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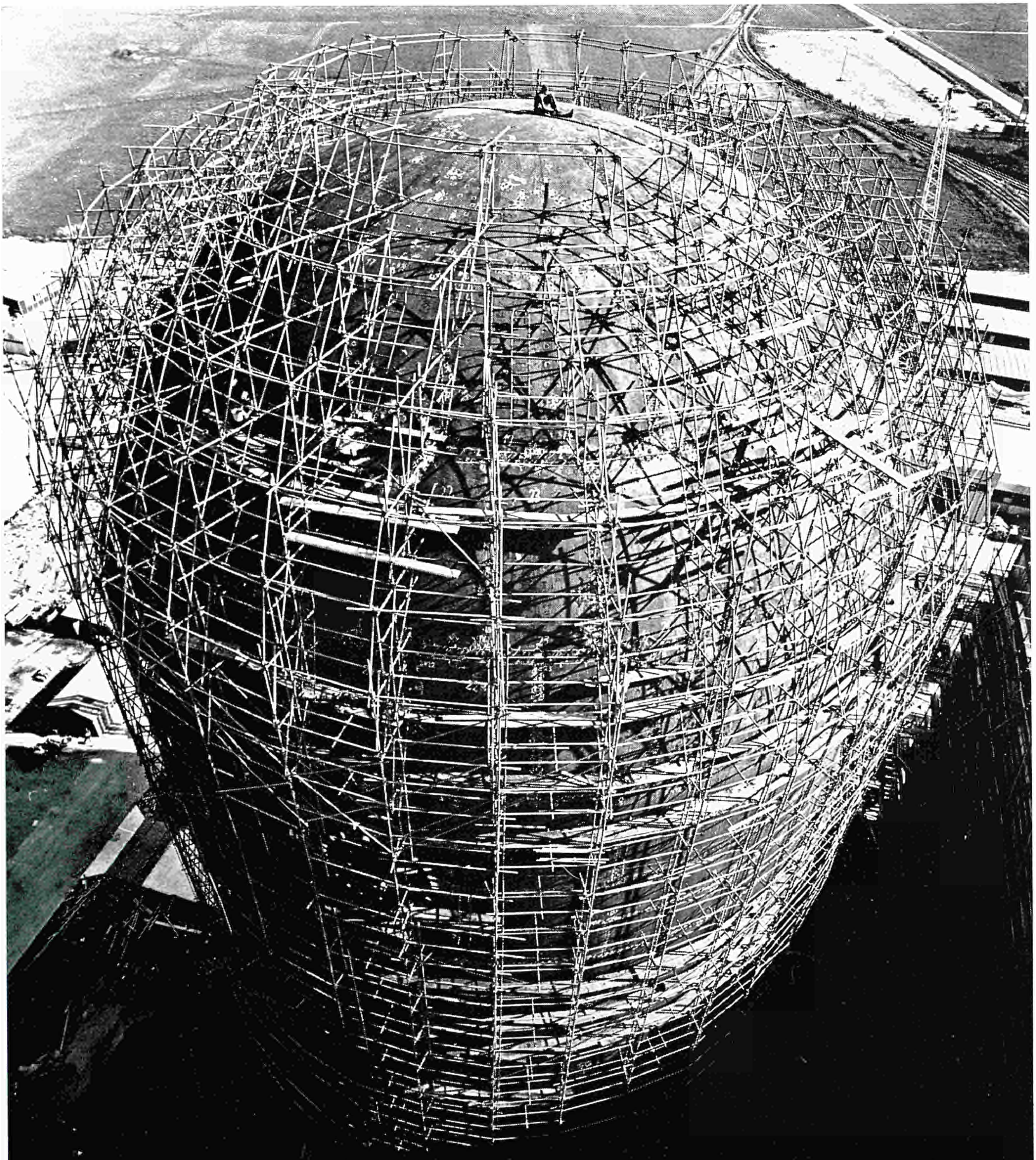

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Quarterly Information Bulletin of the European Atomic Energy Community (Euratom)

1965-2

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1965-2

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

"Proven" reactors, "advanced" reactors, "fast" reactors . . . these terms, which are creeping into nuclear parlance, do not, strictly speaking, denote a hierarchical order, since all the reactors thus described are kings in their own way; the proven reactors are the kings of today, the advanced converters—three types of which are discussed in this issue—will reign tomorrow, and the fast reactors are destined to become the monarchs of a more distant future. According to this scheme, therefore, what we really have is a dynasty.

Undoubtedly, it is on the fast reactors that the ultimate hopes are pinned, because owing to their amazing prospects as "breeders" of fissile material they show promise of solving the energy-supply problem for a long time to come.

Viewed in this perspective, the advanced thermal reactors would indeed seem ordained to replace proven-type thermal reactors, but they will merely constitute a temporary expedient to span the period of development of fast reactors, which looks like being fairly protracted. In short, their destiny is to fill the interregnum.

This outline already leaves room for numerous minor variations, not to mention those that the future will no doubt bring. For example, certain advanced thermal reactors themselves aspire to a breeding function and would thus appear to rival their legitimate successors. Actually, however, the notion of rivalry is misplaced, since the former call rather for a thorium/uranium-233 fuel cycle and the latter for a uranium-238/plutonium cycle. Consequently, it is more a case of co-existence.

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Nuclear Energy in Germany

DR. WOLFGANG FINKE, *Federal Ministry for Scientific Research, Bad Godesberg*

The development of nuclear research and technology in West Germany has hitherto proceeded under conditions which in major respects differ from those obtaining in the United States, Britain and France. There are two reasons for this which should be particularly stressed: firstly, Germany made a late start and took a fairly long time to get its programme off the ground and secondly, all activities were restricted to the atoms-for-peace field.

However, with the quickening pace of development of nuclear research and technology and the increasing importance of the economic aspects of nuclear energy these two factors will gradually decrease in significance in comparison with the other conditions affecting the overall context of scientific and industrial activities. But for the time being they should not be ignored in any attempt to assess the situation in Germany.

Late start

It was not until 1955 that systematic work in the field of nuclear research and development could be commenced in Germany, with one or two isolated exceptions. At that time, however, only a few branches of research were blessed with a sound basis for rapid advance, and in the majority of fields of science and in almost the entire field of industrial development this basis had first to be created, with regard to both staff and material facilities.

This resulted in a fairly long warming-up period, during which it was often impossible to assess the full extent of the efforts required, so that at the outset estimates frequently fell short of reality. It was thus only in the course of time that a realistic picture of the situation in Germany emerged which could be taken as the foundation for a long-term development programme. Since then much has been done to carry out this programme, which was at first merely mapped out in broad outline, the details not being filled in until later. The gap in nuclear research and technological development between Germany and the leading countries in this field has so far, of course, with a few exceptions, not been closed completely, but it has narrowed considerably, particularly in the last few years. The work since started on the new research centres and the growing industrial potential should help to speed up this development, so that the consequences of Germany's late start are likely to be overcome in the foreseeable future, the German Federal Republic attaining a level of scientific and technical achievement in the nuclear field within the next decade which will correspond to the position occupied by it in other fields on the international plane.

Peaceful uses only

Under the terms of the Paris treaties of 1954 the German Federal Republic expressly waived any right to develop or manufacture nuclear weapons. Since then, this pledge has been reiterated by the German Government on numerous occasions. Right from their commencement in 1955 all Germany's activities connected with research into and the development and utilisation of nuclear energy have been devoted solely to its peaceful applications. This self-imposed curb has had an enormous effect on the pace and intensity of Germany's efforts to get in the swim of world nuclear development.

The effect was two-fold: firstly, these activities were right from the outset not marked by that sense of urgency which military matters are—rightly or wrongly—accorded throughout the world and which usually calls for relatively generous appropriations from public funds; secondly, the restriction of German development work to the pursuance of peaceful aims resulted in a liberty of choice with regard to the lines to be followed which the country's limited resources would otherwise certainly not have permitted.

Furthermore, as a result of these circumstances, no need was felt, in West Germany, in the national organisation set up to promote nuclear research and technology, to follow the example of the major nuclear powers. Only a small ministry of conventional type was established, largely supported in its work by outside scientific, technical and economic advisers. Research and development work in institutes and industrial enterprises is to a large extent free of government interference.

General conditions

By virtue of the conditions governing nuclear research and technology in West Germany, the special position which these fields have come to occupy in German science and industry is much less prominent than in many other countries. Conversely, therefore, the general factors conditioning the development of nuclear energy are of greater importance in Germany than elsewhere. These factors may be broken down into the following particular headings:

— the organisation and efficiency of the scientific effort in general,[1]

— the structure, the degree of industrialisation and the potential of the national economy[2],

— the pertinent legal provisions,[3]
— the effectiveness of international co-operation and competition in the scientific, technical and economic fields,[4]
— the situation on the energy market with regard to supply and demand, in particular prices and the availability of rival primary energy sources.[5]

The last few years have shown that nuclear development work in Germany has been much more influenced by these general factors than the special programmes and projects in the various fields would at first indicate. There is a great deal of evidence to suggest that, as nuclear energy gets closer and closer to normalcy, the influence of these factors will be enhanced, especially since it is Germany's avowed policy gradually to put nuclear research and technology, which at present occupy a special position in science and industry respectively, on the same footing as conventional research and technology, and to restrict state interference in these fields over the long term as much as is compatible with the particular character of nuclear energy.

The German nuclear programme

The first concrete plans for a German nuclear programme date back to the years 1955-56. They came about as a result of, primarily, the first Geneva Conference and, secondly, recognition of the fact that Germany was lagging way behind the world's leading countries in the field of nuclear power. The essential feature of these plans was the realisation that training facilities and general research centres would first have to be set up, special large-scale research projects and major technical development programmes being left till later. The consensus was also that the modernisation and expansion of existing institutes would have to have priority, special new research centres being set up in only exceptional cases. Opinions were divided, however, as to whether it would be better to buy research reactors abroad or to develop them in Germany. Ultimately both paths were chosen.

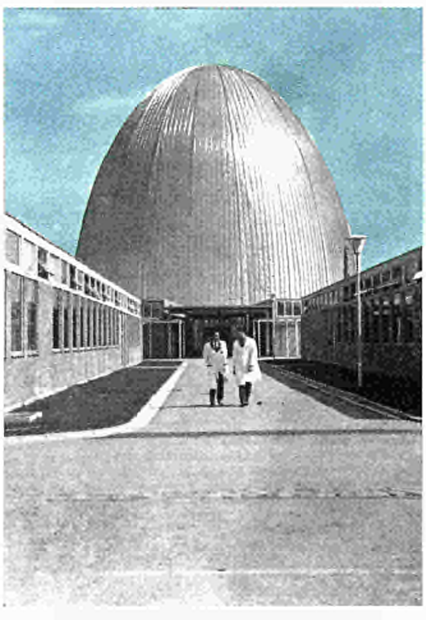
The DESY *electron-synchrotron in Hamburg*

However, no official all-embracing, hard-and-fast German nuclear programme was drawn up at the time, developments being limited to the loose co-ordination of the various projects under a general framework. It is only in the case of power reactor development that these considerations since the beginning of 1957 have crystallised into a kind of overall plan, the so-called *Eltville programme*, which in 1959 formed the basis of a recommendation put forward by the German Atomic Commission. The *AVR* gas-cooled high-temperature reactor in Jülich, the multi-purpose research reactor at Karlsruhe and the Lingen nuclear power plant are largely based on the general plans drawn up at that time, even though the detailed execution of the projects entailed major departures from the previous proposals.

A comprehensive German atomic programme was not drawn up until 1962-63, its implementation being recommended to the Federal Government by the German Atomic Commission in May 1963. However, this programme, too, which covers the period 1963-67, "does not embody any commitments in the form of a rigid plan", but as the Federal Minister for Scientific Research states in the preface, "constitutes a guideline for the most appropriate promotion and planning work in the public and private sectors, which can be constantly improved and adapted in line with the increasing pace of development".

The aims of this government sponsoring are, according to the programme, firstly to open up all fields of nuclear energy, ranging from basic research to reactor construction, to set up and extend the appropriate supply industries and to provide for raw material procurement; secondly, to utilise nuclear radiation and radioisotopes in other scientific and technical fields and, thirdly, to generate applications in other fields.

These recommendations refer in particular to the sponsoring of co-ordinated basic



Reactor building of the research centre at Garching, Munich.

1. 2. 3. 4. 5. See page 43

research in institutes of technology and universities, *Max-Planck* Institutes, research centres, federal and Land research establishments, industrial institutes and especially, in the sphere of international co-operation, Euratom's joint research centre. Moreover, applied research would be supported, particularly in the research centres and industry, great stress being laid on the need for careful co-ordination. With regard to technological development work, the programme favours the more intensive support of promising projects undertaken by firms and is thus opposed to technical development projects being carried out solely by research centres. Finally, the need is emphasised for public backing for the construction of prototype plants.

The main feature of the German nuclear programme for the period 1963-67 is the emphasis laid on research, in particular basic research. In the case of technological development work, with special reference to reactors, it confines itself to supplying the criteria by which the various projects are to be assessed. Only with regard to the long-term view is the programme decidedly in favour of the development of fast and thermal breeder reactors. Under the reactor construction programme, the recommendations cover the building by 1967 of a total of three large power plants equipped with proven-type reactors, three experimental reactors and a marine reactor. The construction of a small-scale plant for spent fuel reprocessing is also advocated.

It will cost about 2,500-2,700 million DM to carry out this programme. This calculation is based on the assumption that the annual expenditure will increase by almost 20%, being thus roughly double by the end of the programme. The total expenditure for the programme breaks down into 1,100-1,200 million DM for basic research, about 400 million for applied research and 1,000-1,100 million for technological development and the construction of nuclear installations, not including the large power plants. The data on the funds required have deliberately been kept somewhat imprecise in the programme, so that no accurate comparison with actual expenditure can be drawn. Nonetheless it may be noted that during the first two years the programme targets were largely attained and no major cut-backs are expected in the third year either.

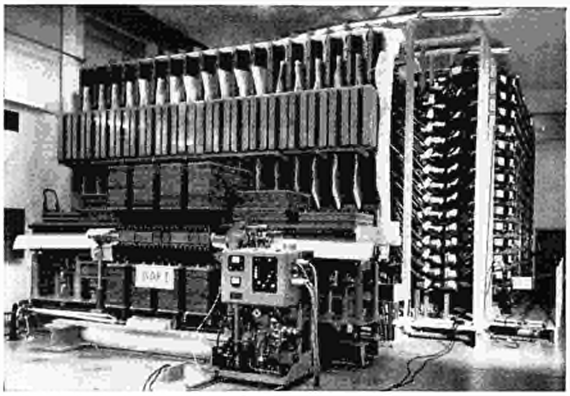
Germany's nuclear research centres

As was mentioned above, nuclear research in Germany is largely carried out in the traditional research centres, particularly the technical colleges and universities and in the *Max-Planck* Institutes, but also to an ever increasing degree in industrial research establishments. The general facilities available in these institutes and laboratories have been adapted to modern requirements during the last few years at a relatively high cost. Many of them have small or medium-sized particle accelerators and several possess training, testing or research reactors of their own. Some establishments, e.g. the *Hahn-Meitner Institute for Nuclear Research* in Berlin, occupy a rather special position by virtue of their organisational setup, but these, too, are fundamentally also institutes of technology.

Specifically nuclear research centres are thus an exception in Germany. Those that do exist were usually set up because the equipment required or task imposed was beyond the reach of traditional facilities. In two cases they were formed in order to put into the same harness a number of different disciplines and by so doing to

I. TRAINING REACTORS

Name	Location	Operator	Manufacturer	Reactor Type	Thermal Power in kW	Max. thermal neutron flux n/cm² sec	Coolant	Moderator	Fuel	State of construction or commissioning date
1. SUR 100	Garching	Siemens-Schuckert-werke AG	Siemens-Schuckert-werke AG	solid homogeneous reactor	0.0001 (for limited periods 0.001)	approx. $5 \cdot 10^6$ ($5 \cdot 10^7$ at 0.001 kW)	none	polyethylene	enriched uranium (20%) 0.7 kg	28.2.1962
2. SUR 100	Berlin	Institut f. Allgemeine u. Kernverfahrenstechnik, Inst. of Technology Berlin	"	"	"	"	"	"	"	17.7.1963
3. SUR 100	Darmstadt	Institut für Reaktortechnik, Darmstadt Inst. of Technology	"	"	"	"	"	"	"	23.9.1963
4. SUR 100	Hamburg	Staatliche Ingenieurschule, Hamburg	"	"	"	"	"	"	"	15.1.1965
5. SUR 100	Stuttgart	Institut für Hochtemperaturforschung, Stuttgart Inst. of Technology	"	"	"	"	"	"	"	under construction
6. SUR 100	Aachen	Institut für elektr. Anlagen u. Energiewirtschaft, Aachen Inst. of Technology	"	"	"	"	"	"	"	under construction



1.5/2.6 megajoule capacitor bank at the Max Planck Institute for Plasma Physics *at Garching, Munich.*

make for a particularly intensive cross-fertilisation of ideas in the study of a wide range of scientific and technical problems. The first category includes in particular the *German electron-synchrotron* in Hamburg, the *Institut für Plasmaphysik* at Garching, near Munich, and the special nuclear research centres at Neuherberg and Geesthacht. The second group is made up of the multi-purpose nuclear research centres at Karlsruhe and Jülich.

The *German electron-synchrotron (DESY)*, at present one of the largest electron accelerators in the world, went into service early in 1964 after four years' construction work. It reached a particle energy of 6 GeV without any major difficulty, and its peak should be around 7 GeV. The path diameter of the accelerator is 101 m, the maximum particle beam totalling 5.10^{12} electrons/sec. A hydrogen bubble chamber of 80 cm diameter was built for the tests in collaboration with the French nuclear research centre at Saclay. The total cost of the project was 110 million DM, 83 million of which are provided by the Federal Government, 17 million by Hamburg and 10 million by the Volkswagen Foundation. The operating costs, which are expected to rise to 40 million DM in the coming year, are borne equally by the Federal Government and the Länder. The staff now totals over 600.

Equally impressive funds and staff are involved in the first expansion stage of the fusion research centre of the *Institut für Plasmaphysik*, at Garching near Munich, with an estimated investment bill of 96,500 million DM, 90% of which is borne by the Federal Government and 10% by Euratom. The operating costs, which are estimated at about 15 million DM for 1965, are to be divided up equally between Euratom, the Federal Government and the Länder. The staff has in the meantime passed the 700 mark. The research installation is used for studies on controlled thermonuclear reactions. The Institute's activities are at present focussed on research on stationary plasmas, fast theta-pinch, linear-Z-pinch and tubular-pinch discharges, plasma diagnostics and plasma theory. Other research on plasma physics is being carried out in the *Jülich Nuclear Research Centre*, being mainly aimed at studies on fast magnetic compression and the development of a plasma accelerator. Euratom is also participating in these activities under a contract of association.

The *German electron-synchrotron* in Hamburg-Bahrenfeld and the *Institut für Plasmaphysik* in Garching near Munich are special centres for basic physics research. The *Gesellschaft für Strahlenforschung*, on the other hand, with its institutes for medical biology, radiation protection and haematology (with research teams in Munich and Freiburg) and its physico-technical department, is engaged on basic studies in the field of biology and medicine and on training in radiation protection techniques. It also has research teams for radiation measuring methods and the deep storage of radioactive effluents. The Gesellschaft, which is still in the process of formation, employs a staff of about 300 at the moment. By 1967-68 the investments spent on its establishments of Neuherberg, near Munich, Munich, Freiburg and the former salt mine at Asse will run to about 43 million DM. Euratom is to participate in the work of some of them and is to make a contribution to the appropriate running costs. The remainder of the expenses are borne by the Federal Government.

The facilities of another special nuclear research centre, in Geesthacht-Tesperhude near Hamburg, are directed less at basic than at applied research and technical development in the marine reactor field. It is equipped with a swimming-pool reactor with two reactor cores, each of 5 MW thermal power, a rolling test-rig, a critical experiment installation and a radioactive waste storage facility. The owner and operator of this establishment is the *Gesellschaft für Kernenergieverwertung in Schiffbau und Schiffahrt*, Hamburg, whose capital is mainly derived from the Federal Government and the Länder of Bremen, Hamburg, Lower Saxony and Schleswig-Holstein. These Länder and the Federal Government also bear jointly (40% and 60% respectively) the Gesellschaft's investment and operating expenses which are not covered by revenue. More than 350 persons are at present employed in Hamburg and Geesthacht. The most important project now being carried out is the construction of the nuclear-powered ship *Otto Hahn*, fitted with a pressurised-water reactor, which is to be completed by 1967. Euratom also is to participate in the cost of the reactor.

The research and development establishments described hitherto are of a specific nature, whereas the centres in *Karlsruhe* and *Jülich* are multi-purpose nuclear research institutes. They were originally planned with different aims in view—Karlsruhe was at first mainly engaged on development work, in particular the construction of a 12 MW research reactor, coupled with a strong emphasis on the industrial side, while Jülich was primarily intended as a joint research establishment for the universities and technical colleges of North Rhine/Westphalia, industrial matters being left pretty much in the background to start with. In the meantime however both centres have acquired more or less the same all-round character. So that in addition to the ties with industry, Karlsruhe also maintains close relations with the universities and technical colleges of Baden-Württemberg, while Jülich's links with the universities and technical colleges of North Rhine/Westphalia have been broadened by collaboration with industry. Both Karlsruhe and Jülich have for some time now been financed solely from public funds, the breakdown being 75% from the Federal Government and 25% from the Land of Baden-Württemberg in Karlsruhe and roughly 90% from the Land of North Rhine/Westphalia and just under 10% from the Federal Government in the case of Jülich. In accordance with present plans, investments will total about 650-700 million DM in each of

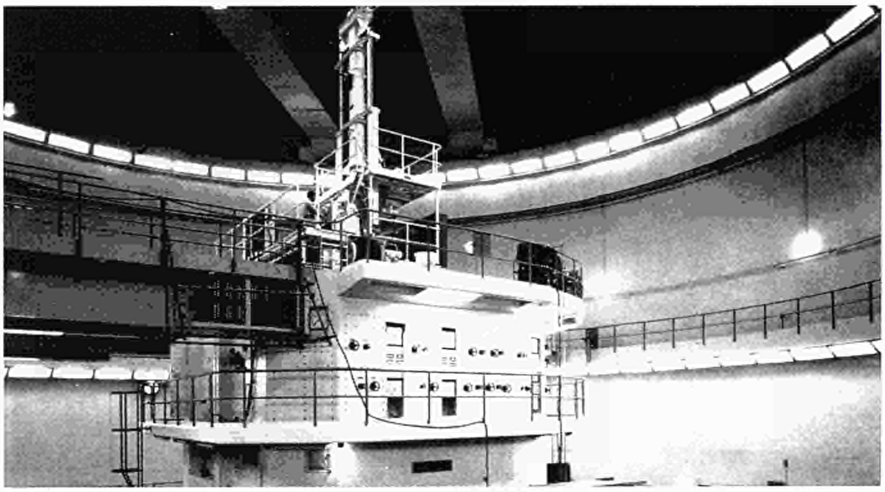
II. RESEARCH REACTORS

Name	Location	Operator	Manufacturer	Reactor Type	Thermal Power in kW	Max. thermal neutron flux n/cm² sec	Coolant	Moderator	Fuel	State of construction or commissioning date
1. FRM	**Garching**	Laboratorium f. Techn. Physik, Munich Inst. of Technology	AMF Atomics, Div. of American Machine & Foundry Co., USA, and MAN	swimming pool	1000	$2.5 \cdot 10^{13}$	light water	light water	enriched uranium (90%) 4 kg	31.10.1957
2. FRF	**Frankfurt/ Main**	Institut für Kernphysik Univ. of Frankfurt	Atomics International, Div. of N. American Aviation Inc., USA, and AEG, BBC, Mannesmann, SSW	homogeneous solution reactor	50	approx. 10^{12}	light water	light water	enriched uranium (19.7%) 1.4 kg as uranyl sulphate in aqueous solution	10.1.1958
3. BER	**Berlin**	Hahn-Meitner-Institut für Kernforschung Berlin	Atomics International and Arbeitsgemeinschaft AEG, Borsig, Pintsch-Bamag, SSW	homogeneous solution reactor	50	approx. 10^{12}	light water	light water	enriched uranium (19.7%) 1.3 kg as uranium sulphate in aqueous solution	24.7.1958
4. FRG	**Geesthacht**	Ges. f. Kernenergieverwertung in Schiffbau u. Schiffahrt mbH	Babcock & Wilcox Co. (USA) and Dt. Babcock & Wilcox	swimming pool	1. reactor 5000 2. reactor 5000	$3 \cdot 10^{13}$	light water	light water	1. Reactor: enr. uranium (20%) 5.4 kg 2. Reactor: enr. uranium (90%) 3 kg	1. Reactor: 23.10.1958 2. Reactor: 15.3.1963
5. SAR	**Garching**	SSW	SSW	Argonaut (graphite water)	1 (for limited periods 10)	10^{11} (at 10 kW)	light water	graphite light water	enriched uranium (20%) 2 to 5.7 kg	23.6.1959
6. AEG PR 10	**Großwelzheim, nr. Aschaffenburg**	AEG	AEG	Argonaut (graphite water)	0.1	$3 \cdot 10^{8}$	light water	graphite light water	enriched uranium (20%) 2 to 5.7 kg	27.1.1961
7. FR 2	**Leopoldshafen nr. Karlsruhe**	Ges. f. Kernforschung mbH	developed by operators	heavy-water reactor	12,000	$3 \cdot 10^{13}$	heavy water	heavy water	natural uranium (thorium) 6 t U (1 t Th)	7.3.1961
8. FRJ 1 (MERLIN)	**Jülich**	Kernforschungsanlage, Jülich	TNPG—The National Power Group Ltd., Great Britain and AEG Rheinstahl	swimming pool	5,000	approx. $8 \cdot 10^{13}$	light water	light water	enriched uranium (80%) 3.2 kg	23.2.1962
9. FRJ 2 (DIDO)	**Jülich**	Kernforschungsanlage, Jülich	Copied by German firms (AEG Rheinstahl) from design developed by Head Wrightson Processes Ltd., Gr. Britain.	heavy-water tank reactor	10,000	approx. $1{,}6 \cdot 10^{14}$	heavy water	heavy water	enriched uranium (80-90%) approx. 2.88 kg	14.11.1962
10. STARK	**Leopoldshafen nr. Karlsruhe**	Ges. f. Kernforschung mbH	Arbeitsgemein. SSW, Lurgi, Pintsch-Bamag (developed by operators)	fast thermal two-zone Argonaut reactor	therm: 0.0065 fast: 0.0035	thermal zone: $8 \cdot 10^{7}$ fast zone: $1.5\text{-}2 \cdot 10^{7}$	light water	graphite light water	enriched uranium (20%) therm.: 5.6 kg fast: 90 kg	11.1.1963 (originally Argonaut) 24.6.1964 (converted for fast breeder development)
11. FRMZ	**Mainz**	Institut für Anorganische Chemie und Kernchemie, University of Mainz	General Atomics (USA)	TRIGA-pulse reactor	100: pulse 250,000	approx. $1 \cdot 10^{12}$ pulse: approx. $8 \cdot 10^{12}$	light water	zirconium hydride	enriched uranium (20%) 2.2 kg	under construction
12. PTB-Measuring Reactor	**Brunswick**	Physikalisch-Technische Bundesanstalt (PTB)	Deutsche Babcock & Wilcox	swimming pool (tank)	1000	$6 \cdot 10^{12}$	light water	light water	enriched uranium (90%) 3.2 kg	under construction
13. SNEAK	**Leopoldshafen nr. Karlsruhe**	Ges. f. Kernforschung mbH	Ges. f. Kernforschung and SSW	fast plutonium reactor			air		plutonium uranium (0.3 t Pu + 0.5 t U)	under construction

the centres by 1970, roughly 50% of which is accounted for by the construction and equipment projects already drawn up. The staff totals at present about 3,000 and 2,700 in Karlsruhe and Jülich respectively. The annual operating costs are 75 million DM for Karlsruhe and somewhat less for Jülich.

The main research facilities of the *Karlsruhe Nuclear Research Centre*, which was set up and is largely operated by the *Gesellschaft für Kernforschung*, consist of a tank-type heavy-water reactor with a thermal power of 12 MW and a thermal neutron flux of 3.10^{13} n/cm².sec, an Argonaut fast/thermal reactor *(STARK)* and a fast sub-critical assembly *(SUAK)*. A 200 MW (th) heavy-water "multi-purpose" reactor with a thermal neutron flux of 10^{14} n/cm².sec is almost completed, a fast zero-energy assembly *(SNEAK)* is under construction and a sodium-cooled, zirconium hydride-moderated experimental reactor is planned, work being scheduled to begin this year. In addition, the equipment available includes an isochronous cyclotron with a maximum particle energy of 110 MeV, several neutron generators, including several of the pulsed type, and an IBM 7070 computer. Several hot cells are under construction, and preparations are being made for a small irradiated fuel reprocessing unit. The Centre comprises institutes for neutron physics and reactor technology, applied nuclear physics, experimental nuclear physics, applied reactor physics, reactor components, hot chemistry, radiochemistry and radiobiology, including neutron biology and isotope laboratories, an electronics laboratory and a school of nuclear technology. It also incorporates the *European Institute for Transuranium Elements* and a number of research establishments operated by other bodies, such as institutes for nuclear engineering, isotope techniques and foodstuffs irradiation technology.

The *Jülich Nuclear Research Centre of the Land of North Rhine/Westphalia* has a *MERLIN*-type research reactor and a materials-testing reactor of the *DIDO*-type, with a thermal power of 5 and 10 MW and a maximum thermal neutron flux of 8.10^{13} and $1.6.10^{14}$ n/cm².sec respectively. At Jülich there are also two 3 MeV Van de Graaff generators. A cyclotron with a maximum particle energy of 180 MeV is now under construction, as well as a hot-cell unit. On the edge of the Centre's premises, the gas-cooled high temperature *AVR* test reactor, which is rated for a thermal power of 49 MW, is nearing completion. The Centre itself includes institutes and research teams engaged on botany, zoology, agriculture, medicine, physical chemistry, radiochemistry, chemical engineering, high-energy physics, nuclear physics, neutron physics, technical physics, reactor components, reactor development, reactor materials and plasma physics, together with central institutes for applied mathematics equipped with their own large-scale computer, and reactor experiments as well as central laboratories for electronics and chemical analysis. Moreover, research and development work in the field of isotope separation is carried out by the state-financed *Gesellschaft für Kernverfahrenstechnik*, which does not come under the organisational setup of the Centre but is located on its grounds.

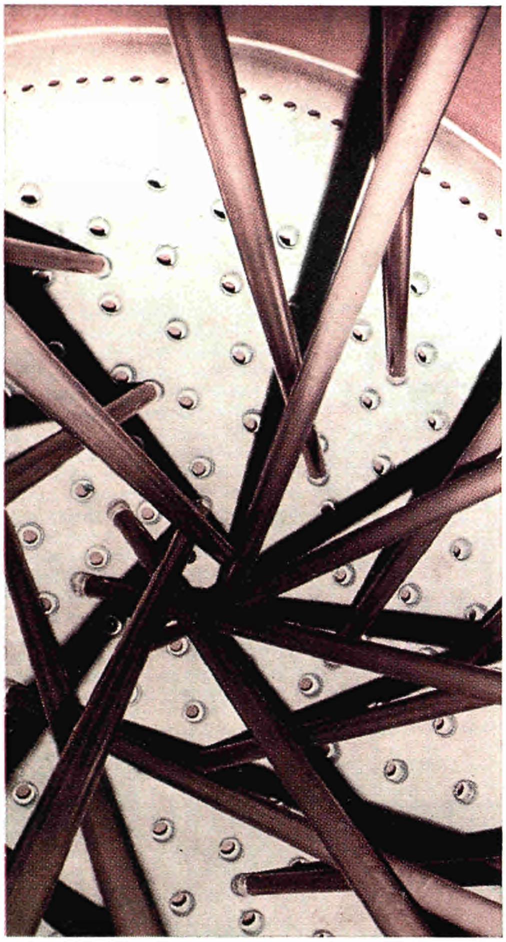


The FR2 *reactor of the Karlsruhe Nuclear Research Centre*

Technical Developments

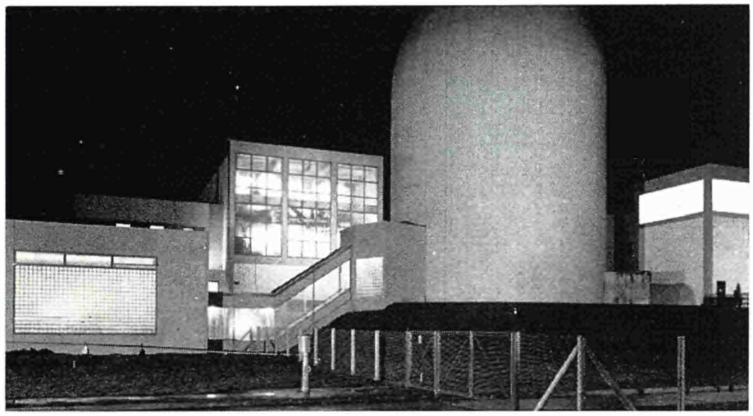
The work being conducted by Germany in this field is focussed on a three-pronged reactor development programme, consisting of a power plant demonstration programme, a medium-term development programme for improved converter reactors and a long-term development programme for fast and thermal breeders. Other development work is aimed, in particular, at effecting improvements in the fuel cycle and in nuclear materials. A small test facility for the processing of uranium ores is in operation in Ellweiler. The construction of a prototype reprocessing plant to operate by the aqueous solution method is envisaged in Karlsruhe, as mentioned above.

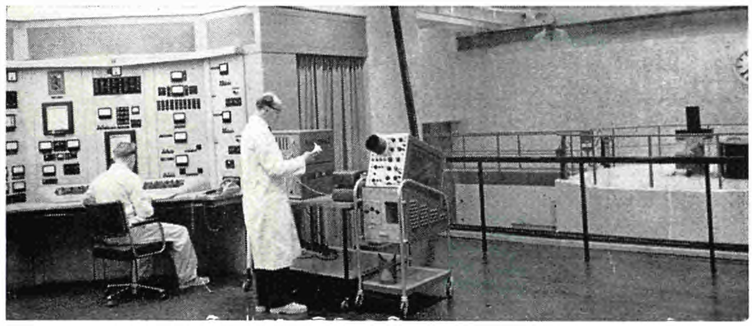
The *nuclear power plant demonstration programme* covers at the moment three power plants fitted with proven-type light-water reactors. Its purpose is to enable German industry and the German power economy to gain their own know-how in the construction and operation of large nuclear power plants and thus to create a sound basis for future decisions on investment. Since joint-enterprise status has been granted to two of the projects, as provided by the Euratom Treaty, and has been applied for in the case of the third, it will be possible to put the experience thus acquired at the disposal of the industries and power producers of the other Community countries. In all three instances the choice of reactor was governed not only by technical, but also and to a marked degree by economic considerations. Particular importance was assumed here by the general economic circumstances of the country and the structure of the electrical economy.



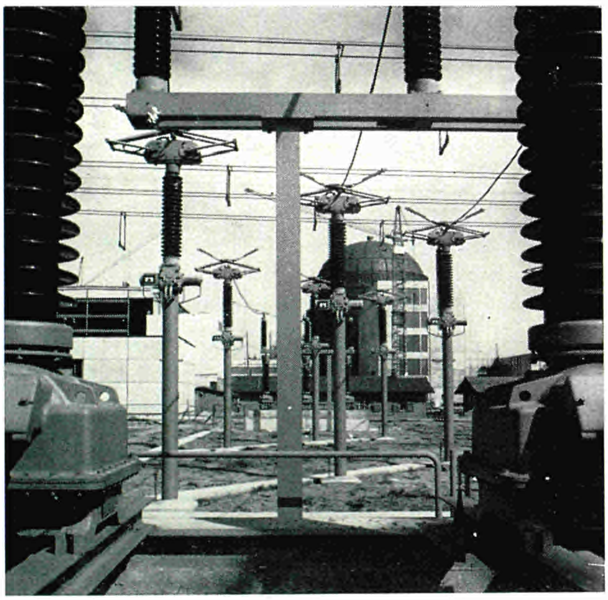
Nuclear Research Centre, Karlsruhe: view inside the pressure vessel of the multipurpose research reactor.

The reactor of Germany's first nuclear power plant at Kahl am Main.

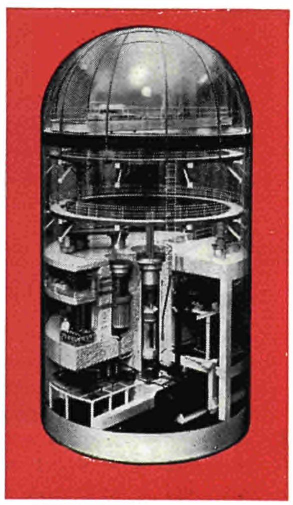




Control room and view of the atomic pile at the Hahn-Meitner Institute, *West Berlin.*



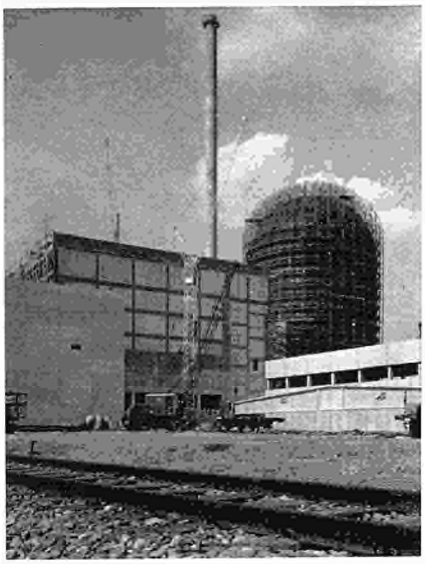
Nuclear Research Centre, Karlsruhe: view of the multipurpose research reactor.



Model of the 20 MWe sodium-cooled reactor in Karlsruhe, designed by Interatom.

All three projects are sponsored by the Federal Government, which assumes wide responsibility for the financial risks of operation, grants *ERP* loans and, together with the Länder, guarantees other loans and provides easy amortisation terms. The *Kernkraftwerk RWE-Bayernwerk* in *Gundremmingen* is also accorded special privileges under the Euratom/US nuclear power plant programme and the Euratom programme of participation. In the case of the *Lingen* and *Obrigheim* power plants, on the other hand, the Federal Government is to provide subsidies for the research and development costs incurred under these projects and for the initial fuel supply.

Consideration is now being given to the possibility of continuing the programme in modified form with the construction of a Franco-German joint power plant with a reactor of the gas-graphite string, to be followed up by a German-French reactor of the gas/heavy-water string as soon as technical and economic conditions permit.



The nuclear power plant at Gundremmingen on the Danube

The *medium-term development programme* is aimed at the design of power reactors with improved characteristics, in particular a higher fuel utilisation. The relatively extensive programme at present in being covers six construction projects, of which some are very far advanced while for others the decision to build has only just been taken, together with a number of design studies. These construction projects include the *AVR* experimental power plant at Jülich, the multi-purpose research reactor at Karlsruhe, a superheated steam reactor to be built alongside the Kahl experimental power plant, the Niederaichbach CO_2-D_2O prototype nuclear power plant, the compact sodium-cooled nuclear energy installation at Karlsruhe and the pressurised-water reactor for the *Otto Hahn*. The bulk of the cost of building these plants, including the cost of the corresponding research, development and test programmes, is borne by the Federal Government. Third parties contribute about 20% of this 650-700 million DM programme, variously distributed. Other studies supported by the Federal Government under the medium-term development programme relate to other reactor types, although it is as yet impossible to assess whether these projects are to be pursued further on completion of the present work.

The *long-term development programme* has two focal points: the development of fast breeder reactors, which today forms one of the major tasks of the Karlsruhe Nuclear Research Centre, and the development of thermal reactors with conversion factors of 1 or more, which in future will be concentrated more round the Jülich Centre. The work to be carried out under the long-term programmes is in both cases still in its initial stages. No decisions on the construction of large experimental and prototype facilities are likely before 1968. Until then, fairly comprehensive preparatory research and development programmes will be carried out, close-knit co-ordination on the European level being fostered by contracts of association with Euratom covering the entire field with regard to fast breeder reactors and, for thermal reactors with a high conversion rate, covering the gas-cooled, graphite-moderated thorium high-temperature design. Furthermore, participation in the *European Nuclear Energy Agency*'s *DRAGON* Project and agreements with similar projects in the United States help to create a spirit of fruitful international collaboration transcending the boundaries of the European Community.

Costs and Financing

The total outlay of the German economy for the research and technical development of nuclear energy amounted to about 3,000 million DM by the end of 1964, the bulk of this being provided from public funds, i.e. just under 2,000 million DM from the Federal budget and about 850 million from the Länder, while an estimated 250 million came from industry. Under the 1965 Federal budget a total of 523 million DM is earmarked for nuclear research and development. Of this sum, 349 million DM are intended for German projects and 174 million, or roughly a third, as contributions to international organisations or projects. Moreover, long-term loans totalling 35 million DM are to be granted from the *ERP* Special Fund for the Lingen and Obrigheim demonstration power plants. In addition, the Länder budgets are likely to allocate about 190 million DM for nuclear research and development in 1965.

The bulk of the public appropriations has hitherto been used for research purposes,

III. EXPERIMENTAL AND PROTOTYPE REACTORS

Name	Location	Owner	Manufacturer	Reactor Type	Thermal Power in MW	Net electric power MW	Coolant	Moderator	Fuel	State of construction or commissioning date
1. VAK	**Kahl/Main**	Versuchsatomkraftwerk, Kahl GmbH (RWE Bayernwerk)	AEG, GE (USA)	boiling-water	60	15	light water	light water	enriched uranium (2.3-2.5%)	in operation since 12.7.1961, full load 5.1.1962
2. AVR	**Jülich**	Arbeitsgemeinschaft Versuchsreaktor GmbH, Düsseldorf	Brown Boveri/Krupp Reaktorbau GmbH	gas-cooled high-temperature reactor (pebble-bed reactor)	49	15	helium-neon mixture	graphite	enriched uranium (20%) as uranium carbide with thorium carbide as breeder material in graphite spheres	under construction since January 1961, scheduled to go into operation in 1965
3. MZFR	**Leopoldshafen nr. Karlsruhe**	Gesellschaft f. Kernforschung mbH	SSW	D_2O pressure vessel reactor	200	50	heavy water	heavy water	natural uranium	under construction since 1961, scheduled to go into operation in 1966
4. FDR	**Marine reactor for „Otto Hahn"**	Gesellschaft f. Kernenergieverwertung in Schiffbau u. Schiffahrt mbH	Deutsche Babcock & Wilcox/Interatom	pressurised-water integral design	38	10,000 shp	light water	light water	enriched uranium (3.6%)	under construction since early 1964, scheduled to go into operation in 1966
5. KKN	**Niederaichbach, near Landshut**	Bayernwerk AG, Gesellschaft f. Kernforschung mbH	SSW	D_2O pressure tube reactor with CO_2 cooling	400	100	carbon dioxide	heavy water	slightly enriched uranium	planned
6. HDR	**Kahl/Main**	Gesellschaft f. Kernforschung mbH	AEG	boiling water with nuclear superheat	100	25	light water	light water	enriched uranium	under construction since March 1965, scheduled to go into operation in 1968
7. KNK	**Leopoldshafen near Karlsruhe**	Gesellschaft f. Kernforschung mbH	Interatom	zirconium-hydride moderated sodium-cooled reactor	58	20	sodium	zirconium hydride	slightly enriched uranium as uranium carbide	planned

largely for the construction of research installations. Recently, however, there has been a marked shift in the general pattern observed so far, in that the current expenditure on actual research and also on technical development work has been stepped up. This trend will probably become even more pronounced in the future. At the same time, a rise in total expenditure is to be expected during the next few years.

Future outlook

The coming years will probably see a marked increase in Germany's nuclear research activities as the new research centres are gradually completed as well as an attendant intensification of international exchanges. With regard to technical developments, the foreseeable future will most likely be influenced primarily by the experience gained in the construction and operation of the various experimental projects which are provided for under the medium-term programme, and which have attained varying stages of advancement, and by the drawing up of the decisions which will point the way ahead for the long-term programme. The really decisive factor governing nuclear developments in the near future is likely, however, to be found in the utilisation of nuclear energy, which is gradually becoming economically competitive.

At the moment there is only one nuclear power plant in operation in West-Germany, namely, the Kahl experimental power plant. However, this year the Karlsruhe multipurpose research reactor and the *AVR* power plant at Jülich will also be completed. In 1966 the *Kernkraftwerk RWE-Bayernwerk* at Gundremmingen is to go into operation, followed two years later by the Lingen and Obrigheim power plants and the Kahl superheated steam reactor. Furthermore, more contracts for the construction of large nuclear power plants are to be expected, which will probably be rapidly succeeded by others once the demonstration plants have been successfully put through their paces. An installed nuclear power output of 5000 MW by the mid-seventies and of 20,000 MW by 1980 now seems quite feasible for Germany.

It is obvious that such a course of events must have decisive repercussions on the scientific and technical programmes in progress and also on the entire set-up of Germany's activities in the nuclear field. In certain instances this may well mean winding up projects in which so much hope is still placed, but generally speaking such

a development is likely to fulfil precisely the expectations which we have of our activities in a field of whose existence there was little more than a vague idea less than three decades ago.

Notes

1. A survey of the position occupied by science in Germany is given in *Bundesbericht Forschung I*—a Government report on the status and inter-relationship of all the measures taken by the Federal Republic to promote scientific research, together with a forecast of the country's financial requirements for the period 1966-1968, published by the Federal Ministry for Scientific Research, Bonn, January 1965.
2. The gross national product of the Federal Republic, including West Berlin, in 1964 was DM 412,500 million. In real terms this represents an increase of 6.5% over the preceding year and a yearly average of just under 5.4% since 1950. National investment in fixed assets reached nearly DM 108,000 million in 1964.
3. The legal framework for activity in the field of nuclear energy was provided by the law of 23 December 1959 for the amendment of the Constitution and by the law on the peaceful use of atomic energy and the protection against nuclear hazards *(Atomic Energy Law)*, both of the same date. By virtue of the law several further ordinances have since been promulgated, among which are the *First and Second Radiologica Protection Ordinances*, the *Nuclear Plants Ordinance* and the *Nuclear Insurance Ordinance*.
4. The Federal Republic of Germany is at present a member of the *International Atomic Energy Agency (IAEA)*, the *European Nuclear Energy Agency (ENEA)*, the *European Atomic Energy Community (Euratom)*, the *European Organisation for Nuclear Research (CERN)* and the *European Atomic Energy Society*. In addition, the Federal Republic has bilateral agreements with Britain, Canada and the U.S.A. and close relations are maintained with a number of other countries as well as between German and foreign research institutions and between German and foreign industrial enterprises, particularly French, British and American concerns.
5. The total consumption of primary energy in the Federal Republic, including West Berlin, in 1963 was 250 million tons coal equivalent (t.c.e.), of which 124 million t.c.e. was accounted for by hard coal, 34 million by lignite, 81 million by oil, a little under 2 million by natural gas and the remainder by hydroelectric power and miscellaneous energy sources. The gross output of electricity in 1964 amounted to 164,500 million kWh, of which about 92% was generated at thermal power stations. The installed power-plant capacity at the end of 1964 was roughly 36,000 MW; this is likely to increase to nearly 100,000 MW by 1980.

IV. POWER REACTORS

Name	Location	Owner	Manufacturer	Reactor Type	Thermal Power in MW	Net Electric Power MW	Coolant	Moderator	Fuel	State of construction or commissioning date
1. KRB	Gundremmingen	Kernkraftwerk RWE-Bayernwerk GmbH	IGEOSA, AEG, Hochtief	twin-circuit boiling-water reactor	801	237	light water	light water	enriched uranium (2.4%)	under construction since 1962, scheduled to go into operation in 1966
2. KWL	Lingen/Ems	Kernkraftwerk Lingen mbH (Vereinigte Elektrizitätswerke Westfalen AEG, group of banks)	AEG	boiling-water reactor with fossil superheater	820	250 (90 fossil)	light water	light water	enriched uranium (2.6%)	under construction* since end of 1964, scheduled to go into operation in 1968
3. KWO	Obrigheim am Neckar	Kernkraftwerk Obrigheim GmbH (Energieversorgungsunternehmen in Baden-Württemberg)	SSW	pressurised-water reactor	908	282	light water	light water	enriched uranium (3%)	construction commenced on 15.3.1965 scheduled to go into operation in 1968

EUBU 4-6

Energy out of "pebbles"

PROFESSOR R. SCHULTEN, *Director of the Institute for Reactor Development, Head of the THTR-Project, Jülich Nuclear Research Establishment*

The *AVR* reactor shares with other gas-cooled high-temperature reactors *(Dragon* and *Peach Bottom)* certain basic features:
— graphite is used both as moderator and structural material for the core;
— the fuel is made up of particles of uranium and thorium carbides coated with carbon and dispersed in a graphite matrix, which in its turn is held in a graphite structure;
— the coolant gas is helium.

The *AVR* reactor therefore shares with *Dragon* and *Peach Bottom* certain advantages inherent in these systems, such as:—*high efficiency*, since the high gas temperature permits the use of modern steam cycles;
— *good conversion* of fertile elements into fissile elements (in particular thorium-232 into uranium-233) because of the absence of neutron-absorbing structural materials in the core;
— *compactness* (and therefore moderate cost) because the presence of heat-resistant ceramic materials only in the core makes it possible to rely on large temperature differences rather than large heating surfaces for good heat transfer.

The AVR reactor

At the end of 1961, the firm of *Brown-Boveri Company-Krupp*, acting on the instructions of the *Arbeitsgemeinschaft Versuchsreaktor* company *(AVR)*, a subsidiary of 15 German municipal electricity authorities, began work on the construction of an atomic power plant of 15 MWe equipped with a gas-cooled high-temperature reactor fuelled with pebble-type elements. Work with the reactor has now progressed so far that it is likely to be commissioned by mid-1965.

The pebble-bed principle

There is however one important feature which is unique to the *AVR* reactor. This is the shape of the fuel elements, which are spherical "pebbles" of 6 cm diameter. There is no question of fitting these pebbles neatly into channels; they are designed, literally, to lie in a heap. Loading a new fuel-element into the *AVR* core boils down simply to dropping a pebble on to this heap; unloading a fuel element just means extracting a pebble from the bottom of the heap. Protection against this indisputably rough way of handling a fuel element is afforded by the pebbles' outer graphite casing. Repeated tests have shown that a casing of 10 mm thickness is adequate to reduce the probability of damage to a small value. Although it is unlikely that anyone would dispute the simplicity of this system, it may give the impression of being somewhat haphazard. Yet this is not so: it is possible to predict the "flow" of the pebbles through the core within fairly fine statistical limits, as experimental work on models has shown. In view of the fundamental simplicity of the arrangement, loading and unloading during operation of the reactor present few problems. The *AVR* core is therefore very flexible.

Whereas most reactor designs have to permit adjustment of the core's reactivity through the provision of neutron-absorbing control rods, which are inserted into the core at the beginning of its life and gradually removed as the fuel is spent, control rods are superfluous in the case of the pebble-bed; admittedly *safety* rods cannot be dispensed with for obvious reasons, but the day-to-day need for fine adjustments to the reactivity can be satisfied simply by removing or adding pebbles, by reshuffling them, by modifying the gas flow etc. The neutron economy of the system is therefore substantially improved.

Moreover, the design offers the possibility of checking the burn-up and the physical condition of the pebbles at the unloading points and thus deciding whether they should be removed or reinserted and, if so, where. The idea of making a computer make all these decisions is being developed.

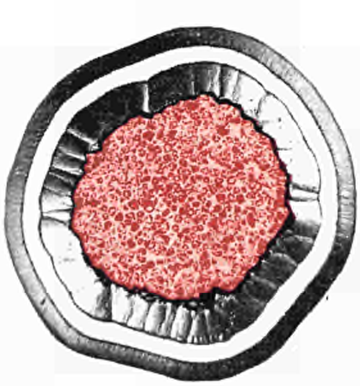
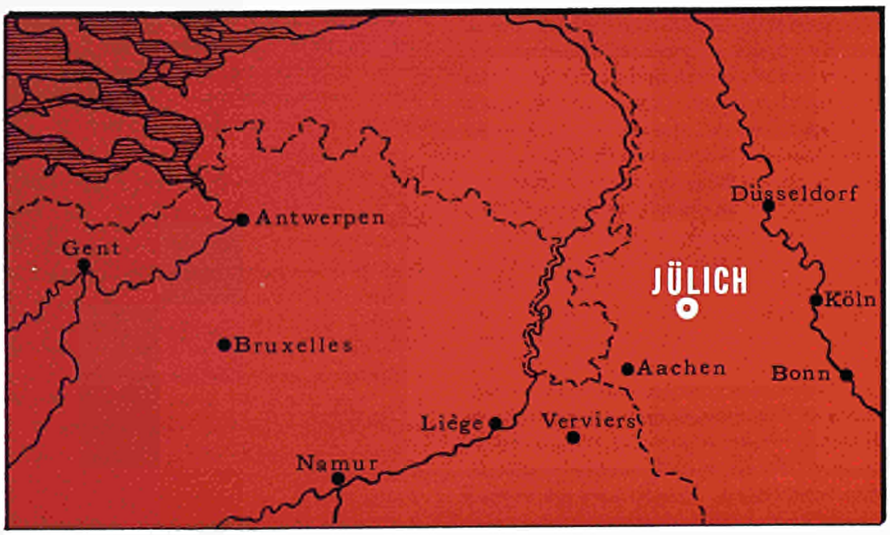


Figure 1: Cross-section of a fuel particle coated with graphite, silicium carbide and again graphite. The average diameter of such a particle is 0.5 mm. The main purpose of the coating is to keep in fission products.

The *AVR* reactor was originally designed to run on fuel fitted with no special cladding material, apart from graphite, so that a high level of activity could be expected in the primary (gas) circuit in view of the porosity of graphite. As a result of more recent development work, especially on coated particles (see figure 1) fuel elements can now be employed which retain fission products and therefore reduce activity from this source to a low level. With regard to gas-cleaning and radiation safety, therefore, the reactor is overdimensioned. In the future, it will thus constitute an excellent means for testing fuel elements to failure. The initial charge of 30,000 pebble-type fuel elements is to be delivered in June 1965. They consist of hollow graphite balls, fitted with screw-plugs, into which a mixture of coated particles and graphite is pressed. The thermal conductivity of this mixture will be high enough to ensure that an internal temperature of 1,250°C is not exceeded at an average power of 1,5 kW per element.

Previous experiments with coated particles are very encouraging and burn-ups of up to 200,000 MW days per ton of heavy metal and at temperatures of up to 1,400°C show a fission product release which is sufficiently small to enable various components, such as the gas blowers and the heat-exchangers, to be removed or inspected soon after full-load operation without complicated remote handling operations.



The reactor is to be run on helium at an outlet temperature of up to 850°C, so that modern steam conditions can be achieved in the steam circuit.

Figures 2 and 3 show the layout of the reactor in the main building. The reactor is of integral design, i.e. the core and the steam generator are situated in the same cylindrical vessel. Although this layout is not strictly necessary in the case of a small reactor, it is a must for larger units, so that the *AVR* reactor is a genuine prototype of a future large power plant.

The THTR programme

The reactor family of which the *AVR* is the first member has been given the name *Thorium-Hochtemperatur-Reaktor (THTR)*. A programme aiming at the development of the concept is being pursued under a Euratom contract of association, which was signed in June 1964. Besides Euratom, the partners of this contract are the *Kernforschungsanlage des Landes Nordrhein-Westfalen (KFA)* and *Brown Boveri & Cie/Krupp*. The programme subdivides into three parts:

— *a research and development programme* covering the physics and chemistry of the reactor, the development of the fuel (under sub-contract to *Nukem*) and the solution of various technological problems raised by the concept: loading and unloading circuit, complete with automatic measurement of each pebble's burn-up, pressure vessel, heat-exchangers, blowers, etc.;

— *the design of a prototype* sufficiently large to be capable of being scaled up to units of 1000 MWe;

— *the utilisation of the AVR reactor*, which as soon as it starts up, will be able to serve as a large-scale experiment for the compilation of first hand practical data on the way a pebble-bed reactor works and on the way the fuel stands up to irradiation. This part of the contract will be covered by a subcontract which is being negotiated at the moment with the *AVR* Company, the owners of the reactor.

Prestressed concrete for the pressure vessels

Initial investigations have shown that the use of steel pressure vessels is not economic for units of more than about 300 MWe.

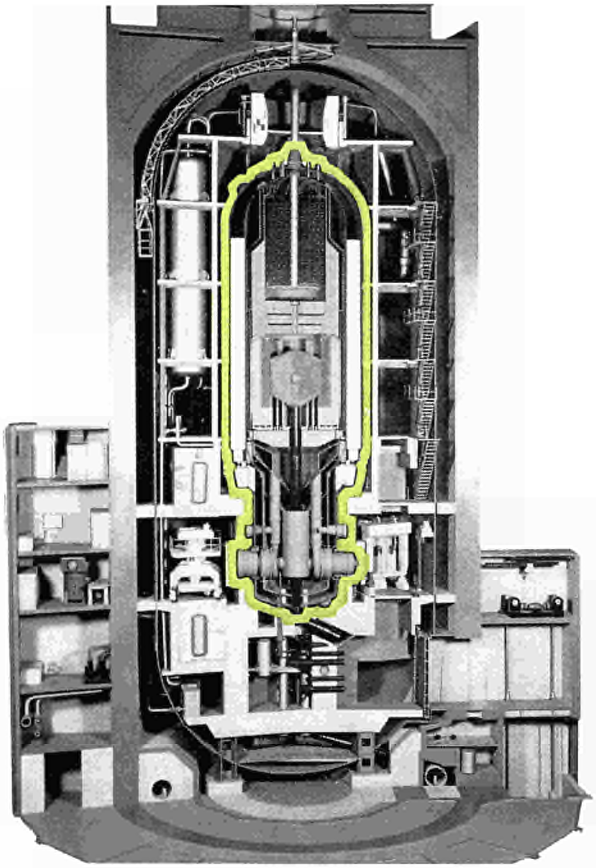


Figure 2: Model of the AVR reactor

For units of this size and larger, prestressed concrete vessels are preferable. As technological developments stand at present, experience gained in this field in France and Britain indicate that in the next few decades there should virtually be no limit to the capacity of high-temperature reactors from this point of view. Under the development programme models are to be built and design work carried out aimed at making the construction of a prestressed concrete vessel for the high-temperature reactor possible by 1967. The contract for the design and development of the vessel has been let to a German-French-Italian group.

Helium-lubrication for the blowers

According to the provisional information supplied by one of the firms participating in the contract, the results obtained in the development of helium-lubricated blowers are so promising that such machines of over 1000 kW can be expected to go into use within the next few years. The use of helium-lubricated blowers has therefore been provided for in the design of the prototype *THTR* reactor, instead of the oil-lubricated blowers used in *AVR*. Estimates for the more distant future suggest that blowers of this type of up to 3000 kW can be built, so that even larger nuclear power plant units can be fitted with these blowers.

Integrated steam generators

Considerable advances on the *AVR* design are also planned in connection with steam generators. If a pressure of 50 atm. of helium is used, the power density of the steam generators can be increased considerably. Construction experience hitherto shows that in particular very meticulous testing of the tube material is necessary in order to obtain steam generators having a low water leak-rate and a sufficiently long life.
A provisional sectional drawing of the *THTR* prototype is given in figure 4, from which it can be seen that the entire gas-circuit of the system is inside the reinforced concrete vessel. The reactor, complete with safety rods and fuel-removal gear, is mounted at the bottom, while above it are the steam generators, separated from the reactor by thick shielding, designed to reduce activity by neutron capture to a minimum. Thanks to this shielding and to the low fission product release made possible by the use of coated fuel particles, there should be no difficulty about removing faulty steam-generators soon after full-load operation. The steam generators are subdivided in such a way that an individual defective tube system can be switched off, resulting simply in a heating surface loss of less than 1%. Since a margin of at least 20% is built in the design from the start, it is unlikely that any component of the steam generators will have to be removed during the lifetime of the reactor.
The upper part shows two helium-lubricated blowers which convey the gas back to the reactor on the cold side of the steam generator.

Capital costs

Calculations of the cost of *THTR* type power reactors carried out to date show, as in the case of a preliminary design study carried out on a 500 MWe *Dragon*-type reactor,[1] that capital costs will at the most be equivalent to those of light water reactors. It is the higher efficiency of high temperature reactors (40-44% as opposed to some 30% in the case of light water reactors) which justifies these expectations. It must not be forgotten that a substantial portion (not much less than half) of the capital cost of a power reactor is at the moment accounted for by the conventional part of the installation. Thanks to the high efficiency of the system, modern, compact and therefore less costly steam turbines, for instance, can be used. The result is a saving which can amount to about 40% of the cost of the conventional plant.

High burn-ups . . .

The *THTR* will, like the *AVR*, use coated particle fuel elements. The high burn-up possible with this type of fuel can be exploited by the use of a loading method involving two types of fuel elements. In the first type thorium is mixed with fissile material in equilibrium concentration. The second type consists of pure fissile material

1. cf. *Euratom Bulletin* 1964 No. 3, pp.20-23

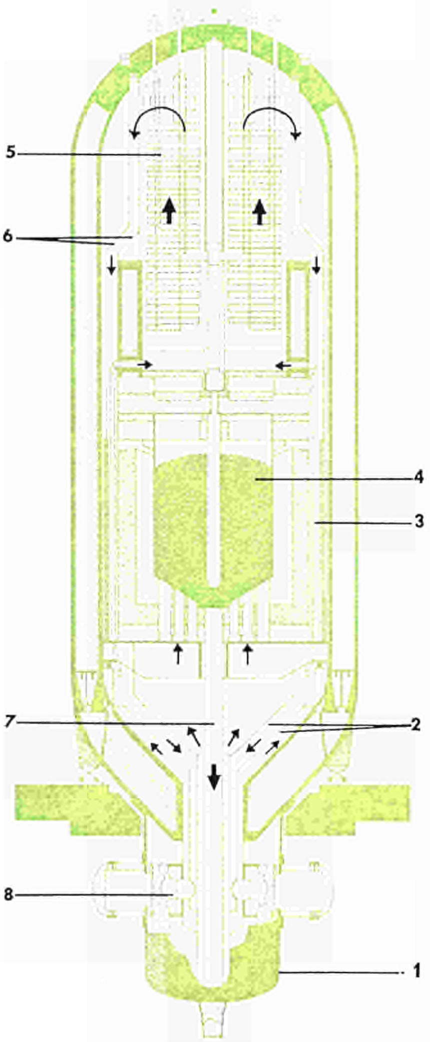


Figure 3: Vertical cross-section of the AVR reactor

1. *Part of pressure-vessel housing blowers and charge/discharge equipment*
2. *Cooling gas jackets*
3. *By-pass tubes for direct cooling of steam generator*
4. *Reactor core*
5. *Steam generator*
6. *Gas return jackets*
7. *Fuel extraction duct*
8. *Blower*

(e.g. uranium-235). While the bulk of the energy is produced by the first type of fuel element with the maximum possible lifetime (up to 200.000 MWd/t), in which at least 80% new fuel is also produced by conversion processes, the second type of fuel element is principally used to generate the extra neutrons necessary for maintaining the chain reaction. When this mixed loading method is used, a conversion factor of about 0,8 can therefore be expected at the burn-up given above.
If *THTR* reactors are designed on this principle, the fuel cycle will have the advantage of being simple: the fuel will be left in the core up to the maximum burn-up and then disposed of. An objection to this solution is that it does not give the *THTR* concept its full chances as a converter: high burn-ups entail the presence of large amounts of fission products which capture many of the neutrons produced in the reactor and thus divert them from the more useful task of sparking off the conversion of thorium into uranium-233. Nevertheless, there is every indication that *THTR* reactors can be economical under these conditions, giving lower fuel costs than light water reactors, for instance.

. . . or higher conversion factors

Studies are to be carried out later to determine whether a higher conversion factor is economically more advisable. The fuel cycle is then no longer so simple, of course: the average burn-up must be kept lower and the fuel has to be reprocessed to permit reutilisation of both the unspent and the freshly produced fissile material.
It should be noted that with the *THTR* design, whatever solution is adopted, a high specific power can be attained, which means that only a relatively small amount of fissile material per kW will be required. A specific power of 5 kW*th* per gramme of fissile material can be expected. Thus, for a given output, a relatively small investment in fissile material will be needed and this may be extremely important in the future in view of the possible shortage of uranium. If it is decided to take the conversion factor up to 1.0 or more (in which case it would be more appropriate to talk of breeding), additional measures have to be taken in the design of the reactor. If large reactors with an output of about 1,000 MWe are considered, the neutron leakage is so small that it is reasonable to expect a conversion factor of approximately 1.0, but there do occur a few neutron losses, which can be briefly summarised as follows:
The conversion of thorium-232 into uranium-233 involves an intermediate step, which is the formation of protactinium-233 after neutron capture. Protactinium-233 then decays to uranium-233 with a half-life of a little over 27 days. This happy result is unfortunately not achieved if, in the meantime, the protactinium nucleus captures a neutron, because it then leads to the formation of a non-fissile, and therefore useless, uranium-234 nucleus.
The conversion process into fissile uranium-233 occurs most readily when the impinging neutrons are at high *energies* whereas well-moderated neutrons are required for the neutron efficiency of the fission process. Furthermore, it is clear that if the neutron *flux* is low, less events will take place and the protactinium-233 formed will have a better chance of decaying into uranium-233 before it is, as it were, tempted to capture a neutron and turn into uranium-234.
It follows that if the advantages of both a high specific power and a high conversion ratio are wanted, there is competition between the two aims: a high specific power requires a high flux of well-moderated, fully thermalised neutrons; a high conversion ratio requires a low flux of neutrons at higher energies. Both aims can however be largely fulfilled providing a sufficiently high investment is accepted on the reprocessing plant. Recent measurements of nuclear phenomena of paramount importance to the thorium fuel-cycle have actually shown that previous measurements overestimated neutron capture by protactinium-233 and underestimated the neutron yield of uranium-233. In these circumstances, the cost of reprocessing and re-fabrication per unit sent out need not become prohibitive.

The use of plutonium as the initial fuel charge is perfectly feasible in a *THTR* reactor and preliminary calculations show that this fissile material could compete with uranium-235 in the future, but some development work must first be carried out on the fabrication of coated particles with plutonium.
Reactors like the *THTR* in which high conversion ratios can be attained offer the

The 15 MWe pebble-bed experimental power reactor of the Arbeitsgemeinschaft Versuchsreaktor (AVR) in Jülich

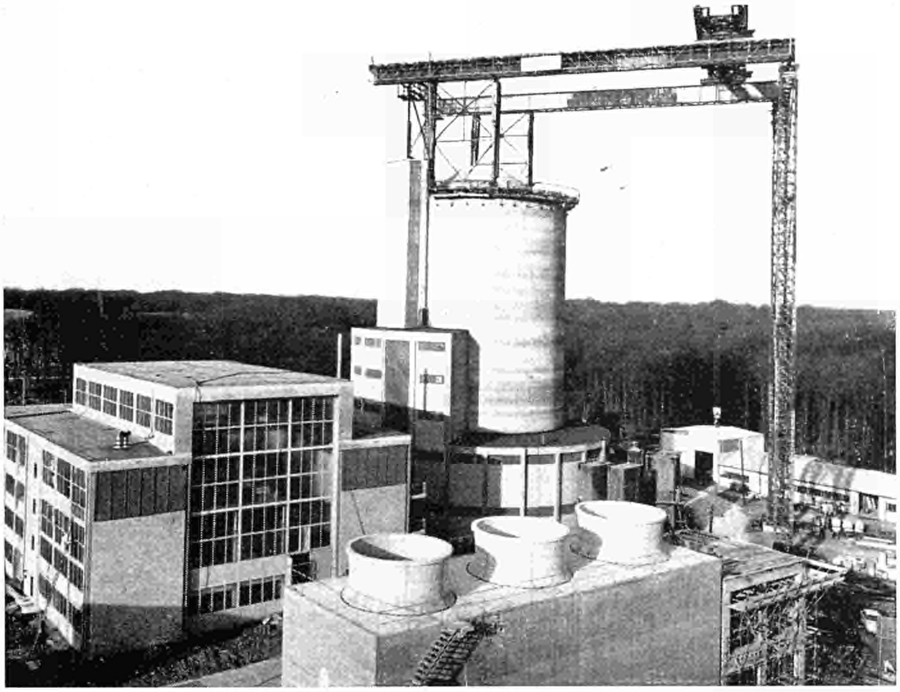
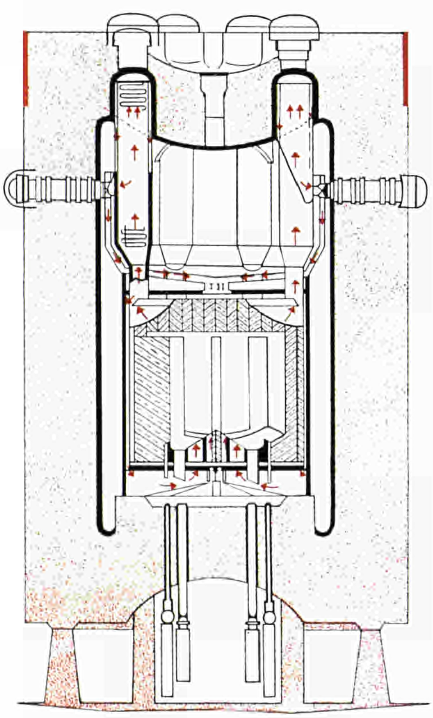


Figure 4: Provisional sectional drawing of the THTR-prototype



advantage that costlier natural uranium deposits can be used without any appreciable rise in fuel costs, in view of the fact that the bulk of these costs is represented by fabrication and refabrication after reprocessing. It is estimated that, with a conversion ratio of about 0.8, a 100% increase in the cost of uranium ore would involve a 10% rise in fuel costs. This could lead, in view of the fuel shortage predicted for the next few decades by some sources, to a considerable enhancement of the long-term prospects for converter reactors and could even give them a permanently assured position.

This would be truer still if the conversion ratio were pushed up to 1, since the cost of uranium ore would then merely be reflected in the payment of interest on the cost of the initial fuel charge.

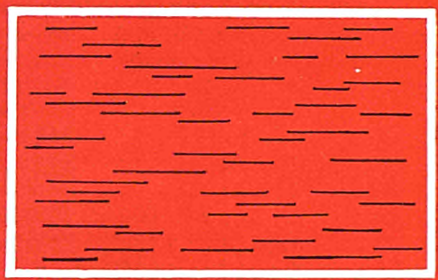
As opposed to the other high-temperature reactors, the *AVR* and the *THTR* are based on the technology of pebble-type fuel elements. At the present moment it is impossible to state categorically whether the use of pebble-type or rod-shaped fuel elements offers the better solution in high-temperature reactors. The pebble-type permits continuous loading of the reactor, thus obviating the need for excess reactivity to offset the depletion of the fissile material and the accumulation of fission products. In addition, it is an advantage that the mechanical and nuclear properties of the fuel elements can be observed during their lifetime in the reactor. The frequently expressed view to the effect that a higher relative blower power is required for a reactor run on pebble-type fuel elements is not valid, since lower gas velocities are required in the pebble-bed for the same power density. Furthermore, the pressure drop in large reactor units is caused not so much by the reactor itself as by the steam generator. In the *THTR*, for instance, the pressure drop in the reactor will be about 0.5 atm. and in the steam generator about 1.0 atm. The total blower power required is relatively small and totals about 2-3% of the net electric power.

The *THTR* is a reactor offering much lower fuel costs in comparison with light-water reactors, at about *the same* capital cost. In the next decade it will certainly be competitive with other reactor systems.

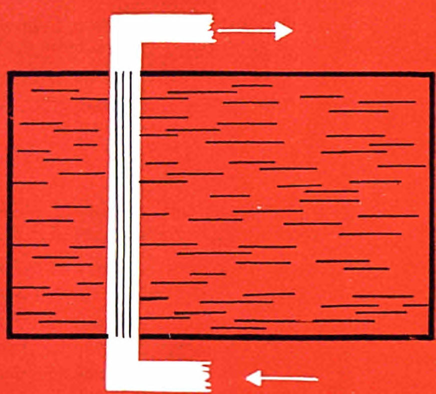
What is ORGEL? A brief "recap"

ORGEL belongs to the heavy-water-moderated reactor family. Its reference fuel is uranium carbide (UC) and it is cooled by an organic liquid.

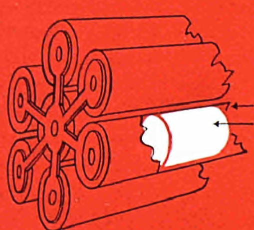
As currently conceived, an ORGEL reactor would have:—a cylindrical steel vessel containing heavy water,



traversed by ORGEL channels in which the fuel is contained and the coolant circulates:



The fuel element is in the form of a cluster of uranium carbide rodlets:

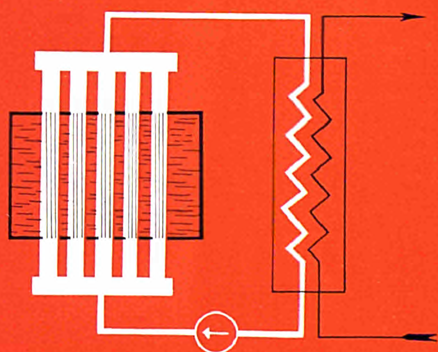


The organic coolant has an advantage over water by virtue of its low vapour pressure. For example, at a temperature of 350°C water would have a pressure of 170 kg/cm², whereas that of the coolant adopted for ORGEL is 7 kg/ cm².

On the other hand, the organic coolant has several drawbacks, such as its decomposition under the action of heat (pyrolysis) and radiation (radiolysis); hence the need for a purification system. However, another point in favour of organic liquids is their chemical compatibility with most materials in common use, such as aluminium. There is accordingly no need for large-scale investments in costly materials such as stainless steel.

Lastly, through the use of heavy water as a moderator, ORGEL has a good neutron economy, which means that fuel enrichment is not essential and that high conversion rates (e.g. uranium-238 into plutonium-239 or thorium-232 into uranium-233) are possible.

The nuclear part proper of an ORGEL reactor is supplemented by a conventional installation in which heat is employed for the generation of electricity by means of a water/steam circuit.



How far have we got with the ORGEL Project? See the next four pages.

EUBU 4-7

How far have we got with the Orgel Project?

SERGE ORLOWSKI, *Directorate-General for Research and Training, ORGEL Project Directorate – JRC, Ispra*

In 1960 Euratom launched the so-called *ORGEL* project, a design study for a heavy-water-cooled organic-liquid-moderated reactor.

For the last four years research, experiment and construction have succeeded one another under this programme, through contracts with private firms or national organisations and with the growing support of the Joint Research Centre's establishment at Ispra. How do we stand today?

The broad lines of the concept are unchanged: the reference fuel is still the uranium carbide chosen at the outset as combining the ability of ceramics to withstand high temperatures with a better heat conductivity and uranium content than can be obtained with uranium oxide; the basic organic coolant consists of terphenyl isomers (meta-, ortho- and paraterphenyls) mixed in suitable proportions (Fig. 1), while SAP (Sintered Aluminium Powder) is still the foremost core structural material, although advanced studies are also in progress on zirconium and its alloys.

This is not for lack of alternatives: the organic coolant, in contact with the structural and cladding materials, and, in the event of failure, with the moderator and fuel, displays low powers of chemical attack, a property which opens up prospects of a number of combinations (Fig. 2); nor is it due to a spirit of conservatism, for certain of these combinations are studied alongside with the reference solution, though less exhaustively. The fact is that research so far has corroborated the fundamental choices made at the beginning, particularly as regards the fuel and the coolant.

The Research Programme

Coolant

The terphenyl mixture OM2 adopted as reference fluid has proved, after several in-pile irradiation campaigns, the most stable at high temperature (up to 450°C bulk temperature and 500°C wall temperature, the latter having no effect on the decomposition rate).

Of the petroleum derivates studied under contract, some, such as the methylnaphthalene fractions, were abandoned two years ago as their vapour pressure was distinctly higher than that of OM2; others, like the alkylphenanthrenes, display a lower pyrolytic stability than OM2 and can hardly be used in-pile above 380°C; they were nevertheless investigated until recently because they are liquid at ambient temperature, which might simplify the problems involved in the initial startup of *ORGEL* reactors.

Hence a considerable effort has been made to find out all about the terphenyl mixtures and their pyrolitic and radiolytic decomposition products. Methods of analysis, of measuring physical properties and determining impurities have been developed and extended beyond, sometimes well beyond, 400°C.

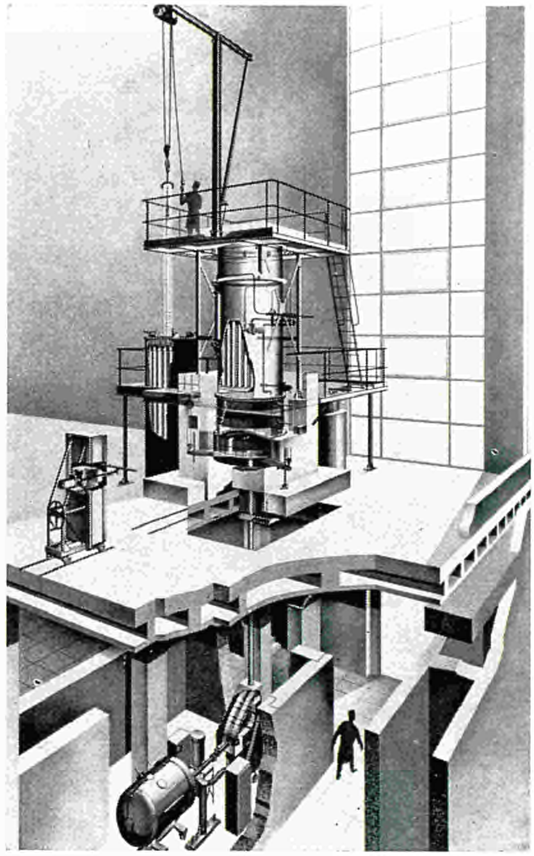
Current cleanup techniques make it possible to avoid fouling at the operating temperatures envisaged, even though its basic mechanisms are not yet clearly understood.

Fuel

As regards uranium carbide, the initial problem was how to produce it on a larger than laboratory scale and under stringent conditions of composition, homogeneity and dimensional tolerances. One difficulty is that strict controls must be applied to obtain the UC monocarbide required, otherwise one gets mixtures of UC-U (substoichiometric) which appear to be less stable under irradiation, or UC-UC_2 (hyperstoichiometric) which may raise problems of chemical compatibility with the cladding. The manufacturing question can now be regarded as solved, since an order of 7 tons of carbide rods for critical experiments in the *ECO* reactor has been delivered to Ispra, and the various tolerances stipulated by Euratom have been complied with. For this "small pilot" contract the price of the finished rods was 90 EMA u.a. per kg contained uranium; further research is now aimed at the industrial manufacture of this fuel at a price which should not exceed 50 EMA u.a. per kg, as well as at irradiation tests which, as will be shown below, had already begun satisfactorily during the fuel element development work.

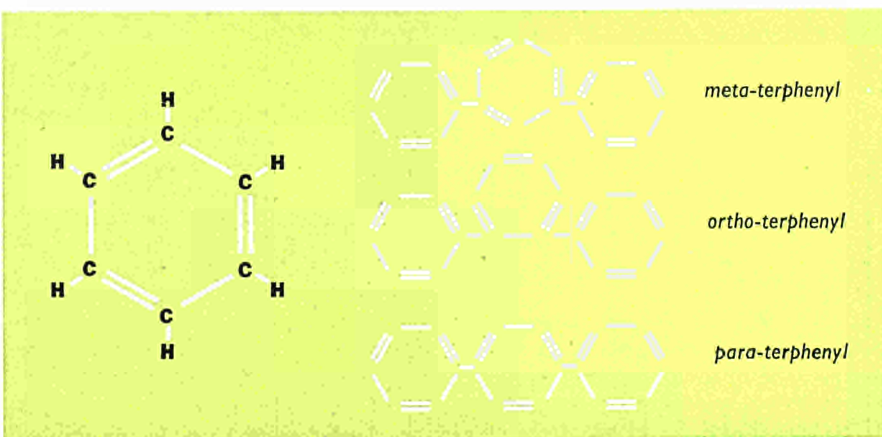
Cladding and structural materials

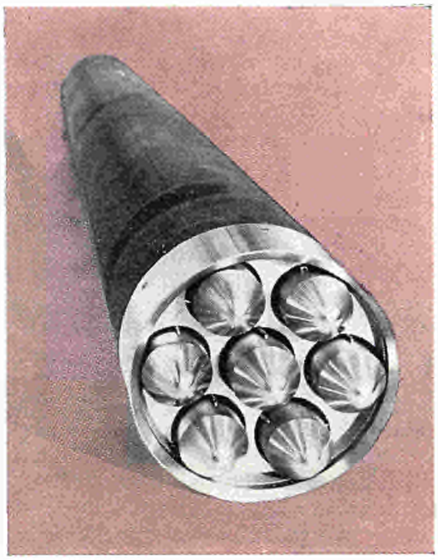
Work in this area was focussed on sintered aluminium-alumina (SAP). This material, which is not new (the first patent was taken out in 1946 by the firm *Alu-Suisse*), seems to have been quite well known as early as 1960. But really valid test results can only be obtained on finished products subjected to the conditions which they will encounter in use; and it was very soon found that SAP was liable to blistering towards 500°C, a phenomenon which was eliminated by thorough degassing of the starting powders. The next step was to develop commercial methods of manufacturing tubes of small diameter (for cladding) and large diameter (for channels), side by side with processes for analysis, for destructive and non-destructive inspections, and for purifying the starting powders; as far as the clads are concerned, this milestone can be considered as passed. And although SAP is still a brittle material in the case of slow phenomena (creep), the irradiation experiments carried out on SAP-clad carbide pins in the *NRX* reactor under the Euratom/Canada agreement showed that the concept is perfectly viable at the burn-ups envisaged (10,000 MWd/t).
At the same time active research is going on into zirconium and its alloys, which may make for more flexible design; it is hoped in this way to benefit from the technological development work devoted to water reactors. The problems involved in the formation of zirconium hydride and the resultant embrittlement, which affect both reactor strings, still remain to be overcome, however. Lastly, preliminary studies carried out at Ispra indicated that magnesium might also be suitable for an organic-liquid-cooled reactor under certain conditions; in that case, it would probably be necessary to use tubular fuel elements, not unlike those which are planned for the French *EDF 5* gas-graphite plant.
All this means that, although the *ORGEL* reference fuel element is still a bundle of 7 or 4 SAP-clad uranium carbide pins, there are other variants awaiting exploration, particularly in the field of tubular



EXPO *exponential experiment, Ispra*

Figure 1: Terphenyls. The basic coolant adopted for ORGEL *is a mixture of the terphenyl isomers. Their molecules are composed of three benzene rings (on the left); alteration of the position of the rings in relation to the central ring produces the three isomers:*





ORGEL *pressure-tubes*
Two variants, one with four and the other with seven fuel pins.

elements, which might be produced very cheaply.

What has actually been built?

Plant has been constructed for the testing, from both the neutronic and technological angle of *ORGEL* reactor sub-assemblies ranging from the smallest to the largest. Thus, besides special-purpose in-pile and out-of-pile loops (for specific studies on fouling, heat exchange, corrosion, the *ORGEL* channel, pumps, etc.), two experiments due to be run in parallel, a critical experiment—*ECO*—and an exponential experiment—*EXPO*—have been set up, the former under contract and the latter by the Ispra establishment. *EXPO* is a less ambitious but more flexible installation than *ECO*, operating in sub-critical conditions and yielding results of less precision than *ECO* will be able to supply. Nevertheless, *EXPO* alone has already furnished initial data on the reactivity of uranium carbide/organic liquid/heavy water lattices—which proved incidentally to have a greater reactivity than we expected.
The *ESSOR* complex (*ESSai ORgel*) includes two hot laboratories in addition to the reactor which, of course, is intended for testing the whole setup comprising channel, joints and fuel elements under *ORGEL* power-reactor conditions. One hot laboratory is to be used for fuel element and the other for pressure tube inspection. The civil engineering for the internal structures of both reactor block and pressure-tube laboratory is finished, the leaktight metal containment will be completed during the summer, and construction of the fuel-element is under way. The entire civil engineering works should be finished by the end of 1965.
All the supply contracts have been placed and the first deliveries are arriving at the Ispra site. The reactor is scheduled for criticality in 1966 and normal operation in 1967.

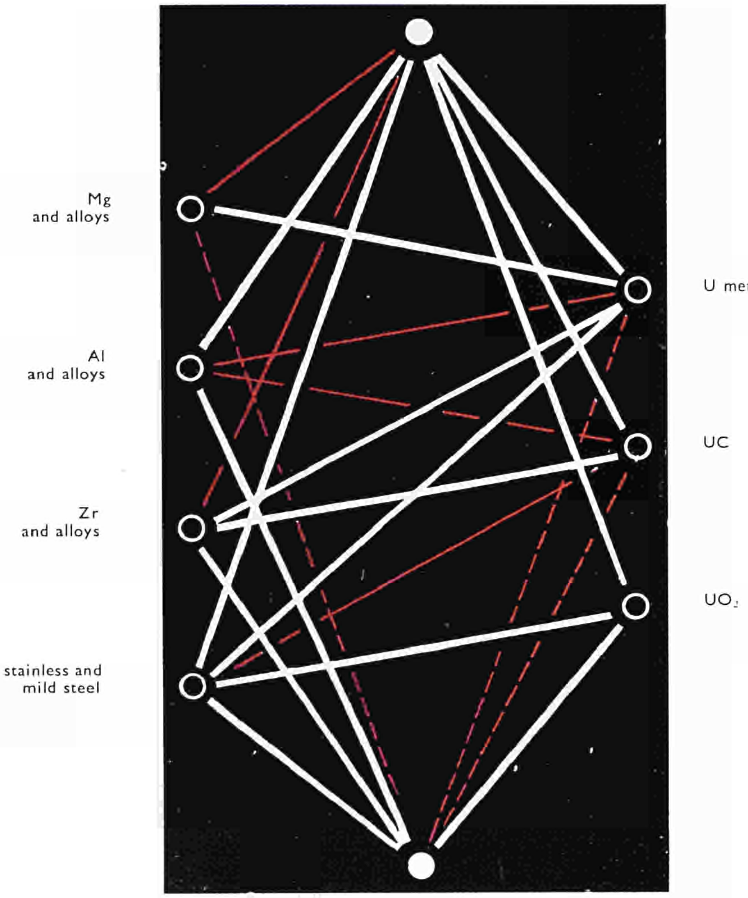
Table I
Production cost of electric power supplied by an ORGEL *power plant (in mills/kWh).*

Installed power	100 MWe	250 MWe	500 MWe
ORGEL natural U	—	5.5 to 6	5 to 5.5
ORGEL slightly enriched U	6.5	5	4.5

The outlook

Construction of an *ORGEL* reactor is already a practical prospect. A study carried out under contract by a group of industrial firms in 1962 did, in fact, show that no insuperable problems would be encountered in building a 250 MWe power plant, and recent developments in the research programme have confirmed this view.
The same contract had also provided confirmation that investments in such a plant would be moderate (160 EMA u.a. per KWe installed, excluding indirect charges but including site and heavy water).
But the attraction of the concept lies in the very low fuel-cycle cost, allied to the simple "throw-away" cycle, i.e. without reprocessing to recover the residual fissile materials.
It is already safe to estimate a figure of 1 mill/KWh for a full-scale reactor, the burn-ups that can be expected with natural uranium reaching 8,000-10,000 MWd/t. With slight enrichment (1% absolute, as compared with 0.7% for natural uranium), which would not entail reprocessing or affect either the fissile material economy or plutonium production, it would be possible to bring costs down to 0.6 mills/kWh by boosting the burn-up to 20,000 MWd/t.
If we consider the cost of fossil fuel in the Community (3.1 to 5.2 mills/kWh for fuel-oil and around 3 mills/kWh for lignite), it is obvious that, in view of the "minimum marginal cost" principle applied by the operator, the *ORGEL*-type plant is highly suitable, even throughout practically the entire reactor life, for base load operation, irrespective of any extra supplies which may be made to the overall power grid

organic coolant
(250/400°C)



heavy water
moderator
(70°C)

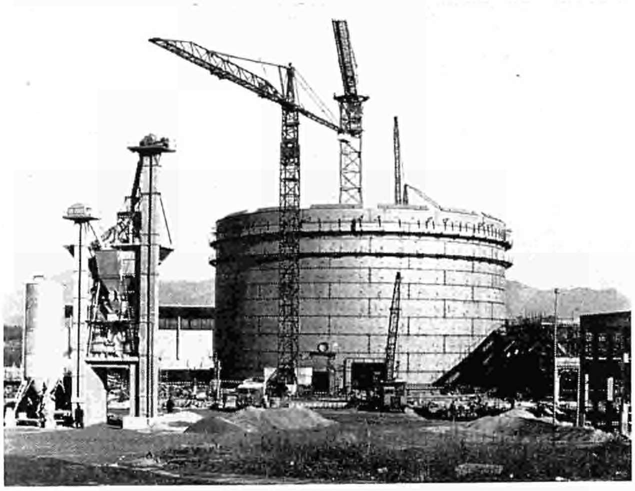
Figure 2: Flexibility of the ORGEL *concept*
Among the possible reactor components there are a number of compatibility relationships.

— *incompatibility*

— *fair compatibility, improvement under study or already achieved (platings, barriers, inhibitors, etc.)*

— *excellent compatibility*

by conventional or light-water nuclear plants.
The total production cost of power supplied by an *ORGEL* power plant are given in Table I. Capacities above 500 MWe are not shown as we have no sufficiently accurate data on prices; it is certain, however, that the cost continues to drop as capacity rises, and the higher capacities are particularly attractive because they involve no serious technological difficulties with this type of reactor. For example, *ORGEL* has no pressure vessel, so that there are none of the technological problems that arise where the vessel walls have to be proportionately thickened as the size is increased.
It was on such grounds that the *United States Atomic Energy Commission*, at the meeting of the *Atomic Industrial Forum* in December 1964, announced a ten-year research and development programme and the construction of three power reactors basically of the same type as ORGEL, the first of which will produce electricity only and will be commissioned towards 1970. This should be followed up by the building of giant reactors, particularly for water desalination projects.
At the same time the Euratom Commission, pursuing the aims set out in 1960, has just taken a decision in principle to ask Community industry to undertake a detailed design study on an *ORGEL* prototype. The construction of this prototype will then be the final step needed to reach the target —the commercial power reactor.



The ESSOR reactor (ESSai ORgel=ORGEL test) under construction at the Euratom Joint Research Centre's establishment at Ispra.

EUBU 4-8

A natural-uranium boiling-water reactor project

ABRAHAM BAHBOUT, *Directorate-General for Research and Training, ORGEL Project Directorate—JRC, Ispra*

Boiling water and natural uranium—the combination may appear paradoxical, since, as we know only too well, the facility afforded by the use of light water as a reactor coolant is offset by the need to employ enriched uranium as fuel. Nevertheless, a power reactor of the *CIRENE* (*CIse REattore a NEbbia—CISE fog-cooled reactor*) type is designed simultaneously to generate light-water steam in the core, this steam being conveyed by direct cycle to the turbine, and to consume natural uranium. How is this possible? In the first place because the reactor is moderated with heavy water, which has the important asset of enabling a good neutron economy to be obtained, and secondly because there is in the core only a limited amount of light water, a material which, on account of the hydrogen it contains, impairs the neutron economy even when employed in conjunction with the heavy variety. This reduction is achieved by maintaining in the core a "fog", or more precisely a low density water/steam mixture, since, in view of the considerable disparity between the densities of these two phases, the greater part of the volume occupied by the coolant consists of steam as soon as it accounts for 10% to 15% of the weight of the mixture.

Genesis of the project

The *CIRENE* project is being carried out by the *CISE*[1] in Milan, with technical assistance and close co-operation from the *Italian Atomic Energy Commission (CNEN)* and Euratom. The project is being financed jointly by Euratom and the *CNEN*.

As far back as 1958, the *CISE* was conducting research into the cooling of reactors by water/steam mixtures. Then, following the conclusion of an agreement in October 1959 between Euratom and the *United states Atomic Energy Commission* for the Study of wet-steam-cooled light-water-moderated reactors, a first research contract was signed with the *CISE/Ansaldo/NDA* group.

The results derived from this contract were such that two other contracts were subsequently concluded with the *CISE*, the aim being to take the information already acquired in the heat-transfer and corrosion fields a stage further.

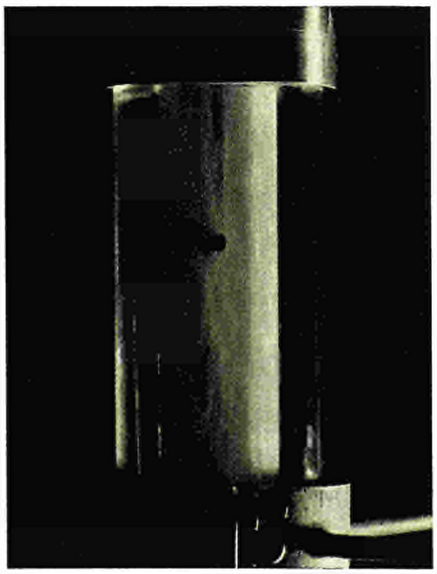
However, the design studies showed that the benefits of cooling by low-density water/steam mixtures had the greatest effect, not in light-water-moderated reactors, but in reactors with a good neutron economy, such as those moderated by heavy water and graphite.

Preliminary optimisation studies geared to these two alternatives had the effect of focussing interest on heavy-water moderation. This led to the signing in July 1963 of a first contract, *CIRENE-1*, with the *CISE*, which was followed up by contracts *CIRENE-2* and *CIRENE-3*, the chief aims of which were:

— to conduct a research and development programme on a pressure-tube heavy-water reactor, cooled by a low-density water/steam mixture;

— to study a *CIRENE*-type high-power reference generating plant and to assess the economic aspects of energy production from such a plant;

— to draw up a draft design for a prototype reactor.



Tests for visual observation of dispersed-phase flow, by analogy, in the hydrodynamic loop of the CISE *at Milan. The photograph shows the effect of an obstacle on the flow of the water film.*

Other projects similar to CIRENE

The *CIRENE* project has therefore got beyond the conceptual stage. It is not an isolated operation, either, since the development of a reactor with similar characteristics constitutes the principal short-term objective of the Canadian nuclear programme, the intention being that such

1. The *CISE (Centro Informazioni Studi Esperienze)* is a limited liability company, incorporated in 1946 and controlled almost entirely by the *ENEL*. In addition to its activities under the *CIRENE* project, the *CISE* carries out research into semi-conductors, lasers, metal physics, etc.

CIRENE

a reactor should become the successor of the *CANDU*-type pressurised-heavy-water-cooled design. There is already some talk of converting the *NPD-2* reactor to boiling-light-water cooling. Furthermore, *Atomic Energy of Canada Limited (AECL)* has launched a study of a 500 MWe reference reactor of this type and has recently decided to embark upon the construction of a 250 MWe prototype reactor.
The *UKAEA*, for its part, has since the beginning of 1963 been engaged in the construction at Winfrith of a 100 MWe prototype *SGHWR (Steam Generating Heavy Water Reactor)* which will however run on slightly enriched fuel (we shall see why later).
Quite recently, Japan too opted for boiling-light-water cooling for the development of its pressure-tube heavy-water reactor.
Finally, the first 100 MWe prototype reactor of the power plant at Beloyarsk (USSR), which is moderated by graphite but cooled by a light-water fog, started regular supply to the grid, in April 1964.

CIRENE operating principle

Figure 1 represents a simplified flow diagram for a *CIRENE*-type power reactor. The coolant enters the reactor in the form of slightly under-saturated water. The water circulates in vertical channels and vaporises as it takes off the energy generated in the fuel. The outlet steam content is relatively high. The water/steam mixture is then conveyed to a separator, where the two phases are separated. The water is tapped off and then recycled in the reactor, and the steam is channelled directly to the turbine.

What makes the CIRENE concept attractive?

The value of the *CIRENE* concept lies not only in its intrinsic assets from the technical and economic angle, but also, and particularly, in the benefits deriving from the fact that the concept is based largely on experience acquired with proven-type reactors already in existence.

Technical and economic advantages

These stem from the fact that the *CIRENE* concept combines characteristics peculiar to natural-uranium pressure-tube heavy-water reactors, on the one hand, and boiling-water-type reactors, on the other hand, without displaying any of their drawbacks.
Heavy-water moderation ensures a good neutron economy, which makes it possible to utilise natural uranium fuel and to boost the burnup to comparatively high values (around 10,000 MWd/t), even without enrichment. The use of natural uranium not only makes for "nuclear independence" in the matter of fuel supply but also eliminates spent fuel reprocessing.
Since the Seaborg Report to President Kennedy was published, however, heavy-water moderation has been found to possess other attractions. In point of fact, the emphasis has merely shifted to another aspect, namely that heavy-water reactors are the most effective at conserving natural-uranium resources[2] and, for the same natural-uranium consumption, produce the most plutonium.
But this is taking a very long view of things. For producers and consumers of electrical energy in a free-market economy, the cost of the energy produced is the main criterion in weighing up the advantages of particular reactor types and drawing comparisons between them.[3]
The design and development of pressure-tube heavy-water reactors was not motivated solely by the desire to make things easier for the next generation by conserving natural nuclear-fuel resources. The primary aim is to maintain their foothold in a competitive energy market. In this struggle, there is one definite item on the credit side i.e. the fuel-cycle cost is appreciably lower than for light-water reactors, on the grounds already stated—unenriched fuel, no reprocessing in the fuel cycle and relatively high burnup.
A *CIRENE*-type reactor is equipped with pressure-tubes, that is to say the temperature and the pressure of the coolant are contained entirely by the channels in which the energy is generated. The moderator is kept in a cold state and at virtually atmospheric pressure.
The channels may be thought of as so many individual reactors linked neutronically by the moderator. The mechanical and thermal problems are reduced to the dimensions of the channel, the basic unit.
One of the salient features of this setup is that it makes for much more flexible fuel management. Access may be had to each individual channel, and both ends of the channel may be used, thus affording a choice between various loading/unloading machine designs. The reactor may be fuelled

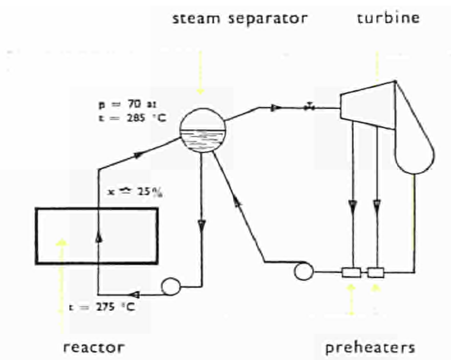
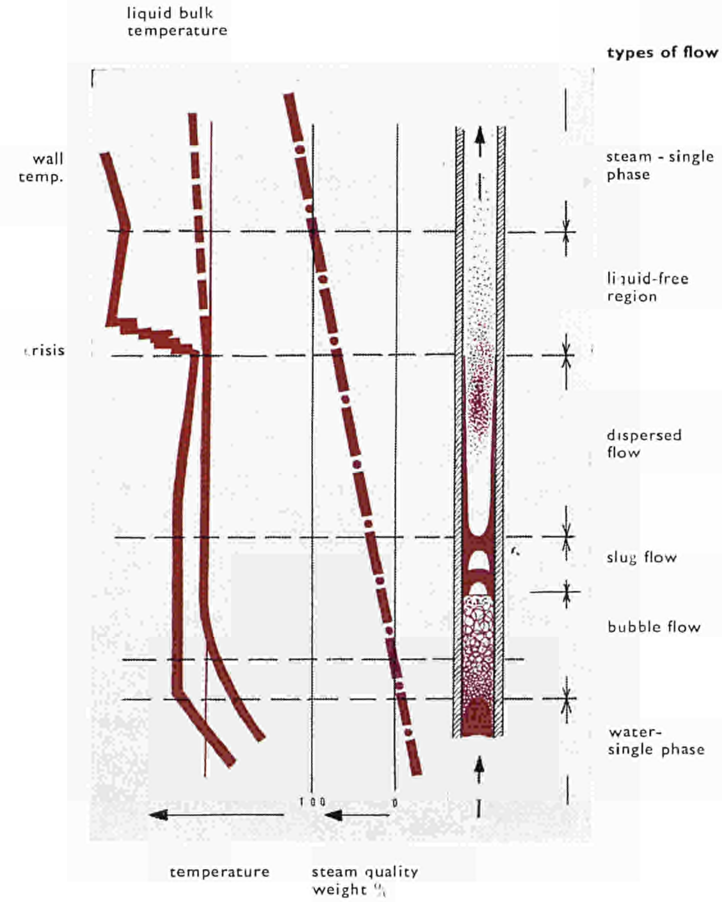


Fig. 1: Flow scheme of a CIRENE-*type power reactor*

2. The lower "yield" of enriched-uranium reactors is due to the fact that the enrichment process in an isotope-separation plant involves a certain wastage in the form of depleted uranium, which is unusable but contains a fraction of uranium-235.
3. This was recently evidenced by the British *CEGB*'s decision to invite tenders for US-type power plants to be constructed in the United Kingdom under a second nuclear energy programme.

not only when in operation but also continuously.
Continuous fuelling is obtained by replenishing at regular intervals a fraction or the whole of the fuel content of each channel. It obviates the need for initial storage of large reactivity excesses to compensate fuel depletion, which eases the task of the control system, cuts down flux distortions and enhances reactor safety.
Moreover, the substantial volume of heavy water required for moderation makes it possible to install between the channels tubes which may be emptied or flooded with heavy water as required, in order to check the degree of lattice moderation.
This check is most useful after the reactor has received a complete fresh fuel charge. It is then possible to reduce moderation, and consequently raise the mean energy of the neutrons, thus increasing the probability of their capture in the uranium-238. This procedure has the advantage not only of absorbing reactivity excesses but also of making use of them, since the uranium-238 nucleus is converted into plutonium after capturing one neutron.
Furthermore, pressure-tube heavy-water reactors offer greater safety than pressure-vessel reactors by reason of the fact that the failure of a pressure-tube is less serious than a vessel rupture.
Finally, we come to the advantages of boiling-light-water cooling.

Let us briefly recapitulate them:
— Light water is a well-known and very cheap fluid;
— Leakages of light water or light-water/steam from the coolant do not raise any recovery problems, as they do in *CANDU*-type reactors, or any contamination problem due to tritium formation;
— As the coolant also serves to drive the turbines, the elimination of intermediate exchangers means substantial savings in capital costs;
— The thermodynamic cycle being a direct one, there is no degradation of heat in the exchangers and the overall efficiency is thereby improved;
— For an equal thermodynamic efficiency, the reactor pressure is appreciably lower than that in an indirect-cycle pressurised-water reactor. This affords savings in absorbent structural materials in the core;
— Lastly, in the event of a severe steam circuit failure, the pressure surge in the



What is fog cooling?

Imagine a vertical tube whose walls are subjected to a constant thermal flux and which is fed by an upward flow of sub-saturated water. As it passes through the tube from end to end, the coolant undergoes various flow regimes as its enthalpy or quality increases. At the tube inlet, so long as the quality is negative (sub-saturated water) and the tube-wall temperature remains below saturation temperature, heat transfer takes place by forced convection of the water, which is the only phase present.
As soon as the wall temperature rises slightly above saturation temperature, the quality being still negative, the wall heat is removed by sub-saturated or surface boiling: the steam bubbles forming on the heating surface then condense partially in the still sub-saturated bulk of circulating water.
When the water reaches saturation temperature (nil quality) the steam bubbles can no longer condense. We then have what is known as *bubble flow*—discrete bubbles of steam circulating in a moving liquid continuum. It is this type of biphase flow that we find in the *BWR* reactors. It covers a relatively narrow range of steam qualities, with an upper limit of the order of a few percent.
Higher in the tube, when the bubble volume becomes appreciable, i.e. when the quality has reached a certain value, the flow system alters—the bubbles coalesce forming steam "slugs" which assume the shape of the duct. At the point where these conditions are set up, we find alternating steam slugs and water slugs—the latter, however, also contain steam bubbles. This flow system, which is unstable, is called *slug flow*.
A little higher in the tube, above a certain quality value, when the void flow rate appreciably exceeds that of the liquid volume, the *annular or dispersed flow* regime is reached: the liquid phase covers the hot wall in the form of a thin rising film, and the rest of the tube in this section is full of steam. The two phases are thus separate. Actually the steam also contains water droplets in suspension, so that it looks like fog.
As the water-steam mixture continues to rise in the tube and the quality increases, the thickness of the film on the wall decreases by evaporation and entrainment. The water droplets entrained by the steam gradually evaporate. After a certain

point the film on the wall vanishes altogether, a liquid-free zone is reached and a so-called "crisis" is reached in the heat transfer process (see below).
Higher still, all the water droplets suspended in the steam disappear completely, and the quality of the coolant has reached the value of unity.
From this point onwards, the heat would serve to superheat the steam.
Actually the bubble and slug flow take up only a small fraction of the length of heating duct while the length occupied by annular flow (fog) is in direct proportion to the outlet quality.

Slow and fast boiling crises (burn-out)

Since with this dispersed flow system heat transfer is not effected by bubble emission from the heating surface, presumably the *burn-out* phenomenon, the bugbear of *BWR* reactors, can be ruled out. For *burn-out* occurs when the bubbles formed on the wall of the heating surface are not carried away fast enough; an insulating layer of steam then forms on the wall, giving rise to a temperature surge which in turn may lead to the melting of the heating material and so to destruction of the surface. This fast crisis, or *burn-out*, is typical of nucleate boiling.
With dispersed flow, as we have said, the water film covering the heating wall gradually moves along the channel, in the same direction as the flow, and vanishes as a result of evaporation and entrainment. After thinning considerably, to virtually nil thickness, the film breaks, splitting up into little streams of water running round dry patches. The overall wall temperature, measured at a given point of the surface, begins to oscillate. These temperature oscillations (or "noise"), grow in amplitude as the number of dry patches increases, then gradually diminish as the film disappears. The wall heat is then carried off entirely by forced steam convection. By definition, crisis is reached when the temperature oscillations, or noise, exceed a certain "threshold" amplitude. *Burn-out* conditions cannot be said to obtain, as the amplitude is not enough for the heating surface to be destroyed. Two main factors account for this. First, the heat flux at the critical point is relatively low, as compared with the critical flux obtained in nucleate boiling, which has a low steam quality. Secondly, with dispersed flow, the steam throughput is high, which means relatively high steam velocities and hence relatively high convection heat-exchange coefficients. The deterioration in heat transfer is slower and less significant than in a nucleate boiling *burn-out*. Consequently the temperature oscillations which accompany the crisis can be withstood by a fuel clad without damage.
This finding, which has now been thoroughly corroborated by experiment, opens up the possibility of a "once-through" heating method. If the fuel in a reactor channel can stand up to the temperature oscillations specific to the transition from dispersed flow to forced steam convection—and a fuel such as Zircaloy-clad uranium oxide looks like having this capability—the fear of burn-out will no longer inhibit the steam quality value at the channel outlet. The channel is then said to be operating in "ultra-crisis" conditions.

reactor building is smaller than in a liquid coolant reactor, the total energy stored in the coolant being relatively small on account of the low density of the steam phase.

Reaping the benefits of past research

We have already underlined the fact that the basic component of the reactor is the channel. It is accordingly channel development which gives rise to most of the new problems to be tackled.
Now assuming that the fuel used is uranium oxide in a cluster array, it will be seen that the majority of the problems have already been solved during the development of other reactors. For example, the constituent materials—oxide fuel and Zircaloy-2 for cladding, pressure and calandria tubing, have now been thoroughly investigated and are able to do the job expected of them. There are virtually no materials problems, no metallurgical "gambles". In addition, the oxide/Zircaloy-2/boiling water combination has been adequately tested from the compatibility angle.
Consequently, the research and development programme involved boils down to the adaptation of the existing elements to the dimensions and conditions specific to the *CIRENE* project.

Long-term prospects

By equating a *CIRENE*-type reactor to a boiler, it is easy to visualise ways and means of improving the concept, at least as far as thermodynamic efficiency is concerned. The first step would be to attempt to reduce the water content in the reactor by increasing the outlet quality. Taken to the logical conclusion, this would lead to a "once-through" reactor, in which the liquid coolant is fed at the channel inlet to emerge at the other end as superheated steam.
A second possibility is nuclear superheating, independent of "once-through" heating, in which the steam produced in the boiling channels is superheated, after separation, in superheat channels. The first Beloyarsk power plant reactor is operating along these lines and the *SGHWR* prototype is to include 8 experimental superheat channels out of a total of approximately 200.
Finally, with superheating, the steam-pres-

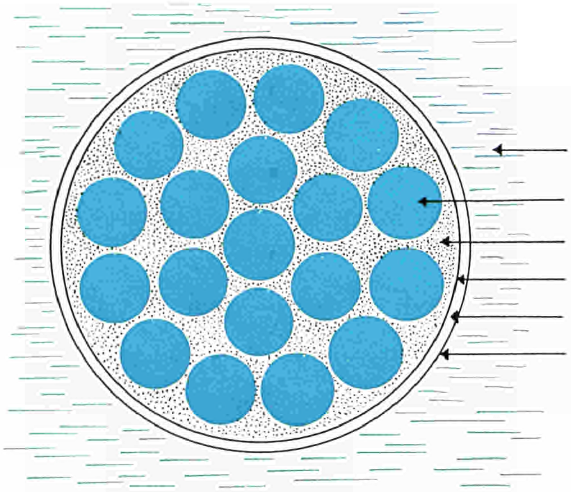


Fig. 2: Uranium-oxide version of a CIRENE*-channel (horizontal cross-section)*

sure increase becomes a paying proposition. In the extreme case, we may imagine a hypercritical-pressure and once-through-heating reactor.

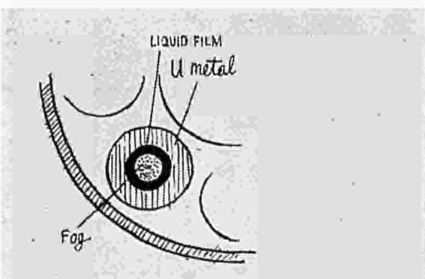
Among these three possible lines the stepping-up of the coolant's steam content to a value slightly below saturation appears to be the most feasible in the medium if not in the short term. Nuclear superheating, whether in superheat channels or in once-through channels, raises the problem of the development of a corrosion-resistant cladding material other than Zircaloy.

Specific problems of the CIRENE project

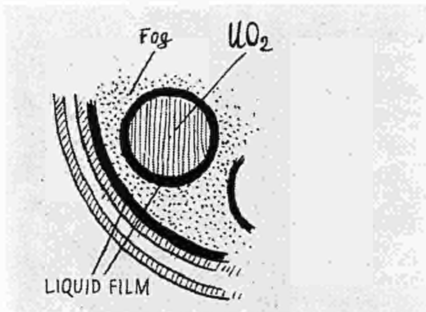
Choice of fuel

At first sight the most suitable fuel for a *CIRENE*-type reactor is Zircaloy-clad uranium oxide in the form of fuel pins in a cluster array, (see fig. 2), which was thoroughly tested out during the development of various water reactors, in particular the *CANDU* reactor.

Nevertheless, the dispersed-phase flow of water/steam mixtures lends itself best to tubular channels. Indeed, in the latter case the heating surface is the inner wall of the tube, which is the only wall wetted by the liquid film of the mixture as it flows past:



in the case of an oxide cluster, on the other hand, the liquid film which wets the inner surface of the pressure tube does not contribute to the heat transfer:

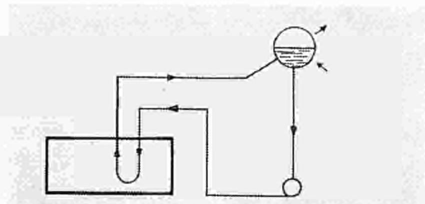


in other words, in a tubular cross-section less water is required for the removal of a given quantity of energy than in a cluster cross-section. As will be seen later on, this coolant density plays an essential part. It was for these reasons that a study was also carried out to explore the possibility of using tubular uranium-metal fuel elements with internal cladding and cooling.

In the uranium-metal version the internal clad serves also as pressure tube. This design is identical with that of the fuel elements at the first Beloyarsk power plant. In this way the quantity of "idle" construction materials present in the core is reduced. This, coupled with the fact that uranium metal has a high density, results in higher core neutron efficiencies than are obtainable with the oxide version.

The fuel tubes can be fabricated by co-extruding the uranium with the cladding. A few prototypes of such fuel elements, the fabrication cost of which is low ($ 30 per kilogramme), have already been produced for the *CIRENE* project by the American firm *Nuclear Metals*. The uranium tubes are arranged in the form of a crown around a graphite core the function of which is to reduce the flux depression in the bundle. The bundle is surrounded by a jacket of Zircaloy-2 which isolates it from the heavy water moderator. The space between the various tubes is filled with a stationary inert gas so as to cut down heat leakage to the moderator (see fig. 3).

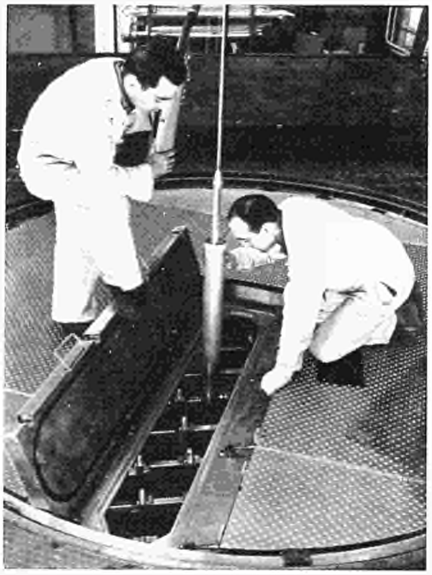
The uranium-metal version currently under study assumes a "hair-pin" type coolant-flow in the fuel element similar to that adopted for the first Beloyarsk power plant:



the coolant enters the top of the reactor in the form of slightly undersaturated water. It circulates from top to bottom in a central Zircaloy tube which passes through the graphite core. At the bottom of the channel the water is introduced into the peripheral heating tubes in which it circulates from bottom to top while gradually changing phase. The coolant likewise leaves via the top of the channel. The entire channel may thus be likened to a dipper, which is immersed in the heavy water and which can be completely withdrawn from above.

The neutron economy of this kind of lattice was checked experimentally in a series of critical tests carried out in the *Aquilon-II* reactor at Saclay early in 1964.

There are, however, a number of technical uncertainties which argue against the use of metallic uranium.



A tubular-type CIRENE *fuel element being loaded into the* Aquilon-II *reactor at Saclay.*

For, in the first place, there has as yet been no systematic study of the irradiation stability of uranium metal tubes, although a few tubes in the Savannah River *HWCTR* pile exposed to burnups of over 6,000 or 7,000 MWd/t have yielded extremely encouraging results. It is therefore planned to carry out some *CIRENE* tube irradiations in the near future, to confirm these results. Secondly, there are a number of problems connected with the compatibility of uranium metal with the hot coolant, a subject on which little is known. An experimental programme has been started at *CISE* to clear up the uncertainties in this area.
Under such conditions, we may conclude that one proven fuel element for the *CIRENE* type of reactor is already available, and it is always possible to turn to this solution (uranium oxide in clusters) if the uranium metal version proves impracticable.
Lastly, mention should be made of the large-scale loop which is to be fitted in the *ESSOR* reactor at Ispra, for testing *CIRENE* prototype power channels and fuel elements.

General reactor design

To obtain dispersed flow throughout the full length of the channels, the coolant would have to enter the core in the form of a steam-water mixture. The only advantage of this over a slightly sub-saturated form would be a slight reduction in the average coolant density in the channels.

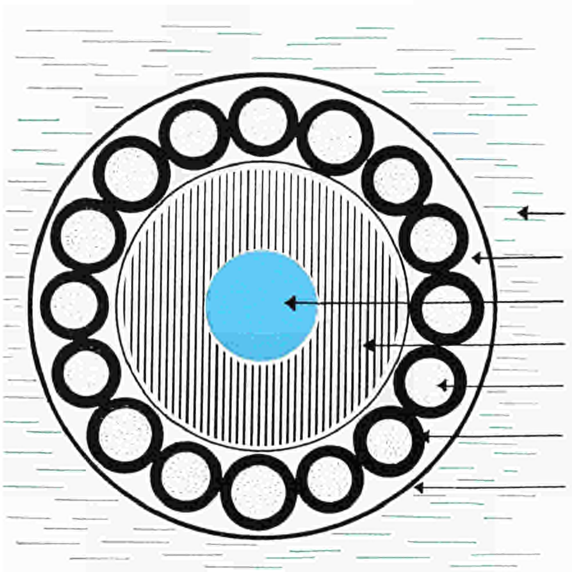
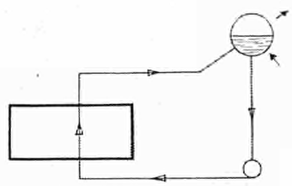


Fig. 3: Uranium-metal version of a CIRENE *channel (horizontal cross-section)*

But there are numerous drawbacks—it would entail recirculating hot, highly pressurised steam drawn off from the separator, mixing the two phases at the channel inlets, etc.
Consequently for *CIRENE*, as also for the Beloyarsk reactor, *SGHWR* and the Canadian project, it was decided that the coolant should be sub-saturated at the inlet. Thus a small portion of the full channel length, nearest to the inlet, operates under nucleate boiling conditions, whilst the remainder is under dispersed flow. This fact, and purism, are the reasons why the Canadians do not use the word "fog" for the cooling system proposed in their studies, even though they are envisaging extremely high channel-outlet steam qualities, so that the dispersed regime occupies the greater part of the channels.
The channel light-water content is of major importance. On the one hand, light water has such a voracious appetite for neutrons that it has a ruinous effect on the neutron balance of a non-enriched uranium lattice. Yet to avoid burnout the clads must be covered all along the channel with a film of water to protect them from temperature surges.
The search for a compromise between these conflicting requirements narrows down, for a number of reasons, to the choice of an optimum channel length, which in fact lies at around 3.50 metres for a 500 MWe reactor.
The channel outlet steam quality varies from 20 to 30%. Thus the coolant density ranges from 0.75 g/cm³ at the inlet (saturated water), for instance, and 0.20 g/cm³ at the outlet.
As already stated, in the uranium metal version the coolant enters and leaves the reactor at the top. In the oxide version, the coolant enters at the bottom and leaves by the top of the reactor:



The specific power, expressed in W/cm³ of fuel, is practically the same in both versions, varying around 160 W/cm³.

Dynamics and stability

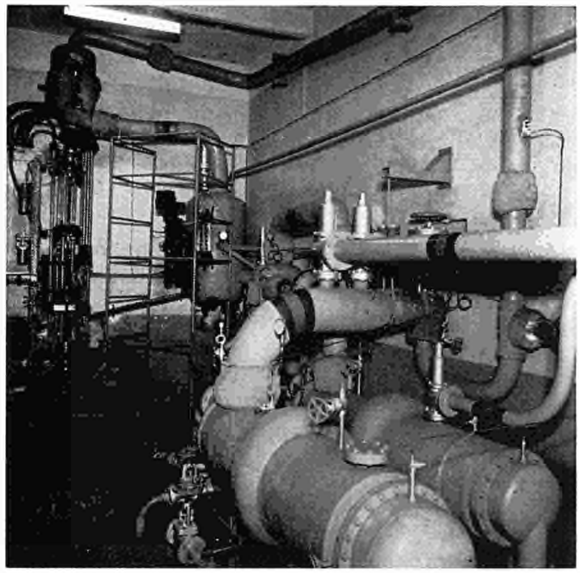
The novelty and main snag of a *CIRENE*-type reactor, as compared with the *SGHWR*, is the need to accept a positive void coefficient.

In order that the reactor may run on natural uranium, the quantity of light water present in the core must be kept very low. Consequently the lattice continues to be moderated chiefly by heavy water. A reduction of the coolant density means lower absorption in the light water—resulting in increased reactivity—and also by lower moderation by the hydrogen contained in the water—causing a loss of reactivity. The former effect prevails over the latter owing to the small quantity of light water present, and this causes the reactivity to rise in direct proportion to the volume occupied by the steam; in other words, the reactor has a positive void coefficient. In the case of the *SGHWR*, on the other hand, a negative void coefficient is obtained by narrowing the lattice pitch and enriching the fuel, which means that the light water plays a greater part in lattice moderation. This positive void coefficient is the main component of the *CIRENE* reactor power coefficient. In fact, the counter-reaction due to the negative Doppler effect of the fuel as the latter heats up is a weak one and is insufficient to offset the positive effect of the void coefficient. The reactor therefore has a positive power coefficient and could be intrinsically unstable.

It must be remembered, however, that when the reactor power rises or falls there is a tendency for the pressure in the entire water/steam circuit, including the out-of-pile section, to follow suit. Consequently, in a reactor of the *CIRENE* type the pressure can be caused to vary in such a way that the coolant density remains constant and thus independent of the power level. But the corrective action of the pressure must not be too delayed; this condition can be fulfilled by reducing to a minimum the volume of the water/steam circuit.

The positive effect of the void coefficient can thus be neutralised and the overall power coefficient becomes zero or negative. In principle, moreover, this system offers the possibility of making the reactor power depend on the loading of the turbo-alternator unit; control of the reactor is then exercised solely via the opening of the team inlet valves to the turbine, no recourse being necessary to any other means of reactor control.

Loop for large-scale heat-transfer tests, installed at Genoa.



Safety

A serious fracture in the reactor steam circuit can give rise to a sudden drop in the pressure, and consequently the density, of the coolant, and this is reflected by a surge of reactivity in the reactor owing to the positive void coefficient.

This situation, at first sight a disastrous one, can, however, be remedied. In the first place, it is possible to limit the rate at which the water/steam mixtures escape through a piping fracture by increasing the number of lines linking the outlet-side headers of the reactor with the steam separator and by providing flow restrictors on the steam lines to the turbines. Secondly, the consequences of a leak can be mitigated by splitting the reactor up into a number of zones, the channels of each zone being connected to an independent external circuit. In short, the accident is cut down to the scale of a power excursion which can be limited by the rise in fuel temperature (through the Doppler effect) without damaging it.

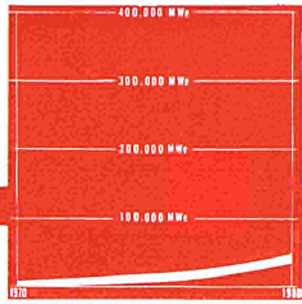
Finally, it should be borne in mind that the "external" consequences of a steam-line fracture (pressure rise in the reactor building) will be less severe than in the case of other reactors. On the one hand the total energy stored in the coolant is fairly low in comparison with pressurised- or boiling-water reactors, owing to the lower coolant density; on the other hand the division of the reactor into hydrodynamically independent zones means that in the event of a fracture only a small part of the energy stored in the coolant will be released.

Concluding remarks

At a time when "convergence" is considered essential and the trend is towards less diversification of reactor strings, it may seem surprising that countries such as Britain, Canada and Japan should be starting to develop reactors of the *CIRENE* type.

It will readily be appreciated, however, that this concept, which combines the advantages of heavy-water and pressure-tube designs with those of boiling-water reactors and opens up extremely promising prospects for the future, enjoys a privileged position in that its development calls for very little effort.

EURATOM NEWS



Nuclear energy from today until 2000 - forecasts for the European Community

According to Article 40 of the Treaty establishing Euratom, the Commission undertakes to publish periodical "programmes indicating, in particular, the production targets for nuclear energy and the various types of investment required for their attainment". The Commission recently drew up such a "target programme" relating to the period 1970-1979, with extensions up to the year 2000, and outlining the probable increase in electricity consumption, the percentage of these requirements to be satisfied by nuclear energy and the resultant repercussions on the various sectors of the nuclear industry with regard to both production costs and investments.
Basing its conclusions on the data contained in the target programme, the Commission also drew up a document entitled "Factors affecting industrial policy", which dealt with the structural and organisational problems raised by the transition of nuclear energy to the industrial stage and mapped out the course of action to be pursued.
The Commission will not be in a position to publish the final version of these documents until certain consultations have been held, but the main ideas behind the first of them, the "target programme" are sketched out here.

The first assumption of the programme is that electricity consumption will rise as follows during the period 1965-2000:

1960-65	*1965-70*	*1970-80*	*1980-2000*
(average annual increase during each period in %)			
7,5	7,0	6,5	6,0

The estimates for annual electricity consumption that follow are (285 milliard kWh in 1960):

(milliard kWh)			
1965	409	*1980*	1,080
1970	574	*1990*	1,930
1975	789	*2000*	3,450

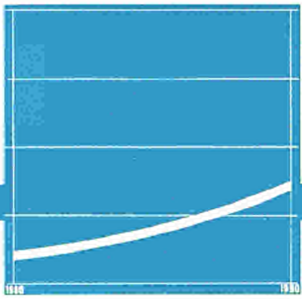
The rôle of nuclear energy in satisfying this demand is based, for the immediate future, on known projects and on programmes already announced. Over the longer term, it is estimated that between 1980 and 2000 two thirds of new power station capacity will be nuclear. This, however, is regarded as a minimal hypothesis; in view of the technical maturity likely to have been achieved by nuclear power and its economic advantage over conventional plants, for which fuel costs are expected to rise, it could well be that all new capacity installed after 1980 or 1990 will be nuclear. This would give a capacity of at least 500,000 MWe in 2000, to furnish around two-thirds of electricity, and a third of total energy, needs.
According to the hypothesis actually adopted one half of the Community's installed capacity in 2000 will be nuclear; slightly more than half the total electricity produced and around one quarter of the energy consumed would come from nuclear sources.

The following table sets out the nuclear power production "objectives" for the period 1970-2000:

	Total installed power on 1 January (1000 MWe)	*Load factor* (hours/year)
1970	3,7	
1975	12,0	6,000
1980	40,0	
1990	140,0	5,500
2000	370,0	5,000

Some observations on the above

— Energy imports, which have been rising from 5% pre-war to 27% in 1960, around 50% in 1965 (of which 80% from Arab countries in the Middle East and North Africa) should rise to between slightly over half and a little under two thirds in 1975, in spite of the (still modest) recourse to nuclear power. Even with the implementation of this "programme", however, imports will still furnish nearly 50% of energy requirements in 2000. Thus this growth in the recourse to nuclear power is indispensable if too great a dependence on external sources is to be avoided,
— furthermore, even if a large part of the fissile materials are imported, nuclear energy will nevertheless contribute to the security of supply owing to the *much lower* cost of nuclear, as opposed to classical, fuel sources, the cheaper and more favourable conditions of transport etc.
— furthermore, assuming a stable price of $ 13,50 per ton of coal equivalent for the fossil fuels used in most conventional power plants i.e. coal, fuel-oil and natural gas, during the period, and assuming that the nuclear power plants to be installed will produce 17,000 mrd kWh before 2000 and 34,000 mrd kWh after (i.e. altogether 140 times Community electricity production in 1964) it is estimated that savings in the cost of electricity production will be $ 93 mrd or 36% (in absolute values) and $ 23 mrd (or 30%) in real values.



The reactors to be installed

Four sets of hypotheses are put forward on the types of plant to be installed. These are purely tentative in that the evolution of reactor development and installation will certainly be much more complex. The set of assumptions selected foresee that nuclear power stations will be equipped with "proved" reactors (i.e. equal proportions graphite-moderated-gas-cooled and water-cooled) until 1975, that they will be complemented by the installation of advanced converters from 1975 and fast breeders from 1980. The installation of "proved" reactors will cease when they form one half of total capacity, in about 1990. By 2000 fast breeders will represent one half of the total, advanced converters and "proved" reactors by then taking respectively only 30 and 20% of the total. It is assumed that power reactors will be installed along the general lines shown in Table I.

Table I: Forecasts of installed capacity (in MWe) of the different reactor types from today until 2000.

	"Proved" Graphite-gas	"Proved" Water	Advanced converters	Fast breeders	Total
1970	2,000	1,500	200	—	3,700
1970-74	5,500	5,000	1,500	—	12,000
1975-79	17,000	17,000	5,000	1,000	40,000
1980-84	25,000	25,000	19,000	6,000	75,000
1985-89	35,000	35,000	51,000	19,000	140,000
1990-94	35,000	35,000	98,000	60,000	228,000
1995-99	35,000	35,000	115,000	185,000	370,000

This programme is selected since it conforms with the present industrial situation, with the state of R & D in the Community and with a harmonious evolution of techniques. According to it the total investment will be lowest, it requires the smallest quantities of fissile materials, whether enriched uranium or plutonium, and it is based on the progressive introduction of fast breeders, thus making the most rational use of the Community's limited resources, as well as those of the Western World as a whole.

Among the factors of uncertainty which could modify these forecasts are:
— the two types of "proved" reactors might not develop in parallel; this would have little impact on any of the succeeding forecasts other than that concerning enriched uranium consumption;
— enriched uranium needs could be reduced by the use of plutonium as fuel for water reactors from 1975;
— the introduction of the thorium/uranium-233 cycle is not taken into account owing to insufficient technical and economic data. Its use could help to solve supply problems; it is most likely to be first employed for high temperature reactors.

Outlook for reactor types to be installed

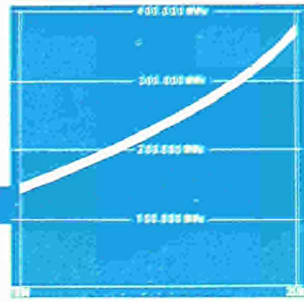
The report makes a number of observations on the reactors to be installed under the "programme" selected. Among them are:

"Proved" reactors

A decisive advantage cannot reasonably be attributed to one type or the other. The two "filières" have reached in the Community roughly the same point of industrial development, the economic prospects are similar, as is their consumption of fissile materials. It is therefore expedient to admit that the parity between the two types will be maintained approximately until the end of the period under consideration and this independently of the various hypotheses about the technico-economic success of more advanced types of reactors. Furthermore "at least for the first years of the programme, and having regard for the present equilibrium, it is certainly wise not to concentrate exclusively on one technique. Indeed, nuclear industry in the Community is equally interested in both". While the investment in enriched uranium reactors may be lower, gas reactors offer a greater autonomy of supply and permit the exploitation of techniques which from the beginning have been developed in the Community. Lastly, the choice of reactor could be affected by such political considerations as the need for independence from a monopolistic supply situation.

Advanced converters

Among the numerous types of advanced converters, *heavy water moderated* and *high temperature gas* reactors appear most worthy of consideration for the Community. Both types hold promise technically and economically and for both the use of thorium may be envisaged. Their excellent neutron economy should permit a lower specific consumption and a higher "in situ" use of the fissile materials produced, thus making for an optimal utilisation of the natural resources of uranium and thorium. In addition to these advantages which apply to both types: heavy water reactors alone



can function using natural uranium as fuel, their requirements being minimal; high temperature gas reactors hold great promise in the field of thermodynamic yield and breeding.

While the relative advantages of the two types appear roughly equal, it is impossible to estimate the shares the two will take in the "programme".

Fast breeder reactors

It is assumed that fast breeders will have been perfected towards 1980. A key factor regarding the pace at which they are introduced will be the availability of plutonium. The greatest uncertainty, however, lies in the rate of technological development, in particular when fast breeders will have become "proved" industrially and economically advantageous. It is assumed that the risk of delays in their development will be avoided by an intense and concerted effort.

Some implications for nuclear industry

Materials

It is estimated that this programme will require 43,000 tons of *natural uranium* during the period 1970-79 and 109,000 tons during the following decade, and altogether 281,000 tons by 2000. *Enriched uranium* needs are put at 5,000 tons during 1970-79, 12,000 tons 1980-89 and 12,000 tons during 1990-99. As known Community reserves of uranium suffice for only half the expected consumption to end-1979, *prospection* must be intensified. Community enterprises would also need to develop access to resources in third countries. As regards enriched uranium, *isotopic separation* capacity in the Western World appears to be sufficient for the period to 1980, but a Community action might thereafter become necessary in this field.

As regards *plutonium*, forecasts are limited to production in "proved" and advanced converter reactors: during 1970-79 34 tons, during 1980-89 177 tons and by 2000 altogether 557 tons. The quantities produced and required by breeders are difficult to estimate.

Estimates are, however, made of *retreatment* capacity for the recovery of plutonium: by the end of 1979 2,000 tons/year of natural uranium elements and 400 tons/year of enriched uranium elements. By the end of 1989, capacity will be 6,500 tons/year of natural uranium elements (including the fuel for the blankets of fast breeders) and 850 tons/year of enriched uranium elements (including the core of fast breeders).

Investment volume

The volume of investment required for nuclear power stations during the 1970s, the 1980s and the 1990s is put at \$ 7,17 and 37 milliard respectively, i.e. a total of about \$ 60,000 million.

The 17,000 milliard kWh produced during 1970-1999 would cost \$ 90 milliard to generate, the average cost per kWh unit being about 5 mills. This figure includes costs both of the installations and of the fuel.

The installation of nuclear power plants

In addition to the 4,000 or so MWe of *proved reactor* capacity in service, under construction or planned for end-1969, a dozen such plants of 500-1,000 MWe capacity would be brought into service during 1970-74 (7,000 MWe total) and 25 during 1975-79 (23,000 MWe). Two or three *advanced converters* of 200 or 300 MWe would come into service during 1970-74, one of which would be a prototype industrial *Orgel* reactor to come into service, if a decision were taken early enough, at the beginning of the period. Between 1975 and 1979, six further 500-1,000 MWe advanced converters could come into service.

A prototype fast breeder reactor (100 MWe) would have to be brought into service towards 1972 along with two full-scale power plants (500 MWe or more) towards 1979 for these reactors to achieve their "maturity" by the end of the decade.

Fuel element manufacture

By 1980 a total natural uranium fuel element capacity of 4,000 tons/year will be needed; for highly enriched uranium elements around 1,000 tons/year.

Irradiated fuel reprocessing

By end-1979 irradiated fuel to be reprocessed would have risen annually to 1,720 tons/year (uranium contained) for natural uranium elements (graphite-gas reactors), 230 tons for natural uranium elements (intermediate reactors) and 350 tons for slightly enriched uranium elements. In addition to Eurochemic (slightly enriched uranium) and Cap de la Hague (natural uranium elements), a new plant for slightly enriched uranium elements with a capacity of over 400 tons/year will need to enter into service once water reactor capacity has exceeded 4,000 MWe, i.e. towards 1974. Between 1980 and 1989 retreatment capacity would need to rise from 2,000 to 6,000 tons/year for natural uranium elements. Three 2,000 ton plants might have to be built during the period for the natural uranium elements and a 400 tons/year plant for enriched uranium elements will be built towards 1989. The fast breeder blanket and core elements will have to be reprocessed by the natural uranium and enriched uranium reprocessing plants respectively.

ORGEL—Safety studies

Since mid-1964 a rig for studying the effects of an explosion in an *ORGEL* reactor channel has been operating at Ispra.
It consists of a vessel with through-channels (a life-size reproduction of the *ESSOR* reactor vessel), a circuit for 4,000 litres of organic coolant which can be heated to over 400°C and pressurised up to 50 kg/cm², and a concrete bunker where the experimenters take shelter during explosion trials.
The experiments are effected by creating an artificial flaw in a pressure tube and raising the pressure of the hot organic fluid until the tube bursts.
Using the apparatus developed by the Ispra technology service for the measurement of fast phenomena, several experiments have already been carried out which have yielded information on the formation of shock waves in the moderator and on the resultant mechanical effects on structures.
Thus the programme as a whole will make a major contribution to research on the safety problems raised by explosion, with regard to the *ESSOR* reactor in particular and the *ORGEL* string in general.

A step towards nuclear superheat in water reactors

Corrosion has been said to be the arch-enemy of the water reactor. This is mainly because any corrosion products released into the water circuits are liable to become radioactive and to give rise to tricky decontamination problems. This factor and a few others, such as the need to avoid excessive neutron capture, have practically restricted the choice of materials to zirconium alloys and stainless steels, which, in view of their high cost, have a detrimental effect on the economics of water reactors. Zirconium alloys and stainless steels (at least in the case of pressurised water reactors) behave satisfactorily at the working temperatures, of the order of 300°C, which have been adopted for these reactors. However, at higher temperatures, which imply the use of superheated steam, corrosion problems are more serious and, in spite of the many research efforts which are devoted to their solution, progress in overcoming them is slow. And yet it is worth pursuing these efforts, since success would allow the design of water reactors equipped with nuclear superheaters, which could take advantage of the high efficiencies offered by modern turbo-alternator units, working at 600°C and over.
An interesting contribution has been made recently by the *Société d'Etudes, de Recherches et d'Applications pour l'Industrie (SERAI)* in Brussels under the United States/Euratom Joint Research Programme. The work of this organisation has been aimed at evaluating the influence of the surface treatment on corrosion in high temperature water and steam.
It was established that, when electropolished, chromium/nickel stainless steel specimens stand up better to corrosion in water than mechanically treated (milled) specimens. On the other hand, a spectacular reversal in the situation was recorded when similar specimens were tested in superheated steam at temperatures in the 400-600°C range. Not only did a mechanical surface treatment improve the behaviour of the specimens but the improvement increased with temperature. (The corrosion rate was 10 times less at 450°C; at 600°C more than 50 times less). Moreover tests carried out over long periods (over 2000 hours) showed the effect was a lasting one.
A satisfactory explanation has been found for this phenomenon. Tests have shown that, thanks to changes in the internal structure of the metal brought about near the surface by mechanical treatment, diffusion of chromium through the surface layer is possible. This is important; it means that the protective film of chromium-rich oxide which forms on the surface can be "fed" with more chromium and thus give better protection.

Section of a stainless-steel platelet which has been exposed to steam for 1,000 hours at 500°C. The central part of the photograph represents a zone scored by a diamond point and consequently cold-worked. It will be observed that the oxide layer in this zone is considerably thinner than on the rest of the platelet.

Production of uranium carbide single crystals at Ispra

A laboratory at the Ispra Research Establishment recently began producing uranium carbide (UC) single crystals. Some of these crystals have a volume of 1 to 2 cm³.
This was not merely a sort of sporting venture—as might be imagined, since uranium composed entirely of single crystals can scarcely be employed as a fuel. Rather was the idea to make available to the various laboratories in the Community quantities of uranium carbide which would be used as a reference material.
It is, of course, known that a single crystal has an almost perfect crystal lattice (no

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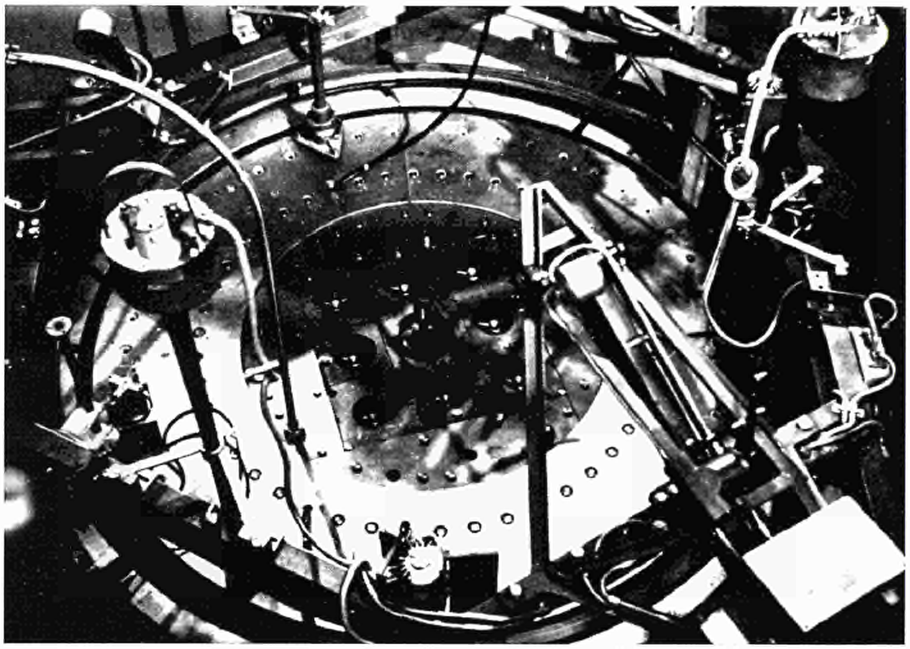
View of RB 1 reactor (Bologna, Italy) used for measuring the infinite neutron multiplication factor of various lattices. The picture shows a part of the lattice of the Latina gas-graphite reactor in the experimental position.



grain boundaries, well-defined orientation) and therefore lends itself more readily to the study of the structure and the basic properties than polycrystalline carbide.
Other Ispra laboratories, as well as other institutes in the Community, have been able, by means of the single crystals, to undertake a more accurate fundamental study of the physico-chemical properties of uranium carbide and also to conduct research into fission gas and fission product diffusion phenomena after in-pile irradiation.
The single crystals were developed by means of special equipment, the uranium-carbide rod being heated in a high-frequency vacuum furnace in such a way that only a limited internal zone is raised to melting temperature. The rod is thus its own crucible. Slow displacement (a few millimetres per hour) of this fused zone is accompanied by a process in which refinement and enlargement of the crystals takes place.

Uranium carbide single cristal

Higher burnups in graphite-gas reactors

On 22 February 1965, Euratom entered into a contract with the *Ente Nazionale per l'Energia Elettrica – Italia (ENEL).* The subject of this contract is the study of the reactivity curve in the Latina reactor, knowledge of which is most useful, as it enables the theoretical burnup peak to be determined beforehand.
This study is also of general interest as regards the development of graphite-gas reactors. Hitherto, the upper limit for the burnup of fuel elements has been estimated at an average of approximately 3,500 MWd/t. A study carried out under Euratom contract has shown that this falls far short of the mark and that burnups of the order of 5,500 MWd/t could be achieved in a reactor optimised for the purpose.
To illustrate what this means, fuel-replacement costs could be cut by 22% if burnup were increased from 3,500 to 4,500 MWd/t and by 36% if it were stepped up from 3,500 to 5,500 MWd/t. It is therefore vital to check the validity of the calculation methods currently employed. The measurements performed by the *ENEL* under this contract will make it possible to effect just such a check. They will be carried out at regular intervals on the Latina reactor when operating at full power.

The experimental reactivity curve obtained on the basis of these measurements will be compared with theoretical curves calculated both by the French and the British method. The results of this comparison will reveal the inconsistencies between the two curves and will give useful pointers regarding the adjustments to be made to the calculation methods.

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