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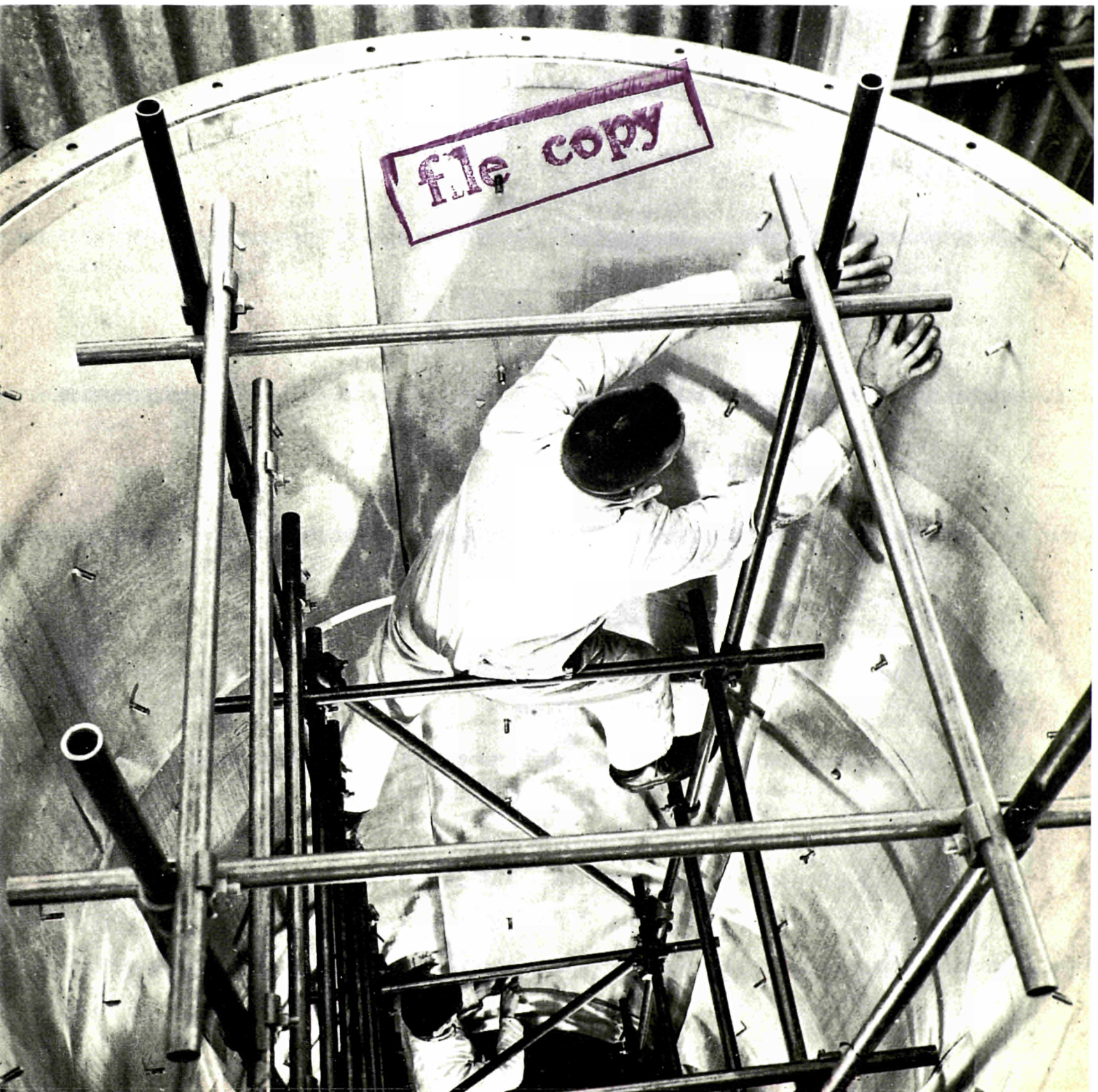

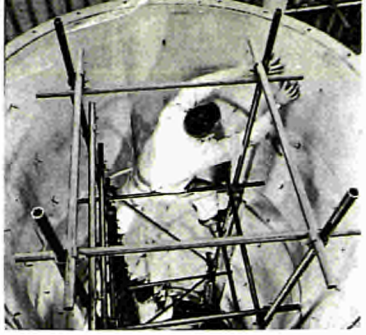

Technicians of the Compagnie des ateliers et forges de la Loire (CAFL) *installing in a mock-up a heat insulation material specially developed for prestressed concrete reactor vessels (see page 19).*

CONTENTS

Quarterly Review of the European Atomic Energy Community (Euratom)

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Quarterly Review of the European Atomic Energy Community (Euratom)

1968-1

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).

Since the earliest days of this periodical, certain readers have told us from time to time that they did not much care for the name "Bulletin"; not that they found it ugly, but simply unsuitable, for, they said: "Euratom Bulletin is not a bulletin but a review".

We have always agreed that they were right and it was only the question of continuity that prevented us from changing the title. They have at last got their way, however, as the "Euratom Bulletin" will from now onwards be called the "Euratom Review".

We may add that with its new name, our periodical cannot be confused with the "European Communities Bulletin", a monthly which gives regular details of the official activities of the ECSC, the EEC and Euratom, and can therefore call itself a bulletin without risk of challenge.

PLUTONIUM

PLUTONIUM IS A fissionable substance which is already contributing largely to the output of nuclear power, for, produced by neutron capture in the uranium 238 of power-reactor fuel elements, it is to some extent consumed on the spot. In natural uranium reactors, where a burn-up of 4,000 MWd/t (megawatts-day per tonne) is reached, it is estimated that roughly a third of the energy is produced by plutonium fission.

Plutonium, a potential fuel for thermal reactors . . .

In the fuel unloaded from reactors there is still some plutonium left which has not undergone fission and can be recovered by chemical processing. In this way, without going through the isotope separating technique, it is possible to obtain a highly concentrated fissionable material which, it is thought, should be able to replace uranium 235 in certain thermal reactors requiring enriched fuel. As plutonium is produced in growing quantities year by year, the economic pros and cons of such recycling in thermal reactors have attracted a considerable amount of research.

. . . but imperative for fast reactors

In so far as fast neutron reactors are considered to have a promising future, however, there is little doubt that they will be the main customers for plutonium. For in a fast spectrum plutonium offers more attractive nuclear properties than uranium 235. The number of neutrons emitted per neutron absorbed in the fissionable plutonium isotopes is greater than in the case of uranium 235, thus giving higher conversion rates. Actually, it is only by burning plutonium that the fast reactors are able to synthesise—by neutron capture in uranium 238—more fissionable material than they consume to produce power.
For the fast breeder reactor, therefore, the development of plutonium fuel is essential.

Plutonium, but in what form?

For the same core volume, a fast neutron reactor, to go critical, needs a far higher concentration of fissionable materials than a thermal neutron reactor, because the cross-sections in a fast neutron flux are substantially lower than in a thermal neutron flux.
In view of this the fast reactor experts envisage using fuel elements containing about 15% of plutonium, the rest being natural or depleted uranium to serve as fertile material. But if the investments in fissionable materials are to be kept down to an economically acceptable level, the fast reactor will have to yield a greater power per unit volume of fuel than a thermal reactor where the enrichment is lower. From an economic calculation which allows for all the consequences of raising the specific power, it appears that the optimum specific power rating lies in the region of 100 thermal megawatts per tonne of fuel, i.e. between five and ten times the value accepted in a light-water thermal reactor.
These enormous specific powers can be carried off if an exceptionally efficient coolant, such as sodium, is employed and if the fuel is finely fragmented, thus increasing the surface in contact with the cooling liquid. Hence the fuel planned for the fast reactors is in the form of pins with an outer diameter of some 6 mm only.

Again for economic reasons, and particularly because of the high cost of reprocessing and of manufacturing plutonium fuel elements, it is hoped to obtain average burn-ups of the order of 10% of the heavy atoms. It should be remembered, for purposes of comparison, that burn-ups of 2% of the heavy atoms in the fuel elements of thermal neutron power plants are regarded as considerable.

Oxide – the present choice

It was very quickly seen that oxide is the form of fuel that would afford high burn-ups, under an optimum specific power, without demanding extensive development work. Oxide fuel has been chosen in the designs for the fast power reactor prototypes which it is intended to build in the near future. This choice is backed by the long experience acquired in the thermal neutron power plants, where uranium oxide has demonstrated—though admittedly under less severe conditions—its outstanding behaviour under irradiation and its perfect compatibility with low-cost cladding materials. As a choice it is a poor best, however, for oxide is a relatively low-density substance; this is a not inconsiderable drawback, because in a reactor where the volume of structural materials and coolant fluid is fixed, the lower the weight and therefore the density of the fuel, the higher the probability of losing neutrons through parasitic captures in non-fissionable and non-fertile materials. Neutron loss leads directly to a drop in the conversion rate.
From this standpoint—and for other reasons too—metal alloy would have been a far preferable form of fuel, which would have afforded a considerably higher breeding rate than the oxide (1.5 instead of 1.2). Unfortunately the U-Pu alloy, even in conjunction with additives such as molybdenum, swells a good deal under irradiation and this, it is feared, would rule out the high burn-ups demanded by economic requirements.
A number of laboratories are now studying the uranium-plutonium carbides, one of whose attractions is their density, which is

distinctly higher than that of the oxides. But they, too, present some knotty problems: the carbide reacts with the customary cladding materials and although the swelling under prolonged irradiation is not so disastrous as with the metals, it is still far from negligible.
Another compound under consideration is the nitride, but not enough is yet known about its properties to allow of assessing its future.
We have outlined the reasons that led to the choice of oxide as the fuel for the first fast neutron power reactors. It was assumed, and rightly so, that the properties of the mixed oxides, containing 15-20% of PuO_2, are not basically very different from those of the pure UO_2. Differences do exist, however, which it would be dangerous to ignore and, in any case, the very high performance demanded of fast reactor fuels is liable to raise new problems which ought to be examined.
That is why the study of plutonium oxides and mixed uranium and plutonium oxides forms a major part of the activities of the European Transuranium Institute today.

tomorrow's fuel

HENRI MATTYS, *European Transuranium Institute, Karlsruhe*

Plutonium-handling difficulties due to its toxicity . . .

This Joint Research Centre establishment specialises in the handling of high-activity alpha-emitters; the work is done in glove boxes (Fig. 1), by carefully trained staff. Plutonium fixes eagerly on certain tissues, particularly on bone, where owing to its very high radiation energy it produces irrevocable damage. To give some idea of the extent of the safeguards needed in our laboratories, it need only be recalled that the concentration of plutonium in laboratory air must not exceed 3×10^{-11} g/m³ and that the venting of air with a plutonium content of more than 10^{-12} g/m³ is forbidden by law.
These very low concentrations are ensured through exceptionally efficient and expensive air filtering equipment and by constant monitoring of premises and personnel (Figs. 2a and 2b).

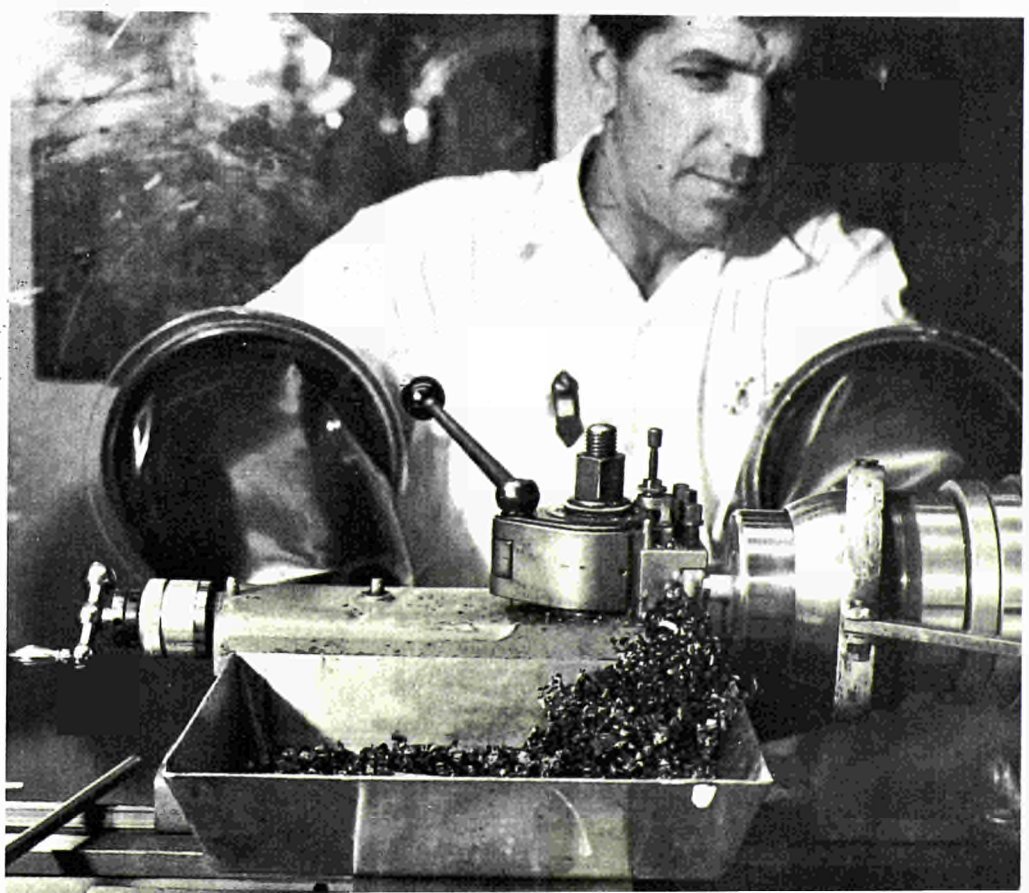


Fig. 1: Glove boxes must be used for work on plutonium.

. . . and its low critical mass

Another property of plutonium which necessitates a very close watch is the low value of its critical mass. A nuclear chain reaction can be obtained with roughly 500 g of plutonium if it is dispersed or dissolved in water, or with about 6 kg of plutonium without any moderator. During manufacture, therefore, the amount of material to be handled in one batch will have to be limited and operations involving dispersion or dissolution in water must be avoided as far as possible.

The mixed UO_2-PuO_2 pellet manufacturing process . . .

The mixed-oxide manufacturing processes are basically similar to those used for the pure uranium oxide. In both cases, pellets must first be pressed from oxide powder and then sintered under hydrogen at a temperature of about 1,600°C. The Transuranium Institute ceramists have so far produced some 15,000 UO_2-PuO_2 pellets, including 7,000 for the "fast" irradiation programme requirements.

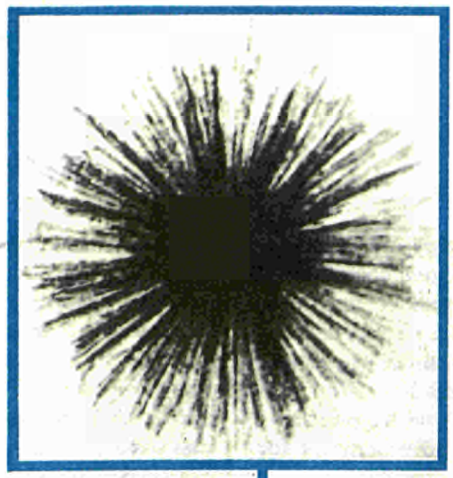


Fig. 2a: Autoradiography of an air sampling filter taken in a contaminated area. Autoradiographic methods are used at the European Transuranium Institute to determine the size of plutonium particles and their distribution. The total magnification is 960 ×; the length of the alpha tracks is approximately 25 microns.

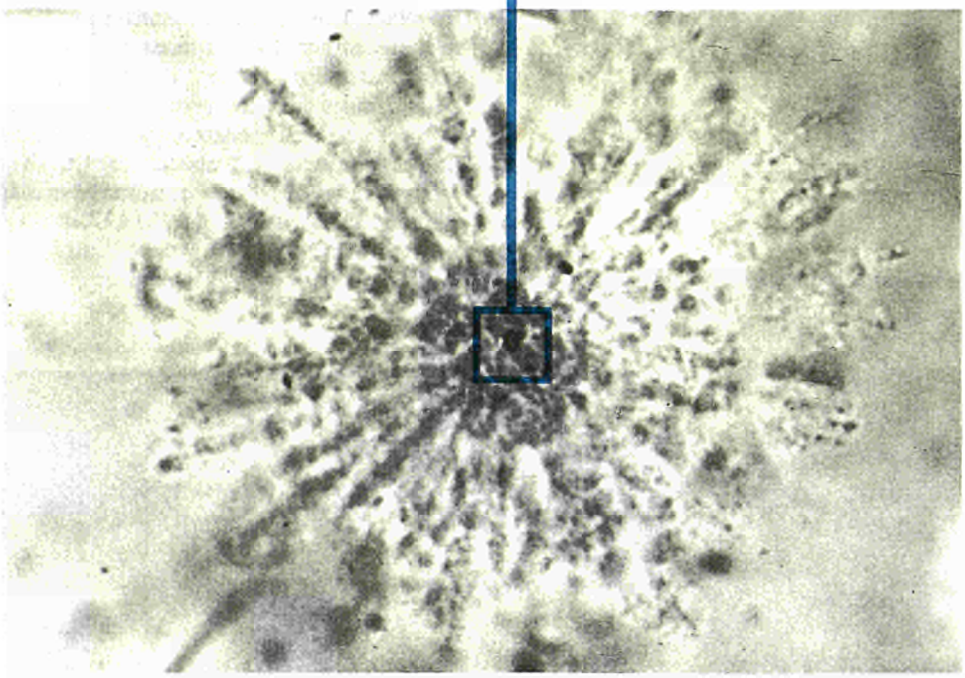


Fig. 2b: In the centre of this picture is to be seen the plutonium particle from which the alpha tracks shown in the previous photograph originate. The alpha tracks from the original autoradiography have been chemically dissolved. The magnification is 1100 ×; the size of the particle is 1.5 microns.

. . . must yield a uniform product . . .

The UO_2-PuO_2 fuel has to comply with homogeneity criteria imposed essentially by reactor safety considerations. In the event of a power excursion the uranium and plutonium temperatures must rise simultaneously, if advantage is to be taken of the negative temperature coefficient of uranium 238 to lower the reactivity of the fuel.

The temperature effect can be explained qualitatively as follows. The cross-sections are very complicated functions of the relative speed of the neutrons and the nuclei with which they react. In particular, very high peaks occur in very narrow energy regions (resonances). Thus the temperature, which directly affects the velocity of the nuclei, is one of the factors that determine the relative speed and has a specially important effect near the resonances. The reactor neutron spectrum, everywhere in the reactor, is shaped by the cross-sections of each of its components and is therefore influenced by their temperature. As the neutron reaction speed depends both on the cross-sections and on the neutron spectrum, it likewise depends on the temperature of each of the reactor components. A temperature rise will thus bring about a reactivity variation which can be negative or positive. In the case of uranium 238 the variation is negative.

Fuel homogeneity, which will ensure that the uranium and plutonium temperatures vary simultaneously, can be achieved by various methods.

In the United States, the co-precipitation of plutonium and uranium is advocated. This powder-manufacturing process does in fact produce almost perfect fuel homogeneity.

At the Transuranium Institute, as indeed in other European laboratories, it is considered preferable to mix dry, separately prepared UO_2 and PuO_2 powders, and sufficient homogenisation is achieved during sintering. This process is more economical than co-precipitation. The UO_2 is the most abundant constituent of the powder mixture (85%) and its sinterability will determine that of the $UO_2 + PuO_2$ mixture. Its production does not entail the use of glove-boxes and above all involves no problems of safeguards against criticality accidents. From this standpoint,

therefore, the choice of process for preparing sinterable UO_2 is unhampered by any restrictions. The PuO_2, on the other hand, does not need to be highly sinterable and the method of preparing it will be chosen mainly with an eye to simplifying the safety surveillance.

. . . of constant dimensions . . .

Another aspect of the manufacturing process used at the Transuranium Institute deserves mention. The diameter and, to a lesser degree, the density of the pellets loaded into the clads have to comply with certain very stringent specifications. The UO_2 producers grind their pellets, after sintering, to specified dimensions with a centreless grinder. During this operation slurries are produced which may raise criticality problems. For this reason, the Institute's ceramists have sought ways of pressing and sintering so that pellets of satisfactory diameter may be obtained straight away. Using methods of the conventional ceramics industry, they have succeeded in achieving a precision margin of $\pm$ 30 microns over the pellet diameter (about 6 mm) without any need for the costly grinding operation and also without complicating the process.

. . . and of well-defined stoichiometry

Chemistry has accustomed us to definite proportions. The formula for plutonium dioxide—PuO_2—implies that for every plutonium atom in this compound there will be two oxygen atoms. In fact, with certain elements, a whole range of perfectly stable solid compounds can be obtained. This is the case with plutonium oxide, whose O/Pu atomic ratio (stoichiometry) may vary (beyond 700°C) from 1.7 to 2.00 without any observable segregation of a new phase (see Fig. 3). We shall see further on that several properties of the mixed uranium and plutonium oxides are sensitive to the stoichiometry, i.e. to the O/U + Pu ratio.
For this reason it may be essential to have available, during manufacture, some method of adjusting the O/U + Pu ratio as required. At the Transuranium Institute the ceramists have therefore developed simple methods by which this can be done.

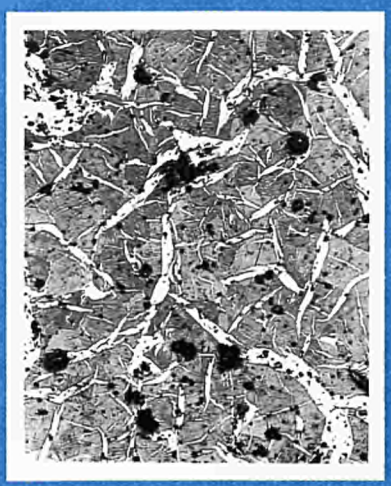
The properties of UO_2 and UO_2-PuO_2 . .

We have already said that the mixed (U, Pu)O_2 fuel must be homogeneous. It was ascertained some years ago that $UO_{2.00}$ and $PuO_{2.00}$ oxides are miscible in any proportion and at any temperature, a property which ensures perfect homogeneity of the mixed oxide. In addition, it may be assumed that the (U, Pu)O_2 solid solution will be as stable under irradiation as the pure UO_2.

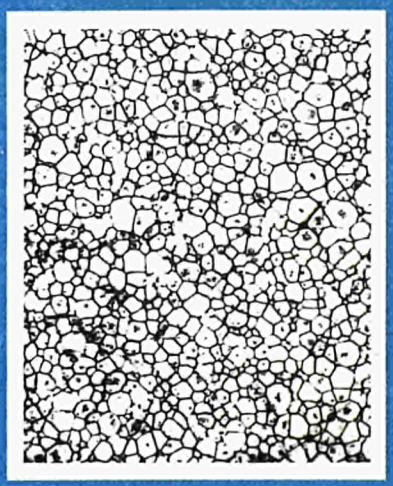
. . . the phase diagram . . .

However, we have seen (Fig. 3) that there are several plutonium oxides, some of which are stable in a very wide stoichiometric range. The maximum O/Pu ratio attainable by plutonium oxide is 2.00, whereas the stoichiometry, or O/U ratio, of the uranium oxides is 2.00 or higher. It was only at a very high temperature that the existence was detected of a uranium oxide with an O/U ratio of less than 2.
It cannot therefore be affirmed *a priori* that complete oxide miscibility is possible with every value of the O/U + Pu ratio other than 2.00, and there are grounds for fearing

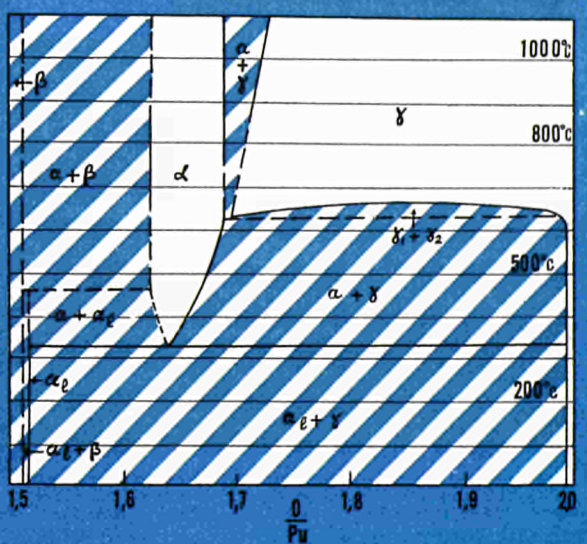
Fig. 3: Phase diagram of plutonium-oxygen system. Depending on the temperature and overall composition, the oxide contains a single type of crystal (single-phase system) or two different types (two-phase system). The micrographs show examples of (right) a single-phase solid and (left) a two-phase solid. (In both cases the black spots are pores).



$PuO_{1,73}$
two-phase



PuO_2
single-phase



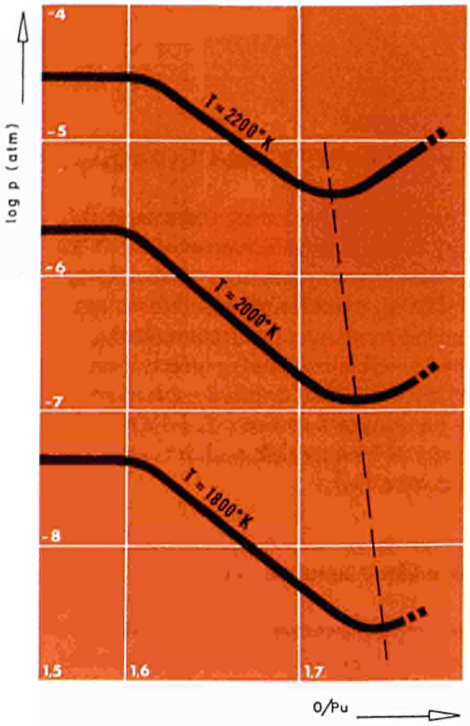


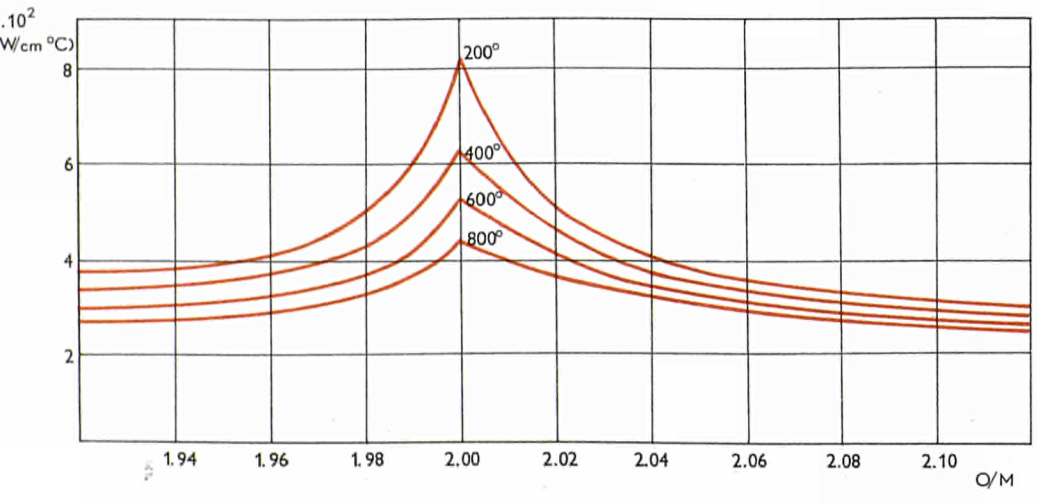
Fig. 4: The vapour pressure of plutonium oxide depends on its composition and on temperature.

that, with certain values of this ratio, plutonium segregations liable to jeopardise reactor safety might occur. Furthermore, certain phases might be less stable under irradiation. Information on this subject is astonishingly fragmentary. A research team at the Transuranium Institute is now trying to fill in this gap by studying the phase diagram for the ternary uranium/plutonium/oxygen system.

. . . the vapour pressure . . .

Knowledge of the vapour pressures of the mixed oxides is also needed in estimating reactor safety. It is easily conceivable that under certain conditions the plutonium might separate from the uranium by fractional distillation. The vapour pressure of plutonium oxides has been measured at various temperatures and with various O/Pu ratios in the solid state by a research team at the Transuranium Institute (Fig. 4), and tests are now in progress to measure the vapour pressure of mixed uranium and plutonium oxides. It can already be stated that if segregation does occur through selective vaporisation, it will be inconsiderable and will be limited to fuel regions where the temperature is over 2,000°C.
Clearly, a knowledge of the vapour pressure will also help in estimating the extent of a nuclear accident: a critical mass will break up proportionately faster as the vapour pressure of its components is higher, and this is a further justification for research on vapour pressures.

Fig. 5: Thermal conductivity of $(U_{0.8}Pu_{0.2})O_{2 \pm x}$ versus oxygen content (O/M). (After H. E. Schmidt and J. Richter).



. . . thermal conductivity

There is a limit to the power that can be extracted from a fuel rod of a given diameter. If, by raising the reactor neutron flux, one oversteps certain power limits, the fuel element will melt, just as even a well-cooled electric conductor melts if subjected to too high a current. The temperature attained in the centre of a fuel rod with a given surface temperature will be the higher, the lower the heat conductivity of the fuel material.
It is thus important to know the conductivity of the fuel material, and in the case of the mixed oxides, whose stoichiometry may vary widely, it was necessary to investigate the variation of conductivity versus the O/U + Pu ratio. This study, conducted by a research group at the Transuranium Institute and actually still in progress (see Fig 5), has shown that the strictly stoichiometric $(U, Pu)O_{2.00}$ material is a better heat conductor than material whose O/U + Pu ratio differs from 2.00.

Irradiation

A nuclear fuel element is only acceptable when it has proved itself under irradiation, in conditions similar to those which are expected in the reactor for which it has been designed.
The Transuranium Institute is at present carrying out an irradiation programme of which we will outline a few typical features, without attempting a full account. We will confine ourselves to a description of current or completed experiments.
The programme has a double purpose. The first aim is to study the overall behaviour of a prototype fuel element. The irradiations in progress or terminated in the Dounreay fast reactor come into this category of experiments.
Secondly, it is hoped to gain a better understanding of certain specific phenomena which occur in the fuels. This category includes the series of capsule experiments (see Fig. 6) planned or already effected, mainly in the *BR 2* reactor at Mol.
By means of the capsule irradiations (see Fig. 7) it has been possible to obtain fuel material irradiated at a very high burn-up in a very short time. Certain pellets now being examined in the Institute's hot cells have exceeded a 12% burn-up.

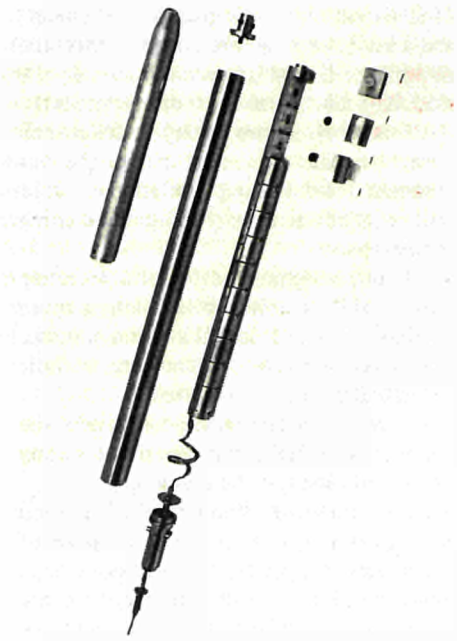


Fig. 6: POM-II irradiation capsule before assembly. In this capsule oxide pellets were irradiated at 100,000 MWd/tonne.

In view of the success of this irradiation and the very interesting results revealed by initial examinations, particularly as regards the swelling of pellets and the distribution of fission products, the programme of irradiation in capsules is being extended. It is planned, for instance, to use this device to study compatibility between cladding materials and fuels, and also the creep of oxide under irradiation.

The irradiations in the Dounreay reactor have two aims:

a) to study the effect of the stoichiometry on the behaviour of oxide under irradiation;

b) to compare several oxide density distributions in the fuel element.

Let us consider each of these experiments.

a) *Stoichiometry:* We have seen that, among the oxides, the exactly stoichiometric one is the best heat conductor. It can therefore be predicted that, other conditions being equal, the temperature at the centre of the fuel element with an O/Pu + U ratio of 2.00 will be lower than that obtaining in the centre of the non-stoichiometric oxides. It is possible, however, that the maximum temperature tolerable in the oxide may be higher when the O/U + Pu ratio is different from 2.00. It is also possible that the swelling phenomena, which largely limit the fuel element's lifetime or performance, may be sensitive to the ratio value. These are the considerations warranting the "stoichiometry" programme in the Dounreay reactor.

b) *Initial radial arrangement of fuel in the element clad.* Very high burn-ups—it is hoped to reach 10% in fast reactors—mean considerable expansion of the fuel, because each fissioned atom gives two fission-product atoms. Moreover, bubbles of fission gas trapped in the fuel cause it to swell under their pressure, and it seems that this swelling becomes particularly drastic at a burn-up of about 6%.

To allow for the swelling and expansion, some 20% of empty space or pores must be provided in the fuel pin. There is some argument over the best void distribution and this is the object of a series of experiments carried out at Dounreay. In these experiments we compare three possible starting layouts (schematised in Fig. 8), for which the production costs may differ considerably. The experiments will be rounded off with a study of the behaviour under irradiation of vibrated fuel, i.e.

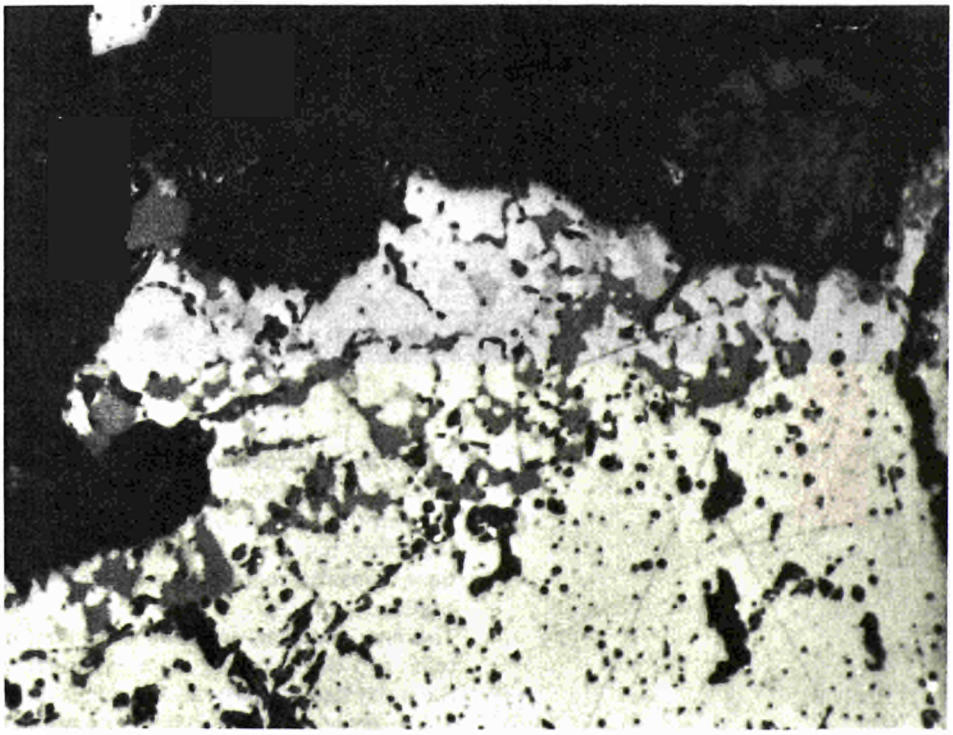


Fig. 7: Micrograph of mixed uranium-plutonium oxide irradiated at 100 MWd/tonne. In the mixed oxide (light grey) can be seen pores (black spots), metal inclusions (white spots) and ceramic precipitates (dark grey) differing in nature from the most abundant phase. Enlargement × 320.

fuel elements obtained by compacting oxide powder by mechanical vibration, a process which it is thought may prove more economical than the pellet-pressing and sintering method.

Post-irradiation examination

The irradiation programme is pointless unless adequate facilities are available for examination in hot cells. The Institute has its own hot laboratory, which has been in service since October 1966. A special feature of the laboratory hot cells is that they give protection not only against gamma radiation, but also, being perfectly leak-tight, against contamination from alpha emitters.

The hot laboratory consists of a line of cells for physical examinations and a chemistry line. The physics line comprises:

— An entry cell used for the dismantling of fuel assemblies, radiography of elements, local analysis by gamma spectrometry, the sampling of fission gases contained in the cladding, and dimension measurements;

— A set of metallography cells which house, besides the microscope, the instruments for cutting the fuel rods into small samples and for polishing and etching the samples;

— A micro-sampling cell, for the selective removal of inclusions which might require detailed study; in this cell gamma spectrometry analyses on a transverse sample section can also be performed;

— A density determination cell, where the phenomena of fuel expansion and swelling after irradiation can be studied.

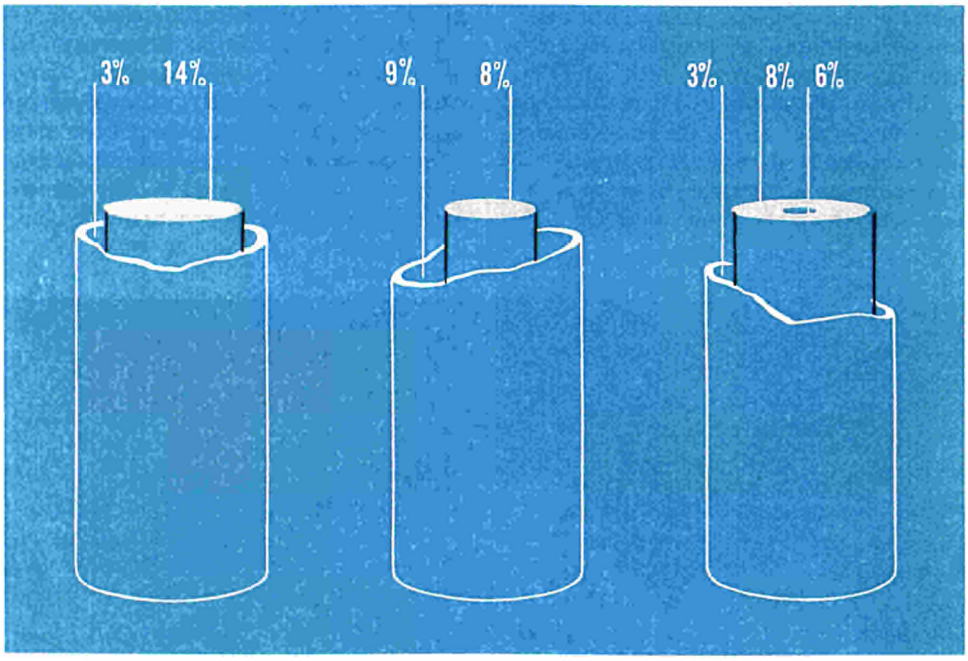
As to the chemistry line (Fig. 9) it is used for preparatory work for the purpose of determining burn-ups or various isotope abundances by mass spectrometry or other physical methods. In this line the following operations can be carried out on highly irradiated material:

— Dissolving of samples of the order of one gramme, with simultaneous sampling of the fission gases released during dissolution;

— Simple chemical assays and various separations of elements with a view to their analysis by mass spectrometry or alpha spectrometry;

— Local microsampling by laser. At a point selected by periscope, the laser vaporises a few microgrammes of material which are condensed onto a target in contact with the sample. The material collected on the target is then analysed by mass spectrometry.

Fig. 8: To allow for the expansion and swelling that occur during irradiation, spaces (or pores) must be provided in the fuel pin. The best arrangement of the spaces is a controversial question, however. An experiment is being carried out in the Dounreay reactor to compare the behaviour of pins in which the same volume of space or pores (17%) is distributed in three different ways.



Conclusions

The Transuranium Institute's work on plutonium oxide and mixed uranium and plutonium oxides is justified by the potential value of these fuel materials for the fast neutron reactors.

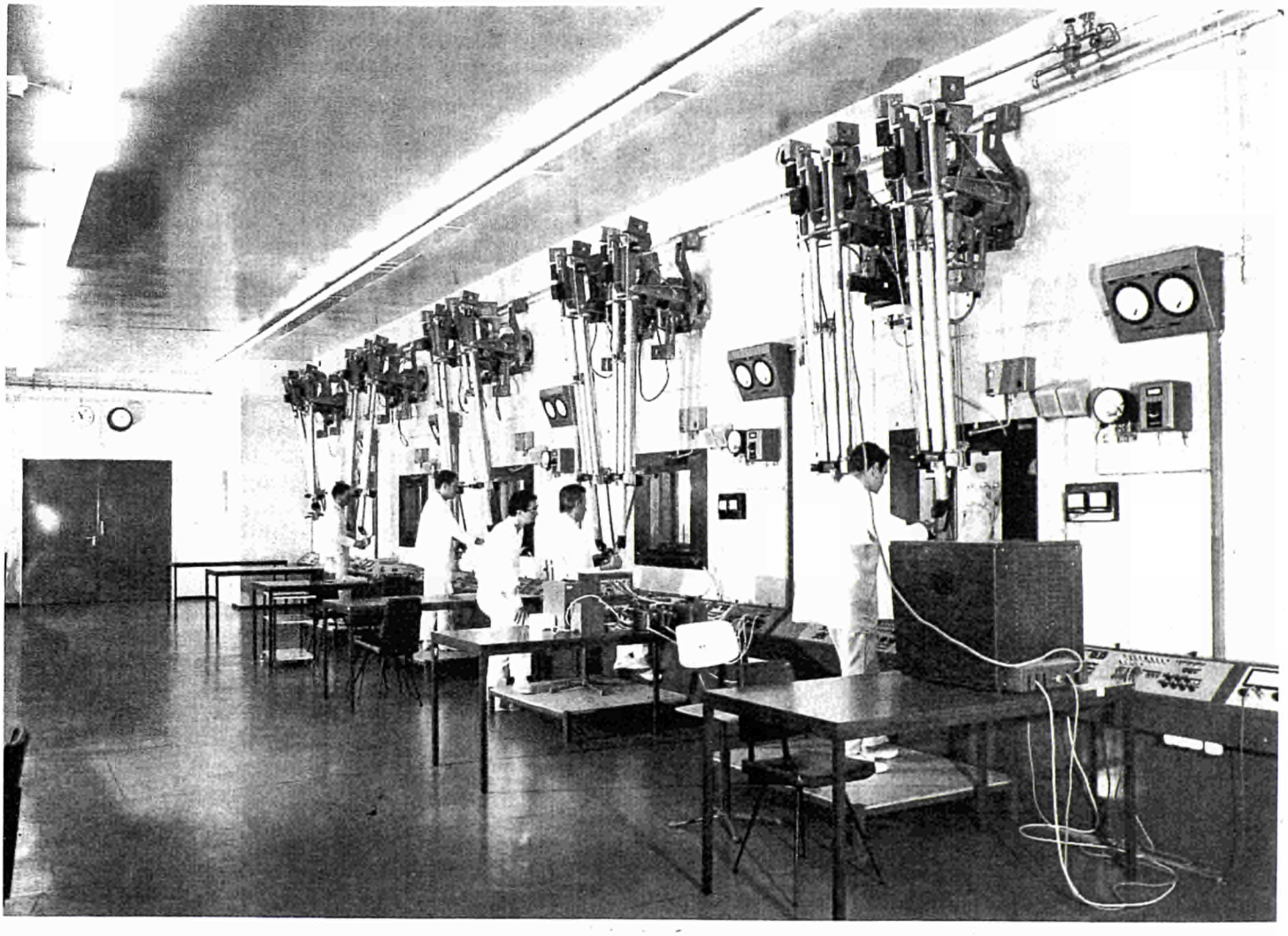
The basic research effort has been confined to a few special subjects of immediate technological interest. Other lines of research could have been developed, but there was a risk that too widespread a field of work would endanger efficiency. Obviously the amount of oxide fuel produced, which was enough to cover the actual requirements of the Institute's

irradiation programmes, is only a tiny fraction of the output to be expected from industry in future years. Nevertheless, emphasis was given to processes that can, practically, be extrapolated to the industrial scale.

As regards irradiations, the effort has been cruelly handicapped by financial and technical limitations: unfortunately the world has too few fast neutron test reactors where fuel elements for fast power reactors can be irradiated, and they are therefore very expensive to use. It is certain, however, that there will be more fast facilities available later, so that the situation will improve.

The high-activity hot laboratory, doubly shielded against contamination from alpha emitters and against gamma radiation, is an invaluable tool for the Community. It rounds off the experimental possibilities of the Transuranium Institute for the study of plutonium-oxide based fuels for fast neutron reactors; yet its usefulness extends beyond this subject, and a great part of the examinations carried out in this laboratory concerns the study of thermal neutron reactor fuels. (EUBU 7-1)

Fig 9: The chemistry line in the hot laboratory—European Transuranium Institute.



Are heavy-water reactors on the way out?

ABRAHAM BAHBOUT, *ORGEL Project Directorate*

FOR ANYONE INTERESTED in the fate of heavy-water power reactors, 1967 has been particularly rich in events. First and foremost, there was the pigeonholing of the American *HWOCR* development programme, and a few months later the Spanish shelved their *DON* project. No less important are the power run-up tests at the Douglas Point and Brennilis (*EL 4*) power plants. Then the Winfrith *SGHWR* power plant and the *ESSOR* test reactor at Ispra went critical, and lastly we had the decisions on the building of the *CIRENE* prototype in Italy and of two other big power units at Pickering, in Canada.

In a general context particularly affected by the recent advance of light-water reactors in the United States, it seems timely to have another look at the heavy-water reactors and review their present state and prospects.

Five families . . .

At the present time there are five different families of heavy-water reactors being developed or built throughout the world. Four of them are of the pressure-tube type, differing from one another by the nature of their coolant—pressurised heavy water, boiling light water, pressurised gas, or organic liquid. The fifth is the pressure-vessel type, cooled and moderated by boiling heavy water, with a direct or indirect steam cycle according to the sponsors.

. . . four with pressure tubes

The reactors of the pressure-tube type offer well-known advantages. First, owing to their cellular structure, they lend themselves readily to an apparently limitless raising of the reactor unit capacity by the mere addition of power channels. This characteristic furthermore enables the core components to be renewed if necessary, thus affording a longer reactor lifetime, and permits individual access to each channel, so that a variety of fuel management systems are possible. In addition, the power channels can be connected up in sets, to independent primary circuits, thus limiting the consequences of an accidental coolant loss through rupture of one of the circuits. Lastly, because of the leaktight containment of the coolant and the fact that it is separate from the moderator, various types of coolant can be used, the moderator can be kept at low temperature and pressure, and its volume or chemical composition varied independently.

The drawbacks are also well known. The energy dissipated into the moderator by radiation or through the channel insulation, amounting to 6—7% of the fission power, is degraded and lost. Furthermore, a special circuit is needed to cool the moderator. Last and most important, the material used for the pressure tubes must have low neutron absorption and also good mechanical properties in hot conditions and under radiation.

Let us see how matters stand with the various pressure-tube families.

Cooling by pressurised heavy water

The chief advocates of the pressurised heavy-water cooling family (see Fig. 1) are the Canadians. The 20 MWe *NPD* power plant, completed in 1962, was the first one to be built, and was followed by the reactor at the Douglas Point 200 MWe power plant, whose power run-up is still going through tests. Construction of a power plant, at Pickering, equipped with several power units of 500 MWe each was commenced in 1965 (see Table 1). Power plants of 200 MWe are also being built in India and in Pakistan, by Canadian constructors.

The steam quality at the turbine inlet is fairly mediocre. On the other hand, the burn-up is relatively high for natural uranium fuel. The basic problem with this family lies in the investment in heavy water, of which the quantity must be cut to the minimum, and in the leakages. Numerous leakages of heavy water have indeed been recorded. However, the greater part of the leakage during normal operation is recovered by drying the air in the areas containing the heavy-water circuits.

The Douglas Point power run-up, which began at the start of 1967, has been extremely arduous, beset by such bugs as the failure of the primary pumps, with major heavy water losses, the holing of a calan-

dria tube due to the vibrations of a booster rod, frequent outages due to major variations in the coolant pressure, and so forth; but these incidents do not cast any doubts on the viability of the concept as such.

A reservation should be made, however, as regards the heavy-water leakages, which amount on average to 4 kg per hour of operation. As the recovery system, which has been tried out successfully at *NPD*, has not yet been installed at Douglas Point, it is still too early to drawn any final conclusions on the economic effect of the heavy water losses. (Even so, by extrapolation from experience with the *NPD*, the rate of irrecoverable losses must be around 500 g/hr, corresponding to a cost of 0.1 mills/kWh, to which must be added the cost of re-enriching about 3.5 kg/hr of heavy water. These figures do not include accidental heavy-water losses.)

Since heavy water that has been in a reactor contains tritium, another consequence of leakages is that protective clothing fitted with an internal fresh air circulation system has to be worn during maintenance work.

Setting aside the reservation as to the heavy water losses, which still threatens the family's future, the kWh to be generated by the Pickering reactors seems to be reliably competitive, judging from the tenders submitted for the first two reactors now being constructed. The main reason is still the very low fuel renewal cost, which amounts to 0.58 mills/kWh, thanks largely to an all-in fuel cost of $ 40 per kg. (We should remark, incidentally, that this low fuel cost will give the heavy-water reactors a longer economic lifetime than that of the enriched-uranium reactors.) The installed kWe cost is $ 232, indirect costs included. It is expected that this figure can be lowered to $ 137/kWe by 1980 by building several reactors on the same site, increasing the unit power per reactor, and so on.

In short, the family with pressure tubes and pressurised heavy water coolant has reached the marketing and large-scale construction stage. On the economic plane it easily stands comparison with the American light water reactors. The only shadowy patches are due to the heavy water losses. We shall see further on that this family of heavy water reactors is the only one to have reached this stage of progress. It should perhaps be borne in mind that it is the only one that has not been developed in Europe.

Gas cooling

Gas cooling cuts down the cost of investment in heavy water and the accompanying risks of leakage and contamination, and also offers the advantage of high temperatures and therefore high thermal yields. The net efficiency of a power plant equipped with this kind of reactor is diminished, however, by the low specific heat of the coolant and by the relatively high pumping power it needs.

The first construction is a 70 MWe prototype, named *EL 4*, at Brennilis, in France, which is cooled by carbon dioxide (see Fig. 2 and Table 2). Fuel for the first reactor charge consists of enriched uranium oxide with stainless steel cladding, but this is only a temporary solution. The *EL 4* power run-up, which started at the beginning of 1967, has been gradual, experiments being performed at every operating level. The power plant will not be finally brought into service until the beginning of 1968 at the earliest.

The chief difficulties so far have been due to the nature of the coolant—swelling, blistering and tearing away of elastomer seals in CO_2 at the ends of the channels; surprising behaviour of the CO_2, which reacts on components such as the fittings and seals; "hunting" (instability) of the blower impellers at low flow-rates. It was also found that owing to defective heat in-

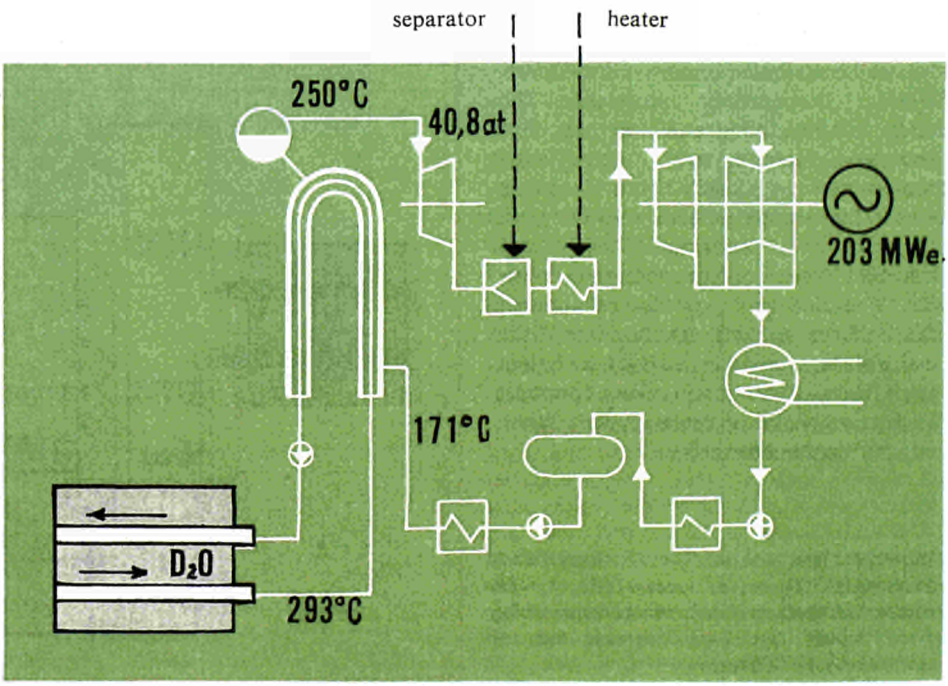


Figure 1: Flow scheme for a pressure-tube reactor cooled by pressurised heavy water (Douglas Point power plant). In all reactors of this family, the channels are horizontal-axis with hot pressure-tubes in Zircaloy-2. The vessel is of stainless steel and the calandria tubes are Zircaloy-2. The fuel consists of clusters of 19 or 28 UO_2 pins with a Zircaloy-2 cladding which, being thin, absorbs few neutrons. A rolled joint is used between the pressure tube and its extension. In the latest constructions, the primary circuit is of carbon steel and the main pumps are of the packing and leakage recovery type. To reduce the amount of heavy water needed, the coolant circulates in counter flow in two adjacent channels, thus dispensing with the need for a coolant return duct from one end of the reactor to the other.

Table 1: Main characteristics of the Pickering pressurised heavy-water power plant.

Net electric power	505 MWe
Average burn-up	9,500 MWd/t
Coolant pressure	112 atm
Steam temperature at turbine inlet	250°C
Steam pressure at turbine inlet	41.4 atm
Net overall efficiency	29.7%
Specific heavy water investment	0.81 t/MWe

sulation, heat losses from the channels were unexpectedly high. These troubles will presumably be overcome in time, and can be avoided in future constructions. The fundamental problem continues to be the fuel question.

Of the two alloys still being studied with an eye to cladding, zirconium-copper is the only one that still offers reasonable chances of success. A statistical experiment is necessary, however, and this can only be obtained in *EL 4*. But the performance of this cladding is limited by the internal hardening caused by diffusion of the oxygen from the UO_2 and by external hardening due to oxygen from the carbon dioxide gas.

It is not at present proposed to embark on the construction of a large gas-cooled power plant. Nevertheless, the extrapolation of the *EL 4* technique to large capacities has been studied under a *CEA-Sulzer-Siemens* co-operation agreement. The design adopted advocates the use of a prestressed concrete vessel, with as many components as possible built into the vessel. This would produce a very compact unit, and it is anticipated that a considerably lower coolant pumping power will be needed and that the containment can be dispensed with. On the other hand, new problems would arise with the balancing of the moderator and coolant pressures (as there would be no more pressure tubes, but only guide channels).

Cooling by boiling light water

The family of reactors cooled with boiling light water has a direct thermodynamic cycle. In this way there is a saving on heat degradation in the exchangers and on the cost of the latter. Furthermore, the concept relies largely, as to materials, on the experience acquired with the American water reactors and the pressurised heavy water family.

The chief problem with this method of cooling lies in the light water content of the power channels, which must be kept to the minimum for reasons of neutron economy. Moreover, if natural uranium is used as fuel, the limited presence of water gives the reactor a positive void fraction (see *Euratom Bulletin*, Vol. IV (1965) No. 2, p. 60), which affects the regulating and safety of the installation as a whole.

The first achievement in this line was the building at Winfrith Heath, England, of the 100 MWe *SGHWR* power plant, which went critical last September (Fig. 3). For this first installation, the British preferred to base their design on a negative or nil void fraction, which is obtained at the price of under-moderation of the lattice and enrichment of the fuel. Of the 112 core channels, 104 are boiling and 8 are superheat channels. The power plant's net efficiency will be 31.6%, thanks partly to the contribution from the superheat channels.

A mine of information can be expected from operation of the *SGHWR*, but it will not answer all the stability and safety questions raised by the natural or very slightly enriched uranium version of this reactor. Indeed, it is hard to see where the *SGHWR* will fit in, fuelled as it is with an enrichment not much different from that of a light water reactor. The project sponsors have in fact started research on a natural-uranium variant.

The Canadians are ahead of them in this field, having begun construction in 1966 of a 250 MWe prototype at Gentilly, in the province of Quebec *CANDU-BLW*—Ta-

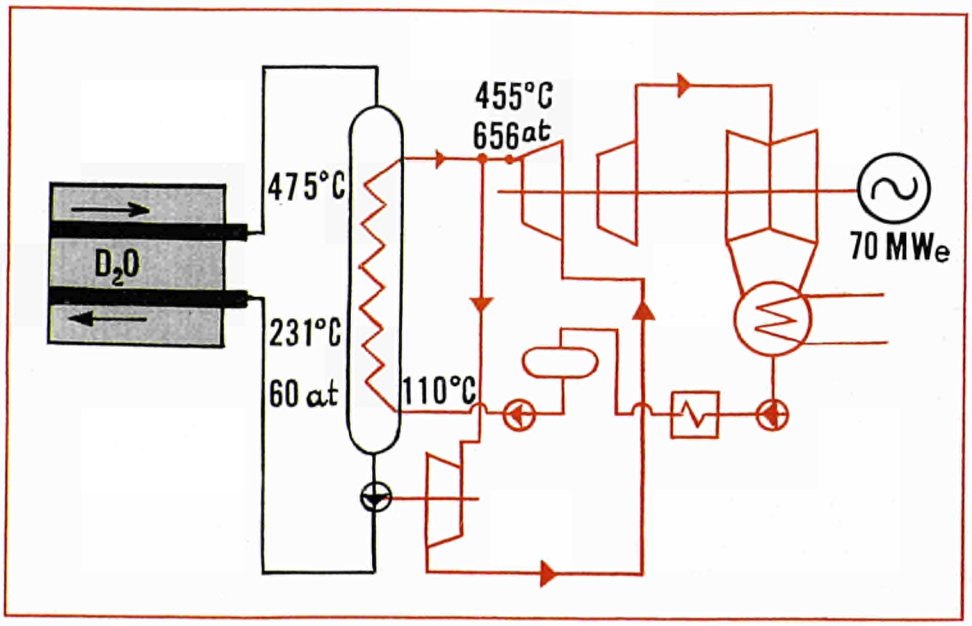


Figure 2: Flow scheme for a heavy-water moderated, CO_2-cooled reactor (EL 4). *The reactor is horizontal and the pressure tubes, of the "cold" type with internal heat insulation, are of Zircaloy-2.*

Table 2: Main characteristics of EL 4.

Net electric power	70 MWe
Coolant (CO_2) pressure	60 atm
Steam temperature at turbine inlet	455°C
Steam pressure at turbine inlet	65.6 atm
Net overall efficiency	26.2%

Table 3: Main characteristics of the CANDU-BLW-*250 reactor.*

Net electric power	250 MWe
Average burn-up	7,000 MWd/t
Steam pressure at turbine inlet	53 atm
Steam temperature at turbine inlet	265°C
Net overall efficiency	31.3%
Specific heavy water investment	0.85 t/MWe

ble 3). The reactor power is held at its set value by a control rod, whose response speed, incidentally, does not seem to be any faster than that used in the Douglas Point reactor. The thermal inertia of the UO_2 fuel slows down the coolant density perturbations due to power disturbances and leaves sufficient time for the control system to act and forestall the void effects.

The coolant circuit is of ordinary steel, as the water is conditioned with ammonia. The power plant is scheduled for commissioning in 1971. In future generations of this family, it is hoped one day to use a fuel based on uranium silicide (U_3Si) whose density overall would be 30% greater than that of UO_2. The outlet steam quality will also be raised gradually, with superheat as the possible goal. For this purpose a zirconium-base cladding resistant to hydride formation would be developed.

Japan too has decided to develop a family of advanced converters of the *CANDU-BLW* type, but oriented more specifically towards plutonium recycling, to improve the economy of the concept. In addition, the effect of the plutonium on the reactor void fraction would tend to improve the stability and safety of the installation. A preliminary design for a 200 MWe prototype has been completed, and work should begin shortly on the detailed design. Construction of the prototype should start in 1969 and operation is planned for 1974. We should mention that the research and development work still has to begin.

In the Community, we must note the decision taken by Italy in 1967 to build, on the Latina site, a 35 MWe prototype of the *CIRENE* family which, apart from a few details, is identical with the *CANDU-BLW* family. The *CIRENE* family development programme has been the subject of a joint Euratom-*CNEN-CISE* effort since 1963. One can but deplore the small size adopted for the *CIRENE* prototype, which may be

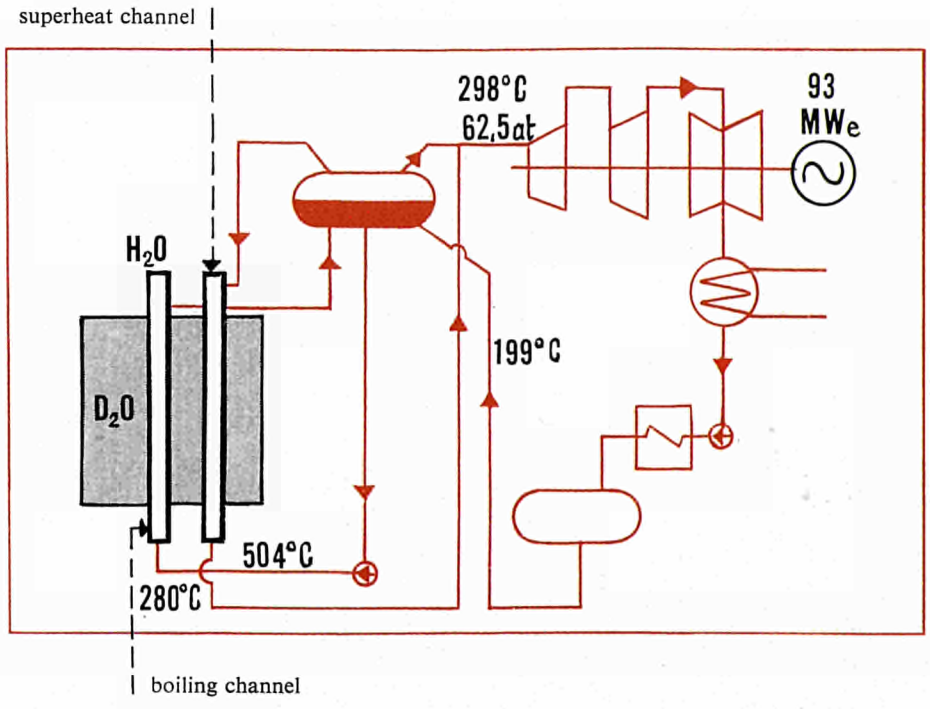


Figure 3: Flow scheme for a heavy-water moderated, light-water cooled reactor (SGHWR). *The* SGHWR *has a vertical core, composed of 112 channels (104 boiling and 8 superheat). The vessel, like the calandria tubes, is made in an Al-Mg alloy. The hot pressure tubes are of Zircaloy-2. The boiling channel fuel consists of clusters of 36 UO_2 pins with Zircaloy-2 cladding and an equilibrium enrichment of about 2.3%, giving a maximum burn-up of 18,000 MWd/t. The steam quality at the boiling channel outlets will be 13%. A natural-uranium version of this formula is under construction at Gentilly, in the province of Quebec. This reactor, designated* CANDU-BLW-*250, is also vertical. The fuel consists of clusters of 19 natural UO_2 pins. The pressure and calandria tubes are of a zirconium alloy. The average steam quality at the outlet is 16%, and the burn-up is about 7,000 MWd/t.*

termed an anachronism in the present-day nuclear context.

Cooling by organic liquids

With organic cooling the energy produced by the reactor can be carried off at a relatively high temperature level, yet at low pressure and with a negligible pumping power (Fig. 4). The neutron balance is favoured by the use of uranium carbide as fuel, which, together with a SAP cladding, also lends itself particularly well to high specific powers.

This family has been studied in major research and development programmes in Canada, the United States (*HWOCR* project) and at Euratom (*ORGEL* project). Two test reactors, *WR 1* in Canada and *ESSOR* at Ispra, have been built for that purpose. The reference reactor for this family (Table 4) is a vertical-axis type with hot pressure tube of SAP or Ozhennite 0.5, a zirconium alloy which offers better resistance to hydride formation than the other commercial alloys. The fuel would consist of clusters of seven UC pins with a finned SAP cladding. Irradiations of this type of fuel in the Canadian *NRU* and *WR 1* reactors have achieved burn-ups of 10,000 and 16,000 MWd/t without any trouble. The UC swelling seems to be substantially less than was forecast.

The coolant technology is now thoroughly mastered, and there are known methods to keep fuel fouling within acceptable limits. The economic incidence of coolant decomposition through radiopyrolysis is assessed at 0.2 mills/kWh, assuming that the decomposition products are not regenerated.

Yet, in spite of all these positive results, no one is planning to build a prototype reactor of this family. In March 1967, the *USAEC* decided to postpone *sine die* the construction of a large *HWOCR* prototype, and all activities connected with this family, apart from the irradiations of SAP-clad UC in the Canadian reactors, were halted in September 1967.

On the other hand the *USAEC* intends to pursue a very extensive programme of design studies and development on heavy water reactors, centred on water cooling.

As for the *ORGEL* project, an industrial group of Community enterprises has been working, since the autumn of 1967, on the

Table 4: Main characteristics of a large ORGEL *power plant.*

Net electric power	475 MWe
Average burn-up	8,000 MWd/t
Coolant pressure	20 atm
Coolant temperature at core outlet	400°C
Steam temperature at turbine inlet	385°C
Steam pressure at turbine inlet	112 atm
Net overall efficiency	35.8%
Specific heavy water investment	0.56 t/MWe

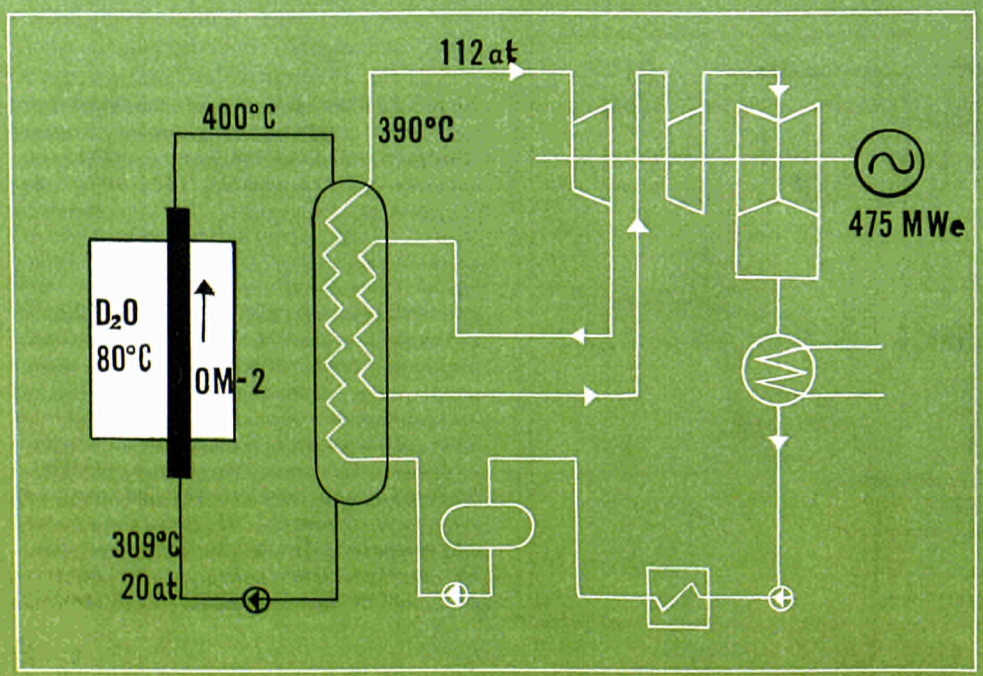


Figure 4: Flow scheme for a heavy-water moderated, organic-cooled reactor (ORGEL).

design of a 250 MWe prototype; this is to be completed, and accompanied by a firm-price tender for construction, by the end of 1968.

Pressure-vessel reactors

Technologically, the pressure vessel reactors are comparable to the American light water reactors. The first vessel reactors, *Ågesta* and *MZFR*, were of the pressurised heavy water type. The operation of the 65 MWth mixed power plant at *Ågesta*, in Sweden, since 1964 has been remarkable from several points of view. The heavy water losses, moderator degradation, primary circuit activation, and frequency of spurious reactor scrams have all been remarkably low. The *MZFR*, at Karlsruhe, appears to function normally if we discount certain bugs such as the failure of the primary pumps, which involved losses of heavy water.

Table 5: Main characteristics of the Marviken power plant (for both operating styles, with and without superheat).

	Saturated steam	Superheated steam
Net electric power	132 MWe	193 MWe
Burn-up	13,000 MWd/t	13,000 MWd/t
Coolant pressure	49.5 atm	49.5 atm
Average moderator temperature	170°C	170°C
Steam temperature at turbine inlet	259°C	472°C
Steam pressure at turbine inlet	47 atm	41 atm
Net overall efficiency	28.5%	32.5%

But the *Ågesta* and *MZFR* sponsors consider that the future of this family lies in boiling. Since 1964 the Swedes have been building the Marviken power plant (Fig. 5 and Table 5) which has boiling heavy water, natural circulation and a direct steam cycle. To keep down the heavy water investment, they have had to dimension the various steam circuit components in minute detail. Special precautions have been taken to minimise leakages of heavy water vapour and the ingress of light water. Marviken is due to come into service in 1969.

When it comes to extrapolating the Marviken design to large capacities new problems arise. The biggest, perhaps, is connected with the natural coolant circulation which, in a large reactor, may lead to flow instabilities between parallel channels, sufficient to impose a ceiling on the reactor power.

This problem can be got round by adopting a forced circulation system, as has been done by the *CEA*, *Sulzer*, *Siemens* and *SOCIA* in a joint study programme on 600 MWe power plants (Fig. 6 and Table 6). The forced circulation, however, seems to be accompanied by a positive void coefficient, and the stability of such a sys-

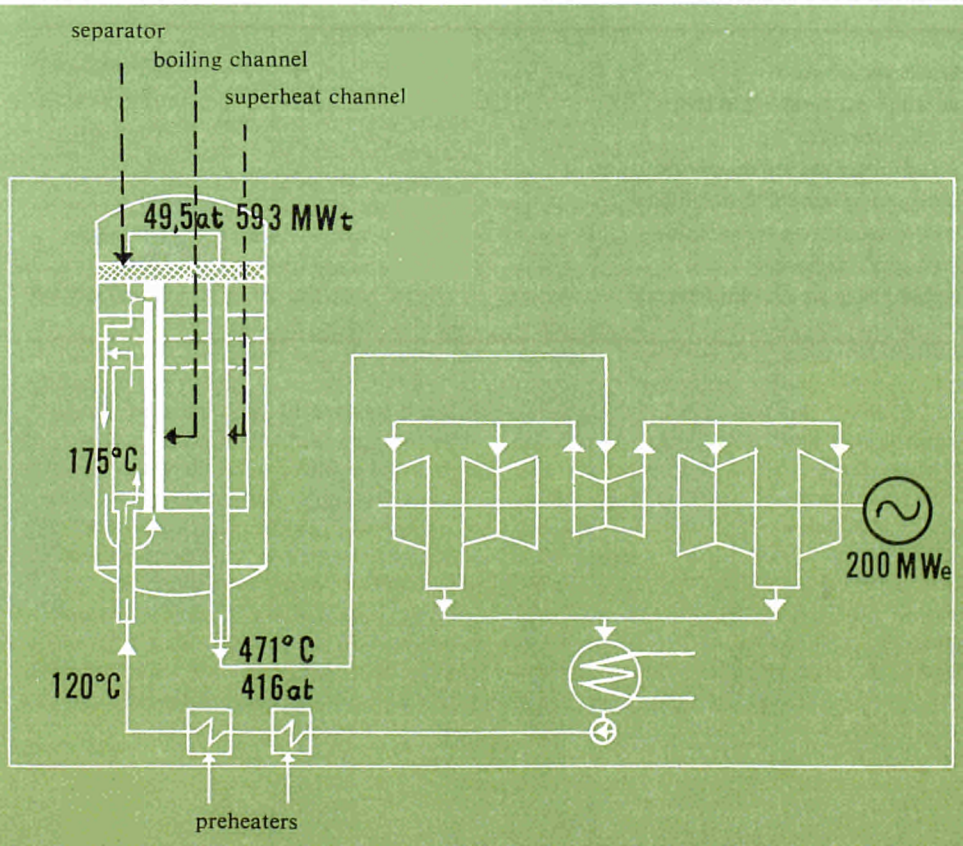


Figure 5: Flow scheme for a boiling heavy-water reactor (Marviken power plant, Sweden). The reactor is equipped with both boiling and superheat channels. The boiling channel fuel consists of Zircaloy-2 clad UO_2 clusters, housed in Zircaloy-2 guide tubes. The superheating fuel clads are of Incaloy.

tem would probably resemble that of a *CANDU-BLW*. The effects on safety are nevertheless less serious, because in a pressure vessel reactor draining of the channels always entails dumping of the moderator, with resultant reactivity loss. The reactor design provides for integration of the core, the fuelling machine and the heat exchangers in a prestressed concrete vessel. The thermodynamic cycle is indirect.

"You must hang together lest you hang separately"

What are the main conclusions to be drawn from this review? The first, unquestionably, is that in Europe and certainly in the Community, heavy water reactors are still at the development stage, unlike the Canadian family with pressure tube and pressurised heavy water cooling (Pickering plant). The position is that:

— *EL 4* will have no successor until a viable, statistically tested fuel has been developed. This will take several years.

— The *ORGEL* and *CIRENE* lines have not yet had the benefit of any industrial construction; any prototypes of these families will not be ready for service until 1972 or 1973 at the earliest.

— The *SGHWR*, which went critical last September, must first prove its worth. In its present version it can hardly be considered for building in the Community, first, because of the initial enrichment of its fuel, and secondly because, as the sponsors themselves say, it is intended for the 300-350 MWe power range.

— The Marviken power plant will not be operational until near the end of 1969 by the most optimistic reckoning. As we saw above, direct extrapolation to powers of the order of 600 MWe is ruled out. The forced circulation version studied by *Siemens* comprises such novel factors that it would require a substantial back-up development programme should the construction of a 600 MWe unit be decided without going through an intermediate stage.

In other words, heavy water reactors in Europe have not reached the industrial and marketing stage. One may be tempted to wonder, in view of the foothold gained by light water reactors, whether it is worth going on with the development of heavy water reactors. In other words, let us look

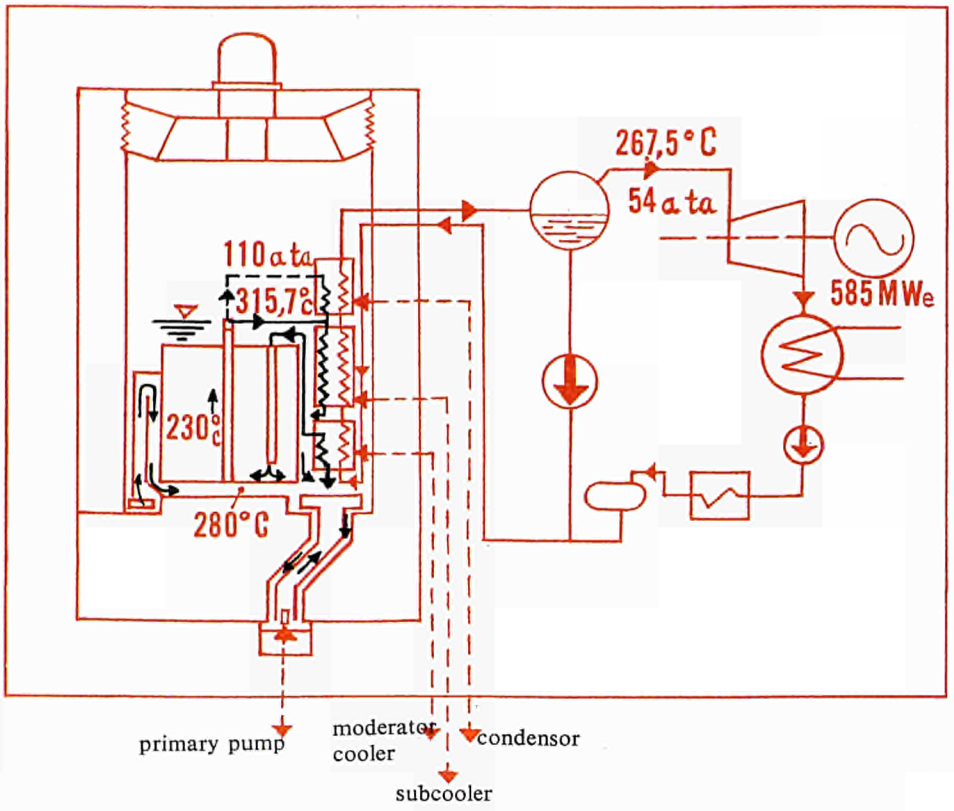


Figure 6: Flow scheme for a forced-circulation boiling heavy-water reactor (BHWR). *The fuel consists of clusters of 36 UO_2 pins with Zircaloy-2 cladding. The average steam quality at the reactor outlet is 10%. The insulation recommended for the concrete vessel is of the gaseous type.*

Table 6: Main characteristics of the boiling heavy-water reactor (BHWR) *designed by* CEA, Sulzer, Siemens *and* SOCIA.

Net electric power	585 MWe
Average burn-up at extraction	8,500 MWd/t
Coolant pressure	100 atm
Average moderator temperature	230°C
Steam temperature at turbine inlet	274°C
Steam pressure at turbine inlet	60 atm
Net overall efficiency	32%
Specific heavy water investment	0.62 t/MWe

at the root of the question: what trumps do the heavy water reactors still hold?

In the first place, we should remember one basic fact: the economy of the pressure-tube type cooled by pressurised heavy water is already completely viable today, as can be seen from the firm tenders for the construction of the Pickering power plant. In addition, Canadian and American studies have shown that the *ORGEL* and *CIRENE* families are even more promising from the economic standpoint, which means that the primary condition governing the construction of any type of reactor is fulfilled.

The standard trump—the use of natural uranium—may have lost some of its value in Western Europe with the spread of light-water reactors, but it still holds its full significance and power in a great many countries in Eastern Europe, Asia, Africa and Latin America. For that reason, if the Community hopes one day to win a share in those potential markets, she will do so far more easily by offering a heavy-water, rather than a light-water reactor; moreover in the latter case, she would find herself competing with the American producers.

But there is one extremely powerful card which should ensure a long life for the heavy-water reactors. By using the thorium cycle, these reactors promise to solve the problem of long-term energy resources in the same way as the fast breeders, but with less technical hazards and uncertainties as to economic profitability. It would, to say the least, be prudent not to stake everything on the adoption of fast reactors and to "hedge" with an alternative device the development of which would not call for an unduly heavy outlay.

Lastly, let us not forget that the use of natural or very slightly enriched uranium in a heavy-water reactor offers the best safeguard against possible U_3O_8 price increases in the future.

It must be remarked that these various trump cards do not all count equally in the eyes of every decision-making body concerned in the promotion of nuclear power: the economic aspects appeal principally to electricity producers, the export prospects affect nuclear power plant constructors, and the thorium cycle is first and foremost a matter for the State or parastatal bodies responsible for the development of nuclear power. The whole hand, however, adds up to a total of vital importance on the national or Community scale. In other words, we consider it imperative to continue promoting the heavy-water reactors.

We come now to the second question: how? The chief factor to be reckoned with is time. We do not think it is too late to launch out into heavy-water reactors, provided that not much more time is lost. A first requisite is that there must be no delay in starting the building of prototypes, and the second is that the prototypes must be big enough, of the order of 300 MWe, so as to speed up the process. A third essential is close collaboration between the various groups who are responsible for promoting the different heavy-water reactor families, and effective co-ordination of their efforts.

We can think of two steps that would be useful. The first is to set up a Community-scale permanent conference of sponsors, project heads, and heads of heavy-water reactor sub-assemblies, with an efficacious technical secretariat. One of the conference's first tasks would be to carry out a comparative study of the various families of heavy-water reactors with a view to reducing the number of prototypes to be constructed. The second would consist in having these prototypes designed and constructed by industrial groups, preferably multi-national so as to take advantage of all the experience that has already been acquired in the Community.

Which families should be kept and which dropped? Actually, a certain amount of natural selection has already taken place in the Community. The pressure-tube, gas-cooled family has reached a point where, depending on the fuel development programme, either there will be no successor to *EL 4* or, if there is, it will be a case of building a high-power unit under conditions of pure economic viability, without any resort to subsidies. That leaves three families in the field, *ORGEL*, *CIRENE* and the Siemens *BHWR*. For the Community this is not an excessive list, but it could be further simplified on the basis of a valid comparison of the various families, taking into account their state of progress, the efforts still needed, their prospects and so on. This task in particular would be rendered easier by the existence of a permanent conference of the promoters of heavy-water reactors. (EUBU 7-2)

Concrete

The technology of reactor pressure vessels in prestressed concrete is currently developing rapidly in the European Community. A conference attended by 250 experts was held in Brussels during November to take stock of the present situation.

A FEATURE OF THE rapid development of nuclear power is the trend towards very large power plants. While it cannot be said that technical possibilities of steel pressure vessels have already been exhausted, it has to be admitted that their size and weight present increasingly difficult fabrication and transportation problems.

The technique of prestressing concrete with cables was discovered almost 90 years ago and has been used for many purposes for some forty years now: large buildings, bridges and structures, chimney-stacks, sewers, storage tanks, etc. It was only natural that the authorities responsible for nuclear development work in Europe should turn their attention to this technique, which is particularly well suited to the large power plants typical of the graphite-moderated gas-cooled family adopted in Britain and France. A comparison of steel and prestressed concrete for use in the construction of pressure vessels soon showed concrete to have the edge. This judgment was naturally reinforced as dimensions and capacities increased. Moreover, because of their intrinsic safety, concrete vessels require no second outer containment, which is normally essential for reactors with steel pressure vessels. In the case of 1,000 MWe power plants this secondary containment costs several million units of account.

Prestressed concrete has therefore quickly become a matter of general interest in the nuclear field, with potential applications far beyond the graphite/gas family which first used it. Euratom has been interested in it for several of its programmes: first, proven-type graphite/gas reactors and then boiling-water and high-temperature gas reactors. Its contribution, albeit limited, has nevertheless been an important stimulus, notably to the development of new methods and the formulation of new concepts.

Two-day conference

During the two-day conference held in Brussels on 7 and 8 November all the work in this field carried out under Euratom contract was reported on to a very wide European audience. The results achieved by industrial firms not under contract to Euratom were also described: this extension of the discussions to cover all the studies in progress was one of the factors which ensured the success of the conference.

A notable feature of the meeting, which covered not only the actual reactor pressure vessels in prestressed concrete but also their thermal insulation, was the wealth of ideas marking the general approach to this complex problem. The object was not to classify ideas in order of merit or value, which would be presumptuous, but to submit them to the 250 experts from the six Community countries responsible for nuclear technological and industrial development.

Papers on concrete covered two aspects: concrete as a material and concrete as a pressure vessel component.

Concrete as a material

Interesting data on the mechanical properties, the thermal and irradiation behaviour of concretes of varying composition were provided by *SNAM Progetti* and *Bredero* with regard to concrete resistant to heat and thermal cycling. Discrepancies which would warrant closer investigation were noted in the results on barytes concrete.

Interesting initial results were obtained in research on heat-resistant concretes by the French Atomic Energy Commission (CEA). Since conflicting conditions have to be taken into account in their use, their

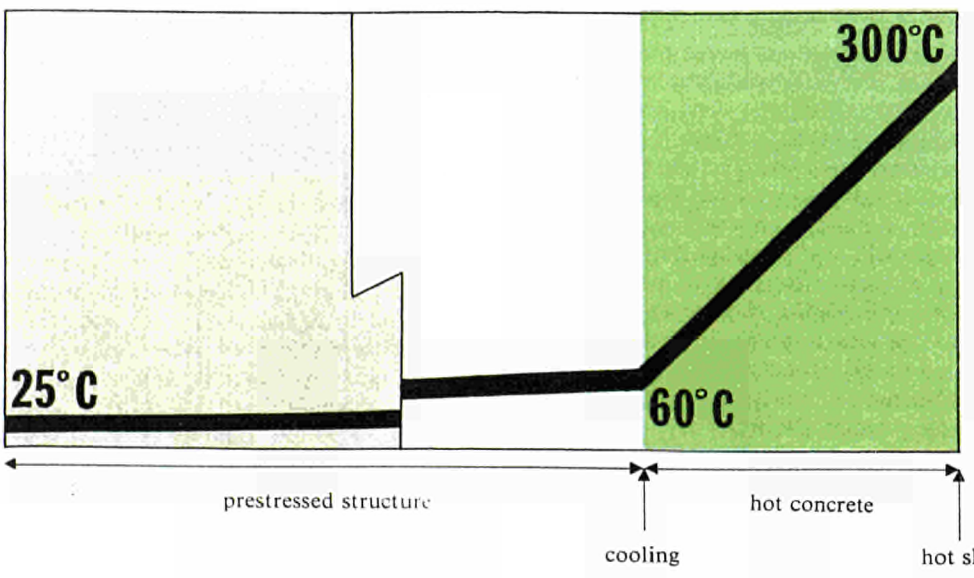


Fig. 1: Profile of temperatures through the wall of a "hot-skin" prestressed concrete pressure vessel (designed by SEEE*).*

composition has inevitably to be a compromise between the different requirements. There seems to be a trend towards concretes based on Super Secar cement, with vilmolith for the fines and globular alumina as the aggregate.

Concrete as a pressure vessel component

A striking feature of the conference was that all the papers dealt with new methods which represented a clear-cut departure from the conventional prestressing processes and the normal calculation of

It is particularly significant that a number of these studies have proceeded as far as the 1/20 to 1/5 scale model stage and that tests to failure of the structure have been carried out after measurements in normal and exceptional operating conditions in order to check the safety coefficients used in the calculations.

Each of these methods has original features but also raises problems, not all of which have been solved.

At the present stage of developments and despite certain detailed experimental studies, it is difficult to draw definite conclusions on the validity and industrial

30 centimetres of concrete in the vessel (Fig. 1). It can be used with any coolant.

Another attractive solution is the *CITE* "*overall prestressing*" method, consisting of prestressing by a jack, first liquid during the initial phase of relaxation of the tendons and creep of the concrete, then solid by the injection of a cement grout which sets hard (Fig. 2). Since the concrete of the vessel is always under compressive stress, it cannot crack and the tensile stresses are transferred to a well-reinforced outer zone.

The "*multi-layer*" vessel designed by *Friedrich Krupp Universalbau*, in which each layer has a separate function, is based on

reactor pressure vessels - an assessment

PAUL FERNET, *Directorate-General for Industry and Economy, Euratom*

containments. This is readily explained by the fact that the dimensions of the cavity and the temperature and pressure in the new designs have already reached the limits allowed by conventional prestressing techniques. The object of both publicly and privately financed research now in progress is to push back these limits by means of a fresh approach to the problem.

Hence projects have been submitted for pressure vessels of the following types:

a) "hot skin" prestressed concrete *(SEEE)*;
b) with "overall prestressing" *(CITE)*;
c) in "multi-layer" prestressed concrete, each layer having a separate function *(Krupp)*;
d) in prefabricated units *(Siemens)*;
e) in two-layer prestressed concrete *(CITRA/Société Générale d'Entreprises)*;
f) in prestressed concrete for high-temperature gas reactor *(BBK)*;
g) in prestressed concrete with a metal cover *(SEEE)*.

applicability of any of these methods. Many more tests have still to be carried out in the majority of cases.

The wealth of ideas which has resulted in the concepts examined is a particularly encouraging sign of the technological vitality of Community firms and their potential in this advanced field.

Provided that the considerable problems to do with the long term behaviour of the leaktight metal skin can be overcome, the *SEEE* "*hot skin*" method appears to be an extremely simple and elegant solution, in which the function of the internal thermal insulation is carried out by the first 20 or

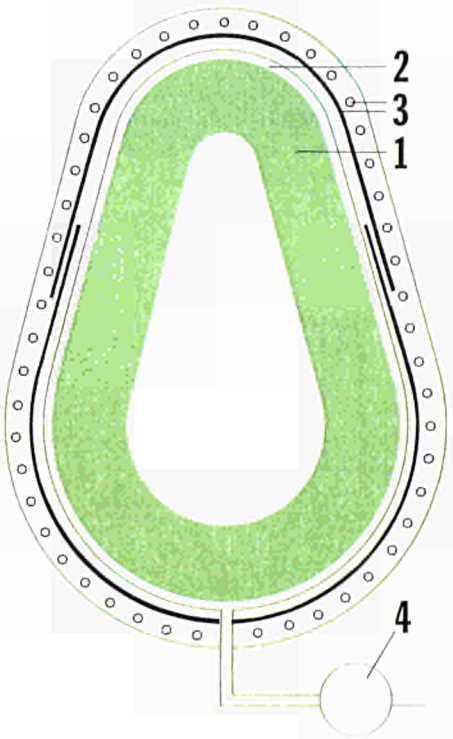


Fig. 2: Diagram of principle of the "overall prestressed" concrete pressure vessel made by CITE.
1 Prestressed zone
2 Leaktight cavity
3 Tightened tendons
4 Tightening device

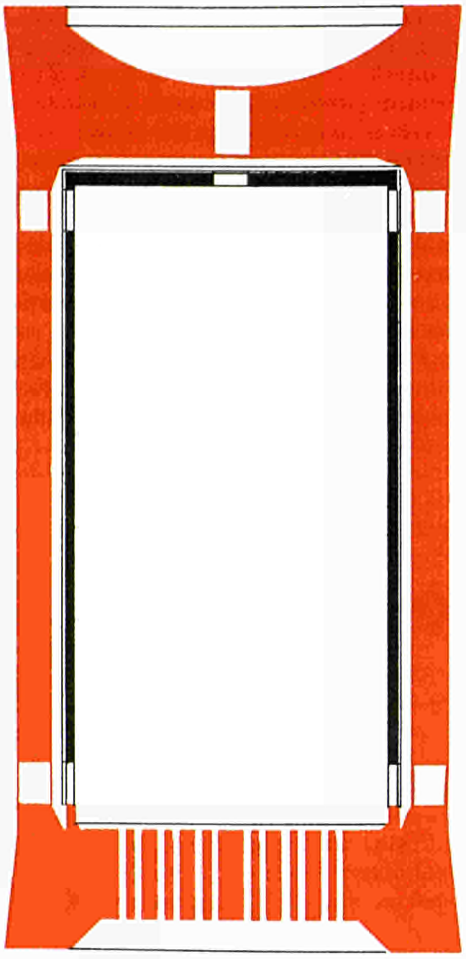


Fig. 3: Plan of the multi-layer concrete pressure vessel proposed by Friedrich Krupp Universalbau, *in which the different layers have separate functions.*

a different principle (Fig. 3). Starting from the central cavity, a first layer of special hot concrete serves as a support for heat-resistant refractory concrete based on clay or expanded ceramic material developed by *Krupp*, which is easy to mount on site. These first two layers are surrounded by a film of cold water at the same pressure as the reactor coolant. The conventional prestressed concrete outer containment is thus protected from all thermal stresses and its design is hence simplified.

Siemens have made a model of a vessel consisting of small *shop-made prefabricated*, and therefore very uniform units which can be made under the best conditions for stabilising their dimensions, allowances being made, in particular, for shrinkage variations. The blocks are assembled on site, concrete being poured into the cracks left between them for the purpose, after radial prestressing by jacks. Special techniques have been studied for the bottom end and upper slab.

The use of a *two-layer vessel*, as designed by the *Société Générale d'Entreprises* and *CITRA*, prestressing at points only where strictly necessary, represents a considerable saving in steel and concrete. In the examples given in the paper the authors estimate the saving at between 20 and 50%, or even 70%; thus it would be possible to design large or high-pressure vessels, the thickness of which would be prohibitive in a conventional design. Further work still has to be carried out on the heads.

The design for a 300 MWe *high-temperature gas reactor vessel* was submitted by *Brown Boveri/Krupp* and tested on a 1/20 scale model by *ENEL/ISMES*. It is a conventional design with a high safety factor.

SEEE, using a prestressed concrete cylindrical body in conjunction with a steel cover (Fig. 4), proposes an interesting solution for lightwater reactors, giving very easy access to the core for loading and unloading as with all-steel vessels. The trickiest problem, obviously, is how to attach the steel cover to the concrete wall, and solutions have been put forward.

Considerable progress has been made in recent years on thermal insulation and shielding, in both the fundamental and the applied fields.

Fundamental research on thermal insulation

Several contracts have been concluded

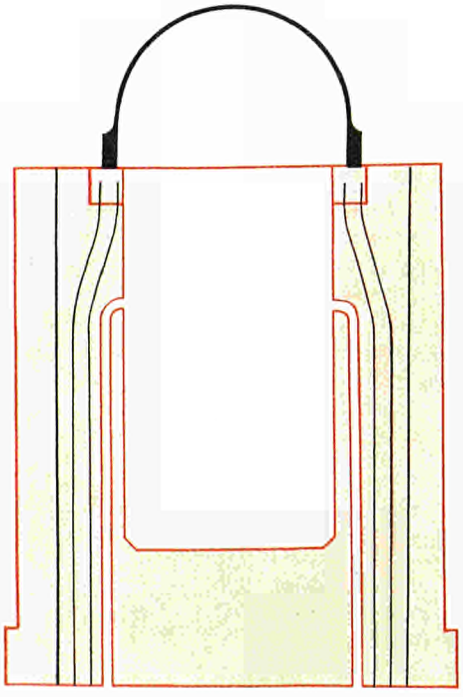


Fig. 4: Plan of the prestressed concrete pressure vessel with metal lid, proposed by SEEE.

Fig. 6: The Bertin *thermal test rig, for studying the phenomena of natural convection in confined spaces. It consists of three main components—the mock-up (centre), a control console and an assembly where the water flow-rates in the coolant circuits are measured. The mock-up consists essentially of two symmetrical cells situated on either side of a heating plate.* ►

Figure 5: Natural convection rendered visible.
The photograph shows a closed cell filled with glass wool soaked in chlorobenzene. The right-hand vertical surface is heated, while the left-hand one is cooled. The heat transfer between these two sufaces takes place essentially by natural convection in the chlorobenzene. The dark area shows the progress of a thread of coloured liquid injected by syringe a minute earlier along the vertical line seen in the centre of the picture. The photograph reveals a turning movement: there is a rising boundary layer on the heated side and a descending one on the cooled side, whilst at the centre of the cell the liquid is practically motionless.

with *Bertin et Cie.* covering studies of natural convection phenomena in confined spaces and on a flat wall. The various possible flow modes were demonstrated and the correlations for plotting them worked out. The transition zones between the different types of convection flows were determined accurately. At the same time the associated thermal phenomena were studied and represented mathematically (Figs. 5, 6 and 7).

These studies were a very important stage in the collection of information necessary for optimising the design of cellular thermal insulants in CO_2 flow. They are also of great interest for forecasting the performance of the thermal insulating material in other fluids. They led to better understanding of the phenomena responsible for heat transfer between metal walls separated by a fluid film, as in foil or cellular insulants, and hence enable them to be better controlled. Once the results for single cells are known they can be extrapolated for use in actual cases of cell assemblies, in which macroconvection phenomena are superimposed on microconvection in each individual cell.

Fundamental research on convection flows in fibrous media is in the hands of *Saint Gobain* and *Bertin* in collaboration with Euratom. The theoretical approach adopted and the measuring methods developed already guarantee the success of the research, the results of which will be as significant for fibrous or porous heat-insulating material as the above-mentioned research on cellular insulation.

It should be noted that, despite its scientific nature, this fundamental research has a well-defined practical objective. The close collaboration established between *Bertin*, public institutions and a number of large industrial concerns has favoured this combination of scientific research and practical development which is a guarantee of success.

Bertin owns or operates for third parties a number of thermal insulation test rigs, ranging from the smallest to the largest in the world.

Other research by *Bertin* has been on analy-

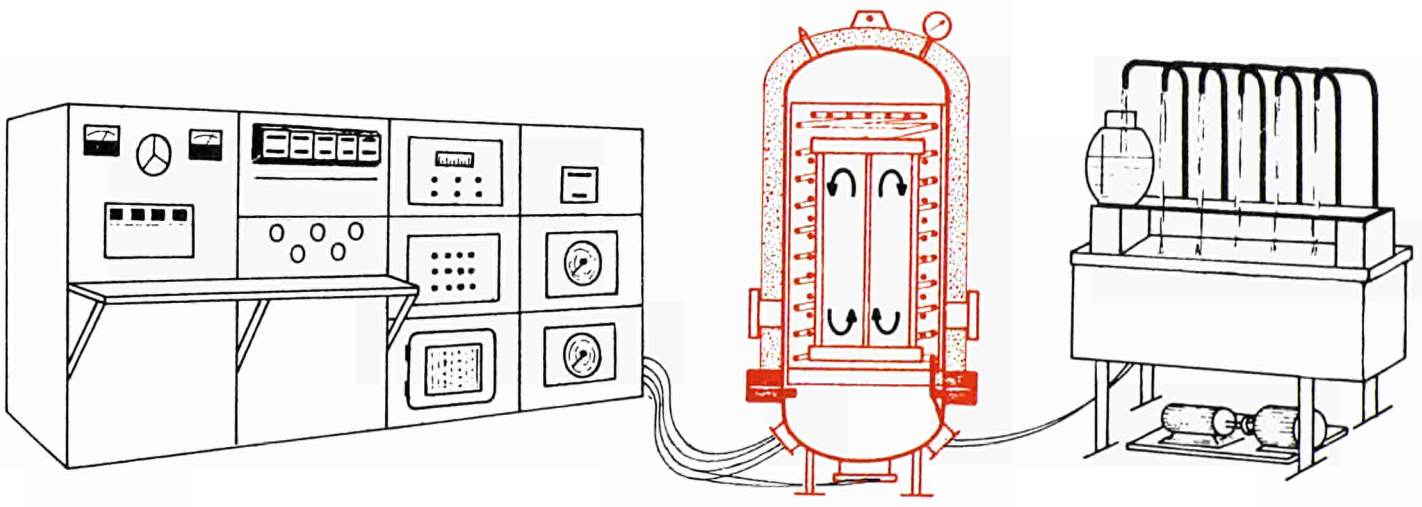


Fig. 7: Cylindrical mock-up for large-scale testing of heat insulation materials in the "Megabidon" at Chinon (France). The "Megabidon" is housed in the cavity at the bottom left of the photograph. In the background, the Chinon 1 nuclear power plant (EDF 1).

tical, graphic and numerical methods of calculating the cooling circuits of the leaktight skin, which removes the heat passing through the thermal insulation. This research clearly demonstrated that the optimisation of a thermal insulation system must be applied to all its component parts, i.e. insulation, leaktight skin and cooling circuit, as a single whole.

Industrial solutions to the insulation problem

Before the Euratom programme was launched in 1962 the types of thermal insulation employed industrially inside prestressed concrete reactor pressure vessels were very few in number: pumice concrete layer in France, *Darchem* metal foil in Britain. Since that date extensive research, in which Euratom has also played a part together with other organisations, has resulted in the development of several types of thermal insulation suitable for various families of reactors.

Special mention should be made of the "water shield" (Fig. 8) and "gas shield" methods in which the insulating structure and the heat removal system passing through it are combined inside the vessel. These methods have been developed by *SOCIA* and *Deutsche Babcock und Wilcox*. Various configurations have been studied and the most interesting from every point of view adopted.

Unlike conventional thermal insulation systems, where the heat is removed by channels embedded in the concrete and

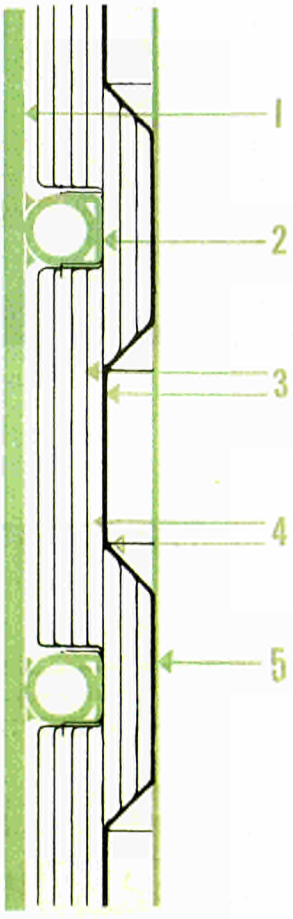


Fig. 8: Vertical cross-section of a panel for a "water screen" insulating system (SOCIA *and* Deutsche Babcock und Wilcox).
1 Leaktight skin
2 Cooling pipe
3 Metal sheets
4 Mineral wool pads
5 Casing

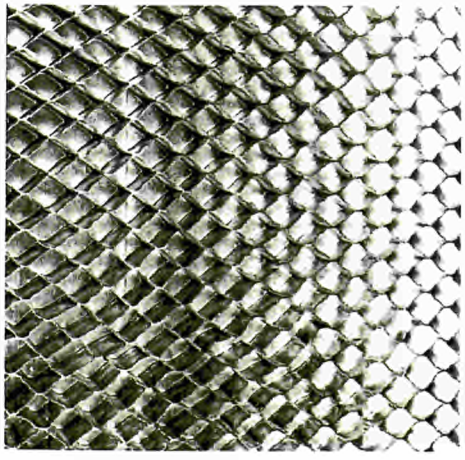


Fig. 9: Single panel of a stainless steel "honeycomb" insulating system (NIDA—*Sud-Aviation*).
The individual square or hexagonal cells, about 4-10 mm wide, are constructed by bending steel strips and welding them together. The prisms thus formed are closed at both ends by plates welded on to them.

fastened to the outside of the leaktight skin, the shield system is inside the cavity, before the leaktight skin. In the case of gas reactors there is an additional advantage in that the heat is removed by a cold gas flow at the same pressure as the coolant gas, thus stepping up the safety of the unit and simplifying the design.

Tests on models of representative sizes are performed in a large vessel under 40 bars pressure fitted with a blower (called "*Figaro*"—*Forschungsanlage für die Isolierung Gasgekühlter Reaktoren Oberhausen*).

The other existing types of thermal insulation developed in the Community and described at the conference are as follows:

a) Metal honeycomb structures;
b) Wire-gauze structures;
c) Glass-fibre-base structures;
d) Diamond-shaped cellular structures.

"*NIDA*", a structural material in current use in the aeronautical industry, is being studied by *Sud-Aviation* (Fig. 9). Its honeycomb structure, high mechanical strength, light weight, cleanness and malleability make it an intrinsically good material. The axes of the cells can be manipulated to lie parallel to the heat flux direction, which is best from the mechanical point of view, or perpendicular to it. Measurements made on small panels, while satisfactory on the whole, were subject to boundary effects and need to be made on larger panels to give definite conclusions. The practical application of the material is a further point requiring closer investigation.

A stainless steel wire-gauze cushion developed by the *Compagnie des Ateliers et Forges de la Loire (CAFL)* has an extremely low apparent thermal conductivity, which makes it a very effective device, even at low thicknesses, against heat transfer by conduction, radiation or convection. Tests made in a very large drum as well as specific tests on corrosion resistance gave extremely good results. The flexibility of the cushions facilitates their fitting to the walls which are to be insulated.

The *Compagnie de Saint Gobain* and *Saint Gobain Techniques Nouvelles* have developed a thermal insulation system in the form of a cellular structure of rectangular section, the free volume of which is completely filled with a fibrous product (glass or silicon fibre, for instance, depending on operating conditions).

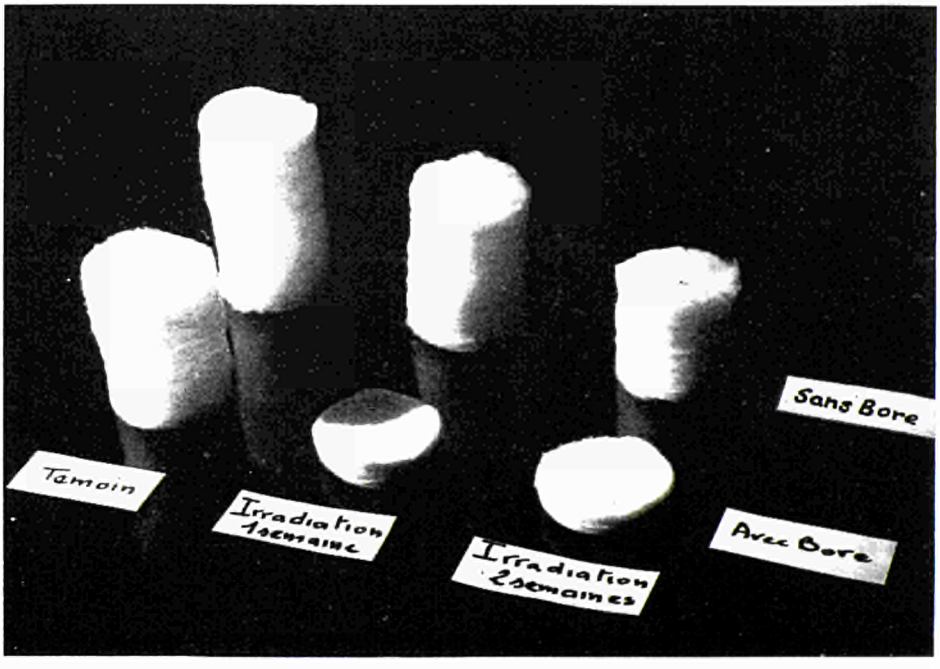
The mineral fibre industry has shown that it can produce insulating materials capable of withstanding the most severe operating conditions in reactors. Indeed, glass and silicon fibre specimens irradiated up to quite representative cumulative doses have shown no sign of embrittlement or dust formation (Fig. 10).

"*Ondulinox*" consists of corrugated metal foil enclosing elongated cellular spaces with a rounded diamond section. It has been developed by *Alsthom*. Series of tests were carried out in a number of installations, both in CO_2 and helium. The helium series was performed with a view to its possible use in a high-temperature gas reactor developed by *Brown Boveri/Krupp*. These general trial series do not show the relative influence of the different phenomena occurring and could usefully be supplemented by a more detailed study of heat fluxes in the cells and determination of the behaviour of an assembly of cells in parallel and in series.

All in all, if most of the methods described have not yet passed the test of industrial application, whether they concern pressure vessels or thermal insulation, some of them nevertheless appear capable of replacing the accepted methods in the near future and by so doing boosting technology and cutting the cost price per kilowatt-hour.

The Community's position in this sector is sound and even in advance of achievements abroad. Community industry has a chance for the future which it should not let slip. These advanced techniques will not reap their full reward, however, unless the industrialists concerned, i.e. users and manufacturers, work together to find the best practical solutions, once the necessary pruning has been done. The Commission is ready to offer its support, to the best of its ability, to all those needing it. (EUBU 7-3)

Fig. 10: Glass wool samples. Normally all glass contains boron, which detracts from its behaviour under irradiation. The mineral fibres industry has therefore been attempting to produce a boron-free glass.
The sample containing boron (bottom row in photograph) deteriorates badly during irradiation, whereas in the boron-free sample (top row), the deterioration is not only less pronounced but stops after a certain cumulative neutron dose.
In the meantime boron-free samples have been subjected to a special heat treatment; it was demonstrated that under those conditions no deterioration occurred.



SHOULD A LARGE isotope separation plant be set up in the European Community? With the increasingly clear-cut prospect of a large number of power reactors in Europe burning enriched uranium, this question is frequently being brought up. We shall endeavour in this article to provide the ingredients of an answer.

Viewed in the widest context, the installations of an isotope separation plant assumes the aspect of a cornerstone of a Community energy policy. The primary aims of such a policy as defined in official documents are, on the one hand, advantageous and stable prices in the long term and, on the other hand, dependability of supply.

We know that in the years ahead nuclear power plants will make a major contribution to the fulfilment of these aims. Not only will the cost of the kilowatt-hours which they produce be lower than in the case of conventional power stations, but also they will restrain and even reverse the trend which is making the Community increasingly dependent on vast-scale imports of fossil fuel, in particular oil.

The full significance of this latter consideration, however, would only become manifest provided the Community were able to cut its imports of nuclear fuel to a strict minimum. But it is not; its internal resources in fissile material account for only a small fraction of its foreseeable long-term needs.

Nuclear energy nevertheless makes a substantial contribution to dependability of supply, and this is attributable to a number of reasons, the chief of which are perhaps that uranium imports are obtained from countries other than those which supply oil and that the extremely small quantities involved, because of uranium's high energy value, limit the risks in the event of a shortage of shipping space.

A distinction must be made, however, between the case of enriched uranium and that of natural uranium. As regards the latter, the Community will be dependent on imports for only part of its requirements and, even if this is a preponderant part, there are still various ways of improving security of supply, for instance by boosting indigenous production and geographical spreading of imports. Enriched uranium, on the other hand, offers scarcely any possibilities of this kind and at the present time the Community has to rely upon deliveries by the *United States Atomic Energy Commission (USAEC)* to meet practically all its requirements. Subject to the appropriate measures being taken, this situation appears likely to persist more or less unchanged in the foreseeable future.

Among proven-type reactors, there is a marked preference throughout the world for those burning enriched uranium on account of the economic advantages which they offer. Where advanced-type reactors are concerned, the majority of the techniques currently being developed show an increasingly distinct tendency towards the use of enriched uranium.

Should the European Community produce its own enriched uranium?

HANS MICHAELIS, *Director-General, Commission of the European Communities*

The European Community's requirements

In the "First Target Programme", published by the Euratom Commission in April 1966, it was estimated that by 1980 installed nuclear power in the Community would be at least 40,000 MWe. With the advances which have taken place in the meantime, however, this estimate has been revised upwards to 60,000 MWe.

As regards the distribution of capacity between reactors using natural uranium and those fuelled on enriched uranium, the strict equilibrium between these two categories as envisaged by the "First Target Programme" would appear to be untenable.

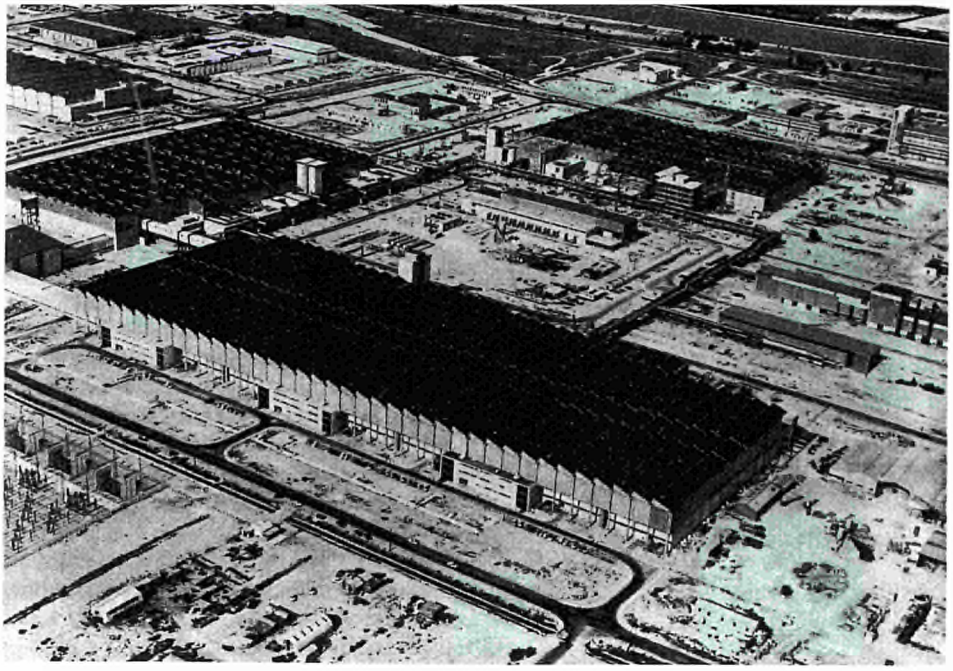
Having regard to this state of affairs, two alternative hypotheses have been considered, namely:

(1) all nuclear capacity commissioned between 1970 and 1980 will be of the light-water type, save for some projects already decided upon;

(2) two-thirds of this capacity will be of the light-water type, the remaining third being of the graphite-gas type or of another group using natural uranium[1].

These two alternative hypotheses are simple ones. They do not cover either advanced converters or fast breeders, since the share of these two reactor classes in nuclear energy output will certainly be

1. A different proportion has been applied further on in this article (page 25, col. 2) for the whole of the *OECD* area; allowing for the United States' share in the total, the ratio adopted is 3/4 enriched uranium—1/4 natural uranium.



Uranium isotope separation plant at Pierrelatte. In the foreground, the low enrichment plant; behind it, to the left, the medium enrichment plant.

quite small during the period under consideration.

If these two hypotheses are combined with the twofold estimate of installed nuclear power in 1980 (target programme: 40,000 MWe; upward revised programme: 60,000 MWe), we obtain a composite set of evaluations which should adequately provide for the various contingencies.

It should first of all be pointed out that under the aforementioned set of hypotheses, the quantities of natural uranium required during the period 1970-1980, for both direct use and conversion into enriched uranium, are between 60,000 and 90,000 tonnes. The proportion of these quantities which would have to be subjected to enrichment may be set between 40,000 tonnes and virtually the total amount comprised in the higher figure, i.e. 90,000 tonnes.

On the other hand, the annual quantities of uranium required in 1980 for supplying nuclear reactors in operation and for building up the inventories of new plants will be between 11,000 and 16,500 tonnes of natural uranium, of which 8,000-16,000 tonnes would be set aside for isotope separation purposes.

As regards the longer term, it must be stressed that fast breeders, on which, again according to the target programme, faith is being pinned for the period beyond 1980, will be quite a long time in coming to the fore, their rate of installation hinging on available plutonium supplies. Consequently, annual enriched uranium requirements will only reach their ceiling about 1995 and the pace of their subsequent regression will be fairly slow, the 1980 level not being regained until much later.

Western world's requirements

In 1965 the total installed electricity production capacity for all the *OECD* countries (Western Europe, North America, Japan) was of the order of 500,000 MWe; the corresponding figure in 1980, according to the "Energy Policy" report[2], will be 1,300,000 MWe. Thus power will have to be increased by 800,000 MWe, to which must be added replacements of about 100,000 MWe, which means planning for 900,000 MWe in new installations. If we adopt a cautious hypothesis for nuclear energy development (a third of the new installations instead of half as anticipated in the United States), we may conclude that between now and 1980 approximately 300,000 MWe of nuclear capacity will be taken into service in the *OECD* countries. This estimate can be cross-checked by totalling the following constituent estimates:

United States	150,000 MWe
Euratom	60,000 MWe
Other *OECD* countries in Europe	40,000 MWe
Canada and Japan	30,000 MWe
	280,000 MWe

If we add to either of the foregoing estimates the power plants expected to be set up in the non-Communist countries not affiliated to the *OECD*, we finally arrive at an installed nuclear capacity of at least 300,000 MWe in 1980.

It may be estimated that at least three-quarters of the reactors in question will use slightly enriched uranium.

Enriched uranium production capacities

The present capacity of the US isotope separation plants would suffice to cover the annual requirements of light-water reactors totalling approximately 200,000 MWe. However, if account is taken of inventories, that is to say, the quantities of uranium needed each year to set up new power plants, the *USAEC* facilities could only feed a lower capacity, the level of which is conditioned by the rate of growth of nuclear installations. Thus with a probable growth rate of about 15% by 1980, giving rise to considerable inventory requirements, the capacity which could be fed by the existing separation units is no more than 140,000 MWe.

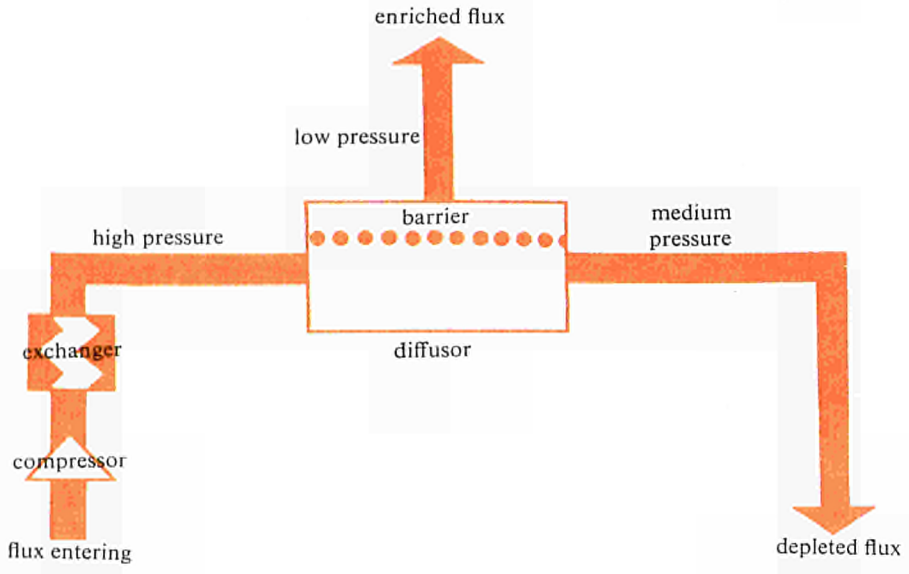
The present capacity of the three American facilities would therefore seem to be insufficient to cover the total needs of the non-Communist world in 1980; judging by appearances, it might even be hard put to it to satisfy between half and two-thirds.

What of the isotope separation capacities outside the United States, namely the European plants at Capenhurst and Pierrelatte? With their present volume, with that resulting from the planned expansion of the British plant, or even with that which might possibly be obtained by a comparable enlargement of the French facility—assuming all this capacity to be available for civilian users—they could not do very much towards solving the problems involved.

Imbalance in enriched uranium supply position

If no steps are taken to obviate such a situation, therefore, a serious imbalance will arise, even before 1980, between the enriched uranium needs of the non-Communist world and the capacities available for meeting them. This situation would be liable to have considerable repercussions

2. Energy Policy, Problems and Objectives, *OECD*, Paris, 1966.



Gaseous diffusion—plan of an enrichment stage.
In all large enrichment plants now in use gaseous uranium hexafluoride is diffused through porous walls, or barriers as they are called. The radius of the pores in these barriers is small compared with the average free path of the gas molecules, i.e. the average distance covered by a molecule before it collides with another in thermal agitation. If the pores were larger there would be more intermolecular collisions, all the heavy and light molecules would become mixed up and the concentration of the flux emitted would remain unchanged. However, since the pores are appreciably smaller than this distance, the number of collisions is limited and it is the collisions between the molecules and the walls which affect the velocity of the former. It is observed here that the velocity of U^{235}-base and U^{238}-base UF_6 is inversely proportional to the square root of their molecular weight. The gas diffused through the barriers thus receives a U^{235} enrichment of about $4\,^0/_{00}$.

or the Community, which would find itself dependent upon external capacities falling short of total requirements.
The remedy which leaps to the mind is to plan for additional enriching facilities. First of all, however, it must be ascertained whether no other solutions present themselves.

Possible solutions . . . use of natural uranium only . . .

The use of reactors fuelled on natural uranium would *ipso facto* eliminate the problems inherent in enriched uranium supply.
It must, however, be borne in mind that the United Kingdom is abandoning natural uranium graphite-gas (Magnox) reactors and that in the Community, with the possible exception of France, reactors of this type do not seem marked out for large-scale development. On the other hand, while natural-uranium heavy-water reactors may be envisaged as a possible solution in the future, the time required for bringing them to industrial maturity is such that they will only assume a place of any real significance in programmes towards the end of the next decade.
To conclude this section, it may be said that the hypothesis consisting in exclusive or at least major reliance on natural uranium must to all intents and purposes be rejected.

. . . recourse to plutonium or thorium. . .

The use of plutonium bred in the reactors operating between now and 1980, and more particularly its recycling in thermal reactors instead of uranium 235, is certainly an interesting proposition, and efforts must be made to develop the requisite technique. According to studies carried out for the purpose of drawing up the "First Target Programme", however, this would only go a short way towards solving the problem of enriched uranium supplies and would do very little to reduce requirements for the period up to 1980.
Certainly a more radical solution might be to optimise reactor operation for maximum plutonium production. This solution has the dual drawback of involving a disruption of, and of being distinctly more costly than, the current pattern of development.
Finally, there is the possibility of recourse to thorium, which, if theoretically attractive, can only be planned on a fairly small scale as far as the next ten years are concerned, since development of the relevant technology has to all intents and purposes not even got off the ground.

. . . improvement of purchasing conditions in the United States. . .

The present schedules of prices and enrichment services are applied to American and foreign users without any discrimination. Furthermore, as from 1 January 1969, when the system permitting toll enrichment comes into force, supply conditions for both US and foreign reactor operators will be exactly the same under this head. However, operators will be legitimately concerned about having sufficient enriched uranium to cover the requirements of their reactors during the latter's lifetime. The necessary safeguard may be obtained by entering into long-term supply contracts at the time when the decision to build is taken; as far as the Community is concerned, however, such contracts must be concluded under the US/Euratom Agreement for Cooperation and must not run beyond the date of expiry of this Agreement, i.e. 1995. The effect of this restriction will be felt increasingly as time goes on. On top of this, the contracts in question must also comply with the quantitative limits assigned to them by the United States Congress, under the head of total quantities set aside for civilian uses, by decision of the President. In this procedure, the possibility that an increase in the quantities concerned will

at a given moment be subordinated to political considerations cannot be ruled out.

. . . diversification of supply sources

The only alternative to obtaining supplies from the US would be to turn to the United Kingdom. It is known that the Capenhurst facilities, previously employed for military purposes, is to be turned over to the production of nuclear fuel. Capenhurst's supply potential is, however, very limited; it may be assumed that its annual capacity will in 1970-71 amount at the very most to 3 %, and by 1980 to about 15 %, of the total US capacity.
Consideration could, of course, be given, especially in view of the possibility of Britain's joining the Community, to a wider extension of the Capenhurst facilities than that currently planned, which is geared to the United Kingdom's own requirements. But then the authorities would be faced with the alternative of deciding either on a minor additional extension —which would be a very incomplete solution from the standpoint of the Community as a whole—or on vast-scale expansion, which would really amount to constructing a new plant, which would not necessarily be set up at Capenhurst.

The problem in a nutshell

At first sight, therefore, the existing US facilities plus the one planned for construction are, in practical terms, the sole source to which the Community can apply in order to cover its enriched uranium requirements. Whatever the commitments and guarantees surrounding supplies from the US, such exclusive dependence arouses, with good reason, considerable anxiety as to the future of nuclear activities in the Community. This anxiety is deepened by the fact that the American authorities plan to turn over at least part of the existing isotope separation capacity to the private sector, which will also be assigned the responsibility of building a new facility in due course. Over and above the risks inherent in the present situation, there is the danger that the interests of the owners, some of whom will no doubt be reactor constructors, will exercise an influence on enriched uranium sales policy.
For these reasons, and failing satisfactory alternative solutions, the possibility of building or adapting an isotope separation plant in the Community must be examined in its various aspects.

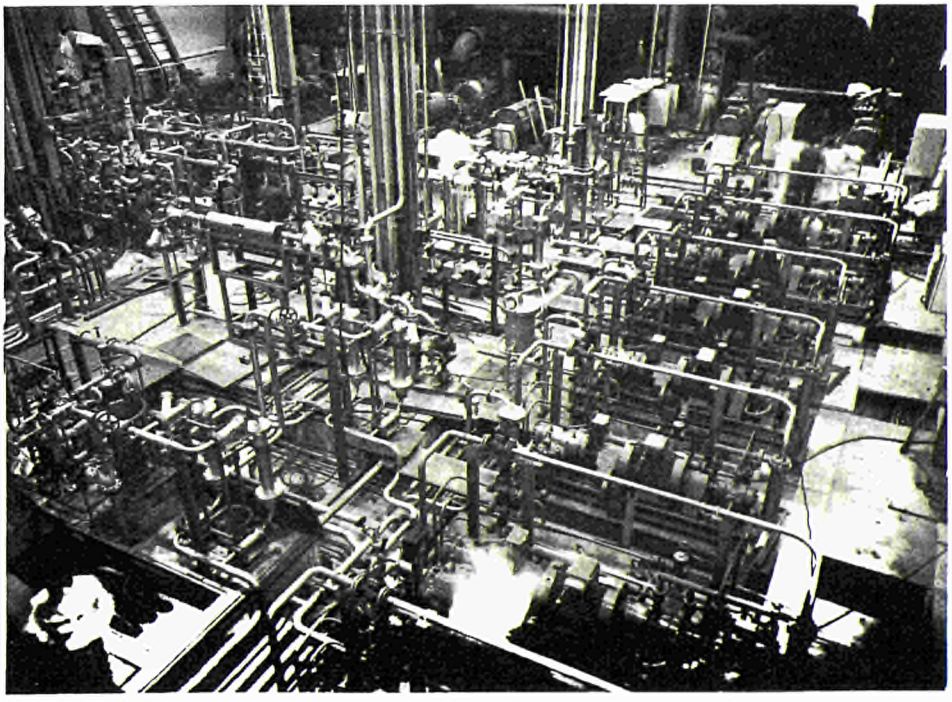
A Community plant – basic conditions

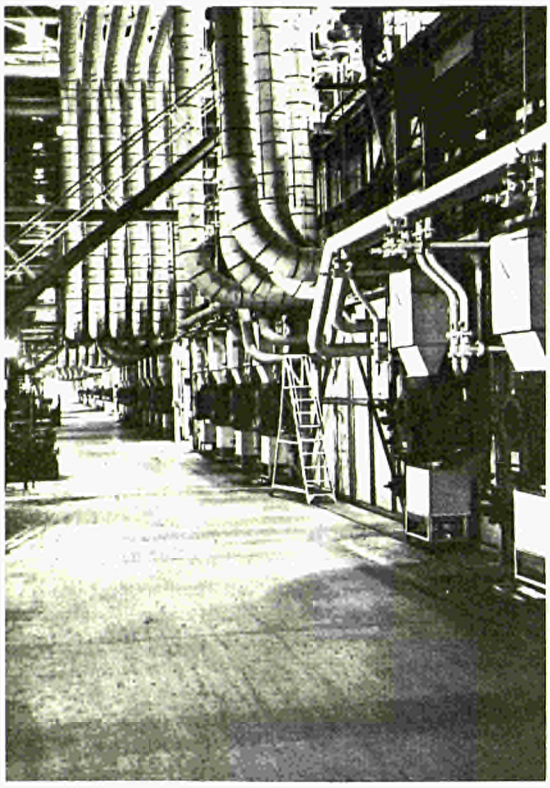
Any decision to construct an isotope separation plant in the Community will hinge on three classes of factors.
— *Political factors.* We shall confine ourselves to summarising these very succinctly by saying that the intention to build such a plant is inherent in the concept of a common energy policy.
— *Economic factors.* The plant represents an investment of a magnitude never reached in the Community. However, if this project has a reasonable chance of operating on an economic footing, it should not be impossible for the Community as a whole to muster the necessary capital, which, it should be noted in passing, is of an amount approximately equal to that which is already being invested by the six member countries each year in electricity generating stations of all types.
— *Technical factors.* A *sine qua non* for the construction of a competitive plant is access to the knowhow at present possessed only by the United States, the United Kingdom and France. None of these countries is prepared to permit wide-scale dissemination of this information. A conceivable way of overcoming this difficulty would be for a single enterprise, under the direct supervision of one of the aforementioned countries, to be assigned the dual role of architect-engineer and constructor of the classified parts of the isotope separation facility.
All these factors point to the conclusion that construction of a large-capacity isotope separation plant by a single member country is out of the question; quite to the contrary, co-operation between several countries is essential.

Extension of existing capacities or construction of a new plant?

Without attempting to examine the problem in detail here, we may make the

Pierrelatte – The vacuum plant





Part of the diffusion plant at Capenhurst near Chester.

following preliminary observations. A possible method might be to set up the facility on the Pierrelatte site by extending the existing installation. However, if this extension were to be carried out along the lines assumed for the first stage of the adaptation of Capenhurst, the resultant capacity would only cover a very small percentage of the Community's enriched uranium needs. An extension more on the scale of these needs would in effect be tantamount to the setting up of a new plant.

What process should be adopted?

The enrichment facilities in the United States, the USSR, the United Kingdom and France use the gaseous diffusion process. As things stand at present, and although other processes, such as centrifugation, offer worth-while possibilities for development, only gaseous diffusion can be considered as economic from an industrial standpoint. We shall therefore confine ourselves to this one process.

The facility envisaged would be a low-factor plant, with an enrichment factor limited to 3 or at the most 4%. The primary reason for this is that the greater part of the requirements for the reactors which it is planned to set up in the Community over the next 15 or 20 years will consist of slightly enriched uranium.

Fast reactors require for start-up purposes either plutonium or medium-enriched uranium (15-30%), although under the terms of the Target Programme preference should be accorded to the first of these types of fuel. Medium-enriched uranium, which would be needed to make good any plutonium deficiencies, should be obtained from enrichment facilities outside the Community. The same considerations apply to highly enriched uranium, which might be required by high-temperature reactors. Reliance on external sources for what would probably be fairly low quantities of medium-enriched or highly-enriched uranium would be a minor worry compared with the political problems and the additional expenditure involved in the construction of a high-enrichment isotope separation plant. Moreover, it is not impossible that Pierrelatte will one day be in a position to supply small quantities of these grades of enriched uranium for civilian purposes. Capenhurst, too, might fulfil this function, especially in the event of Britain's joining the Community.

Economic features of the European plant

According to estimates, the production cost per isotope separation unit of work is at present $ 30. The price fixed by the *USAEC* as from 1 January 1968 is $ 26, which will apply in particular to toll enrichment services.

If the isotope separation cost in Europe were in line with that in the United States, it would not be necessary to impose a customs duty for enriched uranium or to draw up a protectionist supply policy. This, however, assumes that the European facility would have a production capacity approximating to that of the US plants, for, provided access to the relevant knowhow were available, a European facility's production costs, based on the same design and the same parameters, should not differ appreciably from those obtaining in the United States.

European plant capacity and construction costs

Economic considerations dictate fixing the capacity of the European facility at about 7,000,000 separation units of work. This accords with the average capacity of the US facilities. Similarly, a working hypothesis in which the plant's rated output is set at 2,000 t/yr of 3%-enriched uranium gives, with a tails assay of about 0.25%, a capacity of 7,500,000 separation units of work per year.

On the basis of the US figures, it is possible to give a few pointers as regards the expenditure and investment required by an isotope separation facility with this capacity. The most significant figures are as follows:

Rated capacity	7,500,000 units
Highest degree of enrichment	3%
Rated 3% enriched uranium production	2,000 t/yr
Natural uranium feed	12,000 tonnes U contained per year
Absorbed electrical energy	19,000 million kWh/yr
Power requirement	2,200 MWe
Manpower required	2,500 to 3,000
Investment for facility proper	840 million u.a.[3]
Operating assets (materials immobilised and operating capital)	150-200 million u.a.
Annual fixed charges	138 million u.a.
Annual proportional charges	87 million u.a.
Investments for nuclear power plant to run facility	250 million u.a.

3. 1 u.a. (unit of account) = 1 US $

A 7.5 million capacity would therefore require considerable financial resources. If we add to the facility proper the ancillary installations, immobilisation of materials and the power plant, we arrive at a total investment of approximately 1,300 million u.a. (not including the operating capital). This figure is equal to a fifth of what is laid down in the Target Programme for the construction of nuclear power plants up to 1980. It does not, however, exceed the amount at present required each year for increasing the Community's overall electricity capacity or what in fifteen years' time will be the annual portion of investment for new nuclear power plants.

Legal problems

What legal problems would be involved in setting up a European isotope separation plant? First of all, it may be wondered whether such a facility—and, secondarily, the ancillary installations—should be owned by the European Community, probably not Euratom at that time but a single Community. It may certainly be hoped that the Community will gradually acquire more "teeth". Even so, it is doubtful whether it will carry early enough the political weight and have the legal powers and the resources essential for constructing and then operating as the sole owner an undertaking of such importance and such size. Another question which arises is whether such ownership accords with the task laid upon the Community, seeing that Euratom, for instance, is not empowered to build and manage power reactors.

Having regard to the foregoing, the legal form to be adopted would be rather that of a company incorporated by the Community member countries. Possibly the company could subsequently be enlarged through the admission of other states, but at all events non-member countries would not have to be blackballed from it.

As regards the legal status in the strict sense, that of a joint enterprise within the meaning of the Euratom Treaty would appear to be the obvious choice, as in that case the criterion of "fundamental importance to the development of the nuclear industry" would scarcely give rise to any controversy. Moreover, in introducing this concept, the authors of the Treaty had precisely such a project in mind. Again, this status would be particularly appropriate on account of the possibilities it would afford as regards tax relief and favourable financing terms.

In the United States, the authorities are known to be planning to assign part of the ownership of the existing facilities to the private sector. Analogously, an arrangement could be made in Europe by which those of the Member States which as a matter of principle do not allow the public authorities to carry on any direct economic activity could assign their shares to appropriate enterprises in the country concerned. By way of example, the facility could be owned jointly, irrespective of the position of the public authorities, by the utilities, the reactor constructors, the fuel element suppliers and the natural uranium producers. The Member States would thus be free to choose, within the limits of their participation, between a public and a private status for the enterprise. Assignment of shares should not, however, rule out the maintenance of supervision by the assigning state over the management of the facility.

Industrial and economic repercussions

The construction and operation of an isotope separation plant in the Community would have considerable repercussions in the industrial and economic fields. The vast scale of investment would be reflected in orders, the main beneficiaries of which would be the technologically advanced sectors (metallurgy, engineering, ceramics, chemistry, electricity, electronics).

Furthermore, it is obvious that a stable and independent nuclear reactor industry can only develop to the extent that users can be sure of obtaining from their power plant supplier, or from firms working with him, a full and guaranteed fuel-cycle service. This applies to the internal market but perhaps even more to the export sector, as the opportunity thus opened up for meeting external demand would be a major trump in the hand of European reactor constructors. In addition, the existence of an isotope separation facility in the Community would give far more elbow-room to the European reactor industry, which could develop the most promising types without being subjected as regards its enriched uranium supplies, and quite apart from the peaceful-use undertaking, to constraints which might constitute a well-nigh insuperable handicap in its attempts to obtain outlets on the world market.

At all events, a decision on the construction of an isotope separation plant is a very serious matter and of capital significance as regards the future of the European Community in the nuclear field. It must accordingly be based on as exhaustive as possible an analysis of the problems raised in this article. It is, indeed, by no means certain that such an analysis would bear out the options which seem to emerge from the present stage of the preliminary studies. The importance attaching to energy, however, warrants devotion of the necessary resources to a complete study and enlistment of the co-operation of those quarters whose assistance is indispensable. Failure to carry out in the months ahead the studies advocated for this purpose would be tantamount to denying the Community, without even assessing all the consequences, the opportunity sooner or later to avail itself of a key factor in the development of a nuclear industry—uranium isotope separation facilities.

(EUBU 7-4)

Towards a scientific and technical community

An important milestone towards wider co-operation in science and technology was passed by the Community countries on 31 October 1967, thanks to a series of major decisions taken by the Council of Ministers, who had met at Luxembourg.

At this meeting the Council and the Commission of the European Communities expressed their intention to launch, in connection with the Community's medium-term economic development programme, an energetic campaign to revamp and stimulate scientific and technical research and industrial modernisation. This attitude is inspired by the realisation that scientific and technical progress is a basic factor in economic growth, but especially that Europe is dangerously behind in this field, particularly in comparison with the United States.

One of the decisions taken by the Council on 31 October aims at the active continuation of work, already started in the context of economic unification, on Community company status, Community patents, etc., in short, on all legal and fiscal measures that tend to help the creation of more efficient commercial structures.

The Council also entrusted the "Scientific and Technical Research Policy Working Group" of the Medium-Term Economic Policy Committee with a number of tasks, with which the Commission will be associated: comparison of the national methods and general plans, programmes and budgets relating to research; examination of ways of setting up a Community technical data processing and dissemination system (incidentally, as regards nuclear matters, the Euratom automatic documentation system, which is one of the most advanced in the world, is already in operation); examination of ways of providing co-ordinated training and more intensive exchanges of scientists.

The working group must also—and here perhaps is its most important duty—investigate the scope for co-operation in six particular fields—data processing and telecommunications, development of new methods of transport, oceanography, metallurgy, nuisances and meteorology. Moreover, this list is not exhaustive, and other sectors can be considered for inclusion. The Council asked the working group to report back on such possibilities of co-operation by 1 March 1968, a deadline which it may prove difficult to meet as the political attitudes of the Member States must in some respects still be made to tally.

It should be observed in this connection that the Council asked the working group to search for ways by which other European States could participate in this co-operative effort.

Second symposium on nuclear power plant components

Fifty-one representatives of practically all present and future operators of power reactors in the European Community met in Paris on 30 November 1967 to attend a symposium held by Euratom on the theme: "exchange of constructional and operational experience with nuclear power plant components". This meeting was a follow-up of a first symposium on the same subject, held in Amsterdam in November 1966.

Joint reports were presented at the symposium on the results of the work of seven teams, consisting of representatives of the operators and of Euratom, which had covered the following topics this year:

- reactor pressure vessels and internals;
- fuel element handling, storage and transport;
- instrumentation and regulation;
- fuel element damage detection systems;
- control and safety rods together with their drive systems;
- water clean-up;
- corrosion and mass transport by working media and coolants.

In the course of the discussion the participants stressed once more that it was not the components of the nuclear part of the plant that gave most trouble but those of the conventional part.

The discussion covered not only questions of technical detail, but also general questions concerning the construction of nuclear power plants, such as the choice between the ordering of turnkey plants and construction on the operator's own responsibility, the influence which the operator should have on the design and the quality control of components, the activities which should receive public support, etc.

In addition, it was suggested that action should be taken jointly with a view to eliminating the obstacles still in the way of a common market. Priority tasks in this connection would be the working out of a standard form for plant component specifications, together with the establishment of guide lines for the activities to be promoted by the European Communities in the fields of quality control and standardisation of reactor components.

Interim programme of European Atomic Energy Community for 1968

In its meeting held on 8 December 1967 the Council of Ministers, on the basis of proposals put forward by the Commission of the European Communities, drew up an interim programme and a draft research budget for the European Atomic Energy Community for the financial year 1968.

Under the programme the work now under way in the establishments of the Joint Research Centre and at the Commission's Headquarters is to be continued, no new projects being undertaken. It also provides for the further employment of Euratom personnel hitherto working outside the Joint Research Centre establishments so that, in particular, the work of the Communities' contractual partners is continued, in cases where this is feasible and desirable.

The maximum sum earmarked for the completion of this programme is set at 41 million u.a., no account being taken of the funds carried over on 1 January 1968.

At the same time authorisation was given for a maximum total payroll of 2,750 on 31 December 1968.

With regard to the future activities of Euratom, the Council of Ministers adopted the following programmes on 8 December:

A joint programme which must be as wide as possible and which will be financed by all the Member States by means of a budget drawn up according to a fixed scale of contributions. The activities included in the joint programme will be incorporated in one or more programmes extending over several years, pursuant to Article 7 of the Euratom Treaty.

Supplementary programmes in which, for cases where it is impossible to achieve unanimity, only interested Member States will participate, after reaching a special agreement.

The *joint programme* will cover:

– *the activities of the Joint Nuclear Research Centre*. The programmes now being carried out at the Joint Nuclear Research Centre must be reviewed.

In this connection, the optimum use must be made of the existing equipment and the staff now employed. As far as the legal position allows, research may also be extended to non-nuclear activities, in particular in the fields mentioned in the Council's decision of 31 October 1967. The administrative structure and the management of the Joint Nuclear Research Centre must be improved in order to make its work more efficient.

– *certain activities which are at present, or which could be the subject of Association Agreements*, in as far as they are in the Community's interest or can be made the subject of scientific co-operation between the various national programmes, and do not lead to useless duplication of effort. In this connection the Council instructs the Permanent Representatives' Committee to examine, on the basis of a report from the Working Party on Atomic Questions, the activities which could be made the subject of these agreements, paying particular attention to fundamental research and to basic programmes concerning reactor development. As regards the existing associations, the Permanent Representatives Committee will report to the Council before 1 March 1968.

The training of research workers, and documentation.

Co-operation in the framework of the *supplementary programmes* may take various forms, which must be examined in order to determine the method of financing, the part to be played by the Commission, and the provision of information to non-participating States. Joint Undertakings in the sense of Chapter V of the Euratom Treaty seem to be particularly appropriate for this purpose.

When future programmes are being drawn up, it will be necessary to distinguish between activities in respect of which industrial problems do not arise, and those whose continuation or starting up presupposes a common industrial policy.

When research programmes are being worked out, the possibility of inviting European countries outside the Community to participate, within the framework of the Euratom Treaty, should be examined in certain cases.

A special study group of the Consultative Committee on Nuclear Research will have to examine questions relating to long-term supplies of enriched uranium. The Commission will submit appropriate proposals to the Council on the basis of this study.

The questions of the organisation and administration of Euratom must also be examined with regard to future programmes and structures. On the occasion of this examination it will be advisable to introduce into the staff regulations, including those for the Joint Nuclear Research Centre, much better opportunities for the recruitment of temporary staff on contracts for a period of several years, in order to attain the flexibility which is necessary in the field of research as regards the management of staff in the scientific, technical and administrative services.

In co-operation with the Commission, the Council will continue to endeavour to define the activities which will fall under either the joint programme or the supplementary programmes. As soon as a decision has been taken on the choices to be made, the Consultative Committee on Nuclear Research will be asked to work out the technical details so the Council can take a final decision, on a proposal from the Commission, in full knowledge of the facts.

The Council will do its utmost to reach its decision before 30 June 1968.

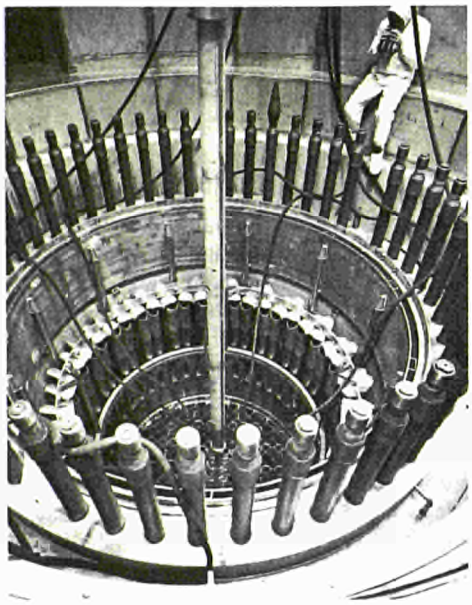
EURATOM NEWS

Five European concerns join forces to promote high-temperature reactors

On 18 January 1968 five European concerns signed an agreement to co-operate in the field of high-temperature gas reactor technology. They are: *The Nuclear Power Group Ltd.*, *BelgoNucléaire S.A.*, *Gutehoffnungshütte*, *Snam Progetti S.p.A.* and the *Groupement Atomique Alsacienne Atlantique S.A.*

The chief object of the agreement is to consider the forming of a European industrial association to promote sales of nuclear reactors using high-temperature gas reactor technology, with special reference to *Dragon* reactors.

The high-temperature gas reactor is the logical outcome of the gas/graphite reactor line. The innovations introduced by the "high temperature" type are mainly connected with the use of helium as coolant and the adoption of a coated-particle ceramic fuel.



First chain reaction at Lingen

On 31 January 1968 the second large German nuclear power plant, under construction at Lingen on the Ems, went critical. The Lingen power plant, capacity 240,000 kW net, will supply the first power to the grid in the late summer of this year.

The *Kernkraftwerk Lingen GmbH*, in which the firms *AEG-Telefunken* and *VEW* are associate partners, was granted the status of a joint enterprise within the meaning of the Euratom Treaty by the Council of Ministers on 12 December 1964.

Lingen nuclear power plant. A reactor engineer sees that the fuel elements are inserted correctly.

AVR-reactor on power

On 18 December 1967, the world's first reactor to use spherical fuel elements, the 15 MWe *AVR* power reactor at Jülich (Federal Republic of Germany), started delivering electricity into the grid of *Rheinisch-Westfälisches Elektrizitätswerk* AG.

The reactor was developed and constructed by *Brown-Boveri/Krupp Reaktorbau GmbH* for *AVR*, a group of German utilities. It is operated by *AVR* with financial support from the *THTR-Association* between Euratom, the *Jülich Nuclear Research Establishment* and *Brown-Boveri/Krupp*.

The reactor is of the high-temperature helium-cooled type and the fuel consists of graphite spheres of the size of a tennis ball containing 1 g of fissionable material, in the form of uranium 235, and 5 g of fertile material in the form of thorium. (For more details see *Euratom Bulletin* 1965, Vol. IV nº 2, pp. 44-48).

1000 automatic literature searches by CID in 1967

During 1967 the automatic documentation system developed by Euratom's CID (Centre for Information and Documentation) finally emerged from the experimental stage. About a thousand literature searches were made with the help of the CID's computer, 40% of them at the request of Euratom's own scientific departments, 48% at the request of enterprises or bodies in the Community countries, and 12% in response to requests from other countries.

Thanks to the criticisms asked for and obtained from users, it was possible to improve the system and in particular to enhance the relevance and completeness of the bibliographical references supplied in reply to enquiries. The "customers" comments show that on average 70% of the data were regarded as relevant and 73% of them related to documents hitherto unknown to the customers.

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Recent publications by the Euratom Information and Documentation Centre

– Irradiation Experiments in the HFR and the Laboratories of the Petten Establishment of Euratom's Joint Research Centre and the Reactor Centrum Nederland—A Survey of Possibilities
by J. BUGL and H. RÖTTGER.
EUR 3650. e
Price: B. Fr. 200, $ 4
In this abundantly illustrated brochure, the ways in which the Petten Establishment can help to solve materials problems are described in detail.

– Accelerator Targets Designed for the Production of Neutrons
Proceedings of the 3rd Conference held at Liège (Belgium) September 18-19, 1967
Edited by H. G. EBERT
EUR 3895 d, f, e
Price: B. Fr. 500, $ 10

– Practical Aspects of Activation Analysis with Charged Particles
Proceedings of the 2nd Conference held at Liège (Belgium) September 21-22, 1967
Edited by H. G. EBERT
EUR 3896 d, f, e
Price: B. Fr. 500, $ 10

– Microdosimetry
Proceedings of the Symposium on Microdosimetry held at Ispra (Italy)
November 13-15, 1967
Edited by H. G. EBERT. EUR 3747 d, f, e
Price: B. Fr. 750, $ 15

– Investigation on Nuclear Core Journals
by E. BÖHM. EUR 3887. e
Price: B. Fr. 40, $ 0.80
This report displays the results of a survey carried out on the scientific and technical journals used for the input of Euratom's automatic documentation system. One of the most striking results is that 75% of the journal literature input is effected by scanning only 272 "core journals". This report should prove useful to librarians wanting to draw a maximum coverage from their journal acquisitions.

– Accidental Irradiation at Place of Work
Proceedings of the International Symposium held in Nice (France) 26-29 April, 1966
EUR 3666 d, f, i, n, e
Price: B. Fr. 1,000, $ 20

– The Development of Light Water Reactors in the Community
Proceedings of the meeting organised by Euratom in Brussels 23-24 November, 1966
EUR 3561 d, f, e
Price: B. Fr. 600, $ 12
The central subject of this meeting was the development of European variants of the boiling water and pressurised water reactors.

– Répertoire des installations nucléaires de la Communauté européenne de l'énergie atomique—4ème édition
EUR 3568 f
Price: B. Fr. 165

– Euratom Thesaurus Second Edition, Part II
Terminology Charts used in Euratom's Nuclear Documentation System
EUR 500 e
Price: B. Fr. 300, $ 6

Fuel for the AVR *nuclear power reactor. Each sphere contains 1 g of uranium 235 and 5 g of thorium.*

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