement is based on the English AGR development; steam generators are installed in the cylinder jacket of the prestressed concrete vessel and arranged like telephone dials, in holes of a diameter of 2.5 to 3 m (fig. 8). A final decision as to which of the two arrangements will be more economical has not yet been taken. The pod-boiler arrangement certainly leads to a more complicated design of the prestressed concrete vessel, whereas the other arrangement involves a more complicated gas-flow system with a larger diameter of the inner chamber and an increased helium inventory.

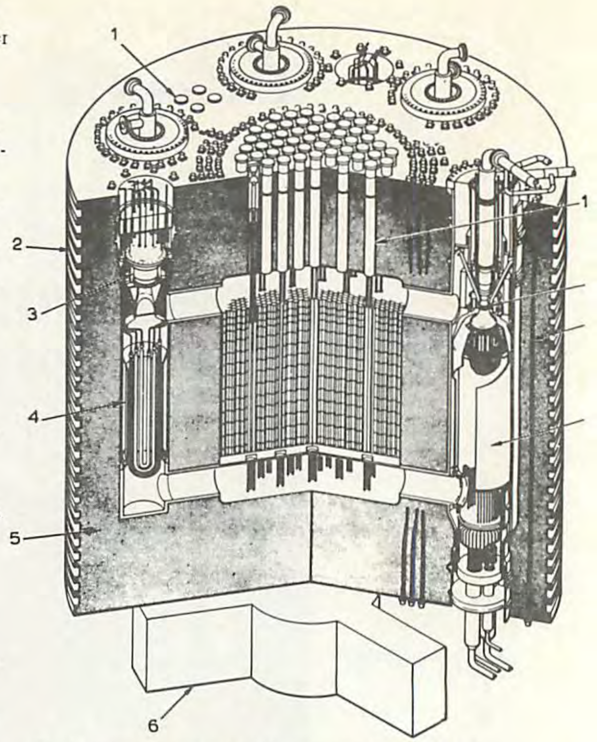

Fig. 8. Cross section of a 1100-MWe reactor concept of Gulf General Atomic. Legend: 1. Helium purification wells; 2. Circumferential prestress channles; 3. circulator; 4. Core auxiliary heat exchanger; 5. Prestressed concrete pressure vessel; 6. PCRV support structure; 7. Steam generator; 8. Vertical prestress tendons; 9. Circulator; 10. Refueling penetration housing control vod mechanism.

2. Future potential of high-temperature reactors

2.1. *Gas outlet-temperatures*

The average gas outlet-temperature for all HTR projects is close to 750°C, except that of AVR; the AVR reactor operates at outlet-temperatures of more than 800°C. These gas outlet-temperatures were determined from optimizations for the different projects, which in all cases are two-circuit systems for electricity generation. These optimizations have to consider maximum fuel temperature, power density, blower capacity, heat-transfer surfaces, and material questions. These are the most important factors. However, gas outlet-temperature determined on all projects does not means represent an upper limit for high-temperature reactors. The gas-outlet temperature of an HTR with block elements can essentially be increased by an adequate design of fuel elements, better adjustment of coolant channels, more frequent element reshuffling and, if necessary, by lowering the power density. If one additionally takes into account the possible increase of the maximum permissible temperature for coated particles, the gas outlet-temperature might then certainly be increased to 1000°C. Investigations carried out during the past two years on the pebble-bed reactor have proved that merely by a variation of the feed rhythm the coolant-gas outlet-temperature can be increased by about 200°C, while the maximum temperatures of fuel elements will remain on the same level. Since this procedure is of high importance for the further development of the pebble-bed reactor, it should be briefly outlined in this paper.

In contrast to normal THTR loading, fuel elements are not circulated several times during their lifetime through the core, but are removed gradually, since their full burn-up is already reached after single pass through the core. The processing time will then amount to about 3 years in accordance with the burn-up time of the fuel elements. This feeding method is called OTTO – Once Through, Then Out. The three-dimensional reproduction of fig. 9 shows the manner in which the concentration of fissile material will be subject to variation over the core. In the present case a fuel cycle was computed with low-enriched uranium. Fresh fuel elements are added at the top of the core and reach their burn-up during their passage through the core. The concentration of fissile material will be reduced accordingly. As a result of the considerable reduction of fissile material during the processing time, a neutron-flux profile is formed, which is shown in fig. 10; its maximum is found at the boundary of the upper reflector. Adequate calculations have shown that fuel elements of this feed system leave the core very homogeneously burnt-up. In fig. 11 the power density, coolant-gas temperature, and maximum fuel temperature over the core axis are illustrated. The power density in the upper. cold part of the reactor is very high, wheras, in the lower part, it is low. Due to the low performance of fuel elements in the lower, hot part of the reactor, the difference in temperature between coolant-gas and fuel becomes smaller and smaller with increasing coolant-gas temperature, thus avoiding a fuel temperature maximum, which usually arose within the core; coolant-gas outlet temperature can be chosen by approx. 200°C higher when fuel temperature is being maintained.

It should further be mentioned that the OTTO feed system also involves many other advantages for the pebble-bed reactor. The buckling of the neutron flux causes an increase of absorber-rod efficiency, so that the amount and introduction depth of absorber-rods can essentially be reduced. The lower part of the core represents a subcritical unit. Moreover, the feeding unit can be simplified substantially, since the burn-up measuring unit and other automatic testing equipment as well as pneumatic recycling of fuel elements into the core are omitted.

On the whole, it may be stated that the development potential of all high-temperature reactors is to be found especially in the increase of the gas outlet-temperature.

2.2. *Future application*

Apart from the application of high-temperature reactors for electricity generation in two-circuit systems with steam-turbine, a possibility was investigated to apply high-temperature reactors also in other systems and for other purposes. Just the possibility of reaching very high coolant temperatures plays an important role in these considerations. Provided they will be realized, these applications would place high-temperature reactors into a dominating position. Relevant studies were focussed on two alternatives: (1) the direct application of a helium gas-turbine in a single-circuit system, and (2) the direct utilization of nuclear process heat for chemical and metallurgical procedures (fig. 12).

2.3. *Gas turbine*

The idea of combinating the high-temperature reactor in a direct circuit with a gas-turbine in principle is as old as that the HTR itself. During the past years, however, gas-turbines experienced a rapid growth in conventional application, so that relevant experience along with sufficient knowledge on high-temperature

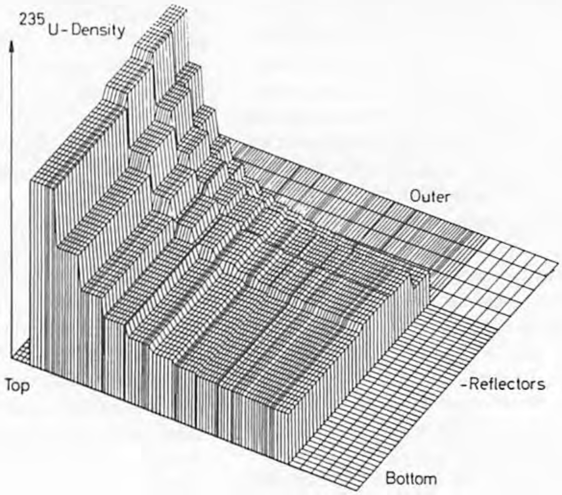

Fig. 9. Three-dimensional drawing of the ^{235}U concentration in a pebble-bed reactor core when refuelling is done following the "OTTO" concept.

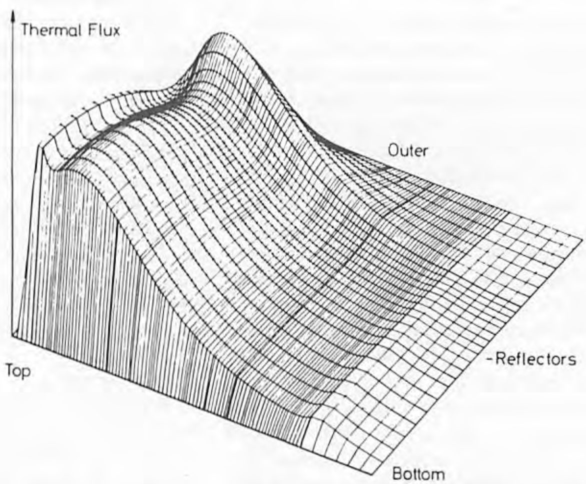

Fig. 10. Three-dimensional drawing of the thermal neutron flux in a pebble-bed reactor core when refuelling is done following the "OTTO" concept.

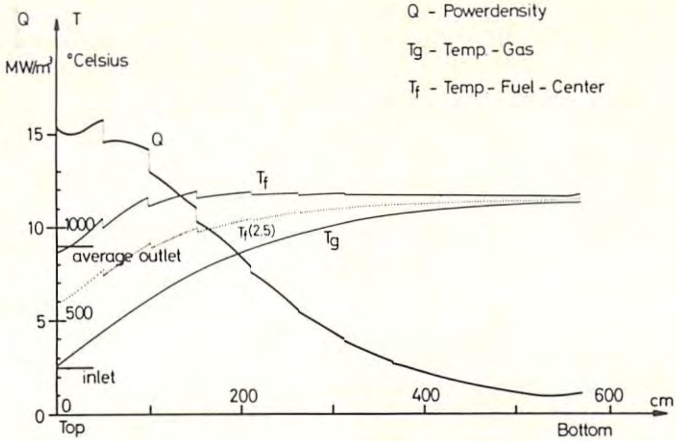

Fig. 11. Dependence of power density, gas temperature and fuel center temperature on core depth calculated for the core axis for a pebble-bed reactor core when refuelling is done following the "OTTO" concept.

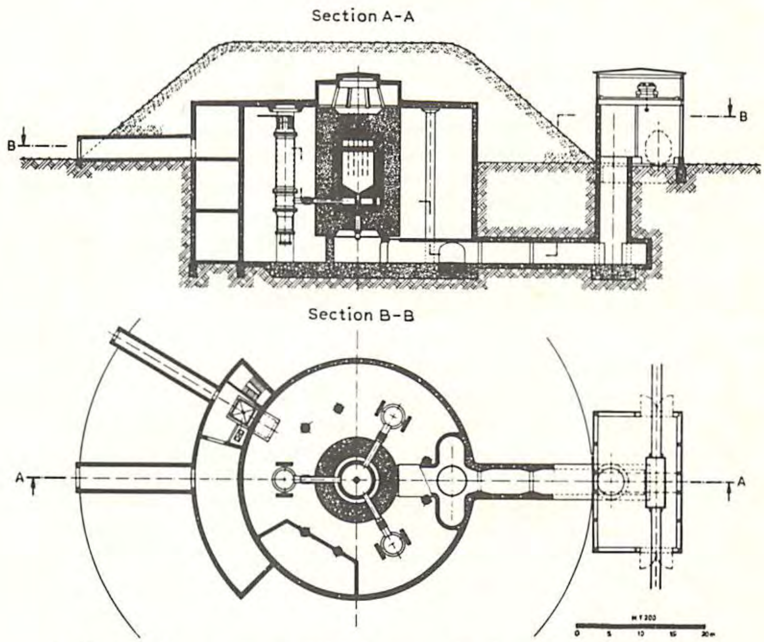

Fig. 12. General arrangement of a 500 MW_{th} nuclear plant for process heat application.

reactors encouraged this development. The advantages expected in comparison with a HTR steam power plant can be summarized by the following items:

A gas-turbine can be built more compact and cheaper. The necessary space for erection and assembly of gas-turbines can be reduced considerably. Presently it is studied whether a gas-turbine could probably be arranged in the prestressed concrete vessel.

Instead of steam generators and condensers, heat exchangers and coolers, which can be built smaller and cheaper, are used in connection with the gas turbine process.

At an advanced stage of development higher temperatures can be applied. Therefore, a further enhancement of the efficiency rate will be possible.

Waste heat is eliminated between 35 and approx. 120°C. For steam-turbine processes, however, the condensation temperature is limited to a maximum of 40°C. The transfer to air-cooling which in future will result from problems of environmental protection can therefore be performed with much lower forfeits than those for steam power plants.

Due to these advantages one expects a reduction of system, fuel, and operating cost.

A difficulty might arise from a contamination of the helium turbine in the primary circuit which would substantially complicate maintenance and repair work. However, in this connection, initial scruples were largely dissipated by the most favourable operating experience made on HTR.

The Jülich Nuclear Research Center is at present engaged in detailed studies for a gas-turbine project.

2.4. *Process heat*

Many fields of industry, especially chemistry, have a high demand for heat at high temperatures. Previously, this demand was completely covered by fossil energy carriers, whereas on the basis of nuclear energy, only high-temperature reactors are able to contribute. Endothermic chemical reactions being suitable for application of reactor heat on high temperature level and requiring considerable energy rates are gasification of coal, fuel oil and methane cracking. cracking of refinery gas, benzine cracking, and olefine production.

Fossil fuels are partially burned today; heat resulting from these processes could be replaced by cheap HTR heat with the necessary reactor outlet-temperatures of the coolant being in the range of approx. 950°C, except in the case of pressure gasification of coal, where the reaction temperature is nearly 1050°C. In addition to economic advantages, the employment of HTRs would result in an extension of fossil fuel reserves and in an essential decrease of envirionmental pollution.

Nuclear reactors for generation of process heat basically can be used in the following fields of application: steel production, heating, production of plastics, production of chemical basic material, general heat consumption of industries, peak current, and steam generation for industrial purposes. Hydrogen, synthesis gas, methanol, ethylene, and methane are products which can be produced by exploitation of reactor heat and which can be utilized in the individual fields of energy industry.

The successful performance of a development program with emphasis on the following points is the prerequisite to the introduction of HTR process heat into the energy market:

Adaptation of steam reforming technology and cracking procedures to the nuclear reactor circuit;

development of procedures for coal gasification under pressure;

continued development of the high-temperature reactor with regard to higher coolant outlet-temperatures, especially for coal gasification under pressure.

The question whether the potential of nuclear reactors will once be exploited in the way described here will mainly depend on the future trend of economic conditions in the energy market. However, already today we are in a position to make the following statements in favour of HTR: The cost of heat generated in the high-temperature reactor is practically invariable as compared with the increase in cost of uranium ore, contrary to the energy generated from fossil fuels, the cost of which is directly dependent on prices of raw material. Processes covering their energy demand by nuclear heat will contribute far less to the pollution of air than the procedure being in use up to now.

If one analyses the potential market share up to year 2000, the potential demand for an installed reactor capacity will be far more than 200 000 MW (th) merely for the Federal Republic of Germany.

The application of nuclear reactor heat for chemical processes will be particularly economical in connection with very large units being used for electricity generation. For several fields of application and also with a view to marketability, it is important to know whether already power plants of smaller size, e.g. of 500 MW (th), could be of interest in economical respect. By a study being actually carried out, it will probably be proved that already a 500 MW (th) HTR counter pressure unit for porcess steam generation would be superior to fossil fuelled units and could complete with major light-water reactor units.

While studying this question we found some interesting aspects, which should now be briefly explained, for they might be important for the future HTR development. For this study, it was necessary to take account of the fact that in nearly all cases nuclear processing plants are located near towns with dense population. It was therefore intended just from the beginning to provide for an underground construction method. The reduced height of construction required for a pebble-bed reactor without any elevated feeding unit is well suited for this type of construction. Furthermore, a non-integrated construction was chosen for this study, contrary to the general trend of HTR power plants. The reason for this choice lies in the fact that requirements set on heat exchangers by the most diversified processes are widely divergent. Therfore, a non-integrated construction is more flexible. A substantially different requirement set on the heat-exchanger will not require then a perhaps completely new design of the total plant.
The prestressed concrete pressure vessel is very small in case of the non-integrated construction method. The lateral ground pressure and the load of the earth fill may properly be considered for the design of the safety vessel. An important argument against the non-integrated construction involves a possible rupture of gas conductor tubes. A corresponding safety-technical analysis in the frame of this study showed that not the loss properly speaking will represent the problem of the technical safety system, but far more the velocity of gas loss. In the hypothetical case of disturbance by a complete rupture of one of the three gas conductor tubes, the conditions arising as a result of the sudden outflow of gas will be of such a complexity that the functional ability of the other two systems cannot be proved with sufficient certainty. These two systems, however, are necessary for transport of after-heat, which needs to be put into operation 20 minutes after such a case of disturbance has occurred. This means that THTR also requires a limited outflow. This problem can be solved relatively safe and cheap at least for simple tube conduits, e.g. by introducing a tube designed for full pressure into the outer pressure-bearing cold tube of the coaxial line. The tube to be introduced need not be connected and is not loaded during normal operation. It is merely in the case of rupture of the outer tube that it will serve as an outflow limitation during the time of pressure reduction.

By the preliminary results of this study obtained so far, it may be concluded that the costs of such a non-integrated model are not higher than for an integrated construction. Therefore it seems worthwhile performing another comparison of this kind in the light of recent knowledge.

Herewith, the survey on the present stage of HTR development and possible trends for the future is concluded. The following section 3 gives a short survey on structure-mechanical requirements and problems of a high-temperature reactor, referring to the relevant specialized discourses given in various sessions of this conference.

3. Structural-mechanical requirements and problems

3.1. *Fuel material*

As mentioned before, all high-temperature reactors utilize coated particles as their fuel material. From the mechanical point of view, the pyrolytic carbon layer around the fuel particles is a small pebble container being exposed to different strains during reactor operation. In addition to the internal pressure due to fission products and, consequently, mainly caused by noble gases, stresses will be arising by different thermal expansions of kernel and coating. Furthermore, there will be stresses due to shrinkage or expansion under neutron irradiation. This problem is discussed in a detailed manner in session C 1 of this conference.

3.2. *Graphite matrix*

Stresses are caused by different orders of dimensions of the graphite matrix and penetrated coated particles. Their role is the more important the denser the coated particles are packed in the matrix. As mentioned before, Gulf General Atomic was successful in finding a technically unobjectionable solution for particles connected by only one binder.

3.3. *Fuel element and graphite structure*

A high-temperature reactor utilizes graphite not only as moderator, but also as structural material for fuel elements. A separation of the problems described is therefore neither possible nor necessary. Moreover, reflector problems could be involved in most cases. Problems arising with the construction of a graphite core and reflector result from reactor-specific loads like temperature change, high fast neutron dose, and neutron flux gradients. A situation becomes complicated by special material properties of graphite. Different sorts of graphite show a different and partially very high anisotropy. It was only in the past years that graphite could be developed with a low anisotropy. Furthermore, the change of dimensions under neutron irradiation largely depends on the sort of graphite and partially changes its prefix increasing neutron dose and irradiation temperature. The creep factor, which plays an important part for the reduction of internal stresses, is dependent on temperature, neutron flux, and the type of graphite. It was and sometimes still is one of the main difficulties to find enough reliable data for an adequate theoretical treatment of these questions. Most of these data can be determined only by very expensive and wearisome reactor experiments which must be repeated for each new sort of graphite. Meanwhile, we have sufficient experience permitting a safe design of the graphite structure for a high-temperature reactor. Further studies, however, will still enable additional experience and, thus, a further improvement of design.

The pebble-bed reactor represents a special case with regard to mechanical problems of the graphite reactor core. A large number of questions resulting from a solid graphite construction can be ignored, because the core consists of statistically packed spheres. Therefore, the mechanical behaviour of pebble-beds has to be studied most exactly. Also in this connection, the investigations carried out during recent years have largely clerified the problem.

All of these intricate problems of graphite core mechanics are being dealt with in a large number of discourses in sessions D2, D3, and D4.

3.4. *Steel installations in the reactor and tube conduits*

Steel installations close to the reactor core are avoided in all HTR concepts. Apart from some specific cases, steel installations are not subjected to high neutron doses. No general or HTR-specific problems will be arising. As far as tube conduits are concerned, there are special questions of design arising due to high coolant temperatures, in particular for thermal insulation; conventional insulation material cannot be applied for reasons of gas purity and endurance. Instead of this, metal foils or phase insulations will be applied at present. For these insulations the emphasis is laid on endurance strenght from a mechanical point of view. Relevant loads are resulting from thermal cycles, possible quick pressure reduction of the coolant-gas in the case of disturbance and gas vibrations occurring under unfavourable conditions.

3.5. *Prestressed concrete pressure vessel*

The concept of prestressed concrete reactor pressure vessels is essentially basic for the development of gas-cooled power reactors. Problems of design analysis of prestressed concrete are being treated in considerable detail in division H of this conference, and therefore, the reader is referred to the relevant reports in the individual sessions.