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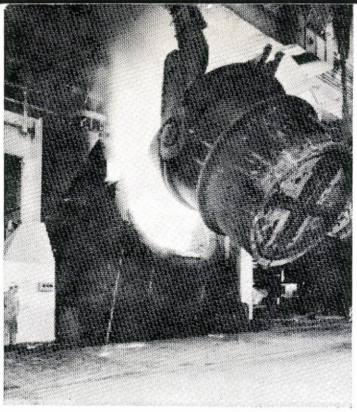

The steel industry turns to nuclear techniques – the oxygen content of this molten steel was determined by activation analysis (see p. 59).

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Quarterly Review of the European Atomic Energy Community (Euratom)

1968-2

The Community's mission is to create the conditions necessary for the speedy establishment and growth of nuclear industries in the Member States and thereby contribute to the raising of living standards and the development of exchanges with other countries (Article 1 of the Treaty instituting the European Atomic Energy Community).

The central feature of this issue is the high-temperature gas-cooled reactor. What stage of development has it reached today? What has contributed to making it an attractive candidate for a new generation of nuclear power plants? What prospects are there of further technological improvements? An attempt has been made to supply answers to this kind of question.

What is immediately striking about the high-temperature reactor is that, unlike some other reactor types, it has not led to a proliferation of rival variants. There are two basic variants only, and in any case the differences between them are essentially confined to the shape of the fuel elements and the method of handling them. This is a decisive advantage, at least on the purely practical plane, as it virtually suppresses the need for making agonising choices.

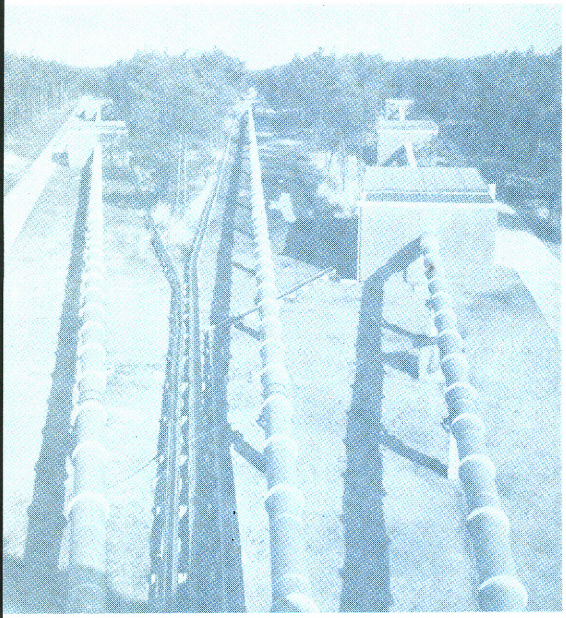
Another striking aspect is that this is a technological field where Europe seems to have a head start on the United States. But Europe must turn its know-how quickly into efficient hardware; otherwise it will be, once more, beaten at the post.

The solution to this problem is not to be found in international co-operation at the public level only. Industrial firms from different countries must be prepared to pool their experience, thereby losing some of their independence but giving themselves the chance, through a powerful grouping, of operating well beyond the frontiers of their respective lands in a highly advanced branch of technology.

This may have something to do with the politics of European integration, but it is also quite simply a matter of coming to hard business decisions.

The target - an underestimate

CHRISTIAN ALLARD, *Central Bureau for Nuclear Measurements, Euratom, Geel*



Three flight paths of the CBNM *linear accelerator.*

The control room of the CBNM *linear accelerator.*

ONE OF THE main tasks of the Central Bureau for Nuclear Measurements (*CBNM*) is to provide exact data on the interaction between neutrons and various materials in the form of total, capture and fission cross-sections, etc. Such precise information is of great importance if one wants to design and predict the behaviour of nuclear reactors without resorting too much to approximations.

One of the instruments used by the *CBNM* for carrying out neutron measurement programmes is a linear accelerator. Since neutrons cannot be accelerated by electric fields, they are produced by bombarding a target with an electron beam. This target is usually regarded as an accessory since the cost of the target is quite naturally minimal in comparison with the accelerator itself. Without undue exaggeration, it could even be said that it is understandable since, as will be seen, it is quite easy to construct a neutron-producing target from any metal. The impatience of physicists is such that the first neutrons to be detected are put to immediate use. The better target thus has to wait.

This view of things had to be modified at the *CBNM* since the time taken for a measurement becomes a primary factor affecting its cost and the efficiency of the laboratory:

The physicist using the installation requires a minimum number of neutrons in order for a given experiment to be of an acceptable accuracy. Thus, an experiment which would take 200 hours with a low neutron yield target (tungsten, for instance), will take about 100 hours with a natural uranium target and 60 hours with a target of 93%-enriched uranium. It goes without saying that, by increasing the target yield, the yield of the *whole* installation is stepped up. This represents a considerable saving, mainly through the fact that depreciation of the installation accounts for a smaller percentage of the cost of the experiment under consideration. It can thus be seen that it is not only desirable but even essential that great care should be taken over the design of a target.

The problems arising from this and the solution that we consider most suitable to aim at will now be examined with reference to *CBNM* experience.

Possible choices

There are two possible ways of designing a target: either one can select targets which are technologically simple and very reliable, consisting of metals with high melting points, such as tungsten or tantalum, but of moderate neutron-producing capacity, or it is possible to use a metal with a high neutron-producing capacity, such as uranium, although the less favourable metallurgical properties of this type of metal necessitate far more careful design, if an acceptable level of reliability is to be maintained during operation.

Despite the risks involved, the *CBNM* adopted the second choice, since the alternative would either have led to a 50% cut in the yield of the laboratory, or would have necessitated doubling the power of the existing installations in order to reach the same "output". Needless to say, this would have meant enormous investments (some several hundred thousand units of account) to obtain the same result.

This once again emphasises the disproportion already mentioned between the im-

omponent of linear accelerators

portance of the electron accelerator and the small target for which it is built.

Problems arising from the design of a target

First of all, it will be remembered that with a uranium target, the electrons of the pulsed beam produced by the accelerator are "braked" in the uranium, hence the emission of so-called gamma "braking" radiation *(Bremsstrahlung)*. In turn, this radiation causes neutron emission by two types of reaction: the (γ, n) reaction for about two thirds, giving photoneutrons, and the (γ, f) reaction for the other third, giving fission neutrons.

The design of a target is closely linked to its use and thus to the types of neutron measurements made from it. The characteristics of the accelerator are moreover fixed by this very consideration. With the *CBNM*, the accelerator at present delivers a beam with a mean guaranteed power of 4.2 kW, but this may subsequently be almost tripled.

Thus, our target had to be designed to absorb safely an output of around 4 kW, this being followed by the development of a target capable of absorbing all the available power of the beam. Two problems arise—one concerning the type of measurement, the other the power of the beam.

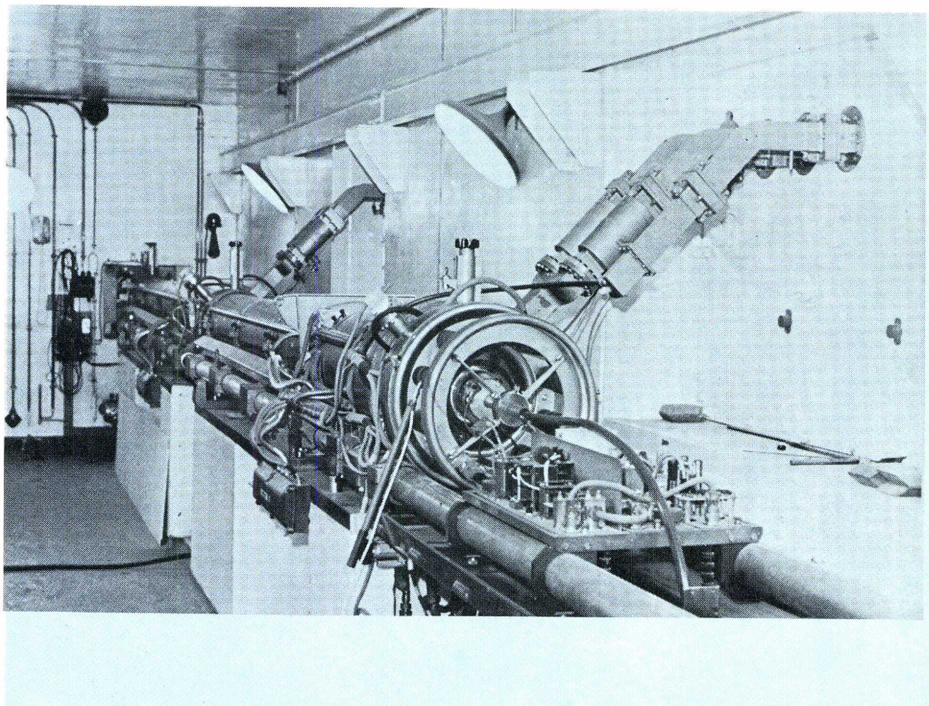
Problems arising from adaptation to type of measurement

Since most measurements carried out by the *CBNM* are neutron measurements using the time-of-flight method (see *Euratom Bulletin* Vol. IV (1965) No. 1, p. 3), the design had to be such that the target was roughly similar to a point; it therefore had to be small. In addition, its structure should not cause a high neutron relaxation time due to elastic scattering. In other words, the neutrons should not be left enclosed in the target for too long before being emitted, since this would lead to the neutron packet produced by an accelerator pulse being scattered in time and hence distort the data.

Measurements are carried out in predetermined directions (flight paths) (Fig. 1), which converge on the centre of the target. The accuracy depends notably on the precision of the measurement of the time taken by the neutrons to travel from their emission point to the detectors connected to the sample which is placed at a given distance along the flight path.

Problems arising from adaptation of the target to the accelerator

Without going into details, it would be useful to make a few rapid observations on the characteristics of the *CBNM* linear accelerator (Fig. 2) to understand the requirements which the target must meet. The operational characteristics of our accelerator both in short (10-50 nanoseconds) and long impulses (2 microseconds) at present require a beam power



The two sections of the CBNM *linear accelerator.*

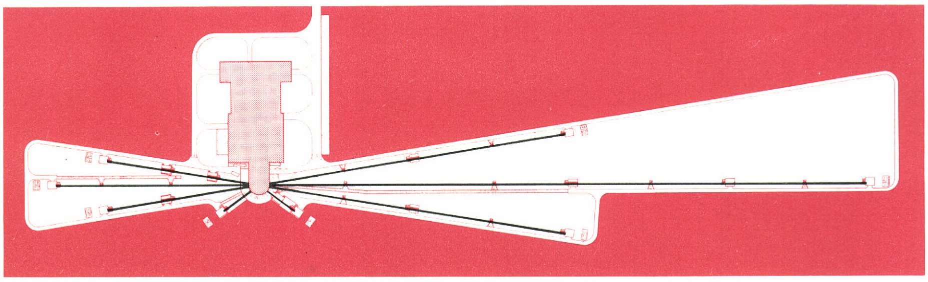


Fig. 1: Plan of the layout of the CBNM *linear accelerator and its main flight paths. The neutron detectors are housed in detection "cabins" distributed along the flight paths at set distances from the target.*

of some 4 kW. The designer of the accelerator is endeavouring to raise the beam power to about 10-12 kW[1]. It is therefore imperative that corresponding targets be developed to take these higher powers.

In all cases, the beam has a diameter of no more than 1 cm and, theoretically, 80% of the beam power is to be found in a diameter of 0.5-0.6 cm, or even less in the case of impulses of 1-2 microseconds. This enables the diameter of the active part of the target, where the electrons are slowed down, to be fixed. A diameter of 2.5-3 cm, however, is favoured since this will allow for the beam to be thrown off centre by up to one cm.

There is one arrangement of the electromagnetic optics when the beam leaves the accelerator (see Fig. 2, point 4) which, in addition to the conventional horizontal aim, permits vertical bombardment of the target, whose axis must also be vertical. This latter arrangement was adopted to direct the "gamma flash" towards the ground, since this can often perturb neutron detection and the quality of the measurements. It also gives perfect neutron symmetry of the flight paths.

The solution adopted by the CBNM

Since it was necessary to have both vertical and horizontal beams, the use of a rotating target was first of all considered. This idea was abandoned for various technical reasons. We chose the solution embodying two separate targets, as shown in Fig. 3. These two targets, which must be interchangeable, can fit onto a lift. A cell fed by a remote-controlled bogey had to be built for storing a maximum of four targets (see Figs. 4, 5 and 6).

Our first target was based on the one used at Saclay; this worked satisfactorily and trouble-free with a maximum mean power of the order of 4 kW. After 2,500 hours of use with powers of less than 4 kW, the window in contact with the uranium (see Fig. 7b) was found to be broken, however, probably owing to prolonged misalignment of the beam. No immediate contamination was detected and, with the aid of the handling and storage device, the target was replaced within an hour without any further inconvenience being caused.

To understand the general trend of the development work and the aim in mind, it is useful to cast our minds back. Our

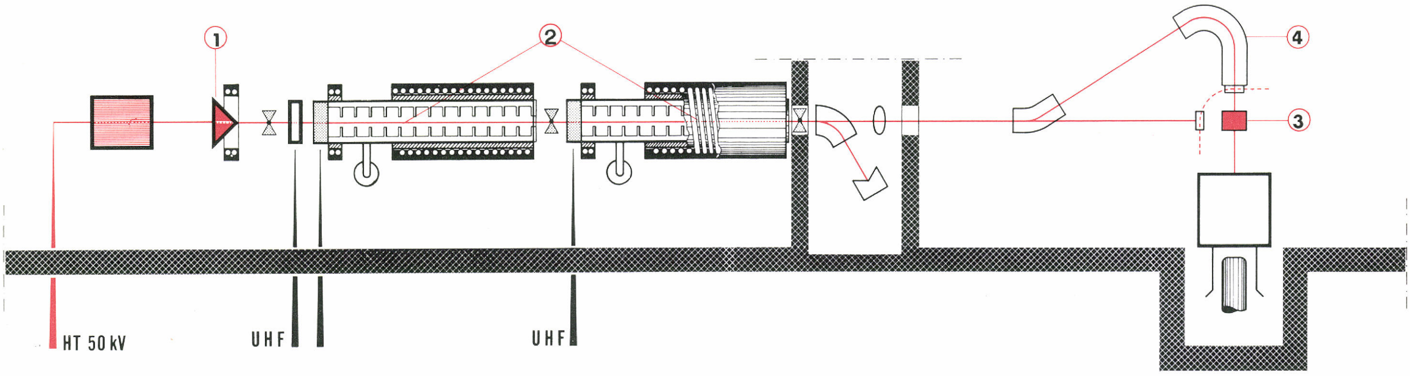


Fig. 2: The CBNM *linear accelerator.*
Electrons are produced in the electron gun (1) and accelerated. So far only two sections of the accelerator (2) have been installed but others can be added as required. The electron beam is aimed at the uranium target (3) and produces Bremsstrahlung *which in turn generates neutrons. The beam can either be aimed directly at the target or deflected (4) so as to hit the target vertically. The advantage of this second method is perfect horizontal symmetry of the neutrons in all the flight paths.*

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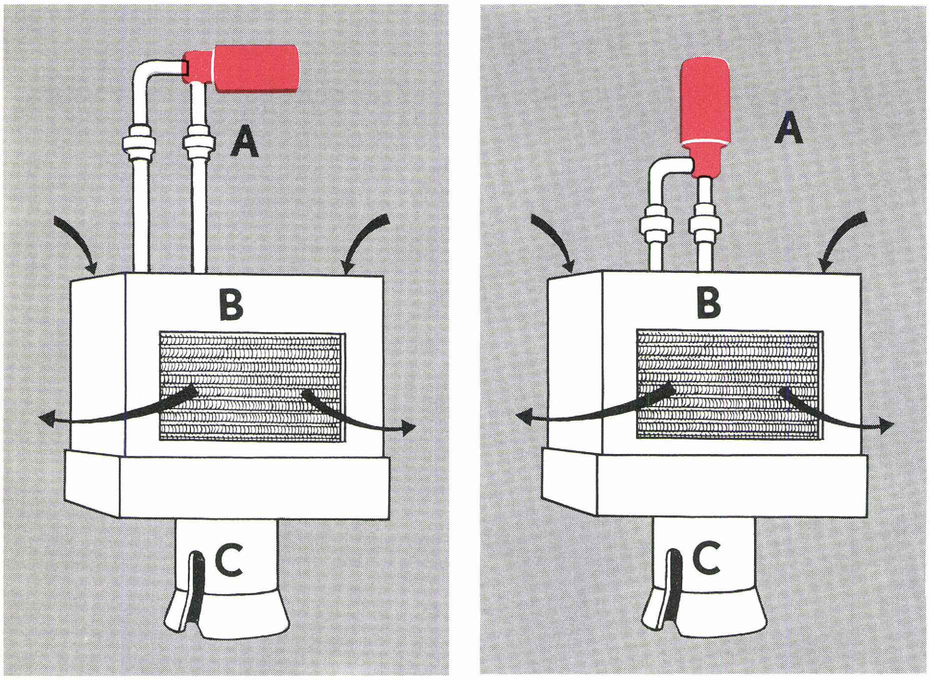
[1] The beam power was in fact raised to 10 kW in March 1968.

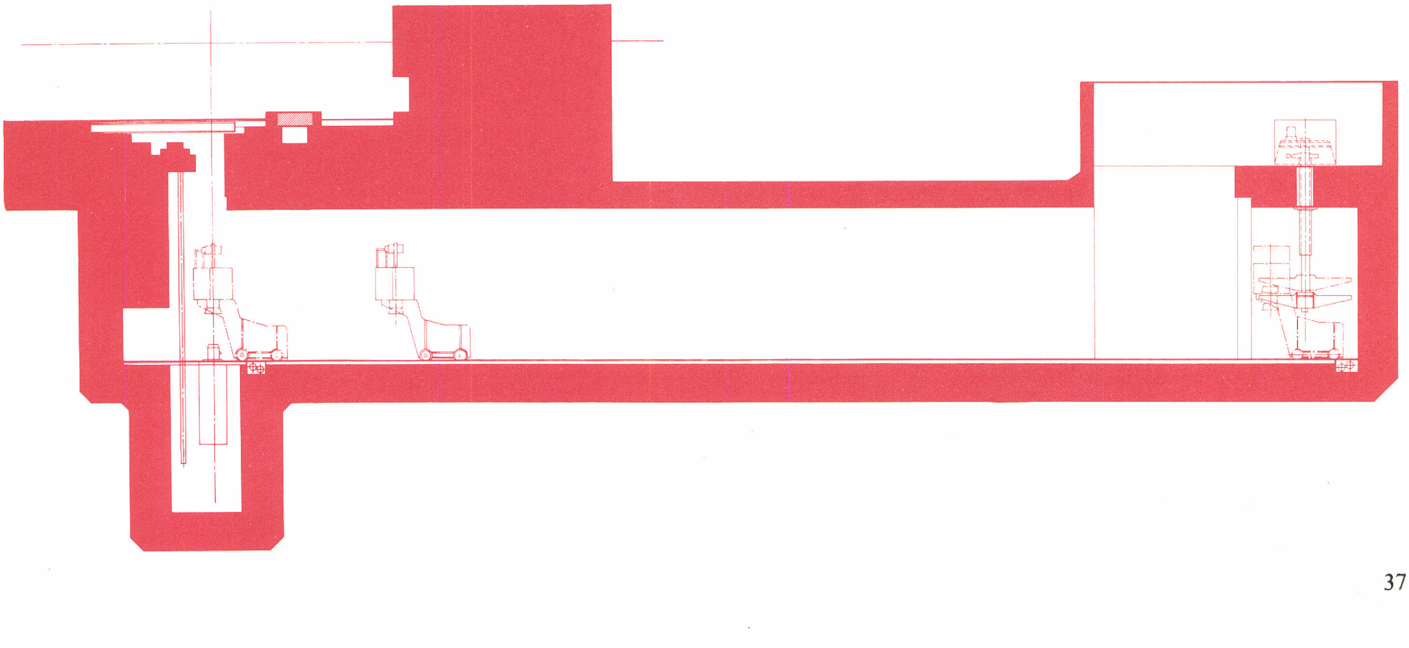


horizontal target

vertical target

Fig. 3: For the sake of convenience, the name "target" has come to be given to the unit comprising the target itself (A) and its heat exchanger (B). The target proper, which receives the beam, has the appearance of a cylinder 6 cm in diameter and 15 cm long. It is connected to the heat exchanger by two transmission systems which supply and remove the mercury coolant from the uranium inside the cylinder. The mercury is circulated by an electromagnetic pump inside the heat exchanger. This unit can be fitted onto a lift by means of a slightly conical guide (C) underneath the exchanger.
The axis of the active part of the target is aimed either horizontally or vertically depending on the beam exit direction selected.

Fig. 4: The diagram shows the positioning, handling and storage device for the CBNM *targets. On the left there is a lift, shown here in the lowered position, on which the selected target is being placed by a bogey. This bogey, which is remote-controlled, moves from a storage cell (at the right of the diagram) through an underground passage (centre) about 15 m long until it reaches the bottom of the lift shaft.*
Inside the storage cell, a maximum of four targets are automatically placed by the bogey onto a rotating tray. This can also move vertically, thus enabling the target to be positioned on the bogey. All these operations are remote-controlled from the accelerator control room more than 50 m away. Monitoring is carried out by television.



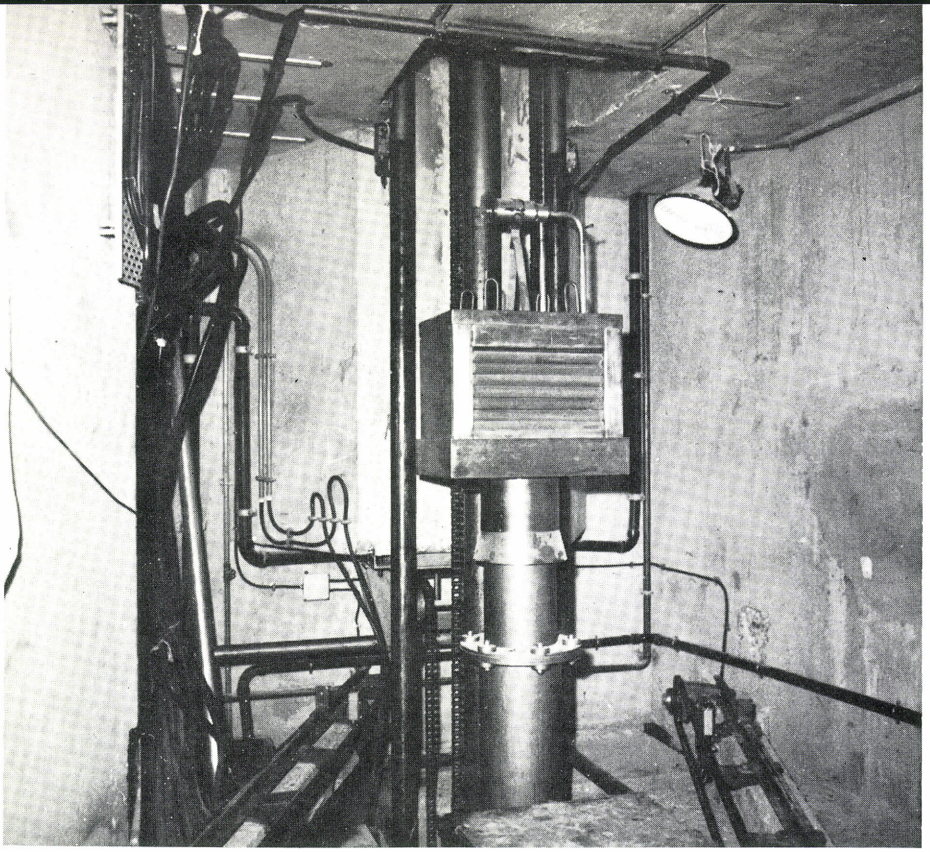
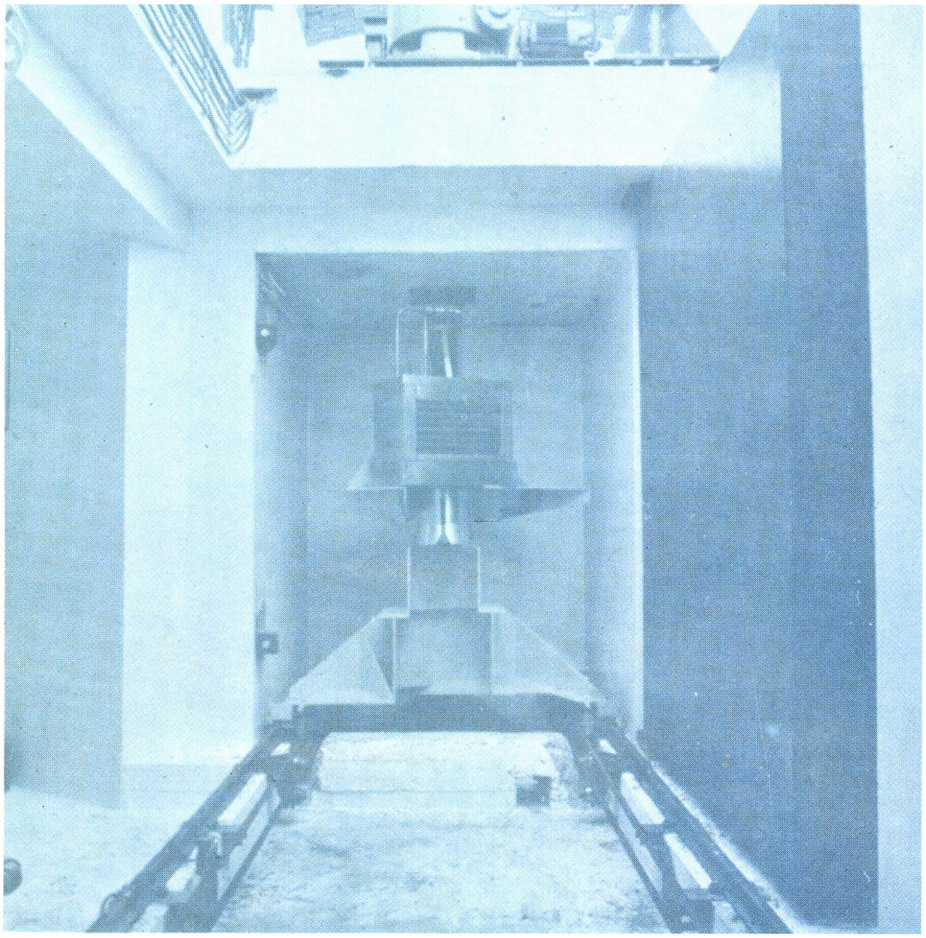


Fig. 5: View of the target in position on the lift in lowered position.

Fig. 6: View of the target storage cell: the bogey has just placed a target on the storage tray.



first target was of a simple design, in view of the lack of opportunity to carry out a systematic study of the subject and the desire to use the accelerator within a reasonably short time. It had been tested at Saclay with a maximum beam power of around 2 kW.

As in the Saclay target (see Fig. 7a), a cylindrical uranium core 3 cm in diameter and 10 cm long was used. A conical surface was retained for receiving the beam since this has the advantage of spreading the impact of the electrons and thus the heating of the uranium.

It will be seen that such a target can be regarded as infinitely thick and thus as having a maximum theoretical neutron yield for a beam power compatible with the cooling conditions necessary to keep the uranium stable.

Yield considerations

This is not the place to theorise on targets, so let it suffice to recall to mind a certain number of results which are generally accepted as adequate for defining the neutron yield of a target.

It is known that the number of neutrons produced by slowing down electrons in a substance depends primarily on the choice of substance—the greatest number of photoneutrons are produced by heavy nuclei.

Moreover, this number of neutrons depends on the thickness of the substance in the path of the electron beam and is practically constant beyond 10 λ radiation lengths[2] for a given beam power. Beyond 10 λ the target is equivalent to a target of infinite thickness. If we take natural uranium as an example, the neutron flux produced by the reaction (γ, n) and (γ, f) with a target of infinite thickness is $2.6 \,.\, 10^{12}$ n/sec. 4π . kW.

According to our definition, such a target will have a neutron yield of 100%. Experiments have shown that the yield of infinitely thick lead and mercury targets was about 50%. It is thought that the same should apply in the case of tungsten. On

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[2] There is a radiation length for each substance, it being defined as the distance that an electron beam must travel for its energy to be reduced by the proportion 1/e.

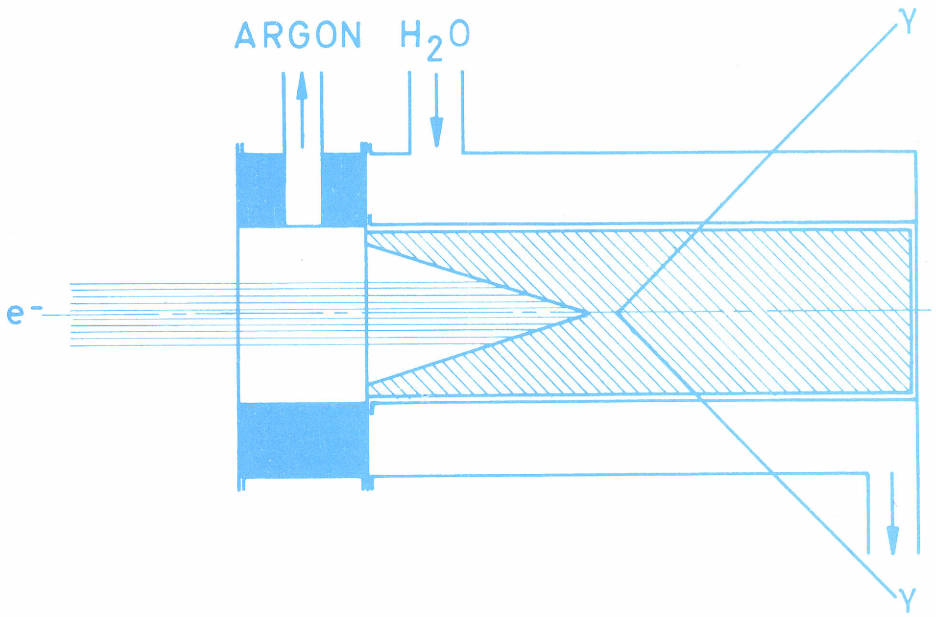
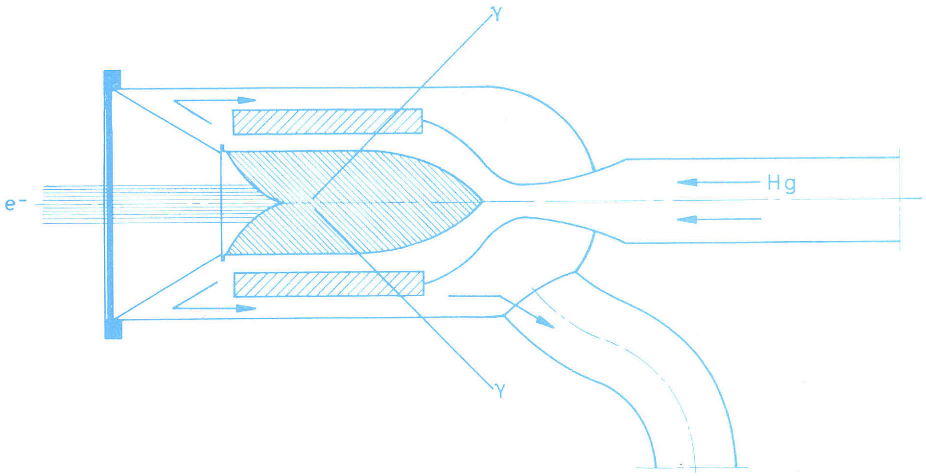


Fig. 7a: First water-cooled Saclay target. Rated power 2 kW. Since then, a new target having a nominal power of about 6 kW has been developed at the Neutron Measurements Division (SAMNF) *at Saclay.*

Fig. 7b: First mercury-cooled CBNM *target based on the Saclay model. The core is enclosed by a uranium crown. The double window through which the beam enters is cooled at the edges. Rated power 4 kW.*



the other hand, it is accepted that, in the same conditions, the neutron yield of a target of 93%-enriched uranium would be 160%. It should be noted here that when a target is of optimum dimensions, *the number of neutrons that it produces is proportional to the electron beam power that it absorbs.*

A homogeneous core

Our aim was to obtain a target which could absorb a beam power of up to 10 kW while keeping the 100% design yield of target *a* (Fig. 7a). This is why we opted at the outset for a target whose natural uranium core had the same dimensions but improved cooling conditions. An increase in the power absorbed causes the temperature to rise, which may destroy the target.

Since it is planned to work towards a core of 93%-enriched uranium in the future, water was straight away ruled out as a liquid coolant because it presents a serious reflection danger for neutrons, which can result in criticality problems. Mercury was the preferred choice since, as already noted, the beam can produce an appreciable amount of neutrons as it slows down.

Even though a tungsten or tantalum core would have been more reliable owing to the greater heat resistance, such a core was not used since the target yield would have been 50-60% lower in neutrons.

We did not want to make a target with a divided core for easier cooling since, in such a case, the coolant fluid takes the place of the uranium and the neutron rate drops, or, if it is to be maintained, the dimensions of the target would have to be increased and it would thus lose the advantage of being a quasi-point source, which is of considerable value to the physicist using the apparatus. Thus, attempts were made to arrive at a homogeneous core working to capacity without unduly increasing the possibility of incidents.

Possible incidents

The quality of the target depends upon its being immune to a certain number of possible incidents, connected mainly with the thermal behaviour of the core and the window.

To ensure that the uranium core does not melt, there should be a good thermal contact between the liquid coolant and the uranium, and at the same time the core should be large enough for the power to be evacuated. The core cladding should not break, either immediately because of differential expansion, or in the long term under the action of thermal cycles, as a result of a change in the structure of the uranium which brings about, by a process that is difficult to predetermine, a gradual and irreversible elongation of the sides of the core.

As for the window, it should be remembered that a beam of some kW dissipates about 100 W in a stainless steel window 0.5 mm thick. The resistance of this window to temperature and structural modifications of the steel under the action of electron bombardment is difficult to determine but of vital importance to the quality of the target.

General trend of development work

We thus attempted to carry out initial improvements to target *a* (Fig. 7a) by developing it along the lines of target *b* (Fig. 7b). The main characteristics of this latter are improved window cooling, uranium stabilisation by adding 10% molybdenum to prevent distortion of the core and the fitting of a uranium crown for improved gamma ray absorption. Thermal contact between the uranium and its cladding is in the form of helium injected at low pressure.

The next step was the design of type *c* (Fig. 7c) with a shorter core than target *b*. The uranium crown was replaced by a cap which almost completely enclosed the core and gave optimum yield. It is known that, for the mercury flow provided by the existing pump, there is a maximum power beyond which the uranium core in which the beam power is absorbed will melt. *The limits imposed on the use of the target are fixed by this maximum power.* The problem is that the core should be so dimensioned that the liquid coolant gets as near as possible to the theoretical hot spot (in this case calculated as 6 mm behind the peak of the beam entry cone) while maintaining a core cooling surface compatible with the number of W/cm^2 that the mercury can evacuate.

There is a great temptation to produce a type *d* (Fig. 7d), in which the core has exactly six radiation lengths and the hot zone of the uranium is completely surrounded by the mercury. There is reason to believe that this gives the best cooling conditions. This extreme solution was not chosen, since the cooling surface of the core would be reduced by some tens of cm^2, which would mean evacuating more

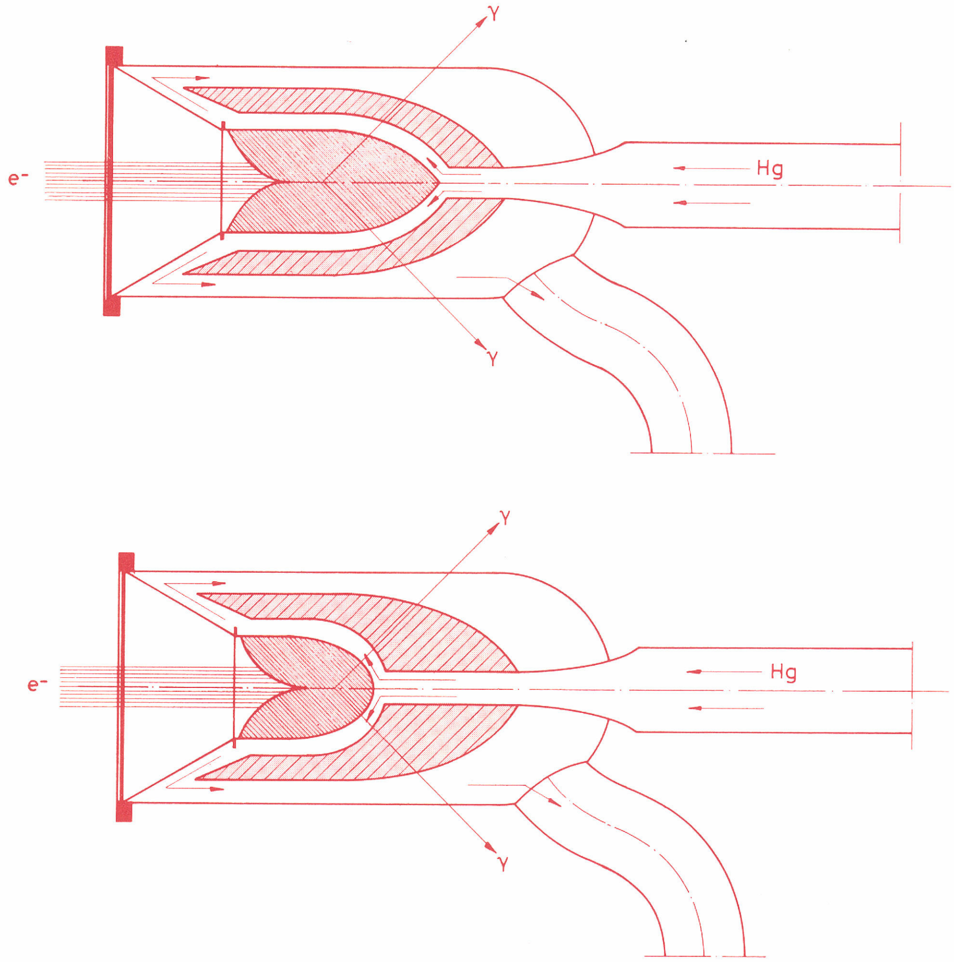


Fig. 7c and 7d: Diagrams showing the layout of the second CBNM *target. The core is almost entirely surrounded by a uranium cap. This layout decreases the "gamma flash" caused by the slowing down of the electrons in the target and distributes the neutron flux almost isotropically around the target without entering the beam axis.*

than 500 W/cm^2 of the surface touching the mercury at 10 kW. To the best of our knowledge, this is virtually impossible. All this led to our producing the type *c* core which lay within more conventional limits. At 5 kW the hot spot has a theoretical temperature of 650° and, by extrapolation, 950° at 8 kW and 1,200° at 10 kW. It will be seen that, as things stand at present, the temperature at 10 kW would be very near to the melting point of uranium-238 with 10% of molybdenum (1,250°) and this was why it was considered unwise to exceed 7-8 kW with the present target. Modification of the cooling circuit may enable 10 kW to be reached without too much trouble.

A stabilised stainless steel double window machined in one piece to avoid welding was used. The space between the windows is filled with 800 g/cm^2 abs. of helium for good heat transfer in the window. It would have been possible to use a window directly cooled by the mercury, but the risk of mercury entering the target chamber and contaminating it for an undetermined length of time if the window should break was too great. Such a solution would also have led to a drop in the neutron yield.

Target monitoring

The target is situated about 60 metres from the accelerator control console. Primarily it can be observed indirectly by a television camera. In the event of failure of the exchanger ventilation system or of the flow from the electromagnetic pump or in the case of a mercury leak, a safety system cuts off the beam from the accelerator. From the control console, it is possible to determine the capacity of the pump, read off the temperature of the uranium core on a digital voltmeter and observe its curve either as a function of the beam power or, perhaps more interesting, of its centring, and plot the inlet and outlet temperatures of the mercury in the target.

Recovery of a damaged target

In order to be able to recover the electromagnetic pump and exchanger unit in the event of an accident in the target itself, this latter is connected to the exchanger cooling pipes by special mercury-tight connections (up to 150° C). This system enables the damaged element to be easily replaced by an identical one in a series of operations. Naturally, these are carried out with due precaution when the unit has been deactivated for a sufficiently long period.

Conclusions

Our first choice led us to a type of target in which the beam is used with a maximum neutron production yield. It is felt that this solution will not permit a beam of more than 10 kW, but since the present accelerator does not have the capacity to work over this figure, it suits our present needs.

When this target has been in use long enough to prove its reliable service under the beam, the present core will be replaced by one of uranium 235. In this way it should be possible to obtain a flux of about $4 \,.\, 10^{13}$ n/sec. 4π with a 10 kW beam. Should we wish to double the characteristics of our accelerator, we would have to adopt another solution which might entail splitting the uranium core for improved cooling. This would decrease the neutron yield of the target unless a solution, at present still on the drawing board, can be worked out for reaching 20 kW without altering the structure of the target. This would be a target cooled by a considerable flow of high-pressure helium in which the uranium would be clad in aluminium similar to a reactor fuel element. Preliminary calculations seem to show that such a solution is well within our grasp. By using this, a high target yield can be maintained and the accelerator would thus be fully exploiting its theoretical possibilities. (EREA-A 7-5)

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The moment of truth for high-temperatur

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THE DEVELOPMENT of high-temperature gas-cooled reactors has been actively pursued in Europe for almost ten years now. The *Dragon* project was inaugurated in 1959 under the auspices of the European Nuclear Energy Agency of the *OECD* and, at about the same time, work began on the German pebble-bed reactor project.

Euratom participated in the *Dragon* project from the very beginning, both for its own benefit and for that of the six member countries of the Community, but it was only some years later, in 1964, that Euratom officially took part in the German project, becoming a signatory of the *Thorium High-Temperature Reactor Association.*

The high-temperature gas reactor constitutes a logical development of the gas-graphite reactor type, which scored its greatest successes in France and Britain. The innovations introduced by the "high-temperature" reactor, as opposed to the earlier gas-graphite reactors, mainly relate to the use of helium as a coolant instead of CO_2 and the adoption of an entirely ceramic fuel of the coated particle type (cf. the article by Claudio Vivante, on pages 46-51).

Economically speaking, the "high-temperature" reactor augurs well for the future. Several studies have already been completed which indicate that, for an initial high-power plant of around 500 or 600 MWe, the cost price per kWh would not exceed 4 mills, i.e. would be approximately the same as that of the present proven-type reactors. Obviously, this cost would decrease as the new type is developed commercially and technically.

> *Thanks to the programmes embarked on in Europe and the United States, high-temperature gas reactors may now be said to have reached a stage where their large-scale industrial application is possible.*

The present situation

Thanks to the programmes embarked on in Europe and the United States, high-temperature gas reactors may now be said to have reached a stage where their large-scale industrial application is possible. The present situation can be summed up as follows:

— *One 20 MWth experimental reactor with prismatic fuel elements (Dragon)* has been built and put into operation. It went critical in August 1964, reached half-power in September 1965 and full-power in May 1966. The first charge was irradiated until September 1966 and the second will be removed during March 1968. In the feed zone of the second charge, a burn-up of about 250,000 MWd/t of metal was reached—one very important result of the project. Start-up, power run-up and on-load operation of the *Dragon* reactor did not present any major difficulties. The operational results have sometimes even exceeded design specifications. For instance, the leaktightness is considerably higher than that specified, the gas temperature at the core outlet has reached 850°C (instead of only 750°C) and the total activity of the primary circuit has not exceeded a few Curies. The fission product release rate has always been about 0.001% and the total concentration of chemical impurities in the primary circuit has always barely come into the sensitivity range of the detection apparatus. By virtue of the *Dragon* reactor project, it has already been possible to acquire a considerable amount of knowledge on the irradiation behaviour of very varied fuels, of both the spherical and prismatic type. Since *Dragon* is the only high-temperature gas reactor in the world with the necessary operational flexibility in this respect, it constitutes a very important test facility.

> *In view of the challenge presented by the American decisions, the commercialisation of high-temperature reactors only has a chance of success within the framework of an international consortium, where every member company is free to develop fully its individual abilities beyond the frontiers of its own country.*

— *The AVR 15 MWe pebble-bed reactor*, which went critical in August 1966 and began its power run-up in September 1967, has already been producing electrical energy for the German grid for several months. Its performance confirms the operational flexibility already noted in *Dragon* and as a result a considerable fund of knowledge is now available on the problems relating to the pebble-bed reactor concept.

— *A 40 MWe prismatic element reactor* has been built at Peach Bottom, in the United States, by *General Atomic.* This reactor went critical on 3 March 1966 and reached full power on 25 May 1967. It has been functioning satisfactorily at powers of up to 45 MWe ever since.

— *The development of coated particle fuels* has made a substantial contribution to the success of high-temperature gas reactors. The *Dragon, AVR* and *Peach Bottom* reactors, which were already under construction before their design had been finalised, have been adapted without any difficulty. Thanks to these fuels, contamination of the

reactors

primary circuit should, under normal conditions, be maintained at a level which permits access to the principal components of the primary circuit for inspection and maintenance purposes.

— *Reference designs* for indirect-cycle power reactors with prismatic and spherical fuels have been produced by *Dragon* (see Fig. 1) and the *Thorium High-Temperature Reactor Association.* Estimates seem to indicate a sufficiently low total initial cost so as to make the new type of reactor very attractive economically. As far as the cost of the fuel cycle is concerned, this is low, not only in the case of the high-enriched uranium/thorium cycle, but also in that of the low-enriched uranium cycle (about 1.1 mills/kWh). It is no surprise that assessments carried out by *Dragon* and the *THTR Association* have led to similar conclusions. The two concepts studied differ in their core structure and in the loading machine used, but the characteristics of the remainder of the plant are basically the same.

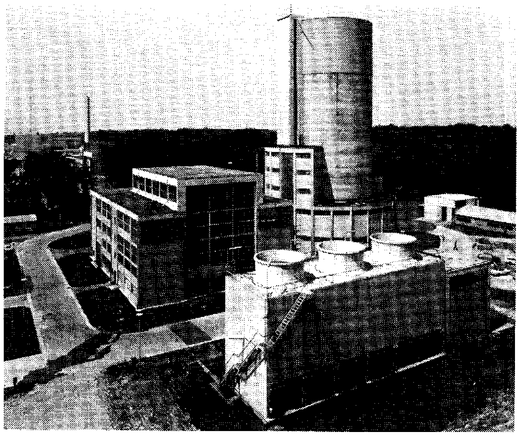
Industrial initiatives

After being invited by the *CEGB (Central Electricity Generating Board)* to submit bids for the Hartlepool power plant, British constructors came up with a tender incorporating high-temperature gas reactors in December 1967. The proposed design only partly exploits the high-temperature gas reactor's potential, but already promises to enable substantial savings to be made in comparison with the *AGR* (Advanced Gas-cooled Reactor).

Within the Community itself, the interest shown by the companies of *Brown Boveri/Krupp (BBK)* and *Gutehoffnungshütte (GHH)* is worthy of mention. Under the *THTR Association*, *BBK* participated in the development of pebble-bed reactors and it is at present drawing up a tender for a 300 MWe power plant. *GHH* is working on a tender for a power plant of equal power, employing prismatic fuel elements, and for some time now it has shown an interest in the construction of high-temperature power plants using direct-cycle gas turbines.

In Switzerland the interest displayed for some time now by *Brown Boveri* (Baden) should be noted. This company has been conducting an investigation into a prismatic core high-temperature reactor with a view to putting it onto the market and has called on the *Dragon* project to act as a consultant.

The interest in this type of reactor manifested in the United States by the *Gulf Oil Co.* must also be mentioned. This company has taken over *General Atomic*, which has been active in this field for a number of years. *General Atomic* undertook to build a 330 MWe nuclear power plant equipped with a high-temperature reactor for a utility company *(Public Services of Colorado)*. This project has the official backing of the *USAEC* under its advanced converter demonstration programme. *Gulf General Atomic* is now offering nuclear power plants equipped with high temperature reactors of 100 MWe unit power, and has made an offer along these lines to the *Washington Public Power Supply System.*



The AVR *nuclear power plant at Jülich, Germany, equipped with a 15 MWe high-temperature gas-cooled pebble-bed reactor.*

The experience gained from both the *Dragon* and the *THTR* projects and from the gas-graphite reactors built has possibly resulted in a technological gap putting America behind Europe in this field. *Gulf*'s recent financial support for the development of the *HTGR* is, however, an important factor which must not be overlooked.

Some figures will clarify the situation. The European Community has, until now, spent in the region of 75 million u.a. on the development of the *HTGR* (to which must be added the 40 million invested by the United Kingdom and other European countries participating in the *Dragon* project); on the other hand, no financial

Table I: Characteristics of HTGR *reactors at present in operation*

Reactor	Dragon	Peach Bottom	AVR
Site	Winfrith (United Kingdom)	Peach Bottom (United States)	Jülich (Germany)
Power MWth	20	115	46
MWe	—	40	15
Power density MW/m³	14	8.3	2.2
Coolant	He	He	He
Coolant pressure (atm)	20	25	10
Coolant outlet temperature (°C)	750-850	750	850
Moderator	graphite	graphite	graphite
Fuel			
cycle	U/Th	U/Th	U/Th
form	coated particles	coated particles	coated particles
elements	prismatic	prismatic	balls
Pressure vessel	steel	steel	steel
Criticality	1964	1966	1966

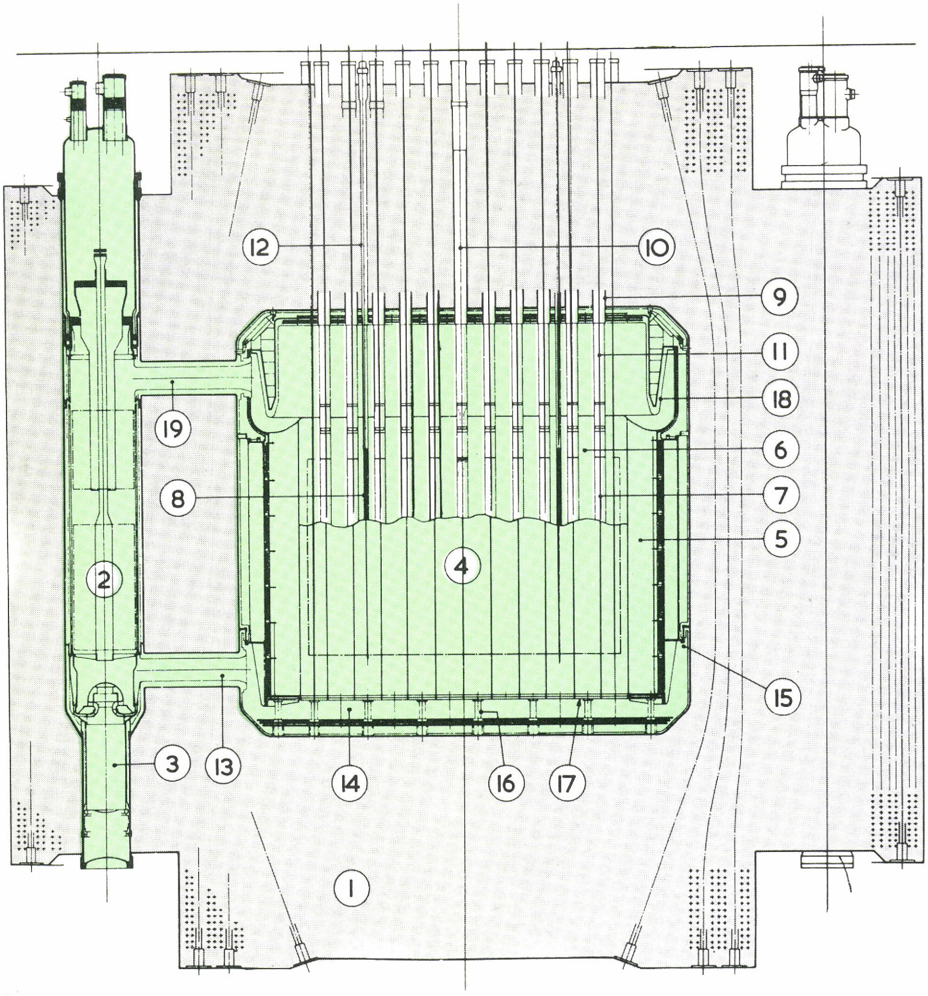


Fig. 1: General arrangement of 528 MWe low enrichment high-temperature gas-cooled reactor of the Dragon *type.*
1 Pressure vessel; 2 Boiler; 3 Circulator; 4 Core; 5 Radial reflector; 6 Moderator columns; 7 Fuel elements; 8 Control rod; 9 Refuelling stand pipes; 10 Stand pipe plug; 11 Stand pipe plug extensions; 12 Control stand pipe; 13 Inlet duct; 14 Under-core plenum; 15 Under-core plenum seal; 16 Stand-offs; 17 Core mountings; 18 Suspended shielding; 19 Hot outlet duct.

commitment has yet been undertaken as regards research or the commencement of industrial operation. The United States, for their part, have by now invested a sum in the region of 85 million u.a. in high-temperature reactors, i.e. somewhat less than that invested in Europe, but *they are now contemplating appropriations of more than 90 million* for the construction of the 330 MWe prototype mentioned above and for the research and development programme associated with the project.

It is, therefore, a fact that the Community, although ahead in the purely technological field, is lagging behind as regards the commercial exploitation of this hardware. Here we have a situation the study of which should perhaps be recommended to those who seek a diagnosis and cure for this typically European malady, which, very inappropriately, has been called the "technological gap".

A few months ago, however, a glimmer of hope emerged. Constructors in the various countries of the Community and Britain agreed to negotiate for the creation of a joint company with a view to putting high-temperature reactors on an industrial footing. It can only be a cause for rejoicing to see that the fruitful co-operation initiated by the European public authorities has been followed up by collaboration between industrialists in various countries. In view of the challenge presented by the American decisions, the commercialisation of high-temperature reactors only has a chance of success within the framework of an international consortium, where every member company is free to develop fully its individual abilities beyond the frontiers of its own country.

Supply problems with regard to fuel . . .

Several possible fuel cycles have been studied. Although, in the long run, the natural resources can best be utilised by employing the U^{235}/Th cycle, satisfactory short-term solutions, with total costs of about 1.1 mill/kWh are obtained with the Pu/Th and Pu/U^{238} cycles, as also the low-enriched uranium cycle (3-5%). Assuming that the price of natural uranium increases, one advantage of the thorium cycle with recycling is that, thanks to the high conversion rates possible with it, a cost is arrived at which is only dependent on the price of uranium to a limited extent.

The choice between the different cycles is dictated by the price of plutonium and the availability of 93%-enriched uranium. It is, however, probable that, owing to conditions in the supply and reprocessing fields, a reactor based on the low-enrichment concept is at present preferable, as it calls neither for very-highly-enriched uranium nor for the building of special reprocessing plants, different from those required for water-cooled reactors, for example. The work involved in isotope separation per MW installed and per kWh produced is substantially less in the case of the high-temperature reactors than in the other thermal reactors fuelled on enriched uranium. This fact can only favour the high-temperature reactors as opposed to light-water reactors.

. . . and helium

As far as the supply of helium is concerned, considerable quantities can be extracted from the reserves of natural gas at a cost which is low enough to ensure that the

It must be stressed that high-temperature gas-cooled reactors still possess a very important development potential, notably by the adoption of thorium fuel cycles and the introduction of gas turbines for the conversion of energy instead of steam turbines.

price does not have a major effect on the total cost of the energy produced. Taking a helium cost of 5 u.a./m^3 as a reference, the amount of helium in the primary circuit could be estimated as corresponding to an investment of 0.2 u.a./kWe; taking into account a realistic leak-rate (0.1% of the contents of the primary circuit per day), the cost of the helium lost would be only about 0.01 mill/kWh.

High-temperature reactors compared with the proven-types and fast reactors

For the European industry, the high-temperature reactors represent a means of facing up to the competition of the "proven" reactors developed on the basis of the American types. It must be stressed that they still possess a very important development potential, notably by the adoption of thorium fuel cycles and the introduction of gas turbines for the conversion of energy instead of steam turbines (cf. the article by E. Böhm, pp. 52-58). This should facilitate a considerable reduction in costs and improve the utilisation of the available fuel resources. The high-temperature type will not, therefore, become obsolete quickly. There are several reasons to believe that the operational flexibility of these reactors, combined with the low investment in fissile materials, will enable this type to operate alongside fast breeders in a spirit of mutual coexistence (for a discussion of this question, see *Euratom Bulletin*, 1967, No. 1, pages 2-5, "Long life for converters").

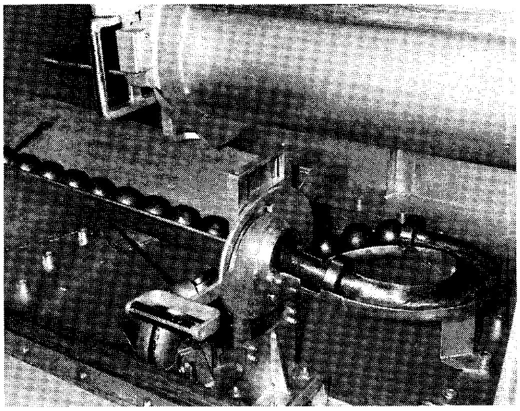
Should the public sector provide backing?

The first industrial projects will most probably require limited support from the public sector, since these reactors can draw on the technology which has already been developed and proved for the gas-graphite reactors. The experience which will be acquired on the behaviour of irradiated graphite at high temperature and large doses, as also on the behaviour of fuel and various components, will enable modifications to be factored into the design of successive reactors, so as to lead gradually to an improvement of the high-temperature steam-cycle reactor. To aid this development, the investment of public funds is essential in order to ensure operation of the plants required for the irradiation experiments and to co-ordinate research and assessment programmes devoted to, in particular, graphite and fuel fabrication and reprocessing.

The *HTGR* power plants—regardless of whether the core is designed for prismatic or spherical elements—ought to undergo a marked improvement, as has already been seen, by the introduction of gas turbines. This is an important step to be taken, and one involving the adoption of new ideas which could have repercussions on the overall design of power plants, including the nuclear installation. Public backing could be very useful for speeding up this progress by promoting research programmes and the construction of the necessary prototypes.

Applications other than the production of electrical energy

Only the use of a high-temperature gas reactor for the production of electricity has been discussed in this article. By way of a conclusion, it should be pointed out that this type of reactor also merits consideration as an industrial source of heat. Its application in the iron and steel industry, for example, is a possibility already being studied (cf. *Euratom Bulletin*, 1967, No. 4, pages 115-120).



Fuel elements being fed into the AVR *charging machine.*

It is difficult at this stage to list all the possible applications and the industrial interest and support from public funds which may be required. It is nevertheless a recognised fact that this is an important field of activity which merits particular attention in future programmes. (EREA-A 7-6)

Bibliography (1) Papers presented at the symposium held by the British Nuclear Energy Society on high temperature reactors and the Dragon Project, London, 23 and 24 May 1966 —published in the *Journal of the BNES*, Vol. 5 (1966) No. 3 (July) pp. 235-461. (2) European Nuclear Energy Agency, OECD. *Eighth Annual Report 1966-1967*—OECD High Temperature Reactor Project Dragon.

Installation of reactor internals at Peach Bottom, *Pennsylvania, U.S.A. The* Peach Bottom *40 MWe high temperature gas-cooled nuclear power plant is now in commercial operation.*

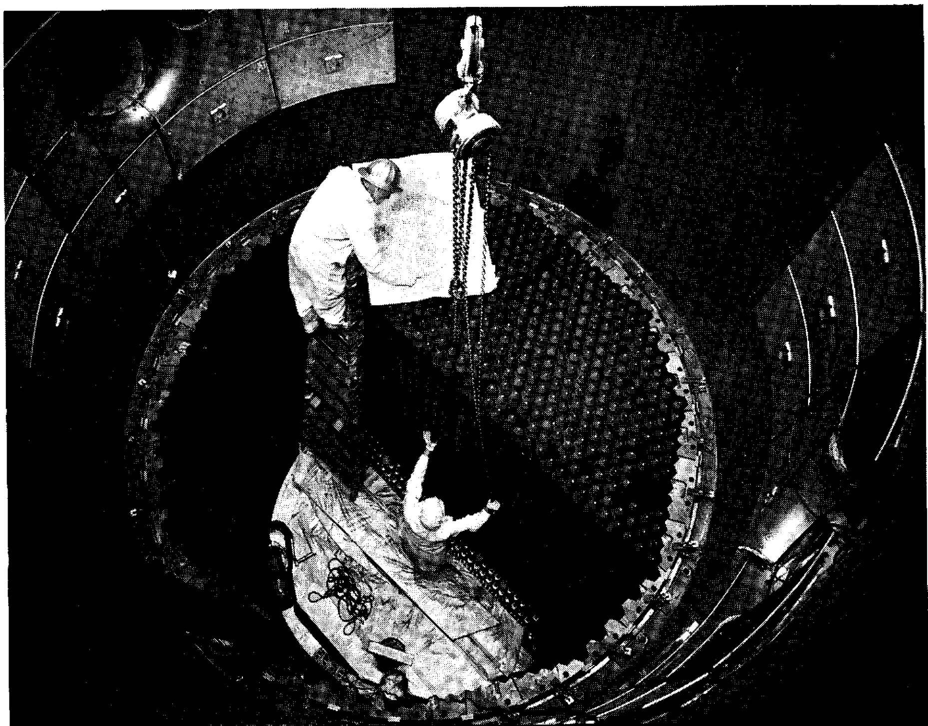
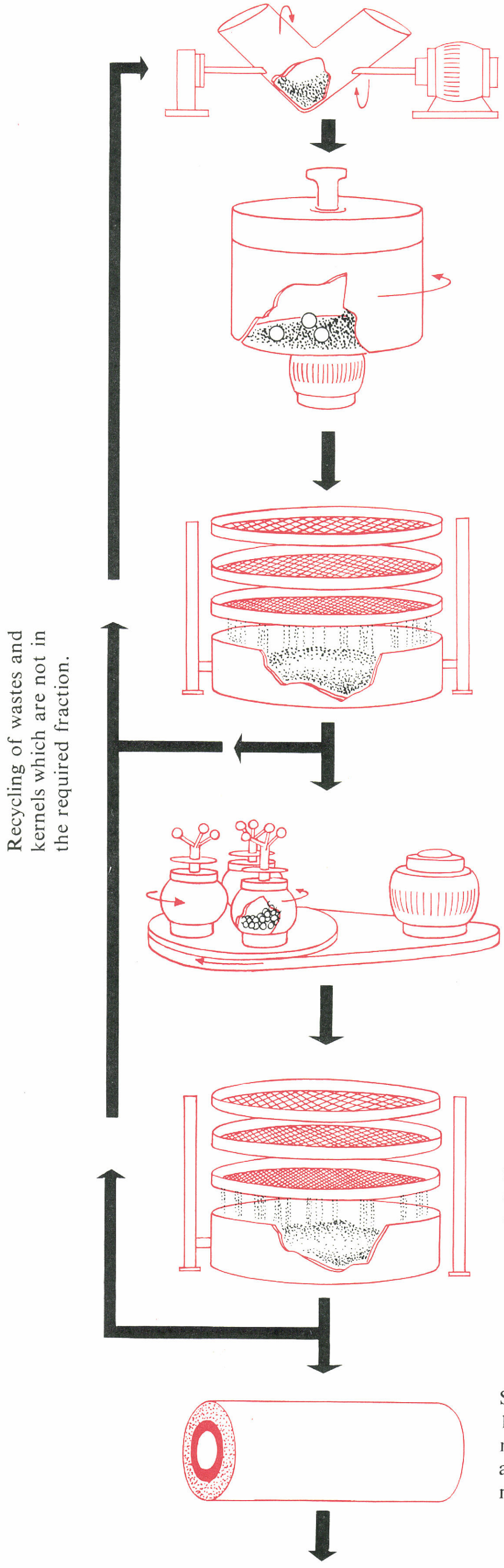


Fig. 1: Fabrication of UO_2/ThO_2 kernels by the powder metallurgy process.



Mixture of $UO_2 + ThO_2$ powders (30 microns approx.) with 4 wt% of paraffin.

The damp powders are placed in a porcelain bowl with three steel balls and are rotated for one or two minutes. A very hard product is obtained which can be granulated.

Granulation through screens with meshes of 353-1,000 microns.

Spheroidisation in a planetary crusher at a speed of 300-3,500 rpm.

Screening of the spherical kernels to select the required diameter.

Sintering at 1,650-1,850°C in a graphite resistance furnace under an inert CO_2 or nitrogen atmosphere.

Sintered kernels ready for fluidisation

The following sentence appears at the beginning of the Eighth Annual Report on the Dragon project, published in October 1967: "In particular, the continuing successful development of the coated fuel particle concept has made this reactor system (Dragon) not only technically and economi-

Coated fue

ONE OF THE most promising concepts in nuclear fuel technology is that of "coated particles", which have been studied over the last ten years in several European and American laboratories.

These little spherical fuel particles coated with ceramic materials have many interesting properties, one of the most important being their remarkable ability to retain fission products.

This leaktightness suggests their use particularly in high-temperature reactors, in which conventional metal cans cannot be used to contain the fission products, as they would melt or be near to melting in the hottest regions. For this reason, since the early stages of high-temperature reactor development it was planned to use a fuel dispersed in graphite, which is a heat-resistant material. Research workers were at first reconciled to the need for large units for the continuous purification of the cooling gas, but owing to the success of the work on coated particles it has been found that fairly small installations will do the job. This has been proved during the operation of the *Dragon* experimental

cally attractive but also one of the safest”. The same could certainly be said of the two other types of high-temperature reactor now being developed in Germany and the USA. What exactly are these coated particles?

That is what this article will try to explain.

particles for high-temperature reactors

CLAUDIO VIVANTE, *Euratom*

reactor, the fission product release rate being found to be about one-thousandth of one per cent.

At present all the high-temperature reactors in operation (*Dragon*, *Peach Bottom* and *AVR*) use a mixed uranium and thorium carbide fuel in the form of spherical particles (diameter 0.2 to 0.8 mm) covered by layers of pyrolytic carbon (thickness 0.05 to 0.15 mm), or composite layers of pyrolytic carbon and silicon carbide.

How are these particles fabricated and how do they behave when irradiated?

Fabrication of the fuel kernels. . .

Two main methods are used to fabricate the fuel kernels of the particles, namely, powder metallurgy and a chemical process known as “sol-gel”.

. . . by powder metallurgy. . .

The powder metallurgy method, as used with mixed uranium and thorium oxides, is described in more detail in Fig. 1. Briefly, the UO_2 and ThO_2 powders are blended in the desired ratio, a binder is added and they are mixed together. A very hard product is obtained which can be granulated through sieves with meshes ranging from 350 to 1,000 microns, depending on the diameter desired for the kernels. The granules are then given a spherical shape and finally sintered.

The density of the kernels varies directly with the temperature chosen for sintering. The resultant possibility of regulating the porosity of the kernels (up to 50% of the maximum theoretical density) is one of the advantages of the powder metallurgy method, since, owing to the fact that fission products can accumulate in the pores, the kernels can be prevented from expanding under irradiation.

If on the other hand it is desired to obtain kernels with a density very close to the maximum theoretical density, the kernels (this applies to the carbides) are blended with a carbon powder before sintering, and then raised to a temperature of 2,500 to 2,600° C, i.e. slightly beyond their melting point, in a furnace with an inert atmosphere.

. . . or by the sol-gel process

The sol-gel process (see Fig. 2) consists in obtaining a hydrosol, a stable suspension of an oxide (ThO_2, UO_2 or PuO_2 in the case with which we are concerned, or a mixture of two or three of them) in water. The particles in suspension, which are obtained by hydrolysis of the corresponding nitrates, are very small (30 to 100 Å).

In order to form the spherical particles, the sol is dispersed in droplet form in a liquid which will not mix with it and which reacts with it to cause “gelation”, i.e. in this case dehydration of the drops.

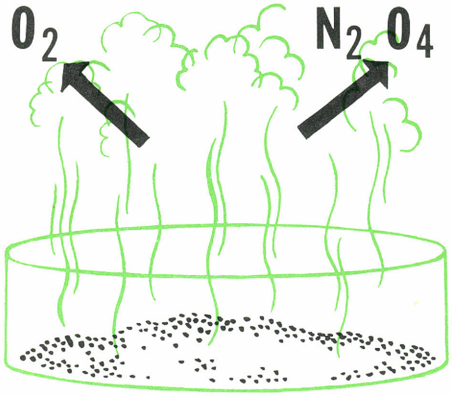
The particles thus obtained, whose diameter can be made to vary between 10 and 1,000 microns, look very like caviar once they have been separated from the gel solution. They are then dried and sintered in the same way as the particles obtained by the powder metallurgy process.

One of the advantages of the sol-gel process as compared with powder metallurgy is that it makes it possible to obtain high densities while operating at relatively low temperatures (1,150° C for thorium oxide instead of 1,650° C with the other process). Secondly the particles have a higher mechanical resistance (about 6 kg as against 3 kg per particle) and their kernels are more homogeneous. Lastly, since the sol-gel process consists essentially in the handling of liquids, it is easier to conduct operations by remote control, and this is very important if it is desired to fabricate plutonium-base fuel or to reprocess irradiated fuel.

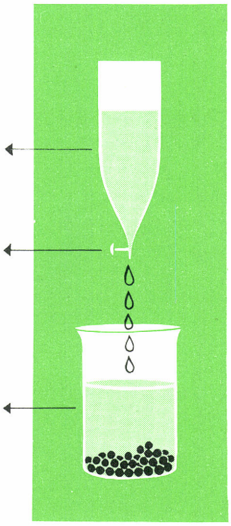
The coatings

Once the fuel kernels have been formed and sintered, they must be provided with a leaktight coating. The best known and most highly developed technique consists in coating the particles with a layer of “pyrocarbon”, produced by the pyrolysis of hydrocarbons (usually methane) on a surface raised to a temperature of 1,400-2,200° C. This high-temperature decomposition of a hydrocarbon placed in contact with the fuel kernels is performed in a “fluidising furnace” (Fig. 3).

By varying the treatment it is possible to



$Th(NO_3)_4 \rightarrow ThO_2 + 2\,N_2O_4 + O_2$
(theoretical reaction)



Sol—
$ThO_2 + 2H_2O \rightarrow Th(OH)_4$

The dimensions of this capillary determine the diameter of the particles.

Gel—Gelation and formation of the particles by solvent extraction—
$Th(OH)_4 \rightarrow ThO_2 + 2H_2O$

Fig. 2: The sol-gel process (as used in the fabrication of nuclear fuel).
A solution of nitrates of thorium, uranium or plutonium is used to produce oxides or carbides. The preparation of thorium oxide, for example, is effected in the following four stages:
1) drying and denitration of the thorium nitrate in order to produce a reactive oxide;
2) dispersion of the thorium oxide in order to form a hydrosol;
3) gelation by evaporation or by solvent extraction;
4) the particles thus formed are separated from the liquid.
In order to obtain the fuel in its final ceramic form, the last stage is the sintering of the particles.

obtain pyrocarbon deposits with a wide range of densities and crystalline structures. Under the microscope two main types can be distinguished: the first is of basaltic aspect and consists of conical grains (Fig. 4); the second has flat microstructure and is known as "laminar" (Fig. 5). The density of these deposits may range from 1.5 to 2.2 g/cm^3, but they are always extremely impermeable to gases, their helium diffusion coefficient, for example, being only 10^{-12} cm/sec.

Although pyrocarbon is entirely satisfactory as regards leaktightness, other solutions are under study. In the USA, for example, a study is now being carried out of the properties of heat-resistant oxides such as alumina, together with the methods of producing them.

Fission gases raise major problems, but it is also necessary to take into account the diffusion of *solid* fission products, including thorium and uranium. Through pyrocarbon this diffusion is relatively fast. The *Dragon* project has therefore conducted a parallel study of silicon carbide coatings, which have turned out to be much superior in this respect. Fig. 6 shows a silicon carbide coating sandwiched between two layers of pyrolytic carbon.

Examination before irradiation

During the development work coated particles were, of course, subjected to all kinds of examinations, including irradiation tests. A summarised account is given below, beginning with the pre-irradiatory examination.

This is performed by all the laboratories producing coated particles, but we shall concentrate here more particularly on the particles tested under the *Dragon* project.

The fuel kernels are analysed before being coated, and their porosity is evaluated. After coating, the thickness of the deposit is measured by microradiography and its cohesion and structure are checked by metallography. Its density is also measured by slightly crushing some particles and dissolving their kernels in an acid.

The surface contamination of the particles is then assessed. It has been shown that the emission of fission products by healthy particles is governed by the quantity of fissile material present on their surface; it is thus imperative to reduce this contamination to a minimum. α- counting constitutes a simple and rapid method of evaluation which yields preliminary data. It is followed, however, by a systematic activation analysis, a method admittedly more expensive, but at the same time more accurate.

In addition to a thorough metallographic examination, the coated particles are subjected to a chemical analysis aimed at the accurate determination of the weight of uranium which they contain; their density and crush resistance are also measured.

Irradiation of the particles

The performance of the coated-particle fuel has been studied by various devices, such as sealed capsules, which yielded information on the mechanical behaviour of different batches of particles in varying and known temperature and irradiation conditions, capsules flushed by an inert gas, making it possible to evaluate the retention characteristics of a given type of fuel, and irradiation loops supplying information on the behaviour of full-size cartridges.

Samples have been irradiated in the Studavik (Sweden), Risø (Denmark), *Pluto* (England), Julich (Germany), *ORR* (USA) and *Triga* (USA) reactors. However, the most important tests have been or are being made in the *Dragon* and *AVR* reactors on fuel which is placed in elements having either the geometry of these reactors or geometries which are conceivable for a high-temperature power reactor.

Lastly, tests have been performed in the *HFR* Petten reactor and the *DFR* Dounreay reactor with the aim of studying the behaviour of particles under high fast-neutron fluxes.

These various irradiation tests were carried out at temperatures ranging from 1,000 to 1,800°C and at burn-ups of up to 48%. An attempt is made to summarise the results briefly below.

The effects of irradiation on the kernels

Under the influence of temperature and irradiation, the kernels undergo various transformations. It might be expected, for example, that the fuel kernels would swell owing to the accumulation of fission products.

While this is true of melted kernels and also to some extent of cores produced by the sol-gel method, which often have a density near to the theoretical value, the same cannot be said of sintered particles, which undergo a kind of further sintering in the reactor, causing them to shrink.

This phenomenon has been observed even at fairly low temperatures and has been proved to be particularly pronounced with uranium carbide kernels. Not only does a certain shrinkage occur, but the pores also tend to fill up with free carbon.

. . . and on the coatings

During irradiation the inner layer of the coating is bombarded by fission products and neutrons due to the reactions which take place inside the kernel. This bombardment may lead to a crystalline reorganisation of the pyrocarbon, accompanied by shrinkage. The graphitising action of the uranium may also help to increase the density of the carbon. As a result radial cracks occur, which then spread near the kernel and cause a characteristic "spear-head" attack, which may culminate in a total fracture (Fig. 7). The spear-head attack generally takes place at temperatures of below 1,450° C. Above this temperature another phenomenon occurs, namely the diffusion of the uranium into the coating, as a result of which the particle will end up by behaving as if it were not coated at all. No satisfactory method of countering this has yet been found.

On the other hand, many methods have been suggested for preventing spear-head attack, which occurs in particles subjected to very high burn-ups; two of them have given particularly satisfactory results. The first consists in placing a layer of silicon carbide at a certain distance from the kernel between two layers of pyrocarbon (Fig. 4). The other method may almost be termed homeopathic, since it consists in the application of a suitable heat treatment before irradiation to cause the uranium to diffuse into the first few initial microns of the coating, thus causing the

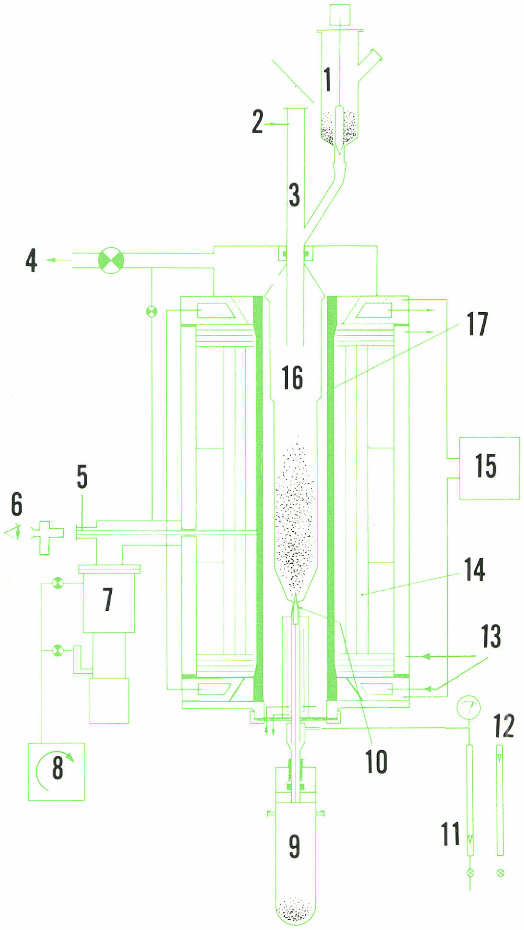


Fig. 3: Schematic diagram of a laboratory-scale fluidising furnace.

1 Feeding container; 2 Argon purge; 3 Feeding tube; 4 Exhaust; 5 Argon purge; 6 Pyrometer; 7 Diffusion pump; 8 Vacuum pump; 9 Container for coated particles; 10 Nozzle system; 11 Methane; 12 Argon; 13 Water cooling; 14 Radiation shields; 15 Transformer 0-25 V, 30 kW; 16 Fluidising vessel; 17 Graphite heater.

Fig. 4: Basaltic coating deposited at 1,700°C on an initial laminar layer —(ThU) C_2 kernel.

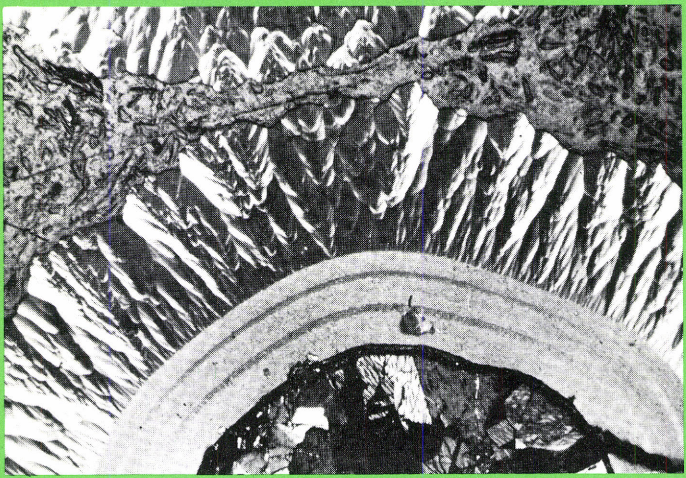


Fig. 5: Laminar coating deposited at 1,500°C—(ThU) C_2 kernel.

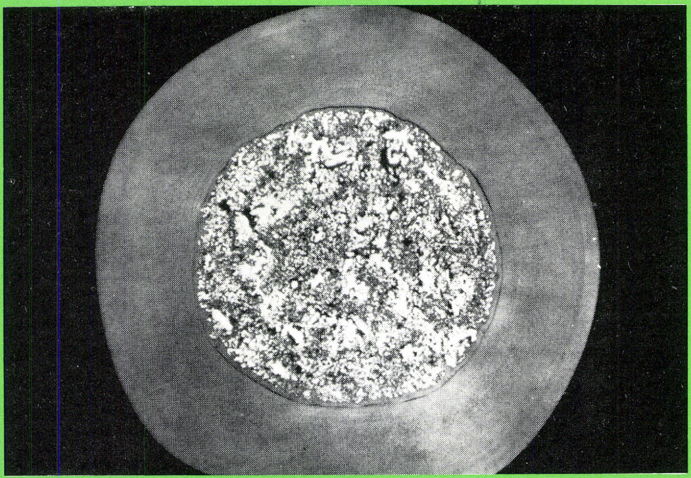


Fig. 8: Emitted fraction of the fission gases (R/B—see Table I) as a function of their period for a typical irradiation test on coated particles. ▶

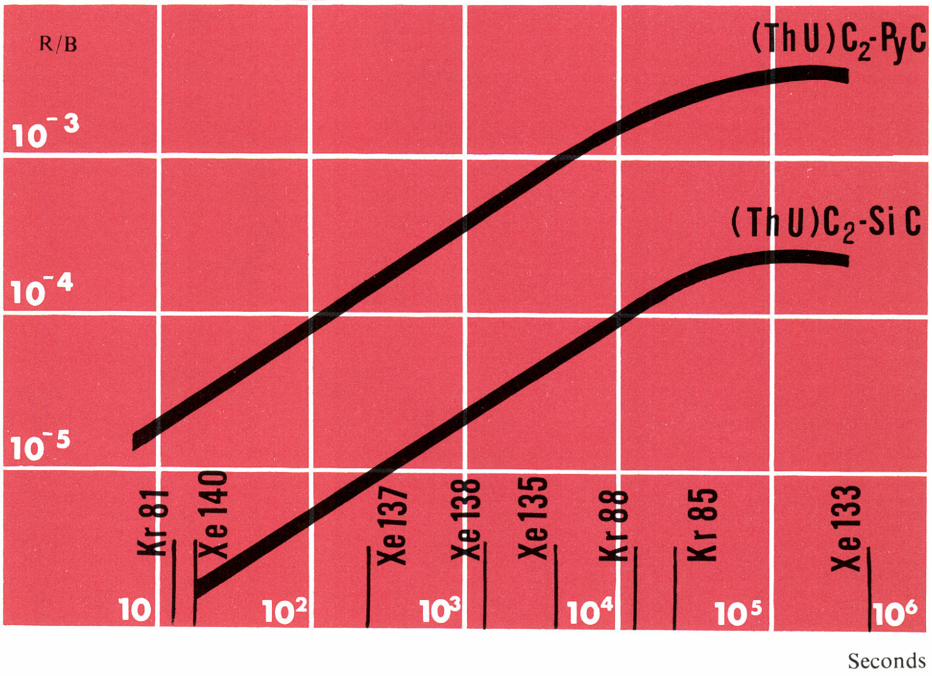


Fig. 6: Layer of silicon carbide sandwiched between two layers of pyrolytic carbon—(ThU)C_2 kernel.

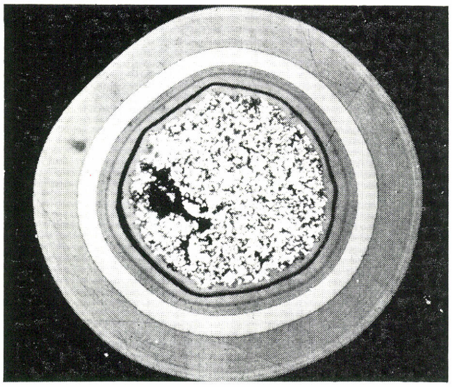
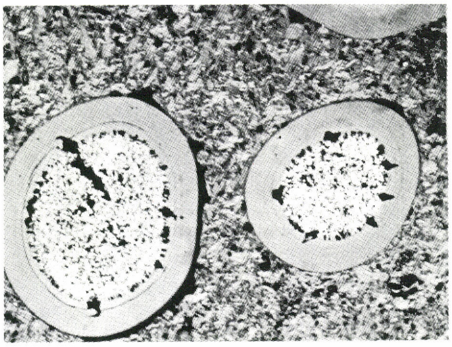


Fig. 7: Particles irradiated at 1,200°C and coated with a layer of pyrocarbon. Note the characteristic "spear-head" attack—(ZrU)C kernel.



immediate graphitisation of the pyrocarbon; it is the uniformity of the layer thus formed which doubtless makes it resistant to spear-head attack.

The effect of fast neutrons

Fast neutrons have relatively little effect on silicon carbide, but a big effect on pyrocarbons. They particularly influence the density, isotropy and size of the pyrocarbon crystallites.

The tests have shown that in order for a pyrocarbon coating to be stable when exposed to a fast neutron flux, it must have a high density (2.1 g/cm^3), low isotropy and a fairly large crystallite size (L_c = 200 Å).

Release of fission products

During the tests particular attention was, of course, paid to the fission product release rates. It was found that with healthy particles retentivity is at first proportional to the thickness of the coating, indicating that the phenomenon is linked to diffusion. Retentivity also depends on the surface contamination of the particles, which governs the lower limit of fission product emission.

Other factors also play a part, such as the size of the kernels. For cores of between 450 and 600 microns, for example, the optimum thickness of the coating is about 120 microns.

We have already seen that it is possible to obtain good fission gas retention factors with either pyrocarbon or silicon carbide coatings. However, the factors for silicon carbide are at least 100 times better than those for pyrocarbon. This is shown in Fig. 8, which represents the fraction of fission gases emitted as a function of their period for a typical irradiation test.

The solid fission products which diffuse most rapidly are barium, caesium, strontium and yttrium. Radiochemical analyses have shown that these products scarcely move along the coolant circuit and usually settle in the graphite parts near the fuel from which they came. Here again the retention properties of silicon carbide proved to be much superior to those of pyrolytic carbon, as shown in Table 1.

Conclusion

The systematic irradiation tests on this type of fuel made it possible to study a multitude of variables and to reveal the remarkable properties of coated particles. Particles irradiated up to burn-ups of

300,000 MWd/t at temperatures of more than 1,400° C released only a thousandth of one per cent of the fission products formed in them. Certain unsolved problems still remain, especially if it proves necessary to have particles capable of continuously withstanding temperatures of more than 1,450° C. It can nonetheless be stated that the technology of coated particles is already sufficiently advanced to make it possible to produce fuel elements for a high-temperature prototype with a minimum of risk. (EREA-A 7-7)

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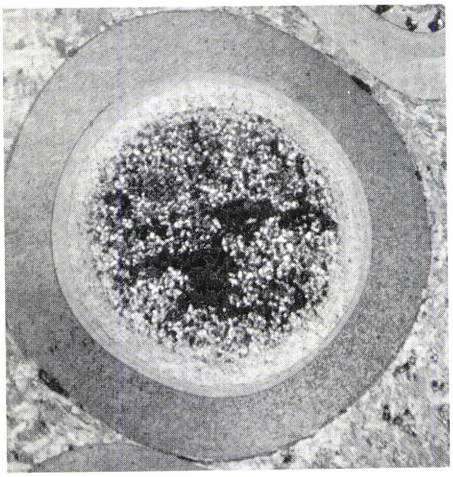


Fig. 9: Irradiated particle covered with two layers of pyrocarbon—(ThU)C kernel.

Fig. 10: Irradiated particle covered with SiC sandwiched between two layers of pyrocarbon —(ThU)C_2 kernel. ▼

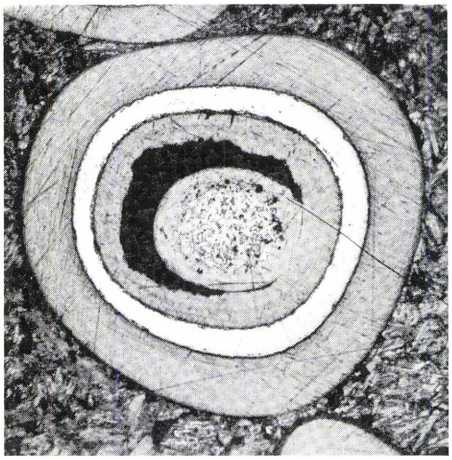


Table I: Release of solid fission products in certain irradiation tests under the Dragon *project.*

Test	Max. irrad. temp. °C	Irrad. rates % (FIMA)[1]	Fractional release (R/B)[2]				
			Cs^{137}	Sr^{89}	Ba^{140}	Ce^{141}	Zr^{95}
Particles coated with PyC							
HPD 10	1,300	4	5×10^{-2}	3×10^{-2}	3×10^{-2}	3×10^{-4}	
Studsvik 6 C	1,420	7.8	1×10^{-3}	6×10^{-2}	2×10^{-4}	1×10^{-4}	
Particles coated with SiC							
HPD 13	1,300	5	4×10^{-5}	1×10^{-5}	—	9×10^{-5}	5×10^{-5}
Studsvik 6 B	1,410	5.6	4×10^{-4}	7×10^{-5}	2×10^{-6}	2×10^{-5}	3×10^{-5}

1. "Fissions per Initial Metal Atoms"—ratio between the number of fissions which occur and the number of metal atoms initially present.

2. During a given period of time, the ratio between the quantity of fission products released and the quantity formed in the fuel (Release/Birth).

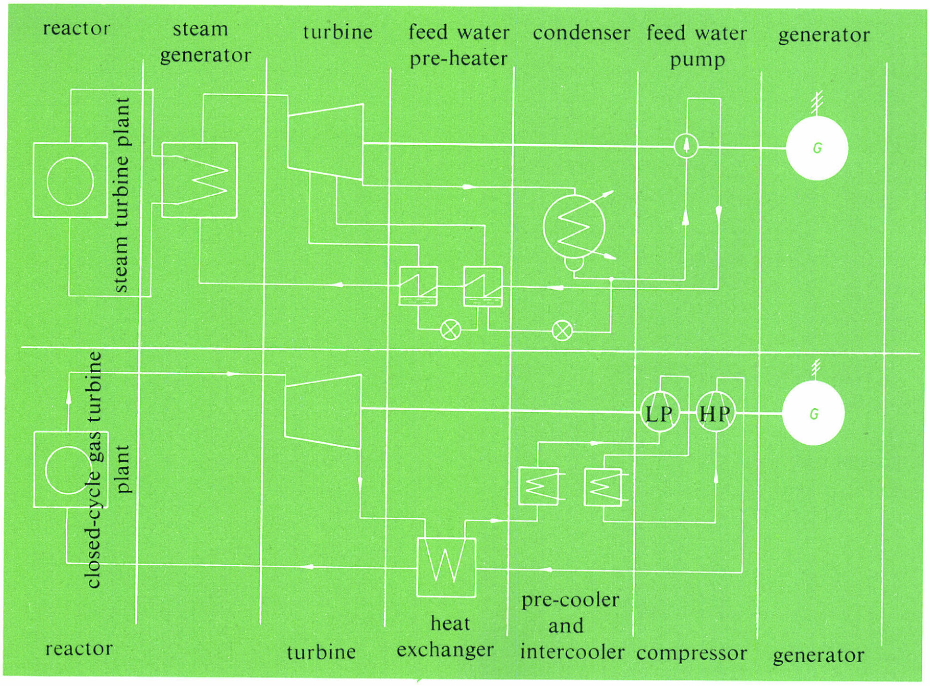


Fig. 1: Comparison of the diagrams of a steam turbine power plant and a helium turbine power plant.

Gas turbines for nuclear power plants

EDGAR BÖHM, *GHH—Gutehoffnungshütte, Oberhausen (Rhineland)*

ADVANTAGES ARE offered by incorporating the high-temperature reactor *(HTR)* in a closed gas turbine cycle since a particularly simple cycle operation is made possible by the direct coupling of the reactor and the turbine and, in addition, the coolant temperature level of a helium-cooled high-temperature reactor meets the requirements of a gas turbine. The extensive experience at present available as a result of the successful operation of both the *Dragon* test reactor and the 40 MWe *Peach Bottom* nuclear power plant in the field of helium-cooled high-temperature reactors can be put to good use in coupling the HTR with a helium turbine.

Unlike the *HTR* using a steam turbine, the closed gas turbine cycle obviates double heat transfer between the reactor and the turbine. This not only enables the steam generator to be dispensed with, but also ensures that better use is made of the temperature gradient between the reactor and the environment.

Open gas turbines cannot be used in nuclear power plants because radioactive fission products released in the event of fuel element damage can escape directly into the atmosphere, the air in the reactor is comparatively speaking highly active and air reacts severely with the graphite fuel element cladding at high temperatures.

The use of air as the working medium in nuclear closed-cycle gas turbine plants is ruled out for the reasons just given. Helium would appear to be the best choice for the high-temperature reactor, since this does not react with the graphite moderator even at high temperatures and to all intents and purposes cannot be activated.

Development work is centred particularly on questions relating to the coupling of the *HTR* to the helium turbine. At the same time, these investigations are making an important contribution to the development of nuclear power plants using helium-cooled fast breeders.

Development of the helium turbine

In recent years the development of closed-cycle gas turbines has reached a technical level comparable to that of steam turbines. The fundamental investigations into the closed circuit gas turbine process were carried out in particular by J. Ackeret and C. Keller (since 1935) and K. Bammert (since 1943). The aim of this work was to produce a high-grade heat engine of simple design without the use of high pressures, in which the working medium stays clean, so that the efficiency of the flow equipment and the heat exchange apparatus does not change during the operating lifetime. As far as the efficiency of closed-cycle gas turbines is concerned, such a plant can clearly be compared with a steam turbine, also of the closed-cycle type. Gas has an advantage over steam in that high temperatures, and thus high process efficiency, can be reached at low pressures. The simplified circuit diagrams of a steam plant and a closed-cycle gas turbine plant are given as an illustration of this in Fig. 1.

In the last ten years, a variety of closed-cycle gas turbine plants fired with coal-dust, gas or oil have been built and operated successfully. Conventional gas turbine plants have already been operating for more than 50,000 hours in Germany, Japan and the Soviet Union. The plants in Germany powered by closed-cycle gas turbines and built by the *Gutehoffnungshütte* are given in Table I. Fig. 2 shows a cutaway view of the 14.3 MWe power plant at Oberhausen.

Table I: Closed-cycle gas turbine plants constructed in Germany by GHH.

Location	Power MW	Temperature upstream of the turbine	Fuel	Commis-sioned	Operating hours up to 31 December 1967
Ravensburg	2.3	660	pit coal	1956	69,000
Coburg	6.6	660	pit coal	1961	45,380
Oberhausen	14,3	710	high-ash fines	1960	46,700
Haus Aden	6.4	680	natural gas/coal	1963	36,970
Gelsenkirchen	17.25	710	waste gas/oil	1967	520
Geesthacht II	24	750	nuclear	1971/72	

The helium turbine in the nuclear power plant

The simple layout in a nuclear power plant with a high-temperature reactor and a helium turbine can be seen from the circuit diagram in Fig. 3. The gas is heated in the reactor to over 700°C and let to the turbine, where it gives off energy to the blading. It then passes through the heat exchanger and is then cooled down to the compressor inlet temperature in the pre-cooler. After being compressed in the low-pressure compressor it is re-cooled in intercooler 1 and compressed in the intermediate pressure unit. This is followed by further re-cooling in intercooler 2, after which the gas is finally compressed to the final pressure in the high-pressure compressor. In the heat exchanger it picks up the remaining usable heat from the low-pressure gas that has expanded in the turbine and then passes through the reactor, where it is heated to the final temperature.

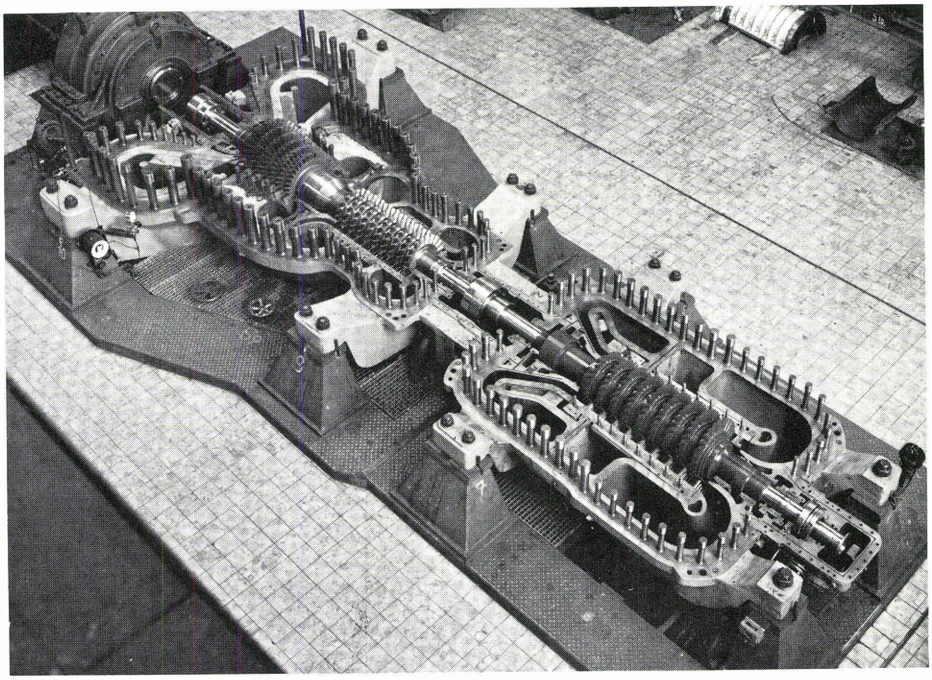
It is advantageous for the whole process if the quantity of mechanical energy expended during compression is small. According to Fig. 3, the compression process using two intercoolings constitutes a stepwise approximation to the theoretically most favourable case of isothermal compression. More than two intercoolings entail design complications which for the most part are not offset by the thermodynamic gain.

In theory, the working medium in the turbine should also be isothermally expanded. As with compression, gradual approximation is also possible by repeated heating, but, because of the high temperature range during expansion and the high gas volumes, this would entail design complications and was thus abandoned.

For the gas temperatures reached today in a high-temperature reactor (from 800°C onwards), the advantages of directly coupling the high-temperature reactor to a helium turbine lie not only in the increased efficiency but also in the fact that such a system results in simplification and reduction in size of the plant, elimination of the steam generators and, in comparison with steam turbines, a small gas turboset. From the safety angle, the suppression of the high-pressure steam cycle also represents an advantage.

Fig. 4 shows the net efficiencies that can be obtained if the turbine inlet temperature for a high-temperature reactor is raised from the present 800 to 1,000°C. The values are those for a 300 MWe plant with double intercooling and a pressure upstream of the turbine of 60 atm. Conservative empirical values were used for determining the individual efficiency ratings of the turbine and the compressors, the pressure drop in the circuit and the capacity of the regenerative heat exchanger. It can be seen

Fig. 2: Cutaway view of the Oberhausen closed-cycle gas turbine. Bottom right, the low pressure compressor, top left, the high pressure casing with turbine and high pressure compressor.



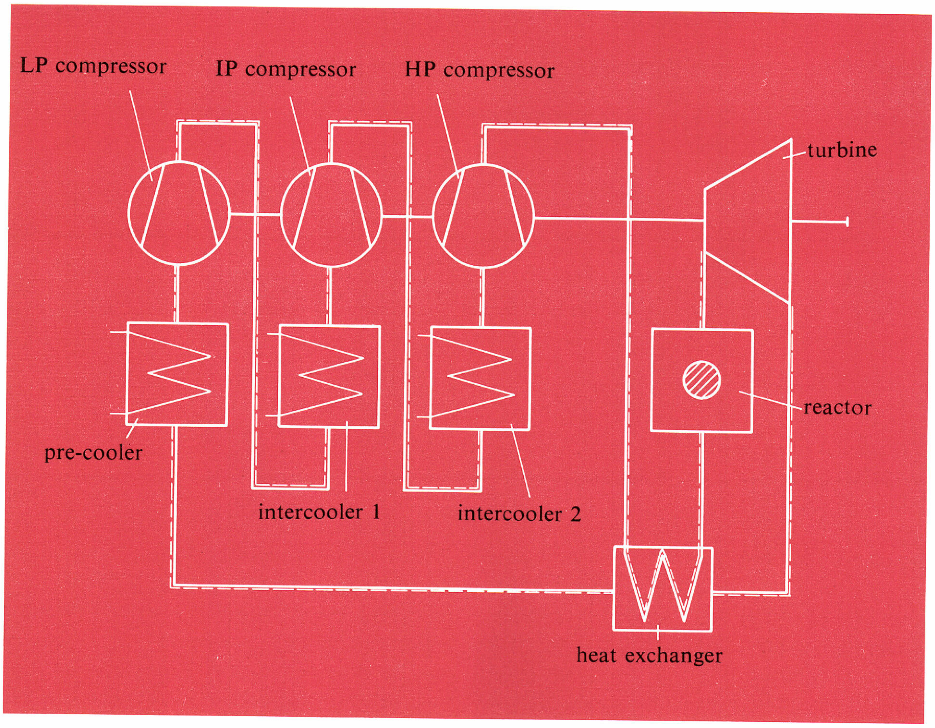


Fig. 3: Circuit diagram of a nuclear power plant with high-temperature reactor and helium turbine.

that at only 800°C the net efficiency is more than 45%—some 10% higher than for a conventional steam plant. Higher temperatures in the gas turbine are possible either by using rotor blade root cooling or molybdenum alloys. With a turbine inlet temperature of 1,000°C, the efficiency would be about 53%.

The high coolant temperature is characteristic of the helium turbine process. This means that a good deal of the waste heat occurring in the circuit process is hot enough (100-120°C) to be used economically for heating purposes, desalination of sea water in flash evaporators or as chemical process heat. If the electrical energy is used at the same time as the usable waste heat, an overall factor of between 65 and 85% is obtained for the utilisation of the energy permanently supplied to the process, depending on the size of the plant.

The amount of cooling water required for removing the waste heat is relatively small owing to the high temperature level in the gas turbine process. It represents only about one-quarter to one-third of the cooling water required by corresponding steam units and thus often facilitates the siting of such a power plant. Air cooling is also very suitable.

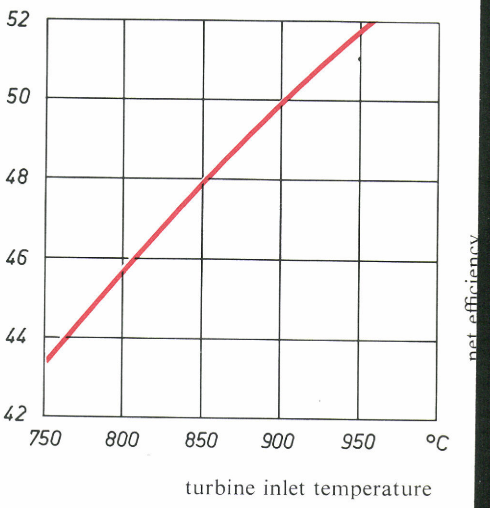
A characteristic feature of gas turbine plant operation is load change due to alterations in the pressure level, with which an almost constantly high plant efficiency is obtained even with small partial loads. High load change rates can be produced in these plants by an additional regulating system in which the turbine and the reactor can be bypassed with part of the working medium. Examination of the overall power balance reveals that the electrical power is zero when the reactor and turbine powers drop by only about 20%, the system temperatures remaining practically constant. It follows that a change in electrical power from full load to idling can be effected as quickly as the reactor power can be cut back from 100 to 80%. Calculations for nuclear power plants with gas turbines and experiments on existing closed-cycle air turbine plants have shown that the power can be raised from idling to the rated output in about one minute. This is of particular importance for a plant used for ship propulsion.

Design of the helium turbines

Owing to its low molecular weight compared with machines using other working media, the use of helium leads to an increase in the number of stages in the turbine and compressor. However, since the permitted peripheral speeds of helium turbines and compressors depend only on the limits imposed by the strength of the material used and in the case of helium restriction of the peripheral speeds due to the damaging influence of high Mach numbers is unnecessary, the resulting designs do not differ greatly from those of closed-cycle air turbines.

Helium turbines can be divided up into two main groups as regards plant lay-out. In the *single-shaft plant*—see diagram in Fig. 3—the turbine drives the three compressors and generates the effective power at the same time. The design here is that normally used for stationary closed-cycle air turbines, where the plant is usually operated at constant speed. With small and medium powers, step-down gearing is necessary to bring this down to the speed of the generator, since the design speed of the turboset in this effective power range is usually higher—about 8,000 to 10,000 rpm. In the case of larger units (300 to 1,000 MWe and over) the whole turboset can be designed for the generator speed of 3,000 rpm. In this way, the basic lay-out remains unchanged even for very large

Fig. 4: Net efficiency of power plants using helium turbines (for a capacity of 300 MWe).



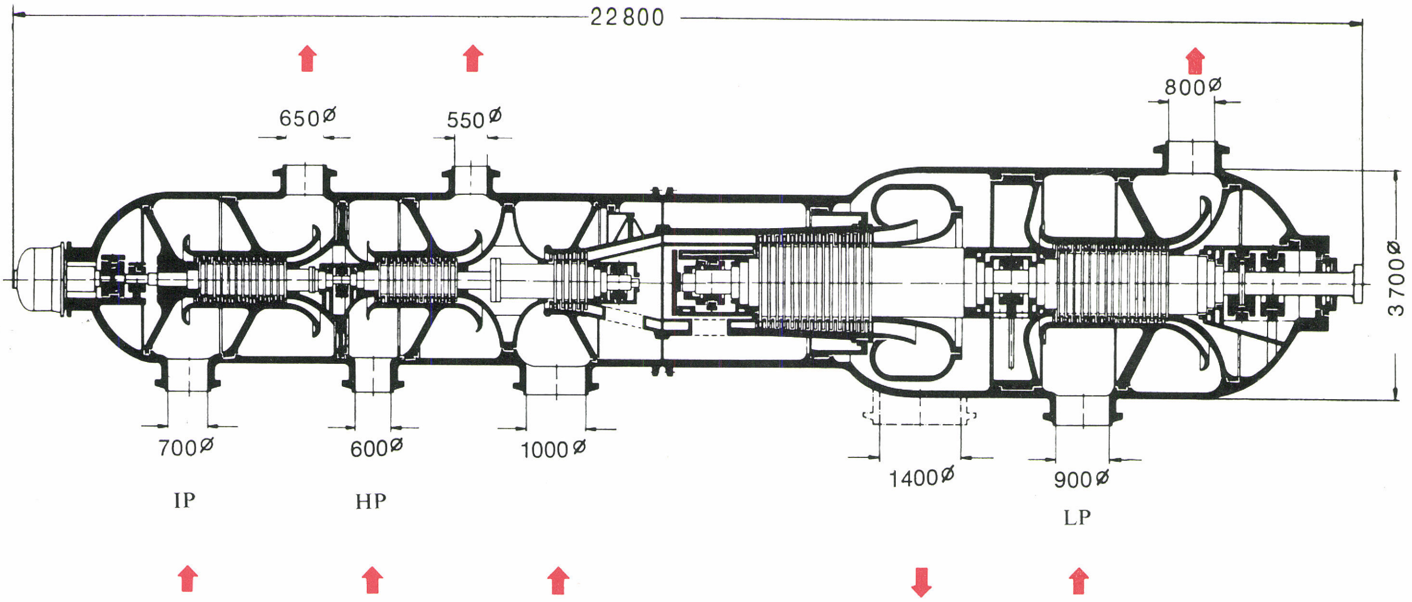
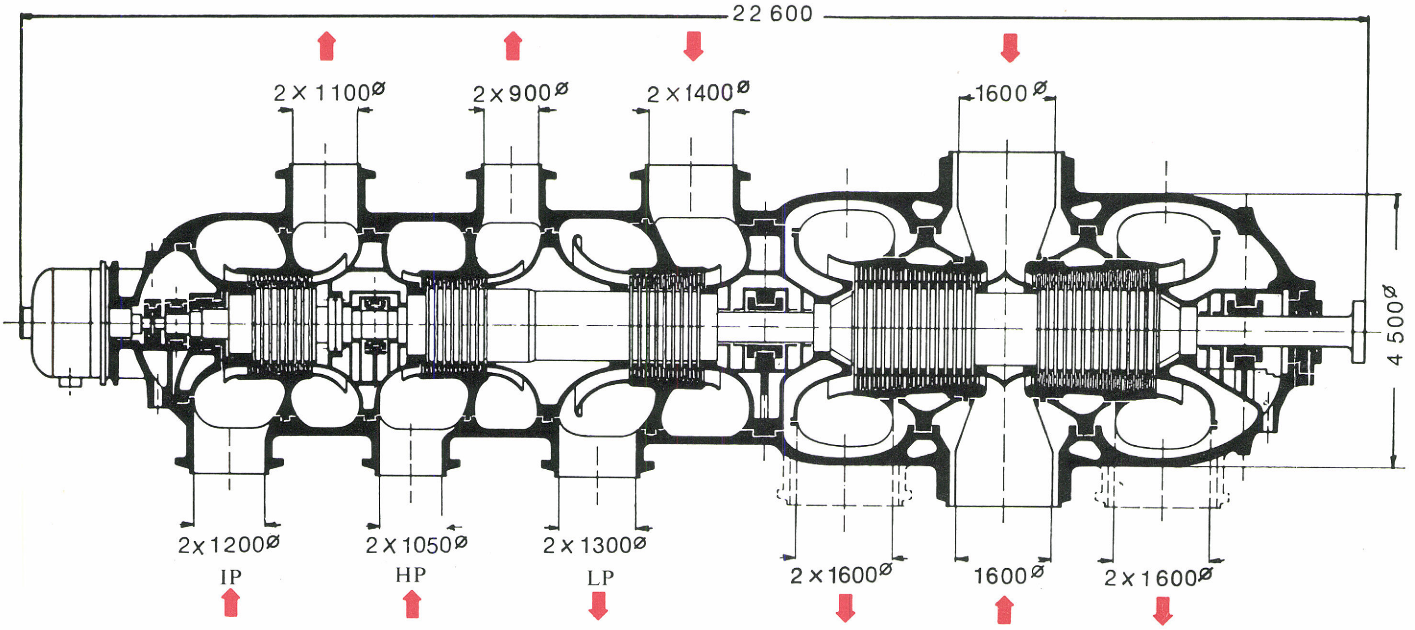


Fig. 5: Longitudinal cross-section of a 100 MWe helium turbine.
The turboset is of two-shaft design. From left to right: 1) the intermediate pressure compressor, the high-pressure compressor, the turbine high-pressure section, which drives the compressors,—all on the high-speed shaft; 2) the low-pressure section of the turbine and the low-pressure compressor, both mounted on the low-speed (3,000 rpm) shaft, which is coupled to the generator.

Fig. 6: Longitudinal cross-section of a 600 MWe helium turbine.
This is a single-shaft turboset. From left to right: the intermediate pressure compressor, the high-pressure compressor, the low pressure compressor and the two-flow turbine.



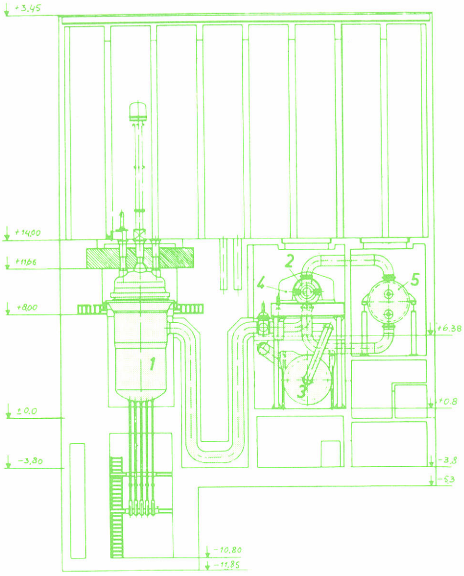


Fig. 7: 25 MWe nuclear power plant with high-temperature reactor and helium turbine for Geesthacht (Schleswig-Holstein).
1 High temperature reactor
2 Helium turbine
3 Heat exchanger
4 Generator
5 Intercooler

units, although for powers above 300 MWe two-flow design is used.

It has been shown that it is worthwhile to design the compressors for a speed greater than 3,000 rpm in the transition range from 30 to 250 MWe. This results in the *two-shaft plant*, ensuring better conditions for the blade channels in the intermediate and high pressure compressors, which handle relatively small volumes. In this case the turbine is divided so that the high-pressure section just covers the power of the intermediate and high pressure compressors, whereas the low-pressure section of the turbine drives the low-pressure compressor and generates the effective power at the same time. In such a plant the high-pressure group is designed for a higher speed (4,000-6,000 rpm) and the low-pressure groups run at the generator speed.

The entire output range of power plants is covered with these three helium turbosets (0.5-30 MWe, 30-250 MWe and 300-1,000 MWe and over). The individual turbosets differ only slightly from each other, the design of the helium turbines remaining simple for all but very-high-output units. The turboset of a 100 MWe plant is shown in Fig. 5 as an example of a two-shaft plant. Compared with single-shaft plants, this machine is relatively long (22.8 m). The diameter of the turbine casing is about 3.7 m.

The single-shaft turboset of the 600 MWe plant shown in Fig. 6 is considerably more compact. Despite the six-fold increase in power, it is only 22.6 m long with a casing diameter of 4.5 m, the turbine being of two-flow design. Both turbine and compressor are designed for the generator speed of 3,000 rpm. This design can also be used for 1,000 MWe units.

The outer casing of the helium turbosets has to be welded owing to the helium leaktightness required. The dimensions of high-output helium turbines differ only slightly from those of open-cycle gas turbines already in operation today. The rotor and casing dimensions of a 300 MWe helium turbine correspond roughly to those of an open-cycle 30 MWe gas turbine plant. Needless to say, greater wall thicknesses and thus more sturdy general designs are necessary owing to the higher pressure level of the helium turbosets. Construction of a plant of this output, however, can be based on experience acquired with open-cycle gas turbines of approximately the same dimensions which have been in operation for a number of years, as well as on experience in the design and operation of closed-cycle air turbines.

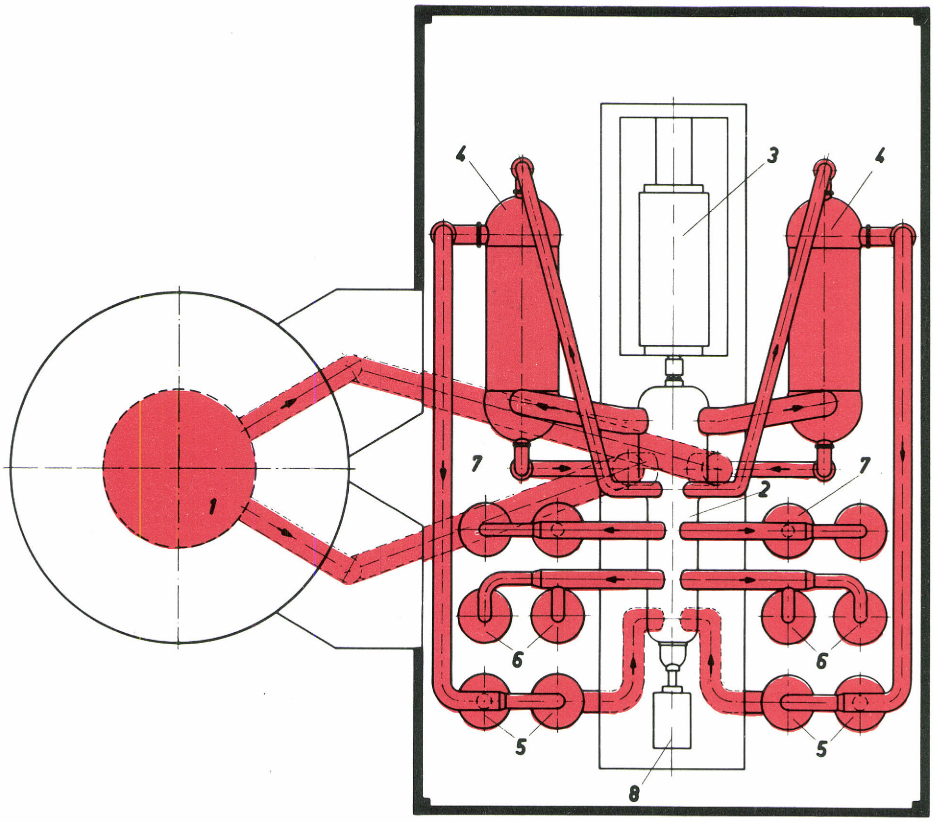
Nuclear power plants with helium turbines

For the first time in the world, a stationary plant with a coupled high-temperature reactor and helium turbine is being constructed at Geesthacht in Schleswig-Holstein as part of the German nuclear programme. The *Gutehoffnungshütte* will begin construction work on the compact 25 MWe experimental nuclear power plant later in 1968. A characteristic feature of this plant is that the main circuit and all the important ancillary systems are housed in one building, planned as a cubic concrete construction. Fig. 7 shows the arrangement of the reactor, helium turbine and heat exchange apparatus, which is roughly the same as that of a conventional closed-cycle air turbine plant. With a turbine inlet temperature of 750°C there is a net efficiency of over 37%, which is comparatively high for the small plant output of 25 MWe. The reactor is controlled by hydraulically operated absorber rods inserted in the reactor core from below. The fuel elements are changed through nozzles in the pressure vessel cover when the reactor is shut down. Many important questions relating to the direct coupling of a high-temperature reactor with a helium turbine can be examined in the experimental plant, in particular the effects of possible fission product deposits, the dynamic behaviour of the overall system, the emergency cooling of the reactor and the sealing of the helium turbine. The experience acquired during the construction and operation of the Geesthacht plant will form an important and necessary basis for the construction of large nuclear power plants and ship propulsion units of this type.

The helium turbine/reactor lay-out chosen for the Geesthacht plant can be retained for larger plants. Fig. 8 shows a diagram of a 300 MWe nuclear power plant. This method of construction draws to a large measure on design and constructional experience obtained with conventional thermal power plants and *HTR* nuclear plants with a steam circuit. The reactor core is surrounded by a prestressed concrete pressure vessel and the turbine, together with the compressors, heat exchangers and coolers, is housed separately. The helium flows down the reactor core and is fed to the turbine via two concentric pipes. After expansion, cooling, compression and re-heating, the helium enters the pressure vessel as "cold gas" at a temperature of about 450°C. This high inlet temperature necessitates a special insulation and cooling system for the prestressed concrete, since this must be kept at an average temperature of 60°C. The turbine, complete with the compressors, is in one single-shaft unit; heat exchangers and coolers, mainly because of their size and the diameter of the connecting pipes, are divided into two parallel units.

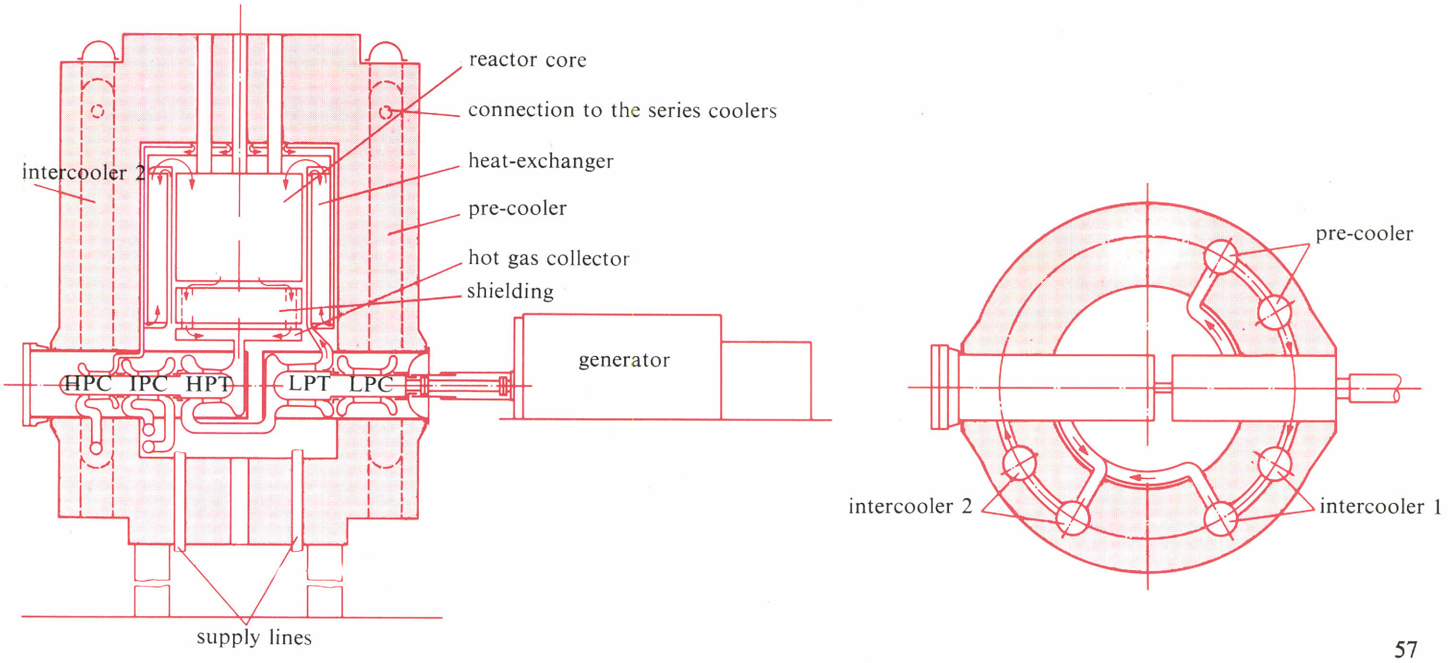
Special safety precautions to prevent an uncontrolled loss of helium following a pipe fracture are necessitated by this separate reactor-turbine arrangement. In the event of this occurring, the gas, which is at high pressure and temperature and possibly slightly radioactive, might enter the turbine house. This must therefore be designed as a gastight and pressure-resistant structure so that the helium inside can be retained for a certain time before being passed off to the atmosphere via suitable filters.

Fig. 8: 300 MWe high-temperature reactor with helium turbine of conventional construction.
1 Reactor
2 Turbine-compressor group
3 Generator
4 Heat exchanger
5 Pre-cooler
6 Intercooler 1
7 Intercooler 2
8 Starting motor



The total space taken up by the reactor and turbine unit can be considerably decreased if integrated design is used. The entire helium circuit and the core can be housed in a prestressed concrete pressure vessel (integrated system) owing to the relatively small length and diameter of the helium turbines and the great flexibility in the design of the coolers and heat exchangers. This leaves only the generator outside the pressure vessel. In order to make better use of the available space and to step up reliability, the turbine and compressors are divided into several shafts with this compact arrangement, similar to that mentioned for helium turbine plants in the 50 to

Fig. 9: High-temperature reactor with helium turbine of integrated design.



Primary pressure	60 atm
Turbine inlet temperature	800°C
Reactor inlet temperature	460°C
Compressor inlet temperature	15°C
Turbine efficiency	92%
Compressor efficiency LP/IP/HP	90/89/88%
Overall pressure drops	8%
Heat exchanger pinchpoint	20°C
Pressure ratio	2.7
Coupling efficiency	50.0%
Net efficiency	49.1%

Table II: Data of a 1000 MWe nuclear power plant with high-temperature reactor and helium turbine (double intercooling).

250 MWe range. The turbine is thus divided up into a working turbine with an LP compressor, rotating at the same speed as the electric generator (3,000 rpm), and one or more high-pressure turbines to drive the IP and HP compressors. These compressor-turbine groups can be operated at higher speed and their dimensions thus kept small. Fig. 9 shows an integrated arrangement in which the turbogroup is mounted beneath the reactor core behind the shielding, and the heat exchangers and coolers vertically in the walls of the prestressed concrete pressure vessel. The coolant flows from top to bottom through the reactor core into a hot gas collector and then to the HP and LP turbines. This is an improvement on the conventional method in that the pipework is considerably simplified. In addition, the helium from the compressors can be used directly for cooling the prestressed concrete walls. Owing to the new demands placed on the circuit components, including the turboset, the solution of the problems presented by the integrated design method should take some time yet.

We have seen that the high-temperature reactor with a helium turbine offers special economic advantages even for high outputs. Efficiency increases of 10-15% are possible compared with the high-temperature reactor/steam turbine system. Table II shows that, at 800°C, a 1,000 MWe plant has a net efficiency of almost 50%. In addition, these plants can be made very compact, so that capital costs may well be reduced even further compared with corresponding steam plants. The main features of this new energy-generating system can be summed up as follows:

— High plant efficiency, even for medium and small plant outputs.

— Compact design because of high power density, high helium temperature, simple process operation and few auxiliary units. Suitable as a ship propulsion system.

— Practically constant high efficiency at partial load due to pressure level control.

— Low cooling water requirements—about one-quarter to one-third of that of a corresponding steam unit. Air cooling is also suitable because of the high waste heat temperature.

— Improved safety through relatively low pressure, constant temperature in the gas circuit and avoidance of the danger of water ingresses in the reactor cooling circuit.

— Excellent fuel utilisation and low fuel costs owing to high specific power (up to 1 MWe/kg of fissile material in large plants) high conversion rate and high burn-ups with long fuel element lifetimes.

— Flexibility of the reactor core permits the use of various fuel cycles. The use of slightly enriched uranium is thus especially attractive, as investigations of the *Dragon* group have shown.

— Economy through low plant costs, high efficiency and good fuel utilisation as well as possible use of waste heat (for instance, desalination of sea water).

— Good potential for upgrading to large power units and higher gas temperatures, together with the considerable increases in efficiency thus achieved. (EREA-A 7-8).

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Activation analysis in steel production

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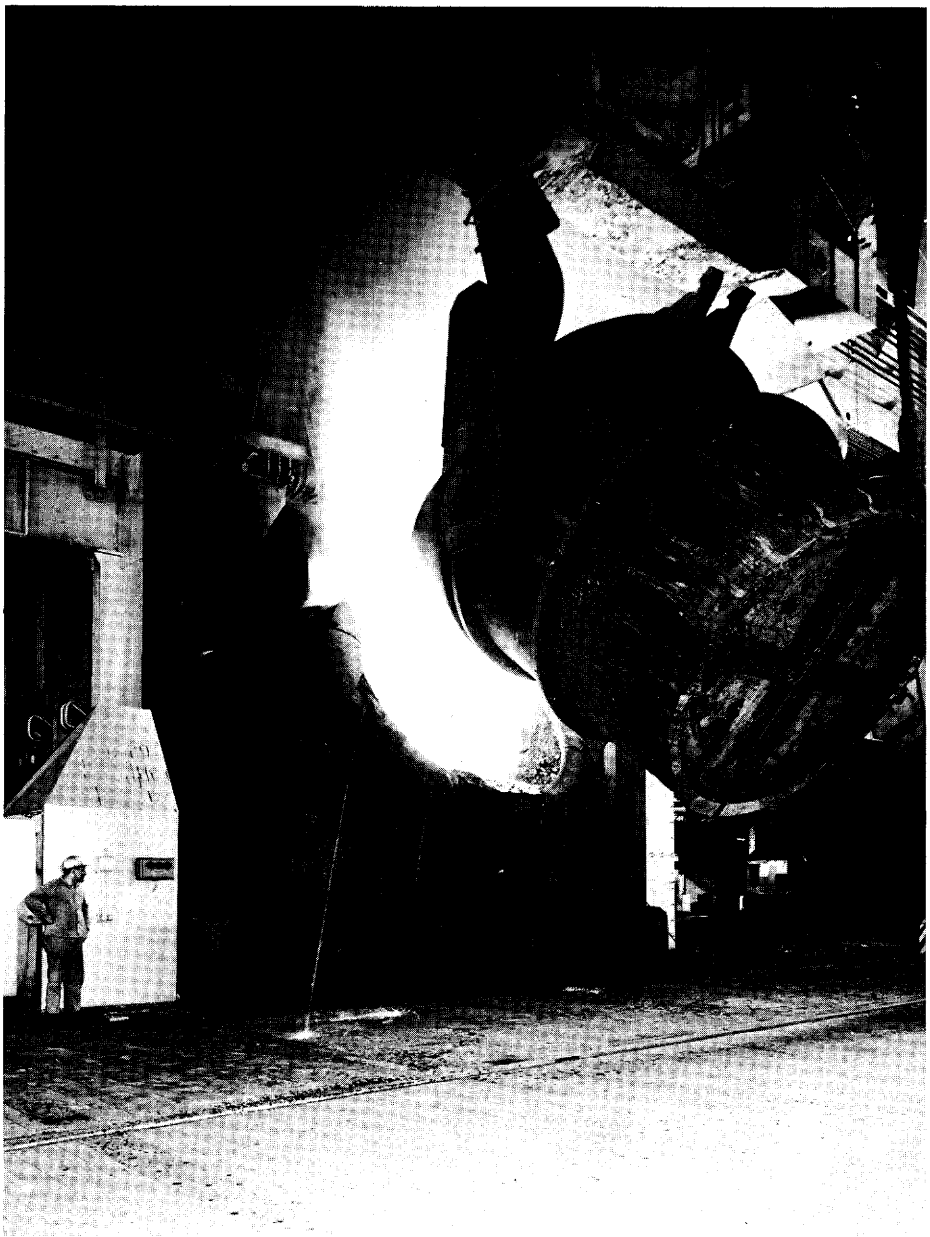
FOR MORE THAN six years now the *Bureau Eurisotop* has been active in promoting the use of isotopes and radiation in industry. As the years have gone by, it has been realised that gradually more and more industrial firms are becoming acquainted with the new techniques involved in applying these "by-products" of the nuclear energy business.

These techniques include measurement and control methods, the use of radioactive tracers in research and industry, the irradiation of chemicals, food etc. and many other applications. This article will deal with another important item in this complex field, namely activation analysis, particularly as applied to the analysis of oxygen in steel.

What exactly is activation analysis?

Through this technique, the analysis of a substance is carried out by means of its radio-activation: the substance is irradiated—mostly by neutrons or by other nuclear particles—and develops new radioactive nuclei emitting gamma radiation, the nature of which is characteristic for each element, whereas its intensity is a measure of the quantities present in the substance. In fact so few nuclei need to be transmuted, that the method remains essentially non-destructive.

Originally the only irradiation facilities available for this type of analysis were nuclear reactors. Thus, activation analysis was practically restricted to large research centres equipped with reactors. However, the development of compact and less expensive machines, producing nuclear particles in sufficiently high amounts, has completely changed this pattern. Neutron-generating machines being the most advanced among them, activation analysis nowadays generally makes use of irradiation by neutrons. In these machines deuterons are accelerated in a high vacuum to several hundred kiloelectronvolts and directed onto a target containing tritium, thereby sparking off the reaction: ${}_1H^2$ +



One of the two 175 m³ capacity crucibles of the Cockerill-Ougrée *(Seraing, Belgium) L.D. steelworks being fed by means of a casting ladle.*

${}_1H^3 \rightarrow {}_2He^4 + {}_0n^1$. Neutron fluxes of up to 10^{11} per second and with energies of up to 14 MeV are the result. Other types of activation, for instance by means of table-top cyclotrons, are on the way however; activation-analysis will irresistibly enter industrial laboratories and research institutes as a routine technique in the forthcoming years.

Oxygen in steel

An illustration of this rapid development is to be found in the steel industry, especially for the rapid and precise determination of oxygen. Why is the steel expert so interested in oxygen-content, especially in the case of molten steel?

It is the search for a high-quality steel that makes him concerned about this oxygen level. Oxygen is used to oxidise the impurities in the molten steel. Part of it will, at these high temperatures, also react with the iron itself, with manganese, etc. and will carry them into the slag in the form of oxides. The greater part of the oxygen is finally removed by deoxidation, for instance with aluminium, forming aluminium oxide, most of which again moves into the slag.

Now as each type of steel requires a specific amount of deoxidation, it is of utmost importance to know the oxygen content in the melt during the process of deoxidation. Besides, it goes without saying that excessive amounts of deoxidation materials in the steel are undesirable. Moreover, these materials are not cheap. Consequently, the steel-maker has to evaluate the oxygen content very carefully in order to bring it to the desired level.

This situation of course requires quick decisions during steel production itself. It is therefore essential to be able to rely on a rapid and precise oxygen analysis of the contents of the steel melt.

Action taken by the "Bureau Eurisotop"

The *Bureau Eurisotop*, which is playing an active part in the promotion of activation analysis in the European Community, has been aware of this situation for some years. Given the fact that oxygen, with the fast neutrons produced by the neutron generators described above, reacts in the following way:

$${}_8O^{16} + {}_0n^1 \rightarrow {}_7N^{16} + {}_1p^1$$

the possibility of determining oxygen by activation with fast neutrons, taking the gamma radiation of ${}_7N^{16}$ as a measure, is evident.

As the energy of the gamma radiation from ${}_7N^{16}$ is very high, discrimination from the gamma radiation emitted by other radioactive nuclides is simple. Furthermore, the half-life of ${}_7N^{16}$ is only 7.4 seconds, which allows short irradiation and measuring times and thus makes for a rapid determination, which can be repeated at will. In view of these prospects, a number of contracts were concluded in order to get to grips with the problem of steel analysis and to develop adequate means of coping with it.

The results of these contracts were made public at a meeting, held in November 1967 in Brussels and Liège. About 70 representatives of the steel industry, of nuclear science and of the neutron generator-manufacturing industry met to discuss, first, a comparative study of activation analysis and "classical" means of oxygen-determination in steel, second, the development of an automatic set-up for the determination of oxygen by neutron activation, and third, the practical lessons to be derived from the actual use of this apparatus in a steelworks.

Activation compared with other analysis methods

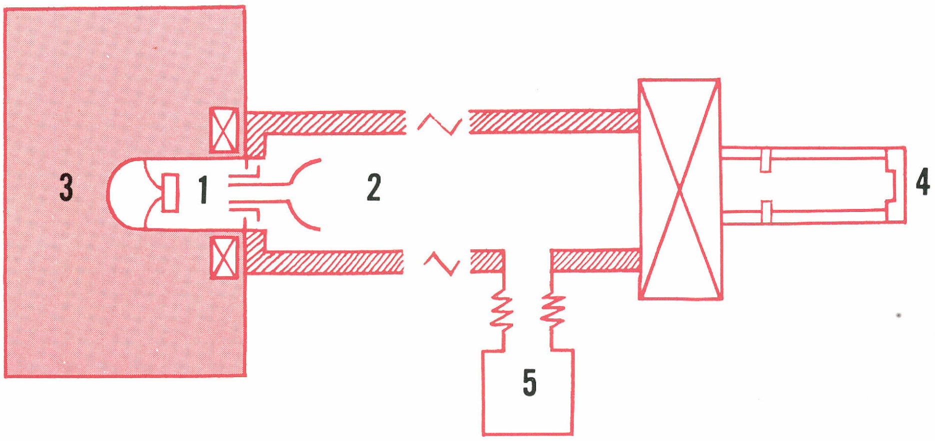
The comparative study was carried out by the steel firm *ARBED* (Luxembourg) with the help of a great many research institutes and industrial firms all over the world. Steel samples were analysed by means of several types of gas and vacuum fusion techniques as well as by activation. They were sent around from one laboratory to the other and went as far as Japan and California.

The conclusion of the co-ordinating firm *ARBED* is that results obtained by fusion and activation show a good correlation, but *only with activation analysis is a sufficiently rapid and precise determination of oxygen in steel attainable.*

Development of an automatic set-up

The automatic set-up was developed by Prof. Dr. J. Hoste and his co-workers of the University of Ghent (Belgium). The apparatus consists essentially of a neutron generator from the firm *SAMES* in Grenoble (France), a measuring system for counting the radioactivity induced in sam-

Figure 1: Simplified diagram of the neutron generator developed by SAMES.
Deuterium gas is supplied to the source (1), where it is ionised. The ions are accelerated in the tube (2) through the action of the high-voltage generator (3) and directed onto the tritium target (4), where a reaction takes place, leading to the production of neutrons. A pump (5) keeps the tube under vacuum.



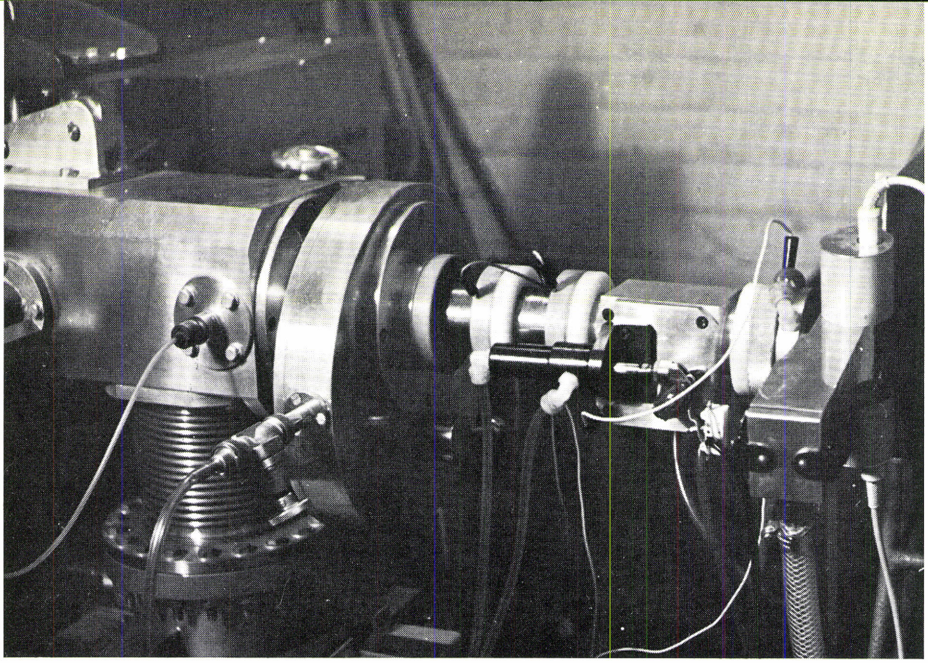


Figure 2: The acceleration tube of the SAMES-*J-type neutron generator installed at the* Cockerill-Ougrée *L. D. steelworks.*

ple and standard by the neutron irradiation, and a pneumatic conveying system between the generator and the counters (figures 1, 2 and 3).

Important features of the apparatus are the irradiation and counting of both the standard and the sample at the same time under strictly controlled conditions and the direct printing of the results in parts per million of oxygen. The machine can handle as much as 30 steel-samples per hour, weighing up to 40 grammes, completely automatically. The weights of these samples are about 50 to 80 times greater than in classical analysis, which clearly makes them more representative.

Practical results

A replica of this set-up was installed and used in the steel-works *Cockerill-Ougrée-Providence* at Seraing near Liège (Belgium). The group of Prof. Hoste and the *Bureau Eurisotop* co-operated here with the *European Coal and Steel Community (ECSC)*, the sub-commission "*Oxygen*" of the *International Commission for Gas Analysis in Steel*, the *Centre National de Recherches Métallurgiques* in Liège and, of course, the steel-works itself.

The participants of the meeting held by the *Bureau Eurisotop* had the opportunity of visiting this installation, whereby it became evident how completely the activation-analytical method is already incorporated into the framework of this steel firm. The management and staff of *Cockerill-Ougrée-Providence* made no secret of their unanimous satisfaction with this contribution of nuclear science to the quality-control of their steel. This opinion was founded on the results of some thousand oxygen determinations.

As distinct advantages of the new method their experts mentioned:

— the method is non-destructive and thus repeatable at will;

— it is very selective for oxygen and has a higher precision than classical methods;

— the method is absolute; it can rely on comparisons with pure oxygen compounds as standards;

— samples are much larger and thus more representative;

— determinations are extremely rapid and enable every change in the oxygen content of the bath to be followed;

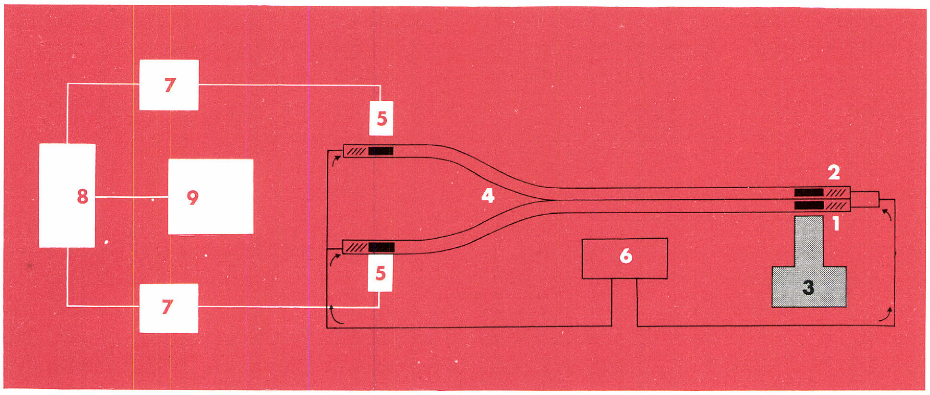
— it costs less than classical procedures;

— the analytical apparatus may be used for other applications in the steel-works as well.

The *Cockerill*-firm may be proud to be the first steel industry in the Community to have introduced this advanced analytical method in their works. (EREA-A 7-9)

Bibliography (1) N. STOLL, A. WAGNER, L. GOEDERT (ARBED): Etudes des possibilités d'utilisation industrielle de l'analyse par activation pour le dosage de l'oxygène et éventuellement de l'azote et de l'hydrogène dans les aciers. *Euratom Report 3161 f, I, II and III.* (2) E. VERNIN ET J. PERDIJON (SAMES): Conditions optimales de fonctionnement du générateur de neutrons. *Euratom Report EUR 3210 f, II and III.* (3) J. HOSTE, D. DE SOETE, A. SPEECKE: The determination of oxygen in metals by 14 MeV neutron activation analysis. *Euratom Report EUR 3565e.*

Figure 3: Simplified diagram of the activation analysis set-up developed by Prof. Hoste and coworkers for the analysis of oxygen in steel.
The sample to be analysed (1) and the standard (2) are activated by the same neutron beam for a set length of time. The beam is produced by the neutron generator (3). After activation, sample and standard are each conveyed pneumatically (4) to scintillation detectors (5) for counting of the induced activity. Compressed air is supplied by the pumping unit (6). The signals given out by the detectors are processed (7) and fed into a computer (8). The final results are printed out (9) in ppm of oxygen.



EURATOM NEWS

Commission surveys Euratom's future role

Community nuclear power programmes are gathering pace. Altogether 2,300 MW of electricity generating capacity is in operation, 2,000 MW under construction, and a further 5,100 MW planned. This means that by 1971-1972 the Community will have nearly 9,400 MW of capacity in operation, of which 3,100 MW in France, nearly 2,500 in Federal Germany and 1,300 MW in Italy. For the longer term the Commission has raised its 1980 estimate from 40,000 MW to 60,000 MW, compared with the US Atomic Energy Commission's forecast of 150,000 MW in the United States by the same date.

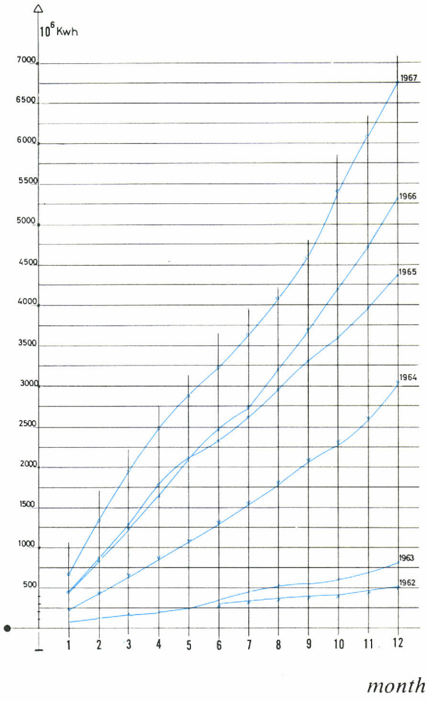
Now that nuclear power has reached the industrial stage in the Community, what problems face the organisation of the Community's nuclear industry on rational lines in order to be able to meet foreign competition both at home and on the export market? What could Euratom's role be in continuing the work of integration? These questions are examined in a communication submitted in March by the Commission to the Council. The document "Future Research Activities of Euratom" sets out the options open to the Community regarding the methods which could be employed to introduce a maximum of co-ordination between research and development programmes and to tackle various problems of industrial organisation; and it suggests the areas of research which could be brought into a Community programme. Its purpose is to instigate a discussion in the Council of Ministers leading to agreement on terms of reference which would act as guidelines for the Commission in the preparation of a detailed programme to be submitted to the Ministers at a later date. Euratom's Second Five-Year Research Programme expired at the end of 1967. Various circumstances prevented the preparation of a new programme to run from January 1968; instead, an austerity programme was approved for the single year 1968 (cf. *Euratom Review* Vol. VII (1968) No 1, p. 31) during which the procedure for the preparation of a new programme, running from January 1969, could be carried out. The Commission's communication can be considered the first move in the preparation of a new pluri-annual programme.

The document suggests a number of basic principles governing Euratom's future activities. First, it points to the need for a massive intervention of the public area in reactor development since industry cannot support all the necessary research on its own. Nearly 1,100 million $ is to be spent during 1968-1972 on fast reactors alone according to known programmes. Expenditure of this order demands a maximum of co-ordination if wastage is to be avoided; secondly, publicly-financed basic, support and "public service" research must be carried out in a number of fields. Thirdly, the need for a Community nuclear industrial policy is emphasised. While in the United States orders for 50,000 MW of nuclear power capacity have been received since 1966, shared between four constructors, 5,000 MW have been ordered since the same date in the Community, shared between no less than twelve constructors. Larger, perhaps multinational, groupings are required, as is a wider market, the co-ordination of public purchasing policies and a better dissemination of information. In addition, a greater concentration of research linked to present projects is required.

A very important problem on which a Community decision must be taken sooner or later is that of enriched uranium. Virtually all advanced (second generation) reactors, the light-water cooled family of "conventional" (proven-type) reactors, as well as fast reactors, perhaps, will be consumers of uranium 235. Hitherto the Community has been and, for some years to come, will be, dependent on the United States. But isotopic separation facilities to manufacture uranium 235 will one day be needed in Europe, and the study of the various possibilities is under way (cf. *Euratom Review* Vol. VII (1968) No 1, pp. 24-29 "Should the European Community produce its own enriched uranium?" by H. Michaelis).

The Commission document goes on to deal with the options, the methods of action and the categories of activity. As to methods, in addition to the Joint Research Centre, its staff of 2,250 and the 300 million $ invested in it, the association

Cumulative net electricity production of nuclear origin in the European Community. In 1967 the output of electricity of nuclear origin amounted to 6.742 million kWh within the Community, i.e. 1.6% of the net total electricity production.



contract (partnership) formula and industrial contracts (all methods employed hitherto), the possibilities of greater resort to Joint Enterprises and ad hoc financial support for research projects are cited.

Joint Enterprises, in which the Commission would participate (the formula is in fact very flexible) are held to be an appropriate means of putting the construction of prototype-reactors—a major segment of the work to be done in the reactor field—onto a Community basis. A Joint Enterprise is suggested for fast reactor prototypes (three are currently projected), along with complementary research, and is considered appropriate for the construction of a heavy-water reactor prototype. Other reactor work in which the Community could participate would be high-temperature reactors (e.g. *Dragon* and the German *AVR* project), "proven-type" reactors (with the aim of enabling the Community to compete on the world market) and such areas of back-up research as uranium processing, the long-term supply of uranium 235, the reprocessing of fuel and the disposal of waste.

As to "basic research", the document notes the existence of thermonuclear fusion research projects (hitherto linked together under a network of Euratom association contracts) which together put the Community "on an equal basis with the most advanced countries" in scientific research. Other categories of research under this heading include nuclear physics, the proposed *SORA* reactor (an instrument to provide an intensive source of neutrons), scientific information processing and the direct conversion of energy—in the context of the provision of fuel for space vehicles.

The category "support research" could cover reactor physics, reactor construction materials, uranium prospection methods, the security of installations and economico-technical studies.

Lastly, "public service research". The document covers the continuance of the Community's work in the following fields: training research staff, dissemination of information, the operation of the Central Bureau for Nuclear Measurements (an establishment of the Joint Research Centre, at Geel, Belgium), biology, health protection, the use of radioisotopes and fissile material safeguards.

The document is necessarily hesitant on the subject of costs, for it is based on partial information and represents only the first phase of the preparatory work. Moreover, a confrontation procedure (the communication of national programmes) has yet to be carried out and could well give rise to new Commission proposals. However, the total cost of the work to be performed in the Community in which Euratom could participate (i.e. the fields outlined above) is put at 1,882 million $, Euratom's share of this total, apart from the 300 million u.a. reckoned as the cost of running the Joint Research Centre and the payment of personnel, would depend on the type of participation and the Community's share of the financing.

Eurex plant officially handed over to CNEN

The *Eurex* plant of Saluggia, Italy, for the chemical reprocessing of irradiated fuels with a high U 235 enrichment, was officially handed over to the *CNEN (Comitato Nazionale per l'Energia Nucleare)* on 26 March by the constructors. This concludes the construction stage of a plant which the *CNEN* had placed in the hands of European industry. The deadline has been met and European industry has supplied an excellent proof of its credentials.

The *Eurex* plant will make it possible to gain valuable experience at the pre-industrial level and to acquire all the know-how necessary for the construction of reprocessing plants on an industrial scale.

The commissioning of the plant is scheduled for next autumn after completion of the cold tests.

Treatment of fodder by irradiation

The working party set up by the *Bureau Eurisotop* on the use of irradiation techniques for the decontamination of fodder held its second meeting at Bilthoven in March 1968. This party is made up of representatives of the Public Health Institutes of the Member States, of the European Federation of Manufacturers of Composite Foodstuffs *(FEFAC)* and of the Commission of the European Communities.

The party has agreed to undertake a programme of pilot tests on the irradiation of mixed fodder flours of current type destined for feeding animals (poultry, pigs, etc.). The aim of this treatment is to eliminate bacteria (salmonella) present in the fodder, which influence the growth of the animals and are the source of poisoning in man (salmonellose) through the consumption of contaminated foodstuffs. According to the estimates in epidemiological reports, this disease affects about 300,000 persons per year in Community countries, with an averagem ortality of 300. The proposed work aims to study the irradiation conditions of fodders and the physiological and anatomo-pathological effects by raising a group of 100 pigs with irradiated foodstuffs and a reference group. The results of the tests should demonstrate the efficacy of the irradiation treatment and supply the information necessary for the introduction of this method in the fodder industry.

Organisations and persons interested in this programme may apply to the Commission of the European Communities, Bureau Eurisotop, 51, rue Belliard, Brussels 4, Belgium.

Plastic treatment of wood with radiation-polymerised monomers

Information Bulletin No. 22 of the *Bureau Eurisotop* on the plastic treatment of wood with radiation-polymerised monomers has just appeared. The author is Dr. Arno Burmester, Member of the Federal Institute for Materials Testing in Berlin and a well-known expert on wood treatment.

The bulletin reviews the physical and technological properties of radiation-treated wood. The discussion centres on a remarkable new material which possesses properties of both wood and plastic. It is known either as "wood-plastic compound' or as "polymer wood". This polymer wood has a number of considerably superior characteristics as compared with untreated wood. These include hardness, surface lustre, resistance to chemical attack and scratching, dimensional stability, etc. It is suitable for such uses as boat parts, sports apparatus, parquet flooring, furniture, handles for instruments and tools and surfaces exposed to the weather.

The bulletin has appeared in German, and a French edition is to follow. Two further bulletins on the subject of polymer wood will appear in the summer of 1968. One bulletin will deal with the production of this material and the other with its applications.

Persons interested in obtaining information on wood-plastic compounds may obtain the bulletin free of cost by writing to:

Commission of the European Communities, Bureau Eurisotop, 51, rue Belliard, Brussels 4, Belgium.

Second international conference on thermionic electrical power generation

More than 150 persons met at Stresa, Italy, from 27 to 31 May 1968 to take part in the "Second international conference on thermionic electrical power generation", organised under the patronage of the European Nuclear Energy Agency of the *OECD* in collaboration with the Ispra Establishment of Euratom's Joint Research Centre. The first conference took place in London three years ago.

Thermionic emission is one of the phenomena permitting the direct conversion of heat energy into electrical energy to which the greatest amount of research has been devoted (see *Euratom Bulletin*, Vol.V (1966) No. 1, pp. 24-30, "The thermionic converter and its use in a reactor" by Josef Bohdansky).

More than 120 communications were presented by delegates from Belgium, Bulgaria, France, the Netherlands, Rumania, the United Kingdom, the Soviet Union, the United States and West Germany, together with Euratom. The strong participation of the Soviet Union, which alone presented almost one third of the communications, attracted particular attention.

For the first time detailed information of a technological character was made public on nuclear devices developed in the United States and suitable for use in space. In particular, some of the American papers referred to a unit, consisting of a nuclear reactor and a thermionic converter, powerful enough to be contemplated as a system for feeding an electrical propulsion unit in a space vehicle.

Publication of a third series of technical notes

The Commission of the Communities recently issued a third series of technical notes, the purpose of which, it will be recalled, is to inform industrial circles of the existence of certain technical data collected in the course of the execution of Euratom's research programme which could be exploited commercially.

This series comprises the following notes:

— *Note 1/C:* "Amplifier with threshold discrimination". This amplifier, which was designed for gamma spectrometry using semi-conductor detectors, is characterised by a low-noise differential input stage with field effect transistors.

— *Note 2/C:* "Inhalation chamber for biological experiments on small laboratory animals". The chamber, which is characterised by a calibrated-leakage device and a special ultra-violet light absorption measuring section, was developed for the purpose of studying effects of vapours and aerosols of organic compounds on human beings.

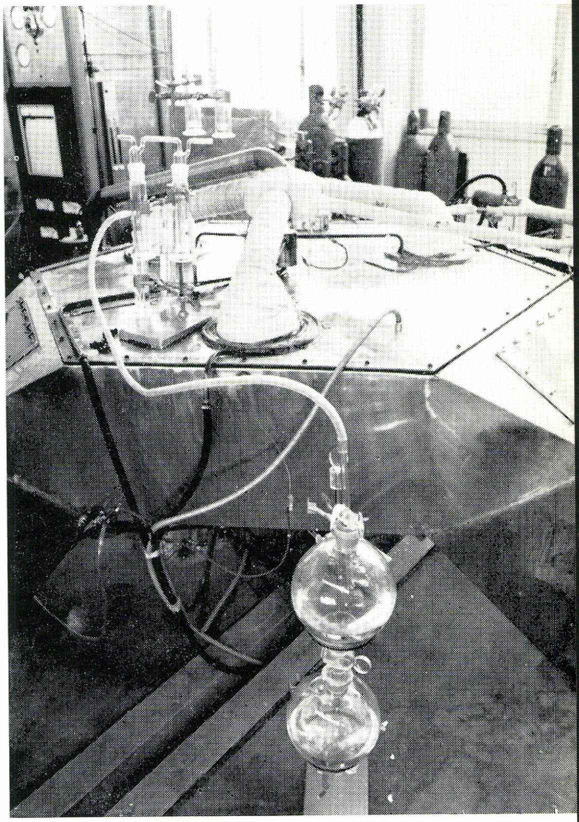
— *Note 3/C:* "Vertical scanning rig for the ultrasonic inspection of tubes". This rig, which was built specially for the ultrasonic detection of defects in finned cladding tubes, can be used for the semi-automatic scanning and the recording of the flaws detected.

— *Note 175:* "Circuit for parallel comparison of binary numbers". Each binary stage of the circuit consists of only four magnetic cores, which are statically regulated by associated bi-stable elements. An auxiliary signal is introduced at the stage corresponding to the highest value in order to reach the stage at which the inequality is greatest, which then gives the stage indicating equality.

— *Note 191/273:* "Furnace for the preparation and recovery of uranium carbide". The furnace is used for the preparation of uranium carbide by means of calciothermy or magnesiothermy from uranium halides. The furnace is provided with a high-frequency induction coil with a field concentrator. The reactive mixture is heated in the crucible and the volatile and possibly radioactive products formed during the reaction are condensed in the internally cooled shaft.

Further information may be obtained by writing to:

Commission of the European Communities,
Directorate-General XIII,
Directorate A,
51, rue Belliard,
Brussels 4 (Belgium).



Inhalation chamber for biological experiments on small laboratory animals, developed at the Ispra Establishment of the Euratom Joint Research Centre.

Readers ask for bibliographies

As some may have noticed, we have for the first time, in this issue of *Euratom Review*, systematically added a bibliography to each major article. In doing so we hope to have met a need expressed by a large number of readers when replying to the questionnaire "Would you like to help us to improve Euratom Review?" By the way we found the replies to this questionnaire quite fascinating. For a full report on readers' criticisms and suggestions read the next issue of *Euratom Review*.

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