EUR 3000.e

ASSOCIATION
European Atomic Energy Community - EURATOM/
Brown Boveri / Krupp Reaktorbau GmbH, Düsseldorf/
Kernforschungsanlage Jülich des Landes Nordrhein-Westfalen - e.V.

THTR
STATUS REPORT 1965

1966

THTR 4

"Thorium-Hochtemperaturreaktor" (THTR) Association No. 003-63-1 RGAD

LEGAL NOTICE

This report is on sale at the addresse

at the price of FF 16,50 FB 165.--

When
which

EUR 3000.e

ASSOCIATION
European Atomic Energy Community - EURATOM/
Brown Boveri / Krupp Reaktorbau GmbH, Düsseldorf/
Kernforschungsanlage Jülich des Landes Nordrhein-Westfalen - e.V.

THTR
STATUS REPORT 1965

1966

THTR 4

"Thorium-Hochtemperaturreaktor" (THTR) Association No. 003-63-1 RGAD

THTR STATUS REPORT

The Kernforschungsanlage Jülich des Landes Nordrhein-Westfalen (KFA), Euratom and Brown Boveri/Krupp Reaktorbau (BBK) have entered into an Association agreement in May 1964 with the objective of developing the pebble-bed high-temperature gas-cooled reactor concept.

The THTR Association has organized on December 16, 1965, its first public meeting called "Status Report", where the advance of the laboratory work performed and of the engineering study of a 300 MWe prototype was reviewed.

C O N T E N T S

Introduction by the President of the Steering Committee

Dr. P. Caprioglio

1) The Kernforschungsanlage Jülich des Landes Nordrhein-Westfalen (KFA), EURATOM and BROWN BOVERI/KRUPP Reaktorbau (BBK) have entered into an Association agreement in May 1964 with the objective of developing the pebble-bed high temperature gas-cooled reactor concept.

A Steering Committee fixes the yearly program and the budget of the Association contract and directs its execution.

It is composed of six members :

two designated by KFA	including one representative of the Federal Ministry for Scientific Research (Bundesministerium für Wissenschaftliche Forschung) who supports part of the KFA financial burden
one designated by BBK	
three designated by EURATOM	

The project leader, Dr. R. Schulten, is the Chief Executive and is responsible with a small staff (Projektleitung) of the coordination and execution of the work by the signatories and other contractors.

As you will see during today' presentation, the programme of the THTR Association bears a very close similarity in many fields with the Dragon Project, as well as with the HTGR programme in the USA. This is why it was felt useful to have cooperation agreements with both of them. These agreements have allowed between other things, the possibility of making use of the Dragon Reactor, which is successfully being operated at 10 MW. Furthermore, this allows a very fruitful cooperation with the USAEC in the field of fuel elements.

2) All reactors under development will have chances of success only as far as they can beat both present type of reactors and the conventional power station on economic grounds.

The conventional power station to beat - as operational in the 70's - will have a capital cost of around 120 $ kW - i.e. 480 DM/kW - and a fuel cost of around 3.5 mills/kWh - i.e. 1.2 - 1.4 Pf/kWh -. If one could do better for both items, one would make nuclear energy competitive not only for base load operation but also for annual utilizations as low as 3000 hrs.

All reactor types under exploitation today for electricity production are capable of better performance as far as the fuel cycle cost is concerned. In fact, they all show a fuel cycle cost of less than 2 mills/kWh. In this respect they are therefore already very good. The reason why nuclear energy is not yet competitive, or, if it is so, is competitive with so many restrictions -

base load operation, very large sizes of stations - is because the capital cost of present reactor types is too high.

These figures show that, if the aim of improving the fuel cycle is always a worthy one, it is not the essential one, even when increased uranium ore costs are considered. The real and difficult problem is to improve on the capital cost; here is where large savings have to be made and can be made.

The capital cost of a reactor plant is determined more than by any other single item by the moderator and the coolant, that are chosen. The coolants which have been used for the present generation of near-competitive power stations are light water and CO_2 gas. Let's first see which are the possibilities of further development of present established reactor systems :

Light Water Reactor

Although the reactor as such is a rather cheap piece of equipment, the implications of an essentially poor thermodynamic steam cycle on the attached conventional equipment (turbines, steam ducting, etc.) heavily outbalances the system towards relatively high capital costs. The need for better steam cycles comes not so much from the desirability of higher efficiencies, but rather from the need of cheaper components. Nuclear superheating is therefore looked at as the natural follow-up. It is enough here to mention that a superheated steam is, from the heat transfer point of view, so much more similar to a gas than to water, to understand that the attractive simplicity and compactness of a PWR or BWR will be impaired by the adoption of superheating. After all this is a consequence of changing coolant and adopting a corrosive and poor heat transfer medium.

Gas Graphite Reactor

The latest version of it, the AGR looks still rather expensive as far as capital cost is concerned. The main achievement is indeed to have acquired a "conventional" steam cycle. The next step is obviously to improve the reactor and heat exchanger portion and this can best be done by adoption of a better heat transfer coolant, allowing higher power densities.

This leads us quite naturally to the High Temperature Gas Cooled Reactor :

The HTGR is one of the most proven amongst the advanced reactor concepts. Not only three experiments are being built in the world (Dragon, AVR and Peach Bottom), but also one of them, Dragon, has now been operating with full satisfaction for some while at rather high power level. The Helium cooling technology can be considered as acquired, since most problems connected with it (leak tightness and coolant purities, for instance) have found a satisfactory solution. The problems connected with the new fuel concept involving coated particles in a graphite matrix have also been overcome and another clear year of satisfactory performance of Dragon will increase our confidence. The advantages connected with the He cooling technology, leading to a very compact and cheap reactor design, should enable the HTGR to compete with light water reactors on capital cost.

The all-ceramic fission product retaining fuel concept, together with the excellent neutron economy in conjunction with the use of the Thorium cycle should make the fuel cycle of this particular reactor one of the cheapest, not only today, but also tomorrow when increased uranium costs will have to be considered. It should be pointed out that it is only for U-ore costs as high as 40 \$/lb U_3O_8 that the fuel cycle cost of an HTGR working as a converter would begin to jeopardize the economics of the system. This gives us reasonable assurance that, should the HTGR prove to be competitive today, it would remain so for a very long time to come. It is our conviction that if fast breeders were present at long term, their use could best be restricted to base load duty. Good converters, such as the HTGR, would profitably take care of the lower load factors without solution of continuity.

The HTGR as it stands today is already a rather advanced concept, and yet has a tremendous potential for further development towards reduction of both capital and fuel cycle cost, should this prove necessary. Further capital cost reduction can be achieved by adoption of a direct gas turbine cycle. This would eliminate He to steam heat exchangers and would give place to a more compact rotating machinery. This last statement should not surprise too much when one considers that a noble gas such as Helium in the range of temperatures achievable in an HTGR (up to 900°C) and in the range of pressures that we already envisage today(up to 40 to 50 atm) can carry a tremendous amount of energy per unit volume and can, therefore, give place to high power compact machinery.

As far as the fuel cycle is concerned, development in fuel reprocessing technology and adoption of Beryllium oxide in the reactor core should achieve conversion factors sensibly above unity, which means breeding. It is, however, important to note that although the possibility of achieving very high conversion factors exists, this does not mean that we have to take immediate advantage of it.

One of the things that, to our minds, makes the HTGR so attractive, is the extreme flexibility of its fuel cycle. Conversion factors can always be optimized according to the actual cost of fissile material, so that the state of development of reprocessing and refabrication technology can allow any fuel cycle to be chosen from once through cycle to breeding.

In a year or so, time will be ripe for a decision upon the construction of a prototype. After all, that is just what is going to be done on the other side of the Atlantic, where an electricity producer has already decided to buy a 330 MW_{el} prototype. A turnkey contract has recently been signed between Public Services of Colorado and General Atomics for a capital cost of 139 $/$kW_{el}$ (556 DM/kW) installed, a guaranteed fuel cycle cost of 1.6 mills/kWh (0.64 Pfg./kWh) over the first 8 years of operation. USAEC will contribute with about 40 million dollars including research and development as well as a contribution for "first of kind" components. Keeping into account that this is the first big step towards gas cooling technology in the USA, one can conclude that the Colorado plant is going to be one of the cheapest prototypes ever built.

Introduction by the Project Leader

Prof. Dr. R. Schulten

This is the first time that we have reported on the THTR project to a large audience. After the contract covering this association had been signed in May last year, a steadily increasing number of staff members worked on this project in an attempt to have the development work and the construction documents for a THTR prototype plant ready by the end of 1967. About 220 persons are at present working on this project. Within the next two years, a staff of about 300 members will be available.

The high-temperature reactor is a continuation and a further development of the gas-cooled graphite reactor type. It is a wellknown fact that this reactor type- above all the Magnox reactors built in Britain - today accounts for the bulk of nuclear power production and that at present the greatest operating experience has been obtained with this type of reactor. It is therefore an encouraging confirmation of our development work to see that the AGR, as a continuation of the Magnox graphite reactor, shows a clear trend towards the high-temperature reactor. Many components of the AGR are also used by us, such as the prestressed concrete vessel with heat insulation, the blowers, the graphite blocks, the heat exchangers and typical measuring and regulating gear. However, new features are the use of helium instead of CO_2 and graphite coated particles instead of UO_2 pellets. The employment of the thorium circuit instead of slightly enriched uranium is an additional innovation.

The following considerations show the real advantages to be gained from the further development of the AGR to a high-temperature reactor:

In comparision with the AGR, the initial temperature of the high-temperature reactor is increased from 670° C to more than 750° C. This temperature increase, in conjunction with the use of helium instead of CO_2, enables the power of the AGR to be increased by more than 100 %. This gives a decisive reduction in the capital cost of the plant. The utilization of coated particles permits higher burn-ups of more than 100,000 MWd/t, as the results of our irradiation experiments indicate. Higher burn-ups mean lower fuel processing costs. Finally, the use of the thorium uranium fuel cycle results in a major reduction in the fissionable material consumption and thus a reduction in fuel costs. Another characteristic of the thorium cycle is the fact that a comparatively low fuel input is required, the advantage of which is confirmed by the reduction in capital fuel costs.

Figs. 1 and 2 show the 300 MW prototype, construction of which is planned under the German atomic programme. The gas flow for the transfer of the heat from the reactor is from top to bottom; the gas is collected underneath the core and passed to the steam generators at the side, where it is cooled and then transferred through the blowers along the reactor shell back to the upper part of the reactor.

In contrast to high-temperature reactors developed elsewhere, loading and reloading of the reactor core can be carried out continuously during operation of the THTR. Our development work indicated that both continuous loading as well as continuous

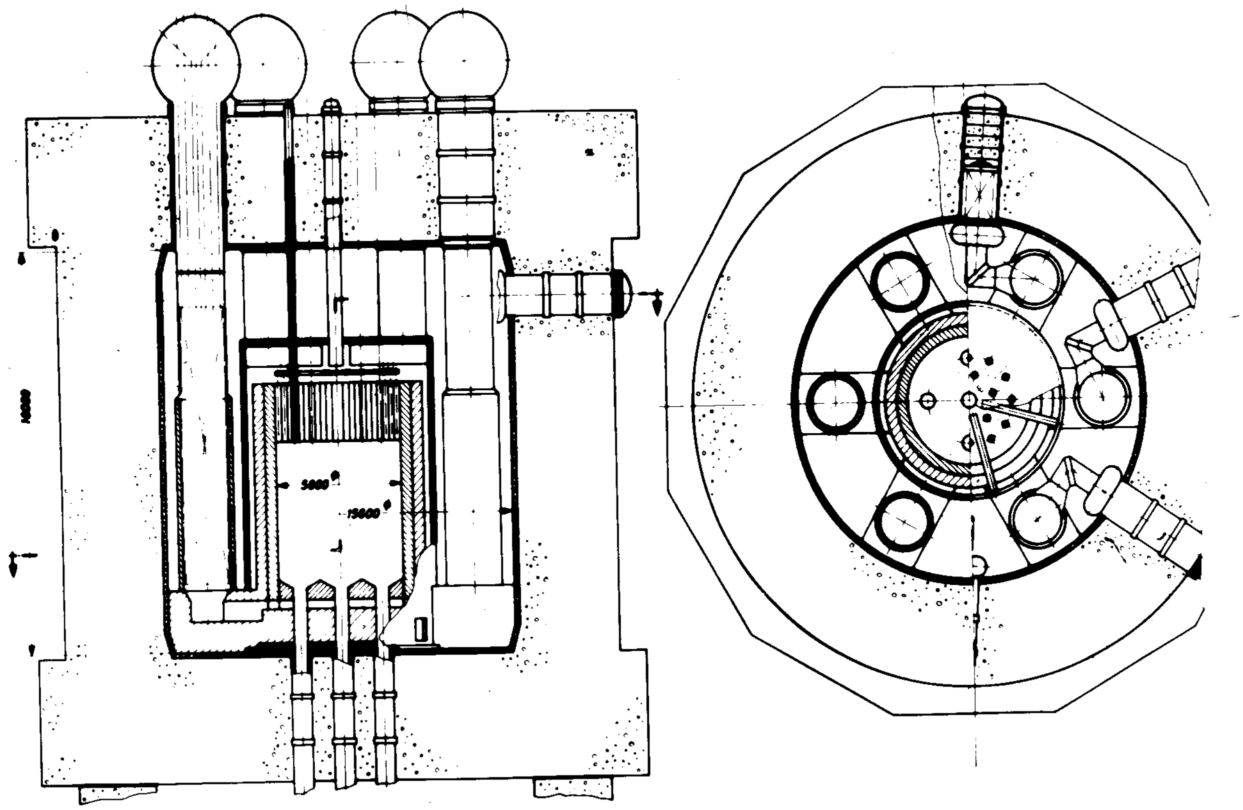


Fig. 1 — Sectional view of the 300 MWe THTR prototype.

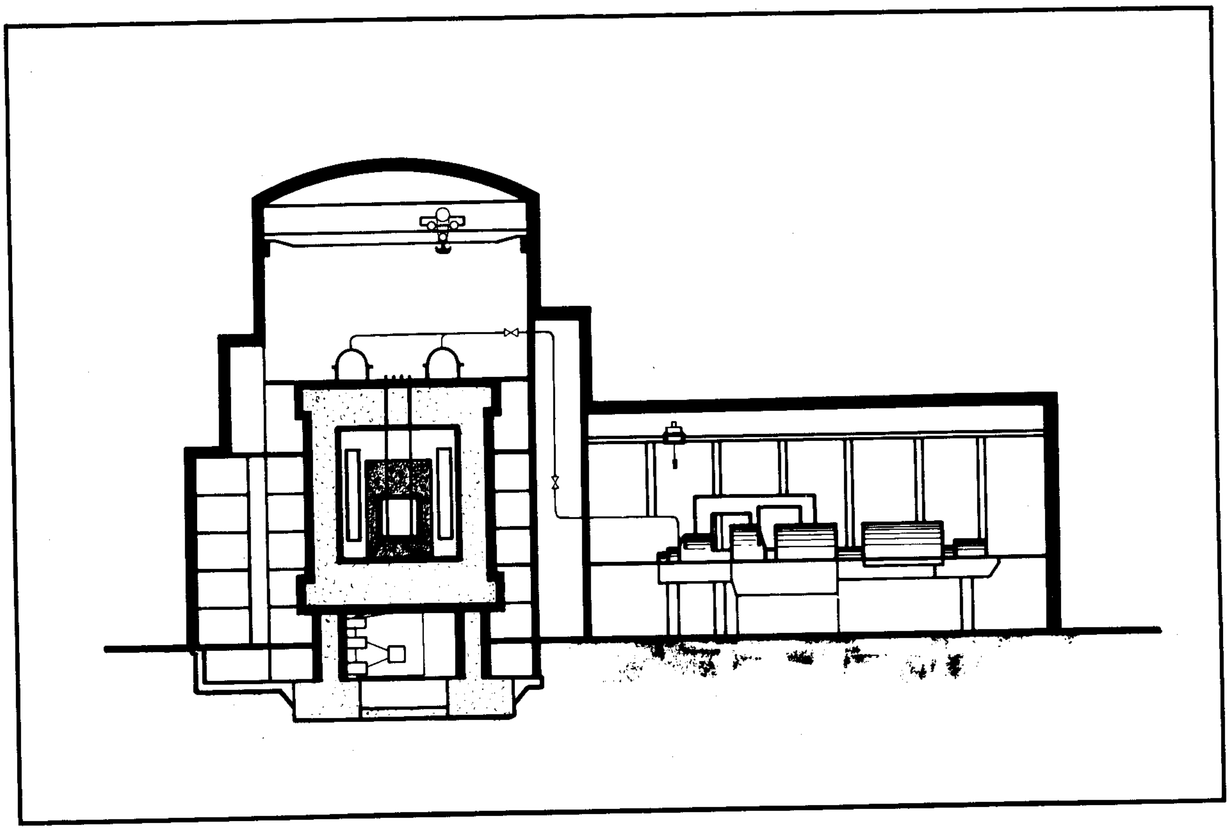


Fig. 2 — 300 MWe - Prototype.

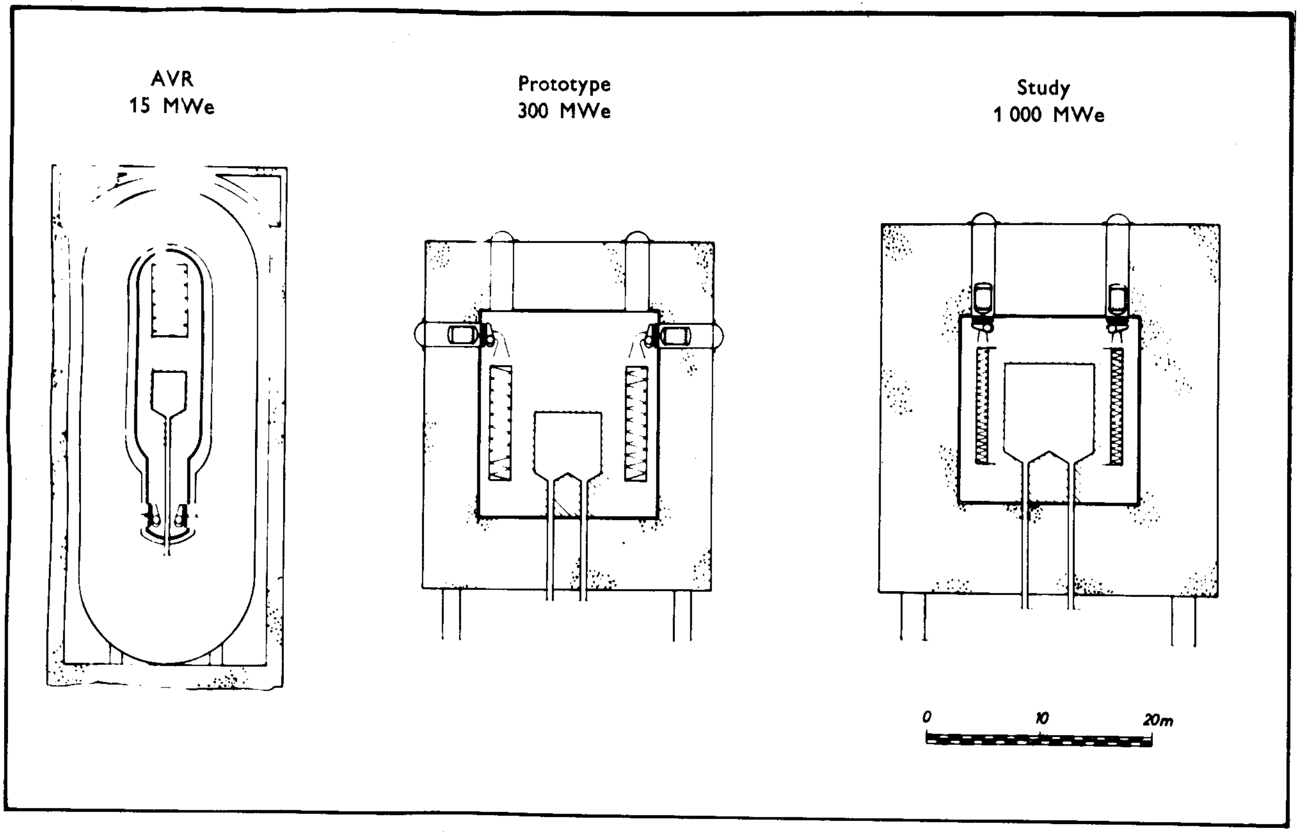


Fig. 3 — Size comparison between the AVR, a 300 MWe and a 1 000 MWe prototypes.

reloading of the reactor are favorable for the operation and for the fuel cycle, since the capital costs can thus be reduced appreciably.

As the fuel loading, the flow of the fuel elements through the reactor and many important parts of the loading plant are similar to the AVR reactor. The removable blowers are mounted laterally. The steam generators can be replaced if necessary. It is assumed that this facility can subsequently be dispensed with. This will permit the size of the concrete vessel to be reduced considerably. The plant is of the so-called integral design, i.e. all the parts of the primary circuit, including the steam generators are located in the same pressure vessel. The nuclear part of the power plant consists essentially of the reactor, the gas purification plant and the helium tanks. Auxiliary equipment e.g., for the removal of the steam generators, are housed in the same building.

Fig. 3 shows the advantages of the prestressed concrete vessel. The volume of the 300 MW plant is not much greater than that of the AVR reactor and the 1000 MW plant is only slightly bigger than the 300 MW plant. The relative volume profit obtained from the transition from 300 to 1000 MW is based on the fact that the steam generators for a 1000 MW plant need not be exchangeable.

A comprehensive experimental programme is an essential prerequisite of the construction of these nuclear power plants. In the past, Brown Boveri/Krupp has already carried out important experimental work, and the work performed under the THTR association was in continuation of this. It is not possible to deal with all these experiments in detail in our reports today, for which reference should be made to our annual reports. Broadly speaking, the experiments covered the following fields:

1) research on graphite: physical, chemical and mechanical properties and irradiation damage;

2) the investigation of heat transfer and helium flow;

3) the study of mass transport and gas purification problems;

4) the behaviour of fission products: diffusion and deposition in the circuit;

5) the verification of reactor components, e.g., valves and fittings, drive mechanisms and penetrations;

6) the development of methods for testing gas-tightness;

7) studies on dry bearings in high-purity helium.

I should like to mention certain experiments which are characteristic of our THTR reactor, namely:

8) the behaviour of the pebbles as they flow through the reactor core;

9) the determination of the burn-up condition of the fuel elements;

10) the development of an economical gas purification system on a chemical basis.

For the development of the fuel elements numerous studies were also carried out under our contract of Association. Spherical fuel elements with fuel in the form of coated particles are used in the THTR. These particles are expected to give a maximum operating temperature of 1250° C and a burn-up of 100,000 MWd/t. The particles are either randomly packed in hollow spaces or embedded in a graphite matrix. At present, the Association has set up a production capacity of up to 8 kg/d through a subcontractor, which could meet up to 25 % of the current requirements of a 300 MW power plant. These studies and the testing of the fuel elements and particles under the irradiation program will be discussed in greater detail this morning.

The development work on a new reactor type are only justified, in view of the present situation with regard to atomic energy, when genuine economic advantages can be expected. Fig. 4 shows the calculations of the capital costs of high-temperature reactors. They indicate that for high-temperature reactors in the region of 600-1000 MWe the capital costs will be from 600 to 500 DM/kw.

The following points are to be made in connection with Fig. 4:

1) With regard to the Colorado power plant, mention should be made of the fact that, in addition to the capital costs given here, another 40 million dollars is being paid by the AEC for development work, fuel loading and one-time purchases.

2) Although a precise cost breakdown cannot be given until after completion of our design work, it is possible to state even now that the high-temperature reactor offers a good development potential.

   a) Owing to the use of prestressed concrete, one part of the reactor is cheap to build;

   b) Because of the excellent thermal properties of helium, the steam generators have a high heat flux;

   c) On account of the steam conditions which can be achieved, the turboset and the other parts of the conventional equipment are cheap, while in larger nuclear power plants these account for the majority of the construction cost.

It is well known that the majority of power plants can only be operated with load factors of less than 60 %, in order to meet the requirements of the electricity market. In this connection, the future possibilities of operating nuclear power plants with low fuel costs but higher construction costs are overestimated since the energy generating cost is in this case much more dependent on the low load factor demanded by the market. For this reason our development work is aimed at reactors having load factors of 50-60 %. Not enough attention has been paid to this fact in the cost calculations effected so far. In comparison with other reactor types, the high-temperature reactor occupies a favorable position in this respect, in particular because the capital costs for construction and the fuel are considerably lower.

Fuel studies were also carried out by the Association and other development groups, leading to the following results (Fig. 5): In the figure the fuel costs are given as a function of the total capacity of the reactors. Up to 10,000 MW no allowances were made for reprocessing, but above this value reprocessing was assumed. A fairly large number of high-temperature reactors with a total capacity of 15,000 MW are shown to the right of the figure.

We have studied one group of problems which are frequently overlooked when assessing fuel costs. For obvious reasons, a limited number of reactors having a low output entail fairly high fabrication and reprocessing costs. Only when the total capacity is higher are these costs likely to fall. The fabrication costs drop fairly fast when the power is stepped up. As technical developments stand at the present, on the other hand, reprocessing costs are only low enough for large capacities if the units are only used for one type of reactor, as is assumed here. This is the particular economic problem involved in the introduction of a new reactor type, and perhaps greater attention should be paid to it in the future. Precise consideration of the matter shows, for instance, that substantial difficulties arise when a new reprocessing unit is required. The considerable cost involved can then only be justified by the operation of several nuclear plants of the same or a similar type at the same time, which presents a problem in the case of new developments.

In the case of high-temperature reactor, this problem is to be solved in two ways:

1) By using a fuel cycle without reprocessing as long as the capacity is not sufficient to reduce the costs;

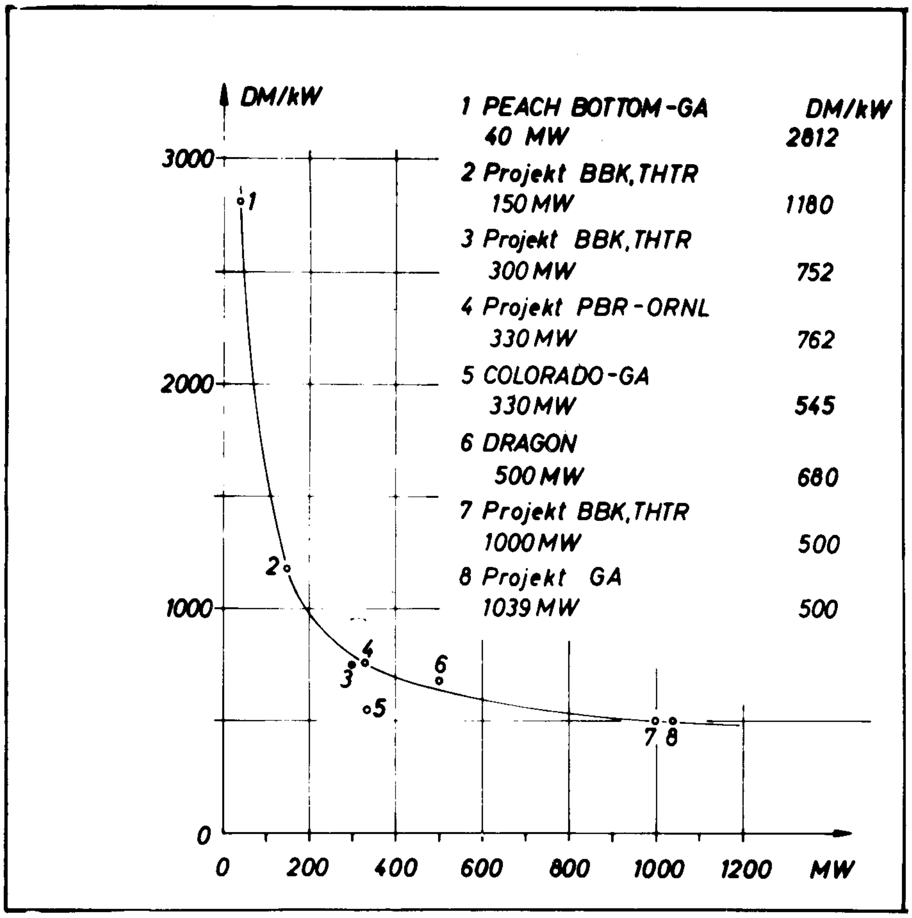


Fig. 4 — Specific capital costs of high temperature reactor stations as a function of the reactor power.

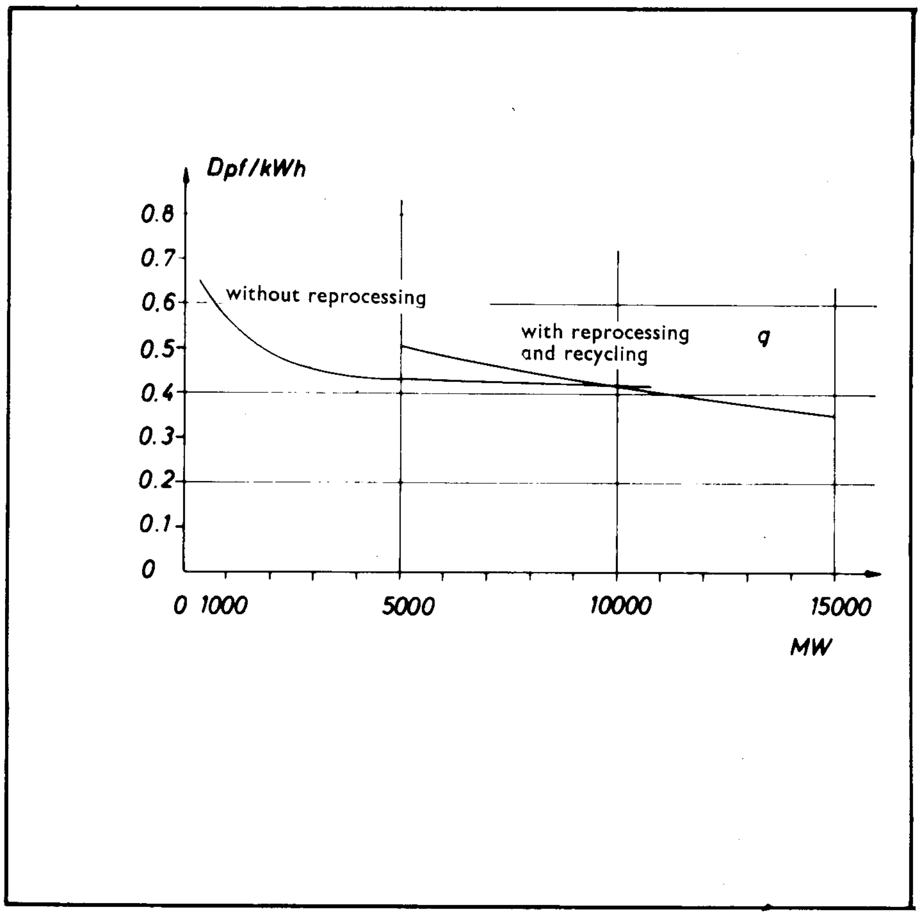


Fig. 5 — Fuel cycle costs as a function of the reactor power.
Load factor 0.73 $\simeq$ 6 400 h/year.

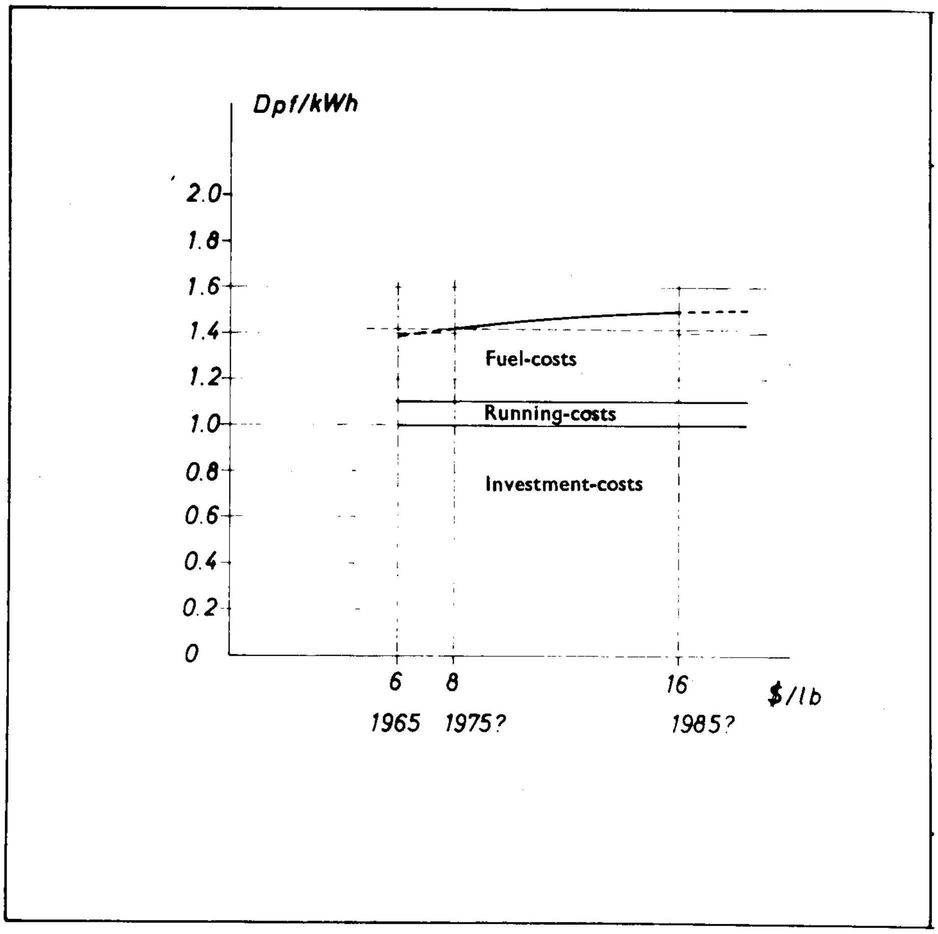


Fig. 6 — Electricity production costs as a function of the uranium ore price.

2) By adapting the fuel cycle to existing reprocessing units, i.e., by developing a special head end in collaboration with Eurochemic or the Thorex plant in Italy, for instance.

It can be seen that the two-type loading (so-called feed and breed concept) which is planned by the Association and which forms the basis of our calculations will give very favourable fuel costs. The one element is here a so-called breeder element containing thorium and fissile material in equilibrium concentration; the other contains only fissile material and is used to control the reactivity and power density.

The next figure shows the total costs for one kWh, the capital costs and operating costs being taken from other publications (Fig. 6). The percentage of the costs which depends on the price of uranium ore will therefore be less than 5% of the total cost per kWh. The operation of the high-temperature reactor thus seems to be largely independent of the ore price. A twofold increase in the ore price which is possibly to be expected by the year 2000 will hence have no major effect on the cost and consequently the operation of such reactors. It may be stated here that the capital costs have a much greater influence on the total costs. Moreover, we feel that owing to the relatively low capital costs, as has already been said, high-temperature reactors can also be operated in greater numbers in the future with loading factors of less than 70 %.

The most important problems and their resulting studies for the following two years will be as follows: the results hitherto of the test programme and the experience acquired during the construction of the AVR can be utilized in the 300 MWe prototype plant. An important factor here is the large-scale simplification which should be possible by virtue of the favourable results obtained in connection with the low fission product release, the low leakage rate and the low helium gas impurity concentration.

The irradiation programme is concentrated on two specific fuel element types and to definite graphites to be used for building up the core. Furthermore, the burn-up measurement and the transposition of the chemical cleaning process for the helium developed by us are regarded as particularly urgent problems. All the studies are based on the assumption that extrapolation of the components from a 300 MWe plant to a 1000 MWe plant is possible.

To sum up, perhaps the present status of the project can be outlined as follows:

1) The research carried out under our Association will during the next two years be mainly devoted to the problem of radiation damage and graphite corrosion at high neutron doses and high temperatures. In this way it is hoped to work out precise specifications for the most suitable types of graphite for the reactor internals.

2) The gas purification and burn-up measurements techniques, which have so far only been tested in laboratory experiments, must be studied on a larger scale.

3) The development work and the finalization of the data have reached the stage where the design and construction of the 300 MWe prototype can begin in 1966 and, parallel to it, the elaboration of a design study on a 600 MWe unit.

4) Apart from the oxide particles, which is a special problem to be tackled next year, work on the fuel and fuel elements has progressed so far that a start can be made on large-scale production. The 1966/67 irradiation program will concentrate on the testing of these products. The production process is largely automatic.

With regard to the future potential of the high-temperature reactor, I should like to quote the following arguments, which, incidentally, in some cases also apply to other reactors working on the thorium fuel cycle:

1) The high-temperature reactor is flexible with regard to the choice of fuel. Both slightly-enriched uranium and plutonium and also highly-enriched uranium with the correspondingly adapted conversion factor can be used. Even on the most pessimistic assumptions with respect to uranium ore prices, the calculated fuel costs are only slightly in excess of those given for other reactors, which can most likely be more than offset by cost reductions obtained by breeding. These are to form the subject of studies next year.

2) No fissile-material-producing reactors of another type are required to introduce this system.

3) With the construction and operation of the 300 MW demonstration plant, the development can be completed in 6-8 years.

4) Now that the fuel has been shown to have good retaining properties with regard to fission products, there are apparently no longer any safety problems.

The high-temperature reactor offers an economic and competitive solution to the power supply situation not only for the next few decades, but also over the long term.

The further development to a breeder is mainly a question of the size of the reactor and of the development of a suitable reprocessing method. Also, the use of BeO, which can be employed in the reactor as a structural material in addition to graphite and gives a better neutron balance, is a matter which does not imply a modification in the reactor concept itself, but a further development of the fuel concept.

In the course of the following talks we shall try to give you as accurate as possible a picture of what has been achieved so far and of the problems which still have to be solved. It is with this in mind that the subjects dealt with were selected.

The Experimental Work Performed for the Development of the THTR Project.

Dr. C.B. von der Decken

Out of the vast research and experimental work carried out, I would like to limit myself to three important subjects :

- the mechanical behaviour of the pebble bed,
- the problem of graphite corrosion and mass transfer
- the gas purification

The mechanical behaviour of the pebble bed is a major problem specific to the THTR reactor. The work on this is carried out on different models. Fig. 1 shows the 1 : 1 model of the AVR reactor core. On the left-hand side the exterior of the model can be seen. The movement of the balls on the periphery of the core can be observed through a perspex window. On the right-hand side there is a view of the interior of the core. You can see the core bottom with the gas entry slits and the central outlet tube for the balls as well as the noses - all made out of graphite. The model contains 90,000 graphite spheres. So far, more than 10 million balls have been circulated, which corresponds to reactor operating time of more than 20 years.

Fig. 2 shows a scaled-down model containing a fluid. This arrangement enables the movement of the balls to be observed inside the pebble bed and the velocity profile of the ball movement for different fuelling systems, to be measured directly. Another scaled-down model using steel balls is shown on

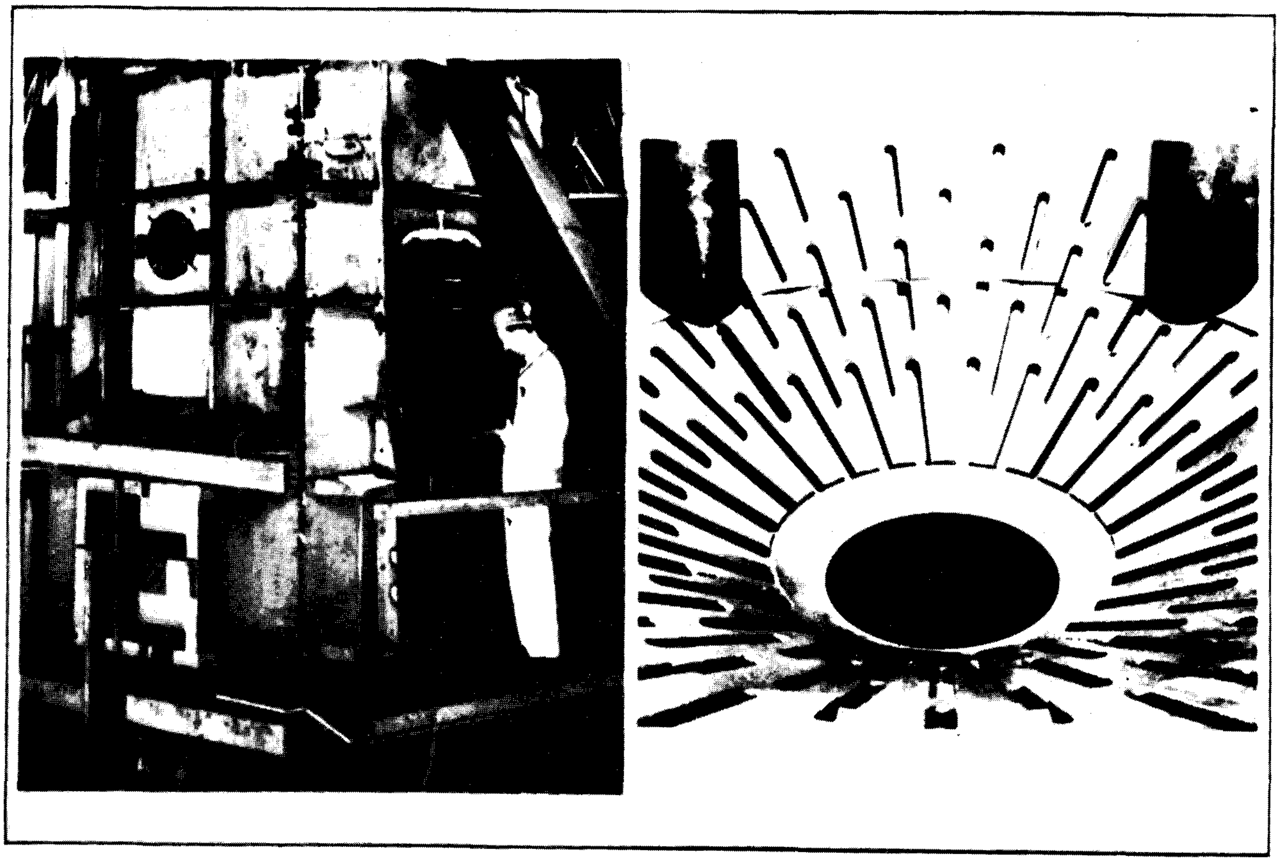


Fig. 1 — Model of the AVR core.

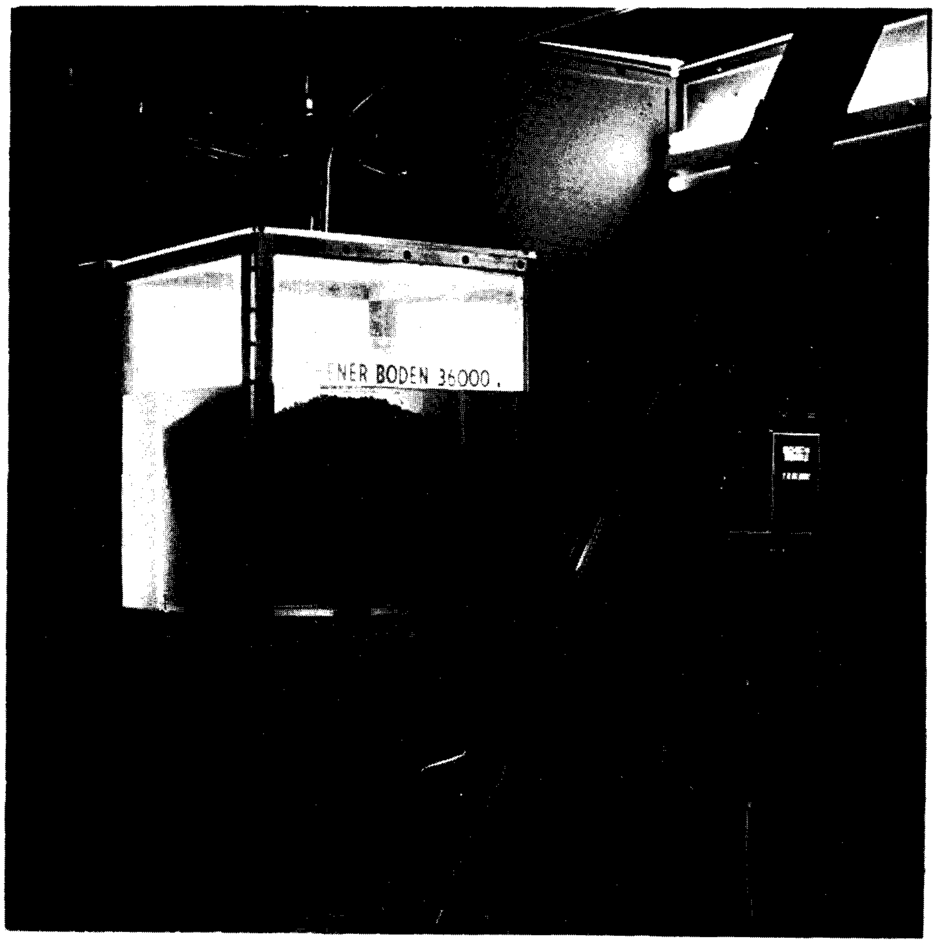


Fig. 2 — Model of the AVR core with glass balls.

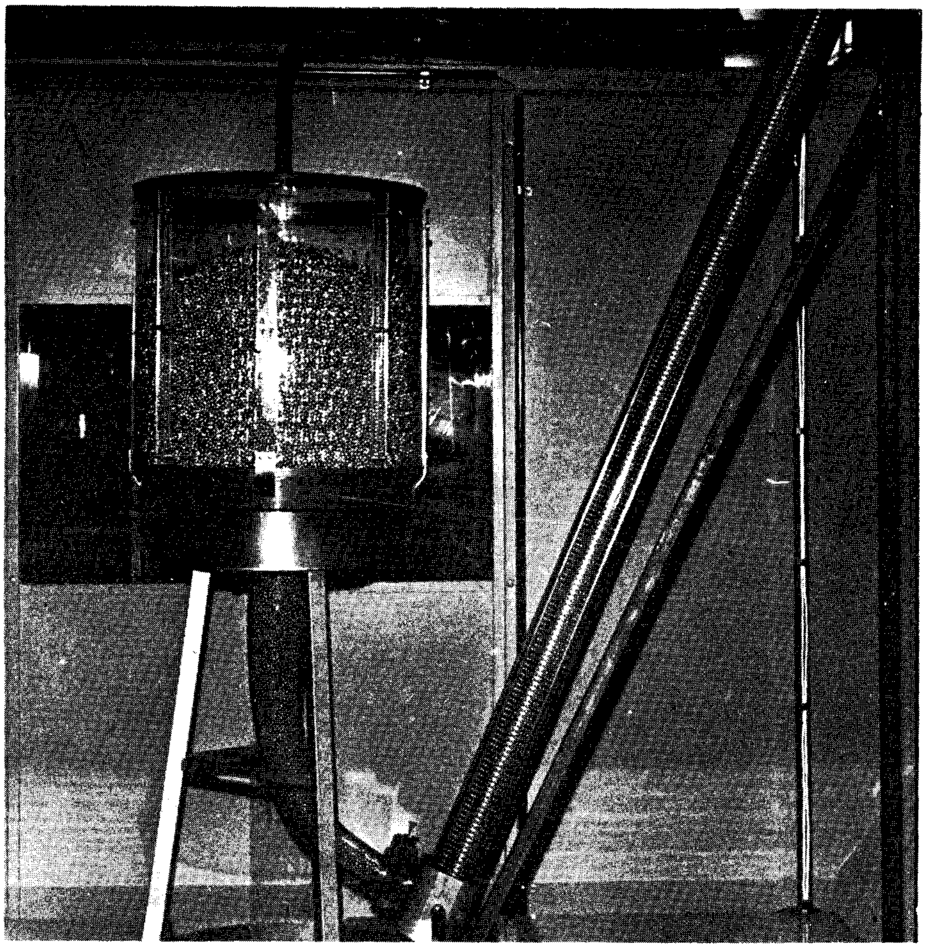


Fig. 3 — Model with steel balls.

Fig. 3. All measurements on this model can be performed easily and more quickly and with far less personnel and lower cost than the 1 : 1 model.

The work on pebble bed mechanics can be divided into two stages. For the first stage it must be assumed that a 1 : 1 model is available and that all results obtained from smaller models can be checked by selecting a small number of comparative measurements between the 1 : 1 model and the smaller models in order to find out whether they are valid for the 1 : 1 model. For the investigations on the AVR reactor, this was the method used.

Broadly speaking, we could do the same for the THTR reactor but one has to realize that the technical equipment required for a 1 : 1 model for the THTR reactor would be extremely complicated, not only on account of the 800,000 graphite balls and the larger dimensions used, but also because the amount of personnel and time required would be unrealistically high.

This is where the second stage begins, which is where we are just now. Our aim is to avoid the construction of an expensive THTR 1 : 1 model by working out semi-empirical theories and similarity laws. We were able to show that this is possible in principle, so that we could already go ahead without a 1 : 1 model. To work out the semi-empirical theories and similarity laws, all existing models, including the AVR 1 : 1 model, have to be measured in detail and the results compared.

A considerable step forward was achieved by the discovery that the form of the curves can be very well approximated by potential lines, and this is suitable because the basic provisions of the potential theory - free from curl and incompressibility - and the validity of the continuity equation are complied with for the pebble bed.

A comparison between the curves obtained in the glass pebble model and those obtained in an electrolytic cell shows considerable agreement.

Over and above this work, theoretical considerations showed that it should be possible to prepare a statistical model of the behaviour of the pebble bed.

I would like to summarize the main results of our studies so far, especially in respect of our discussions and the later lectures, as follows. There is a well-defined velocity profile for the balls across the core cross-section. This profile depends on the special geometry, as a result of which the direction of the movement of the balls is almost vertical throughout the whole core. The velocity profile can be changed and optimized by a suitable geometry, such as the number and position of the outlets, and by a special design of the bottom cone.

I now come to the second part of my talk. One of the most specific problems of all gas-cooled graphite reactors is graphite corrosion and carbon mass transfer in the primary system. The aim of our research work here is to work out the specifications for cooling gas, gas purification and graphite. In particular, we have to take into account the catalytic effects of the oxidation of the graphite and the carbon deposition in the heat exchanger. In all these problems we are being helped by the excellent work which has been carried out by the Dragon Project Group. This applies in particular to the catalytic effects of the carbon deposition, the dynamics of the reactions and the inhibition thereof as well as the long term behaviour of steam generator materials.

Under our programme, the following equipment is used to carry out research work in this connection :

1) An apparatus for measuring the corrosion of graphite balls up to 1000°C with corresponding gas impurities;

2) An apparatus to study the local oxidation in the graphite shell when a temperature gradient is present;

3) An apparatus to determine the catalytic effect of solid fission products on the oxidation.

If the results that we have obtained so far are evaluated, we obtain - provided that no carbon deposition takes place - the following specifications for the clean-up system :

Helium outlet temperature		750°C
Leakage of steam generator		12 g/h
Throughput		7300 Nm^3/h
Necessary by-pass		1.25 $^o/oo$
Core concentration	H_2O	0.26 vpm
	CO	1.80 vpm
	CO_2	0.01 vpm
	H_2	1.80 vpm
Outlet concentration after gas purification	H_2O	0.20 vpm
	CO	0.05 vpm
	CO_2	0.20 vpm
	H_2	0.05 vpm

These specifications can be made technically feasible, but there is still some uncertainty with regard to the chemical reactivity with respect to the catalytic effect of the fission products on oxidation. As mentioned before, we will try to clear up these questions in our experimental programme. Furthermore, little is known about the corrosion of graphite under irradiation.

As we have seen, the cooling gas must be kept clean to prevent carbon deposition and mass transfer. Impurities are caused by gases which are adsorbed in the graphite and by leakages from the heat exchangers. Furthermore, small quantities of oxygen and nitrogen are introduced continuously into the helium during the insertion of the fuel elements. The chemical impurities are the following : H_2, H_2O, O_2, CO, CO_2, CH_4 and N_2. The nitrogen does not take any part in the reactions of the mass transfer and is therefore not to be taken into account here.

The customary process today, which is also used in the AVR and Dragon Project, purifies the gas by adsorbing the different impurities at the temperature of the liquid nitrogen. This adsorption method has the advantage that the different adsorbers can be regenerated by heating them up and sweeping them with pure gas. The disadvantage is that such a purification plant is technically complicated and that a large quantity of liquid nitrogen with a high purity level has to be used. In addition the cooling gas has to be cooled down from the temperature at the exit of the heat exchanger to about -190°C for a larger by-pass. This is costly and therefore undesirable.

A further possibility is purification by chemical binding. Our investigations on the gas purification system for the THTR Project have dealt with this problem. Our aim is to remove these impurities at temperatures of 240-300°C, i.e., at the temperature of the cooling gas at the exit of the heat exchanger. All the impurities mentioned can be converted into H_2O or CO_2. The problem is thereby reduced to a three-step procedure :

1. Oxidation into H_2O or CO_2, at the same time removal of oxygen

2. binding of CO_2

3. binding of H_2O

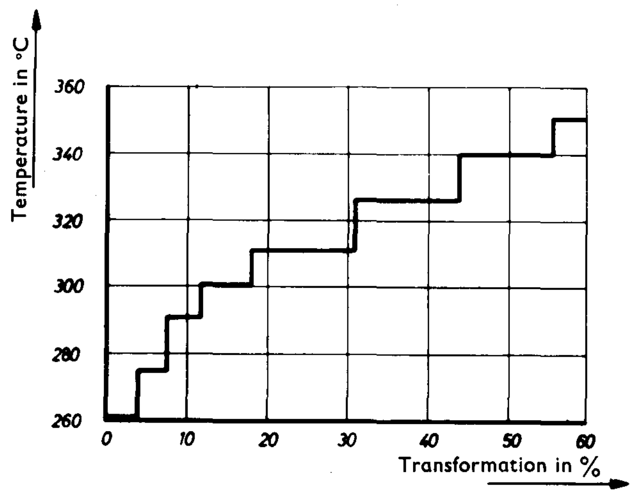


Fig. 4 — Transformation of Ca $(OH)_2$ into $CaCO_3$ as a function of the temperature.

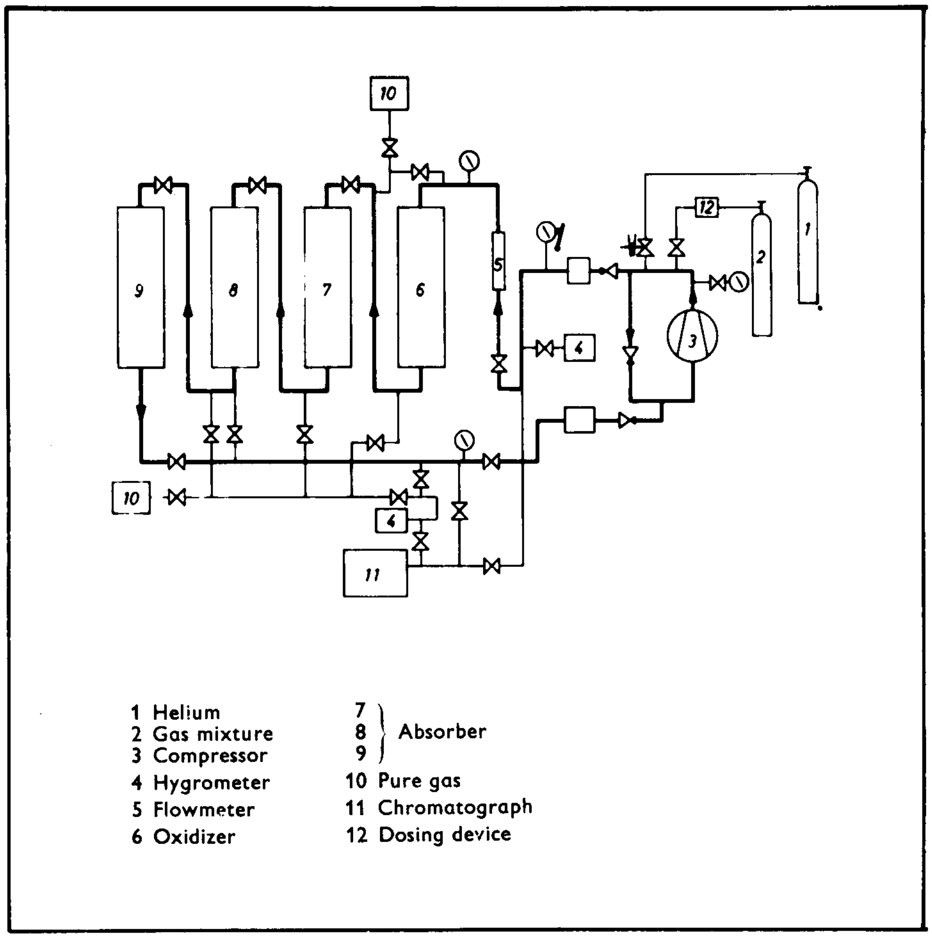


Fig. 5 — Schematic representation of the rig used for the laboratory tests.

Determinations of the oxidation of CO and H_2 on BTS contacts at temperatures of up to 260°C have been carried out. The oxidation, even at lower temperatures (minimum 150°C) does not create any difficulties. The transfer of the less important CH_4 into CO_2 and H_2O takes place quantitatively only at higher temperatures of 500-600°C on copper oxide. Oxygen is quantitatively bound on BTS contact. The BTS contact can be regenerated up to 260°C. In the further course of our examinations the temperature range will be extended to 300°C.

For the binding of CO_2 we examined in particular calcium hydroxide. The reaction follows the equation

$$Ca(OH)_2 + CO_2 \rightleftarrows CaCO_3 + H_2O$$

Technical application of this reaction is hampered by the fact that a layer of $CaCO_3$ is formed on the surface of the $Ca(OH)_2$ particles, which prevents further reaction and limits the absorption ability of the filter. As the CO_2 filter cannot be regenerated, however, a high conversion rate is absolutely essential. Therefore the formation of this layer has to be avoided. This can be done by producing $Ca(OH)_2$ particles with a very high inner surface area and by a special temperature variation during the reaction. A typical temperature curve for the reaction can be seen in Fig.4. With fresh filter material we work, for instance, at temperatures of 260°C. If the outlet concentration of CO_2 increases too much, the temperature of the filter will be increased by 15°C. The reaction ability of the filter is thus increased and the outlet concentration of CO_2 drops again down to the value desired.

The final temperature achieved can be about 350°C. Using this procedure we could obtain a conversion rate of 60%. The outlet concentration of CO_2 at inlet concentrations of 100 and 5000 ppm was lower than 1 ppm.

At the moment we are dealing with parameter studies. We found that the necessary minimum temperature of the filter material depends heavily on the inlet concentration. The maximum temperature of the bed is limited by the water loss from the $Ca(OH)_2$. The aim of the further development work is to avoid the technically expensive temperature regulation of the bed.

For the <u>binding of the H_2O</u> the following two reactions were examined :

a) $CaO + H_2O \rightleftarrows Ca(OH)_2$

b) $BaO + H_2O \rightleftarrows Ba(OH)_2$

The dissociation of $Ca(OH)_2$ above 150°C is already too high. Therefore we feel that the use of BaO filters is a suitable solution. At a filter temperature of 300°C we could reach a conversion rate of 80%. The outlet concentration of water had been so low that we could not detect it with the devices available to us. The water inlet concentration varied between 100 and 10,000 ppm. During further investigations we will try to find out the best filter temperature at gas pressures up to 40 atm.g.

A flow sheet of the rig used for these experiments is shown in Fig. 5. Different absorber units can be connected up. A special apparatus enables us to feed in different mixtures of impurities at controlled concentrations. There is also a flowmeter, a hygrometer, a Wösthoff apparatus and a gas chromatograph. The gas is circulated by a diaphragm compressor.

To sum up, the results obtained so far in the development of a new gas purification plant are very encouraging. The importance of these experiments is obvious when we consider that the gas purification plant uses gas directly as it leaves the steam generator without an expensive cooling system.

A larger by-pass can thus be built at a cost which is technically defensible. With such a gas purification system, the cooling gas can be kept so clean that mass transfer and graphite corrosion are reduced appreciably.

Experimental Studies on the THTR Heat Exchangers

Dipl.-Ing. F. Scholz

The experimental studies on crossed tube arrangements are aimed at ensuring the economic and safe design of the thermally highly-stressed and compact steam generator for the THTR. Crossed tube arrangements, in which the successive tube planes lie direct on top of each other perpendicular to the flow direction make for very compact construction and ensure relatively good utilization of the circular cross section, which is necessary since the integral design and the use of a prestressed concrete pressure vessel, mean that it must be possible to remove these parts.

The first figure shows one of the tube bundles tested with the water inflow and outlet pipework.

It seemed advisable to check the measurements made by other authors recently with regard to the special requirements of the THTR steam generator. In particular, the heat flux should be directed from the gas to the water, the heat fluxes and temperature differences should be as high as possible and all the successive tube planes should be involved in the heat exchange. These studies were carried out in the high pressure gas tunnel, which is particularly suitable for such experiments.

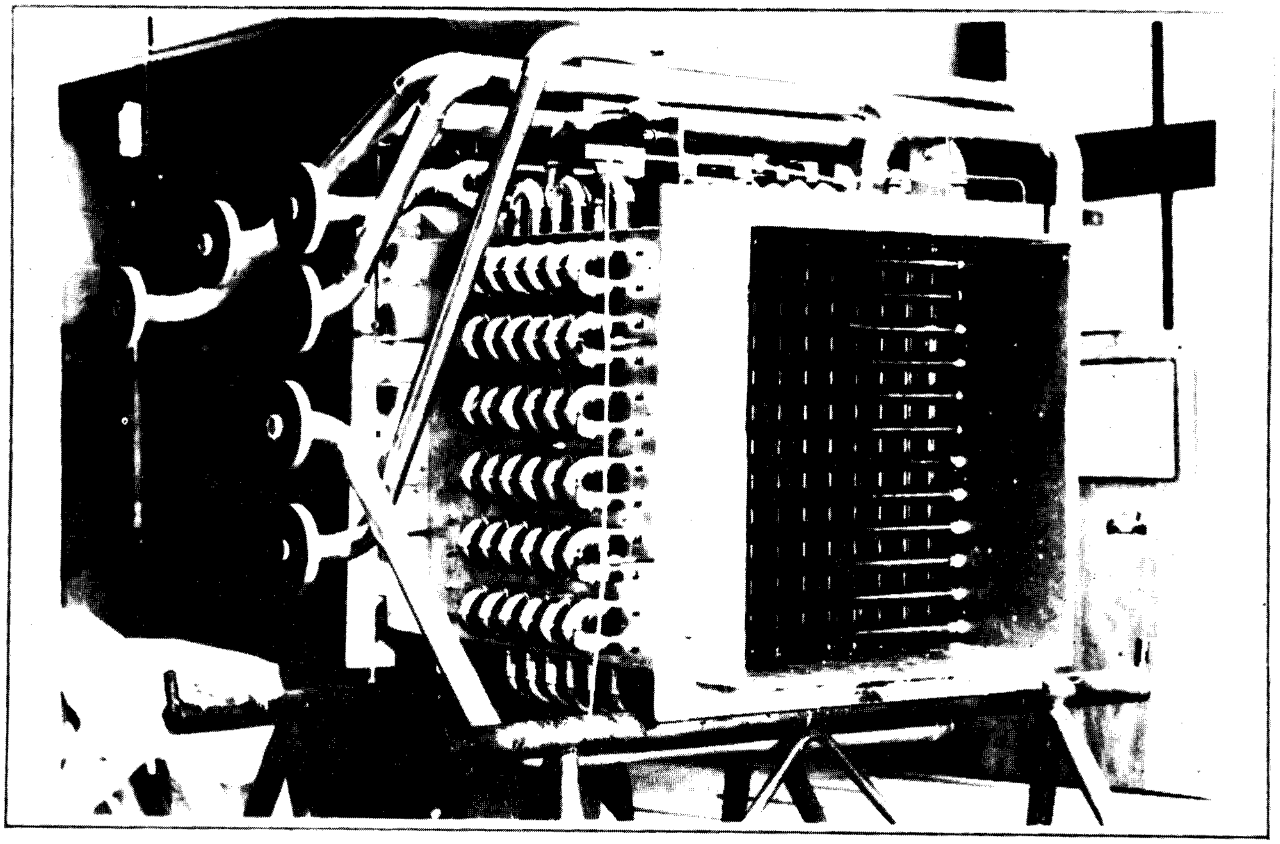
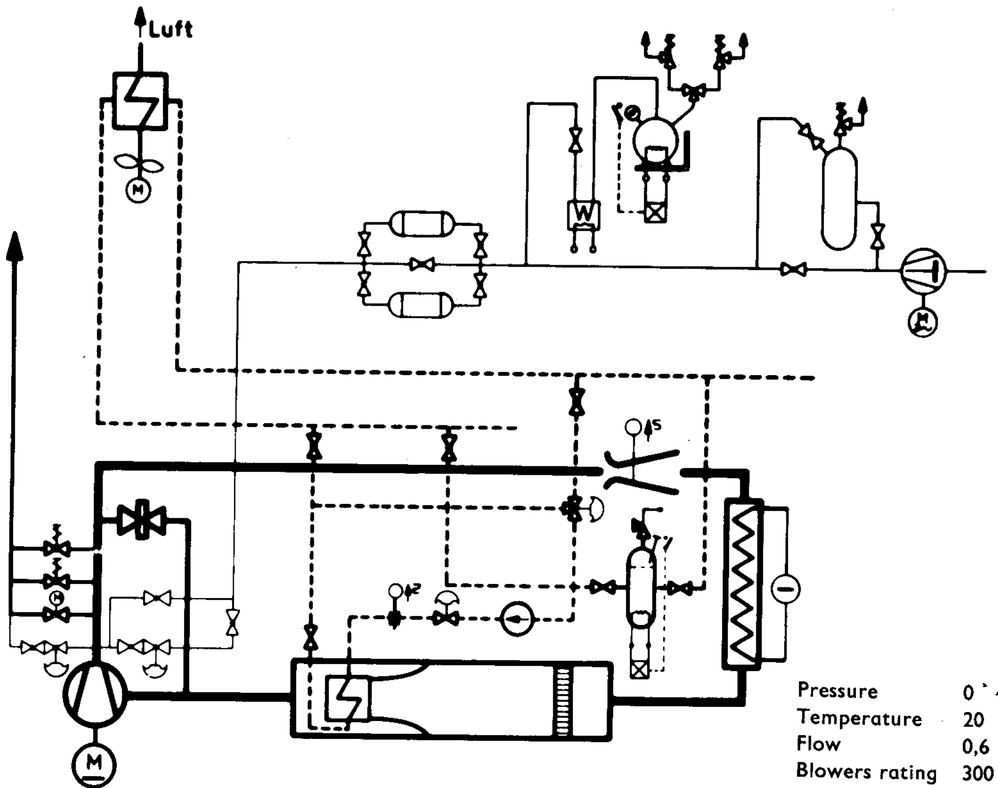


Fig. 1 — Crossed aligned tubes bundle.



Pressure	0 ˙ 40atü
Temperature	20 400°C
Flow	0,6 6m³/s
Blowers rating	300 KW
Heating power	0 2 MW

Fig. 2 — Sketch of the High Pressure Gas Channel.

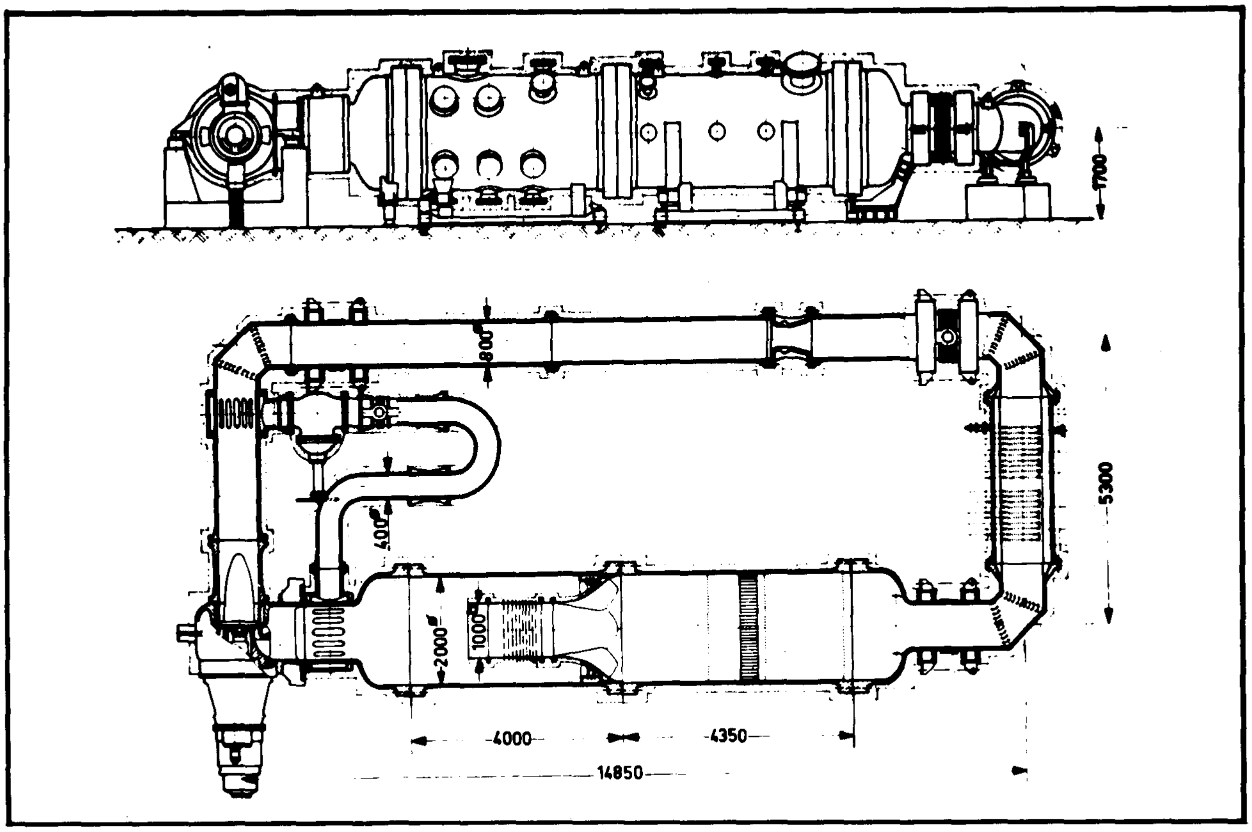


Fig. 3 — High Pressure Gas Channel.

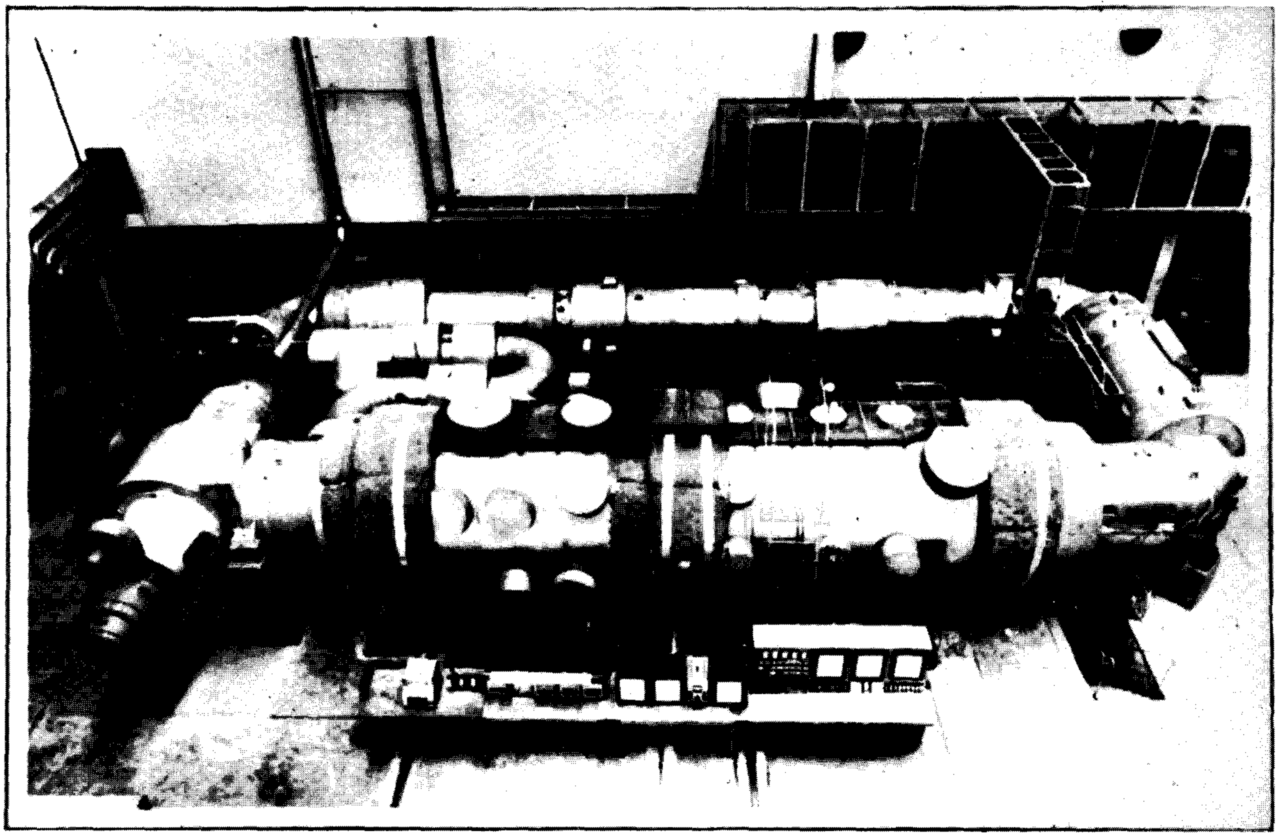


Fig. 4 — High Pressure Gas Channel.

Fig. 2 shows the flow scheme and the main data concerning this experimental facility, which has already been described in the literature. An axial-flow blower circulates the operating gas, which is pressurized air or CO_2, at pressures of up to 40 atmospheres and temperatures of up to 400°C. The gas flows from the blower through a measuring line of about 800 mm inside diameter into an electrical resistance heating system, where a power of up to 2000 kW is produced. From there the gas flows into the settling chamber, which is equipped with screens and a honeycomb, the inside diameter being 2000 mm. For the further improvement of the velocity profile the gas stream is accelerated in a nozzle forming the square test section 1 m by 1 m, transfers the heat to the test object, which is cooled by water from a closed cooling system, and is sucked up again by the blower. The nozzle which converts the circular flow area into a square flow, together with the test object, are mounted in the test section pressure sleeve.

The next figure (Fig. 3) shows a simplified construction drawing with the main dimensions. At the bottom the main structural components are visible, such as the blower, the measuring line, the heating system, the settling chamber and the test section with the nozzle and the test bundle.

Fig. 4 then shows the unit ready for operation with the control desk and the measuring apparatus. The very high gas Reynolds numbers of up to 2×10^6 which can be obtained by CO_2 or by compressed air are, however, not achieved in the steam generator of the THTR, where they are round about a factor of 50 to 100 lower, but these very high Reynolds numbers are necessary to get very high heat transfer coefficients. To obtain the same high heat transfer coefficients for 40 atm. and 400°C with air and helium, one needs air Reynolds numbers 5-15 times as high as in case of helium because of the high thermal conductivity of helium. The factor 5 or 15 respectively depends on the Reynolds range.

The following table shows the main data for the two tube arrangements studied.

Tube arrangement	Crossed in line	Crossed staggered
Tube outside diameter	$D = 32$ mm	32 mm
Longitudinal pitch	$S_l = 32$ mm	32 mm
Transverse pitch	$S_q = 57.6$ mm	48 mm
Number of tube planes in the direction of the flow	12	12
Number of tubes in one plane	12	15

The next figure (Fig. 5) shows the results of the crossed in line arrangement. We have plotted Nusselt times Prandtl to the -0.5 and the pressure loss coefficient ζ over the Reynolds number. The thermal properties are related to the mean gas temperature and the mass flow to the minimum flow area of a tube plane. The reference length is the tube diameter. Experiments were carried out at gas pressures of 1-40 atm., at gas temperatures of 50 to 400°C and temperature differences between the gas and tube surface of up to 300°C. The average mean square deviation of the measuring points from the lines marked here is about 3% in the case of the heat transfer values. For the pressure loss coefficient, such a figure can be given only for the lower Reynolds range, because ζ cannot be described throughout the whole Reynolds range by simple exponential laws. There is a remarkable change in the slope of the heat transfer values and also a typical change in the pressure loss at Reynolds numbers of about 1.5 to $2 \cdot 10^5$, as was found recently also by other authors. In addition, different symbols were used for the different ranges of the mean average logarithmic temperature differences between the gas and tube surface in order to determine the possible influence of large temperature differences. One can see that such an influence is non-existent, or at least is not larger than the scattering of the measured values. What has been said about the results of the crossed in line bundle is nearly the same as in the case of the crossed staggered arrangement.

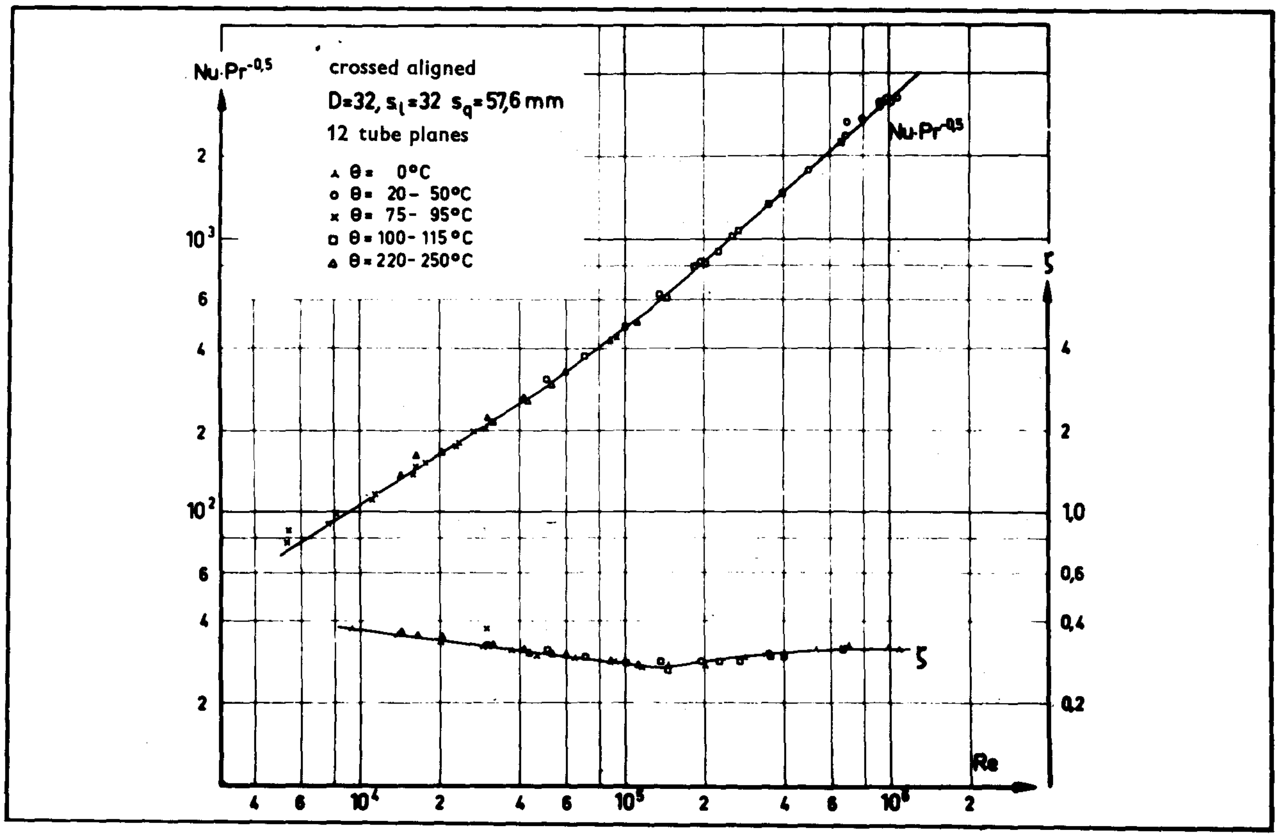


Fig. 5 — Heat transfer and pressure drop for the crossed aligned tubes arrangement.

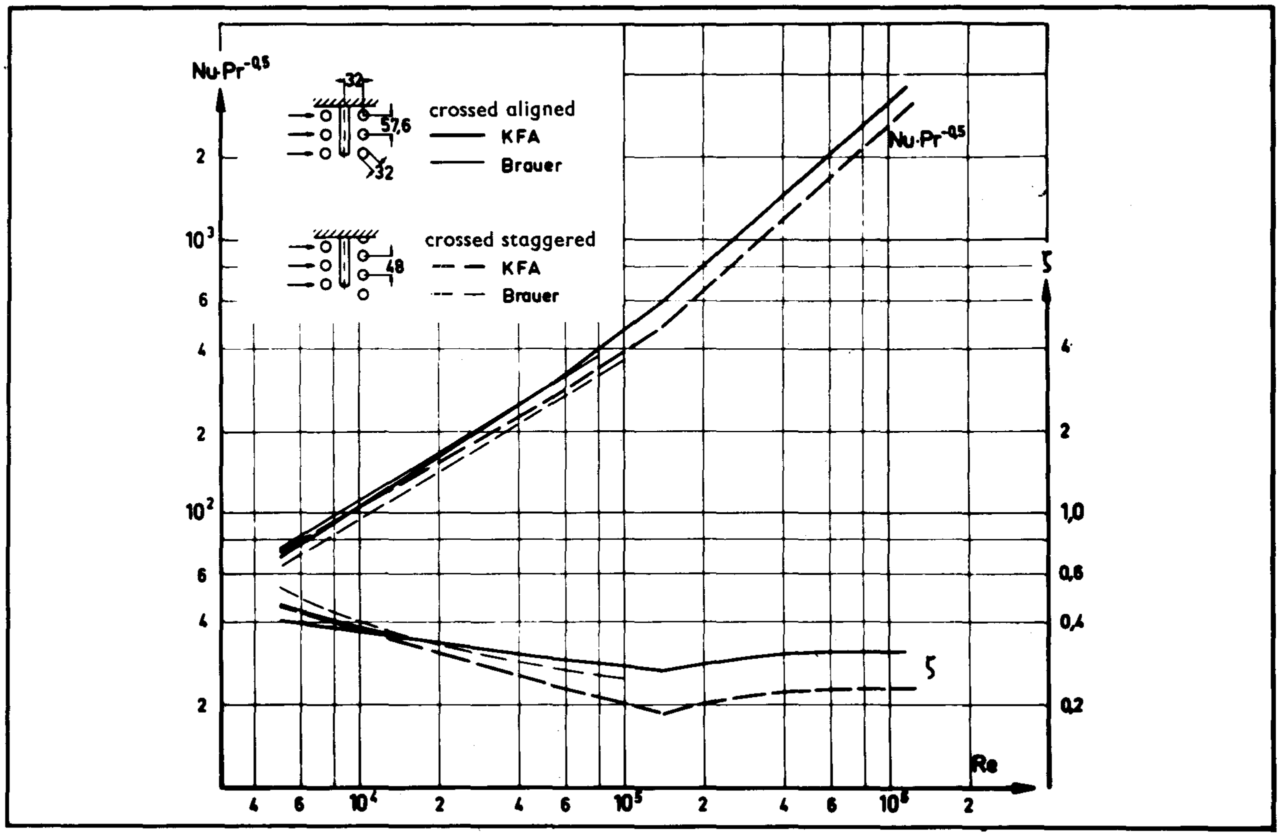


Fig. 6 — Heat transfer and pressure drop
for two crossed tubes arrangement.

Fig. 6 shows a comparison between the two tube bundles and also, in the thinner lines, gives the results which Brauer obtained in the lower Reynolds range. A satisfactory agreement of the results is found in the common lower Reynolds range, but it is also quite clear that it is not sufficient to carry out such research work only in a limited Reynolds range and to extrapolate far beyond that range.

The experiments shown should be regarded as scientifically basic studies on the assembly and experimental conditions. Probable deviations under truly operational conditions should be studied in further experiments. For instance, in an actual steam generator all the tube bends are also in the gas stream. This is a structural fact which influences the heat transfer as well as the pressure loss. Furthermore, it would have to be considered whether the influence of operating gas should be experimentally tested. The high-pressure gas tunnel would be suitable for this. Because of the high cost of such research work, it was decided for the moment not to carry out further experiments of this type. The results of further studies are being awaited, which will fix the precise structural design of the steam generator, including economic and manufacturing problems. After this, two or three of the most promising tube arrangements will have to be examined in detail. It should also be mentioned that, because of the very high heat flux, some special experiments may be required, particularly for flow stability problems in the vaporisation range and in regard to the high thermal stresses to be expected.

Considerations on the Design of Fuel Elements and Structural Graphite

Dr.Dr. H.J. Stöcker

The question of the materials used in the THTR core is peculiar in the sense that only one chemical element is involved, namely, carbon. The core contains about 300 kg of U-235 and about 7000 kg of thorium, about 600 kg of helium as the coolant and about 400,000 kg of carbon in the fuel elements and the reflector. In the case of this reactor, therefore, the earlier name of "atomic pile" would be most applicable.

Depending on whether the fuel elements or the internals are being considered, the graphite has to meet different requirements, both with regard to the radiation resistance and the mechanical properties. The spherical fuel elements must have a high mechanical strength, since they are dropped onto the pebble bed during loading. The strength properties of the reflector graphite do not have to be so good, since it merely has to carry its own weight. On the other hand, the fuel elements receive a lower radiation dose than the reflector graphite, because they are only in the reflector for a short time. In line with these different requirements, the same type of graphite will probably not be used for the internals and the fuel elements.

The design data for the structural graphite and the fuel elements are drawn up on the basis of the following considerations. During a useful life of 30 years, the <u>reflector internals</u> receive a fast dose of about $3 \cdot 10^{22}$ nvt. As a result of this the dimensions of the graphite blocks are altered, because unfortunately no graphite exists which can resist volume changes at this radiation dose. Provided they do not exceed a certain amount, the macroscopic changes in the dimensions can be offset by appropriate structural measures. In addition, flux and temperature gradients occur in the reflector blocks which can cause internal stresses. The question is whether these stresses in the graphite blocks are in excess of the strength values of the graphite.

In the case of the thermal stress this is not the case, as was proved by means of quenching tests on entire reflector blocks. The radiation-induced stresses can only be studied experimentally on a limited scale, since high-flux radiations on large blocks are not feasible. As a result we can merely calculate from the irradiation results available for small blocks that no stresses will occur in the reflector blocks which will cause cracks.

A type of graphite must thus be selected which has been adequately tested in the 700-1000° C temperature range. Reflector graphite types were therefore subjected to long-time testing under the THTR irradiation programme.

The fuel elements consist of graphite balls 6 cm in diameter, containing the fuel in the form of coated particles. The diameter was calculated by optimization calculations for the blower power, the thermal power per ball and fabrication costs per ball. Smaller pellets would require a higher blower power and would also be more expensive, since more would be needed.

With regard to the fabrication costs, a larger diameter would be better, but this has an upward limit, since for the same power density the available surface decreases as the ball diameter increases. This means that more energy must be given off per ball surface, which in turn causes a rise in the temperature gradient and the fuel temperature. As things stand at present, however, this temperature should not exceed 1350° C, since the retention of the fission products is dependent on the fuel temperature. We are thus playing safe, because coated particles can be heated up to 1700° C for short periods without damage.

The requirements placed on the drop impact strength of the elements are fairly high. They must be able to withstand at least 50 drops from a height of 2 m. This is a requirement which is reflected in the fabrication costs, since not all types of graphite fulfil these conditions. A more stringent requirement was at one time imposed by the pneumatic feed unit in the AVR reactor, but we are working on a solution which will enable the balls to fall on the pebble bed somewhat less violently. Cheaper types of graphite could then be used, such as the extruded graphites originally intended.

Other conditions imposed on the fuel elements are good wear resistance, since the formation of graphite dust in the reactor must be kept as low as possible. The graphite should be as pure as possible, not only because of the neutron economy, but also on account of the unwanted catalytic acceleration of the oxidation processes by the presence of impurities on the surface of the elements. As a result of the irradiation tests carried out during the past year in collaboration with Dragon, the oxidation mechanism is now better understood and our purity requirements can be restricted to less special elements, such as iron.

The fuel consists of coated particles. These are small kernels of uranium-thorium carbide or oxide, 0.2 to 0.6 mm in diameter, on which layers of carbon or silicon carbide are deposited from the gaseous phase. These coated particles have especially good retaining properties for fission products. The present record is held by Oak Ridge particles, which at a temperature of 1700° C up to a burn-up of 25 % of the heavy metal used gave off less than 10^{-6} % of the gaseous fission products formed.

A brief account of the development work carried out in the past year is given below. The main effort was directed to perfecting the methods for fabricating coated particles and also to the development of two types of fuel elements.

In the case of the coated particles a considerable development gap had to be made good. It was not until September 1964 that the first 1 kg batch could be made. Since then 60 different types of coating have been applied and tested. Out of these, 17 were selected for irradiation tests. So far eight tests have been carried out and the results have shown that we have in the meantime achieved at least the same quality as that of particles produced elsewhere. In the coming year we are only going to produce a few special types of coatings and place particular stress on irradiation experiments in the FRJ-2 at Jülich, the BR-2 at Mol and the Dragon Reactor. In BR-2, for instance, coated particles are being irradiated with a greater dose of fast neutrons than they would ever accumulate during the useful life of a fuel element.

In the case of the fuel elements themselves, development work has been devoted to three main types, as shown in the figure 1. These three variants are known as the tapestry type, the synthetic type and the annulus type. Those of you who have followed the work carried out for AVR and THTR in the past will see that these basic concepts have not changed much and you will wonder why we haven't thought of something better. In reply, I should point out that a few other types have since been proposed, but on closer examination these were found to be merely variations of the two basic types named.

The purpose and development stage of the three types shown differ appreciably. The first variant - the so-called <u>tapestry variant</u> - has been developed the farthest and is thus the most reliable at the moment. This type is to be used as the AVR replacement charge. It is similar to the first charge supplied by UCC for the AVR reactor, except that the one shown here differs from this variant in that the fuel is arranged in a <u>region</u> as opposed to the <u>homogeneous</u> distribution of the UCC element. This gives a fuel temperature which

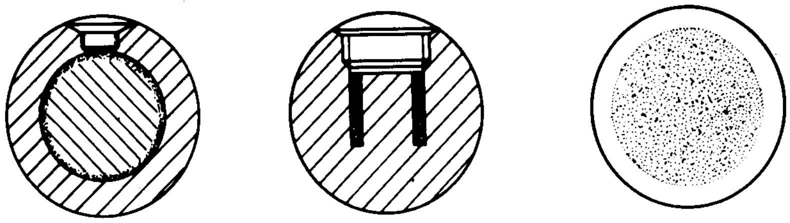


Fig. 1 — Fuel element types for high temperature gas cooled reactor.

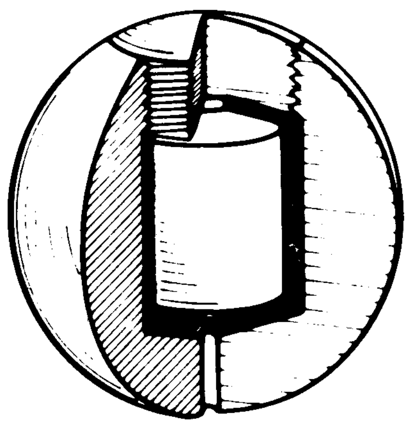


Fig. 2 — Modified annular gap type.

is about 100^{o} C lower. In addition, we have pressed ahead with the development of the cheaper synthetic variant shown here, which is relatively simple to make. The drop impact strength and the other properties required of it are already satisfactory. Since this type is compacted from a mixture of graphite powder, binder and fuel particles, the ball cannot be heated up to the graphitization temperature of 2500^{o} C, since at this temperature uranium would diffuse through the particle coating. The ball thus contains a larger quantity of ungraphitized carbon, which, as is common knowledge, undergoes marked shrinkage under irradiation at high temperatures. A start was therefore made this year on irradiation tests on synthetic graphite and next year entire balls are to be irradiated.

The third type is the so-called annulus variant, containing loose particles. The use of loose particles offers the advantage that if a hole is drilled in the element and the fuel particles emptied out, they can be separated from the element, which is very favourable for processing and possible reprocessing later. The annulus type is also made of electrographite, which is fairly expensive, and we are interested in developing other cheaper types containing internal cavities in which the loose particles can be housed. A possible variant is shown in the next picture (Fig. 2).

It is made in the same way as the synthetic variant, except that one part of the mould pressed into the pellet is made of a material which can be removed by subsequent heat treatment or thermally disintegrated. Loose particles can thus be put in the cavities remaining. Since the pellets do not contain any fuel at the beginning, there is no fixed temperature for the heat treatment.

The Fabrication of Fuel Elements

Dr. B. Liebmann

As was stated previously, the THTR fuel elements will consist of graphite balls containing uranium and thorium in the form of coated particles, just as those for the AVR reactor. Fabrication of the fuel elements can be divided up into three stages: fabrication of the particles, coating of the particles and fabrication of the actual fuel elements.

Two fundamentally different methods are used in the fabrication of the particles: the granulation technique and the sol-gel process. Fig. 2 shows a block diagram of the granulation method for the fabrication of uranium-thorium carbide particles. Details are also given of those stages which are not required in the fabrication of oxide particles, thus showing that oxide particles are cheaper than carbide particles. The basic developement work on the granulation technique is largely completed. For the actual granulation operation, comparative studies were conducted on two possible methods - granulation in the Lödig mixer and granulation on the gyrating bowl. The comparison showed that the gyrating bowl technique is less expensive. The work on the Lödig mixermethod was therefore halted a few weeks ago. A total of a few hundred kg particles were made by the granulation technique in the course of tests and irradiation experiments, i.e., roughly the same quantity as is required for the first AVR core charge.

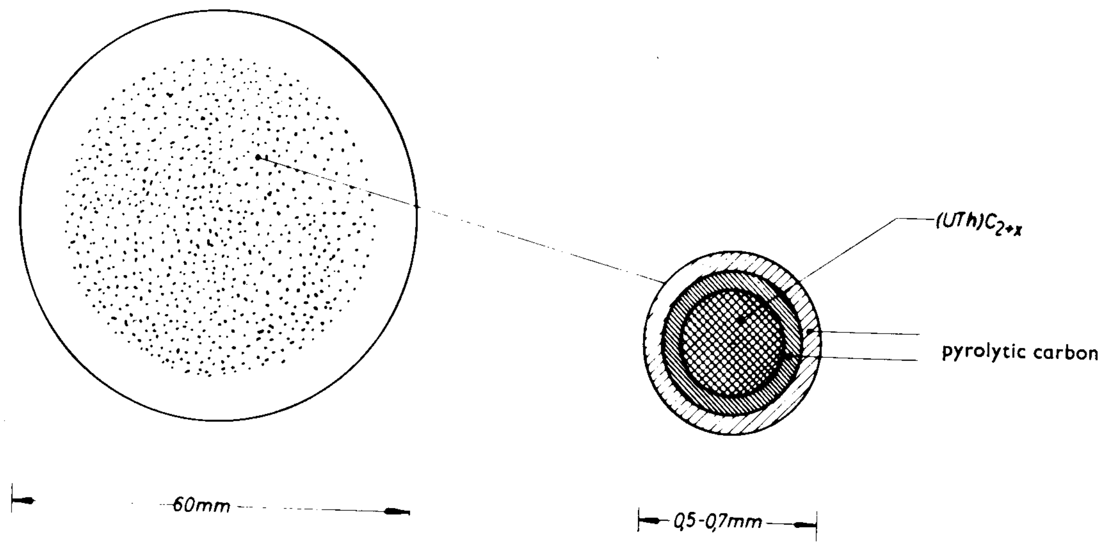


Fig. 1 — Fuel element with coated particles.

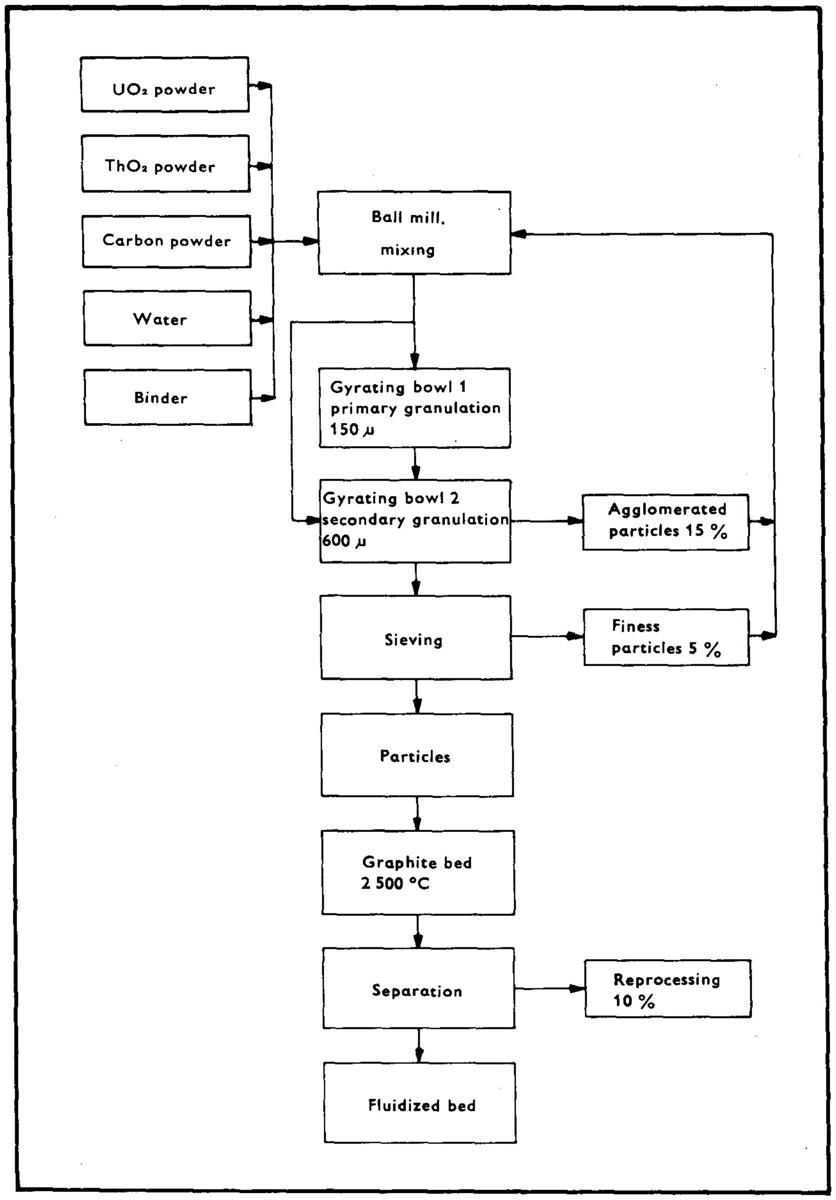


Fig. 2 — Preparation of particles gyrating bowl method.

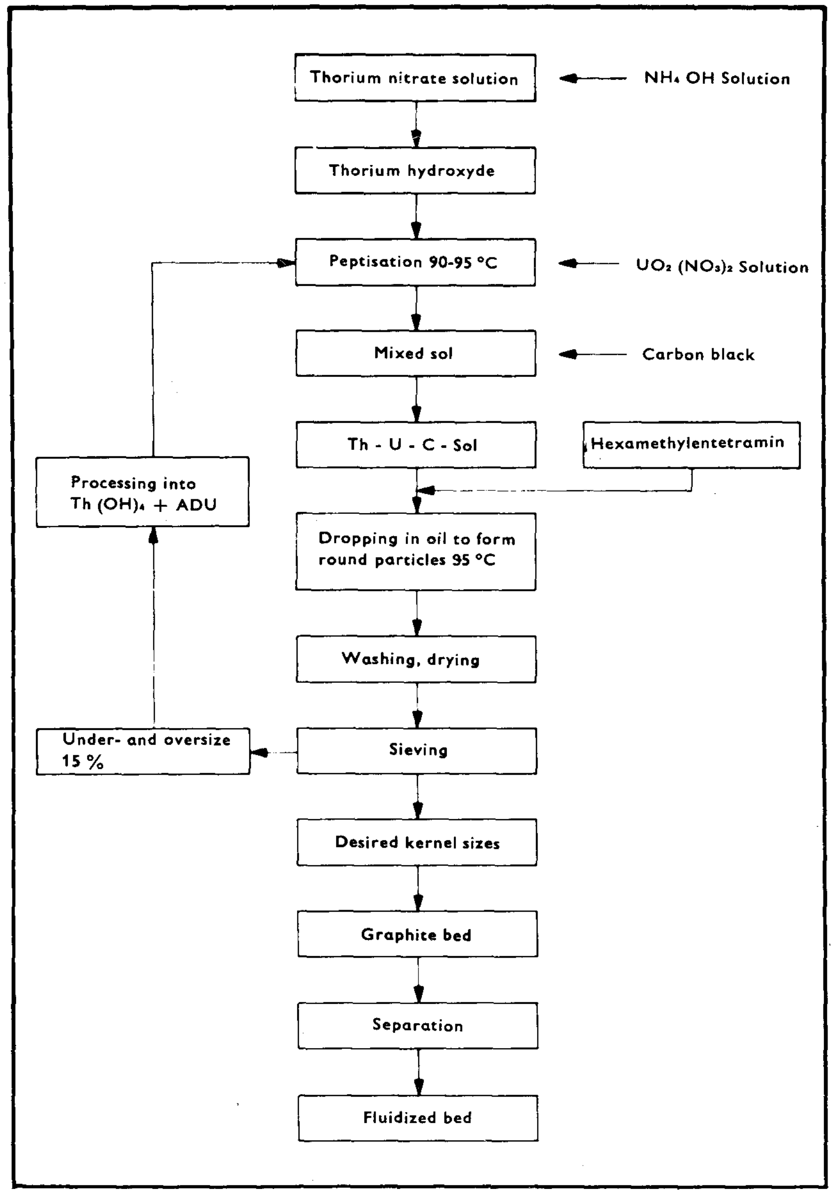


Fig. 3 — Preparation of particles Sol. gel method.

Laboratory-scale development of the sol-gel process for the production of carbide particles is now completed. The first specimens containing particle cores made in this way are to be irradiated in 1966. The development and fabrication of pure UO_2 and ThO_2 particles is still in its infancy (Fig. 3).

On completion of the work now being carried out, it will be possible to decide which of the two techniques is economically more interesting. For the fabrication of relatively small quantities of particles, such as those used for the AVR fuel elements, the granulation method is perfectly adequate. Should the demand increase later, as is likely, however, the sol-gel process would be less costly. Moreover, it would offer advantages in the processing of active U^{233}-containing fuels from reprocessing plants.

For coating the particles with pyrolytic carbon, the fluidized bed has gained wide currency. The following picture shows a flow diagram of our 1 kg bed. With the aid of the fluidized bed shown here, 1 kg of particles can be coated in 3-6 hours, depending on the type of coating required (Fig. 4).

Before we built the fluidized bed, the correlation between the properties of the coatings and various parameters was examined in detail by means of preliminary tests on a small-scale unit. The main parameters were: the type of coating gas, the type of carrier gas, the partial pressure of the various gases and the total pressure and temperature in the coating room.

On completion of the preliminary tests, a decision had to be taken with regard to the size of the fluidized bed to be built. It was soon found that it is hardly possible to transfer experience gained with the coating of 10-20 g charges to charges of the order of magnitude of 1 kg. On the other hand, fairly large quantities of particles are required for development work using kg charges. Since, however, sufficient quantities of particles were available, we decided to build a 1 kg fluidized bed. The apparatus has proved highly satisfactory and we have not regretted our decision. So far, particle quantities containing about 100 kg of heavy metal were handled without incident during the course of the current experiments and the fabrication of irradiation specimens.

With this apparatus it is possible to make pyrolytic carbon of columnar, laminar and isotropic structure. Depending on the structure, the deposition rate can be up to well over 100 μ/hr, the coatings being applied at high temperature (up to 2100° C) and at a density of 2.1 g/cm^3. In addition, SiC layers are also deposited by the thermal decomposition of methyl trichlorosilane. The next two pictures show ground sections of particles covered with different types of coatings.

Coating at underpressure is, we feel, a very interesting development. The bed was appropriately modified. In the tests we carried out total pressures of down to 20 Torr; we obtained in the reaction space, reasonable deposition rates of up to 20 μ/hr and excellent coating qualities being achieved. In the 1400-2000° C temperature range isotropic, laminar and columnar structures are obtained, with very satisfactory density, compression resistance and surface contamination values. By carrying out the coating at underpressure, the amount of carrier gas required could be reduced (normally about 2000 l/hr, cost about DM 30, for six hours coating). Because of this, coating at underpressure has acquired a considerable economic and qualitative importance.

The coated particles can be inserted in the fuel elements in different ways. Basically, there are three types of fuel elements: the tapestry type, the annulus type and the compacted element (cf. Fig. 1 in the previous talk).

The development of hollow sphere fuel elements is most advanced at present. Elements of this type are being considered as the replacement charge for the AVR reactor. The hollow graphite balls are made of special block-compressed graphites supplied by German and French firms. These graphites must possess high density, strength, heat

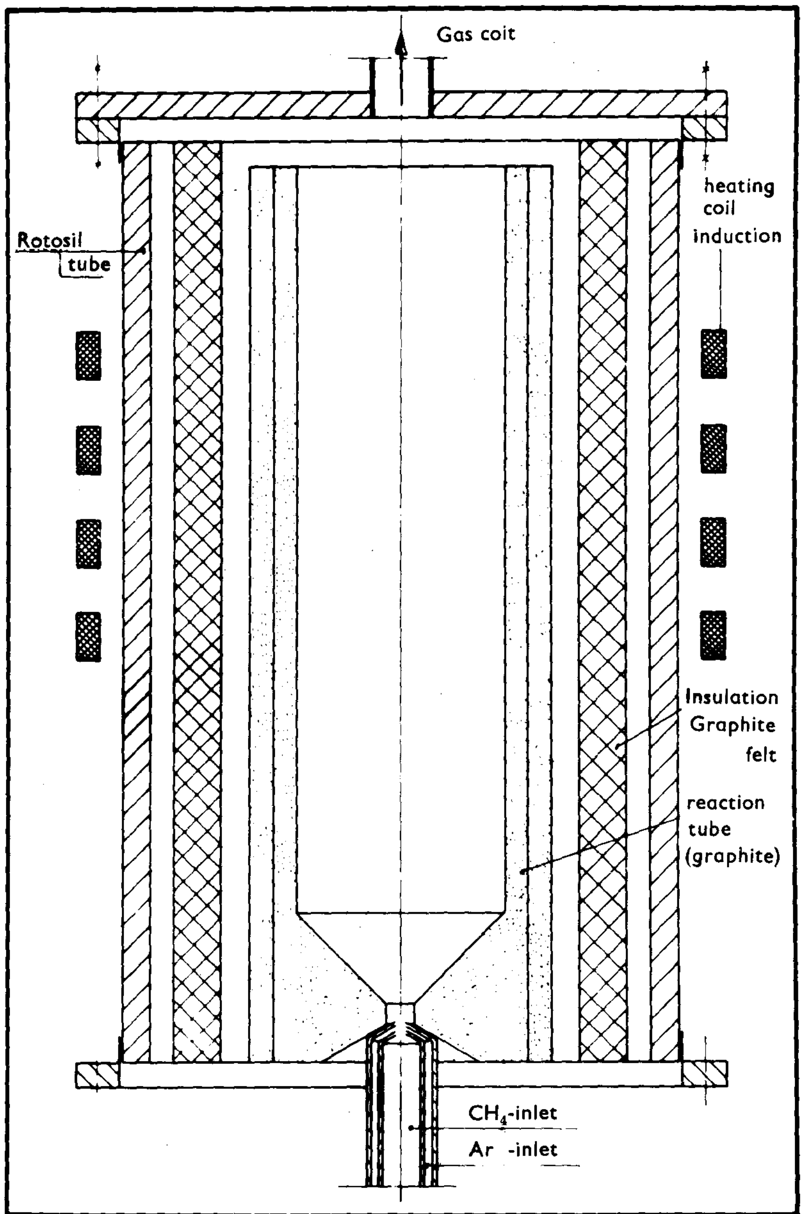
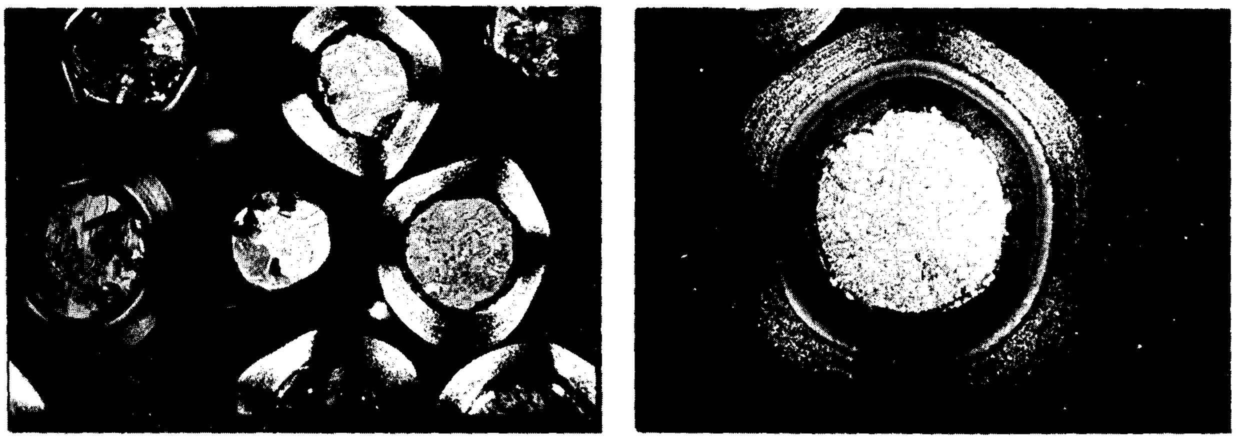


Fig. 4 — Fluidized bed.



B 25

Fig. 5 — Coated particles micrographs.

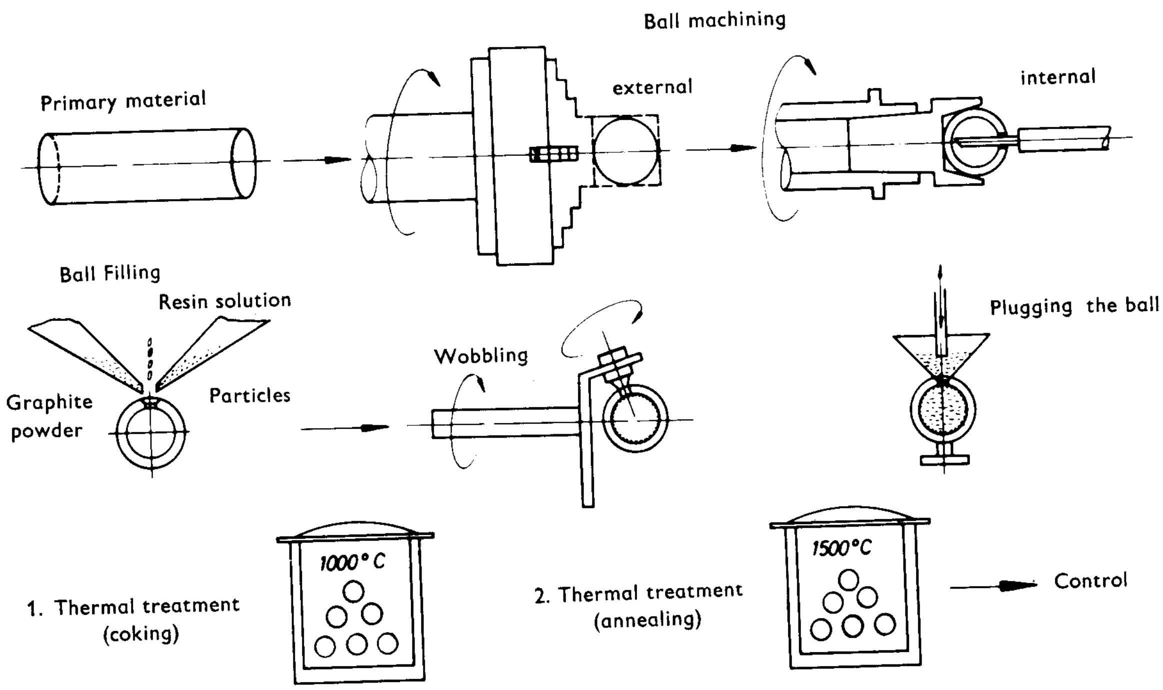


Fig. 6 — Preparation of hollow sphere fuel elements.

conductivity, radiation resistance and oxidation resistance properties. It is especially important that the hollow balls have a good drop impact strength. In the type of fabrication described here, the centre of the fuel elements has no effect on the drop impact strength. All the stresses must be borne by the shell of the sphere. The AVR elements have to withstand at least 50 drops from a height of 4 m on to a graphite bed without suffering any damage. The figure 7 shows the number of drops as a function of the drop height. It can be seen that the number of drops depends to a very marked degree on the height. At a height of 2 m the spheres withstand about 7-10 times as many falls as from 4 m. By means of structural modifications the dropping height in the THTR project is to be reduced to 2 m. In this case the strength properties of the spheres need not be so high, so that cheaper types of graphite could be used.

Hollow sphere fuel elements were irradiated in the Risø test reactor in 1964 under the maximum temperature conditions of the AVR reactor. The power per sphere was a maximum of 2.4 kW. At a sphere surface temperature of 1050° C, this gave a central temperature of 1250° C. No major change was found in any part of the fuel elements even after 20 % burn-up.

Finish-machined graphite balls are not required for the fabrication of the compacted graphite fuel elements, which are simply made by pressing together a mixture of graphite powder and the fuel particles. Since the fuel element balls are made from a mixture of graphite powder and fuel particles, it is not possible to heat the spheres up to temperatures of over 2000° C, since the particles would otherwise be damaged. Whereas a graphitization temperature of 2700-3000° C is normally required for the fabrication of graphite bodies from artificial carbon, a crystalline rearrangement takes place at temperatures as low as 1600-2000° C in the compacted masses used. This leads to a two- to five-fold improvement in the drop impact strength, the thermal conductivity and the oxidation resistance of the graphite spheres as compared with those used hitherto, in which coking of the

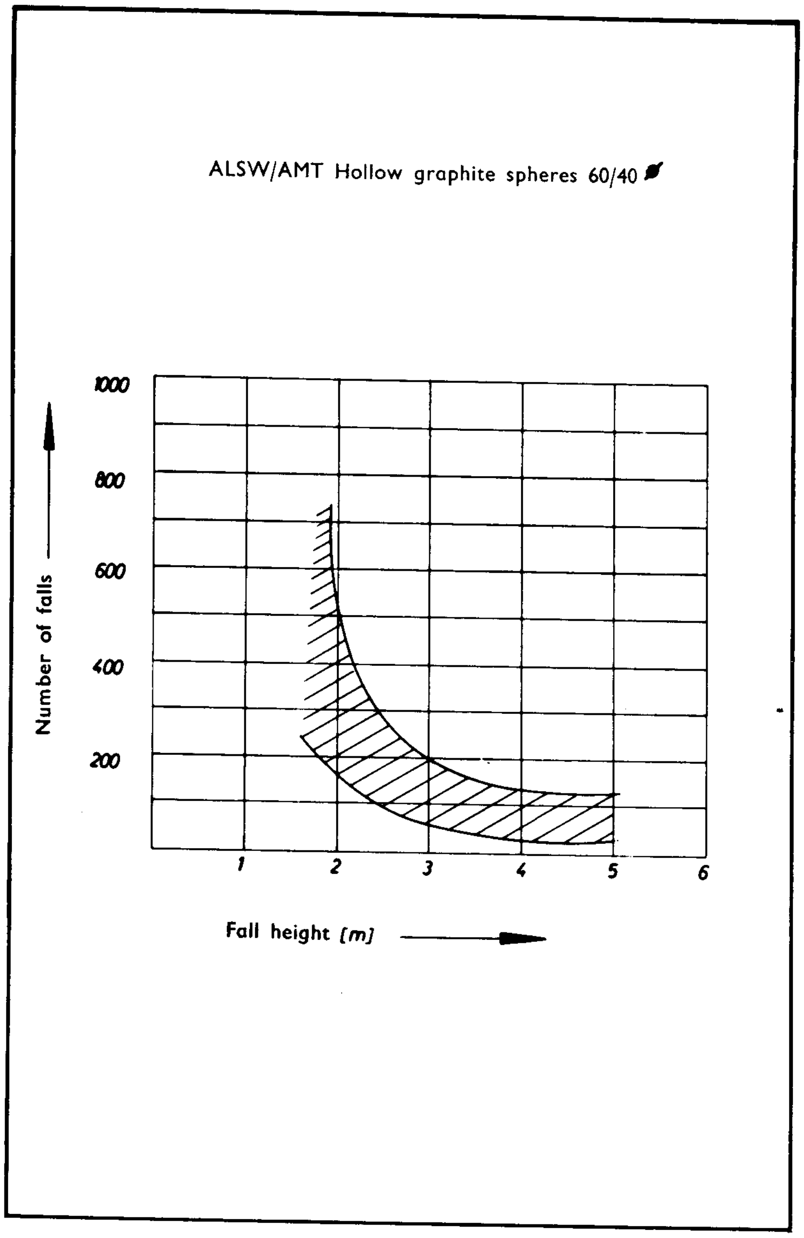


Fig. 7 — Fall tests on a bed af balls.

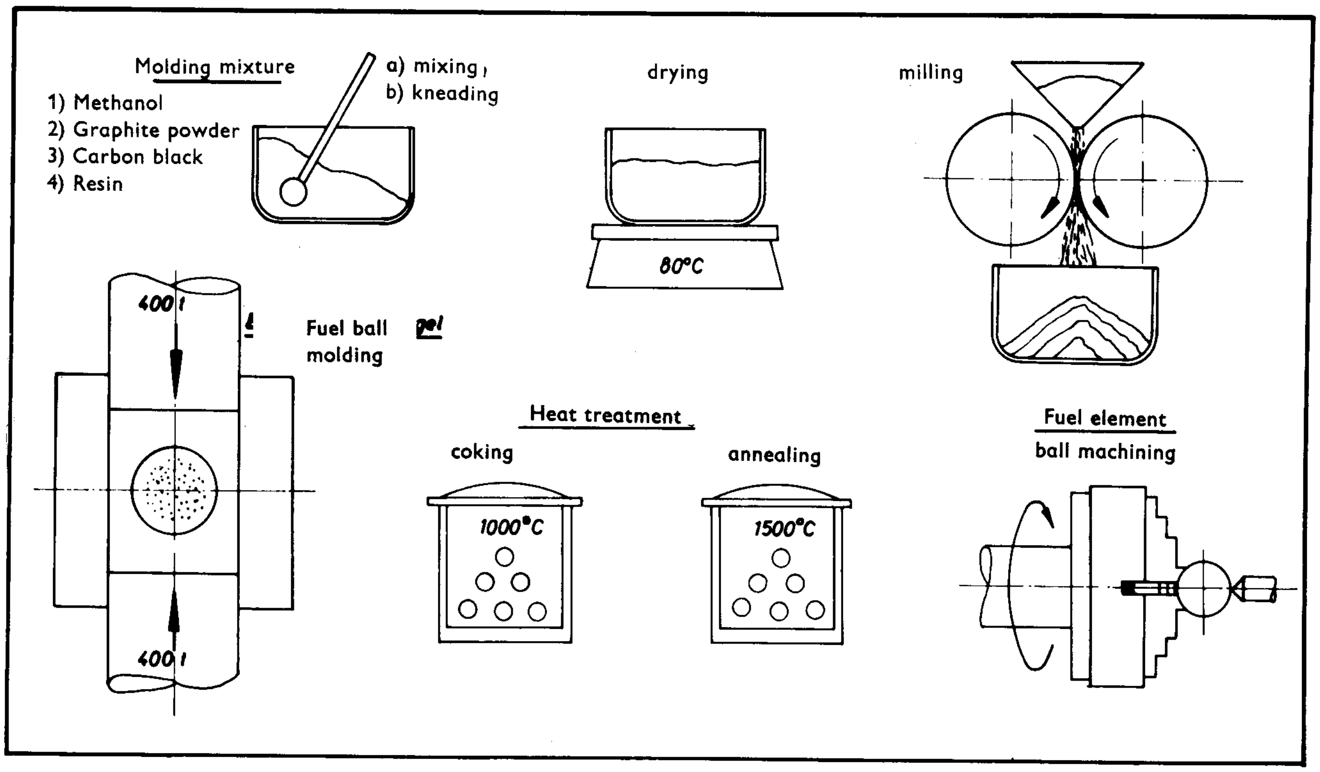


Fig. 8 — Preparation of molded fuel elements.

binder occurs at only 1000° C at the most. The drop impact strength of the spheres is raised from 50 drops to 1500-2000.

Owing to the good thermal conductivity properties of the spheres, it is possible that the power obtained per sphere could be increased to 6 kW if the fuel load were doubled. This output, which would be double that of the original design, would not only lead to a reduction in the fuel element cost in relation to the uranium quantity used, but might also enable the size of the core to be reduced for a higher power density. The various problems relating to this still have to be examined in detail.

In comparison with the two variants of the elements mentioned hitherto, the elements containing loose particles have an attraction in that fuel loading is considerably simplified. They therefore present an advantage, in particular, when the spent fuel is to be reprocessed within the breeder cycle, i.e., as soon as repeated reprocessing is included in the fuel element cost. The annulus elements shown in Stöcker's paper, which are fitted with an annular gap and a threaded plug, are easier to manufacture than other concepts. For spheres requiring lower drop impact strength properties, a smooth circular plug can be used for sealing the spheres. The normal annulus of 25 mm diameter, 25 mm high and 2 to 2.5 mm thick (4-5 cm^3 volume) is enough to hold the fuel particles for power output of up to 2 kW per ball if 6-9 kg heavy metal has to be put in each sphere. Fuel elements of this type were also tested successfully up to 20 % burn-up in the Risø reactor.

The Irradiation Programme

Mr. P. Jelinek-Fink and Mr. L. Valette

The irradiation tests carried out under the THTR programme cover two groups of materials :

1. graphite as structural material for the reactor and fuel elements
2. fuel in the form of coated particles and complete fuel elements

Before the beginning of the THTR work, studies had been carried out elsewhere in both fields. The irradiation tests on graphite started at low temperatures of up to 300°C for the first generation of gas-cooled reactors and were continued at higher temperatures of up to 600°C for the AGR. Fig. 1 shows the results achieved by the British with extruded graphite at temperatures between 385 and 480°C and fast flux doses of up to 1.6×10^{22} nvt. The figure shows that graphite shrinks under irradiation in both directions at the beginning, the shrinkage perpendicular to the direction of the extrusion being less than that parallel to the extrusion direction. Expansion perpendicular to the extrusion is noted again at a dose of about 10^{22} nvt. Moulded graphite behaves in a similar way.

At Hanford and Winfrith irradiations were made at higher temperatures of 800 to 1200°C. It was found that within this temperature range the dimensional changes at high fast neutron doses increase rapidly and depend very much on the temperatures. In addition, differential dimensional changes occur owing to the inhomogeneous temperature and flux distribution, which can set up stresses in larger graphite components. These stresses can lead to ruptures if the creep strength of the material is not sufficient to absorb them. So far only a few studies of radiation-induced stresses and their reduction by creep strain have been made in the temperature range concerned.

Irradiations of coated particles and fuel components have been carried out on a large scale by the Dragon Project, the Oak Ridge National Laboratory and under the AVR programme. These irradiations chiefly covered carbide fuel, and it is only during the last two years that oxide fuel material has been tested at Oak Ridge and by General Atomic. Pyrolytic graphite and silicon carbide were applied as coating materials, and up to three different coatings were deposited on the particles. The results of these irradiations are summarized briefly as follows :

- The release of fission gases can be kept very low by means of the coating. When particles are intact, the release of fission gases is almost exclusively due to surface contamination.

- Solid fission products (Cs, Ba, Sr) are released to a larger extent from the coated particles, but they are retained by the graphite shell of the fuel element.

- A burn-up of 15% fima is the threshold for differentiating the good particles from the bad ones. At this value particles can break owing to the spearhead formation of the kernel, by expansion of the fuel material and by outer corrosion.

- Coated particles with oxide kernels and carbon coatings showed excellent behaviour during the few irradiation experiments conducted. At high burn-ups of up to more than 20% fima the release rates were better than those for carbide fuels.

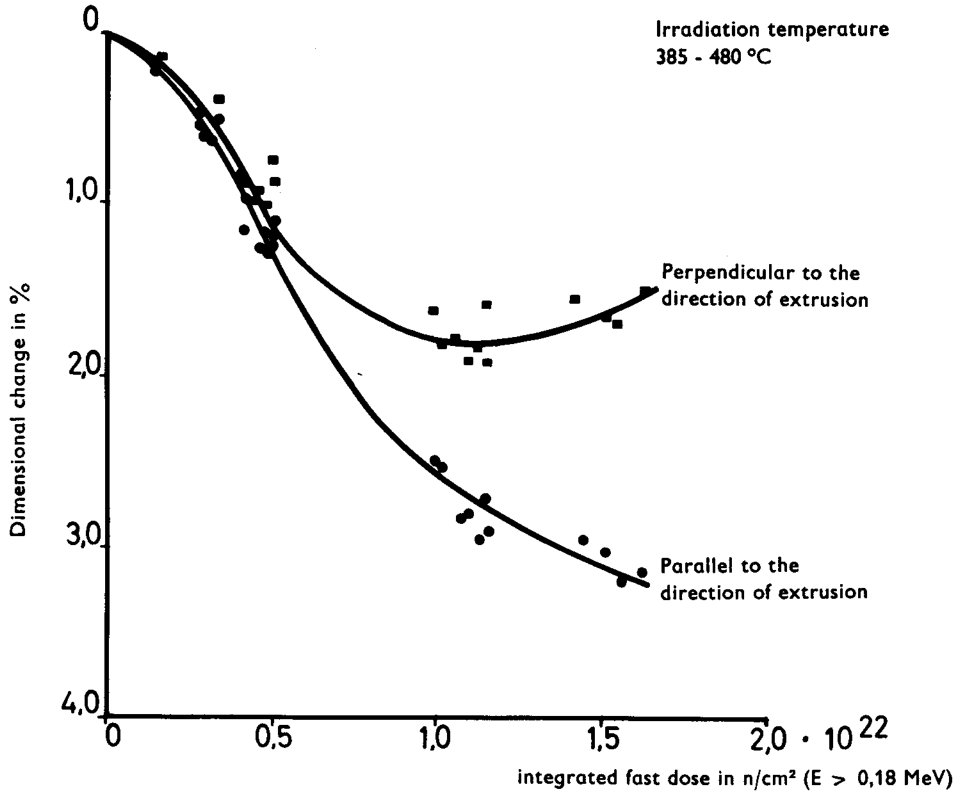


Fig. 1 — Dimensional changes of isotropie graphite.

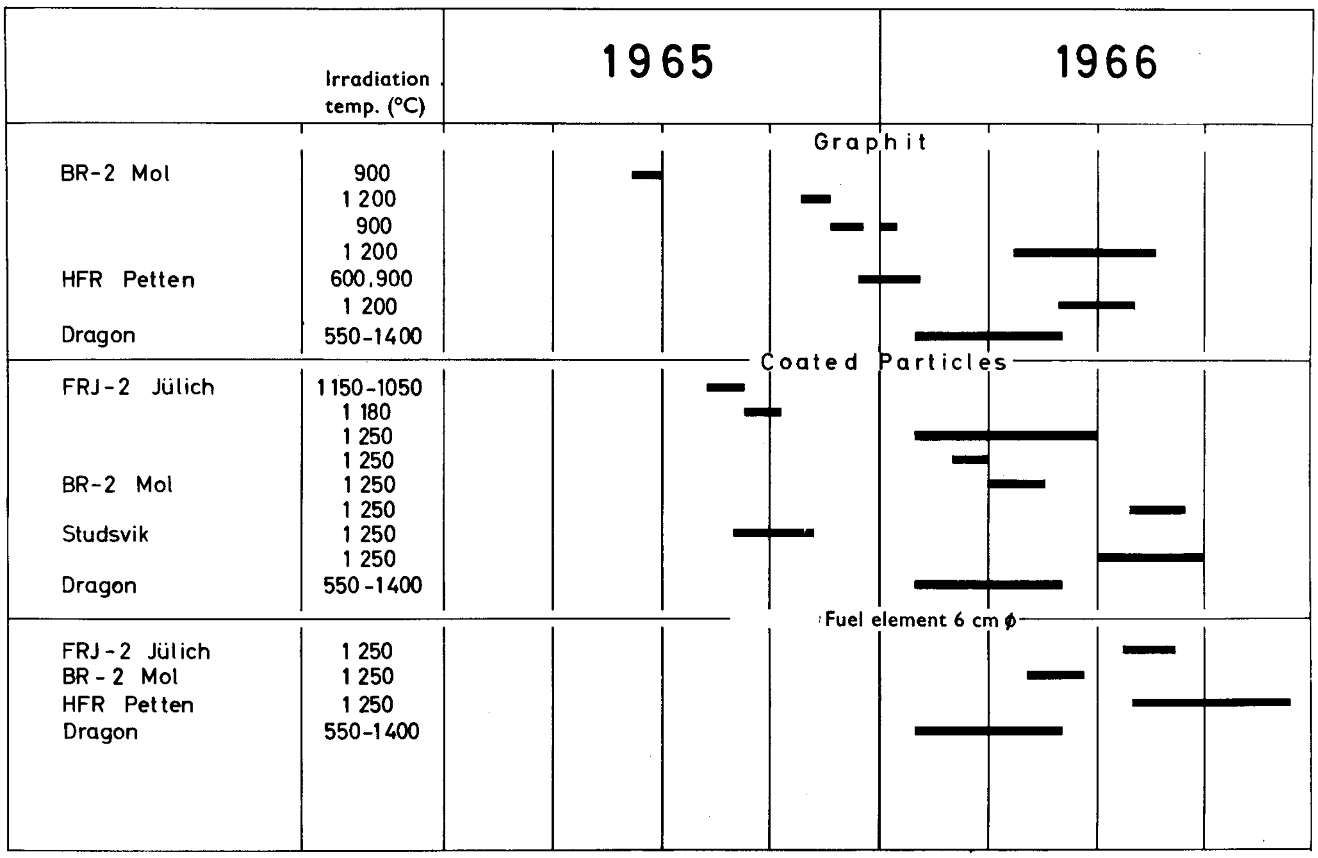


Fig. 2 — Short term irradiation programme.

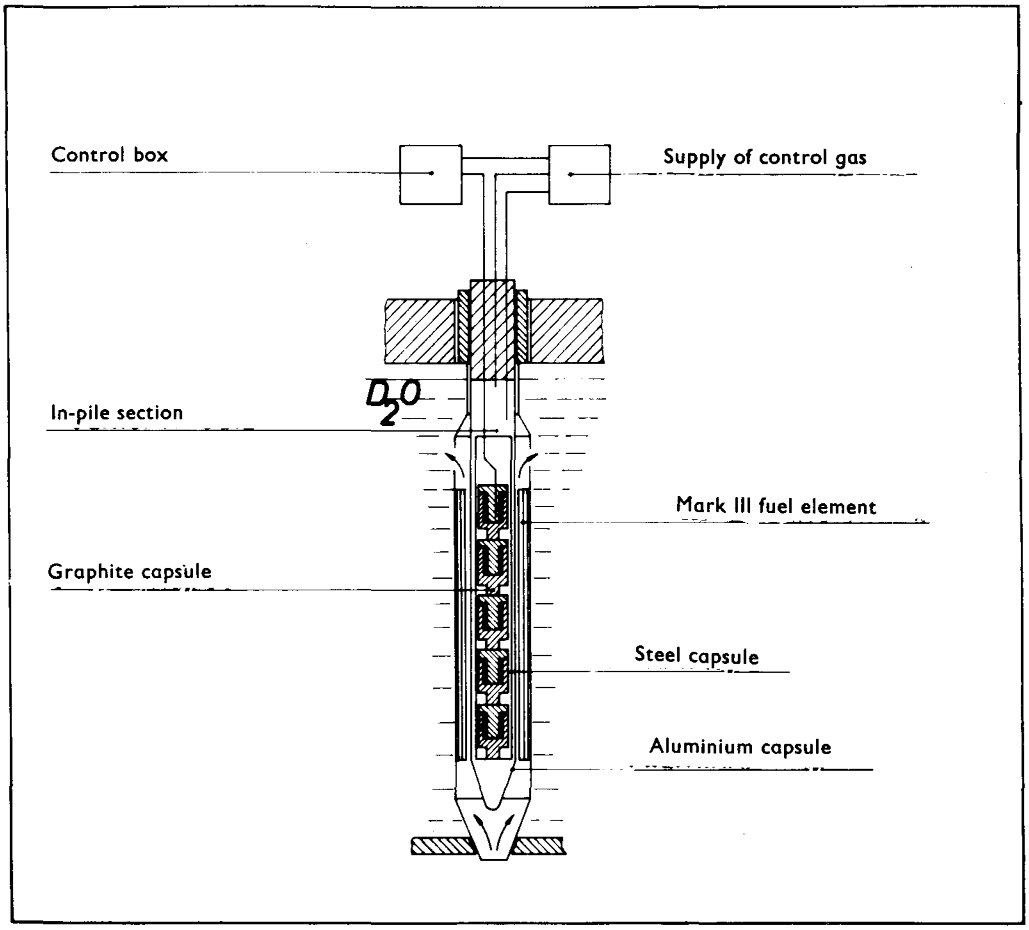


Fig. 3 — Irradiation capsule for coated particles.

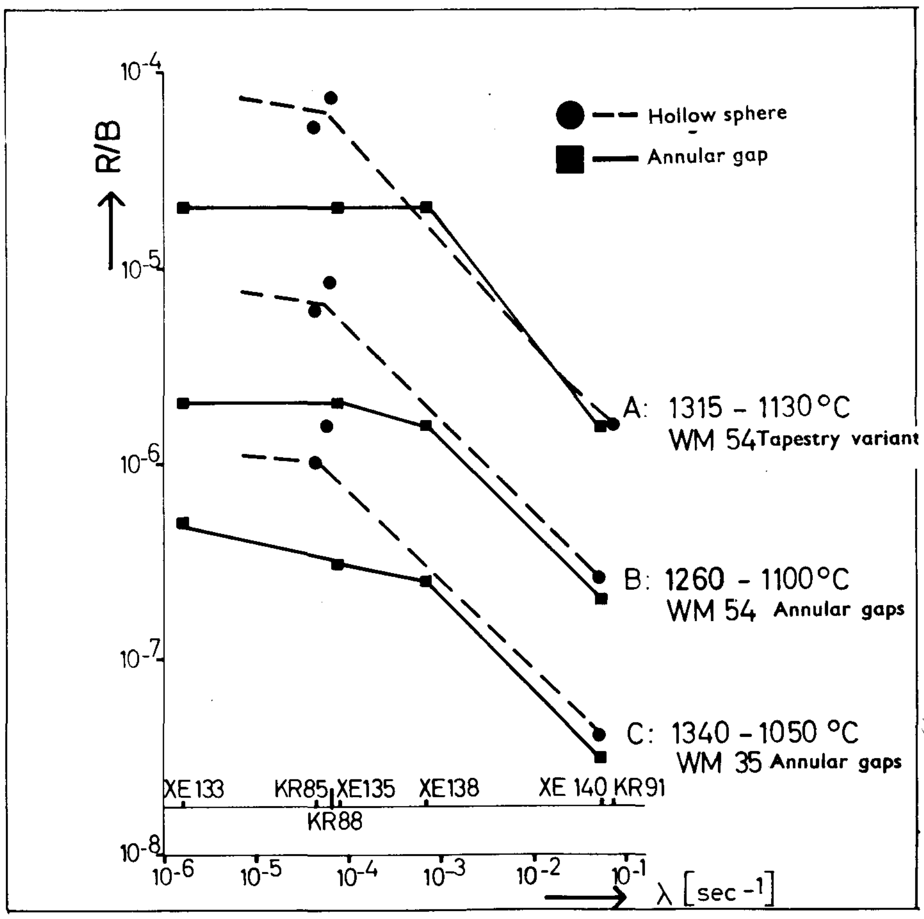


Fig. 4 — Fission gas release of coated particles under irradiation.

The THTR irradiation programme was based on this information. It has two different objectives :

1. To test fuel produced in the Euratom area within a short time for the make-up charge of the AVR reactor, and
2. To produce the necessary data for the development of fuel elements and the selection of structural graphite for the THTR.

Fig. 2 shows the short-range irradiation programme, which concerns mainly the testing of fuel elements ordered by THTR for the AVR reactor make-up charge.

A performance standard for the graphite shell of the AVR make-up fuel is not specifically laid down, but a model calculation shows that a maximum shrinkage of 0.8% may be permitted even without any allowance for creep strain. The fuel is in the reactor for about three years, which corresponds to a fast neutron dose of 2×10^{21} nvt.

Test reactors for graphite irradiations are BR2 at Mol, HFR at Petten and the Dragon reactor at Winfrith. Different European graphites and, as a reference material, Union Carbide graphite, which is being used for the first charge of the AVR reactor, are irradiated. As the figure shows, two graphite irradiations at temperatures of 1200 and 900°C have been completed at Mol. In both cases the fast doses reached were 4×10^{20} nvt. The third rig at Mol contains samples which have already been irradiated in the first rig. The temperature is again 1200°C and the fast dose will exceed 10^{21} nvt. Graphite irradiations are performed in collaboration with the Dragon Project at Petten. In order to investigate the creep behaviour, the present irradiated rig contains samples under tension and compression at temperatures of 600 and 900°C. The fast dose per reactor period is 1.5×10^{20} nvt. In addition to these irradiations in research reactors, large numbers of graphite samples are being irradiated in the Dragon reactor at temperatures of 600 to 1400°C.

The experiments conducted up to now have shown that the European graphites shrink less than the American reference material. Another interesting result observed is that the electric conductivity changes much more with the shrinkage than the thermal conductivity, and that therefore the change in the electrical conductivity is not representative of the change in the thermal conductivity, as was assumed hitherto.

For the coated particles the specification for the AVR fuel states that the steady-state release of Kr-88 should be less than 5×10^{-5} R/B after a burn-up of 9% heavy metal at temperatures of 1250°C.

In 1965 the following irradiation facilities were available for the testing of coated particles :

- the static capsules in the reactor FRJ-2 at Jülich,
- the swept loop at Studsvik.

As an example of an irradiation facility for coated particles, Fig. 3 shows a section through one of the rigs irradiated at Jülich. The rig, which is inserted in a mark III fuel element of the reactor FRJ-2, has an outer diameter of 2 inches, and the coated particles are housed in five separate graphite capsules. The temperature is controlled by different gas-mixtures in a cooling annulus.

The irradiations at Studsvik are carried out in a swept facility which was developed by Dragon. It consists of three loops running in parallel. The first irradiation at Studsvik was carried out in September/October this year. Fig. 4 shows the results of these experiments. Loose coated particles as well as particles which are embedded in a graphite matrix were irradiated. The burn-up obtained was 1.1 to 1.6% fima. The ordinate shows the R/B values, and the abscissa the decay constants. One can see that the particle-type WM 35 is behaving better than particle-type WM 54. Moreover, the diagram shows that the sample in which the particles are housed in a graphite matrix - the design of this sample being similar to the hollow-sphere-type element - has higher release rates. This can be attributed to damage of the particles during sample production. As a result it can be stated that at the low burn-ups achieved the release rates for loose coated particles are better than the specified 5×10^{-5} R/B for Kr-88 and that the particles in a graphite matrix show a lower release rate than the specified rate for fuel elements (the specification value for fuel elements is 5×10^{-4} R/B for Xe-133). Despite this the manufacturing process for the fuel elements of the hollow-sphere type should be improved. For the sake of completeness I should mention that both irradiations at Jülich also had satisfactory results.

For the next year, additional particle irradiations, partly as long-time tests, are planned at Studsvik, Jülich and in the Dragon reactor. Some of these will serve as specification tests.

For the irradiation of fuel elements the following facilities are available or under construction :

- an instrumental capsule for the BR2 at Mol
- fuel element replacement rigs for the FRJ-2
- the so-called five-ball experiment for the HFR at Petten
- the 25-ball experiment in the Dragon reactor.

At Jülich and Mol the influence of processing of coated particles in balls will be investigated, and the experiments at Petten and Mol are specification tests.

Fig. 5 shows the long-term irradiation programme until 1967. As Dr. Stöcker has already reported, the fast dose for the graphite shells of the fuel elements can reach 10^{22} nvt, and values up to 3×10^{22} nvt at temperatures of 700 to 1000°C are to be expected for some parts of the reflector. We think it necessary to confirm the behaviour of graphite under these conditions in spite of the very high cost of such experiments. Of course, such costly irradiations can only be performed with a few samples which have to be carefully selected in advance. The preselection is effected by the irradiations already operating at Mol and in the Dragon reactor.

The long-term irradiations of graphite are to be carried out mainly in two installations :

- in the rechargeable rig in the BR2, where an integrated fast dose of 0.3×10^{20} nvt/day can be obtained and
- by irradiation in the Fermi fast reactor, where fast doses of about 10^{20} nvt/day can be achieved.

The rig at Mol permits irradiations under controlled temperature conditions. Preliminary discussions have been held on the irradiation possibilities in the Fermi fast reactor, and it is planned to insert 64 samples towards the end of 1966. These irradiations will make it possible to obtain high fast doses at relatively low costs, but temperature measurements and controls are not possible. In addition, irradiations are planned at Petten and in the Dragon reactor.

The time schedule for irradiations shown in the table is only provisional. The dose values actually obtained will depend on the time and in particular on the funds available. After the completion of these irradiations we hope to be able to say which European graphite has the best combination of qualities. The change in such essential qualities as dimensional stability and heat conductivity, as well as the creep strain, will be known at that time, so that exact layout data will be available for the reactor designers.

For the irradiation of coated particles in 1966 and 1967, swept capsules will be available at Jülich and in BR2 at Mol. The objectives of the Jülich tests are to study the influence of production conditions and corrosive impurities on the coated particles. In the reactor BR2, the influence of fast neutrons on coated particles will be tested under different burn-up rates. Finally, irradiations in the Dragon reactor will make it possible to obtain statistical information on the behaviour of coated particles. Some of these experiments will give high burn-up rates of more than 20% fima at high temperatures from 1400 to 1800°C. The time schedule given in the figure for the experiments at Jülich is only an estimate. In practice, some of the swept irradiation capsules will stay in the reactor until failure of the particles.

During the second half of next year the irradiations of complete balls will start at Jülich and in the Dragon reactor. Irradiations of complete spheres are so expensive that high burn-ups cannot be achieved in research reactors. The above-mentioned swept fuel element replacement rigs at Jülich will again serve to test production variables and corrosion influence. The five-ball experiment at Petten is to be used, in particular, to investigate the damage caused by fast neutrons. Extensive statistical irradiations will be carried out in the Dragon and the AVR reactor. To sum up, it can be said that since the Summer of 1964 the short-term irradiation programme has made such good progress that we hope to complete the programme by the end of next year. The test schedule for the AVR reactor make-up charge can be retained. As far as the long-term development programme is concerned, only preparatory experiments have been carried out. The preparations have, however, progressed to such an extent that we will be able to start the experiments on a large scale during the coming year.

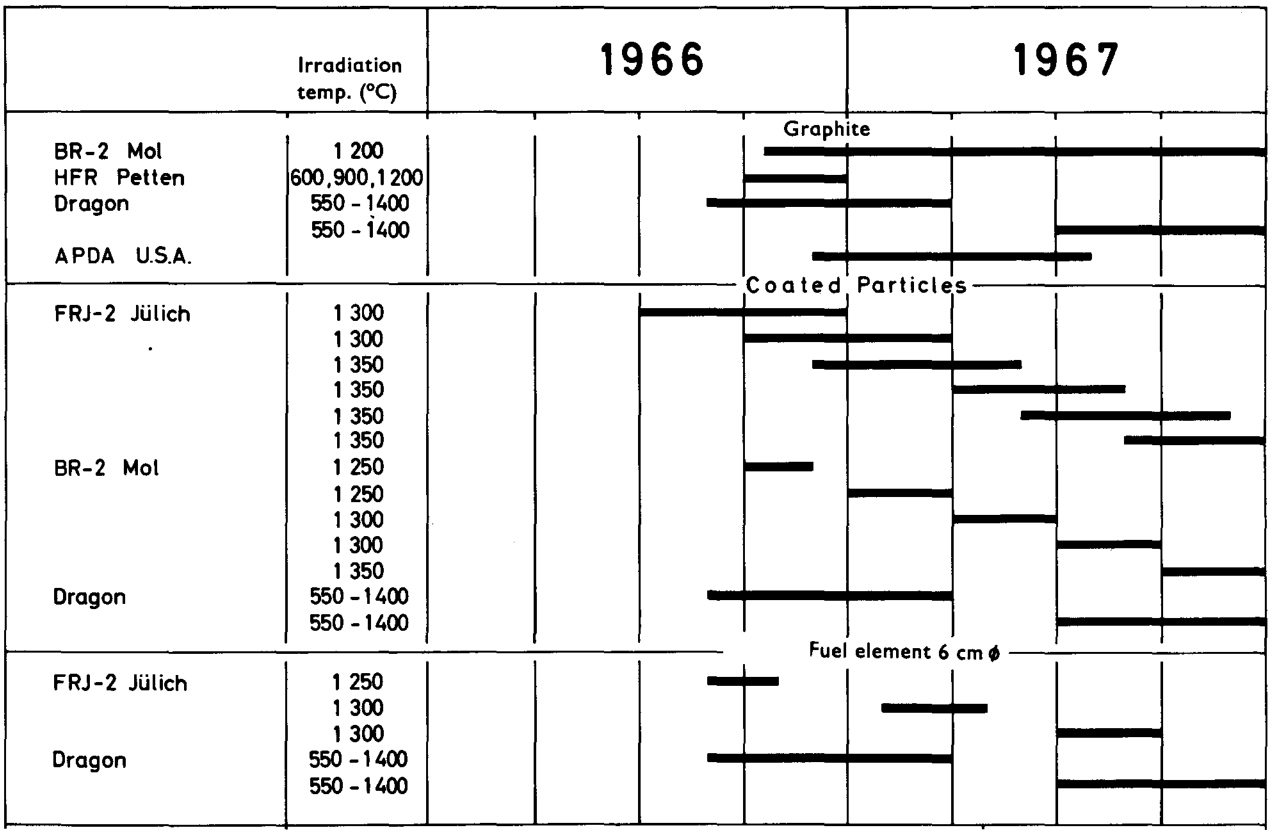


Fig. 5 — Long term irradiation programme.

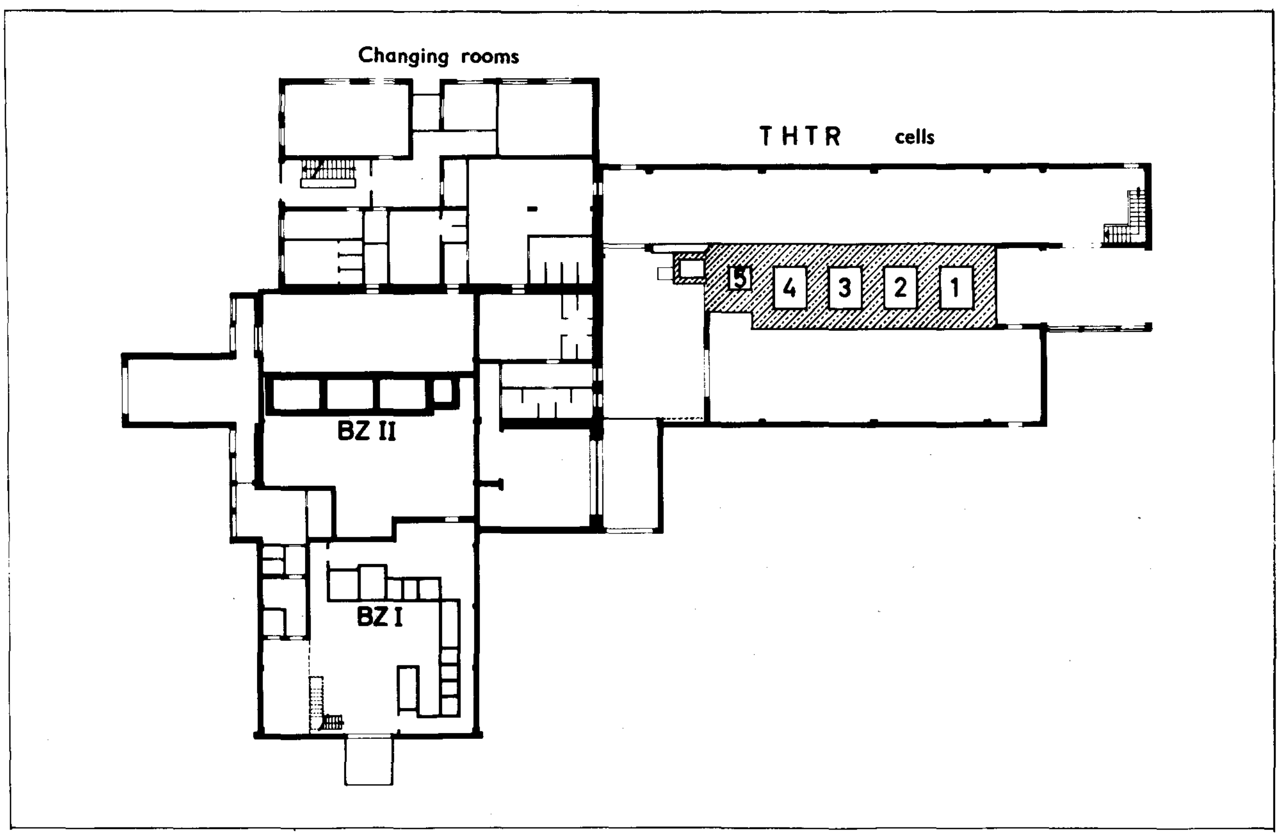


Fig. 6 — Outlay of KFA fuel cells.

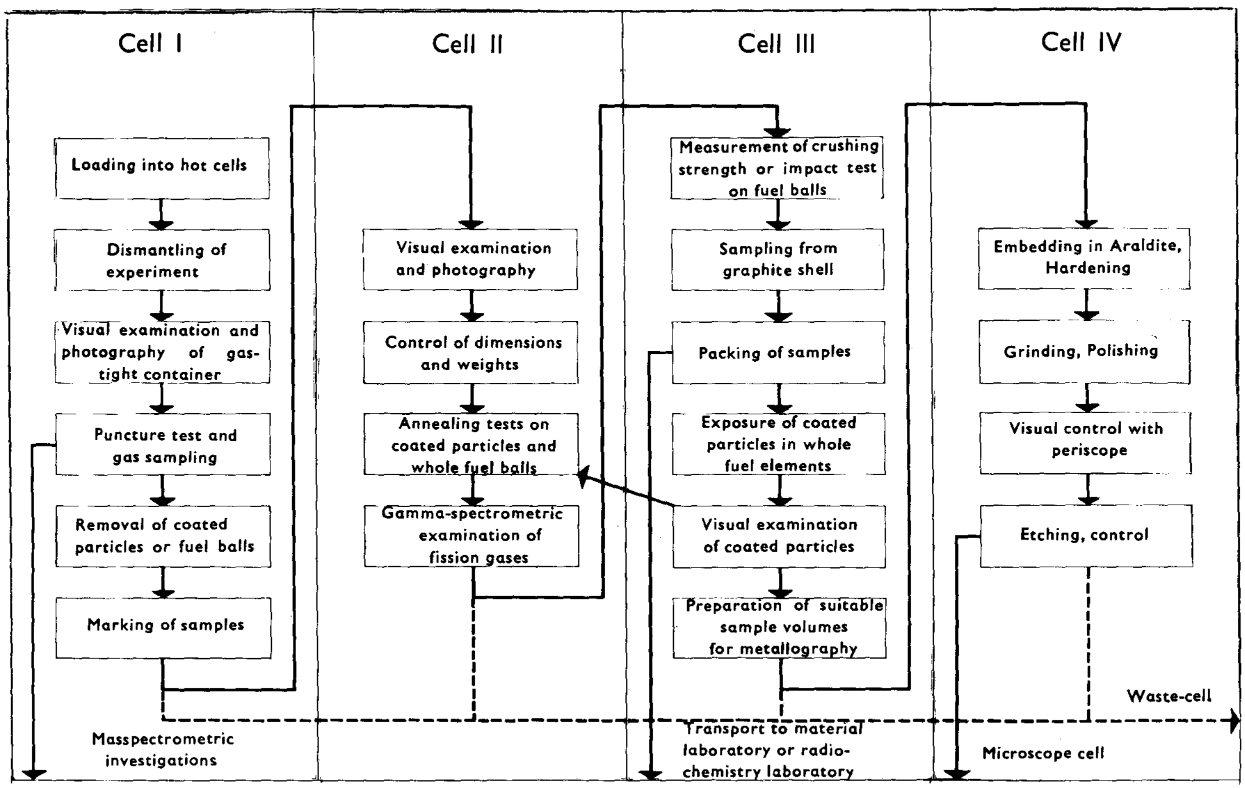


Fig. 7 — Operations in THTR cells.

Such a comprehensive irradiation programme performed under the THTR Project requires large facilities for the post-irradiation examinations, because irradiated fuel samples and fuel spheres will be produced in large numbers. It has become obvious that the capacity of the hot cells and radiochemical laboratories available is one of the factors governing the size of the working programme and that it is desirable to perform the post-irradiation examinations and their evaluation in one place in order to ensure comparable results. As a sufficient number of suitable hot cells were not available in time, it was decided to build a specially adapted hot-cell line. Only by this means can full use be made of the irradiation facilities of Dragon and AVR. In order to keep the costs low, the THTR cells were connected by a common changing room with another laboratory already under construction. A great number of other cost-reducing steps have also been taken. Fig. 6 shows the THTR cells and the existing cells BZ I and BZ II.

The THTR laboratory itself consists of four working cells, a waste cell and a microscope cell. The concrete block enclosing the working cells and the waste cell is laid out in such a way that in cells 1 and 2 a total of 50 fuel element balls can be handled at the same time and five in cells 3-5. All cells are equipped with gas-tight boxes and are connected by a transport tunnel. In the first unit, the dismantling cell, different irradiation devices or transport containers are dismantled and the irradiated material prepared for examination in the other cells. The second cell is used for non-destructive material tests. Here mainly optical measurements, photography, weighing, dimensional controls and annealing tests of the balls will be performed. The next cell, cell 3, is used for removing samples from the ball shell or the fuel material insert, and in the fourth cell metallographic samples will be prepared. The line of concrete cells is completed by the

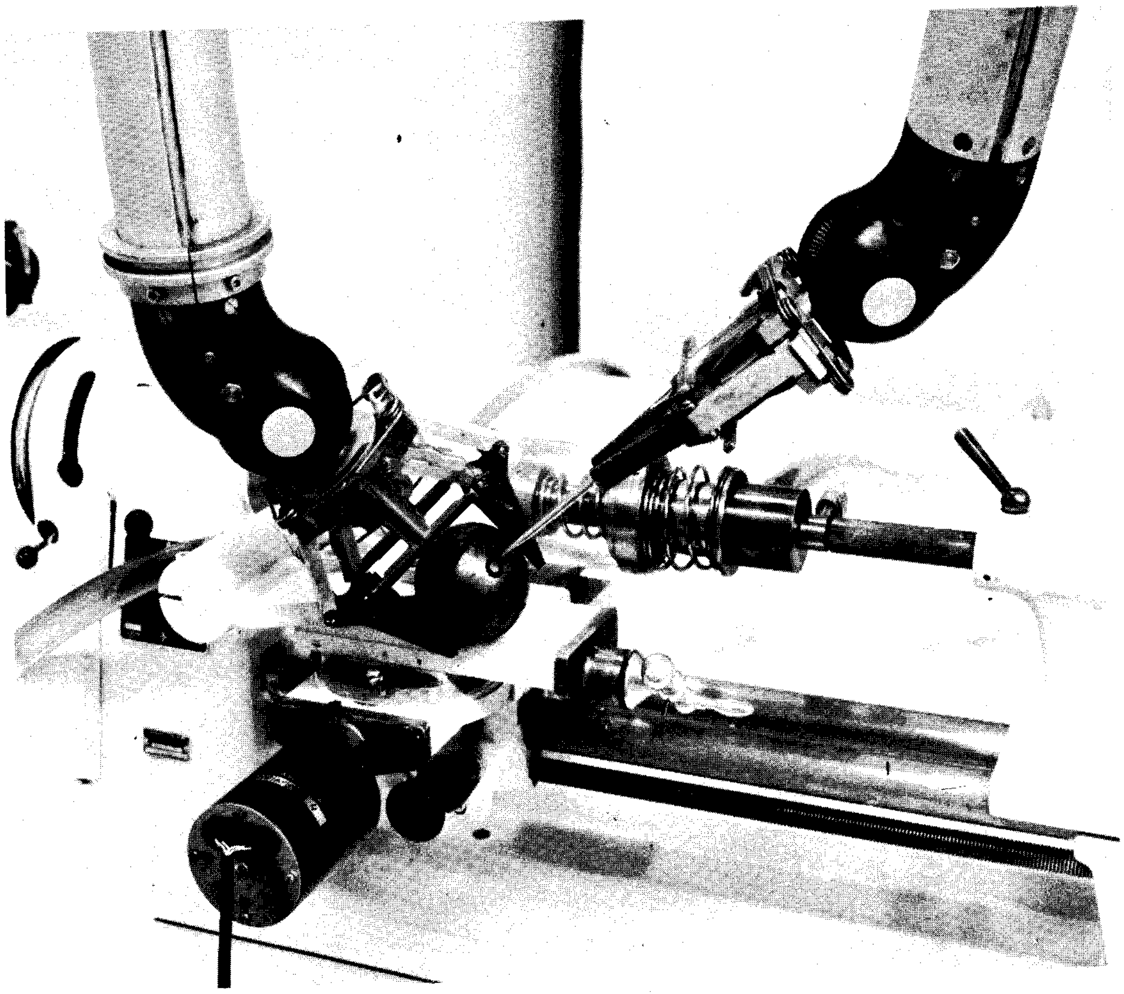


Fig. 8 — Lathe device.

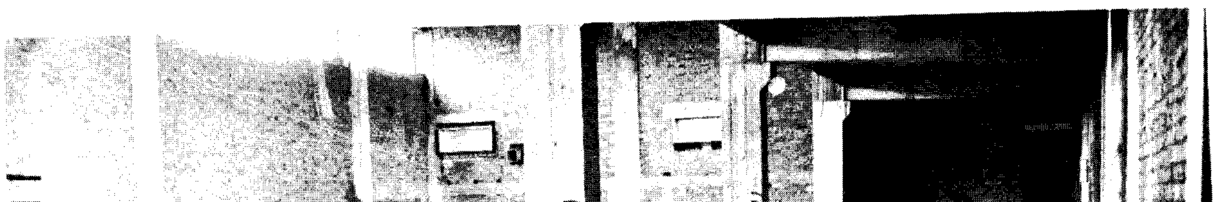


Fig. 9 — THTR Cells.

November 1965

waste-cell. A number of devices for use within the cells had to be developed. The lathe device for cell n° 3 is shown as an example (Fig. 8). This facility is used for producing V- shaped slots on the surface of the balls to facilitate breaking and also permits the trepanning of cylindrical samples. The slide shows the removal of such a sample. The last figure (9) shows the present state of construction of the laboratory.

The progress made indicates that the laboratory will start operation in September 1966. This date corresponds to a building period of about 15 months, whilst the planning period prior to the beginning of construction was about one year.

Neutron Physics Investigations

Dr. A. Schatz

In the work on the physical behaviour of THTR, three problems were mainly considered :

- the fuel cycles
- the normal operational behaviour of the reactor
- the nuclear safety.

These studies, which in the first place dealt with the planned prototype, having an electric power of 300 MW, were always performed with a view to raising the output to 1000 MWe.

The fuel cycle is characterised by the use of highly-enriched uranium, U_{235} or U_{233} as fuel, and thorium as the fertile material. The uranium/plutonium cycle, which is concurrent with the thorium/uranium cycle, is of secondary importance only, because of the low U_{238} concentration.

The efficiency of the thorium/uranium process is greatly reduced by the absorption of neutrons in the fission products, and by the protactinium loss. Both effects are very much dependent on the thorium/uranium ratio and on the fuel inventory in the fresh fuel elements. Therefore these two parameters are chosen in such a way that best burn-up is achieved.

The first aim of the present burn-up calculations was to find a core composition which would be suitable for the loading of the THTR prototype, so a load with only one fuel element type was taken. In these conditions a parameter study was carried out for steady-state loading operations; in other words the core is in equilibrium. The start-up and run-down periods were not taken into account.

The burn-up calculations which were carried out with a Dragon Project code are based on the core data given in the first table:

Power density	6 MW/m^3
Average moderator temperature	625°C
Average fuel temperature	675°C
Neutron losses	5.8%
Enrichment	93%

The total uranium and thorium feed in a fresh fuel element was varied.

The first figure gives the conversion ratio and fifa (the number of fissions per U_{235} atom) as a function of the moderation ratio and the thorium/uranium ratio of a new fuel ball. Since the fuel elements are not reprocessed, only the conversion during the period when the element is in the reactor is of interest. This conversion is, however, determined by the average protactinium concentration. For this reason, in a fuel element cycle without reprocessing, it is not possible, with only one type of fuel element, to raise the conversion ratio above a certain value, that is to say, about 0.7.

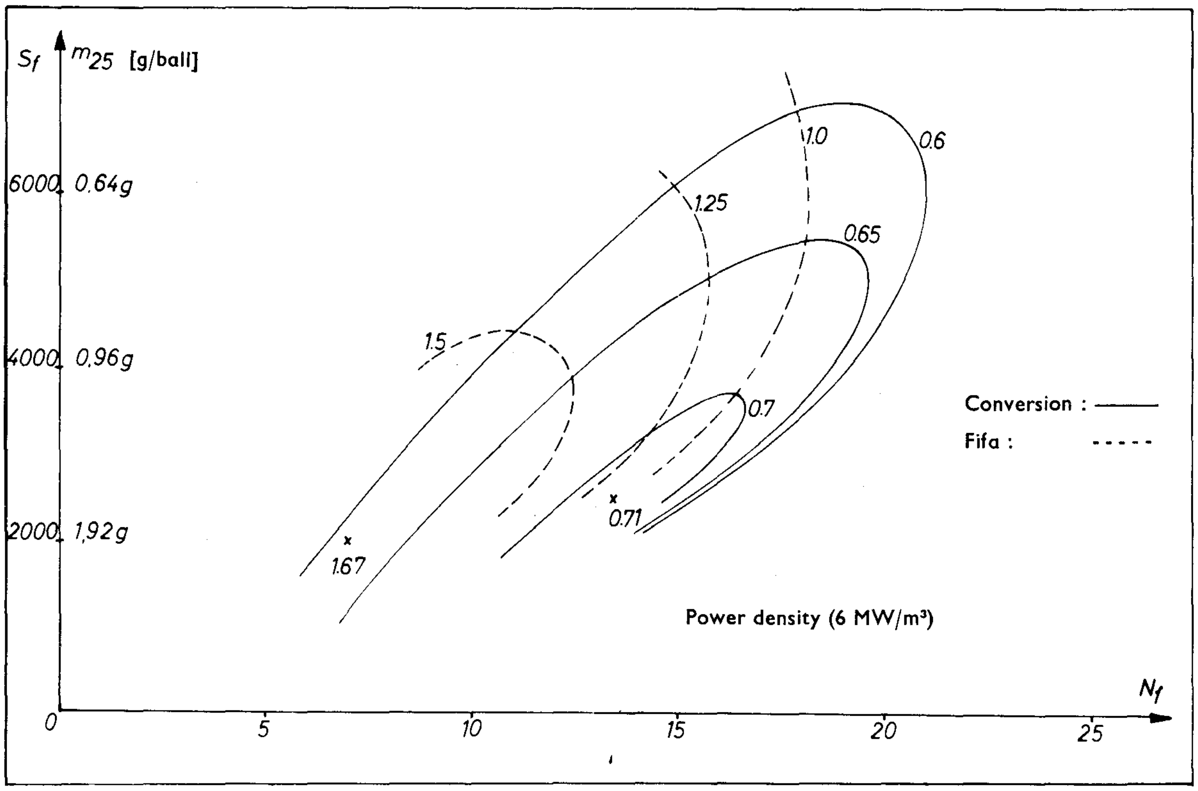


Fig. 1 — Conversion factor and fifa "without reprocessing".

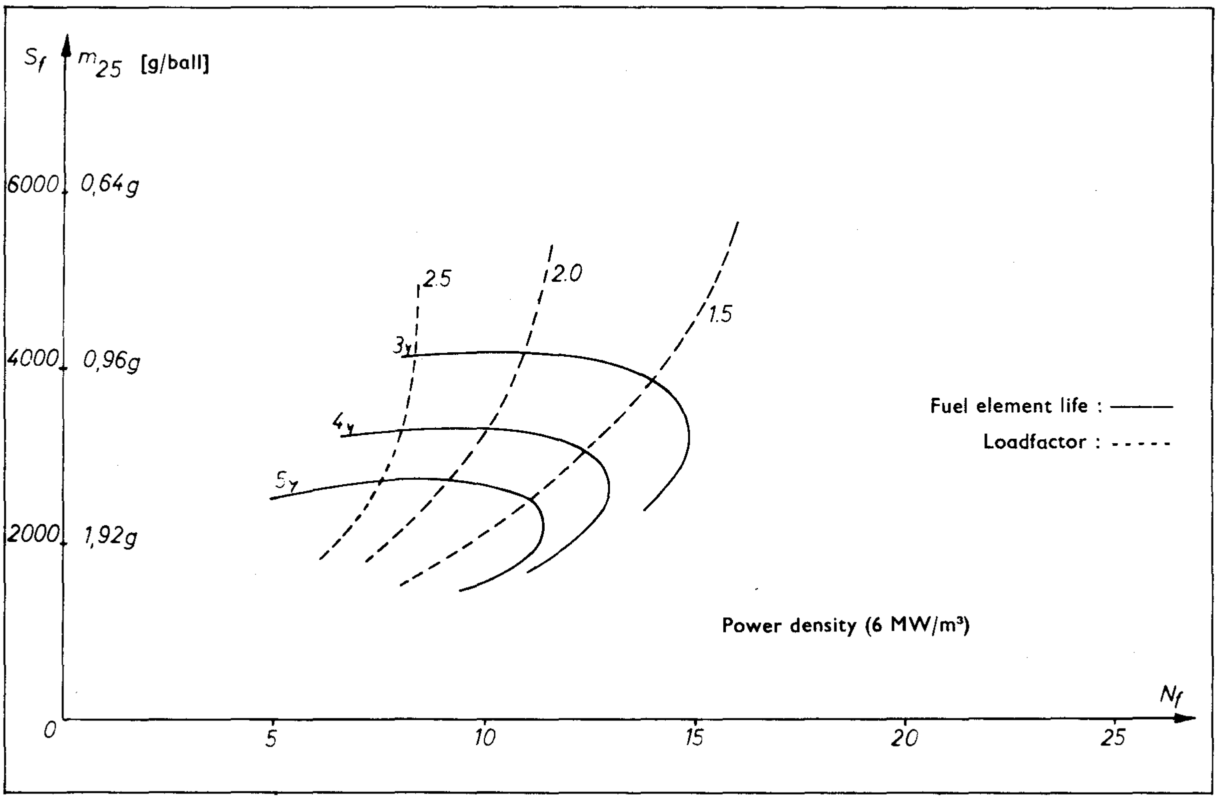


Fig. 2 — Load factor and fuel element lifetime "without reprocessing".

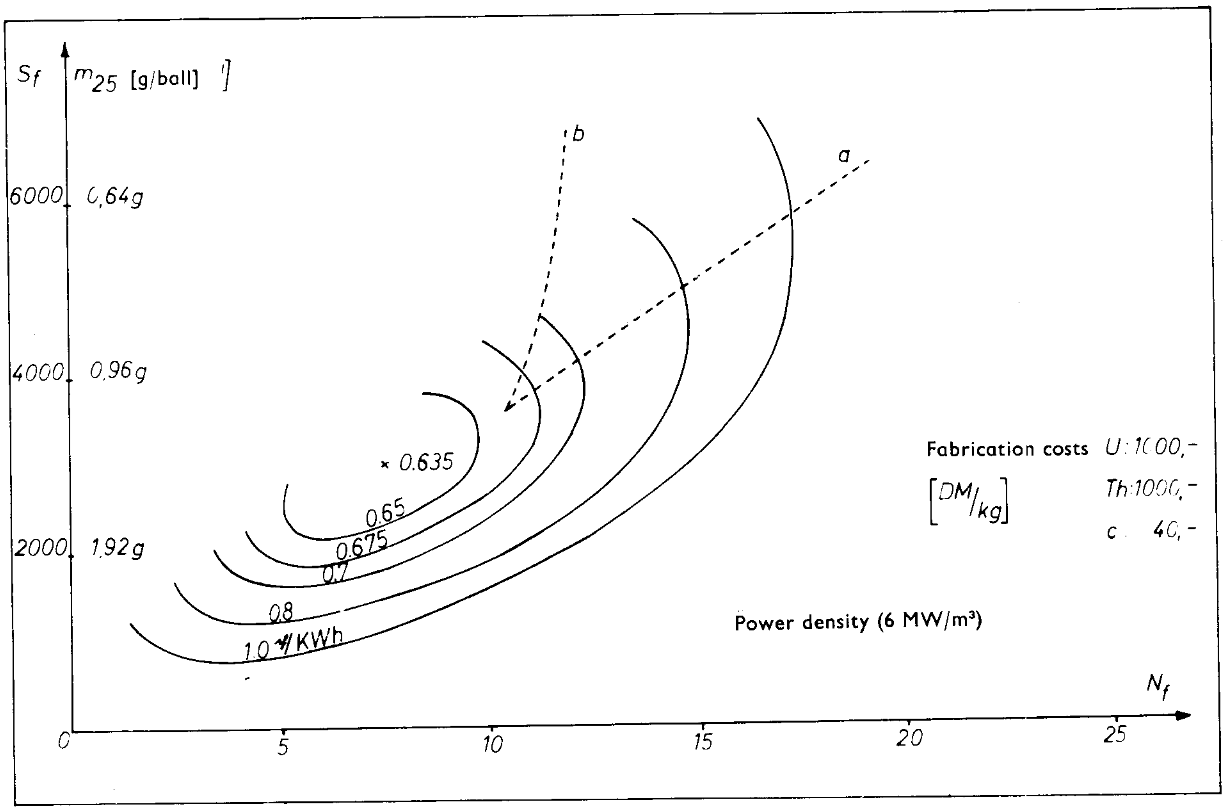


Fig. 3 — Fuel element cost "without reprocessing".

Fig. 2 shows the life-time of the fuel elements and the power factor gamma. Gamma is the ratio between the maximum power of a new ball and the average power per ball. For an expected life-time of 3 to 5 years, the power factor lies between 1.5 and 2.5.

To fix the composition of the standard load of the prototype, we have to fulfil three main conditions, which are given by the choice of the element type, here the so-called "Tapetenvariante" (Tapestry type).

1. The heavy metal used should not total more than 12 gms per ball.
2. The thermal power of a ball should be limited to a maximum of 3.4 kW.
3. The neutron dose for neutrons above 0.18 MeV should not exceed 0.5 to 1 x 10^{20} nvt.

Fig. 3 shows the cost as a function of the same parameters. The first conditions of which I have just spoken, i.e., the limitation of the heavy metal inventory and the maximum thermal power per ball, are shown by curves a and b. The permissible area for this fuel element is between curves a and b. At an average ball performance of 1.1 kW and a radial and axial power shape factor of 1.5, a maximum power factor of gamma = 2 is permissible. This is shown by curve b on the slide. The third condition, i.e., the limitation of the maximum fast dose, on which depends the life-time of the fuel element, may be fulfilled in this case, where the life is about 3.5 to 7 years.

Under theseconditions, the lowest fuel costs are about 0.662 Dpf. per kWh for an average moderator ratio of 8000 and a thorium/uranium ratio of 21.5. These figures for the 300 MW prototype give a feasible standard case which was ensured by physics studies. The second type load which is now under consideration shows even better results with regard to fuel costs.

During the start-up phase of the reactor, special conditions for the core charge are obtained, because only fresh fuel elements are available and the reactor has not been poisoned by long-lived fission products. At a full core load, the reactor would then have a very high excess reactivity. If, however, only enough fuel elements are introduced in the core to ensure that the reactor is just critical at full power and the rest of the core is filled with graphite balls, then the maximum power of the fuel element goes far beyond the permissible amount of 3.4 kW. Therefore, there are two possibilities for the start-up period. In the first place, the number of fuel elements in the core is so high that the reactor reaches its full power. Excess reactivity is compensated by boron balls, or taking the same number of fuel elements in the core, the fission material investment in the fuel element is reduced to such an extent that no surplus reactivity is created. After a certain running period the fuel elements with full fuel inventory will be loaded. Which of the two methods is chosen, or whether the two possibilities are combined, will depend on the final fuel element fabrication costs.

The burn-up of the heavy metal used can be improved if the fuel and the fertile material are separated and located in separate balls. Then the breed elements can stay in the reactor longer since they are not poisoned by fission products as quickly as the fuel elements. The thorium used is then better exploited and the fuel costs can be further reduced.

In a reactor of about 1000 MWe there is a further improvement because the neutron leakage can be reduced to 2-2.5% and therefore more neutrons are available for breeding.

In the breed elements and fuel elements are separated, we expect a reduction in fuel costs of up to 15% as compared with the one-type case. A further reduction in cost is possible if many nuclear power stations of this type are under operation, because mass production is, of course, always accompanied by a marked decrease in fabrication and reprocessing costs. In such a case, i.e., with an installed power of about 15,000 MWe, Carlsmith in Oak Ridge has worked out the cost for the fuel cycle of a pebble bed reactor as being about 0.35 Dpf. per kWh.

The temperature expression in the reactor calculation is rendered more difficult because there is no defined channel for the cooling gas. The gas flow in the reactor is not only in the actual direction but there is a radial flow as well. Therefore, loss of pressure and heat transfer in the core can be well described with the expressions of Sonntag and Denton. With these relations a numerical calculation process could be worked out which permits the calculation of mass flow rate and temperature distribution in the core. A corresponding programme calculates the power distribution in the core, the mass flow rate, the temperature distribution in the cooling gas and the surface temperature in the fuel elements. Input data are porosity, gas pressure, core inlet and outlet temperature, also the dimension of the core and fuel elements. At the same time the programme gives the temperature of the fuel elements and especially the very important value for the fuel particle temperature.

The neutron physics and thermodynamic behaviour of the reactor under normal operation was examined on the basis of a standard case. The standard case is that core composition in which one-type load without any reprocessing gives an optimum burn-up. The data of the standard element are shown in the next table (table 2).

TABLE 2

	Dimension	Value
Ball radius	cm	3.0
External radius for the "wall paper" layer	cm	2.0
"Wall paper" layer thickness	cm	0.25
Average graphite density	g cm^{-3}	1.65
U^{235} loading	g	1.04
U^{232} loading	g	0.07
Th^{232} loading	g	10.08
Maximum permissible power per ball	kW	3.4

TABLE 3

		Dimension	Value
Thermal power		MW	750
Volume		cm^3	125
Power density		MW/cm^3	6
Loading factor		-	0.61
Number of balls		-	$6{,}71.10^5$
Coolant gas :	helium		
	pressure	atm.abs.	40
	inlet temp.	°C	250
	outlet temp.	°C	850
	average temp.	°C	550

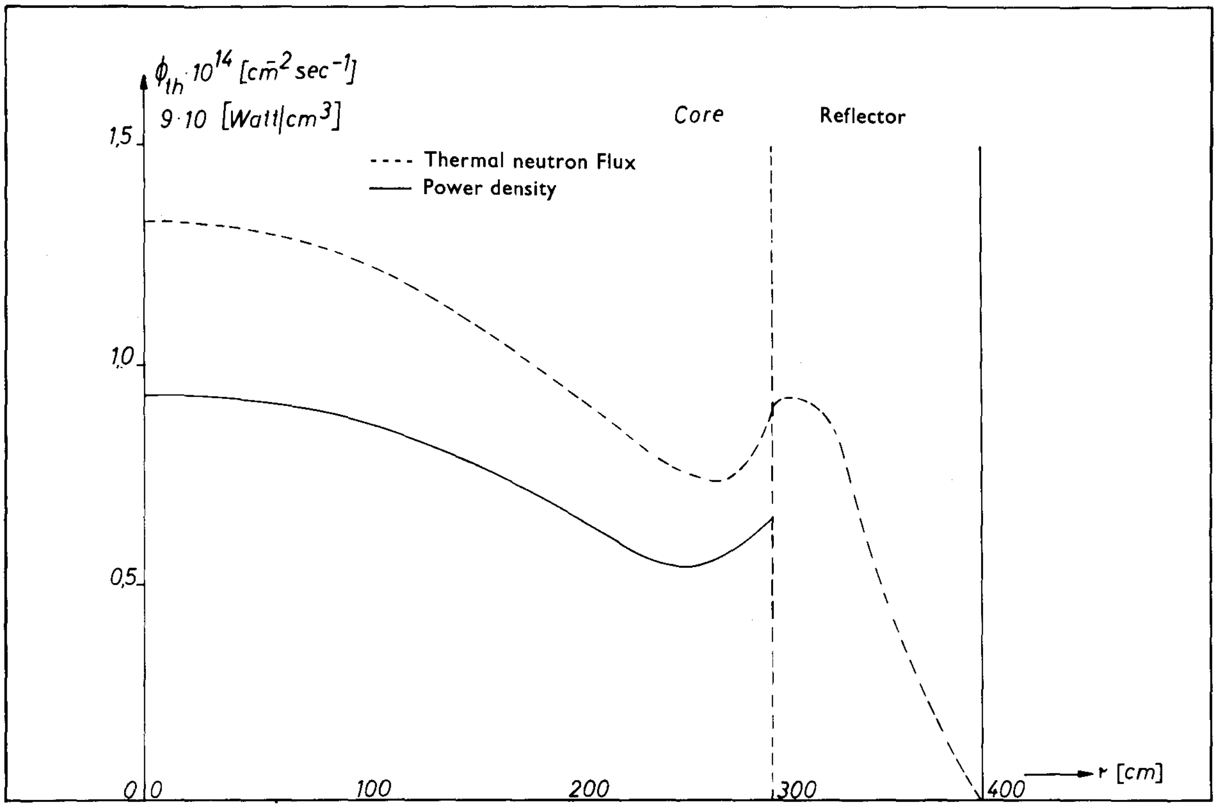


Fig. 4 — Thermal neutron flux and power density in the radial direction (core middle).

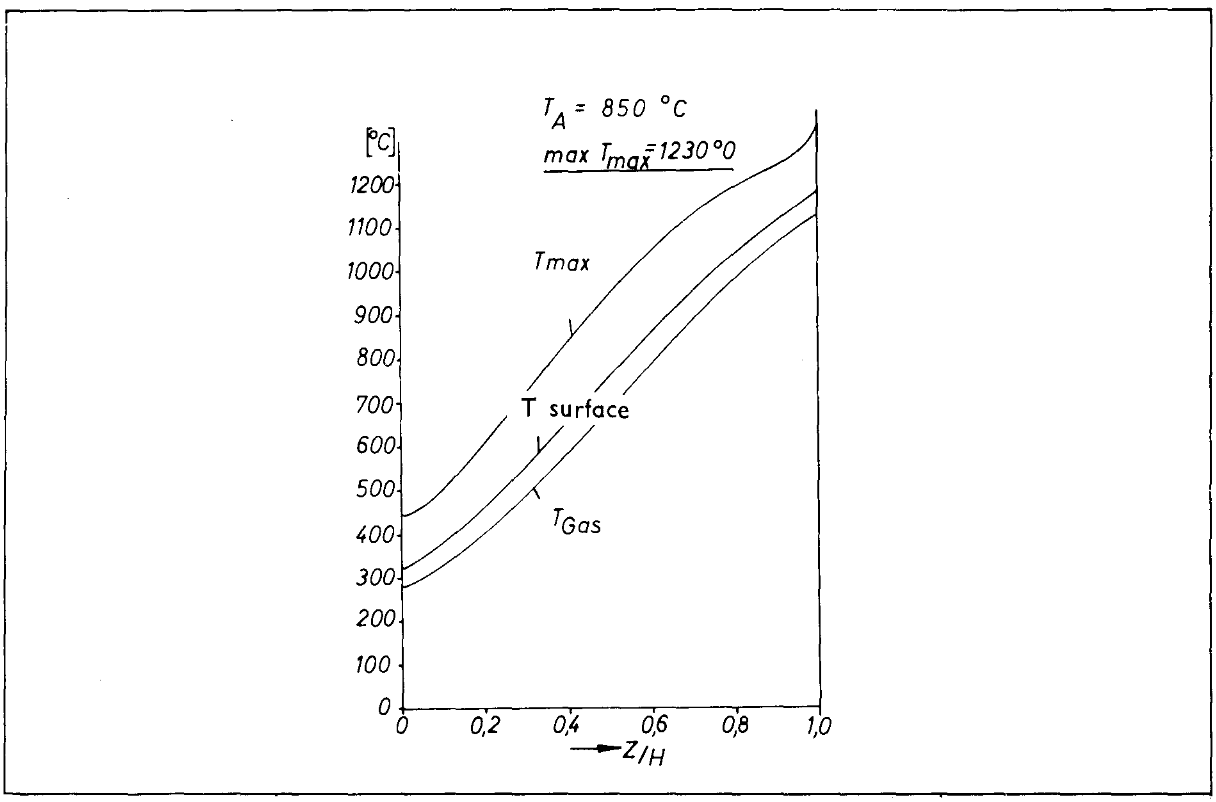


Fig. 5 — Axial temperature distribution.

Table 3 shows the most important data of the reactor core. At a power density of 6 MW per m^3 and a core volume of 125 m^3, the thermal power is 750 MW. In the core there is a total of 674,000 balls. The core itself is surrounded by a graphite reflector both radially and axially.

Table 4 shows the critical data for the reactor. The average ball power is 1.1 kW, the maximum power of an element is 1.72 kW, the maximum power of a fresh element is 3.4 kW.

In the hot critical condition, the reactor does not have any excess reactivity because the reactivity degression by burn-up is offset by the continuous addition of fresh balls and the withdrawal of spent fuel elements. The total fissionable material in the core is 322 kg, the annual consumption of uranium-235 is about 200 kg.

Fig. 4 shows the radial distribution of the thermal neutron flux and the power density halfway up the core, in the area of maximum flux.

The last figure (Fig. 5) shows the temperature distribution along the core axis. The lower curve shows the coolant gas temperature, the middle one the surface temperature and the top curve shows the fuel temperature. Throughout the core the maximum permissible temperature of the coated particles of 1300°C is not exceeded.

The third point now being dealt with is nuclear safety. Nuclear safety of the reactor is primarily ensured by the fact that in all operational conditions of the reactor there is always a negative temperature coefficient. The temperature coefficient of the moderator is $-2.9 \times 10^{-5}\,{}^{\circ}C^{-1}$ and the prompt temperature coefficient of the fuel and fertile material is $-1.7 \times 10^{-5}\,{}^{\circ}C^{-1}$. A reactivity increase in the case of accidents is likely only if water gets into the primary system.

TABLE 4

	Dimension	Value
Average power per ball	kW	1.11
Max. power per ball (average burn-up)	kW	1.72
Mean moderator temperature	°C	627
Mean fuel temperature	°C	677
Moderation ratio	-	8000
Thorium ratio	-	21,5
Buckling	cm^{-2}	5.10^{-5}
Total loading U^{235}	kg	163
Total loading U^{233}	kg	159
Total loading Th^{232}	kg	6826
Fifa	-	1.57
Conversion ratio	-	0.62
Burn-up	MWd/t	$1,4.10^{6}$
U^{235} consumption per year	kg/a	199

Assuming that if a steam generator tube bursts the content of about three tons of water evaporates, there would be an increase in reactivity of 0.3%. This reactivity increase corresponds to an increase in the average core temperature of 65°. This does not create any problems for the control of the reactor, and even if the control system failed, the reactor would not be in danger.

Status of the Development of the 300 MW Prototype

Mr. H.W. Müller

1. Characteristics

The development of gas-cooled, graphite-moderated systems has so far been pursued in Great Britain and France and to a certain extent also in the United States of America. As a result power plants with a total capacity of more than 7000 MW have been built and put into operation. Obvious limits set to the efficiency of this first generation of CO_2-cooled and graphite-moderated reactors led to the development of a second generation, the so-called advanced gas-cooled reactors (AGR), which in Great Britain will be used for future power station units. This AGR, in comparison to the Magnox reactors, is more compact as far as its structure is concerned, and its steam conditions are more up-to-date. The development of the AGR has its limits, however, from a technological point of view, and consequently the logical development of this system will be the next generation in these series, the high-temperature reactor. The construction of high-temperature reactors pursued by the Association is determined by the following criteria:

- a spherical ceramic fuel element, loosely embedded;
- a continuous circulation of the elements;
- a gas-cooled and graphite-moderated reactor core.

2. AVR - First Stage of Development

The first stage of the development carried out by BBK - the development of this reactor system - consisted in the construction of the AVR power plant. This test power plant, built in the immediate vicinity of the KFA, serves the following two essential purposes:

- to verify the functional efficiency and safety of the high-temperature reactor with mobile fuel elements in an integrated primary system;
- to gather design and operational experiences for the future development of this reactor type.

The erection of the test power plant is essentially completed. The cold operational tests are being performed at present. Some of the fuel elements have already been shipped and will probably be charged within the next few months. We do not wish to minimize the difficulties faced during the construction or to diminish our willingness to indulge in self-criticism, but it may be stated that an effective construction period of four and a half years is not too long, taking into consideration the complete novelty and originality of this reactor.

The first figure shows the AVR test power station. I take it as read that the details are sufficiently well known and shall not therefore bother to describe them.

The development of the first pebble-bed reactor, which started more than six years ago, provides the main knowhow for the second stage of the development series, the THTR. This basic knowledge is independent of the size and specifications of the reactor. It is directly transferable and comprises the following:

1) Core-physics questions, e.g., heat transfer and gas flow behaviour of the pebble bed, the flow behaviour of the

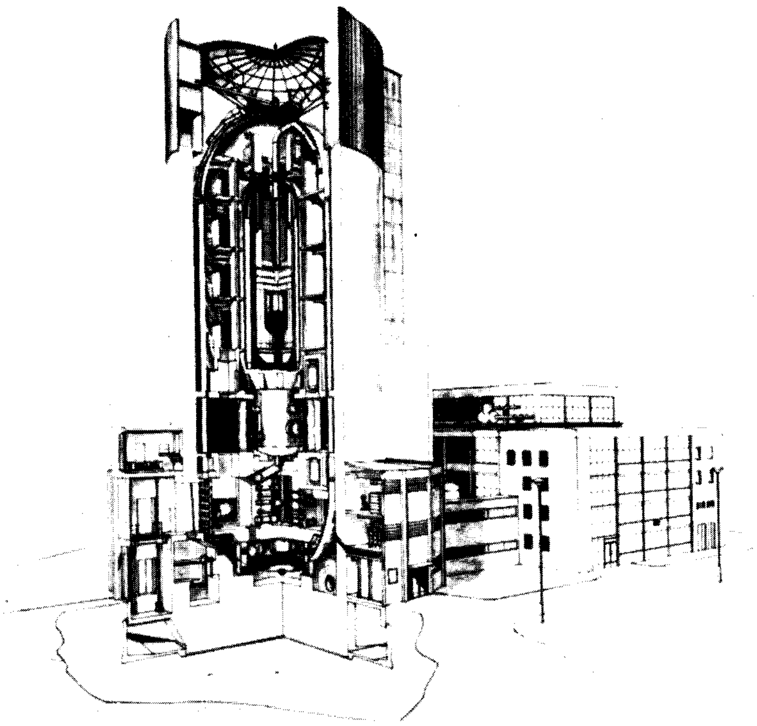


Fig. 1 — The AVR power station.

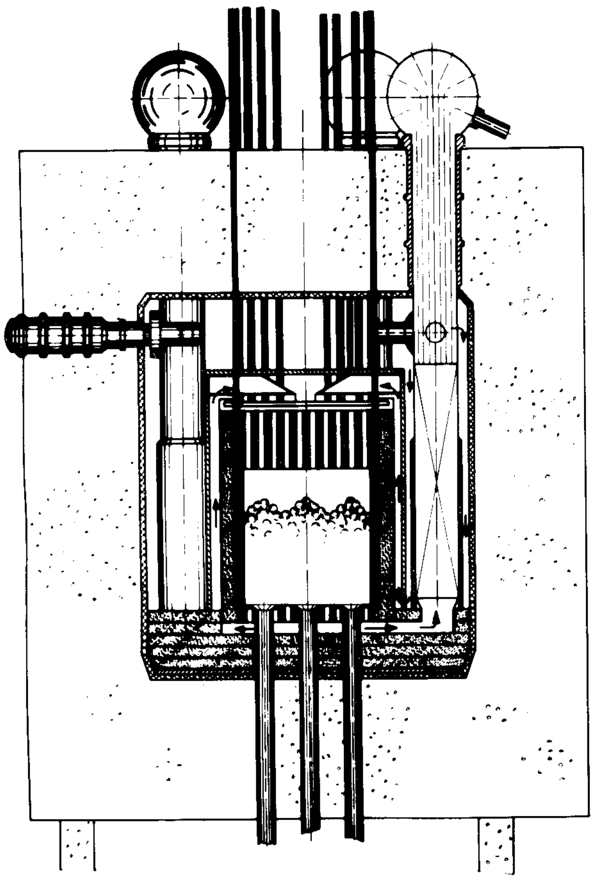


Fig. 2 — Reactor arrangement.

pebbles and the fuel charge.

2) Chemistry of the core, such as corrosion and mass transfer problems, problems of cooling gas purification, questions of friction and lubrication of the parts of the plant exposed to primary gas.

3) Helium leak tightness requirements, including shaft sealings as well as flange and valve packings, cable ducts and the leak tightness of the steam generator weld seams.

4) Development of specific components such as blowers, control rods and tools for the removal of parts.

If we have decided on making basic changes to the THTR design, despite the directly transferable AVR experience, this is because the progress made in the development of ceramic fuel materials makes it possible to dispense with a large part of the safety requirements which were necessary at that time, and also because there are certain limits to the production of huge steel vessels for reactors exceeding 300 MW, which are not applicable to prestressed concrete vessels.

3. THTR Conception

When we started on the design of the THTR prototype, we were faced with quite a number of contradictory requirements.

1) As a genuine prototype the plant should be as simple and as cheap as possible; on the other hand, it should serve as a test station and, therefore, supply as much wide-scale information as possible.

2) One should be able to extrapolate the plant, on the one hand, to outputs of up to 1000 MW, while the installed power of the prototype, on the other hand, should be kept as low as possible for financial reasons.

3) A high conversion ratio is desirable. The current market prices of uranium might, however, make it necessary to shift the fuel cycle somewhat in the direction of maximum burn-up, say towards lower conversion, i.e., towards the "throw-away cycle".

4) The requirements of easy accessibility to new-type components (e.g., steam generators) hamper optimum utilization of the space and volume of the vessel, and consequently a cheaper type of construction.

Compromises cannot be avoided when the requirements are so contradictory and our decisions with regard to the construction should be considered from this viewpoint.

For the prestressed vessel, a cubic cylinder was chosen with an inside width of 16 m and a wall thickness of five meters, enclosing the entire primary circuit. The integrated type of construction - originally used for the AVR - was thus retained, as it offers considerable advantages as to safety and compactness. The design of a vessel of prestressed concrete is determined above all by the required penetrations and the arrangement of the tendons. Temperature limitation in the concrete is achieved by means of a special cooling system which has to be adapted to the insulation of the vessel. We have to attach particular importance to the leak tightness of the vessel. We try to achieve a gas tightness that does not exceed a daily gas loss of 0.1 % of the total contents (this means that we will lose one complete gas filling in three years). The penetrations in the vessel are to be kept to a minimum in consideration of the tendon troughs and the gas tightness.

The steam generator is subdivided into individual units which are laterally arranged. We are of the opinion that in future high-temperature reactor generations unremovable steam generators can be used. At the present state of technology, however, we do not possess sufficient experience to make definite statements concerning the damage to steam generator pipes. We have therefore decided to use removable steam generators for the THTR, although certain design and financial disadvantages are involved. Contrary to the AVR - a cooling gas flow from top to bottom has been chosen for the THTR primary circuit. The downward flow makes it possible to choose a core height/diameter ratio which is independent of physical or thermodynamic considerations. There is no levitation of the uppermost pebble layer which would limit the mass flow, i.e., the power density is merely limited by the blower power in contrast to the upward flow. With regard to the design, the downward flow results in a major simplification of the top reflector and facilitates the arrangement of the control rods. On the other hand, the bottom core plate is exposed to the temperature of hot gases. This problem is solved by using graphite or carbon stone instead of steel for the bottom reflectors and supporting plates.

The requirements of gas tightness are such that the blower used is fitted with a canned motor. By choosing a fast-running engine the rotor is light and of low dimensions, so that the penetrations are also small. Particular importance is attached to the use of blowers with gas bearings, which have passed their tests in the DRAGON project. For the THTR reactor, however, we would have to use 2000 kW blowers with gas bearings. Blowers of this size with gas bearings, however, have not been tested as yet and must therefore be investigated as to their reliability in operation at partial loads and possible failures.

Contrary to the fuel-charging plant in the AVR reactor, in the THTR reactor the spherical fuel elements will most probably be removed from the core through several extraction pipes. In consideration of the cost involved, this extraction system should be conducted to a quick-registering burn-up measuring plant from where it is led to numerous distributors. As we have to consider a rating of

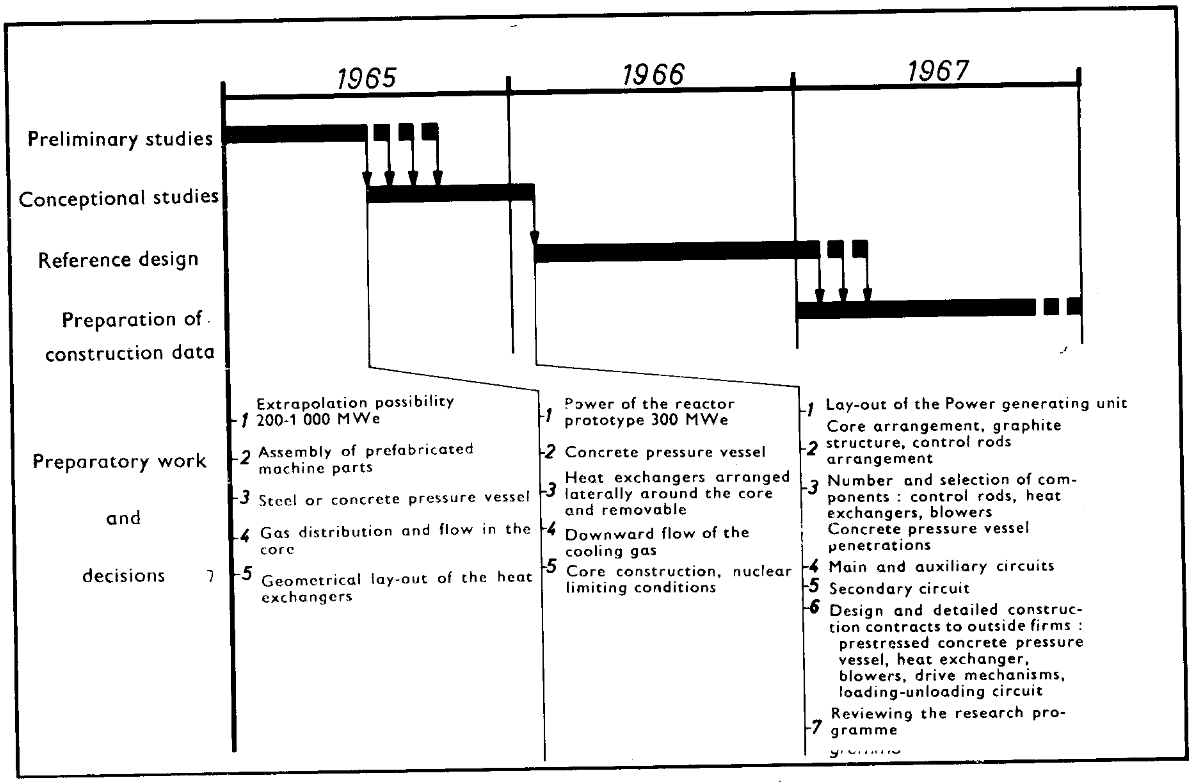


Fig. 3 — THTR planning of the decisions.

300 pebbles per hour under steady-state conditions the control of the charging schedule must be done by means of an electronic programme controller.

In order to be able to operate the reactor with partial loads also, control rods must be provided in the core besides the shut-down rods. The guiding of the absorber rods in zones of high temperatures and high neutron fluxes presents particularly tricky problems. The selection of materials for the absorber rods and the appropriate guides is therefore of utmost importance. Here, however, there is a possibility of dispensing with the rod guides by the introduction of free absorber rods directly into the pebble bed. Experimental tests revealed that owing to the liquid nature of loose pebbles the required power is so small that no damage to the fuel pebbles is to be expected. Before taking final decisions in connection with this especially attractive method for the pebble bed reactor, further tests are to be made to see how the mechanical load for absorber rods and fuel elements can be reduced to a minimum by overlapping static and dynamic forces and simultaneous circulation of the pebbles.

4. Planning Programme

In conclusion I would like to give you some details of our planning programme. Having taken up our theoretical and experimental development work in 1963 and 1964, we started with the layout of the THTR power plant at the end of 1964. In July 1965 the following decisions were taken:

1) The prototype power plant was to have an output of 300 MW;

2) The vessel is to be made of prestressed concrete;

3) The steam generators are to be arranged laterally and are to be removable;

4) Downward flow of the cooling gas;

5) Determination of the core dimensions.

Various test series necessary for the individual parts of the installation were undertaken, and construction and development contracts were drafted and/or awarded. By the second half of this year, i.e. 1965, the data of the primary cycle and the steam conditions had been determined on the basis of investigations of the parameters. Safety considerations with a view to non-steady-state operational conditions were made simultaneously. In February 1966 the details of the reactor components will be known to the extent that the detail design on the basis of the reference design can be started. In 1966 and 1967 the design will be worked up in all details so as to be ready for construction.

Our following lectures deal with the individual problems of the design and the techniques applied.

Main Characteristics of the 300 MWe Plant

Mr. H. Schafstall

The THTR power plant consists of a two-circuit system; the reactor cooling gas and the turbine working medium flow in two separate circuits. The thermal energy of the primary part is transmitted to the secondary part by steam generators. The change of a variable in one of the circuits affects the other data so that these two main circuits must be considered as one unit.

The thermodynamic state variables determine to a great extent the dimensions, design, material and coordination of the various components. Therefore, their influence must be taken into consideration and evaluated carefully before the design data are frozen.

The design work carried out this year should combine the basic concept and the data in such a way as to achieve optimum economics for large plants. The scope of the work was therefore determined by the demand for extrapolation up to 1000 MW.

The demand for optimum economics required cost studies for a wide range of design data. In the concept stage of a new reactor type, this is possible to a certain extent only. Some outstanding features and considerations were therefore taken as a basis in order to obtain a realistic design of the nuclear power plant in the light of the recognizable influences and conceivable development risk.

All variants were investigated under the same conditions. The standard calculation used the following data :

depreciation time	17 years
utilization factor	0.6 - 0.8
annuity factor	13.2%
maximum net power	300 MWe
fuel cost	0.66 Dpfg/KWh

The basic concept of the reactor was determined by the arrangement of the removable steam generator on the side of the core in a prestressed concrete vessel and by the gas flow in the core from top to bottom.

A study of the following conditions was necessary to determine the design data : the suitability of intermediate superheat, live steam condition and gas condition, i.e., core inlet and outlet temperatures and the primary gas pressure. The best feed water preheating temperature and the steam generator temperature gradient were taken into account as well.

These variables affect the efficiency and the plant costs. Especially critical variables that have an influence on the design data are the thermal stresses in the steam generator, carbon corrosion in the core and radiation damage to the graphite structures.

Furthermore, the requirement that the risk be reduced and the reliability of operation be increased limited the possibilities of variation. The demand for a removable steam generator requires, for instance, a compact steam generator design since the openings in the prestressed concrete vessel must not exceed certain dimensions. The gas and steam temperatures are limited by the thermal stresses in the pipe walls on account of the good heat transfer of helium under high pressure.

The high-temperature reactor offers the advantage that the secondary circuit can be designed without difficulty as the circuit of a modern steam power station.

Owing to the varying cost structure, however, double intermediate superheat is excluded from the beginning. There are thus only two variants to be studied : a plant with single intermediate superheat and a plant without intermediate superheat. The plant without intermediate superheat has seemed advantageous up to now, since it would possibly reduce the plant costs and presents the simpler solution with regard to design.

The list below shows, however, that our decision in favour of intermediate superheat was well justified. The plant with intermediate superheat and supercritical pressure (225 atm.abs., 525/525°C) is in line with the data which are customary in power plants at present. The optimum intermediate pressure is about 65 atm.abs. in a preheating range of up to 220°C.

For safety reasons we have at present also chosen a steam pressure above the primary gas pressure for operation at partial load. The pressure before the medium pressure turbine (inlet steam after intermediate heating) is maintained almost constant by throttle valves.

A plant with intermediate superheat is more costly. The costs are increased for safety reasons since shutoff and trimming valves, collectors and water injection and their control are required in addition.

The plant with intermediate superheat offers the following advantages and disadvantages as compared with a plant without intermediate superheat :

Advantages

1) Higher efficiency ~3.8 points

The intermediate superheat increases the efficiency of the plant by about 3.8 points to 40%.

2) Lower fuel cost ~11 million

The cost for fuel is decreased by about 11 million thanks to the increased efficiency over the aforementioned depreciation period and the interest.

3) Lower blower capacity ~15%

The blower capacity is reduced by 15%, which is primarily due to the lower gas throughput.

4) Lower steam quantity ~22% and condensate quantity ~20%

The quantity of steam is reduced by about 22% and the volume flow by more than half.

The sum of the cross-sections of the pipes conducting steam and water is increased by 15% but the number of steam pipes through the prestressed concrete vessel is reduced by about 40%. This is due to the design and type of suspension of the steam generator. The reduction in the condensate quantity means a reduction in the costs for the recooling system and in captive consumption requirements, such as the cooling water pump, cooling tower ventilator and the water-jet induction pump.

5) Lower thermal core power

6) Smaller first fuel charge ~ 9%

7) Lower capacity of the charging system

At a constant core power density the improvement of the efficiency as under items 5, 6 and 7 accounts for a reduction of costs of 9% in each case.

Disadvantages

1) Larger heating surface ~ 33%

The intermediate superheater increases the heating surface of the steam generator by about 33%. The plant without intermediate superheat (110 atm.abs., 525°C) is somewhat less complicated. The costs are reduced as far as the steam generator is concerned.

2) Higher tensions in the pipe walls of the steam generator

The highest thermal stress in the pipe wall is at the end of the superheater since this is located in an area with high gas temperature. The higher steam pressure requires thicker pipe walls and thus the thermal stresses increase. However, at the same time the permissible stresses are increased.

3) Higher feed pump capacity ~ 66%

The feed pump capacity is increased to 66% by the higher steam pressure and the relative increase in the pressure loss on the water side in spite of decreasing feed-water quantity.

4) Larger expansion line of turbine ~ 22%

The larger expansion line of the turbine requires a greater number of stages. However, the lower steam quantity reduces the exhaust steam cross-section and thus the number of exhaust steam flows. The turboset becomes more expensive.

It is to be expected that, as a whole, the cost reduction for a 600 and 1200 MW plant with intermediate superheat will be greater than without intermediate superheat. For the prototype we have therefore chosen intermediate superheat.

Once this decision is taken, the live steam pressure can be chosen within wide ranges. The permissible humidity of 12% at the turbine outlet is reached only at 260 atm.abs. (p_k = 0.05 atm.abs.).

The water or the steam must flow in the steam generator from the top to the bottom since the primary cooling gas is passed from bottom to top. Thus the stability of the flow on the water side must be given particular attention. This problem decreases in importance with using live steam pressure. A supercritical pressure offers the following additional advantages :

- clear flow conditions in the evaporator
- optimum selection of the intermediate pressure
- higher efficiency
- lower blower capacity

These advantages offset the disadvantages of the larger steam generator heating surface and higher thermal stress. Either 180 or 225 atm.abs. and 525°C before the turbine are selected for the steam.

The core outlet temperature could in fact be increased to over 1000°C. The gas temperature is above all limited by the maximum permissible temperature of the fuel particles, which is at present 1350°C. Under these conditions, a preliminary study showed that, in view of the temperature drop in the fuel elements, the gas outlet temperature should be in the range of 750 to 850°C. From an economic point of view, a high gas temperature is desirable.

An increase from 750 to 850°C improves the efficiency by about 1.5% (0.6 points) on account of the reduction in the blower capacity and assuming optimum preheating. Owing to the diminished heating surface and throughput, the reduction in the blower capacity is almost 45%.

The same temperature rise, however, enhances the thermal stresses in the steam generator pipes by 1/3 on account of the higher heating surface load. The required heating surface increases by 8% in the case of a reduction in the gas inlet temperature from 850 to 750°C. By increasing the temperature gradient at the cold end of the steam generator by about 20°C, this can be offset to a great extent, although the blower capacity would again be increased.

There are further disadvantages at a high temperature with regard to graphite corrosion.

Small amounts of water and hence oxygen enter the primary gas circuit through leaks in the steam generator.

The reaction rate of graphite corrosion due to water vapour is at 750°C determined by the radiation influence only. At higher temperatures the reaction rate is considerably increased. The throughput through the gas purification system must therefore at 850°C be greater by a factor of 6 than at 750°C if the graphite corrosion in the core is assumed to be equal in both cases.

The gas temperature also influences the graphite shrinkage in the case of large neutron doses. The functional relation between shrinkage, temperature and dose rate is, however, to a great extent dependent on the type of graphite used. The components that are affected by the shrinkage most are the bottom structure, lower reflector and graphite structures in the core.

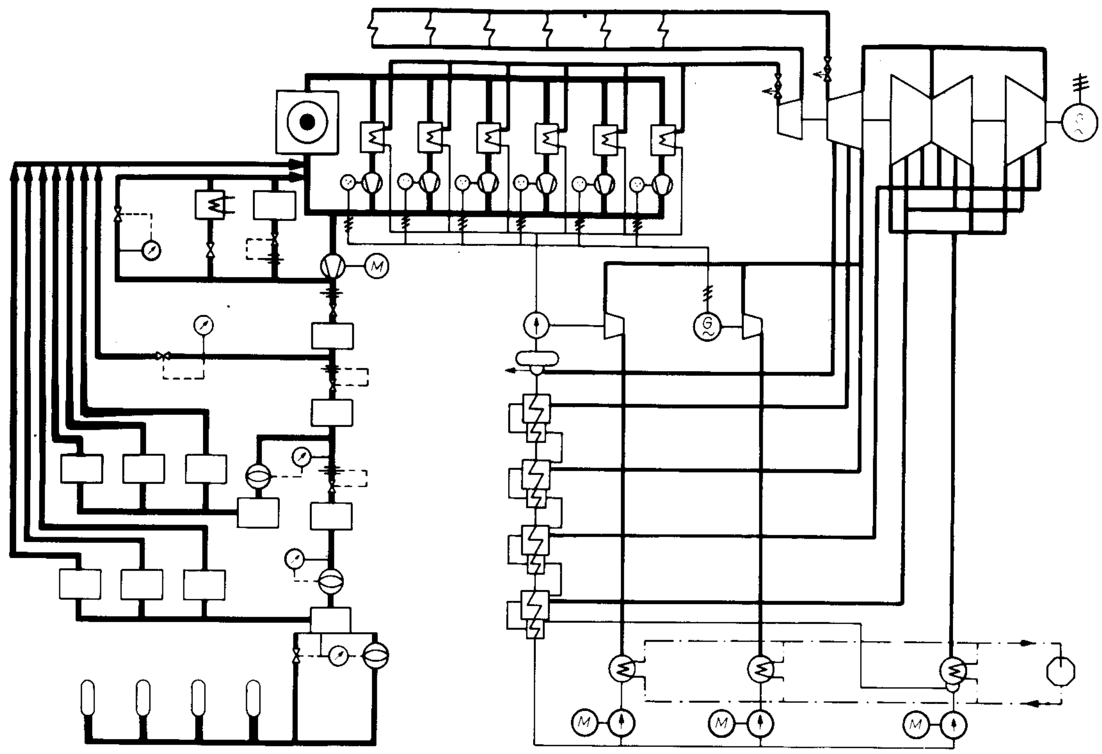


Fig. 1 — Sketch of the Gas-Vapor-Water circuits.

Since removal of the graphite components concerned should be avoided as far as possible, the shrinkage has been taken into account in the design, which is kept within reasonable limits at an average gas outlet temperature of 750°C. We have therefore chosen 750°C.

To determine the system pressure, the dimensions and cost of the steam generator, blower and prestressed concrete vessel have to be taken into account primarily.

The cost of the prestressed concrete vessel increases progressively with rising pressure; for the blower they decrease by the squared power and for the steam generator they remain about constant. 40 atm.abs. turned out to be a suitable gas pressure from the economic point of view.

All these data were based on a circuit plan which is shown in Fig. 1. In the upper centre of the figure you can see the primary gas circuit with the core and the six steam generator blower units connected in parallel. These six units ensure a high measure of safety. Each element can be made in the workshop and installation is comparatively easy. The integrated blowers (blower impeller, bearings and drive motor from one unit) are supplied with current through a turbine-driven variable-speed medium frequency generator.

Feed water is supplied to the steam generators by the main feed pump. They furnish the high pressure turbine with live steam. Between the high pressure and medium pressure turbines are the six intermediate superheaters, which are symbolically shown at the top of the picture. The secondary circuit differs from that of a modern steam power plant only in that it has only five preheating stages (four low pressure preheaters and one mixer heater and degasser). In the lower right-hand corner of the picture you can see the recooling plant with cooling tower, which supplies all the coolers with cooling water in addition to the secondary circuit.

The left-hand part of the picture shows schematically the auxiliary gas circuits. Non-purified gas is required for the fuel ball cycling. Part of the gas flows through the water separating unit. Most of the gas is supplied to the graphite filter. Part of this purified gas is returned directly to the primary circuit, while the remainder undergoes a chemical purification. This purified gas serves various purposes, such as control gas and blanket gas.

The remaining chemically purified gas is led to the decontamination plant. This superpure gas feeds the charging unit locks, acts as a scavenging gas when the primary gas components are removed and also maintains the pressure in the primary circuit. The helium cylinders are supplied with gas through a superpure gas tank.

The preliminary design data are compiled in the next table. The data will be finally fixed early February.

Power	300 MWe
Core power thermal	750 MWth
Plant efficiency	40%
Core dimensions :	
Height	5.15 m
Diameter	5.6 m
Reactor outlet temperature	750°C
Reactor inlet temperature	270°C
System pressure	40 atm.abs.
Pressure drop in reactor	1.2 atm.
Blower power	12.0 MWe
Power density in core	6 MWe/m^3
Number of steam generators	6
Number of blowers	6
Number of control rods	24
Max. coating temperature in the fuel element	1350°C
Max. load per element	3.4 kW
Uranium charge per element	1.04 g U^{235}
Thorium-Uranium ratio	10 : 1
Number of pellets	675,000
Prestressed concrete vessel :	
Height	18.0 m
Diameter	15.6 m
Live steam state :	
Pressure	225 atm.abs.
Temperature	525°C
Intermediate superheat :	
Pressure	65 atm.abs.
Temperature	525°C

Some Special Engineering Problems

Mr. U. Hennings

The steam generators of the THTR prototype are based on a series of criteria which not only differ from conventional boiler techniques, but also partly from their predecessor, i.e., the AVR steam generator:

- They must be removable, but should not require the usual maintenance;
- They should have a small outer diameter and should cause a low gas pressure drop;
- The necessary design volume should be obtained over the length, but the units must be fixed very firmly.
- There must be a good controllability, but only a few penetrations through the reactor pressure vessel should be required;
- At the same time, high demands are placed on operational safety and good partial load characteristics are desirable;
- Compact design should be brought into line with the requirements of easy fabrication, testing and safe layout.

These partly contradictory requirements were stated by Brown Boveri/ Krupp in a tender addressed to the various European boiler suppliers.

It was noted with satisfaction that the industry concerned is greatly interested in such cooperation. We received eight bids, some of which incorporated variants. The evaluation confirmed that our specifications can be met, broadly speaking, although certain points and problems will have to be solved, such as the heat stresses in the pipe wall, to which I shall refer later on. On the other hand, the call for bids, which was disseminated fairly widely, showed that not all companies reknowned in the conventional boiler field but not possessing specific knowledge in the sphere of high-temperature reactors are able to understand, assess and to solve our problems unless very close cooperation is maintained with them.

I now want to discuss another specific question which emerged during the present study, namely, the possible or permissible thermal stresses in the pipe wall. First of all, it may be a source of surprise to hear that thermal stresses are not a conventional but a specific problem, linked up with the development of the THTR reactor. This question is not concerned with the high average temperature in the tube wall, but with the temperature profile, i.e., the difference between the inner and outer tube wall surface. It is quite clear that for the removable steam generator in the prestressed concrete vessel, a compact structure is required which consequently means a higher heat flux. This can easily be obtained by using helium as the heat transfer medium even without extending the surface of the pipes. The heat transfer coefficient on the gas side as the decisive value is unusually high. From the heat conductivity equation it follows that a higher heat transfer Q requires a larger ΔT for a given material and pipe geometry. If thermal expansion is prevented in a component, thermal stresses are set up which can be determined from the formulae given in Fig. 1 for the flat plate and the pipe. These stresses should be added to those created by the pressure in the tube.

I should like to give some figures for the THTR superheater part. The ΔT wall, that is to say the temperature difference between

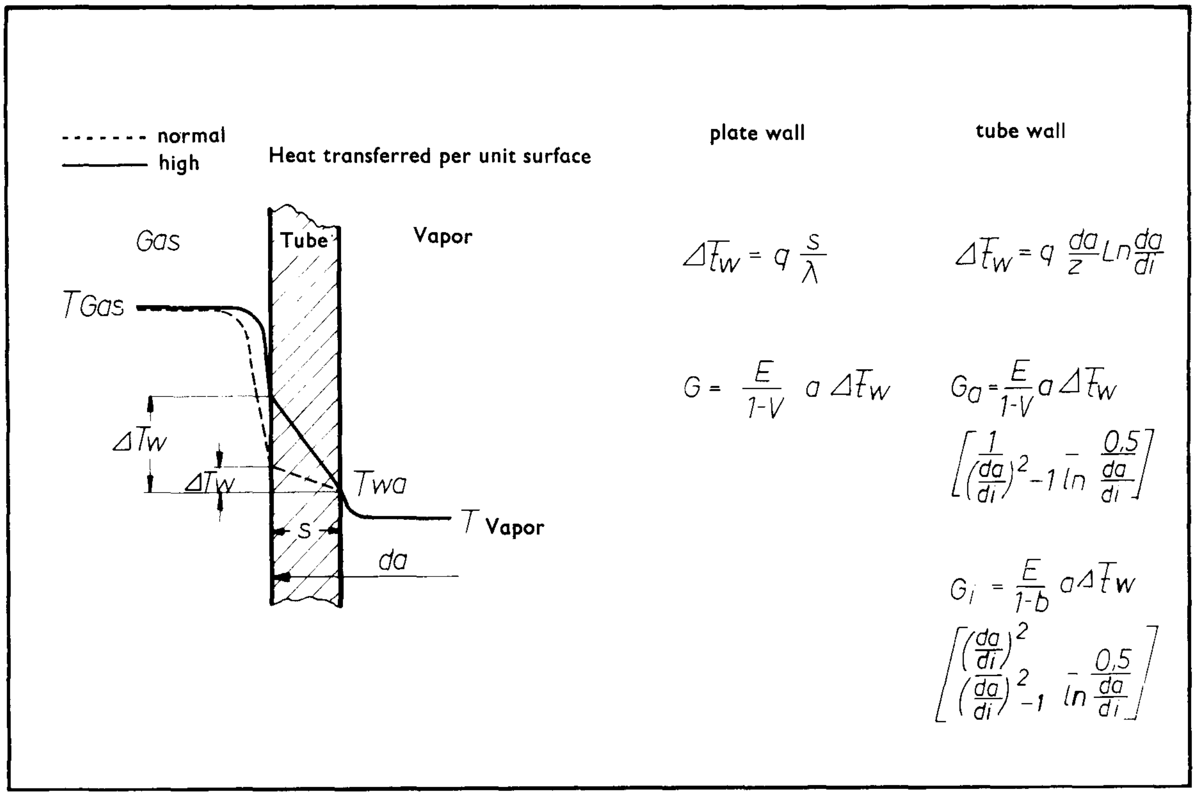


Fig. 1 — Temperature distribution and thermal stresses in the tube.

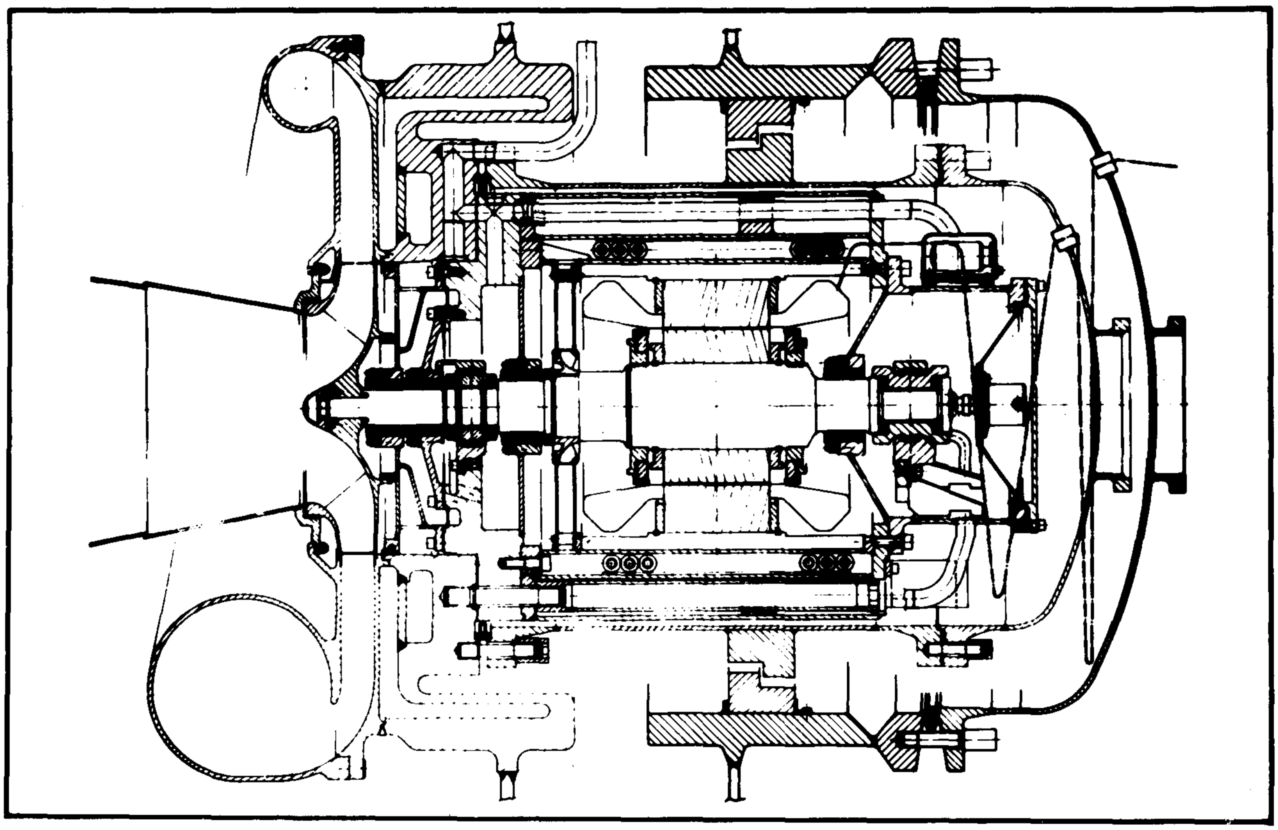


Fig. 2 — Section through the AVR blower.

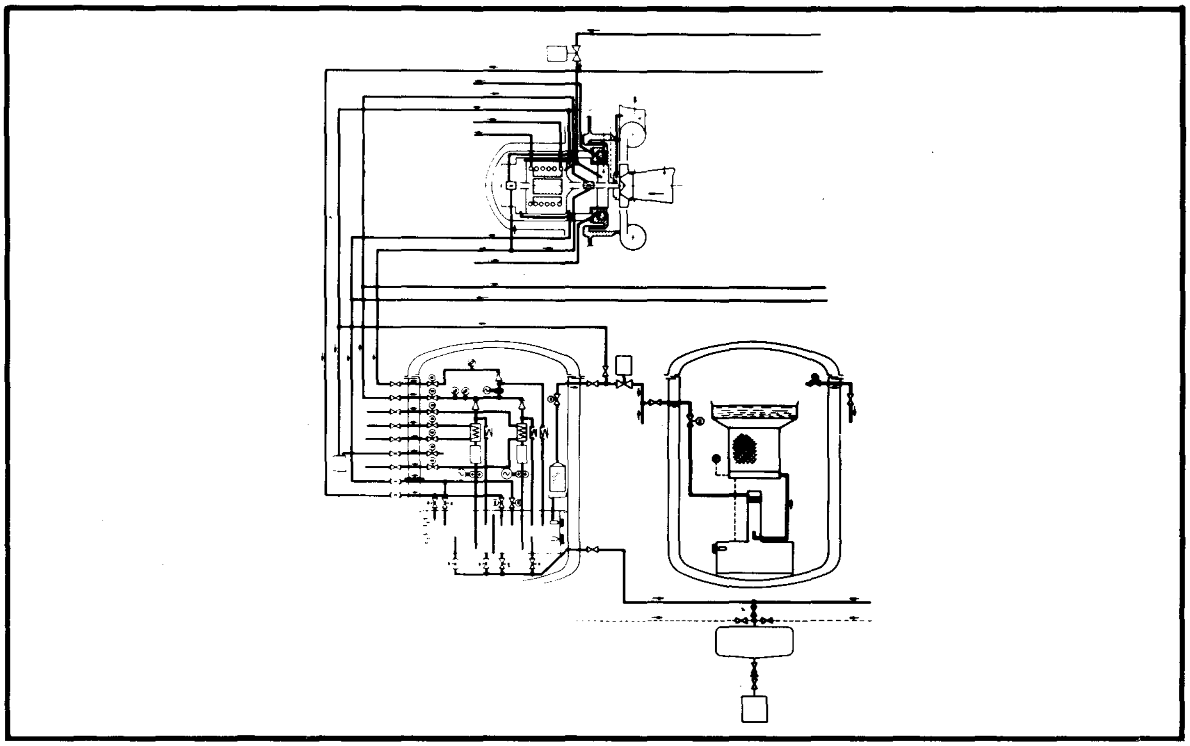


Fig. 3 — Sketch of the oil-gas circuit.

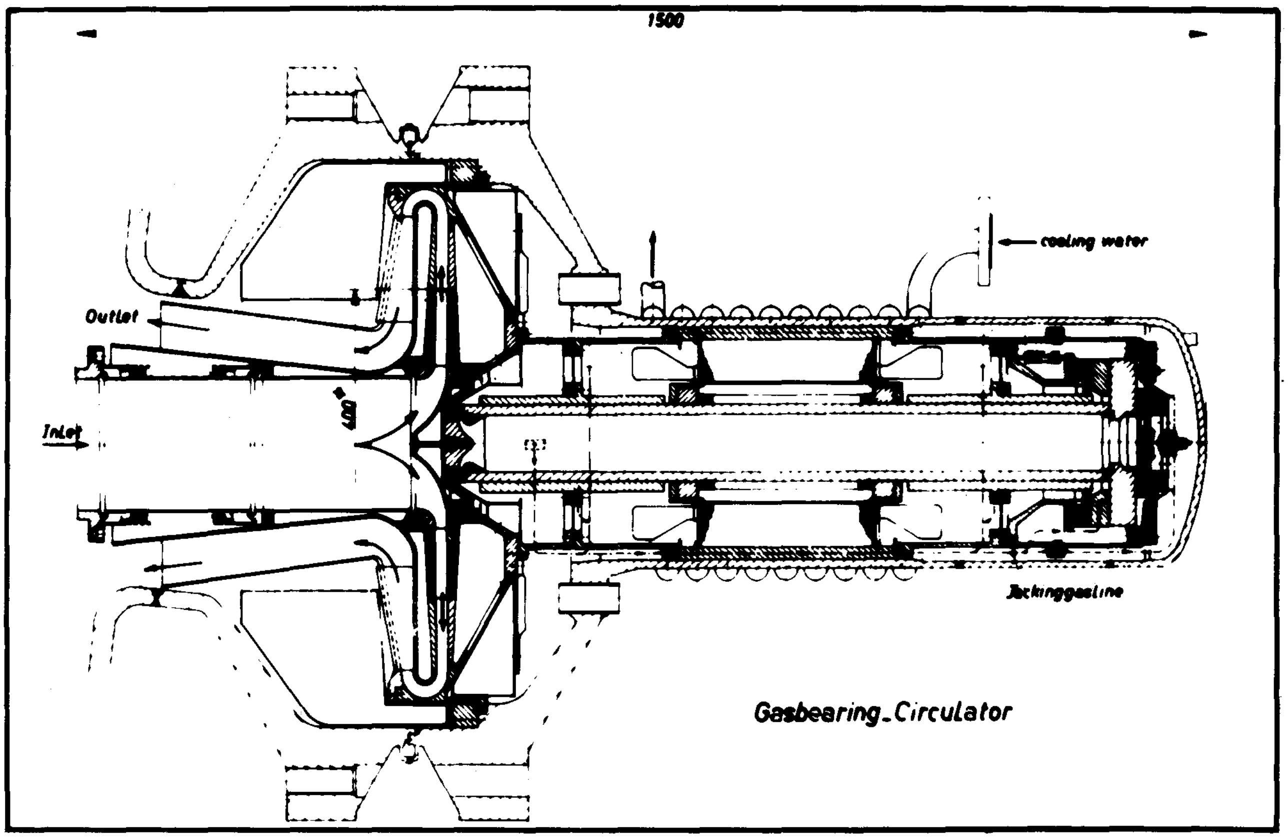


Fig. 4 — Section through the Dragon blower.

the outer and the inner phase is 70 to 100° C for an average wall temperature of 510° C. The heat penetration is 900 to 1000 kilocalories/m^2/hr/° C.

If this were a permanent situation, then the creep of the pipe material would be the decisive material constant. The pipe could absorb thermal stresses and then reduce them by creep. Elongation of up to 10 % would not be dangerous here.

However, this becomes critical in the case of frequent operating changes where the temperature profile is altered in such a way that the thermal tension is reversed. Both the mechanical and thermal cycling stability of a material is lower than under steady-state load. It became obvious that in contrast to the pressure load so far there are no regulations as to the thermal stress to be admitted in boiler pipes. The Fachverband Dampfkessel-, Behälter- und Rohrleitungsbau (association for vessels, pipes and steam boilers) uses an additional amount added to the calculation temperature

for unheated pipes this is 15° C
for convection heating areas 35° C
and for radiation heating areas 50° C.

The American ASME Boiler and Pressure Vessel Code only contains information referring to temperatures up to 370° C. In the THTR steam generator the average pipe wall temperatures are over 500° C.

As we do not have corresponding regulations, Brown Boveri/Krupp applied to various centres or experts in an attempt to arrive at the best possible assessment of thermal stress. These inquiries have not yet been completed. There is general agreement about the fact that this is a time problem and that the frequency of the thermal cycles is primarily decisive. Experimental results do exist from third parties; however, it is questionable whether they can be transferred to our conditions. Oil-fired boilers are used for heat fluxes of the same order as for the reference boiler design of the THTR, i.e., for about 400,000 calories/m^2/hr. In this connection we have thermal stresses of 15 to 20 kg per mm, which has not led to any damage so far. Although one manufacturer considers stresses of over 30 permissible, we consider 20 per mm^2 as the limit value, a value where we do not expect any noticeable shortening of the steam generator's useful life.

Attention should be drawn to the fact that when the load changes, causing local creep, the protective layers of the pipe can be affected.

In view of the enormous influence which the value of the permissible thermal stress has on the layout of the steam boiler, prestressed concrete vessel, blower power and gas temperature, then the importance of the problem of thermal stresses in steam generator pipes is immediately obvious.

For the development of the THTR reactor the primary blowers are just as important as the steam generators. Here again we have new criteria in comparison with the AVR. In the AVR reactor the cooling gas was circulated through radial blowers of 75 kW each. The blower impeller is attached to the motor shaft. The shaft has two oil-lubricated bearings. In order to prevent impurities entering in the cooling gas, measures have been taken to prevent oil from penetrating into the helium circuit. By a labyrinth gland and a sealing system, these requirements are met. In order to increase the operational safety in the blower start-up and the run-down phases, a high-pressure oil lubrication system has been installed. The sealing gas, loaded with oil, is being purified. All in all, this leads to a very expensive oil supply and sealing gas system, with many pipe penetrations to the blower.

The use of gas-bearing blowers makes it possible to simplify the construction and would be an important step forward in the development of gas-cooled reactors. In the DRAGON reactor the step towards gas bearings has already been taken. The low-power blowers have withstood the test so well that for the assessment study for a large power station, gas-bearing blowers are being considered as well as oil-lubricated ones. For a 300 MW station, six blowers each with a power of 2 MW are necessary. If we compare the power of the DRAGON blower with that of the blower for the THTR project, it will be seen that the power is increased by a factor of 30 to 40. It is quite obvious that a simple extrapolation is no longer possible in this connection.

A distinction is made between dynamic and static bearings, depending whether or not the necessary gas pressure for the bearing is produced by the operation itself or by the introduction of pressurized gas. According to the information available this situation is as follows. The blowers used and the experiments conducted so far have shown that dynamic bearings are very suitable for the purpose for which they are required. According to the considerably lower viscosity of gases, (ratio 1/1000 in comparison to oil), the application of the aero-dynamic principle requires lower tolerances in the bearings. Tests have shown that tested bearings of a diameter up to 200 mm and loads of about 800 kg run satisfactorily at normal or partial load. On the other hand, the behaviour of the bearings in a complete blower under genuine operational conditions has not yet been tested.

Static bearings can also be used. When the blower is mounted vertically, this type seems to be particularly advantageous. The static bearing has a somewhat larger bearing clearance, so it is less sensitive to cooling gas impurities and the danger is reduced. However, its application requires a higher consumption of jacking gas. The jacking gas pressure must be a few atmospheres above the system pressure, and it must be produced by an additional and very safe blower.

Two control possibilities are being discussed. The variable speed control and prerotation control. The latter is ideal for gas bearings, because it is not necessary to change the blower speed. The movable blades at the entrance to the impeller, however, make the installation more complicated, so that other movable parts have to be installed and operated within the reactor pressure vessel. Dynamic bearings can only be controlled down to a certain speed.

The companies concerned are already aware of the limits of the gas-bearing technique, i.e., the maximum load which can be used, the lower speed limit which can still be controlled, and the bearing sizes that can be operated and produced.

Design data for complete cooling blowers with a power of more than 1000 kW are available. The blower wheel can be calculated and correctly dimensioned on the basis of experience. This applies also to gas bearings. Gas-bearing blowers of the type envisaged have been tested, but only for lower powers. Additional development work is necessary, in particular the testing of larger units on the test rig under reactor conditions. The construction of the THTR prototype will be carried out in such a way that oil-bearing as well as gas-bearing blowers can be used.

The Fuel Element Cycle

Dr. W. Rausch

The construction line being followed for the THTR Project is so far the only one in which the aim of the development work is a power reactor with differential fuel elements. The project, therefore, differs from the construction lines followed elsewhere, even for gas-cooled high-temperature reactors.

Fuel elements are called differential if their reactivity contribution to the entire reactor is negligible and if their dimensions are so small that within the fuel elements a uniform burn-up is achieved.

This clearly produces very advantageous results, namely, the insertion and withdrawal of individual fuel elements can occur at all stages of operation; a continuous burn-up control offers the possibility of optimal exploitation of the fuel; in addition, the mechanical behaviour of the fuel elements can be constantly checked.

In order to make use of the advantages that follow from this choice of fuel elements, certain installations are required. The layout of these installations will form the subject of my address.

The first figure shows a schematic diagram of the fuel element cycle. Most of the fuel elements are contained in the reactor core. There are about 800,000 of them. Through one or more openings in the core bottom, fuel elements are continuously removed and passed through the devices for checking their mechanical and nuclear characteristics, in particular the measurement of their gamma activity and burn-up. Fuel elements with mechanical defects or fuel elements with an optimal burn-up are removed from the circuit and replaced by new ones. All other fuel elements are depending on their burn-up, reintroduced into the core at suitable points. The transport of fuel elements takes place in the descending branches under gravity and in the rising branches pneumatically by compressed helium.

Computer-guided controls make sure that the fuel elements are properly conducted at the various branch points in the cycle, and they also evaluate the results of the fuel element checkings and the measurement of the operational data of the plant. They thus ensure criticality of the reactor at all stages of the operation, as well as optimum utilization of the fuel.

Now, what requirements must the cycle components meet ?

The first question is that of the circulation rate of the fuel elements, i.e., how many fuel elements should be circulated per hour in the cycle ?

The first answer to this question stems from the requirement that in steady-state operation each fuel element should pass the checking installations several times during its lifetime. If we demand about six checks during a lifetime of about three years, this means that about 200 fuel elements must be transported per hour.

Now it would seem sensible that the fuel elements should be checked more often as they get older. In the first place, as they get older, the probability of mechanical defect increases, and secondly, burn-up control becomes more important as burn-up increases. Here we are helped by a phenomenon that has already been mentioned: the flow behaviour of the fuel element balls in the core and the different times that are needed for them to cross the core.

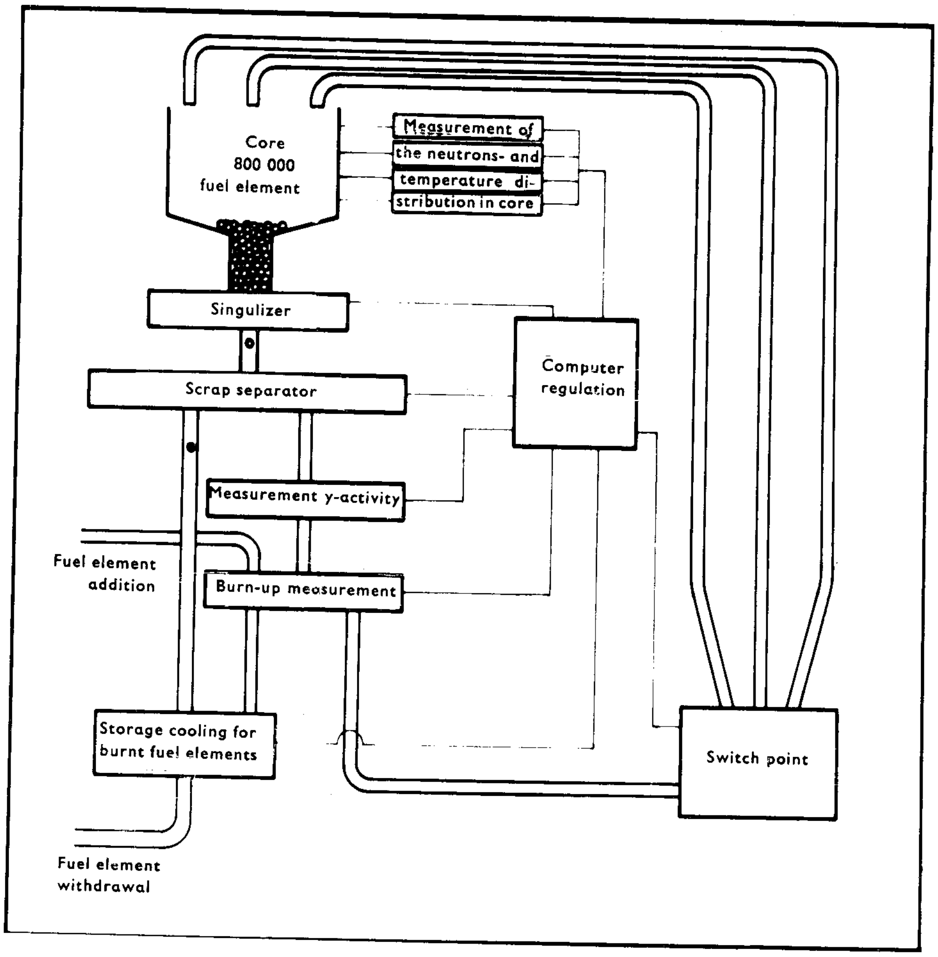


Fig. 1 — Schema of the loading-unloading circuit.

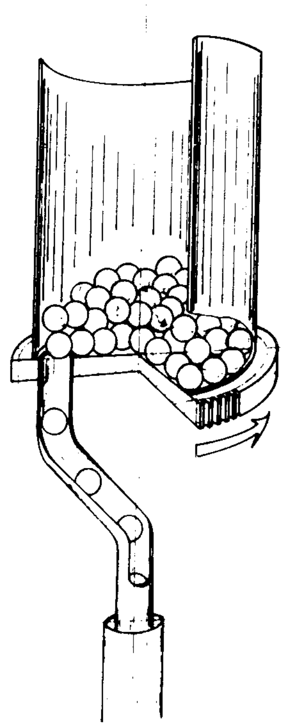


Fig. 2 — Schematic representation of the Singulizer.

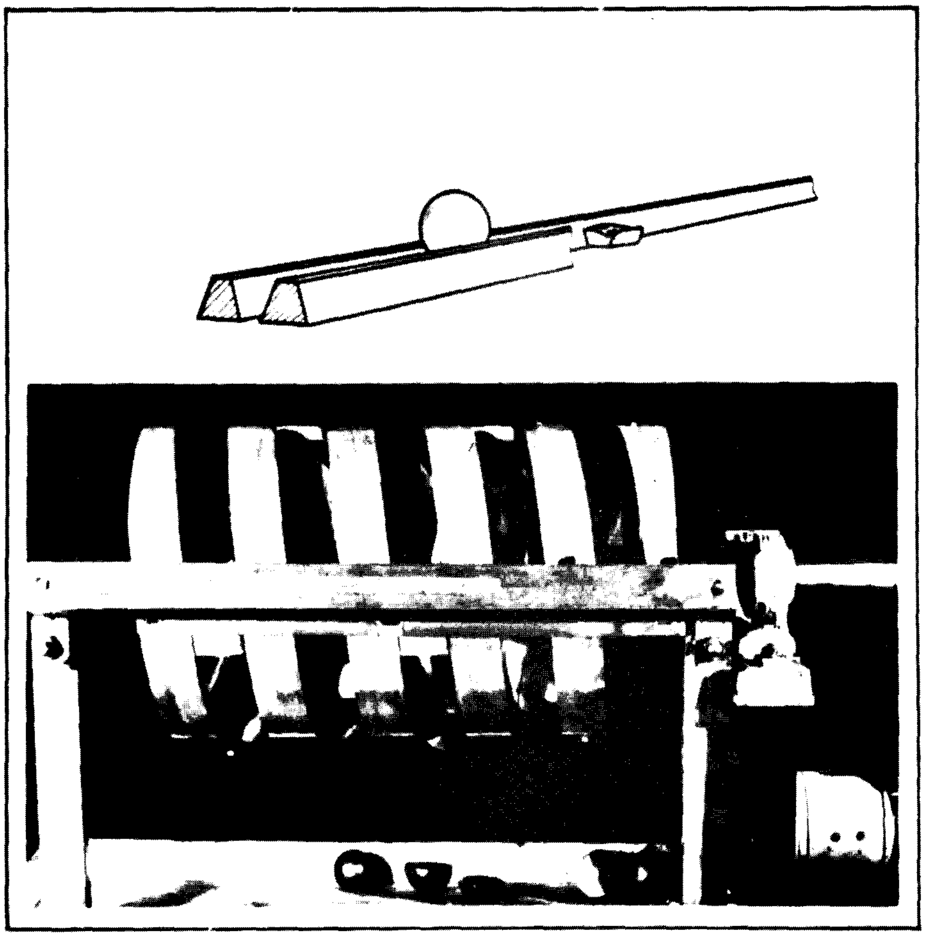


Fig. 3 — Scrap separator.

Because of this, fresh fuel elements can be inserted at positions in the core surface from where they take one year or more to cross the entire core. High burn-up fuel elements can, however, be added in places where a relatively shorter time would be required to cross the core, so that they can be checked more frequently.

The flow behaviour of the fuel elements through the core depends, as we have heard, considerably on the core geometry. It can be influenced by the number of pebble outlet openings and their layout. Another inevitable influence results from in-core structures that may be necessary to insert the shut-down rods. If we take it that only a compromise arrangement will assure complete harmonization between core layout and fuel element cycle, a circulation rate of about 300 fuel elements per hour must be envisaged.

So far we have only looked at steady-state operation. The position is different at reactor start-up because here, apart from fuel elements, pure graphite balls and probably also balls with boron inserts will be used. This requires the fastest possible change of the core configuration, as we cannot afford to wait a year or longer till the last dummy element has left the core. In order to have a certain amount of margin we must increase the circulation rate by a factor of 2 at least, that is, to about 600 elements per hour.

We have seen that the fuel elements leave the core by one or more openings, fitted with outlet pipes which are so measured that there can be no bridge-building. These outlet pipes end at the singulizer disc. The following figure (Fig. 2) shows how, with the help of the rotating singulizer disc, the flow of numerous fuel elements is broken down into a single file of individual fuel elements. The principle shown here is a further development of the device which we had in the AVR reactor, and which has, in a similar form, already formed the subject of a considerable amount of experimental tests.

After having gone through the singulizer, the fuel elements pass the various control checks, the first being the scrap separator. Here broken or damaged fuel elements are ejected.

One version being considered is shown in the next figure (Fig. 3). It is based on the fact that an intact fuel element can roll down between two inclined rails if their spacing is smaller than the diameter of the fuel element. Breakages, or fuel elements which, as a result of damage, have a smaller diameter than the distance between the two rails, will fall between the two rails if, with the help of diverters, it is ensured that practically all the diameters of the balls are tested. If these rails are joined to a spiral, then we have not only a more compact arrangement for the same length of the test course, but also the advantage that the roll-off speed of the balls can be regulated. In the lower part of the slide you see such a spiral scrap separator with which experiments are being performed at the moment.

The elements are then passed through an ionization chamber, where the gamma activity is measured and a distinction made between highly active fuel elements on the one hand and slightly active or dummy balls on the other. This considerably facilitates later differentiation. From here the fuel elements go on to the burn-up measuring device, and here lies the most difficult problem from the point of view of the measuring technique.

Under the more stringent conditions of start-up operation, a burn-up of at least 600 fuel elements per hour would have to be determined. If we had only one measuring device, this would mean that we would have less than six seconds for the measurement of one individual ball. If there is a larger number of parallel burn-up measuring devices, then the margin available for an individual measuring becomes greater. On the other hand, because of additional branching points, the fuel element cycle becomes more complicated and more costly.

Two measuring methods are under discussion at the moment. One is the measuring of fission product activity, particularly that of caesium-137, which has a sufficiently long half-life, and burn-up analysis by reactivity measurement in a critical facility.

Measuring the intensity of the gamma transition from caesium-137 was suggested for the determination of burn-up very early on. The special difficulties in the application of this method to the THTR elements are twofold.

The first results from the requirement of a very short measuring period of only a few seconds, together with the fact that the fission product activity of THTR fuel elements is very low because of the very low content of fissionable material. Conventional, non-differential fuel elements contain more fuel by a factor of 100-1000 than our THTR elements do, and therefore, of course, they produce a higher fission product activity. At the same time, burn-up measuring for conventional fuel elements may take many hours.

The second difficulty stems from the requirement that as few fuel elements as possible should be outside the core. This is to limit the cost of the total fuel element inventory. If we request that not more than 5 % of all fuel elements remain outside the core, then for a circulation rate of 600 fuel elements per hour, this means a time difference between withdrawal and insertion of an element in the core of only 66 hours. The cooling-down period for short-lived fission products therefore amounts to less than three days. This is particularly disagreeable because the energy of the caesium transition is close to the transition energy of the short-lived (78 h) tellurium-132 with an energy difference of only about 12 keV.

Therefore the measuring apparatus must have a high counting yield and at the same time an energy resolution of less than 12 keV. The methods applied in Vienna and Studsvik with a Compton or crystal spectrometer do not meet these two requirements simultaneously. On the other hand, we have an extremely interesting and possibly successful method under discussion here in Jülich. By directly measuring the caesium photo peak with the help of germanium diodes, we obtain a resolution of around 5 keV. At the same time a much higher counting yield than in the Compton and crystal spectrometers is achieved, so that less time is needed for measurements. It seems possible that for measuring times of about 30 seconds the necessary exactness can be reached. At a circulation rate of 600 fuel elements per hour, five parallel measuring devices would therefore be sufficient.

However, the question remains as to how long the cooling-down period must be for short-lived fission products, and whether the 66 hours to which I have referred are sufficient in order to allow the activity of tellurium to run down sufficiently so that the necessary accuracy in measuring the caesium line can be achieved. Further experiments are required for this and are now going on.

The second method that is up for discussion is based on the fact that a reactor with a sufficiently small critical mass reacts with high sensitivity to the insertion of fissile, absorbing and scattering substances, and that the reaction, because of the short life of the neutrons involved, takes place so to speak "promptly". The change in power to be observed depends not only on the amount of material introduced in the reactor, but also on the location of the sample in the reactor. On the next figure (Fig. 4) you can see what we call the influence functions for the various substances. In the upper part of the figure you can see that for fissile substances there is a positive power change, with a maximum in the core centre; for scattering substances the power change is also positive with a maximum at the core edge, which is due to the reflector effect of such samples; samples of absorbing substance, however, bring about a negative power change with a maximum in the core centre.

Corresponding measuring results are shown on the same figure. Here samples were rapidly moved through the central channel of the Danish reactor DR1 in Risø. The resulting power was recorded as a function of the time. The time scale is shown in units of one tenth of a second. As you can see, the length of the signal, or in other words the time the sample takes to pass through the core, amounts to about two-tenths of a second.

The first sample consists of ten grams of pure graphite. Accordingly the power signal has the form of the influence function for scattering substances with the two characteristic maxima. The second sample

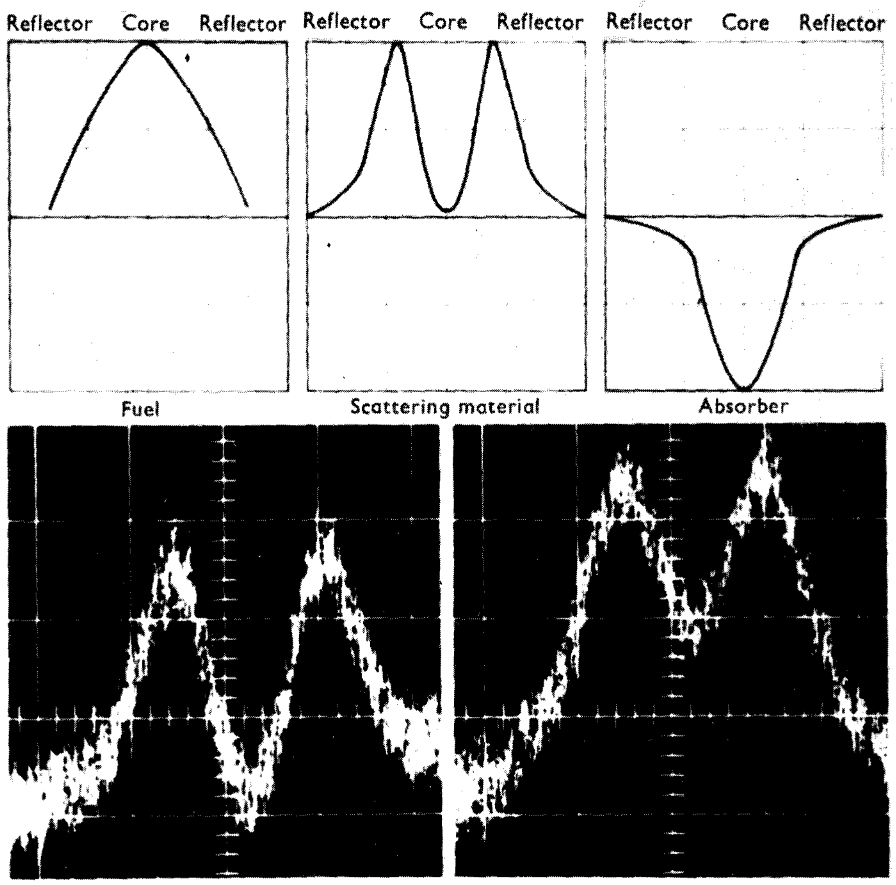


Fig. 4 — Influence function and measured signal of a critical assembly.

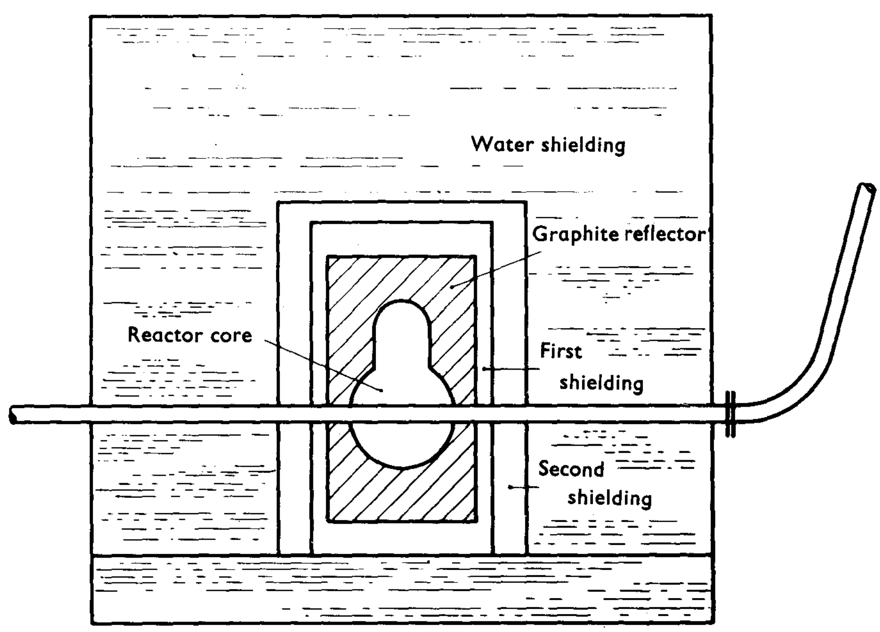


Fig. 5 — Sketch of an homogeneous liquid reactor used for the development of a burn-up measurement.

consists also of about 10 g of graphite, but also contains 0.3 g of uranium-235, i.e., fissile material. The resulting signal is obtained by superimposing the signal from the pure graphite sample on that of the fuel element sample. The difference is very clear. The value for the minimum at the core centre is considerably increased by the addition of a fissile material. The signals measured correspond roughly to the signals of completely new or partly burnt-up elements.

Similar measurements on complete fuel elements are to be performed in a small critical facility, a homogeneous liquid reactor which is to go into operation at Jülich next year. We are quite convinced that this method will permit the measurement of the burn-up of up to 1000 fuel elements per hour with the necessary accuracy. This method also makes possible a distinction between pure graphite balls and those with boron inserts, which cannot be done by gamma spectroscopy.

The next figure shows a cross-section of the burn-up measurement reactor. The fuel elements would in the simplest case be accelerated in an inclined pipe so that they would then cross the core at a sufficient speed. After leaving the measurement reactor, the still usable fuel elements, i.e., the majority of them, will be reintroduced into the core in suitable positions. The optimally burnt-up elements (in steady-state operation we expect about 10 to 25 burnt-up elements per hour) are withdrawn from the circuit and replaced correspondingly by new ones.

A special task consists in arranging the pneumatic insertion of the elements into the reactor core in such a way that the mechanical impact on the fuel elements when they meet the core is reduced to a minimum. This is to be done by improved control of the pneumatics and by certain delaying devices which are installed at the entry of the reactor.

NOTICE TO THE READER

All Euratom reports are announced, as and when they are issued, in the monthly periodical **EURATOM INFORMATION**, edited by the Centre for Information and Documentation (CID). For subscription (1 year: US$ 15, £ 5.7) or free specimen copies please write to:

Handelsblatt GmbH
"Euratom Information"
Postfach 1102
D-4 Düsseldorf (Germany)

or

Office central de vente des publications
des Communautés européennes
2, Place de Metz
Luxembourg

To disseminate knowledge is to disseminate prosperity — I mean general prosperity and not individual riches — and with prosperity disappears the greater part of the evil which is our heritage from darker times.

Alfred Nobel

SALES OFFICES

All Euratom reports are on sale at the offices listed below, at the prices given on the back of the front cover (when ordering, specify clearly the EUR number and the title of the report, which are shown on the front cover).

PRESSES ACADEMIQUES EUROPEENNES

98, Chaussée de Charleroi, Bruxelles 6

Banque de la Société Générale - Bruxelles
compte N° 964.558,

Banque Belgo Congolaise - Bruxelles
compte N° 2444.141,

Compte chèque postal - Bruxelles - N° 167.37,

Belgian American Bank and Trust Company - New York
compte No. 22.186,

Lloyds Bank (Europe) Ltd. - 10 Moorgate, London E.C.2,

Postcheckkonto - Köln - Nr. 160.861.

OFFICE CENTRAL DE VENTE DES PUBLICATIONS DES COMMUNAUTES EUROPEENNES

2, place de Metz, Luxembourg (Compte chèque postal N° 191-90)

BELGIQUE — BELGIË
MONITEUR BELGE
40-42, rue de Louvain - Bruxelles
BELGISCH STAATSBLAD
Leuvenseweg 40-42 - Brussel

DEUTSCHLAND
BUNDESANZEIGER
Postfach - Köln 1

FRANCE
SERVICE DE VENTE EN FRANCE
DES PUBLICATIONS DES
COMMUNAUTES EUROPEENNES
26, rue Desaix - Paris 15e

ITALIA
LIBRERIA DELLO STATO
Piazza G. Verdi, 10 - Roma

LUXEMBOURG
OFFICE CENTRAL DE VENTE
DES PUBLICATIONS DES
COMMUNAUTES EUROPEENNES
9, rue Goethe - Luxembourg

NEDERLAND
STAATSDRUKKERIJ
Christoffel Plantijnstraat - Den Haag

UNITED KINGDOM
H. M. STATIONARY OFFICE
P. O. Box 569 - London S.E.1

EURATOM — C.I.D.
51-53, rue Belliard
Bruxelles (Belgique)