XVII/341/2/71-E
Orig. F

COMMISSION
OF THE
EUROPEAN COMMUNITIES

Directorate-General for Energy
and Euratom Safeguards

XVII/D/1

SECOND ILLUSTRATIVE NUCLEAR PROGRAMME
FOR THE COMMUNITY

1 July 1972

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PREFACE

In this report and the documents annexed to it the costs are expressed in constant EMA units of account at their 1970 value. They thus remain unaffected by the depreciation of money and by the parity changes that have since occurred.

The following documents are annexed to this report:

1. Commission action in the radiation protection field.
2. Structure of electricity production
3. Light-water reactors
4. The fuel cycle
5. Fast reactors
6. High-temperature reactors.

INTRODUCTION

In 1966 the Commission of the European Atomic Energy Community published the "First Illustrative Programme for the EAEC" in accordance with the procedure laid down in the Treaty.

The Commission of the European Communities now considers the time to be ripe to publish a second illustrative programme in order to survey the development of nuclear technology and the context of energy in general. This programme, like its predecessor, has been compiled in implementation of Article 40 of the Treaty establishing the EAEC, which states that "in order to stimulate action by individuals and undertakings and to facilitate coordinated development of their investment in the nuclear field, the Commission shall periodically publish illustrative programmes indicating in particular nuclear energy production targets and all the types of investment required for their attainment".

The first Programme was prepared in a buyers' market atmosphere and when the prices of competing sources of energy were being held down to the very low levels which had ruled in the past. In addition, the economic assessments were based on combinations of nuclear reactor types some of which had still not proved their industrial maturity. The essential goal of the first Programme was thus to endeavour to secure conditions in which nuclear energy would become competitive and then come on to the market.

The backdrop to the present Illustrative Programme differs from that of its precursor on two basic counts. First, the preference of electricity producers in the Community has focussed on light-water reactors, and virtually all the orders for commercial power plants during the present decade will be for this type. This family has emerged as the most reliable and most economic, notably through the considerable boost it has been given in the United States, whereas development work on advanced-reactor families will have progressed little beyond the stage of demonstration plants during this period.

Secondly, the tension which characterized the petroleum market during 1970-71, despite the appearance of natural gas on the market in large quantities, has been reflected in a marked increase in the cost of crude oil supplies, thus giving nuclear energy an indisputable lead in the competition stakes. Above all, however, this tension has served to underline the vulnerability of the Community's oil supplies. It is for this reason that the Second Illustrative Programme has a greater political content than the First.

A massive swing to nuclear energy would help towards solving the problems inherent in the Community's lack of independence in matters relating to energy and it is regrettable that current achievements are lagging behind the targets set out in the First Programme.

Like its predecessor, the present Programme is devoted to the production of electricity by nuclear fission.

It would be premature to forecast a swing to nuclear energy for other purposes - to which it is technically adaptable as of now - such as marine propulsion, desalination of water and industrial heating, the needs of which can be expressed in units of power which lie below the break-even point for nuclear steam-raising plants.

This is even more the case with applications calling for further technological developments, such as the provision of the high temperatures required for the reduction of ore in the steel industry and the on-site gasification of fossil fuels for the mining industry. Thermonuclear fusion certainly constitutes a source of energy with exceptional, but longer-term, appeal. While various approaches to the problem have yielded encouraging results, not all the conditions necessary for the development of a reactor working on the fusion principle have been satisfied but, according to the bodies responsible for carrying out the research, they could be by 1980-85. However, before envisaging industrial application it will be necessary to tackle some extremely complex technical problems, which are unlikely to be solved before the end of the century.

The objectives of the Programme involve the installation of capacity and the creation of an infrastructure which together form the internal nuclear market which is to be open to the industry. It is clear that the companies in the Community will be all the better equipped to compete in the export market if they can capture the internal nuclear market.

Although it has been proved that nowadays the industry possesses sufficient knowhow to produce and market nuclear installations successfully, it still needs to be able, under conditions combining efficiency and competitiveness, to meet the demand, the growth of which, as evaluated in terms of electricity production, will be considerable. This knowhow is certainly susceptible of improvement in many ways and the industry will not fail to apply such improvements to proven reactor types, since commercial competition necessitates continuous updating of techniques.

However, the resources which the Community must call upon in order to achieve its nuclear objectives do not stem from technology alone; they also concern the organization of the industry itself and reduction of the obstacles inherent in nuclear energy and the barriers of all kinds which are responsible for the persistent partitioning of the Community market.

1985 has been set as the deadline for the Programme.

This leaves sufficient time for the formulation of the guidelines to be set as part of an overall energy policy, of which the programme forms a basic element. It also accords with the special characteristics of the nuclear sector, which requires relatively protracted deadlines and a programme of sufficiently long duration for the scope of the projects recommended in it to be duly appraised.

Furthermore, the completion of the Illustrative Programme in 1985 roughly coincides with the coming of age of the technological variants currently being developed.

The Illustrative Programme also envisages advanced-type reactors, the advantages of which as regards the utilization of resources and thermodynamic efficiency could help to solve the energy problems which will arise in the Community beyond 1985, but on which major decisions affecting a more distant future must in any case be taken well before that time.

For this reason it seemed wise to extend the period covered by the Illustrative Programme proper to the year 2000 in order to outline potential trends in the nuclear market, where the various families of reactors can be developed.

The Second Illustrative Programme was compiled for the six-nation European Community in 1970-71.

The enlargement of the Community following the accession of Denmark, Britain, Ireland and Norway, should not, in the medium term involve a change in the objectives concerning production of nuclear electricity in the present Community.

On the other hand, from the point of view of the resources to be deployed in the promotion of a nuclear policy, enlargement will bring about far-reaching changes, stemming mainly from the remarkable technological and industrial potential of the United Kingdom.

With regard to the long-term prospects for advanced-reactor families and, later on, for thermonuclear fusion, harmonization of development programmes should be facilitated by the similarity between the projects under investigation.

It would therefore appear that the Third Illustrative Nuclear Programme which will be drawn up for the enlarged Community, should not challenge the principles underlying the recommendations made in the present report.

PART ONE
THE ILLUSTRATIVE PROGRAMME

SECTION I: Nuclear energy in the Community energy context

The requirements - or, if preferred, the criteria - which the Community energy policy is aiming to satisfy concern the provision of resources which are reliable, adequate, cheap and non-pollutant.

Nuclear energy can fulfil the major criteria, i.e., security of energy supply, and is also advantageous from an economic and environmental standpoint.

1. Supply

For the six countries forming the European Community the period since the end of World War II has been one of sustained economic expansion, accompanied by a rapid upsurge in the overall demand for energy. Demand for fuel rose from 461 to 784 million tons of coal equivalent between 1960 and 1970, a rate of about 5% a year.

Apart from its high rate of growth, energy consumption has also been marked by a radical change in structure. Where it had formerly relied for the most part on indigenous coal, the Community quickly turned to petroleum, which is today its main source of energy.

For a continually expanding proportion of its supplies it has had to have recourse to imports - a proportion which in the space of twenty years has increased from 10% to about two thirds.

The combined effect of the inadequacy of the Community's own resources and the continued expansion of its needs can only increase its dependence on non-member countries and raises the problem of security of energy supplies under the three heads of the availability of resources, the regularity of supply and the holding-down of prices.

These problems are rendered more acute by a new element which is radically transforming the European supply situation. Whereas hitherto Europe has been able to count upon American oil to make good any shortage in times of crisis, various factors such as the exhaustion of certain resources and consideration for the environment have compelled the United States to fall back on the world energy market and to become a net importer of oil at such a rate that as of 1975 it may have to purchase tonnages of Middle Eastern oil equalling the European requirements.

The development of nuclear energy can become a major factor in the Community's security of supply. Deposits of uranium are abundant and widely distributed throughout the world, thus affording a wide range of choice adding a stability factor to supply conditions. The resources are to a large extent located in the Community or are under the control of Community companies. All the phases in the processing of nuclear fuels can be carried out in the Community; however, in order that this may be done efficiently, it is essential for a decision to be reached - and implemented - on the construction of a sufficient uranium isotope separation capacity. Finally, because of its high energy density, nuclear fuel raises far less serious transport and storage problems than do fossil fuels.

2. The economic aspect

The most obvious field of action for fission energy is in electricity generation. The fuel supply situation for electric power plants is currently dominated by petroleum products. Because of their stable, low prices they have gained such an enormous slice of the market that they have become the most important primary source of power for the electricity sector in the Community.

While waiting to be put on a competitive footing, nuclear energy was only able to play a marginal role. Initial construction activities centred on demonstration units aimed above all at providing industrial experience and assisting in the training of reactor operating teams. Nevertheless, the gap between the total cost of conventional and that of nuclear production has gradually been narrowing - and the trend even reversed - first through the adoption of higher levels of unit output and then under the influence of the increases in fossil fuel prices.

themselves to nuclear energy to any significant extent. Apart from the attractive price of fuel oil, this attitude was influenced by several factors, notably, the choice of the type of reactor, the absorption by the power grids of the high unit powers desirable for nuclear installations and the necessity for additional reserves of productive capacity in order to reduce the risk of non-availability. To these were added the very high investment costs characteristic of nuclear power plants.

Towards the end of 1970 the energy market was shaken by a number of factors affecting both supply and demand: an increase in maritime freight charges brought about by a temporary scarcity of tonnage; a rise in demand for fuel oil;
interruptions in the supply of crude oil. The oil market proved highly sensitive to this trend and the price of heavy fuel oil in particular tended to rise quickly.

This situation had scarcely subsided when the producing countries forced the oil companies to adjust their royalty payments to them. The outcome has been a potential increase in the price of petroleum such that the balance between the various forms of power has now tilted in favour of nuclear energy.

Incidentally, it should be noted that the cost of the energy produced by the nuclear power plants derives almost entirely from the value added in the Community at the different stages in the manufacture of the fuels, plant and equipment, and at the stage of production of the electricity itself ; the cost of the energy produced by oil-fired power plants, on the other hand, includes to a large degree the royalties paid to the petroleum exporting countries.

It is certainly possible that market forces will cause fluctuations in fuel oil prices, but these will not settle at a low enough level long enough to cancel the economic advantage of nuclear power.

If, in addition, the average cost of reducing its sulphur content is applied, the upward trend of fuel oil should become firmer.

3. Respect for the environment

Nuclear fission used in the production of electrical energy can consitute a menace in the form of ecological damage caused by, on the one hand, radioactive waste and residue and, on the other hand, thermal pollution. It thus goes the way of all forms of energy conversion which, in one way or another, involve the risk of degrading the environment.

Owing to the different nature of the nuisances caused by fossile fuels it is not really possible to carry out a strict comparison. However, it must be stated emphatically in favour of nuclear energy that, since its initial peaceful applications, it has been subject to close scrutiny backed up by very strict regulations which are continuously being updated with the aim of protecting man and his environment from radioactive radiation and contamination.

Basic standards establishing maximum permissible levels of contamination by exposure to radiation have been defined at international level, and more particularly at Community level. Chapter III of the Euratom Treaty thus lays down "basic standards for the protection of health of the workers and the general public against the dangers arising from ionizing radiations". The definition of safety criteria governing the design and operation of nuclear power plants is based on these standards. The Treaty also states that they can be revised and supplemented and that each Member State shall introduce the appropriate laws and regulations to ensure that they are observed. It further stipulates that a system must be set up for monitoring the level of radioactivity in the air, the water and soil and for ensuring that the basic standards are complied with. Annex I sets out the role assigned to the Commission of the European Communities in this connection.

The radioactive emissions from nuclear installations are thus governed by ever more stringent Community directives and national regulations, and it can be seen that in practice the criteria applied, both for power plant constructors and for electricity producers, are even more severe than those deriving from mere observation of these directives and regulations.

Although the present situation as regards the risks involved in ionizing radiation may be said to be satisfactory overall, it is nonetheless advisable not to relax vigilance concerning this "nuisance", in order to keep the radiation doses from nuclear installations down to a very small fraction of the total dose to which the population is exposed - itself well below the permissible dose.

Special attention must also be paid in this connection to the problem of the final storage of radioactive residue containing high-activity elements and transuranium elements with a very long half-life originating mainly form nuclear fuel reprocessing plants. In particular, the development of technologies guaranteeing absolute leaktightness in storage devices over very long periods is called for.

The introduction of an environmental protection policy at national and Community level will doubtless mean that conventional thermal power plants will be obliged to observe stricter standards than at present as regards the emission of noxious substances. The pollution due to electricity production as a whole will thus be still more tightly curbed and the economic burden of the measures that will be taken will increase the attraction of nuclear energy as compared with fossile fuels.

With regard to thermal pollution, since the thermal efficiency of the current generation of nuclear power plants is lower than that of conventional power plants, the former - for the same electrical output - dissipate into the cooling water about 60% more thermal energy than conventional power plants.

This difference, although appreciable, does not radically alter the problems of thermal pollution, either as regards the choice of sites or from the aesthetic standpoint where the use of cooling towers is necessary.

At all events, these problems should be dealt with under a Community, or even international, siting policy - as is already being done in the case of the Rhine - aimed in particular at making the best use of the natural cooling resources by taking into account the capacity of the river basins and water requirements for other purposes (drinking or industrial water, irrigation etc.).

By reducing the Community's dependence on imports of fossil fuels, and more particularly of petroleum, nuclear energy is not only a diversifying factor capable of rendering energy supplies less vulnerable; it is also able to influence prices of competing forms of energy, since it will henceforth be the cheapest source of energy for electricity generation. This relative state of affairs seems to be stable

(a) because of the fundamental upward trend in the cost of fossil fuels and

(b) because respect of the environment will place a heavier burden on production of electricity from conventional thermal power plants.

Hence there is a clear case for a massive, expanding use of nuclear energy as one of the cornerstones of the Community's energy policy.

SECTION II : Nuclear energy and electricity production

1. The expansion of electricity production

Throughout the Community the expansion of electricity production over the last fifteen years has shown an exponential trend at an annual average rate of about 7.5%, roughly doubling every ten years.. Thus in 1970 demand for electricity amounted to 558 TWh, against 272 TWh ten years before.

An analysis of the factors governing demand, which was contained in the report which is being published by the Commission on the "Long-term energy prospects in the Community" shows that this rate can be expected to be maintained up to 1985 as an expression of probable movements in electricity consumption.

The table below sets out the Community's annual electricity requirements at five-yearly intervals, together with the production figures to be achieved by the power plants (in TWh):

	1970	1975	1980	1985
Annual consumption (including losses)	558	796	1130	1610
Gross production	580	840	1210	1740
Net production	550	790	1140	1625

These prospects are enlarged upon in Annex II, "Structures of Electricity Production".

2. The scope for the use of nuclear energy

The niche which nuclear energy could carve for itself in the future production of electricity does not depend only upon the advantages it offers as regards the environment, costs, and security of energy supply, nor, for that matter, upon the flexibility of site selection conferred by the transport and storage facilities peculiar to nuclear fuels.

The rate and degree of its penetration are subject to various limitations inherent in the electricity generating sector, which preclude nuclear power plants from accounting for all the new capacity to be installed.

In the load diagram certain sources of energy occupy places conditioned by their advantages as regards cost, their local abundance, their flexibility of utilization etc. While lignite and streams thus share with nuclear power the base position in the load diagram, other sources are concerned with peak-lopping equipment, such as lakes, gas turbines and internal combustion engines, which nuclear energy is not attempting to replace, bearing in mind their low utilization factors.

The upshot is that nuclear energy will, generally speaking, only replace the commercial fuels: coal, petroleum products and natural gas, which constitute what is known in electrical generating circles as the "competitive" sector*. The phasing of nuclear plants into the power grids will cause existing equipment to be diverted to the areas of reduced annual utilization. In other words, scope for the use of nuclear energy is limited by the growth of production capacity in the competitive sector.

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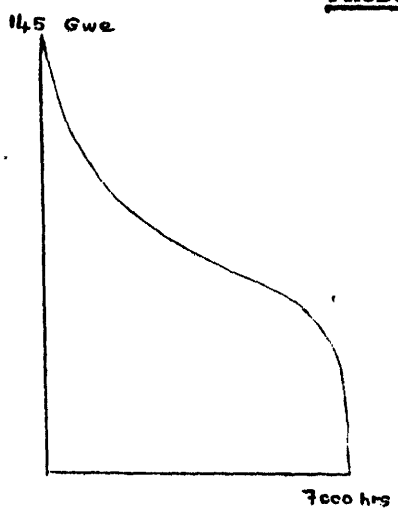
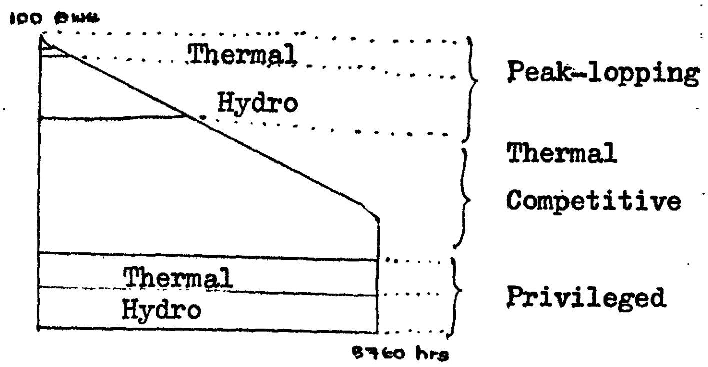
*The Fig. overleaf shows a load diagram

DIAGRAMMATIC REPRESENTATION OF ELECTRICITY PRODUCTION IN THE COMMUNITY



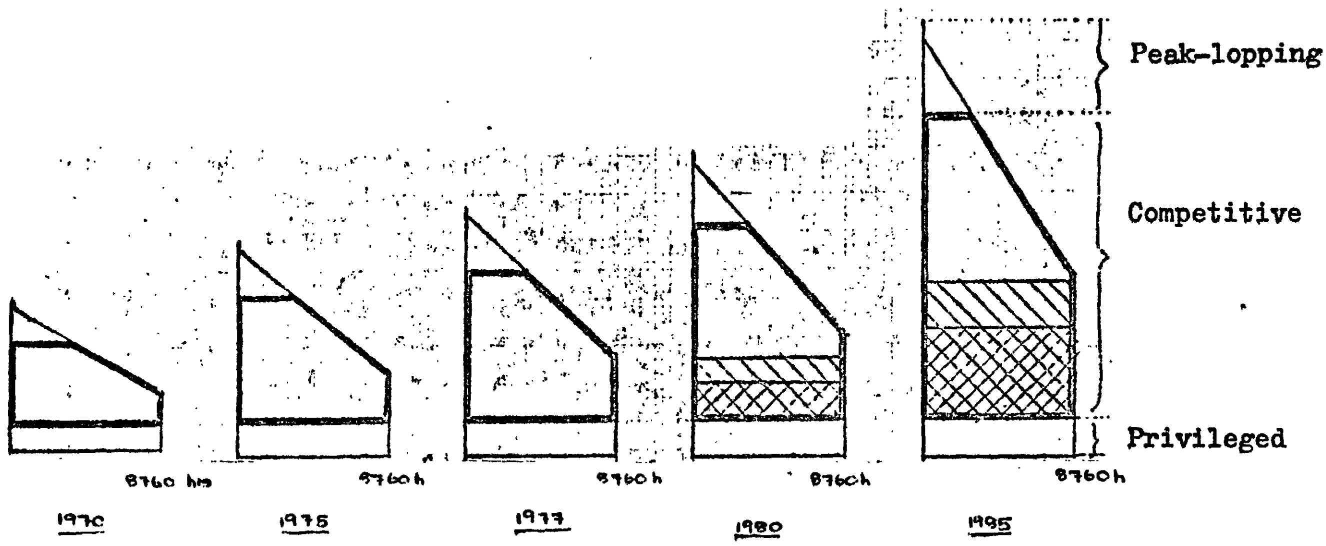
Theoretical monotonic diagram of power-plant utilization



Schematic monotonic diagram of power demand

Example taken from 1970 (rounded-off figures)

Production: 580 TWh	:	Production: 580 TWh
Total power of generating capacity: 145 GWe	:	Maximum grid loading: 100 GWe



Movements in the contribution of the competitive sector to electricity production

Growth of the share of the competitive sector in relation to 1977.

Minimum share of this increase as forecast by the Second Target Nuclear Programme.

NB: As determined here, this proportion does not include production by the nuclear power plants already in service in 1977.

The size of this sector is determined first of all by deducting production from conventional "privileged" sources of energy, namely, water power and geothermal energy, lignite, industrial gases, etc., from total production.

It is generally agreed that the primary equipment (lakes, sluice-controlled heads of water, streams) for water power will not be produced on a large scale in future owing to the growing scarcity of sites and the high specific cost of such equipment. The increase in installed power will derive mainly from the revamping of existing sites and the construction of pumping stations, the result being no more than a slight increase in production.

The Community still possesses considerable reserves of lignite, which are located for the most part in Germany. This fuel is burnt in specific types of plant operating at high utilization factors and will continue to play a part in the growth of electricity production for at least another decade.

Other privileged sources will continue to be used to produce electricity, but their share will remain marginal.

Another factor which must be considered in order to gauge the scope for nuclear energy is the specific equipment and the equipment used to lop the peaks of the load diagram. This makes up a high total capacity but the resultant production is very limited.

The relevant figures, arrived at from schematic models representing the ratio of power demand to production applied by each type of power plant and each category of primary source, can be seen in the following table*.

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*The installed-capacity trend forecast in this table allows for the hypothesis raised later concerning the availability factor of nuclear power plants on being taken into service, and has been adjusted in the light of the nuclear objectives proposed in Section III.

	Net installed capacity (GWe)					Gross production (TWh)				
	1970	1975	1977	1980	1985	1970	1975	1977	1980	1985
Total	134,3	196	229	284	397	580	840	971	1210	1740
Reserve[1]	27,9	34	41	49	60	-	-	-	-	-
Water power										
- Base*	16,2	17	18	19	20	79	82,5	86	89	92,5
- Peak**	19,8	26	28	32	34	41	47,5	51	56	59,5
Thermal power										
- Upper peak	7,3	11	12	15	26	2	2	2	3	
- Privileged	13,3	18	18	20	20	87	115	121	130	133
- Competitive sector	49,8	90	112	149	237	371	593	711	932	1450

*Including geothermal

**Including pumped storage

The decisions taken to date on investment determine the total capacity of the nuclear power plants to be in service in the Community by 1977 and make it possible to estimate the overall volume and structure of the nuclear generating capacity required to satisfy the total demand for this period.

The increase in generating capacity in the competitive sector between 1977 and 1985, which conditions the scope for the use of nuclear energy within the limits of the Second Target Programme, is thus about 130 GWe.

Assuming that nuclear energy fully covers this field, and bearing in mind the 25 GWe already operational[2] in 1977, the Community's nuclear generating capacity could reach 155 GWe gross in 1985, or in round figures 150 GWe net

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[1] For reasons connected with the compilation of the load diagram no figures are given here for energy generated by the reserve capacity; the latter in any case retains a basically operational character.

[2] Excluding some additional capacity intended to replace units that have been phased out in the meantime.

SECTION III: The objectives of the programme

1. Setting the objectives

Price levels for the competing sources of energy will henceforth be high enough to put an end to the querying of decisions relating to nuclear plant investments each time prices fluctuate. The production of electricity by nuclear methods could thus be fixed independently of the delivered prices of the sources of energy in the competitive sector: coal, oil products and natural gas.

However, competitiveness will not suffice to stimulate the maximum possible expansion of the nuclear sector, even when backed up by the desire to reduce the Community's energy dependence and to conserve the environment.

Barriers of various types hinder the requisite speeding up of nuclear power-plant construction: the sub-division of the Community market into national blocs, intensified by divergent technical and safety standards; the piecemeal nature of the industry; a certain holding back of demand, due in large measure to the financing arrangements and to public opinion - recently aware of environmental pollution - which is in some cases hostile to a new form of energy which it regards with suspicion. In addition, a rise in nuclear generating capacity so intense as to increase gross output by about 130 GWe in eight years would imply assurances of fuel availability particularly enriched uranium - which are not inherent in the present prospects.

Faced with this situation, the Commission cannot confine itself to outlining the scope of nuclear energy any more than it could content itself with merely forecasting trends in the use of this form of energy.

It must also lay the foundations of the projects to be undertaken in order to promote successfully the harmonious development of the Community's nuclear sector.

It is accordingly the responsability of the Commission to propose measures which, since they are aimed at the removal of the barriers mentioned above, will make it possible to achieve, and to envisage exceeding a very minor target.

It was with this in mind that the final objective of the current illustrative programme was set, namely, that by the end of the period under consideration - i.e., 1985 - the total power of the nuclear generating capacity in the Community will be at least 100,000 MWe.

The development of the nuclear capacity which will have to be installed in the Community can be projected as follows :
Currently, the nuclear power plants in service produce a total net power of 5500 MWe. Including the installations under construction, total nuclear capacity will be of the order of 12,000 MWe in 1975 and 23,500 MWe in 1977.

Achievement of the target means that at least 45,000 MWe will have to be in service in 1980. Thus the volume of orders must be stepped up considerably as of now. Annual nuclear commitments would have to average not less than 7,000 MWe between 1972 and 1975 and 11,000-12,000 MWe between 1976 and 1980 ; they would thus exceed 50% of the commitments in the competitive sector as of now, and 70% in 1980.

Briefly, nuclear generating capacity would develop as follows:

Year	Nuclear power in service (MWe net)
1970	3,200
1975	12,000
1980	45,000
1985	100,000

Calculations of the corresponding electricity production have been based on hypothetical figures for the dates shown; the relevant movements should be as follows:

Year	1970	1975	1980	1985
Gross production (TWh)	15	65	245	575
Energy equivalent (10^6 toe)	5	20	76	175

In 1985 nuclear energy should account for the following percentages in the Community's energy balances:

10% of total primary energy requirements;
33% of total electricity production;
37% of thermal electricity production.

2. Hypotheses relating to the availability of nuclear power plants

The first hypothesis relates to the time lag between the placing of the order and the phasing of the plant to the line. It has been agreed that this interval, currently five years on average, will continue to apply during the period covered by the Programme, although the reduction of construction time must consitute a permanent goal and the approval procedures may in some cases give rise to prolonged delays.

The second concerns the relationship between power and production. Annual operating time is fixed at 6,500 h, a figure which, however, would only be achieved after a period of run-up to power spread over two years, during which the equipment would operate for 3,000 h in the first year and 5,000 h in the second. This hypothesis, which largely conforms to the present situation, may in time prove to be pessimistic, at least as regards the power run-up period.

The phasingsto the line are to take place half-way through the year.

3. Low-key objectives

Assuming that the nuclear objectives are fulfilled, the structure of electric power and electricity production in the competitive sector between 1970 and 1985 would develop as follows*:

	Net capacity (GWe)					Gross production (TWh)				
	1970	1975	1977	1980	1985	1970	1975	1977	1980	1985
Competitive sector	50	90	112	149	237	371	593	711	932	1450
Conventional power	47	78	88	104	137	356	528	596	687	875
Nuclear power	3	12	24	45	100	15	65	115	245	575

The main point to emerge from these movements concerns the growth in conventional thermal production. Electricity production from fossil fuels has been determined up to 1977 by investment decisions already taken at the present time and will maintain a steady rate of growth between 1977 and 1985, assuming that the nuclear production objectives are simply attained and not bettered. In view of the fact that indigenous fossil fuels (coal, natural gas) will account for a very low proportion of this growth and that the flattening-out of their total contribution towards electricity production can be expected to start in 1980, demand by the electricity sector for imported fuels will thus continue to increase considerably between 1977 and 1985 and even beyond.

In order to put a stop to the growth in imports of fossil fuels for electrical production by 1985, it would be necessary for the targets in the Second Illustrative Programme to be exceeded by a wide margin and to total about 680 TWh by 1985; this presupposes that at that time 115-120 nuclear GWe should be available and thus that the volume of orders to be placed for nuclear power plants during the next eight years should increase by 25% over the minimum figure set in the present programme.

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*This table gives details of the last line of the table on page 18.

However, before analysing the conditions required for the achievement of the objectives it is necessary to make a survey of the resources available to the Community for covering its medium-term equipment requirements and to comply with the provisions of Article 40 of the EAEC Treaty by indicating the market which is being opened up to industry and the investments which the latter needs to make.

SECTION IV: Resources for achieving the objectives: reactor families

In order to achieve its objectives as regards power and energy in the nuclear field and thus help to protect its independence where energy supplies are concerned, the Community must call upon the technological resources of its industry. These resources lie in various families of reactors whose stages of technical advancement and whose commercial potential vary appreciably.

1. The situation in the various reactor families

The development models envisaged in the First Illustrative Programme were based on five types of reactor. The "proven types" consisted of the gas-graphite reactors and the enriched-uranium, light-water moderated reactors. The advanced converters comprised the heavy-water reactors and the high-temperature gas reactors. The last reactor type envisaged was the fast breeder.

Types of reactor	First Programme	Second Programme
Proven	Gas-graphite, Light-water	- Light-water
Advanced converters	Heavy-water, High-temperature	- High-temperature
Breeders	Fast	Fast

Other types of reactor had also been considered likely to find industrial applications, but the hopes held out for them in various quarters have not been fulfilled in recent years. They therefore have no place at all in the potential currently under examination.

Since gas-graphite reactors are no longer included in France's forward construction planning and heavy-water reactors have generally fallen out of favour, the Community's Second Illustrative Nuclear Programme discards these families - although it should be emphasized that they are the only ones to use natural uranium - thus retaining three types, namely, light-water, high-temperature and fast.

Gas-graphite and natural-uranium reactors

France, the only country in the Community to promote the gas-graphite family, has resolutely turned to the other proven-reactor family: thus the Sixth Plan recommends laying down a production capacity using light-water reactors with a total power of the order of 8,000 MWe. A corollary of this decision has been the de facto abandonment of all further uranium-fuelled gas-graphite power plants. It should be noted that in the United Kingdom, the only country outside the Community to have developed this family, the recent decisions on capacity have been mainly concerned with the "Advanced Gas (cooled) Reactor" (AGR) variant, which uses enriched uranium. However, the UK has plans for the construction of light-water reactors.

Heavy-water reactors

Various heavy-water reactor concepts have been studied and developed throughout the world. Canada, in particular, has developed an original design on which it has based its nuclear electricity production. Eight power reactors totalling over 8,000 MWe are in service or under construction at the present time and it is proposed to place orders totalling a further 5,000 MWe in the course of the next two years. The technical and industrial experience acquired, together with the satisfactory operation of these reactors, makes it possible to offer them on export markets. However, although a few small units have been purchased by India and Pakistan, and one medium-sized one by Argentina, no orders, other than those in Canada, have been announced since 1967. Finally, the interest shown by Japan in heavy-water reactors is reflected in the design variant, a prototype of which is to be laid down in about 1975.

Undoubted competence in this field has also been acquired within the Community, notably in Germany, France, and Italy - where the 35 MWe CIRENE prototype is under construction - and even under Euratom's ORGEL programme.

But notwithstanding the construction and commissioning of demonstration units in France (EL-4, 70 MWe) and in Germany (KKN; 100 MWe) it now seems unlikely that there will be a revival of interest in heavy-water reactors in the Community and that they will play a significant part in the Community's construction programmes.

The differences in degree of technical and commercial maturity between the three remaining families in the Second Illustrative Programme are such that a clear distinction has to be made in the actual form of the presentation between light-water reactors and their future competitors, which cannot be expected to contribute on a massive scale to electricity production before 1985. It is for this reason that an analysis of the situation regarding high-temperature and fast reactors has been included in Part Two, which is devoted to the prospects for the period 1985-2000.

2. Light-water reactors

The enriched-uranium light-water family of reactors, in the form of its boiling and pressurized-water (BWR and PWR) variants, is now very much in the ascendant.

In the United States especially, practically all the plants in service, under construction or on order consist of reactors of this type, the total figures for such plants on 1 January 1972 being about 133 units and 113,000 MWe. It is therefore considered that the light-water power plants' generating capacity could reach an output of 150,000 MWe by 1980.

A similar expansion has also begun in the Community, although it is more limited in scope. Light-water plants today account for half the installed nuclear capacity* and by the end of 1975 this proportion will have risen to over 75%. According to the objectives enumerated in Section III, the operational capacity of light-water power plants in the Community should be about 45,000 MWe in 1980 and 100,000 MWe in 1985.

Annex III is devoted to this reactor family.

Technical potential

Although strictly speaking they are already considered to be proven and competitive, light-water reactors will undergo some further improvements and be further modified as regards their design and construction. While the underlying cause of this development is mainly of an economic nature, it can be attributed to differing considerations depending on the circumstances:

(1) Efforts to obtain higher power ratings, which could result in the placing in service of 2,000 MWe units in the 1980's; however, the dimensions of some of the components will raise difficult problems regarding manufacture and transport, mainly the pressure vessels and turbo-alternator sets, and will certainly necessitate the development of new techniques to accommodate the larger unit sizes.

* See Appendix

(2) Consolidation of current technology, the achievement of higher fuel element performance and improvement of the fabrication and quality-control procedures.

(3) Improving the accessibility of the systems for inspection and maintenance purposes.

(4) The formulation of standards, valid on an international scale, relating to the definition and acceptance of equipment and to plant safety criteria.

(5) Finally, and to an extent which is still difficult to forecast, the anxiety of certain constructors to produce original designs in order to free themselves from having to use licences granted by non-Community countries.

Economic outlook

a) Specific capital costs*

During 1969 and 1970, eight power plants equipped with light-water reactors were ordered in the Community. Seven lay within the power spectrum of 770-1150 MWe and the eighth had a more modest rating (450 MWe). The specific capital costs (expressed in constant 1970 values) of these units, which will enter service between 1973 and 1975, vary between 140 and 260 u.a./kWe. Undoubtedly, these costs have not been established from strictly comparable data, but this is only a very partial explanation of the divergence found. This wide divergence stems chiefly from the absence of interpenetration of markets, from the particular situation in any one of these, from the commercial policy practiced by each particular firm and from diverging industrial structures in the countries concerned.

As regards the immediate future, such a disparity could lose its edge, as indicated by the tendency observed in 1971, and the level of the specific capital costs should be in the 170-220 u.a./kWe range (1970 values in the case of the 800-1000 MWe power plants to be ordered between now and 1975.

*The specific capital cost comprises all the direct costs (site, civil engineering, steam-raising plant, turbo alternator, electrical and auxiliary equipment, initial spares) and all the indirect costs (engineering, overheads during construction, unforeseens, interest during construction and operating costs during trials).

Under these conditions, capital costs could work out at 155-190 u.a./kWe for plants entering service between 1980 and 1985, and 145-175 u.a./kWe for those entering service between 1985 and 1990.

It should be noted that the trend of the specific capital costs, expressed in u.a. at 1970 value, takes no account of the additional costs that may result from particular provisions such as air cooling, special safety measures, etc.

b) Fuel cycle cost

In the Community the fuel cycle cost in the case of power plants under construction lies between 1.6 and 2 mills/kWh.

An analysis of the various cost components and the hypotheses on which they are based suggest that the fuel cycle cost can be predicted at roughly the following values:

Plant enters service	Cycle cost - mill/kWh
1980-85	1.5-1.7
1985-90	1.4-1.6

It should be noted that these forecasts agree with those made in the United States* at the end of 1969.

c) Operating, maintenance and insurance costs

The operating, maintenance and insurance costs used by electricity producers in the Community in order to estimate the generating cost of the energy produced by power plants being constructed lie between notional values of 4 and 5 u.a./kWe a year.

*The USAEC estimates the fuel cost for reactors ordered in 1971 at 1.70/1.75 mills/kWh.

A downward trend towards 3 and 2.5 u.a/kWe a year should emerge in the case of power plants entering service in 1980 and 1985 respectively, as a result of improvement and rationalization of the various operating sequences and the formation, in the case of several power plant operators, of joint teams specializing in maintenance operations.

It should be noted that in the United States a constant value of 2.1 u.a./kWe a year from 1975 onwards is generally taken in respect of this item.

d) Overall generating cost of electricity

On the basis of the foregoing hypotheses and an assumed utilization time of 6500 hours per year, the foreseeable trend in the generating costs of the energy produced by light-water power plants is set out in the following table; the figures at either end of the ranges shown derive from the annual charges on fixed assets, which give overall rates of 10- 13%.

Plant enters service:	Energy generating cost (mills/kWh 1970 value)
1975-1980	4.9-7.1
1980-1985	4.3-6.0
1985-1990	4.0-5.5

These costs are to be considered as indicators allowing the economic assesment of expected improvements in nuclear energy and comparisons between different types of reactors.

In the longer term the experience acquired by European designers ought to bring about a reduction in costs (at constant value); if this reduction is to be appreciable, however, greater standardization of the models offered to producers would be necessary, together with repeat orders for each of these models, the replication effects being intensified as the market widens.

SECTION V: The extent of the potential market and the investments required

In order to evaluate the investments of every kind which are likely to be needed to achieve the objectives of the programme, it was assumed that the lowest target, expressed as 100,000 nuclear MWe to be in service by 1985, would be achieved but not bettered and that the new power plants providing this capacity would be exclusively of the light-water type.

It is clear that this assumption does not detract in the slightest from the desirability of exceeding the target figures or from placing significant nuclear production capacity using advanced reactors in service before 1985.

1. <u>The market for nuclear power plants</u>

The table below, which is based on predicted technological development and the trend of costs of light-water power plants, summarizes the estimates of future average annual investments in electricity generating units.

<u>Breakdown of annual investments in nuclear power plants between 1971 and 1980</u> (Markets expressed in millions of u.a. at 1970 value)

Period	1972-75		1976-1980	
Average capacity ordered each year	6,700 MWe		11,000 MWe	
Average number of power plants ordered each year	5-8 800-1200 MWe/unit		6-9 1200-200 MWe/unit	
Total annual investment	938	1470	1700	2090
<u>Principal markets</u>				
Civil engineering[1]	97.5	152.4	177	217
Nuclear steam raising plant[2]	281	441	510	628
Turbo-alternator[3]	234	366	424	522

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[1]Including leaktight containment

[2]Including pressure vessel, steam generators, pumps, fuel handling and storage equipment, instrumentation

[3]Including condenser, preheaters and feedwater circuits.

These movements would seem to be fairly modest for the immediate future, since it will be necessary to await the end of the current decade in order to achieve a volume of orders of 6-9 power plants a year for the entire Community. This being so, it is unlikely that the bottlenecks observed at the major component manufacturers in the United States a few years ago will also occur here.

It should be emphasized that the demand is expressed in units of production rather than in the total power to be installed. The uncertainty which makes it necessary to term such demand "probable" and which produces a fairly wide scatter under the head of "number of power plants ordered each year" in the foregoing table arises from:

(1) The technical feasibility for constructors to develop increasingly large units, and

(2) The technical and economic feasibility for electricity producer distributors to include more and more powerful units in their generating capacity.

At all events, the cost of the energy produced will fall when unit sizes increase and the pressures leading towards economies of scale will continue to make themselves felt.

This overall stimulus, supplemented by certain effects of the competition between constructors - or reactor families - could accelerate the trend towards greater size at the risk of losing some of the potential benefits of standardization.

The outcome of these movements is such that, measured by the number of units to be ordered each year, the market for nuclear power plants could already be more or less static in the Community. This highlights the appeal and importance to European industry of capturing export markets.

2. The fuel-cycle industry

The rise in production of electrical energy generated by nuclear means will involve a similar growth in the nuclear fuel industry, the characteristics of which are dealt with in Annex IV. The rate of increase in the turnover of this recently established industry will be even higher than that achieved by power plant constructors.

Market trends in the fuel cycle industry
(Light-water reactors)

	Annual			Cumulative
	1975	1980	1985	period 1975-85
Demand for natural uranium[1] in tons	4,200	10,800	20,500	126,900
Demand for separative work units in 10^3 kg[1]	1,640	5,920	12,600	70,500
Demand, in tons, for enriched uranium for fabrication purposes: (1) core (2) refuelling	 370 210	 1,030 980	 1,850 2,700	 11,820 12,500
Irradiated fuel elements for reprocessing, in tons of uranium	110	720	1,940	9,360
Total uranium in kg[2]	820	5,450	14,900	71,500

[1] For a tails assay of 0.25%

[2] To this figure should be added what is produced by the gas-graphite reactors, whose capacity of 2500 MWe generates a production of 1.3 tons a year.

While the investments to be made in light-water reactors by electricity producers will probably increase by no more than a factor of 2 between 1975 and 1985, the money spent on orders for natural uranium will increase fivefold over the same period and for the same family, while that spent on orders for irradiated fuel reprocessing will increase tenfold.

Estimated turnover (in millions of u.a. at 1970 value) for the various stages in the fuel cycle

	Annual			Cumulative
	1975	1980	1985	1975-85
Natural uranium at mine	65	170	320	2,000
U_3O_8 conversion	10	25	50	300
Enrichment	50	190	400	2,300
Fuel element fabrication	65	170	320	2,100
Irradiated fuel transport	-	5	10	60
Irradiated fuel reprocessing, including radio-active waste storage	5	25	50	300
Reconversion into UF_6	-	-	5	30
Credit for plutonium recovered from light-water reactors	4	25	75	350
Total net expenditure on the cycle after deducting the credit for plutonium	190	560	1,100	6,700

Different industrial growth rates thus apply to the various stages of the fuel cycle, each stage having a growth rate congruent with the position it occupies in the cycle.

The investment problems will thus affect the various stages in different ways, depending on their inherent characteristics.

Cumulative level of additional investments to be made in the Community (millions of u.a. at 1970 value

(Estimate based on the objectives of the programme)

Cumulative total up to:	1975	1980	1985
Supply of U_3O_8 (exploration and production)	100	350	750
Conversion/reconversion	low	50	150
Enrichment	-	700	1,400
Fuel elements fabrication	-	50	100
Reprocessing*	low	low	100
Cumulative total	100	1,200	2,500

*Owing to an excess of capacity at the present time, new investments will not be made on any significant scale until after 1980.

Services

The irradiated-fuel transport market in 1985 will represent a turnover of the order of 10,000,000 u.a. on the basis of a future estimated price of 5 u.a./kg uranium transported; currently the incidence of transportation of fuel elements for the light-water family is not sufficiently high to warrant reference to an actual transport market, and the widely varying prices applied are not representative.

Radioactive waste storage is more an activity concerned with protection of the biosphere against dangerous substances at the end of the cycle, and largely the preserve of the public authorities, rather than a commercial activity which can be undertaken by private enterprise.

SECTION VI: Conditions required for the achievement of the objectives of the Programme

The Community nuclear-generated electricity market emerging from the minimum target set by the Programme is expressed as a volume of orders for about 75,000 MWe to be placed between 1972 and 1980.

This figure is modest when compared with the size of the market which nuclear energy would gain if, as stated in 3.2, all the production units in the competitive sector entering service from 1977 onwards were equipped with nuclear reactors.

The minimum target of 100,000 nuclear MWe available by 1985 is in line with the national forecasts mentioned previously, as shown in the table below, which, it must be emphasized, is by way of illustration only.*

Nuclear electricity generating capacity in service (GWe net)

	1972	1977	1985	increase 1977-1985
West Germany	2.2	13.6	45	31
France	2.7	6.3	27	21
Italy	0.6	1.4	16	15
Benelux	0.1	2.2	12	10
TOTAL	5.6	23.5	100	77

However, when one considers the technical barriers of all kinds which are liable to hinder the expansion of nuclear energy during the next few years, it would appear that achievement of the Programme objectives does not of necessity stem from a natural trend in demand, which the supply would manage to meet spontaneously.

*In the case of joint ventures by two or more countries the unit's entire capacity is attributed to the country in which it is located.

Whether they have an overall effect on the nuclear sector in the Community, such as the partitioning of markets at a national level, or poor dependability of fuel supplies can do, or whether the effect is regional, such as financing difficulties or the opposition of the general public, these potential barriers show that the Programme objectives will not be achieved unconditionally.

1. Opening-up of markets

An essential condition for the harmonious, rapid development of nuclear energy in the Community is the creation of a common market for equipment, which will raise the efficiency of the industry and make itself felt, particularly at the product level, in the form of lower costs, shorter lead-times and greater reliability. As a further result, the position of the European industry vis-à-vis overseas competition will be strengthened.

Among the causes of the current partitioning of markets is, first and foremost, the traditional links between the suppliers and the electricity producers. The producers can only sever these links if considerably more favourable conditions concerning technology, prices or deadlines are offered them. Another factor is the concern on the part of certain governments for the protection of their industries' development, especially when the nuclear sector is involved, since this is the one which to a great extent mirrors the national achievements in technology and thiwhichertherefore, political influences are inevitably brought to bear at every stage in the decision-making process.

Basically, the opening-up of the Community's internal market is impeded by a series of obstacles in the form of the regulations and procedures peculiar to each country.

The technical specifications and standards and the safety criteria, which are dealt with later on, undoubtedly hamper producers in the examination of tenders which may be submitted by companies in other Member States; similarly, these differing specifications, standards and regulations confront suppliers of electro-mechanical equipment with adaptation problems when they attempt to expand their usual markets.

These barriers to the satisfactory functioning of the common market in nuclear electricity generating equipment will not yield of their own accord. Their elimination calls for the formulation and implementation of appropriate measures.

Furthermore, the nuclear electricity industry in the Community has hitherto consisted essentially of national industries existing side by side and operating in their respective national markets, the development of which they have tended to follow rather than lead. This state of affairs explains the prevailing structural differences and reflects the absence of market penetration to which it also contributes.

A prerequisite for any progress in this field is that every electricity producer should offer manufacturers throughout the Community the possibility of genuine access to his market. In this connection the institution of a procedure for consulting all Community undertakings possessing the necessary qualifications is both to the advantage of the electricity producers and meets the exigencies of the extension of competition as required by the common market.

2. Harmonization of criteria and standards

The opening-up of markets, the stepping-up of intra-Community exchanges and structural reforms in industry involve the removal of the technical barriers to free competition.

This mainly concerns the criteria and standards governing the design, construction and operation of nuclear power plants and the installations within which the various fuel-cycle activities take place. This also concerns the carriage of radioactive substances, notably in the form of irradiated fuel and radioactive waste.

Since these criteria and standards embody far-reaching social aspects because of their relevance to public health, safe working conditions and protection of the environment, there should in no case be divergences between one country and another in this field.

The requirements to be met for this purpose would have to be determined by agreement between the manufacturers, operators and national safety and control organizations in such a way as to reconcile the priority aspects of safety (prevention and limitation of the consequences of accidents) - apart from health considerations - and their consequences of all kinds, notably from a technological and economic point of view. The final goal is uniformity of the technical basis for the national administrative procedures governing the granting of construction and operating licenses for nuclear installations and of permits for the carriage of radioactive substances.

In the particular case of the light-water power plants, it is advisable to be forearmed against the penalties incurred through any delay in the adoption of commonly recognized criteria and standards. From a purely economic point of view, a delay in starting up a 1000 MWe LWR power plant on completion of its construction would involve financing charges of a potential order of two million u.a. a month. Likewise, a shutdown of a power plant of the same power rating constitutes a production loss of over 100,000 u.a. a day. Experience has shown that such delays and shutdowns most often stem from the complexity of the supervisory procedures and methods, as well as from ineffective quality control.

For this reason a simplification and standardization of the methods, criteria and codes applied by the safety and supervisory organizations would make a major contribution towards reducing the time lag between the decision to build and the commencement of operation. Additionally, it would facilitate the standardization of units and components, this leading to increased productivity.

Two avenues are open for the pursuit of harmonization:

(a) A joint study of concrete projects;
(b) A systematic study of designs and techniques and the standardization of components.

First step: joint expert study of concrete projects

The studies concerning specific projects which are carried out by Community expert groups should be conducted as in the past: in conjunction with the authorities and the competent organizations in the countries concerned, without trespassing on the legal and administrative prerogatives of the competent national authorities.

The joint examination by experts of specific technical problems has to date proved the most direct method of comparing the practices employed in the various countries. Henceforth it should be biased towards the examination of new technical problems, e.g.,

(1) Extension of the operating limits of proven-type installations;
(2) Extrapolation of prototype installations to industrial use;
(3) Application of reliability methods of analysis.

Second step: systematic study

In addition, efforts should be directed to systems and components which are most suitable for standardization. Only the designs and equipment of a sufficiently advanced technology and those for which an international market is in existence or is being developed should be taken into consideration.

The working methods should allay the justified fears that they might tend to become restrictive rules which could stifle further developments; rather should they be moulded into syntheses - e.g., in the form of guiding principles or specimen reports - of sound practice, which would, moreover, be subject to periodic updating.

Apart from the new initiatives to be taken, it would be necessary to ensure consolidation of the activities which have occasionally been performed in this field under the aegis of other international organizations, such as the ISO, the CEN (European Committee for Coordination of Standards) and the IAEAC (International Atomic Energy Agency), Vienna.

For technical and marketing reasons, it would seem appropriate in the standardization of components to accord priority treatment to the technical problems inherent in mechanical components, i.e., to pressure vessels in power reactors and other parts subject to pressure in the primary circuit, such as pipework and junctions and, where necessary, valves and pumps.

In line with the recommendations made by UNICE to the Commission, it would be advisable at an early stage to compile a detailed and above all comparative schedule of the various codes, rules and standards in existence at the national level.

International carriage of radioactive substances

The development of nuclear energy also involves an increase in movements of radioactive substances. By their very nature, these come within the category of dangerous goods. In order, therefore, to be performed under the safest and most economic conditions, these transport operations must be governed by regulations which are strict and, in the case of international movements, uniform for all countries.

The harmonization of standards in this field is considerably further advanced than in that of fixed installations.

The IAEA has undertaken the task of formulating rules applicable on a global scale to the transport of radioactive substances. An initial text, in the form of a regulation issued in 1961, was revised in 1966. A new revision has just been completed.

In accordance with the recommendation made by the Board of Governors of the IAEA in 1964, which calls upon the Member States and organizations concerned to use the IAEA transport regulation[1] as a basis for the national and international regulations in this field and to ensure that it is applied to international movements, the relevant IAEA provisions have been incorporated in almost all the international regulations having force of law[2] (and will be so whenever they are revised). In the same way, many national regulations on the subject derive from the IAEA Regulation, thus conferring upon it almost universal scope.

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[1] As regards its legal scope, the IAEA Regulation is only mandatory in the case of operations directly carried out or involving action by the Agency.

[2] RID - International regulation governing the conveyance of dangerous goods by rail (CIM Convention)

IATA - International Air Transport Association

IMCO - Intergovernmental Maritime Consultative Association

ADR - European agreement concerning the international carriage of dangerous goods by road (ECE)

ADN - European agreement concerning the international carriage of dangerous goods by inland waterways (ECE).

This, then, constituted the initial impetus to harmonization of the regulations, which should be continued. It is calculated to ensure and increase the safety essential to the various stages in the carriage of radioactive substances (packaging of such substances, organization and performance of the operations involved) through the observation of the common rules, the preparation and subsequent practical application of which necessitated theoretical and experimental research on a large scale. Furthermore, this harmonization makes it possible to improve the transport economy - e.g., through technological progress following on from the research carried out, through the interchangeability of packaging materials, through the creation of a fleet of special vehicles and through the routine nature of the administrative formalities.

3. Industrial structures

The degree of the Community's nuclear industry readiness to cope with the potential demand, as shown in its skill in designing, producing, marketing and guaranteeing equipment and products, appears to vary both as between countries and among the various sectors of this industry's activity.

A situation such as this is an economic entity unencumbered by barriers would in itself bear the seeds of suitable remedies for this structural weakness and the sheer volume of potential demand would speed up the desired changes.

In order that the Community market may be unified and rendered fluid it is necessary that, in conjunction with the efforts directed towards the removal of technical barriers, the Community industry should accelerate the introduction of its future structures and set them up on the widest possible scale.

A. <u>The nuclear power-plant construction industry</u>

The current situation regarding the structure of the nuclear industry in the Community prompts two comments in particular:

(1) The firms concerned are fragmented and to all intents and purposes without intra-Community links;
(2) They have a low profitability rating.

Another feature of this situation is a certain degree of dependence on American LWR technology.

<u>The large number of firms in the Community and their fragmented nature</u>

In the Community at least seven firms construct LWR power plants for the market.

Three groups develop high-temperature power plants, but to date only one has been given the go-ahead to build a prototype power plant.

In the field of sodium-cooled fast breeders, two groups in the Community are engaged in the construction of prototype power plants, using different technical solutions, one on a multinational scale and the other at a purely national level.

Certain structural reforms have already taken place in Community firms. However, they have always done so on a purely national scale, whereas they ought to be carried out in an international context, as was emphasized by the Commission in its report to the Council on the reorganization of the electro-mechanical industry, dated 22 April 1970. Since the initial stages of transnational collaboration were only embarked upon recently (agreement between KWU, TNPG, SNAM PROGETTI, BELGONUCLEAIRE, etc.), it is as yet impossible to evaluate the consequences.

The low profitability of nuclear activities

The fragmented nature of the industry adversely affects the profitability of the constituent firms' nuclear activities. Even the largest European constructor is expecting its nuclear power plant division to show a loss for several more years (including, admittedly, the writing-off of R&D expenditure). According to recent statements, the American firms too have not yet written off the R&D expenditure incurred during past years, despite the considerable market they have captured (about 100,000 MWe in five years). All the same, it should be stressed that nuclear power plants account only for part - and in some cases a fairly small part of the firms total activities.

Only highly-capitalized companies possessing a diversified financial structure can afford to invest large sums in nuclear activities which may well become profitable only after a relatively long period, especially in the current situation, characterized as it is by:

(a) A persistently low number of orders each year for nuclear power plants;
(b) A market which is still heavily partitioned between one country and the next;
(c) Numerous firms in competition.

Dependence on US light-water reactor technology

With one exception , almost all Community suppliers of light-water nuclear power plants have to rely to a greater or lesser extent on the technology of US companies, namely, General Electric, Westinghouse and Babcock and Wilcox.

The table below sets out the licensing arrangements of the companies engaged in the supplying of light-water nuclear power plants:

Licenser	Licensee	Country	Participants or groups	Remarks
General Electric	AEG	West Germany	KWU	-
	AMN	Italy	IRI	-
	SOGERCA	France	CGE-ALSTHOM	-
Westinghouse	ACEC	Belgium	Westinghouse International Europe	taken over by Westinghouse in 1970
	FRAMATOME	France	Schneider	licence renewed for 10 years in 1970 with extension of exchange of information
	FIAT		a joint venture with TOSI and MARELLI for tendering purposes	-
	BREDA	Italy		-
	Thermomeccanica			-
Babcock and Wilcox, New York	BABCOCK Atlantique	France		previous agreements extended to nuclear field
	BBR	West Germany		ditto

These technological links may constitute a handicap for the industry in the Community since they have the effect of slow-down multinational industrial regroupings, restricting export prospects, or causing companies to neglect their own research and development.

The necessary structural transformation will have to be aimed in particular at the creation of a competitive nuclear industry in the Community which will enable the companies concerned to:

(1) adapt to and satisfy demand by setting up suitable manufacturing capacity, engineering facilities and industrial architecture;

(2) assimilate American technology and gradually free themselves from licensing agreements through the acquisition of knowhow and the taking-out of patents on their own account;

(3) develop new technologies for advanced power plants, as regards both the nuclear steam-raising plant and the fuel cycle.

In its final phase, this transformation must lead, in particular, to the formation of three or four major industrial groups possessing the capacity to design, develop and build both proven-type and advanced reactors and to attack the world market with a reasonable chance of success.

Within these groups, the introduction of increasingly large units will mean investment on a scale which demands that the relevant decisions be taken with a concern for rationalization and specialization.

At the same time, the problem of the harmonization of technical standards will thereby by simplified.

It is a far cry, however, from the present state of affairs to this desired structure. The field of major components, too, is beset by problems of various kinds.

As far as turbo-alternators are concerned, the European industry is not in a position to produce shafts for the turbo-sets installed in plants with a capacity exceeding 900 MWe.

This results in a dependence on overseas supplies, with all the hazards that entails, especially as regards delivery delays.

The pressure-vessel and steam-generator sector, in which the market has already been opened up to some extent, is at present characterized by a substantial excess of supply over domestic demand, an excess which the export market is not adequate to absorb.

The position of Community firms in external markets depends largely on their position in the domestic market. It is also determined by factors unrelated to their technical capability and cost efficiency, an important role being played, in particular, by export financing conditions.

B. The fuel-cycle industry

The nuclear fuel industry consists of several sectors (uranium-bearing concentrate supply, enrichment, fuel fabrication, reprocessing), each of which not only possesses its own particular structure but also is at quite a different stage of development.

Uranium-bearing concentrates: supply and conversion

With the help of the national laboratories, the uranium-mining industry developed its own prospecting, extraction and ore-treatment techniques. It then evolved within a national structure marked by more and more private funding, while occasionally entering into multinational funding arrangements. This industry operates both inside (basically in France) and outside the Community, mainly in Africa, Canada and Australia. Its world-wide character is becoming more pronounced day by day, as regards both its field of operation (exploration and extraction) and its market outlets.

However, the weakness of prices of these ores on the world market may induce fears that this industry's rate of expansion could suffer a serious setback in consequence. This would have a highly adverse effect on the ability of the Community's industry to compete with American companies, which are cushioned by their own highly protectionist domestic market, where the price quoted for uranium-bearing ores is higher than that on the world market.

As regards the conversion of uranium concentrates into uranium hexafluoride, the Community's industry in this field is asserting itself on the markets. This activity is fundamentally bound up with that of enrichment.

Uranium enrichment

At present there is no industrial capacity in the Community able to undertake uranium enrichment for peaceful purposes. This deficiency may well be highly detrimental to the development of nuclear energy in the Community, as shown in para. 5 of this Section.

Fuel element fabrication

Industrial fabrication of enriched-uranium fuel elements for LWR power plants is currently booming.

The production capacity of each of the six plants belonging to six different companies in the Community lies between 50 and 200 tons a year, and three of these companies are planning to step up their capacity to 200 or even 500 tons a year, on the basis of uranium hexafluoride. These plants are financed entirely from private sources.

Each of these various companies is linked with a power plant constructor. There is consequently no independent European manufacturer in the Community, whereas there are already several in the United States. The principal cause lies in the restricted nature of the market at the moment. In the longer term, however, a similar trend can be expected to emerge, thus providing a stimulus to competition.

Unfortunately, it must be pointed out that to date each of these facilities has confined itself to supplying its domestic market, the only exceptions being of a marginal nature. Practically speaking, therefore, no intra-Community trade has taken place in this field.

Furthermore, these different companies are all linked, either financially or by technical or licensing agreements, with the US companies General Electric or Westinghouse, except for the chief of them. However, it should also be noted that links are sometimes forged in the reverse direction.

In view of the ties between these companies and the power plant constructors, the industrial structure of this section of the fuel industry will, at least during the next few years, follow a parallel course of development to that of the power plant construction industry and the conclusions drawn with regard to the one industry likewise apply to the other.

Fuel reprocessing

This industry (which was originally aimed mainly at the recovering of plutonium from gas-graphite power plants) is preponderantly financed from public funds and at the present time it is experiencing a major crisis. It must not only adapt in order to be able to reprocess fuel from light-water power plants, but also bear in mind the fact that at the moment the reprocessing market for these fuels is embryonic and falls well below the forecasts made in 1960-65.

This was the reason underlying the amalgamation of the interests of the three main European concerns in this sector, namely British Nuclear Fuels Limited, the "Département des Production du CEA" and the "Gesellschaft für Wiederaufbereitung von Kernbrennstoffen". A tripartite service company, "United Reprocessors GmbH", has been formed which has declared its readiness to accept other partners. Under Article 85 of the EEC Treaty the Commission has been notified of the formation of this company and is required to state its views thereon.

4. Stimulation of demand

The additional investment required by nuclear generating equipment as compared with conventional plant is for many producers a constraint which is causing them to drag their feet on switching to nuclear energy, despite their faith in its profitability as calculated on the life of the installations.

In the course of time, improvement in the reliability of nuclear power plants, thus reducing the level of the reserves required, together with the effect of a certain degree of standardization in helping to lower specific investments, will cut down the additional outlay. However, there can be no question of waiting for this potential to be realized if the minimum objective set by the current Illustrative Programme is to be achieved.

The solution of the financing problems is primarily in the hands of the electricity producers. Nevertheless, the public authorities can ease this task, especially in the matter of taxes. However, the situation is so urgent that, even before they can benefit from the measures taken in this field, as in that of rating, the producers should have the widest possible access to the capital market.

This being the case, the Commission has proposed that the Council give temporary authorization to invoke Article 172 (4) of the Euratom Treaty, which in particular empowers the Commission to raise loans so that the Community can contribute to the funding of nuclear power plants.

The Commission could thus act as an intermediary between the electricity producers and the world capital market. Through the surety which it would provide for these operations on behalf of the six governments, it could obtain for the producers the best borrowing terms on the widest possible money market.

Arrangements with wider implications in the matter of the budgetary structure and the planning of investments to be made in the electricity sector as a whole should be introduced concurrently with the implementation of measures of this kind, which, although admittedly of a restricted nature, are nevertheless appreciable as regards their immediate effect.

Within this overall view, account should be taken at a Community, and at an even higher, level of the individual or bilateral decisions relating to production and transport equipment by endeavouring to secure a more intimate pooling of reserves.

The gains notched up in production investments would largely compensate for the additional expenditure aimed on intensifying the linking of grids, on condition, however, that long-term insurance agreements - or contracts for reciprocal aid in the event of equipment failure - should be concluded between the producing companies, wider-ranging than the current aid agreements, which are of a very limited duration, and more flexible than the customary planned exchanges of energy.

Gradually, such joint planning would dovetail naturally with arrangements on other problems concerning equipment, and mainly that of siting policy.

Finally, at a time when the loan capital required for the countless needs of expansion threatens to remain in short supply and therefore expensive for a long time to come, a more extensive use of self-financing should be encouraged. However, this process by its nature can only come about gradually and it clearly involves a certain flexibility in the determination of charges and taxes.

5. Dependability of nuclear fuel supplies

In order that the corresponding electricity requirements may be met it is not sufficient that the market for nuclear generating equipment should open up and the demand exceed the supply, nor that the supply should be organized on the scale of the opportunities offered. It is also necessary to ensure the uninterrupted supply of fuel for nuclear power plants.

The known world reserves of uranium, as stated in 1.1, are plentiful and geographically scattered and the problems raised by the carriage and storage of this fuel are of the easiest to resolve. While on the basis of these data, it is legitimate to foster a speed-up in the swing to nuclear power, security of power supply in the Community demands that the successive stages of the nuclear fuel cycle in industry should be kept satisfactorily supplied with energy "feedstocks".

This means that the Community must exercise sufficient control over the production of uranium ores and concentrates, the production of enriched uranium and the use of the plutonium generated in its nuclear power plants.

Uranium-ore supplies and concentrate production

The reasonably dependable reserves located in the Community and those controlled by the Community's industry in non-member countries currently total more than 75,000 tons of uranium. Cumulative requirements will be about 55,000 tons by 1980 and about 140,000 tons by 1985.

Bearing in mind the time-lag of 7-8 years between the initial prospecting and production, the Community's mining industry will have to invest in exploration during the present decade in order to guarantee dependable, continuous supplies from sources within its territory or under its control during the following decade. To this end, a Community strategy integrating the activities of the various companies may prove to be necessary, since until 1980 the world uranium market will probably be characterized by a slackness in demand relative to the known resources and to the means of production that are, or will be, available.

In the present conditions of pressure on prices, financial aid by the public authorities would seem to be inevitable. But such aid should be only temporary, and it is desirable that, outside the protected US market, the prices should settle at a level which will allow the mining industry to carry out unaided the exploration for, and exploration of, the new resources which are indispensable.

The facilities for converting U_3O_8 into UF_6, which are for the most part in France, should be able to cover Community needs until 1975 and possibly 1977-78, since their current capacity is 3,000 tons a year and could be quickly stepped up to 5,000 tons a year.

Uranium enrichment

At the moment the Community is completely dependent upon external supplies, and in effect on a single source, for its enriched uranium. Whereas the short-term outlook regarding availabilities may be considered satisfactory, the same will no longer apply at the end of the present decade. A comparison between the cumulative world requirements for enriched uranium over the next few years and the cumulative production of existing or projected enrichment facilities - mainly American - shows that it will no longer be possible to meet these requirements from around 1980.

The deficit will represent 18-21 million kg of SWU for 1982 and 38-47 million kg of SWU for 1985 (with and without plutonium recycling in thermal reactors respectively). It is important that this situation should not result in a breakdown in the Community's supplies of enriched uranium around 1980.

The problem thus raised must be thoroughly examined and settled before 1974 if the operators are to be sure that the nuclear power plants they order at that time can be properly supplied with fuel.

The installation of an enrichment capacity in the Community would make a fundamental contribution towards achieving the objectives of the existing programme, since without it the development of the Communities' nuclear potential would be seriously threatened at the industrial and still more at the commercial level.

The setting-up of an internal enrichment capacity would enable the Community industry to perform all the activities involved in the fuel cycle and would also hold out the prospect of improved management through the integration of successive industrial operators. In addition, the availability of comprehensive fuel-cycle services would enable the internal market to expand under the stable conditions required and consolidate the position of the Community's industry on external markets, where the fuel-cycle guarantees would constitute a major trump card in competition between reactor constructors.

At its meeting held on 16-17 December 1970, the Council of Ministers authorized a special study group under the Consultative Committee on Nuclear Research to compile a dossier showing the technical and economic characteristics and performance figures for installations, based on the various technologies developed in the Community.

Taking into account the findings of this study group, the Commission intends to update the proposal which it submitted to the Council on 22 May 1969 recommending the creation of independent enrichment capacities in the Community.

The plutonium market

Utilization of the plutonium produced in the Community by proven reactors can in no case provide a solution, even temporary, of the problem just discussed. Nevertheless, the decisions concerning this utilization are to be taken during the period covered by the present programme.

The annual production of plutonium in the Community will develop substantially as follows:

1975	2.0 tons (total Pu)
1980	6.5 tons (total Pu)
1985	15.0 tons (total Pu)

As of 1975 it will be amply sufficient to cover the foreseeable needs of R&D and those of the fast reactors still under construction or already in operation. The surplus of available plutonium will increase by approximately 2.5 tons in 1975 to 9 tons in 1980 and 40 tons in 1985.

The only immediate outlet for this surplus lies in recycling in light-water reactors. However, the quantities available for this purpose will clearly be too small, at least until 1980, to yield an adequate turnover for the Community's plutonium-bearing fuel industry.

It is thus probable that the cost of fabricating plutonium-bearing fuels in the Community will remain appreciably higher than that of uranium-bearing fuels for about ten years and that, in consequence, the price of plutonium will tend to be well below its theoretical energy equivalence value (about 7 u.a./g).

On the world plutonium market*, which is overshadowed by the very large surplus in the USA, the trend towards lower prices can be expected to be less pronounced, and above all less protracted. In view of the quantities of fissile material in existence, the American plutonium fuel industry should be expanding appreciable as of 1973/1975.

In these circumstances, some European electricity producers may be inclined to look elsewhere for more profitable markets for their plutonium output.

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*World surplus 1971-75: 28 tons of total plutonium
1976-80: 140 tons " " "

It is thus to be feared that the small quantities of surplus plutonium in the Community will be exported, at the risk of handicapping the Community's plutonium recycling industry, which would be deprived of the possibility of gearing itself to meet the demand which it could satisfy towards the end of the current decade.

The Commission, together with the quarters concerned, has initiated an examination of the means which should be employed to enable the Community's plutonium fuel industry to come safely through this diffucult period.

6. Public opinion and the environment

As stated earlier the use of nuclear energy in the Community has to comply with basic rules governing the laying-downnand observance of radiation protection standards which the Commission is empowered by the Euratom Treaty to determine. Design and acceptance criteria for nuclear installations will still have to be harmonized if the barriers in the Community market really are to come down, but these criteria have complied with the basic regulation since its promulgation. It has been stated elsewhere that the problems relating to thermal discharges do not differ in essentials as between nuclear and conventional power plants and thus do not require specific treatment.

Even so, the existence of Community rules concerning protection against ionizing radiation has not precluded certain hostile reactions among public opinion, where misgivings have been prompted by incomplete or misleading information. These reactions have shown themselves particularly in the case of site selection for nuclear installations.

Public oppostion constitutes a potential restraint on the development which is to be desired in the nuclear energy field - and which is in some measure justified by ecological criteria. This obstacle is receiving the full attention of the responsible authorities and in particular that of the Commission.

It is necessary in this matter to provide the public with complete and objective information concerning the operation of nuclear installations and to emphasize the important part that accident hypotheses play in the design and construction of these installations.

It is particularly important that such information should make clear the stringency of the regulations designed to protect any person outside a nuclear power plant, even in the case of the maximum credible accident, i.e., an accident resulting from the hypothetical simultaneous occurrence of a series of critical events, which is even less probable than the occurrence of each of them in isolation, the probability of which is itself very slight.

As regards the anxiety about the storage of radioactive waste, it can be stressed that this problem is not yet acute and that adequate solution will have been found by the time they are needed.

In this connection, the Commission has laid before the Council of Ministers a European Communities' programme on the environment in which it advocates, in particular, that the following tasks be carried out jointly and in a context that embraces at least all the Community countries:

(1) definition of criteria for management and long-term storage that will ensure safety and respect for the environment;

(2) study of suitable sites for such storage;

(3) working-out of a formula for the management of storage sites and determination of responsibilities in respect of the materials stored*.

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*The Commission has undertaken to submit proposals to the Council by the end of 1973.

PART TWO

THE LONG-TERM PROSPECTS

1985-2000

The consequences of important decisions to be taken during the period covered by the Illustrative Programme will not be felt in terms of energy production until after 1985. This is particularly the case where the advanced-reactor families are concerned. Despite the extremely uncertain nature of forecasts for a period so far ahead in this field, it is worth while to start analysing the context of their commercial development straight away.

CHAPTER I : Prospects for nuclear energy

1. Electricity production

The probable trend of the demand relates the nuclear objectives to the prospect of the total electricity production doubling every ten years until the end of the century.

Consumption of electricity has, to date, only slackened off in passing phases and there is no reason to believe at the moment that the average rate of growth observed in the past will not continue. When one studies the pattern followed by demand for electricity in countries where it is higher than in the Community - particularly in the United States, where, however, consumption per head of population is almost three times as high - one also finds this stable average trend. The versatility of electricity conduces to its expansion and penetration of many fields. It can replace other sources of power, develop within a given sector find new applications and, in a general way, benefit from the spread of urbanization. It is also conceivable that the intensification of the struggle against pollution will divert to electricity the demand currently being directed to other forms of energy.

2. Nuclear energy's contribution

The gross production required from the total number of electric power plants in the Community will be in this hypothesis :

2,420,000 million kWh in 1990 and
4,840,000 million kWh in 2000

Apart from fossil fuels, which are mainly imported, and nuclear energy, no other source will be able to contribute in any significant degree to the growth in electricity production and it is a reasonable supposition that the contribution of the entire pribileged sector (hydroelectric power, lignite, gases other than natural) will flatten out at the 1985 level.

The pattern of electricity production would develop as follows (in TWh) :

	1985	1990	2000
Gross production	1740	2420	4840
Privileged sector	285	285	285
Competitive sector	1450	2135	4555
of which :			
fossil fuels	875	955	940
nuclear power	575	1180	3615

Bearing in mind the utilization conditions for each source of energy, the nuclear capacity which should be available in 1990 and 2000 in order to provide the production quoted has been estimated at 210,000 and 620,000 MWe respectively.

From 1985 to 2000, the increase in the Community's total nuclear generating capacity would thus be 520 GWe.

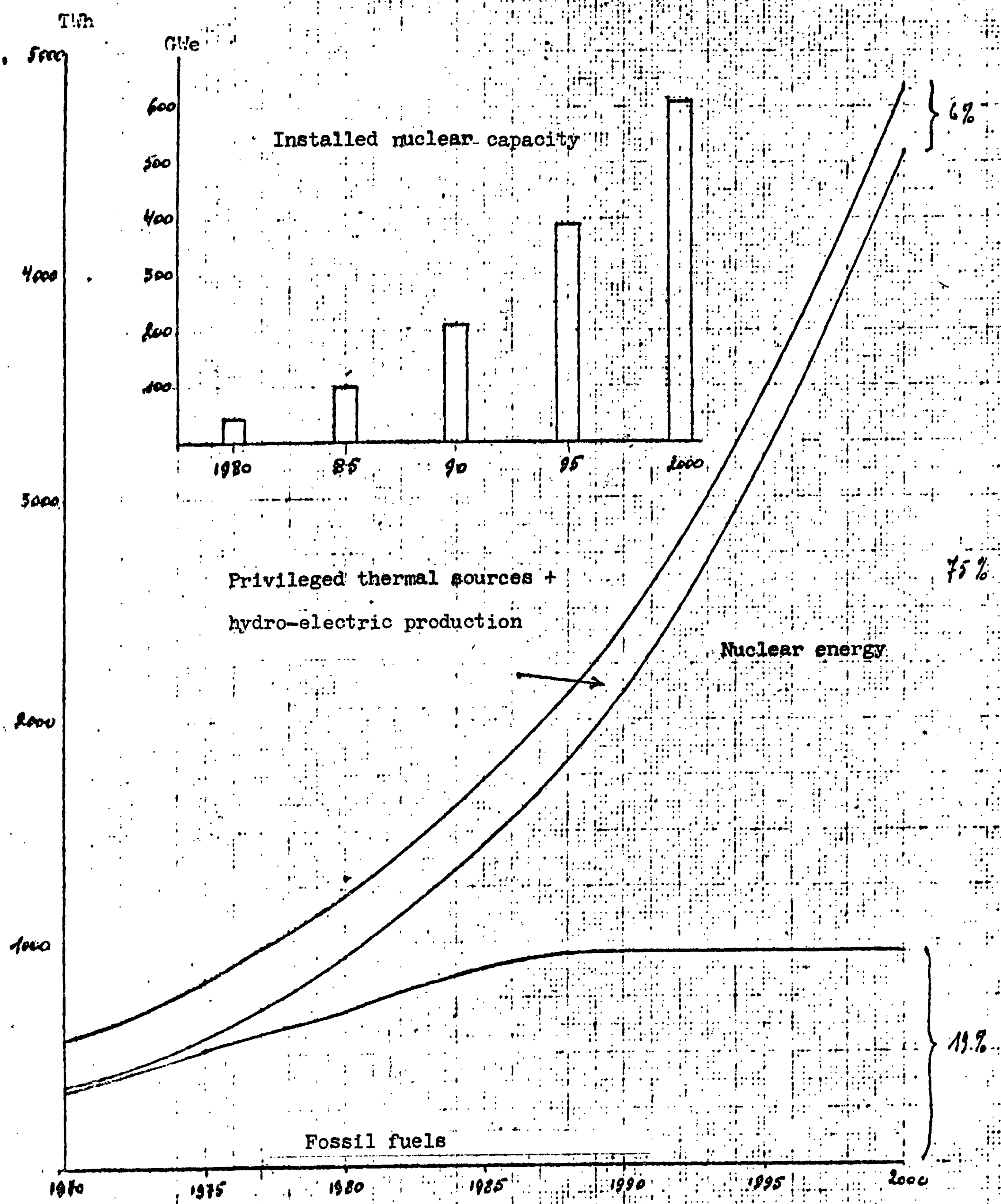
The following diagramillustrates these forecasts.

It will be noted that shortly after 1985, the contribution of fossil fuels to electricity production should rapidly decline in relative value and become fairly stable in absolute value*.

This trend indicates that, as far as electricity production is concerned, it will be 1985 before the use of (chiefly imported) fossil fuels is brought under control and can be adapted to the exigencies of the time.

*This is an overall forecast for the Community in which trends which may differ from one country to another are integrated.

Prospective trend of electricity production in the Community



CHAPTER 2: The new resources available

In 1985, the beginning of the period under review, the existing nuclear generating capacity will come mainly from light-water reactors. Among recent installations it will also include some prototypes or demonstration plants using techniques currently undergoing development, and especially those relating to fast reactors. It can also be taken that it will include high-temperature reactors for a power output which is already significant.

This raises the question of the position which will be occupied by the advanced and proven families on the market after 1985, bearing in mind the three factors by which nuclear power will help to solve the Community's problems:

(1) dependability of supply;
(2) protection of the environment;
(3) economy.

Before examining the factors governing their potential penetration, it will be useful to provide a brief outline of the prospects which the advanced families have of becoming fully developed*, it being understood that, while they enter the lists against the previous type on the basis of arguments not solely concerned with more efficient use of fissile material resources and high thermodynamic efficiency, it must be agreed that the cost curve for the electricity generated by the LWR plants is the target aimed at by their competitors. If they are able to achieve this, the nuclear market during the period following that covered by the Illustrative Programme will be indisputably open.

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*Annexes V and VI, which deal respectively with fast reactors and high-temperature reactors, describe in greater detail the status and prospects of these families.

1. Fast breeder reactors

The concept of fast reactors emerged as one of the earliest developments in nuclear energy. Characterized as it is mainly by breeding - the ability to produce more fissile material than is consumed - it immediately became the subject of widespread research.

Efforts on the fast reactor family are currently being directed chiefly to the variant using sodium cooling (SBR). Modest projects for the development of fast reactors cooled by a gas are under way - and, to a lesser extent by steam - in several countries.

In the Community, development of the fast-reactor family has gone furthest in France. Full-load operation of the Phenix 250 MWe pressure-vessel power plant is envisaged for 1973. Its construction is in the hands of a consortium comprising the CEA, EdF and GAAA. The programme family for this reactor is managed by the CEA, who are currently operating the "Fortissimo" version of the Rapsodie test reactor. Studies are being carried out on a 1000 MWe power plant, construction of which could be given the go-ahead in 1974, ie., after one year's operation of Phenix.

The joint project conducted by Germany and Benelux involves the building of a prototype 300 MWe loop-type sodium-cooled fast breeder (SNR) reactor, construction of which is due to commence in 1972 and on which there is a financing agreement between the German (70%), Belgian (15%), Dutch (15%) and Luxembourg governments. This agreement has given rise to industrial cooperation between Interatom (W. Germany), Belgonucléaire (Belgium), Neeratom (Netherlands) and Luxatome (Luxembourg). The R&D of general interest is being performed by GfK (Karlsruhe), CEN (Mol), RCN (Petten) and TNO (The Hague), as well as by certain departments of the Belgonucléaire and Luxatome companies. They are coordinated in an umbrella programme.

In addition, GfK has undertaken the study of a large fast-flux materials-testing reactor (FR3).

The Italian effort, like that of France, is being performed in a national context, its aim being the construction of a reactor (PEC) for testing fuel elements which could be used in a later, more elaborate generation of reactors than that represented at present by the SNR and Phenix prototypes. In March 1970, construction was placed in the hands of an industrial consortium comprising SNAM - Progetti and the Società Italiana Impianti under a turnkey contract. The CNEN retains responsibility for the manufacture of the core. An agreement covering exchanges of technical know-how has been concluded between the CEA and the consortium. The R&D is being carried out in the centres at Bologna and Casaccia.

On 10 May 1971, an agreement between EdF and RWE was published, to which ENEL subsequently also became a party. These electricity producers have stated that it is their intention to join forces to:

(a) build in France a nuclear power plant equipped with a sodium-cooled fast breeder reactor with a capacity of about 1000 MWe along the lines of the Phenix prototype;
(b) build in Germany a similar type of nuclear power plant, but along the lines of the SNR prototype reactor, one year after the latter has been placed in service.

Each of these projects will be pursued by subsidiaries jointly owned by the producers, the first incorporated under French and the second under German law. The breakdown of the capital among EdF, RWE and ENEL will be 51%, 16% and 33% respectively in the first-mentioned company and 16%, 51% and 33% in the second.

The Benelux electricity producers will be able to lend their cooperation in the second project.

In the United Kingdom, the experimental 60 MWth DFR has been operated by the UKAEA since 1959. The 250 MWe prototype built on the same site should be in service by the end of 1972, i.e., about a year late owing to difficulties in the manufacture of the reactor vessel top cap. Responsibility for the construction has since 1969 lain with TNPG, who took over the UKAEA team which had been assigned to this work from the outset.

Work on the first 1000 MWe power plant could begin about 1975.

In the United States, the major part of the current USAEC budget for civil reactors is earmarked for sodium-cooled fast reactors, the development of which is considered a matter of priority.

The Fermi reactor, which has been constructed and operated by a private company (PRDC), was designed for a thermal output of 430 MWth (150 MWe), but has only been operated at loads up to 200 MWth. Following damage during the experiments involved in the run-up to power in 1966, this reactor has been repaired and placed back in operation.

In the field of experimental reactors, the EBR-2 (62.5 MWth) reactor has been used by the USAEC since 1961; General Electric has built and, in collaboration with the USAEC and the German Karlsruhe centre, operated the SEFOR 20 MWth experimental reactor*. Westinghouse is in charge of the construction of the large 400 MWth FFTF test reactor, which should enter service in 1974.

Three constructors, namely Atomics International, Westinghouse and General Electric, each of which is associated with a group of electricity producers, have submitted proposals to the USAEC as part of the project definition phase of the American programme. The USAEC has stated its intention to promote the construction of two demonstration plants, each with a capacity of 300-500 MWe. Work on the first of these will start at the end of 1972 or the beginning of 1973.

*This will shortly be dismantled, following the success of its experimental programme.

In the USSR the BOR (60 MWth) test reactor went critical in 1968 and the BN 350 (1000 MWth), plant, construction of which was completed at the end of 1971, should enter service in 1972; it will have a capacity of 150 MWe and produce 100,000 tons of fresh water a day. In addition, a 600 MWe power plant (BN 600) is under construction and should be commissioned in 1975 or 1976.

Finally, in Japan the programme is advancing rapidly. The construction of a 100 MWth Rapsodie-type experimental reactor is under way and commissioning is planned for 1973-74. Design work has commenced on a 300 MWe power plant, construction of which should be completed in 1977 or 1978.

Future outlook

The solution of the technological problems governing theuse of fast reactors depends to a great extent upon the efficiency of the organization which the industry sets up at short notice in order to undertake the large-scale construction projects involving demonstration plants of the order of 1000 MWe currently envisaged. Great efforts must be made on the part of both constructors and operators to adapt, if necessary, to an advanced concept based on very sophisticated standards. To this end the industry is able to draw on the vast R&D programme now under way in the research installations.

The main problems lie in the develppment of the steam generators (sodium/water reaction), the fuel and the core as a whole, where the phenomena linked with fast neutron flux (swelling of structural materials) are intensified by the need for high burnups. There also remain safety problems, the economic solutions to which still have to be developed. In this respect the experience gained with the prototypes will be decisive.

From an economic point of view it is admittedly likely that, initially, the specific investment cost of breeder reactors will be higher than that of light-water reactors, mainly because of the implications of sodium technology and the presence of an intermediate circuit. However, the additional cost would be largely offset by the lower cost of the fuel cycle due to the breeding of fissile material. All in all, when the technique has been fully mastered, the cost per kWh should be lower than that for light-water reactors. In addition, the cost of the fuel cycle, which already does not depend much on the cost of the feed material, will undergo a further reduction due to the increase in the breeding rate stemming from the use of carbide fuels and will then, for practical purposes, become indifferent to fluctuations in the price of uranium.

However, it must be expected that this reactor type will only be competitive where unit sizes are very large.

Added to these overall economic advantages are those of improved utilization of the plutonium produced by the light-water family and greater independence of nuclear power plants in relation to enrichment installations.

In the most advanced countries the commissioning of commercial power plants is currently planned for about 1985. However, this could be delayed until after 1990 if the present estimates concerning construction and fuel cycle costs proved to be too optimistic.

2. High-temperature gas reactors

In the industrialized countries there is a growing interest in the high-temperature gas reactors. This is largely due to following characteristics, which were adopted in the original designs:

1. A helium coolant temperature of the order of 750°C, thus facilitating the use of the latest types of steam turbine while reducing the thermal effects on the environment as compared with light-water reactors, owing to their excellent thermal efficiency (about 40%).

2. High-rating fuel elements which achieve high burnups.

3. A good neutron economy stemming from the use of graphite as the moderator, cladding and scattering agent.

The experience gained in the development of coated-particle fuels and of primary circuits integrated in prestressed-concrete pressure vessels gives high-temperature gas reactors the added attraction of intrinsic safety. By drawing on this experience, constructors are now able to tender for power plants.

The family consists of two reactor variants using different fuel elements: these are in the form of either spherical or prismatic elements and use uranium which is either highly enriched to 93% in the case of the uranium/thorium cycle or only slightly enriched (about 5%) in the case of the uranium/plutonium cycle.

In the United States, Gulf General Atomic received letters of intent between September 1971 and July 1972 relating to six high-power reactors.

This interest derives from the experience acquired in the development of reactors using prismatic fuel elements and the thorium cycle which has been pursued for several years now by GGA with USAEC backing. This experience has been turned to practical account mainly in the full power operation of the 40 MWe Peach Bottom reactor since 1967 and in the construction of the 330 MWe prototype reactor at Fort Saint Vrain, for which the power run-up is imminent.

In the UK, as a result of a call for bids by the CEGB, the two consortia TNPG and BNDC have submitted a preliminary proposal, together with a draft research programme, for a low-enrichment, prismatic fuel element reactor with a capacity of over 600 MWe. The decision to build is not expected before the end of 1972.

In Germany, a 300 MWe pebble-bed type of power plant for the Hochtemperatur-Kernkraftwerk GmBH (HKG) has been under construction at Schmehausen since 1970 by a consortium consisting primarily of Brown Boveri and Nukem. Commercial operation should begin in 1976. The design of this plant is based on the experience gained with the 15 MWe AVR reactor, which has been operating on load since 1967, and is the outcome of the work carried out between 1963 and 1968 by the THTR Association, in which the Community participated.

Additionally, the EURO-HKG company was formed on 13 December 1971 by the main electricity producers in the Community and the United Kingdom with the aim of acquiring and pooling technical and economic knowhow in the field of high-temperature reactors and of arranging exchanges of personnel for training purposes.

In France the CEA is currently carrying out an analytical study of the HTGR type with the assistance of a group of industrial companies.

Government and industrial circles, especially in the UK, Germany and France, view with favour international collaboration aimed at introducing this type on an industrial scale.

Finally, the Community is participating in the Dragon programme, which concerns in particular the operation of the 20 MWth Dragon experimental reactor and the development of designs for power reactors using prismatic elements.

Prospects for the steam-cycle type

In the United States the development of high-temperature gas reactors seems economically justified by the fact that they are now competing with the light-water reactors. In addition, this type of reactor offers the advantage of a lower uranium consumption in meeting increasing energy requirements.

In Europe the choice of fuel cycle still remains open. In accordance with the stimulus provided by the advocates of the different solutions, the choice could be made between low enrichment, which has been studied by several European industries to date, and high enrichment, as adopted by Gulf General Atomic. The latter case would entail the creation of a thorium industry in Europe.

The earliest date by which it would be possible to put a total installed capacity of the order of 2000 MWe into operation on a commercial basis would be 1980.

The direct cycle (helium turbines)

It is possible to link high-temperature gas reactors directly to a helium turbine. The consequent advantages would be:

(a) reduced requirements for cooling water;
(b) greater efficiency, even during low-power operation;
(c) possible lower specific investment costs.

This improvement to the family necessitates a programme for the development of fuels and graphite capable of withstanding temperatures of about 1000°C at the reactor outlet. It will also be necessary to develop heat insulators, structural materials (ducting, turbine blades), valves, etc. with the aid of test rigs under helium at this same temperature.

In Germany, certain companies, in collaboration with KFA Jülich, have undertaken an R&D programme in order to solve these specific problems. Also, certain additional prosposals are being negotiated with the German government so that a programme may be drawn up which would lead to the submission of commercial bids around the end of the decade.

Applications not relating to the generation of electricity

The recent major rise in petroleum prices could cause a swing towards energy produced by nuclear means. With helium outlet temperatures of 900-1100°C, high-temperature reactors would constitute sources of heat suitable for steelmaking, petrochemicals and the heavy chemical industry. Applications such as these could be contemplated after 1980.

Several studies have already been carried out on a Community sacle, involving opeperation between nuclear engineering companies and university and private research centres. The additional problem raised, as compared with direct-cycle techniques, stems from the presence of greater quantities of hydrogen in the primary circuit following its diffusion across the heat exchanger walls.

At the national level, KFA Jülich is currently examining various reactors in the 500-3000 MWth range in collaboration with various German industrial groups and research institutes, with the aim of generating steam or linking the reactor to an installation producing hydrogen via the conversion of fossil fuels.

In the United States, Gulf General Atomic, in collaboration with Stone and Webster, was recently awarded a study contract by the State of Oklahoma with the object of adapting an HTGR to a coal gasification plant.

Japan has also shown an interest in the use of this reactor type as a source of heat for industrial uses notably in steelmaking and the construction of a multipurpose 50 MWth reactor is planned.

CHAPTER III : Breakdown of the market by reactor types

In the field of electricity production the growth of the nuclear market will lead to at least a sixfold increase in output by the year 2000 over the figure for 1985. This means that nuclear power plants totalling at least 520,000 MWe will be installed during this period.

Obviously, these estimates are looking far ahead and thus, as regards the long-term prospects, the field is still wide open. In any case, the technical and economic features of the reactor types put into service beyond 1985 are still too much of an unknown quantity for meaningful lines of demarcation to be drawn.

For all this, the analysis of future market conditions and of the trends governing the options and determining this breakdown must continue unabated, so that all subsequent decisions can be taken on the basis of comprehensive background data.

In this context it is certain that dependability of supplies and environmental considerations will continue to play a major role. Owing to the sacle of the procurement programme, based on light-water reactors, carried out prior to 1985, later choices will fall upon advanced reactor designs which meet the above requirements and also enable electricity to be produced under conditions which are at least as favourable as those offered by light-water reactors.

The influence which the size of the nuclear market will exert on the breakdown between reactor types is difficult to assess. But, according to the evidence, the market will be all the more attractive the larger its volume. On the other hand, insofar as the breakdown is based on genuine competition, the influence will probably be mutual, the market being more voluminous the more lively the competition. By the same token, the position of nuclear energy with respect to the other primary sources of energy would become still more dominant and, above all, the competitive position occupied by electricity should be consolidated.

1. <u>An initial outline of a breakdown by reactor types</u>

Unlike light-water reactors, the conversion factor of which is about 0.5, fast reactors - and in particular those cooled by sodium (SBR), which could reach maturity by the beginning of the period under consideration - produce more fissile material than they consume.

Because of this ability, SBRs can utilize 50-80% of the energy contained in natural uranium, whereas LWRs only manage to extract about 1%. Under these conditions it is conceivable that an initial breakdown between reactor types beyond 1985 and probably before 1990 could be based on a coupling of LWRs and SBRs which would minimize the overall cost of the energy produced while providing a considerable fillip to the dependability of supplies via tthe optimum utilization of the available resources of fissile and fertile materials.

Within this two-family system, the accurate and detailed estimation of how the increases in capacity to go into service between 1985 and 2000 will break down between LWRs and SBRs is as hazardous now as it was when the First Illustrative Programme was drawn up. While it has been consistently proved that plutonium will constitute a key factor favouring the development of nuclear power, it is still impossible to quantify in any better way all the variables involved in a model capable of satisfactory simulating the mechanism of medium and long-term supply and demand.

At all events, it is appropriate to raise questions about the guarantees offered by a development model based on two types of reactor.

It is necessary to underline the hazardous nature of this model, which depends on the success of the SBR at both the technical and the economic level. Despite the universal interest shown in this family of reactors since 1944, the forecasts have yet to be confirmed by full-scale industrial experience, which, however, will probably not be acquired before 1985; in the meantime it remains to complete the construction of the prototypes and to build the demonstration plants. Not until these plantssare in operation will it be possible to evaluate with any accuracy the role which SBRs could eventually play.

If the development of SBRs were to end in failure, it might be considered risky in the medium and longer term to assign to LWRs alone the task of promoting nuclear energy to the leading role which it must play in the energy sector as a whole.

Briefly, in a situation in which LWRs represent an initial crystallization of the nuclear techniques which can be exploited economically until 1985 at least, and in which SBRs still offer no more than a hope of ensuring dependability of supplies by taking economic advantage of the breeding process, the question arises whether it is reasonable to base the success of nuclear energy on a system made up solely of these two reactor types.

2. Second outline

The reply to this question obviously depends on the part the high-temperature reactors (HTR) are likely to play as compared with both the LWRs and the SBRs.

It is important to note that the HTRs offer a double appeal in that, in the form of first-generation thermal-neutron reactors (HGTR), they can compete in the medium term with LWRs before the SBRs are able to do so themselves, while in the longer term, in the form of second-generation fast-neutron reactors (GBR), they could prove capable of competing with the SBRs.

In other words, alongside two nuclear techniques which are likely to dovetail smoothly (LWR and SBR), there is a third (HTR) which can compete with both, while also opening up good prospects for the use of nuclear heat for purposes unrelated to the production of electricity e.g., steelmaking, manufacture of chemicals, refining of fossil fuels, etc.

The foregoing arguments in favour of development based on three reactor systems are of course, not in themselves sufficient to justify its necessity.

However desirable Community action to promote such a cause may be, the financial resources and the means required for the development of the HTRs must also be made available, over and above those required for the development of the SBRs, without jeopardizing the equally high-priority effort involved in industrial development work on LWRs.

This indicates the true magnitude of the problems which all the bodies concerned in the Community are being called upon to help in solving, not only in order to enable the nuclear energy production aims recommended by the present Programme for 1972-85 to be achieved, but also, during this same period, to provide it with the new means of production upon which its longer-term future depends to an equally decisive extent.

It is important to note that at a time when the Community is faced with a number of decisions to be taken in this field, its enlargement has in fact become an imminent reality.

This is why, on 20 December 1971, the Council of Ministers took the decision to allow the United Kingdom to participate in the work of the "Coordinating Committee on Fast Reactors" forthwith, because of the extent of its programme concerning the SBRs, and to inform the other three countries applying for membership. This Committee will thus be in a position to point up the beneficial effects of the enlargement of the Community on the development of SBRs and on the prospect of gaining the maximum benefit at the lowest cost from a wider market with an eye also to the export markets offering themselves to a European industry which is of the size necessary for world trade.

The consideration being given to the HTR family within this enlarged framework has already found concrete expression in the fact that, on 13 December 1971, the principal Community electricity producers and the CEGB, together with the HKG company, which is responsible for building the prototype 300 MWe THTR (Thorium-Hochtemperaturreaktor) at Schmehausen, decided to set up a company to be known as Euro-HKG. It is to be hoped that an agreement of this kind will be the prelude to a rationalization of the decisions taken by the electricity producers concerning HTRs and will encourage mutual consultation between constructors on the main choices which will guide the development of this reactor system.

APPENDIX

Nuclear power stations installed, under construction and planned in the Community

Position as at 15 June 1972

1. Net electrical capacity of nuclear generating plants in service, under construction or planned: 28,302 MWe net, broken down as follows:

	Country	In service	under const.	on order or planned	Total MWe
a) Proven-type reactors Gas/graphite					
Chinon 1 / Loire (EDF)	F	70	-	-	70
Chinon 2 / Loire (EDF)	F	200	-	-	200
Chinon 3 / Loire (EDF)	F	480	-	-	480
St. Laurent 1 / Loire (EDF)	F	480	-	-	480
St. Laurent 2 / Loire (EDF)	F	515	-	-	515
Bugey 1 / Rhône (EDF)	F	540	-	-	540
G 2 Marcoule / Rhône	F	40	-	-	40
G 3 Marcoule / Rhône	F	40	-	-	40
ENEL (Latina)[1]	I	200	-	-	200
		2565	-	-	2565
Boiling water					
KRB (RWE/BW) Gundremmingen	G	237	-	-	237
KWL (VEW) Lingen [2]	G	174	-	-	174
VAK (RWE/BayernW) Kahl	G	15	-	-	15
ENEL (Garigliano)	I	150	-	-	150
GKN ((Dodewaard)	N	52	-	-	52
KKW (Preag) Wurgassen, Weser	G	640	-	-	640
KKB (HEW/NWK) Brunsbüttel	G	-	770	-	770
ENEL 4 (Caorso)	I	-	783	-	783
KKP 1 (Badenw/EVS) Philippsburg	G	-	860	-	860
KKP 2 (Badenw/EVS) Philippsburg	G	-	-	860	860
KKI (Bayernw/IsarAmperW) Ohu, Isar	G	-	-	870	870
KKW (HEW/NWK) Krummel, Elbe	G	-	-	1260	1260
		1268	2413	2990	6671

[1] Owing to a permanent non-availability the effective capacity is 153 MWe

[2] Excluding natural-gas superheat

Pressurized-water	Country	In service	under constr.	On order or planned	Total
KWO Obrigheim Neckar	G	328	-	-	328
SENA (Chooz)[1]	F	270	-	-	270
ENEL (Trino Vercellese)	I	247	-	-	247
BR 2 (Mol)	B	10	-	-	10
KKS (NWK + HEW) Stadersand/Elbe	G	630	-	-	630
S.E.M.O. (Tihange /Meuse)[2]	B	-	870	-	870
Centr.Nucl.de Doel(Doel/Scheldt)	B	-	780	-	780
PZEM (Borssele)	N	-	450	-	450
KKB 1 (RWE) Biblis /Rhine	G	-	1146	-	1146
KKB 2 (RWE) Biblis /Rhine	G	-	1178	-	1178
Fessenheim I (Rhin EDF)[3]	F	-	890	-	890
Fessenheim II (Rhin EDF	F	-	-	890	890
KKU (Preag/NWK) Esenshamm	G	-	-	1230	1230
GKN (NeckarW, TWS, DB) Neckarwestheim	G	-	775	-	775
Bugey 2 (EDF)	F	-	-	925	925
Bugey 3 (EDF)	F	-	-	925	925
		1485	6089	3970	11544
b) Advanced converters					
Heavy water					
MZFR (Karlsruhe)	G	51	-	-	51
KKN (Niederaichbach-Isar)	G	--	100	-	100
EL 4 (Monts d'Arrée)	F	70	-	-	70
CIRENE (Latina)	I	-	-	32	32
High temperature					
HKG (Schmehausen)	G	-	300	-	300
AVR (Jülich)	G	13	-	-	13
Sodium/zirconium hydride					
KNK (Karlsruhe)	G	19	-	-	19
Nuclear superheat					
HDR (Grosswelzheim / Main)	G	22	-	-	22
		175	400	32	607
c) Fast breeders					
Phenix (Marcoule)	F	-	233	-	233
SNR 300 (Kalkar/Rhine)[4]	G	-	-	282	282
		-	233	282	515

[1] Franco-Belgian (50/50) power plant

[2] 50% French (EDF) participation

[3] 30% Swiss participation

[4] German (70%)/Benelux consortium

d) Type not yet decided	Country	In service	under constr.	On order or planned	Total
BASF 1 (Ludwigshafen)	G			400	400
Grafenrheinfeld (BayernW/...)				record only	
Grosswelzheim/Gundremm. (RWE)	G			1200	1200
Badbreisig/Mulh.Kärlich (RWE)	G			1200	1200
Biblis III (RWE)	G			1200	1200
KBR -1 Breisach/Rhine(BW/EVS)	G			1200	1200
Schmehausen/Lingen (VEW)	G			p.m.	p.m.
Slechen Rosenheim (BW/IAW)	G			p.m.	p.m.
GKN (Borsele/Maasvlakte)	N			600	600
Enel 5	I			600	600
				6400	6400

2. Type breakdown of reactors installed or under construction (%)

Gas/graphite	2565 MWe	(17.5 %)
Boiling-light-water	3681 MWe	(25.2 %)
Pressurized-light-water	7574 MWe	(51.8 %)
Heavy-water	221 MWe	(1.5 %)
High-temperature	313 MWe	(2.1 %)
Other advanced converters	41 MWe	(0.3 %)
Fast breeders	233 MWe	(1.6 %)
	14628 MWe	(100 %)

3. Breakdown by stage of completion and country of location

	West Germany	France	Italy	Nether-lands	Belgium	Community
Reactor installed	2129	2705	597	52	10	5493
Reactor under constr.	5129	1123	783	450	1650	9135
	7258	3828	1380	502	1660	14628
Reactor on order or planned	9702	2740	632	600	-	13674
	16960	6568	2012	1102	1660	28302

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COMMISSION
OF THE
EUROPEAN COMMUNITIES

SECOND TARGET NUCLEAR PROGRAMME FOR THE COMMUNITY

ANNEX I

ACTIVITIES OF THE COMMISSION OF THE EUROPEAN COMMUNITIES IN THE FIELD OF RADIATION PROTECTION

1st July 1972

ANNEX I

ACTIVITIES OF THE COMMISSION OF THE EUROPEAN COMMUNITIES IN THE FIELD OF RADIATION PROTECTION

One of the tasks of the European Atomic Energy Community (Euratom), which was set up under the Treaty of Rome in 1957, is to lay down adequate conditions for the protection of the health of workers and the general public against ionizing radiation. In Article 2 of the Treaty, the Community is obliged in particular to "establish uniform safety standards to protect the health of workers and of the general public and ensure that they are applied".

CHAPTER III of the Treaty (Articles 30-39) states how these safety standards are laid down and what are the powers and obligations of the Commission regarding the overseeing of the provisions to ensure compliance with these standards and of the surveillance of the radioactive contamination of the environment.

The present document summarizes the activities and results achieved by Euratom in this field, but it should be pointed out right at the outset how the radioactive risk is generally estimated, on what principles the establishment of standards concerning the protection of workers and the population are based, and what significance should be assigned to the limiting values adopted for human exposure and the contamination of the environment.

1. Estimation of the radioactive risk (1)

The majority of estimates of the risks to mankind arising from exposure to radiation are based on observation of individuals or of sections of the population who have been exposed to heavy doses of radiation for relatively short periods of time as a result of accidents, war or medical treatment.

As regards low doses of ionizing radiation, only animals experiments have provided data which have been extrapolated to human conditions and these are still inconclusive and incomplete.

This is trie not only of the somatic effects (and especially of leukemia) but in particular of the genetic effects, and here it has been impossible to detect with any certainty genetic damage resulting from these doses received by man, despite considerable scientific efforts.

With regard to the somatic effects, the studies carried out on the Japanese survivors of Hiroschima and Nagasaki and on certain groups of patients who were exposed for medical reasons) showed that in the case of doses of more than 100 rad there was a greater incidence of leukemia during the 15-20 years following exposure; therefore, at high levels of exposure there is a proportional relationship between dose and biological effect.

An important point to be noted is that diseases such as cancer are not specific to ionizing radiation and occur naturally among the population. The only way of demonstrating the effect of radiation is to carry out a comparison on a valid statistical basis between a non-exposed population group and an exposed group and to examine the extent to which there is a significant increase in the number of cases of the disease under examination in the second group. In the case of the levels of exposure encountered under normal conditions in the pursuit of a nuclear activity, the link between the dose and biological effect ceases to exist and thus makes it virtually impossible to establish a quantitative relationship with any accuracy.

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(1) The basic standards which are laid down by Euratom pursuant to Chapter III of the Traty and are aligned with those proposed by the International Commission on Radiological Protection relate in particular to maximum permissible doses. These doses are expressed in rads or rems; 1 rad corresponds to the absorption of 100 erg/g by the substance under consideration. For radiological purposes the dose is expressed in rems; the figure in rems is equal to the dose in rads multiplied by a quality factor which takes into account the biological effect which, in turn, varies according to the type of radiation in question. For information : the dose due to natural background radiation is about 0.125 rem/year in Europe. The basic standards lay down 5 rem/year as the maximum permissible dose for the most exposed workers, and 0.5 rem/year for the public as a whole.

Radioactivity is expressed in curies (Ci); 37.10^9 disintegrations/sec occur in the quantity of a substance corresponding to 1 Ci.

However, under a double heading of simplification and caution the assumption has been maintained that there is no threshold as such; also, the existence of a linear relationship between dose and effect has been upheld with a view to defining an upper or maximum limit to the risk of somatic effects. The statistical evaluations arrived at via this method must be treated with caution, since it must be borne in mind how and under what circumstances they were arrived at. The confusion between risk and effect is frequent; what is envisaged is an increase in the probability of an increase manifesting itself in a certain disease within the group exposed.

As regards the overall genetic effects, it has been known for 40 years as a result of animal experiments that irradiation is capable of causing genetic mutations. There is no human evidence, even in the descendants of Japanese parents exposed in 1945, of significant genetic damage linked with ionizing radiation. Since animal experiments have yielded incontrovertible evidence, geneticists have upheld the theory of the existence of linear relationship between dose and effect and of the absence of a damage threshold. Nevertheless there might well be a recovery process in the case of low-dose chronic exposure.

It should be pointed out here that the UNSCEAR (United Nations Scientific Committee on the Effects of Atomic Radiation) expressed an opinion on this matter back in 1964[1] by suggesting a linear non-threshold model, for both the genetic and the somatic effects of radiation: "It must be emphasized that the estimates of risk are reliable only in the range of doses, usually high, for which information is available. The use of these estimates for doses outside the observed range may be very much in error, and in the low dose range, where a linear extrapolation to zero dose is used, it can in most cases only be taken as an indication of the upper limit of risk. Thus the linear non-threshold model as used in radiation protection is intended to represent only the upper limit of risk, not the risk, and the true risk can be presumed only to lie somewhere between zero and the value emerging from the linear non-threshold model".

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[1]Report to the General Assembly; 19th session, supplement No. 14 (A/5814; United Nations New York (1964).

These various considerations thus suggest that it is not easy to obtain a mathematical representation which defines the total risk, at acceptable levels, to the population of the peaceful uses of nuclear energy (and in particular nuclear power reactors). It is commonly acknowledged that this risk is very low in the present state of the art and in the light of existing prevention and safety techniques.

In addition, no accident affecting the public has ever occurred to date in a nuclear power plant - obviously a fortunate state of affairs - in contrast with other industrial activities, where a more accurate estimate of the magnitude of the risk is possible as a result of the experience acquired from accidents.

2. Principles and Methods of Radiation Protection

2.1 The International Commission on Radiological Protection (ICRP) is the organization which, since 1928 has defined at a scientific level the principles, concepts and methods in the form of recommendations which must be adopted at a national or international level with respect to the risks involved in ionizing radiations.

A certain number of concepts, which were also drawn on in the establishment of the Euratom Basic Standards, appear in ICRP Publication No. 9 (containing the recommendations which were adopted in 1965 and are still in force). The ICRP, basing itself on the assumption that any exposure to radiation can involve a certain risk of somatic and genetic effects, accepts the linear model which implies that there could be a damaging effect, even at the lowest levels of exposure. As stated above, such an assumption is dictated by caution, since certain effects could only emerge above a minimum dose (or threshold), but the ICRP feels that "the policy of accepting a risk of damage at low doses constitutes the most reasonable basis for protection against radiation". However, it also says: "Unless man wishes to give up the activities involving exposure to ionizing radiation, he must accept that there is always a certain risk and limit the radiation dose to a level at which the risk run can be deemed acceptable in view of the advantages gained".

The concept of an acceptable risk has been readily accepted by man, who considers it to be the price of progress. The radiation protection standards defining the magnitude of this risk represent a compromise between two apparently contradictory aims, namely, to promote the essential peaceful uses of the atom and to eliminate any risk of exposure to radiation.

The ultimate aims of radiation protection are to give advance warning of acute exposure to radiation and to limit the risk of delayed effects to an acceptable level.

2.2 The ICRP "recommends the avoidance of all useless exposure and the restriction of all doses to the lowest values which can be achieved without difficulty, due consideration being given to the social and economic aspects".

The application of this principle has provided the nuclear industry with a level of safety attained in no other industry.

Right from the outset the idea that the doses involved in any nuclear activity should be established accurately before that activity is engaged upon and that once under way it should be subjected to close and continuous surveillance has been applied with remarkable success. It is customary to consider that work involving the risk of exposure to radioactivity can only be performed if the standards are laid down in advance and complied with throughout the entire sequence of operations.

2.3 The prevention of accidental exposure to radiation begins early on

In order to be effective it must be factored into the design of nuclear installations, when the probability of the risk of an accident must already be taken into account and reduced to the lowest level which is compatible with the technical requirements of the planned installation.

From numerous standpoints the prevention of accidental exposure to radiation follows the same rules as the prevention of other types of industrial accident. The precautions are such that an accident could not be caused by one simple error, but by a combination of several elements relating to working procedures, environmental conditions and human factors.

2.4 The prevention of chronic exposure is based on observance of maximum permissible levels for exposure doses and internal contamination. These levels act as guidelines for the planning of installations, the selection of working methods, the drafting of safety instructions and, in general terms, the practical organization of radiological protection.

2.5 Maximum permissible dose is defined as the dose which, at the present state of the art, is unlikely to cause particular trouble to the individual during his life. Maximum permissible doses are laid down for workers.

One-tenth of the maximum permissible doses laid down for exposed workers has been fixed as the maximum permissible dose for members of the general public. The factor of ten has no formal significance from a biological standpoint, but it appeared necessary for purposes of operational or nuclear installation planning, in order to fix an upper limit to which certain members of the public could be exposed with a view, in particular, to setting maximum levels of radioactive waste discharge into the environment.

A third limit has been fixed. This concerns what it has been decided to call the genetic population dose. It deals with the possible hereditary effects among the population in general. This dose can never exceed 5 rem/30 years and is added to the exposure due to natural radioactivity, which is of this order of magnitude, and to medical treatment.

2.6 It should be borne in mind that the limiting levels of exposure or contamination are not exact lines of demarcation between dangerous and harmless doses; rather they constitute guidelines which help to eliminate or reduce the risk of damage to health. Such levels cannot just be left to the judgment of the workers or employers: surveillance must be placed in the hands of experienced, qualified staff who are able to interpret the results or apply them with a full knowledge of the case.

The surveillance of nuclear installations, to ensure the health of the workers, is based on (a) physical and chemical and (b) biological and medical methods. The former perform what it has been agreed to call physical radiation surveillance, i.e., all of the measurements and readings of exposure and contamination. The latter are included under the head of the medical surveillance of workers, which examines how the individual adapts to his work and how his state of health develops in accordance with the contamination to which he may be subjected.

2.7 Regulations also guarantee training and information and in particular ensure that the nuclear worker is informed of the risks to which he may be exposed as well as dealing with the regular training and information of supervisory staff with a view to creating and maintaining an atmosphere of safety.

3. Activities of the Commission of the European Communities

Following on from the brief review above of the general principles underlying the policy of limiting and monitoring the radioactive risk, the present chapter summarizes the development of the Euratom regulations and their application in the six nations of the Community.

3.1 The Commission, the executive arm of Euratom, had conferred upon it by the Treaty a number of specific areas of responsibility with respect to health protection as a result of which it has been able to conduct large-scale activities in this field. In accordance with Article 2 of the Treaty, these activities are aimed above all at the alignment of health protection standards throughout the Community, in order to avoid any discrimination based on the nationality of the worker or the company employing him.

As is stated in Chapter III, the Treaty provides for the establishment of a system of Basic Standards to this end (fundamental and practical protection standards) in such a way that they may "be applied as such without any additional safety coefficient". These Basic Standards lay down:

1. The maximum permissible doses commensurate with adequate safety;
2. The maximum levels of exposure and contamination;
3. The basic principles underlying the medical surveillance of workers.

These standards are in conformity with the recommendations of the International Commission on Radiological Protection (ICRP), which are based on the principles outlined in Section 2.1 above.

They were drawn up by the Commission in consultation with a group of persons appointed by the Scientific and Technical Committee from among scientific experts in the Member States, and in particular from among experts in the field of public health.

The Commission requests the opinion of the Economic and Social Committee on the Basic Standards thus drawn up. After consulting the Assembly, the Council acting on a qualified majority vote on the proposal put forward by the Commission, which records the opinions of the Committees and conveys them to the Council, lays down Basic Standards in the form of directives to the Member States. They can be amended and supplemented in accordance with the procedure recently instituted after their introduction in stages on 2 February 1959, 5 March 1962 and finally 27 October 1966.

The Treaty points out that the legal and regulatory measures intended to ensure compliance with these standards come under the jurisdiction of the Member States, but the Commission can verify their conformity with the standards and is empowered to make recommendations with a view to bringing about the harmonization of these provisions. Euratom thus occupies a special place among international institutions, since it possesses precise, incontrovertible means of action as regards radiological protection.

3.2 The inventory of the provisions in existence before the adoption of the Euratom Treaty in the six countries of the Community, and then the application of those provisions have shown that it is possible to map out a common policy on protection in six countries having different legal and administrative structures. Emphasis should be given to the importance of this situation in relation to the international legislation governing health protection. The overall legal instrument which the Commission has submitted to the states has stimulated national initiatives, interministerial coordination, legislative and regulatory amendments and, in some cases, the promulgation of new legal texts. It would not have been possible to have implemented all of these initiatives in such a short time and with such significant results without the obligation imposed on the Member States of applying the basic standards.

The principle that the States must give the Commission prior notice of activities involving ionizing radiation is embodied in the basic standards and represents one of the essential components of any action aimed at limiting exposure to radiation and ensuring compliance with the standards. Germany, Belgium, France and Italy have fairly clear-cut legislation on these points and a special procedure has been set up for nuclear installations likely to constitute a serious risk to the environment. Scientific commissions composed of experts representing the various ministries and departments involved are consulted before a final decision is taken. The responsibility for issuing the authorization to build and operate remains with the competent national authorities.

3.3 The basic standards also specify the compilation of a medical file which is to be kept up to date on each worker and held in the archives throughout the lifetime of the person concerned, and in any case for at least thirty years after the conclusion of the work exposing him to ionizing radiation. This file must above all contain the individual doses received by the worker and the results of medical examinations. The Member States must take practical steps to ensure that the medical file for each worker is kept regularly up to date and also see to it that all useful information concerning the places worked at by the worker and the doses received by him is passed on within the Community. This provision is aimed at facilitating the practical application of the Community principle of the free mobility of labour.

3.4 In accordance with the basic standards, the Member States of the Community must subject activities involving a risk of ionizing radiation to a system of prior declaration or authorization. If suitable procedures are adopted, the necessary guarantees regarding prevention and protection are provided before the activities are undertaken. Regular checks enable the competent authorities to ensure that the conditions laid down as regards operation are fulfilled and that the levels of exposure at no time exceed the prescribed levels.

The Euratom Treaty also stipulates that the Member State must set up installation enabling the levels of radioactivity in the atmosphere, water and the soil to be continuously monitored with a view to ensuring compliance with the basic standards. Information concerning this monitoring must be passed on to the Commission, which also has the right of access to the installations, the operation and efficiency of which it can then verify.

This Community surveillance of compliance with the standards through the national installations is one of the most original aspects of the radiological protection policy which the Commission needs to carry through successfully, and the outcome of which rests in the power of recommendation and even of direction which the Commission could exercise each time that the basic standards are exceeded or the regulations not observed. Hitherto the Commission has never had to use this power, but it is included in the Treaty and confers upon Euratom a real responsibility for the surveillance of radioactivity which is likely to have an effect on the health of the population.

3.5 The radioactive risk knows no frontiers; it is thus normal for the special problem of radioactive waste to have been the subject of precise measures embodied in an article in the Treaty (Article 37) which obliges the Member States to pass on to the Commission general data on any planned discharge of radioactive waste, regardless of its form, in order to establish whether it is likely to give rise to radioactive contamination of the water, soil or the airspace of neighbouring Member States.

The Commission has laid down the essential aspects of the general data which must be supplied by the Member States. The method used to evaluate the risk of contamination to neighbouring States is based not only on an appreciation of normal operating conditions, but also on exceptional or accidental discharges.

Since 1959 about fifty opinions have been issued by the Commission on planned discharges of radioactive waste from various nuclear installations in the Member States. Article 37 of the Treaty constitutes an important element in a Community policy on the environment.

In future the application of Article 37 will assume particular importance owing to the expansion of nuclear energy and radioecology will be called upon to play a decisive part in the analysis of the sites or areas in which nuclear power plants are set up. An objective assessment of the likely hazard to man and the environment due to the nuclear activities envisaged will be based on ecological considerations.

It would be possible right now to undertake more exhaustive ecological studies than those currently under way, which would bear on the future outlook for a forecast of nuclear expansion.

Current methodology adopted in the matter of radioecology is well-known and based on internationally accepted concepts, such as radiological absorption capacity, the critical transfer path and the critical population group.

3.6 Any regulatory or technical action must be backed up by a research programme. Euratom has appreciated this very well, since for more than twelve years now the Commission and the Member States have been implementing short, medium and long term programmes of studies and research aimed at extending the existing knowledge of the effects of ionizing radiation on man and the environment.

The latest five-year programme, which was approved by the Council of Ministers in 1970, include among its prime objectives a study of the biological effects of the low-dose radiation received and the development of knowhow and methods enabling the radioactive contamination of man and the environment to be analysed with a view to quantifying the radioactive risk more accurately. This programme is closely linked with the aims of radiation protection and will help to verify or update present health protection standards.

The principal route by which man is subjected to radioactive contamination is via the food chain. The radioactivity discharged by nuclear installations ultimately enters the human organism, in dilute or concentrated form, depending on the characteristics of its biological environment. Living creatures change the quantities of radioactivity taken up from the environment. The acquisition of the most precise knowledge possible of the factors involved in the transfer and concentration of this radioactivity is envisaged in several research contracts aimed at determining acceptable levels of contamination in the food chain and the environment, whether in the atmosphere, fresh water, sea water, estuaries, the soil or food.

In addition, the path followed by radioactive substances in man is still not understood fully enough and certain standards relating to the uptake of radioactive substances are currently being amended as a result of studies carried out at a European level.

Conclusion

Owing to the development of nuclear energy, the imminent expansion of its peaceful uses and the resultant increase in the number of potential sources of irradiation or contamination, the coordinated activities of prevention and surveillance, which are already very effective, would have to be stepped up in order to further limit potential risks in the future. This programme is one of the Community's main tasks, since the risk of nuclear accidents goes beyond the national frontiers and therefore justifies concerted action at a Community level.

Other aspects of health protection cover nuclear medicine and hygiene as well as radiobiological research, with regard to which the Commission will continue to act as sponsor and coordinator.

As of now the conditions most favourable to the development of atomic energy in the Community can be considered to have been created by the implementation of a complete programme in the field of health protection.

The succession of stages leading to the fullest use of nuclear energy will enable the necessary technological improvements to be made in good time, thus striking at each stage of development a balance between the advantages and uncertainties still inherent in an accurate knowledge of the effects of radiation.

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COMMISSION
OF THE
EUROPEAN COMMUNITIES

SECOND ILLUSTRATIVE NUCLEAR PROGRAMME FOR THE COMMUNITY

ANNEX II

STRUCTURE OF ELECTRICITY PRODUCTION

DRAFT

(4 April 1972)

STRUCTURE OF ELECTRICITY PRODUCTION

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The Structure of Electricity Production

I. Production and Trends in Demand

The object of this Annex is to indicate the probable overall developments in the field of electricity production in order to supplement the Second Illustrative Nuclear Programme. The main emphasis will be placed on the development of the production structures and on an analyses of the potential part played by the individual sources of energy in the production of electricity.

The production of electricity is bound up even more closely with demand than is the case with forms of energy which can be stored, since each kilowatt-hour is only produced at the moment of consumption. What is stated below on electricity production is directly related to the assumptions concerning trends in demand made in the sections dealing with electricity in the Commission's report on "The Long-Term Prospects for Energy Supplies in the Community".

However, in order to be able to draw conclusions concerning total production requirements from the overall demand as forecast in that document, secondary hypotheses had to be made concerning the development of certain items on the electricity balance sheet. Of decisive importance in production trends is the electrical energy required by the grid, which includes the losses. The corresponding gross consumption also includes the power plants' own requirements and the consumption by the pumped-storage facilities. The gross power generation needed to meet the requirements is obtained from the gross consumption, taking into account the balance on exchanges.

The last two decades have been marked by a continuous, rapidly-increasing demand. If this period is considered in five-yearly stages, there are seen to be only slight discrepancies (less than

0.3% a year) between the rates of increase in demand and supply, and these become even slighter over longer periods.

(in TWh)

	1950	1955	1960	1965	1970	Mean annual growth 1970–1985
Energy consumption (including losses)	119	186	272	392	558	+7.6%
Plants' auxiliary power and consumption of pumped-storage facilities	6	10	16	25	35	
Gross consumption	125	196	288	417	593	+7.7%
Import	2	2	4	6	13	
Gross production	123	194	284	411	580	+7.6%

The period 1950-55 was still characterized by a considerable need to make up lost ground caused by the war, and this led to an average rise of almost 10% a year in consumption. During the subsequent five-year stages the rate of increase flattened out to 7.2-8% a year, the average rate between 1955 and 1970 being 7.6% a year.

In the analysis of demand mentioned above, forecasts for movements in demand covering the next fifteen years are based on a very slight slowing-down in overall demand, within which there will, admittedly, be structural changes. Thus the main expectation is of a disproportionate increase in domestic as opposed to industrial consumption, resulting in a further rise in the level of low-voltage sale.

The anticipated stagnation of the import surplus, together with the faster rate of construction of pumped-storage stations, will necessitate an increase in production equivalent on average to exactly the rate over the last fifteen years (+7.6% a year).

(in TWh)

	1970	1975	1980	1985	Mean annual growth 1985-1970
Energy consumption (including losses)	558	796	1,130	1,610	+7.3%
Plants' auxiliary power and consumption of pumped-storage facilities	35	60	90	140	
Gross consumption	593	856	1,220	1,750	+7.5%
Import surplus	13	16	10	10	
Gross production	580	840	1,210	1,740	+7.6%

These overall data constitute the starting point for the two-part survey of the structure of electricity production which covers the periods 1950-1970, 1970-1975 and 1975-1985. A retrospective look at the period 1950-1970 is included in the survey, since it provides interesting statistical information, not only on the Community as a whole, but also on the varying rates of development in the individual Member States. All the decisions relating to generating capacity for the period 1970-1975 have already been taken, whereas a certain latitude still exists as regards the pattern **of electricity production from the different energy sources. For the period 1975-1985, the development trends emerging today, which should be assumed as probable without taking the criteria of a common policy into closer consideration, are assessed.**

II. Evolution in the Community and the Community countries between 1950 and 1970

1. The part played by the different sources of energy in electricity production

1.1 Over the last two decades a distinct change has taken place in Community electricity production. This was the outcome of numerous investment and operational decisions reached in the light of the possible alternative methods of production.

In those places where the topography was suitable, water power traditionally held pride of place, thanks to a number of indisputable advantages, among which were the continuous natural replenishment of the energy consumed, the very low operating costs and the long operating life of the generating plant.

However, that water power only accounted as to one-eighth for the increase in demand for electricity can be explained by the fact that development potential approached saturation and that the generation facilities themselves became more expensive as a result of the inevitable choice of less suitable sites. In addition, high transport costs are usually incurred as a result of the inflexible siting which characterizes water power. Another reason for the relatively slight growth rate of energy from hydraulic sources is the growing importance of storage and pumped-storage stations, their production being restricted to short-duration, peak-lopping activities only.

The bulk (about four-fifths) of the increased demand for electricity during the last couple of decades has been met by fossil fuels.

At the beginning of the period under consideration, hard coal held a clear lead in all the Community countries possessing their own supplies, which in the case of Germany was enhanced by considerable quantities of lignite. Whereas lignite roughly maintained its position, hard coal's share in the expanding demand dropped under pressure of competition from hydrocarbon fuels after the end of the fifties, and even considerable national subsidies did not prevent an absolute decline in all the Community countries at the close of the sixties.

The use of mineral-oil products in 1950 was restricted essentially to limited quantities of diesel oil for peak-lopping purposes. During the fifties a start was made on the use of fuel oil in power plants, but because of limited availability and the impossibility of concluding long-term contracts it was unable to gain much ground. It was only when large quantities of heavy fuel oil came on to the market at attractive prices as from 1960 that a pronounced upsurge occurred in the use of fuel oil in thermal power stations at the expense of hard coal.

Natural gas reserves of inter-regional importance only became available in the Community after the mid-sixties. Up to that time the deposits at Lacq and in the Po valley had been tapped to a certain extent for electricity-generating purposes but the first real impetus to the large-scale use of natural gas in the electric power sector was provided by the opening-up of the rich Dutch fields. However, until 1970 the main purpose served by this development was the reshaping of electricity production in the Netherlands, where natural gas has already overtaken every other form of fuel used in power stations. An incipient expansion in the use of natural gas for electricity-generating purposes has also made itself felt in Belgium and Germany, both in its rapidly increasing share of the market and in the signing of important contracts.

The expectations aroused by nuclear energy as early as the mid-fifties remained unfulfilled during the following decade. The technical and economic difficulties that had to be overcome in the use of this fuel for power-generating purposes proved greater than initially supposed. In addition, the anticipated shortage of fossil fuels did not come about, and indeed heavy fuel oil in particular was available on attractive terms for consumption in power plants. A start on the construction or planning of nuclear power plants on a large scale in the Community had to wait until the beginning of the seventies.

The developments over the last two decades as illustrated here are shown below in quantitative terms for the Community and the Member States.

1.2 In 1950, over 90% of the electricity produced in the countries which were later to form the Community was still obtained via indigenous sources such as water power and coal (hard coal and lignite). 38% was accounted for by water power (47 TWh), 44% by pit coal (54 TWh) and 9% by lignite (11 TWh).

The proportion accounted for by these fuels had dropped to a total of 57% by 1970, when lignite had registered a slight improvement to 11% (66 TWh), but hard coal and water power had fallen back to 26 % (151 TWh) and 20% (117 TWh) respectively.

Over the same period, the use of mineral oil products (and especially heavy fuel oil) soared in the power station sector. Although the share of these fuels was still quite insignificant at 2.5% (3 TWh) in 1950, by 1970 it had overtaken that of every other form of fuel in use in power stations with a proportion of 27% (154 TWh).

There was also a marked increase in the consumption of natural gas in thermal power stations, although not to the same extent as with fuel oil. As late as 1965 the share of natural gas in electricity production was still only 2% (9 TWh) and it had risen to no more than 8.4% by 1970 (49 TWh).

The proportion of electricty generated via nuclear energy in 1970 was, at 2.6% (15 TWh), still unimportant in comparison with other fuels. The first feeding of nuclear electricity into the grid in the Community took place in France in 1957, followed by Germany in 1961, Belgium in 1962, Italy in 1963 and the Netherlands in 1968.

1.3 Conditions in the individual Member States often diverged considerably from the overall structure in the Community[1]. Basically, the following developments have taken place:

In Germany, it is clear that the bulk of the electricity produced was initially derived from coal (breakdown for 1950: hard coal 52%, lignite 24%). In 1970, the position of these two fuels was still strong (hard coal 39% = 96 TWh, lignite 25% = 61 TWh); at this time fuel oil accounted for 15% (36 TWh), thus occupying a distinctly inferior position in relation to the other Member States (apart from Luxembourg). The proportion of electricity provided by water power slumped from 18,5% in 1950 to 7% in 1970.

In France, about 90% of the electricity produced in 1950 came from water power (47% = 16 TWh) and hard coal (42% = 14 TWh). Up to 1960 the proportion provided by water power rose to 55%, which then dropped to 39% in 1970. At this point hard coal accounted for 25% and fuel oil 22 %.

In Italy, almost 88% of the electricity produced in 1950 was based on water power, its share of the market being an overwhelming 82% until 1960. During the sixties, more electricity was generated via fuel oil and in 1970 the contribution of water power was only 35% as agianst the 48% notched up by fuel oil.

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[1]The tables containing data on the Member States are in the Annexes.

Structure of Gross Electrical Production in the Community

Breakdown according to sources of energy 1950-1970

				Conventional thermal energy							
Year	Water power	Geo-thermal heat	Nuclear energy	Hard coal	Lignite	Mineral oil products	Natural gas	Derivative gases	Other fuels	Total conventional thermal energy	TOTAL
1950	46.7	1.3	-	54.2	11.3	3.0	0.1	6.1	0.3	75.0	123.0
1955	68.9	1.9	-	80.9	23.1	6.6	2.1	10.5	0.5	123.7	194.5
1960	100.2	2.1	0.1	110.9	34.6	14.7	6.4	13.7	0.7	181.0	283.5
1965	106.5	2.6	4.7	148.6	49.6	73.3	8.6	14.0	2.5	296.6	410.4
1970	117.4	2.7	15.3	150.6	65.5	153.8	48.9	21.3	4.9	445.0	580.4
				(in percentages)							
1950	38.0	1.0	-	44.0	9.2	2.5	0.1	5.0	0.2	61.0	100
1955	35.4	1.0	-	41.6	11.9	3.4	1.1	5.4	0.3	63.6	100
1960	35.4	0.7	0.1	39.1	12.2	5.2	2.3	4.8	0.2	63.8	100
1965	25.9	0.6	1.1	36.2	12.1	17.9	2.1	3.4	0.6	72.3	100
1970	20.2	0.5	2.6	26.0	11.3	26.5	8.4	3.7	0.8	76.7	100

In the Netherlands and Belgium, the structure of electricity production in 1950 was broadly similar: in both countries, where water power plays no part, hard coal accounted for just under 90% of the production. While in both countries, and particularly after the beginning of the sixties, increasing use was made of heavy fuel oil in power stations, different paths were followed as from the mid-sixties Over a period of five years the position of natural gas in the Netherlands increased from less than 1% to 47% in 1970. At this time fuel oil provided 33% and coal only 16% of the electricity generated. In Belgium, on the other hand, fuel oil had a dominant position (50%) in 1970, while coal's share had dropped to 26%. Characteristic of Belgium is the relatively high proportion of derived gases (roughly constant between 1950 and 1970 at 10%) consumed in industrial power stations.

Up to the beginning of the sixties, the structure of electricity production in Luxembourg was based almost exclusively on the blast-furnace gas produced during iron-founding. Since 1962, water power has gained a statistically important position as a result of the construction of the Vianden pumped-storage station, but it should be noted that this plant is phased exclusively to the German grid. The upsurge in domestic demand in Luxembourg will, on the other hand, only be covered by imports until economic high-capacity plants are built, possibly on a joint basis with neighbouring countries.

2. Further expansion of generating plants

2.1 A quantitative plot of capacity extensions covering the last twenty years must be restricted to a breakdown according to water power, geothermal heat, conventional thermal power and nuclear energy. A retrospective breakdown of the output of conventional thermal power stations is difficult, since the time sequences for the period under consideration are neither gapless nor comparable[1].

Installed power plant capacity in the Community 1950-1970
(Position at each year's end in MW)

Year	Hydroelectric power plants	Geothermal power plants	Conventional thermal power plants	Nuclear power plants
1950	15,009	220	25,235	-
1955	21,024	250	36,611	-
1960	26,572	309	53,438	96
1965	32,401	339	76,999	1,097
1970	36,289	402	104,017	3,376

2.2 An extension of generating capacity requires particularly careful planning, since electricity as such cannot be stored and its power at anytime is determined by the customer and not the supplier. Should the supplier be unable to meet demand at a given moment, the first option open to him is "brwon-out". If this is inadequate, he is forced to block out certain sections of the grid.

[1] It was not possible to compile a comparative, detailed list of power plant capacity according to types of fuel in the Community until after 1963, and then only gradually and, unfortunately, not on a retrospective basis.

The recording criteria differed so greatly before 1969 that the data concerning the current situation can no longer be compared with those for past years.

Since power plants take several years to be built, during which time unforeseeable increases in demand can occur, an adequate stretch margin to cover future growth must always be included in the planning's. In addition, reserve capacity is needed in order to offset irregular water supplies, plant failures causes by operational defects and fluctuating demand due to climatic or fortuitous factors.

At the beginning of the fifties there was still a certain tightness in reserves. This situation arose from a particularly rapid expansion in demand resulting from both the post-war backlog demand and the boom around 1950. However, at the end of the sixties the thermal reserves available at the time of the winter peak amounted to 25% of the maximum load on the grid (as against 12.17% around 1953). This may to some extent also be explained by the trend towards increased reserves under the influence of placing into service of continuously growing unit capacities.

With more and more nuclear power plants with increasing unit capacities coming on stream, one can proceed on the assumption that considerable reserves will be maintained in the near future.

2.3. Although an increased level of reserves normally leads to retrogressive development in the mean utilization factor of power plants, this has been more than offset in the Community as a whole by an improvement in load conditions. The mean annuel utilization factor of the maximum grid load shows an upward trend in all Community countries with the exception of Italy, where it was allready above the average before 1960. The underlying narrowing of the typical fluctuations in demand, apply to the daily and weekly but not to the seasonal profile. As a matter of fact, the drop in demand is seen to be even more pronounced during vacation periods. On the other hand, consumption during the low-demand periods - weekends and holidays - clearly rises more quickly than consumption on working days, so that the repercussions of shorter working hours have been more than offset. A specific leisure-time demand is reflected in the levelling-out of the weekend trough, and is expanding particularly rapidly in step with the rising standard of living. A contribution towards leveling-off the working-day load curve is made by the electricity-intensive industries, which normally work to the three-shift system and are expanding at a faster rate than overall demand. Finally, consumption during the slack periods is boosted, in particular by use for home- and water-heating purposes, as a result of cheap offpeak rates.

2.4 The changes in the load patterns and reserve keeping affected the utilization of the various types of power plant in different ways.

The installed water-power capacity increased somewhat more quickly than producibility in an average year, through the growth in the proportion of storage installations devoted to peak-lopping.

On the other hand, production in the thermal sector outpaced the expansion of installed capacity. Improved load pattern coupled with an increasing proportion of the base-load being covered by thermal power, are preponderant contributory factors in this trend. A certain influence was also exerted by the decline in the share of the run-of-the-river power stations, which led to a corresponding reduction in reserve capacity needed to offset a water shortage in the Community and in the Alpine areas.

The resultant increase in the mean plant utilization period from about 3,700 hours/year around 1950 to about 4,500 hours/year around 1970 made a considerable contribution to the profitability of investments in thermal power stations. According to a rough calculation, the resultant improvement of about 800 hours a year is attributable in the main to the following causes:

Improvement in load curves	+ 650 h
Cutback on reserves in covering water shortages	+ 450 h
Increase in general reserves	- 300 h
	+ 800 h

2.5 Although the statistical data of the Community over the last two decades do not permit a complete representation of generating capacity as a function of unit sizes, the general trend has been towards a very rapid increase in the standard size of newly-installed thermal generating plant, the aim of which was to take benefit from the attendant cost savings.

While at the beginning of the fifties the size of modern power plants was about 50 MW, around the end of the last decade the peak of technical achievement for plant already in operation was 600 MW, and conventional and nuclear plants had been designed to develop up to 800 and 1,300 MW respectively.

2.6 The trend towards larger generating units has been accompanied by continuous improvements in thermal efficiency. In 1950, the most modern units were able to achieve efficiency ratings of 31-33% under optimum operating conditions (50 MW); in 1960 (125 MW) the figure was 36-37%, a little later (250 MW) it was 39% and in 1970 (600 MW) 40-42%.

The coming to the fore of modern generating plant has caused the average thermal efficiency, related to the net production of conventional thermal power stations in the Community countries, to increase from less than 20% in 1950, through 29% in 1960, to 35% in 1970.

The improved combustion efficiency can be illustrated by the fact that, whereas in 1950 the production of a kilowatt-hour required an average consumption of 640 g of coal, in 1970 the average rate was 350 g and in the msot up-to-date plants no more than the thermal equivalent of just over 300 g of coal is required.

These improvements were achieved via technical advances in the construction of boilers and turbines. The outcome has been a certain degree of specialization of base load plants. Previously base load was covered by the more modern generating sets, whereas older plant were used for peak-lopping purposes (the consequence being more frequent load changes and plant shutdowns). This specialization in the base-load sector led to an intensified swing towards typical peak-load power plants such as pumped-storage stations and gas turbines.

Low investment costs and short start-up times, together with good load following characteristics, make gas turbines particularly suitable for operation at low utilization factors, despite their high fuel costs. Recently, composite steam-gas turbine sets have appeared, in which the waste heat from the gas turbine section is ducted to the steam-turbine boiler, thus raising the overall efficiency of the plant and at the same time improving its load following characteristics. As far as the future is concerned, the prospect which emerges is that of the construction of conventional thermal power plants of a simplified type, in which reduced thermal efficiency is balanced against lower investment costs greater adaptability of operation, the base load sector being dealt with by nuclear energy.

2.7 The rising unit-sizes demanded a rapid extension of the transmission network, which has had to keep pace with the rate of expansion of generating plant. However, in smaller grids the possible expansion proved at times bo be a limiting factor of the unit capacities. The current extension of the 380 kV grid is already the consequence of a wider-reaching planning than was the case with the setting-up of the 220 kV grid. For currently foreseeable unit sizes the 380 kV grid will remain adequate until the next stop to a voltage level of over 1,000 kV will be made. In this way the 760 kV voltage system, already in use in North America and the USSR for transmission over long distances, would be bypassed in the Community. Up to 1985 the beginning of a transition to a higher voltage will in any case not alter fundamentally the situation of the period under consideration.

3. The role of industrial self-producers

3.1 At a given location, there is as a rule only one supplier available to meet the demand for electricity, since otherwise the necessary overlapping of grids would call for economically unacceptable additional investments.

Nevertheless, an alternative capable of competing with the public supply sector is available to the consumer with sufficiently high requirements, i.e., the possibility of generating his own electricity. Admittedly the choice is not always fully open to the party wishing to meet its own needs in all member countries. Even if the choice is open to him, the decision hinges on problems such as the maintenance of reserves and quality of supplies, which are more difficult to solve for the independent supplier than for any public electricity utility, which can rely on the grid.

Accordingly, the decision in favour of setting up one's own electricity generating plant can only be taken of conditions particularly propitious to this measure exist. The following are cases in point:

(1) Where a requirement not only for electricity but also for heat can be fulfilled by combined plants capable of generating both electricity and process steam (energy-steam cycle); or

(2) Where unmarketable or surplus energy can be usefully employed in the "in-plant power station" (e.g., low-grade coal in pit-head power stations, waste heat and blast-furnace gas in ironworks power stations, oil-refinery gas and synthesis gas from chemical plants, etc.).

Another reason for building pit-head power stations is that consumption of coal on as regular a basis as possible in these plants can stabilize output and employment in the mines.

3.2 The importance of industrial self-production of electricity, measured as a proportion of total output, diminished slightly in the Community during the period under consideration, namely, from 32.9% in 1950 to 26.7% in 1970. This relative slump could be explained primarily by the fact that the public utilities are better able to achieve economies of scale via increased plant size and more efficient use of the grid than individual producers. The particularly distinct fall-off after 1965 shows that the demand threshold, above which the setting-up of privat generating plant can yield advantages, has moved upwards.

3.3 Although what has been said regarding the Community as a whole also applies in general to the individual Member States, the considerable divergences shown below should also be noted:

Germany

The high proportion of 40.2% recorded in 1950 for industrial self-production diminished steadily up to 1970, when the figure was 33%.

France

Between 1950 (30.7%) and 1955 (35.2%), self-production grew, but its share then dropped back to 19.7% by 1970.

Italy

The 19.7% recorded in 1950 was the lowest for all the Member States. This had risen to 25.7% by 1965 but five years later it had relapsed slightly to 24.4%.

Netherlands

There was a distinct drop between 1950 (26.7%) and 1970 (15.4%). In 1970, the proportion of industrial self-production of electricity in the Netherlands was therefore the lowest for all the Community countries.

Belgium

Here industrial self-production is also of considerable significance. The highest level in the entire Community (45.3%) was recorded in 1955, the figure for 1970 being 31.4%.

Luxembourg

Domestic requirements were, apart from insignificant amounts of water power, met be iron-foundry power stations. As a result of the commissioning of the Vianden pumped-storage station, which is not connected to the Luxembourg grid, the proportion of the national production accounted for by industrial power stations dropped to 58.7% in 1970.

III. Current situation and short-term developments (1970-1975)

1. Generating capacity in existence or under construction

In 1975, the electricity consumed in the Community will be produced in power plants already in operation or currently under construction. Disregarding the fact that certain delays could occur in the completion of the power plants planned for 1975, it can be taken that there is already a reasonably clear picture as to the generating capacity the Community will have in hand in 1975, which will consist of the following capacities as compared with 1970:

Structure of power generating capacity in the Community

- 1970 and 1975 -

Type of power plan and fuel used	Electrical plant - installed capacity - End of 1970 1,000 MW	%	End of 1975 1,000 MW	%
A. Total	(144.1)	100.0	(203.7)	100.0
B. Privileged sources of energy				
1. Water power	36.3	25.2	42.5	20.9
a) Run-of-river power stations	12.9	9.0	13.7	6.7
b) Storage power stations	23.4	16.2	28.8	14.2
2. Geothermal heat	0.4	0.3	0.4	0.2
3. Conventional thermal power	17.1	11.9	20.3	10.0
a) brwon coal, single-fuel	9.6	6.7	12.8	6.3
b) brown coal, multi-fuel	0.6	0.4	0.6	0.3
c) derivative gases, single-fuel	2.5	1.7	2.5	1.2
d) derivative gases, multi-fuel	4.4	3.1	4.4	2.2
C. Non-privileged sources of energy				
4. Conventional heat energy	86.9	60.3	127.7	62.7
a) single-fuel[1]	58.3	40.5	82.9	40.7
hard-coal	35.0	25.0	28.2	13.8
mineral oil products	20.8	14.4	47.6	23.4
natural gas	1.5	1.1	7.1	3.5
b) multi-fuel	28.6	19.8	44.8	22.0
hard coal/mineral oil products	16.7	11.6	22.9	11.3
hard coal/natural gas	1.7	1.1	1,7	0.9
mineral oil products/natural gas	8.2	5.7	17.6	8.6
hard coal/mineral oil products/natural gas	2.0	1.4	2.6	1.3
5. Nuclear power plants	3.4	2.3	12.7	6.2
Conventional thermal power, total (3+4)	104.0	72.2	148.1	72.7
of which single-fuel	70.3	48.8	98.2	48.2
total multi-fuel	33.7	23.4	49.9	24.5

[1] Including less recent lignite and lignite briquettes and also low, indivisible capacity.

It can be seen from the table above that the proportion of total capacity provided by hydro-electric power stations will diminish still further (1970 = 25%, 1975 = 21%), despite an absolute increase in installed capacity (+ 6,000 MW), which is mainly accounted for by storage - and in particular pumped-storage power stations.

Before 1975 there will be no further developments in geothermal power, which is restricted to Italy; in addition, the number of plants using derivative gases in the Community will probably remain unchanged.

By the same date lignite-fired power station capacity will have increased by slightly over 3,000 MW but its share in the total installed capacity will remain almost constant (6.3%).

A particularly sharp drop has occurred in the number of coal-fired power plants (- 7,800 MW, including low, indivisible levels of production), caused by the high age of many of the smaller, hard-coal-fired power stations, which had become uneconomic, and by the conversion of a number of more modern plants to oil or natural gas. In contrast, a considerable increase in single-fuel, fuel oil and natural-gas-fired power stations (+ 26,800 MW and + 5,600 MW respectively) has to be recorded.

In the case of dual- or triple-fuel power stations the hard coal/fuel oil and fuel oil/natural gas combinations are of primary importance. While the installed capacity of coal/fuel oil fired power stations will increase by 6,200 MW by 1975, the output of fuel oil/natural gas power stations will rise by 9.400 MW.

The installed capacity of nuclear power plants will by 1975 have increased by 9,300 MW (to 12,700 MW) to take a 6.2% share in overall production, which at that date will roughly correspond to the contribution made by lignite.

2. Structural developments in electricity production

Any forecast of the structure of current electricity production in the Community for 1975 must be based first of all on the fact that the gross production needed to meet requirements will be 840 TWh (as against 580 TWh in 1970), this representing an average annual increase of 7.7%.

There is a considerably greater uncertainty attached to the breaking-down of the anticipated production for 1975 than there is in the case of overall capacities. This is due to:

(a) The latitude in decision-making on the use of fuels in multi-fuel plants,

(b) The variability of the load factor applying to the individual plants,

(c) The operability of the plants.

It should be pointed out in this connection that the values for dual- and triple-fuel power plants quoted in the foregoing table do represent their theoretical interchange potential, whereas the actual freedom of choice is considerably restricted by certain factors. The appropriate storage and transport facilities for choosing at will fuels for which the boilers were designed are not always present at plants recorded statistically as multi-fuel.

Even where all the technical conditions governing the use of several alternative fuels exist, the room for manoeuvre in a number of typical cases is severely restricted in practice. In multi-fuel lignite power plants, the alternative fuel is as a rule only used in order to obtain a more flexible response to grid requirements than complete dependence on lignite supplies would allow. On the other hand, multi-fuel plants burning derivate bases, usually facilitate the useful consumption of blast-furnace, coking or refinery gas occurring outside the sphere of influence of the power stations. In both cases the latitude for decision-making is very narrow.

The clear options open to the power plant operator are thus restricted to installations designed to burn coal, fuel oil and natural gas as required, a field where the coal/fuel oil and fuel oil/natural gas combinations are particularly prominent.

The use of natural gas in power stations designed for the purpose will only be restricted in favour of fuel oil or coal, given current price ratios, if the quantities of natural gas supplied under contract in order to meet the full operational needs of these plants are inadequate.

The electricity produced by dual-fuel coal/fuel oil-fired power plants during the 1970 period was, from a plant equipment point of view, about 70 TWh and it could increase to 100 TWh by 1975. It must, however be borne in mind here that some of the plants under examination are restricted as regards their options, owing to the stipulations of the German "electricity-from-coal" law. In 1970, the proportion accounted for by coal in the fuel consumed in dual-fuel coal/fuel oil power plants was an estimated 30% and by 1975 this figure may have dropped slightly in favour of fuel oil.

In a simplified matter, the following fuels used in power stations occupy a special position:

(a) Fuels inevitably arising from other processes which, apart from being usefully consumed in power stations, can find either no market at all or only a restricted one, (e.g., blast-furnace, coking or refinery gas, waste fuels).

(b) Fuels particularly occupying a privileged position in existing plants[1] since they incur low variable costs (water power, geothermal heat, lignite).

On the other hand, it can be assumed that in the case of the other fuels, namely coal, fuel oil, natural gas and nuclear energy, that the extent of their use in power plants is determined by the competitiveness of their position on the fuel market. As regards their use in the power plant sector, therefore, these fuels are thus in direct competition with each other.

It is quite obvious that the uncertainty involved in forecasting affects the first group (hereinafter called "privileged fuels") less than the competing fuels.

Although the privileged fuels still provided as much as 36.5% of gross electricity production in the Community in 1970, this figure is likely to continue to decline and by 1975 be a bare 30%. The predominant position of the competing fuels will thus be consolidated further.

The probable trends of the various fuels used in the Community for electricity-generating purposes up to 1975 are plotted below.

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[1] In the power stations due for construction, these fuels will also be in competition with other forms of energy. Their use is only acceptable from an economic standpoint if not only the variable but also the overall production costs justify such choice.

2.1 The privileged fuel sector

Except in the case of pumped-storage stations, hydro-electricity is dependent to a great extent on climatic conditions. Production for can therefore only be forecast on the basis of a year with normal flow conditions.

Up to 1975 an annual growth rate of only 1.6% is assumed for total hydro-electricity production. This figure admittedly incorporates an almost threefold increase in pumped-storage production between 1970 and 1975. The contribution made by water power in the Community will thus continue to diminish until it reaches about 15% in 1975.

A further expansion of geothermal electricity production in the Community is not envisaged. The plants in existence in Italy will probably achieve only a slight rise in their output, and therefore the share of less than 1% taken by these fuels in 1970 will drop still further.

Electricity produced from lignite will probably increase by an annual average of 6.1% up to 1975. Since lignite-fired power plants operate at low variable costs, it can be assumed that plants of this type will operate at a yearly load factor of about 6,500 hrs operating time in order to cover base load requirements if full capacity is available. The proportion of production provided by lignite would thus fall slightly to about 10% in 1975.

The use of derivative gases and other types of fuel in power plants will increase on average by 4% a year up to 1975. Among these are included coking, blast-furnace, and refinery gas, together with some insignificant fuels such as industrial waste gases, tar and household refuse. The growth in this sector will come mainly from coking gas, which will be replaced by natural gas for public supplies and be used in industrial power plants close to the place of production. In 1975 derivative gases and other types of fuel will account for a bare 4%.

2.2 Competing fuels

The predominant position of this group has already been pointed out. Between 1970 and 1975 electricity production in the Community provided by competing fuels will rise by 221 TWh, while an increase of only about 39 TWh will be accounted for by privileged fuels.

The greatest absolute growth, about 127 TWh (+ 12.8% a year on average), will thus be attributable to fuel oil, with production based on this fuel rising particularly steeply in France (+ 58 TWh = 23.0% a year) and Italy (+ 45 TWh = 12.4% a year), while the position in Belgium will bottom out. Up to 1975, therefore, the proportion of electricity produced via fuel oil in the Community will rise by about 33%.

Natural gas will also experience a sharp rise of about 54 TWh (+ 16.1% yearly average), thereby actually bettering the relative growth rate of fuel oil. This increase will take place largely in Germany (+ 22 TWh = 19.8% a year), the Netherlands (+ 20 TWh = 15.4% a year) and Belgium (+ 8 TWh = 23.7% a year) and on a Community scale natural gas will then provide slightly over 12% of the energy used to generate electricity in 1975.

Since the coal-fired electricity production calculated for 1975 does not lie much below that for 1970 (- 10 TWh = - 1.4% a year) the proportion of production accounted for by this fuel will drop to about 17%. The decline in the use of coal will be especially marked in France (- 13% TWh = 8.3% a year) and the Netherlands

(- 6 TWh = 31.7% a year); while the sharp drop in France is primarily attributable to the conversion of 250 MW coal-fired units to fuel oil, the remaining coal-fired generating capacity in the Netherlands has changed over almost entirely to naturel gas. On the other hand, a slight increase in the use of coal-fired power plants is expected in Germany, Italy and Belgium.

Admittedly coal-fired capacity in Germany will be lower in 1975 than in 1970; however, this drop will probably be more than compensated by the higher utilization factors of the coal-fired power plants entering service at the beginning of the seventies, which take advantage of the favourable terms offered by the "electricity-from-coal" laws. A slight expansion in coal-fired production, based on imported coal, is expected in Belgium, and in Italy increasing use is expected to be made of imported coal in dual-fuel coastal power plants.

Between 1970 und 1975 the Community nuclear electricity production will record its relatively highest growth rate (+ 33% a year) since it started from a very low level. At that time, however, the proportion supplied by nuclear energy will, at less than 8%, still be of little consequence and production will probably be with 65 TWh relatively modest. This rate of production is admittedly based on the hypothesis that nuclear power plants do not achieve a high utilization factor during the year of commissioning[1] and the two subsequent years. Bodies not adopting in their forecasts this emphatically cautious estimate of the utilization factor, which is based on prior experience of nuclear power plants during their start up period, arrive at correspondingly higher anticipated values for nuclear electricity production.

The relevant data are summarized in the table below:

[1] The individual estimated utilization factors during the start up time are shown on page 22 of the "Second Illustrative Programme for Nuclear Energy".

Short-term trend of gross electricity production in the Community

	Share				Average annual growth rate
	1970	1970	1975	1975	
	TWh	%	TWh	%	%
A. Gross production (B+C)	580	100.0	840	100.0	7.7
B. Privileged sources of energy (1+2+3)	211	36.5	250	29.8	3.4
1. Water power	117	20.2	127	15.1	1.6
- primary	114	19.6	119	14.2	0.2
- pumped-storage power stations	3	0.6	8	0.9	20.1
2. Geothermal heat	3	0.5	3	0.4	-
3. Thermal energy	91	15.8	120	14.3	5.5
- lignite	65	11.3	88	10.5	6.1
- derivativgagas, etc.	26	4.5	32	3.8	4.1
C. Non-privileged sources of energy (4+5)	369	63.5	590	70.2	9.9
4. Conventional thermal energy	354	60.9	525	62.5	8.2
- hard coal	151	26.0	141	16.7	1.4
- mineral oil products	154	26.5	281	33.5	12.8
- natural gas	49	8.4	103	12.3	16.1
5. Nuclear energy	15	2.6	65	7.7	33.5

IV. Medium- and long-term trends (1975-85)

The aim of this section is to examine the probable contribution of individual fuels to the production of electricity up to 1985. Development trends currently emerging, including the influences and guidelines emanating from the State, are dealt with in this analysis. In addition, a rate of development is assumed for nuclear power which takes into account the minimum requirements laid down in the "Second Illustrative Programme for Nuclear Energy".

In this section there will be no examination of the extent to which the developments which can be extrapolated from the factors already available will correspond to the energy policies of the Community, or whether it has proved to be desirable or necessary to aim for guidelines taking greater account of criteria relating to a common energy policy.

1. Overall development in generating capacity and gross production

The largely constant rise in electricity consumption and the long-term nature of the investments in generating plant needed to cover this rise require a clear picture, over a relatively long period, of **the developments in power plant capacity and of the total** electricity production needed to meet demand. It can thus be seen that the gross installed capacity in the Community between 1970 and 1975 will increase by about 60,000 MW.

A large proportion of the plant which is to enter service before 1980 is already under construction or at the planning stage. Disregarding the necessary investments to cover plant replacement, it can be taken that installed capacity will increase by over 90,000 MW between 1975 and 1980, and the rise between 1980 and 1985 can be estimated at about 120,000 MW.

Between 1970 and 1975, gross electricity production will average a 7.7% expansion annually, while the corresponding values for the periods 1975-80 and 1980-85 could be about 7.6 and 7.5%.

Probable overall developments in installed capacity and gross electricity production are shown in the following table:

	1970	1975	1980	1985
Installed capacity (in GW)	144	205	298	417
Gross electricity production (in TWh)	580	840	1,210	1,740

2. Probable contribution of the various fuels to the long-term production of electricity

Although the movements in the overall installed capacity of the power plants available and of total electricity production can thus also be forecast fairly clearly for the period up to 1985, this does not apply in the same degree to the structure of installed capacity and applies even less to the fuels used in the production of electricity.

The decision concerning the extent of the future use of various fuels should in power plants depends on a whole series of factors. From a purely accounting point of view, the electricity would have to be generated in power plants at the lowest possible cost, compounded of capital, operating and fuel costs, for a certain number of hours of utilization. Where the annual utilization period is high, power plants with low fuel costs really come into their own. This basically valid principle can, however, only be applied under certain conditions. Water power, for instance, is an extremely cheap fuel, which, however, is only vailable to a limited extent, depending on the region. This also applies to lignite, which is not worth transporting elsewhere. Nuclear power plants too have low operating costs and the main question-marks here hang over the capacity of the individual nuclear power plant manufacturers, financing, operational safety and also the availability of suitable sites - all of which affect the expansion of nuclear generating capacity. In addition, there are fuels, although of secondary importance, arising from linked-production processes like blast-furnace, coking and refinery gas, which have to be burned in power plants. Finally, there are long-term considerations concerning the dependability and continuity of supplies, together with matters relating to social policy.

The long-term prospects for the individual fuels are analysed below, once again, for the sake of continuity, under the heads of "privileged" and "compteting" fuels. It should, however, be pointed out that this method of division appears problematical, especially as regards the assignment of lignite to the "privileged" sector. Lignite is in the keenest competition with other fuels, and particularly with nuclear energy, in the covering of the base load.

In any case, the share of the privileged fuels in electricity production will diminish still further in future and probably only account for about 15% (1970 = 36.5%) in 1985.

2.1. Privileged fuels

(a) Water power

Since the potential of natural water power is a function of both usable flow and gradient, its geographical concentration is largely confined to mountainous areas. Almost three-quarters of the water power exploited in the Community derives from Alpine water, while a further 14% is located in the Apennines and the Pyrenees and only 13% in the uplands and other fluvial regions.

Of the Community countries, France and Italy, and to a slight extent Germany also, are favoured by such conditions, there being very little development potential for natural heads of water in the Benelux countries.

Although account must be taken of the fact that the break-even point has moved downwards in recent years, it can be assumed as a rough guide that about 80% of the economically usable potential in the Community has been developed. It is obvious that the most suitable sites are used for development first and the the economic return is in inverse proportion to the degree of development. Furthermore, the construction of hydro-electric power stations offers considerably fewer prospects of economies of scale than thermal power plants where the trend is towards larger production units.

In future, therefore, the projects undertaken will in the main be only those which also entrain other economic benefits in addition to electricity production, such as canalization, flood protection or irrigation.

For purely electricity supply purposes, on the other hand, increasing use is being made of pumped-storage stations not utilizing natural heads of water, but recycling and storing a head of water by means of cheap, offpeak electricity in order to have a valuable source of peak-lopping power on tap.

As far as the development of natural water power is concerned, the major efforts are being directed towards optimum concentration of the energy thus obtained towards peak periods. This not only means preferential treatment of the water power offering storage facilities, but also symbolizes the trend towards extending the output capacities from existing and future stations, so that the stored water can be disposed of in the shortest possible time.

Between 1970 and 1985, the total installed capacity of hydro-electric power stations will probably rise from 36.3 GW to a bare 55 GW; of this increase, the bulk of about 15 GW will fall to storage stations, which operate at a low utilization factor. The proportion of electricity generated in the Community via water power will thus diminish still further to a figure of about 8.6% in 1985 (1970 = 20.2%).

The importance of hydro-electric power stations cannot, it is true, be judged on the basis of the energy produced. Storage, and above all pumped-storage, power stations yield very low utilization factors, but they do make their output available at peak-load periods, thus playing an extremely important part in levelling out the fluctuations in demand for electricity.

The probable trend of installed capacity, gross producing and the utilization factor for hydro-electric power stations up to 1985 is shown in the table below.

	1970	1975	1980	1985
Installed capacity total (in GW)	36.3	42.5	50	54
of which storage	23.3	28.8	35	38
run of river	12.9	13.7	15	16
Total production (in TWh)	117.4	127	139	149
of which storage	55.4	63	70	75
run of river	62.0	64	69	74
Average utilization factor (hours/year)				
Storage (annual, short-term and pumped storage)	2,400	2,200	2,000	2,000
River	4,800	4,700	4,600	4,600

(b) Lignite

Ninety percent of crude lignite (brown coal and black lignite) extraction in the Community is concentrated in Germany and particularly in the Rhineland brown coal field. Over the last ten years it has hovered around 30 million tce a year, and over the same period the amount delivered to the power plants has increased from 15 to 24 million tce.

Since the market for brown coal briquettes is in a structural decline and new applications for brown coal could not make themselves felt until 1980 at the earliest, movements in brown coal extraction during the period under survey will be predominantly a function of the fuel requirements of the brown coal-fired power plants and in 1985 will amount to about 38 million tce.

On the other hand, the expansion of brown coal-fuelled electricity production presupposes the opening-up of deposits which can be worked economically, a point to be noted here being that the opening-up and working of new lignite deposits will inevitably entail higher costs, mainly due to increasing working depths.

Brown coal-based electricity production demands heavy investments in generating plant and fixed costs also predominate in the fuel costs of brown coal is highly mechanized. Conversely, the very extraction of lignite is highly mechanized. Conversely, the very low variable costs will mean continued use of the installations to cover base load demand - a position from which they could not be ousted by nuclear energy in the modulated part of the load before the beginning of the eighties at the earliest, and then only gradually.

The following orders of magnitude can be used as reference points for lignite (brown coal and black lignite) consumption in power plants, the capacity of these plants and their production up to 1985 :

	1970	1975	1980	1985
Lignite (millions of tons tce)	23.6	32	36	36
Power plant output (GW)	10.1	13.6	15.4	16.0
Gross production (TWh)	65.5	88	100	100
Mean utilization factor (hours/year)	6,500	6,500	6,500	6,250

(c) Derivative gases and other fuels

The forecasts of coking and blast-furnace gas available for pit-head and iron-foundry power stations are based on the assumption that pig-iron production will rise by 3-3.5% a year up to 1985. Here the anticipated technological advances, and in particular the drop in specific coke consumption, have to be taken into consideration.

As a consequence of the rapid spread of the use of natural gas in public supplies, coking gas will be increasingly concentrated about the immediate vicinity of the point of production. Intensified use of coke-oven gas in industrial power plants as well as of flue gas should therefore be taken into account. Again, the continued expansion of refineries and the petrochemical industry will cause the quantities of refinery gas available for electricity production to increase still further.

2.2 Competing fuels

(a) Nuclear energy

The future contribution of nuclear energy is not, as briefly pointed out above, conditioned by movements in the generating costs in nuclear plants alone.

In the "Second Illustrative Programme for Nuclear Energy", details are given of the various factors determining the movements in nuclear-generated electricity and the results to be expected by 1985.

Should the target set out in the "Illustrative Programme" a target corresponding generation requirements, be achieved, it can be taken that in 1985 the Community's installed nuclear generating capacity will amount to slightly more than 100,000 MW. This would mean that nuclear power plants would then account for about 25% of the overall capacity available in the Community.

Nuclear power plants	1970	1975	1980	1985
Installed capacity (in GW)	3.4	13	47	105
Electricity production (in TWh)	15.3	65	245	575
Mean utilization factor[1] (hours/year)	4,600	5,000	5,200	5,500

(b) Natural gas

Natural gas is an eminently suitable fuel for thermal power stations. Not only is it numbered among the so-called "clean" fuels, but also natural-gas-fired power stations incur the lowest plant and operating costs.

Although in Italy and France natural gas from domestic fields (Po Valley and Lacq respectively) was used in power stations before the beginning of the sixties, it only gained its importance at a Community level through the exploitation of the rich fields to the north of the Netherlands during the second half of the last decade. In addition to the advantages mentioned above, sales of natural gas to power stations initially benefited from the producers' interest in signing contracts covering bulk supplies at as constant a rate as possible, which meant that they offered correspondingly attractive conditions.

It is true that this situation has altered fundamentally as a result of a relative shortage of natural gas supplies. The fleeting rise in heating oil prices in 1969 intensified demand, which in turn

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[1] These mean values arise from the hypotheses concerning a reduced availability of nuclear power plants during the first three years of operation as shown on page 22 of the "Second Illustrative Programme for Nuclear Energy".

sent up the price of gas. At present, therefore, it is no longer possible to sign new, long-term contracts for the supply of natural gas on conditions which are conducive to economic consumption in thermal power stations, especially as oil prices have in the meantime risen sharply again.

If no further high-yield fields are discovered in the Community, it will be necessary to seek ways of making natural gas as profitable as possible, since domestic reserves are limited and thus there will no longer be any question of its further intensified use in power stations. At the moment one cannot tell to what extent massive imports of natural gas up to 1985 would serve to promote its use in Community power stations. There are no doubt considerable quantities of natural gas of potential importance for consumption in the Community in the USSR, North Africa and also in the North Sea. However, whether it will be possible to transport large quantities of these deposits to the Community and offer them at prices making them attractive as fuel for power stations remains in doubt.

It is, however, definite that the use of natural gas in power plants will continue to expand considerably as a result of the contracts already concluded. Between 1970 and 1975 alone the installed capacity of single-fuel, natural-gas-fired power stations will increase from 1,500 MW to 7,100 MW and that of the dual-fuel, fuel oil/natural-gas-fired plants from 8,200 MW to 17,600 MW, while electricity production based on natural gas will probably rise from 48.9 TWh to about 103 TWh.

The use of natural gas for electricity generating purposes will also increase up to 1980-1985, but at a rate already likely to be on the decline by then.

In the light of the surveys carried out as part of the "Medium-term forecast of and guideline for gas supplies in the Community" and the "Outlook for long-term energy supplies in the Community", it can be assumed that the following quantities of natural gas can be used as fuel in power stations:

Natural gas power stations (single- and multi-fuel)	1970	1975	1980	1985
Consumption of natural gas (in tce)	15	31	40	48
Gross electricity Production from natural gas (in TWh)	49	103	133	169

(c) **Hard coal**

The future trend of the use of Community coal in power stations will only be determined to a slight extent by purely economic criteria. Social and regional considerations and also certain aspects of supply policy come into play here.

Without appreciable state subsidies, which are indeed granted in all the Community countries possessing their own coalmines, the use of coal as a fuel for power stations would decline rapidly in future.

The construction of new generating capacity earmarked for Community coal will, after 1975, be restricted to Germany, where, in the context of the "electricity-from-coal" laws, new coal-fired power stations with an installed capacity of about 6,000 MW are to be built with the aid of subsidies. If, in addition to these, coal-fired power stations should be built in other Community countries after this period, they would be dual-fuel plants able to use cheap imported coal as required.

In the light of the known extraction plans and declarations of intent on the part of the Member States, it can be assumed that the amount of Community coal available as fuel for power stations will diminish continuously and appreciably during the period under examination.

Since, however, the supply of cheap steam coal could increase on the world market during the second half of the seventies, particularly because of the important potential offered by South Africa, it will be assumed for the purpose of this survey that there will be a constant overall rate of coal consumption in power stations between 1975 and 1985. This assumption is relatively optimistic as regards the future trend of steam coal imports, since the supposition is that the shrinkage of the use of domestic steam coal will be offset by a commensurate increase in imports of hard coal.

Hard coal power stations (single- and multi-fuel)	1970	1975	1980	1985
Fuel consumption (tce)	53	47	47	47
Gross electricity production (TWh)	151	141	142	142

(d) Mineral oil

It seems probable that, at least up to 1985, oil supplies for the power plant sector will offer the greatest flexibility. Even if one assumes that during the period under review there will be fuels offering lower production costs in power plants, it should be borne in mind that these fuels will either not be available in sufficient quantities (lignite, natural gas, cheap imported coal) or it will simply not be possible to create generating capacity beyond a certain limit (in the case of nuclear power plants).

This assumption also applies if future increases in heavy fuel oil prices are taken into consideration. The price of crude oil can be taken as the upper price limit for heavy fuel oil where it is used in power plants, since it is also possible for power plant operators to use crude oil directly as a fuel in their plants, the resulting conversion costs being low.

Although both crude and heavy fuel oil prices fluctuated at a very low level up to the end of 1969, thus barring the way to a more rapid increase in the use of nuclear electricity, from 1970 onwards wellhead crude prices moved sharply upwards under pressure from the producing countries. Since a shortage of tanker tonnage occurred at the same time, prices of crude supplies to the Community rose considerably.

At first the oil companies were in many cases able to pass on the increases to the customer. In some cases it was even possible to charge higher prices on top of the increases.

However, the sharp drop in freight charges for crude oil and the situation as regards competition on the heating oil market, which was intensified by a number of factors becoming involved all at the same time, led in the summer of 1971 to yet another sharp drop in prices of both heavy and light heating oil, and this state of affairs has remained basically unchanged up to the time of writing.

However, it would seem rational to take the long-term view that the trend towards higher crude prices will also give rise to increases in the price of heating oil. There is also the point that in future legislation concerning environmental protection will be tightened up, thus pushing up prices either via the production of low-sulphur-content oil or via the necessary installation in oil-fired power stations of suitable equipment for removing the sulphur from their off-gases.

It nevertheless seems reasonable to assume that, during the period under examination, adequate quantities of heavy fuel oil will be available at prices which do not constitute a barrier to the continued growth of its use, at least up to 1985. Further movements in heavy fuel oil prices will, of course, also play a decisive part in determining the rate of expansion of nuclear generating capacity in the Community.

In the light of the factors currently in evidence, the following trends can be forecast:

Oil-fired power stations (single- and multi-fuel)	1970	1975	1980	1985
Fuel consumption (tce)	49	85	123	167
Gross electricity production (in TWh)	154	281	413	562

The foregoing detailed recapitulation of movements by individual fuels is summarized in the following table:

Long-term trend of gross electricity production in the Community

	Share								Mean annual growth rate				Heat equivalent			
	1970		1975		1980		1985		1970/75	1975/80	1980/85	1970/85	1970	1975	1980	1985
	TWh	%	TWh	%	TWh	%	TWh	%	%				Mio tce			
A. Gross production (B + C)	580	100,0	840	100,0	1.210	100,0	1.740	100,0	7,7	7,6	7,5	7,6	195	271	379	535
B. Privileged sources of energy (1 + 2 + 3)	211	36,5	250	29,8	277	22,9	292	16,8	3,4	2,1	1,1	2,2	73	87	94	99
1. Water power	117	20,2	127	15,1	139	11,5	149	8,6	1,6	1,8	1,4	1,6	38	42	44	47
- primary	114	19,6	119	14,2	129	10,7	137	7,9	0,8	1,6	1,2	1,2	36	38	40	42
- pumped storage power stations	3	0,6	8	0,9	10	0,8	12	0,7	20,1	4,6	3,7	9,7	2	4	4	5
2. Geothermal heat	3	0,5	3	0,4	3	0,2	3	0,2	-	-	-	-	1	1	1	1
3. Thermal energy	91	15,8	120	14,3	135	11,2	140	8,0	5,5	2,4	0,7	2,9	34	44	49	51
- lignite	65	11,3	88	10,5	100	8,3	100	5,7	6,1	2,6	-	2,9	24	32	36	36
- derivative gases, etc..	26	4,5	32	3,8	35	2,9	40	2,3	4,1	1,8	2,7	2,9	10	12	13	15
C. Non-privileged sources of energy (4 + 5)	369	63,5	590	70,2	933	77,1	1.448	83,2	9,9	9,6	9,2	9,5	122	184	285	436
4. Conventional thermal energy	354	60.9	525	62,5	688	56,9	873	50,2	8,2	5,6	4,9	6,2	117	163	210	262
- hard coal	151	26,0	141	16,7	142	11,8	142	8,2	-1,4	0,1	-	-0,4	53	47	47	47
- mineral oil products	154	26,5	281	33,5	413	34,1	562	32,3	12,8	8,0	6,4	9,0	49	85	123	167
- natural gas	49	8,4	103	12,3	133	11,0	169	9,7	16,1	5,2	4,9	8,6	15	31	40	48
5. Nuclear energy	15	2,6	65	7,7	245	20,2	575	33,0	33,5	30,4	18,6	27,5	5	21	75	174
Conventional thermal energy, total (3 + 4)	445	76,7	645	76,8	823	68,1	1.013	58,2	7,7	5,0	4,2	5,6	151	207	259	313

3. Concluding remarks

If the long-term trends were to crystallize in the form shown above, the outcome would be that in 1985 nuclear energy would account for the greater part of generating capacity, while the use of fuel oil would already be in slight decline around 1980. Nevertheless, roughly one-third of electricity production would still be based on mineral oil products in 1985 and the absolute quantity of oil products consumed in power plants would have risen to almost 170 million tce.

The dependence on imports brought about by the large quantities of oil used to generate electricity in the Community will certainly be slightly increased by imports of coal, but their help in dispersing the regions of origin represents a positive factor from the standpoint of the dependability of supply.

At the moment it is still doubtful as to whether imported coal will succeed in offsetting the declining use of indigenous coal in power stations as assumed in this survey. An essential condition would be for the electricity producers to have dependable, continous, long-term sources of supply at competive conditions so that they can undertake on this basis the necessary investments in their power plants suitable for coal-firing.

However, if coal's contribution did not reach the assumed proportions, the probable consequence would be that the demand for oil products on the part of the electricity producers, and therefore their dependence on this type of fuel, would be further intensified.

The extent to which the use of imported natural gas could lead to a diversification of supplies to power plants is as yet not clear enough to afford an idea as regards increased dependability of supply.

Unless large imports of natural gas into the Community will bring about a return to a buyer's market, imports from non-member countries, which are burdened with considerable transport costs, would probably be directed, as indigenous supplies become more and more scarce, to areas able to make more profitable use of such gas than power plants.

The possibility to come back if necessary to indigenous fuels such as coal or lignite, would at best be given in a limited area only.

It should be assumed in the light of the trend visible at the present time that the extraction of domestic coal will decline in all the Community countries as a result of the adverse cost situation, thus causing a reduction in offerings of power plant coal too. The process could be slowed down by massive subsidies.

The scope for an expansion of lignite-fired electricity production is also fairly limited, since there are no further deposits which could be worked cheaply to provide fuel on an economic basis for additional power stations.

The foregoing arguments point to the inevitable conclusions that only via intensified use of nuclear energy for electricity-generating purposes an effective contribution can be made towards reducing the high level of dependence of the electricity supply sector on mineral oil.

A decision on priority for limiting the electricity producers' dependence on oil comes within the political sphere. Should such a decision be taken, however, the outcome would be an urgent need to fulfil the conditions for a speeding-up of the expansion of nuclear generating capacity as set out in the "Second Illustrative Programme for Nuclear Electricity Production".

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Structure of gross electricity production

Breakdown to types of fuel

1950 - 1970

(%)

		Water power	Geothermal heat	Nuclear energy	Convention thermal energy							Grand total
					Hard coal	Lignite	Mineral oil products	Natural gas	Derivative gases	Other	Total	
Community	1950	38,0	1,0	-	44,0	9,2	2,5	0,1	5,0	0,2	61,0	100
	1955	35,4	1,0	-	41,6	11,8	3,4	1,1	5,4	0,3	63,6	100
	1960	35,4	0,7	0,1	39,1	12,2	5,2	2,3	4,8	0,2	63,8	100
	1965	25,9	0,6	1,1	36,2	12,1	17,9	2,1	3,4	0,6	72,3	100
	1970	20,2	0,5	2,6	26,0	11,3	26,5	8,4	3,7	0,8	76,7	100
Germany	1950	18,5	-	-	52,3	23,5	0,5	-	4,7	0,5	81,5	100
	1955	14,9	-	-	50,6	28,0	1,1	-	4,8	0,6	85,1	100
	1960	10,9	-	-	54,0	27,7	2,7	0,1	4,1	0,5	89,1	100
	1965	8,9	-	0,1	49,1	27,1	9,9	1,4	2,5	1,0	91,0	100
	1970	7,3	-	2,5	39,4	25,3	15,0	5,5	3,7	1,2	90,2	100
France	1950	47,4	-	-	41,5	0,9	4,3	-	5,9	-	52,6	100
	1955	50,0	-	-	36,9	1,1	5,0	-	7,0	-	50,0	100
	1960	54,5	-	0,2	29,0	1,3	3,5	5,3	6,2	-	45,3	100
	1965	44,2	-	1,0	34,0	1,8	11,3	3,2	4,5	-	54,8	100
	1970	38,9	-	3,9	25,2	1,8	21,7	4,5	3,6	0,4	57,2	100
Italy	1950	87,5	5,2	-	4,4	-	2,2	0,5	0,2	-	7,3	100
	1955	80,8	4,9	-	2,8	-	5,5	5,4	0,6	0	14,3	100
	1960	82,0	3,7	-	1,7	1,2	6,7	3,8	0,9	0	14,3	100
	1965	51,9	3,1	4,2	1,7	1,2	32,3	3,1	1,6	0,9	40,8	100
	1970	35,2	2,3	2,7	2,8	1,1	48,3	4,9	1,6	1,1	59,8	100
Netherlands	1950	-	-	-	87,0	-	10,3	-	2,3	0,4	100,0	100
	1955	-	-	-	91,4	-	5,8	-	2,4	0,4	100,0	100
	1960	-	-	-	76,3	-	19,6	0,9	2,9	0,3	100,0	100
	1965	-	-	-	49,6	-	46,5	0,6	2,7	0,6	100,0	100
	1970	-	-	0,9	16,4	-	32,6	46,7	3,4	0	99,1	100
Belgium	1950	0,7	-	-	88,2	-	0,2	-	10,9	-	99,3	100
	1955	1,1	-	-	83,2	-	3,3	-	12,4	-	98,9	100
	1960	1,1	-	-	74,7	-	12,2	0,3	11,7	-	98,9	100
	1965	1,3	-	-	64,2	-	26,1	0,1	8,3	-	98,7	100
	1970	0,8	-	0,2	25,9	-	50,4	13,3	9,0	0,4	99,0	100
Luxemburg	1950	0,3	-	-	(1,9)	-	(0,5)	-	(97,3)	-	99,7	100
	1955	0,3	-	-	(1,8)	-	(0,4)	-	(97,5)	-	99,7	100
	1960	1,4	-	-	(4,8)	-	(1,0)	-	(92,8)	-	98,6	100
	1965	39,7	-	-	2,3	-	7,2	-	50,8	-	60,3	100
	1970	41,3	-	-	0,6	-	11,4	0,1	46,3	0,3	58,7	100

Installed generating capacity in the Community countries

1950 - 1970

(year-end figures)

(in MW)

Year	Germany	France	Italy	Netherlands	Belgium	Luxemburg	Community	Germany	France	Italy	Netherlands	Belgium	Luxemburg	Community
	1. Hydro-electric power stations							2. Geothermal power stations						
1950	2.460	5.352	7.169	-	27	1	15.009	-	-	220	-	-	-	220
1955	3.120	7.954	9.896	-	53	1	21.024	-	-	250	-	-	-	250
1960	3.630	10.261	12.612	-	53	16	26.572	-	-	309	-	-	-	309
1965	4.427	12.683	14.297	-	65	929	32.401	-	-	339	-	-	-	339
1970	5.114	15.219	14.962	-	65	929	36.289	-	-	402	-	-	-	402
	3. Conventional thermal power stations							4. Nuclear power plants						
1950	11.910	6.590	1.175	2.512	2.913	135	25.235	-	-	-	-	-	-	-
1955	18.230	8.214	2.275	4.194	3.496	202	36.611	-	-	-	-	-	-	-
1960	27.090	11.534	4.765	5.452	4.338	259	53.438	-	96	-	-	-	-	96
1965	38.807	15.110	10.075	7.528	5.221	258	76.999	16	428	642	-	11	-	1.097
1970	47.382	21.819	17.242	10.685	6.660	229	104.017	888	1.781	642	54	11	-	3.376

ELECTRICITY

Gross production, broken down according to producers

1950 - 1970

(%)

		Total produced	Public utilities					Independents		
			Water power	Geothermal	Nuclear	Conventional thermal	TOTAL	Water power	Conventional thermal	TOTAL
Community	1950	100	31,9	1,0	-	34,2	67,1	6,1	26,8	32,9
	1955	100	30,7	1,0	-	34,6	66,3	4,7	29,0	33,7
	1960	100	30,7	0,7	0,1	36,4	67,9	4,7	27,4	[illegible]
	1965	100	22,8	0,6	1,2	44,8	69,4	3,1	27,5	30,6
	1970	100	18,0	0,5	2,6	52,2	73,3	2,2	24,5	26,7
Germany	1950	100	15,5	-	-	44,3	59,8	3,0	37,2	40,2
	1955	100	12,7	-	-	48,0	60,7	2,3	37,0	39,3
	1960	100	9,4	-	-	51,9	61,3	1,5	37,2	38,7
	1965	100	7,8	-	0,1	55,2	63,1	1,1	35,8	36,9
	1970	100	6,4	-	2,5	58,1	67,0	0,9	32,1	33,0
France	1950	100	41,9	-	-	27,4	69,3	5,5	25,2	30,7
	1955	100	46,1	-	-	18,7	64,8	3,9	31,3	35,2
	1960	100	50,5	-	0,2	21,0	71,7	4,0	24,3	28,3
	1965	100	41,4	-	1,0	32,7	75,1	2,8	22,1	24,9
	1970	100	37,1	-	3,9	39,3	80,3	1,9	17,8	19,7
Italy	1950	100	70,5	5,2	-	4,6	80,3	17,0	2,7	19,7
	1955	100	66,8	4,9	-	10,6	82,3	14,0	3,7	17,7
	1960	100	66,8	3,7	-	7,5	78,0	15,2	6,8	22,0
	1965	100	42,3	3,1	4,2	24,7	74,3	9,6	16,1	25,7
	1970	100	28,6	2,3	2,7	42,0	75,6	6,6	17,8	24,4
Netherlands	1950	100	-	-	-	73,3	73,3	-	26,7	26,7
	1955	100	-	-	-	77,0	77,0	-	23,0	23,0
	1960	100	-	-	-	77,2	77,2	-	22,8	22,8
	1965	100	-	-	-	80,8	80,8	-	19,2	19,2
	1970	100	-	-	0,9	83,7	84,6	-	15,4	15,4
Belgium	1950	100	0,7	-	-	60,3	61,0	0	39,0	39,0
	1955	100	1,1	-	-	53,6	54,7	0	45,3	45,3
	1960	100	1,1	-	-	57,3	58,4	0	41,6	41,6
	1965	100	1,2	-	-	60,5	61,7	0	38,3	38,3
	1970	100	0,8	-	0,2	67,6	68,6	0	31,4	31,4
Luxemburg	1950	100	0,3	-	-	-	0,3	-	99,7	99,7
	1955	100	0,3	-	-	-	0,3	-	99,7	99,7
	1960	100	1,4	-	-	-	1,4	-	98,6	98,6
	1965	100	39,7	-	-	-	39,7	-	60,3	60,3
	1970	100	41,3	-	-	-	41,3	-	58,7	58,7

ELECTRICITY

Gross industrial self-production, broken down according to producers

1950 - 1970

(GWh)

		Total produced	Joint power stations (1)	Coal mining	Refineries	Iron- founding industry	Chemicals	Non ferrous metals	Paper	Textiles	Other	Railways
Community	1950	40.464	365									
	1955	65.710	1.132									
	1960	91.221	1.582	35.321	1.764	15.584	18.064		6.099	3.488	4.967	4.352
	1965	125.810	3.547	44.808	4.366	18.952	29.284		8.324	3.731	8.109	4.679
	1970	154.757	4.542	47.950	6.789	24.136	34.907	4.382	11.123	4.037	10.323	6.562
Germany	1950	18.797										
	1955	31.667										
	1960	46.069		21.949	1.050	5.257	9.314	308	3.063	1.256	2.269	1.603
	1965	63.550		28.106	2.427	5.822	12.149	2.577	3.690	1.317	4.545	2.917
	1970	80.021		30.988	3.090	10.891	15.802	2.422	5.295	1.419	5.207	4.901
France	1950	10.646										
	1955	18.257										
	1960	21.289		9.514	275	5.484	1.490		1.477	567	725	1.757
	1965	26.387		13.101	820	5.484	1.347		1.995	533	1.345	1.762
	1970	28.930		12.999	1.499	5.408	2.420	286	2.367	605	1.685	1.661
Italy	1950	4.853		-								
	1955	6.735		-								
	1960	12.360		-	160	1.698	3.674	1.828	868	1.445	1.695	992
	1965	21.351		-	631	4.427	10.155	1.116	1.623	1.677	1.722	-
	1970	28.693		-	1.586	4.875	14.186	1.233	2.352	1.842	2.619	-
Netherlands	1950	1.978		1.343								
	1955	2.577		1.520								
	1960	3.760		1.790	/229/	(156)	750	.	509	126	200	
	1965	4.812		1.922	381	(117)	1.182	.	749	108	353	-
	1970	6.273		2.182	376	(175)	2.002	. 11	837	59	631	-
Belgium	1950	3.455	365									
	1955	5.306	1,132									
	1960	6.299	1.582	2.068	50	1.551	276	424	182	94	72	-
	1965	8.310	3.547	1.679	107	1.768	249	465	267	96	132	-
	1970	9.579	4.542	1.781	238	1.627	402	430	272	112	175	-
Luxemburg	1950	735		-	-							
	1955	1.168		-	-							
	1960	1.444		-	-	1.438	-	-	-	-	6	-
	1965	1.390		-	-	1.334	44	-	-	-	12	-
	1970	1.261		-	-	1.160	95	-	-	-	6	-

(1) Belgian joint power stations belonging to mining and iron-and-steel industries.

Structure of generating capacity in the Community
Installed capacity 1970 - 1975

(1.000 MW)

Type of power station and fuel used	Community (1)		Germany		France		Italy		Netherlands		Belgium		Luxemburg	
	1970	1975	1970	1975	1970	1975	1970	1975	1970	1975	1970	1975	1970	1975
A. Total	(144,1)	(203,7)	(53,4)	(73,8)	(38,8)	(49,7)	(33,2)	(50,5)	(10,8)	(16,5)	(6,8)	(11,8)	(1,1)	(1,3)
B. Privileged fuels														
1. Water power	36,3	42,5	5,1	6,1	15,2	17,0	15,0	17,7	-	-	0,1	0,5	0,9	1,1
a) river	12,9	13,7	2,4	2,5	6,2	6,6	4,2	4,4	-	-	0,1	0,1	0,0	0,0
b) storage	23,4	28,8	2,7	3,6	9,0	10,4	10,8	13,7	-	-	0,0	0,4	0,9	1,1
2. Geothermal heat	0,4	0,4	-	-	-	-	0,4	0,4	-	-	-	-	-	-
3. Conventional thermal energy	17,1	20,3	12,7	15,9	2,8	2,8	1,0	1,0	0,1	0,1	0,5	0,5	0,2	0,2
- brown coal, single-fuel	9,6	12,8	9,3	12,5	0,2	0,2	0,1	0,1	-	-	-	-	-	-
- brown coal, multi-fuel	0,6	0,6	0,2	0,2	-	-	0,4	0,4	-	-	-	-	-	-
- derivative gases etc., single-fuel	2,5	2,5	1,2	1,2	0,6	0,6	0,5	0,5	0,1	0,1	-	-	0,1	0,1
- derivative gases, multi-fuel	4,4	4,4	2,0	2,0	2,0	2,0	-	-	0,0	0,0	0,5	0,5	0,1	0,1
C. Non-privileged fuels														
4. Conventional thermal energy	86,9	127,7	34,7	46,5	19,0	25,8	16,2	29,9	10,6	15,9	6,2	9,5	0,0	0,0
a) single - fuel	58,3	82,9	25,7	35,6	14,8	21,3	9,0	17,0	4,9	4,6	3,8	4,3	0,0	0,0
- hard coal	36,0	28,2	21,2	19,7	10,2	4,4	0,2	0,1	2,9	2,5	1,4	1,4	-	-
- mineral oil products	20,8	47,6	4,3	10,1	4,1	16,4	8,6	16,7	1,4	1,5	2,4	2,9	0,0	0,0
- natural gas	1,5	7,1	0,2	5,8	0,5	0,5	0,2	0,2	0,6	0,6	0,0	0,0	-	-
b) multi-fuel	28,6	44,8	9,0	10,9	4,2	4,5	7,2	12,9	5,7	11,3	2,4	5,2		
- coal/oil	16,7	22,9	6,8	6,8	2,8	2,8	4,1	9,8	1,9	1,6	0,9	1,7	-	-
- coal/natural gas	1,7	1,7	1,1	1,1	0,3	0,3	-	-	0,1	0,1	0,2	0,2	-	-
- oil/natural gas	8,2	17,6	1,0	2,9	0,9	0,9	2,5	2,5	3,4	9,3	0,4	2,0	-	-
- coal/oil/natural gas	2,0	2,6	0,1	0,1	0,2	0,5	0,6	0,6	0,3	0,3	0,8	1,1	-	-
5. Nuclear power stations	3,4	12,7	0,9	5,3	1,8	4,1	0,6	1,5	0,1	0,5	0,0	1,3	-	-
Conventional thermal energy (3 + 4)	104,0	148,1	47,4	62,1	21,8	28,6	17,2	30,9	10,7	16,0	6,7	10,0	0,2	0,2
of which - designed for single-fuel	70,3	98,2	36,2	49,0	15,6	22,1	9,6	17,6	5,0	4,7	3,8	4,3	0,1	0,1
- designed for multi-fuel	33,7	49,9	11,2	13,1	6,2	6,5	7,6	13,3	5,7	11,3	2,9	5,7	0,1	0,1

(1) Differences between the figures for the Community and those for the Member States are rounded off.

(2) Including less recent lignite, lignite briquettes and low, indivisible levels of production.

Trend of electricity production in the Community countries, 1970 - 1975

(TWh)

Type of power station and fuel	Community (1)		Germany		France		Italy		Netherlands		Belgium		Luxemburg	
	1970	1975	1970	1975	1970	1975	1970	1975	1970	1975	1970	1975	1970	1975
A. Gross production (B + C)	580,4	840,0	242,6	335,3	146,8	210,1	117,4	184,1	40,9	61,2	30,5	47,0	2,1	2,3
B. Privileged fuels (1 + 2 + 3)	211,8	250,1	91,1	114,4	65,7	66,8	48,5	59,9	1,4	1,5	3,1	5,5	1,9	2,0
1. Water power	117,4	127,0	17,7	18,0	57,2	57,0	41,3	50,0	-	-	0,2	1,0	0,9	1,0
- primary	114,2	119,0	16,3	15,9	57,1	56,3	40,4	46,8	-	-	0,2	0,2	0,1	0,1
- pumped-storge stations	3,2	8,0	1,4	2,1	0,1	0,7	0,9	3,2	-	-	-	0,8	0,8	0,9
2. Geothermal heat	2,7	3,0	-	-	-	-	2,7	3,0	-	-	-	-	-	-
3. Thermal energy	91,7	120,1	73,4	96,4	8,5	9,8	4,5	6,9	1,4	1,5	2,9	4,5	1,0	1,0
- lignite	65,5	88,0	61,5	82,5	2,7	3,0	1,3	2,5	-	-	-	-	-	-
- derivative gases, etc..	26,2	32,1	11,9	13,9	5,8	6,8	3,2	4,4	1,4	1,5	2,9	4,5	1,0	1,0
C. Non-privileged fuels (4 + 5)	368,6	589,9	151,5	220,9	81,1	143,3	68,9	124,2	39,5	59,7	27,4	41,5	0,2	0,3
4. Conventional heat energy	353,3	524,9	145,5	193,9	75,4	123,3	65,7	116,2	39,1	55,7	27,3	35,5	0,2	0,3
- hard coal	150,6	140,5	95,7	100,0	37,0	24,0	3,3	7,5	6,7	1,0	7,9	8,0	0,0	-
- oil	153,8	281,3	36,4	58,4	31,8	89,6	56,7	101,7	13,3	15,6	15,4	15,7	0,2	0,3
- natural gas	48,9	103,1	13,4	35,5	6,6	9,7	5,7	7,0	19,1	39,1	4,0	11,8	0,0	0,0
5. Nuclear energy	15,3	65,0	6,0	27,0	5,7	20,0	3,2	8,0	0,4	4,0	0,1	6,0	-	-

(1) Differences between the figures for the Community and those for the Member States are rounded off.

(2) Including less recent lignite and lignite briquettes.

XVII/341/2/71-E
Orig.F

COMMISSION
OF THE
EUROPEAN COMMUNITIES

SECOND TARGET NUCLEAR PROGRAMME FOR THE COMMUNITY

ANNEX III

LIGHT-WATER NUCLEAR POWER PLANTS

DRAFT

(1 March 1972)

ANNEX III

Second Illustrative Programme

LIGHT-WATER NUCLEAR POWER PLANTS

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LIST OF INITIALS AND ABBREVIATIONS

AEG	Allgemeine Elektrizitäts Gesellschaft
AGR	Advanced Gas Cooled Reactor
BWR	Boiling Water Reactor
EEC	European Economic Community
EdF	Electricité de France
GG	Gas-graphite
KWU	Kraftwerk-Union
PWR	Pressurized Water Reactor
USAEC	United States Atomic Energy Commission

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1. INTRODUCTION

1.1 Present status of power plants in the Community and the United States (in operation, under construction and at the project stage)

The light-water type of nuclear power plant is currently undergoing considerable industrial development in its boiling water (BWR) and pressurized water (PWR) variants.

In the United States alone, after the record year in 1967 when orders were placed for 25,800 MWe, at 1 January 1972 133 power plants were on order, in operation or under construction, representing about 113,000 MWe.

Nearly all US nuclear power plants are of the light-water type.

A similar development of light-water plants has started in the Community, although on a smaller scale, whereas at 31 December 1970 the gas-graphite and light-water power plants in operation each represented 50% of the installed capacity, by the end of 1977 light-water plants under construction will account for more than 80% of the total installed capacity.

The balance between the light-water and gas-graphite types (as adopted in the "First Target Programme") will give way to a predominance of light-water reactors.

Tables 1.1 and 1.2 below give the status at 1 January 1972 of the light-water power plants (in operation, under construction and at the project stage) in the United States and the Community.

1.2 Foreseeable development of nuclear power plant installation in the Community up to 1985

The current technology of advanced high-temperature and fast reactors (see Parts III and IV) is such that their commercial introduction can be expected to have little influence on the industrial development of nuclear power plants until 1985. It can therefore be predicted that almost all the power plants ordered up to that date will be of the light-water type.

On the basis of the estimate of the increase in the number of nuclear power plants in the Community (see Part I) and assuming a period of five years between the ordering and commissioning of a power plant, the average annual volume of orders which can be anticipated is about 5-8 units of 800 to 1,200 MWe capacity for the period 1971-75 and 6-9 units of 1,200 to 2,000 MWe for the period 1976-80.

The estimates of orders quoted above represents average values which take no account of possible exports or the enlargement of the Community.

On the strength of the great commercial boost given this type of reactor by the American and European nuclear industries, it is now thought that by 1980 the light-water plants in service will total 45,000 MWe in the Community and more than 150,000 MWe in the United States.

TABLE 1.1

BREAKDOWN OF LIGHT-WATER PLANTS IN OPERATION, UNDER CONSTRUCTION OR ON ORDER FROM THE LEADING US CONSTRUCTORS (number/power in MWe)

(according to year ordered and excluding exports)

Company	1953/1955	1956	1958	1959	1962	1963	1965	1966	1967	1968	1969	1970	1971	Totals
General Electric	1/200		1/68	1/70		3/1805	3/1992	9/7462	8/6931	9/8174	3/3040	5/5186	6/6891	49/41.825
Westinghouse	1/90	1/175			1/575	2/892	3/1945	6/4867	13/10.657	5/5286	3/3021	5/4643	14/13.54	54/45.695
Babcock & Wilcox	1/265							3/2558	5/4204	3/2182		2/2460	2/1880	16/13.549
Combustion Engin.								2/1157	5/3988	1/1190	1/1190	4/4315	1/ 890	14/11.730
Total No.	3	1	1	1	1	5	6	20	31	18	7	16	23	133
Power MWe	555	175	68	70	575	2697	3937	16,044	25.780	16,832	7251	16,604	23,205	112,799

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TABLE 1.2

BREAKDOWN OF LIGHT-WATER PLANTS IN OPERATION, UNDER CONSTRUCTION OR ON ORDER IN THE COMMUNITY AT 1 January 1972
(according to year ordered)

Year ordered	Plant	Type	No./net etc. power in MWe	Owner/ operator	Reactor supplier
1956	BR3	PWR	1/10		West.
1958	Kahl (VAK)	BWR	15	RWE-Bayernwerk	GE/AEG
	Trino Vercellese	PWR	257	ENEL	West.
			2/272		
1959	Garigliano	BWR	1/150	ENEL	GE
1961	Chooz	PWR	1/266	EdF/Centre & Sud	West./ACECO/ FRAMATOME
1962	Gundremmingen	PWR	1/237	KRB	AEG/GE
1963	Dodewaard	BWR	1/ 52	GKN	GE
1964	Lingen	BWR	1/174*	KWL	AEG
1965	Grosswelzheim-HDR	BWR	22	GfK	AEG
	Obrigheim	PWR	328**	KWO	Siemens
			2/350		
1967	Würgassen	BWR	640	Preussenelektra	AEG
	Stade	PWR	630	KWS	Siemens
			2/1270		
1969	Borssele	PWR	450	PZEM	KWU (Siemens)
	Doel	PWR	2 x 390		ACECO
	Tihange	PWR	870	SEMO	ACECO/SFAC/ FRAMATOME
	Biblis A	PWR	1150	RWE	KWU (Siemens)
			5/3250		
1970	Caorso	BWR	783	ENEL	AMN/GE
	Brunsbüttel	BWR	770	KKB	KWU (AEG)
	Philippsburg I	BWR	864	KKP	KWU (AEG)
	Fessenheim	PWR	890	EdF	SFAC/FRAMATOME
			4/3307		
1971	Bugey II and III	PWR	2 x 930	EdF	Creusot Loire Framatome
	Biblis B	PWR	1178	RWE	KWU (Siemens)
	Unterweser	PWR	1230	Pressenelektra/NWK	KWU (Siemens)
	Phillippsburg II	BWR	864	KKP	KWU (AEG)
	Ohu	BWR	870	KKI	KWU (AEG)
	Neckarwestheim	PWR	775	BKN	KWU (Siemens)
			7/[illegible]		

*Taking into account conventional superheating, total power is 240 MWe
**Value increased to 345 MWe from 2 December 1969.

2. Trend and outlook of technology

In view of the considerable industrial development in progress in light-water power plants throughout the world, it is of interest to analyse the main technological stages which have in the past marked this development and to try and foresee the main progress it will make in the next few years.

In the succeeding sections, the main technical parameters affecting the cost of energy produced in BWR and PWR plants will be reviewed.

2.1 Size and standardization

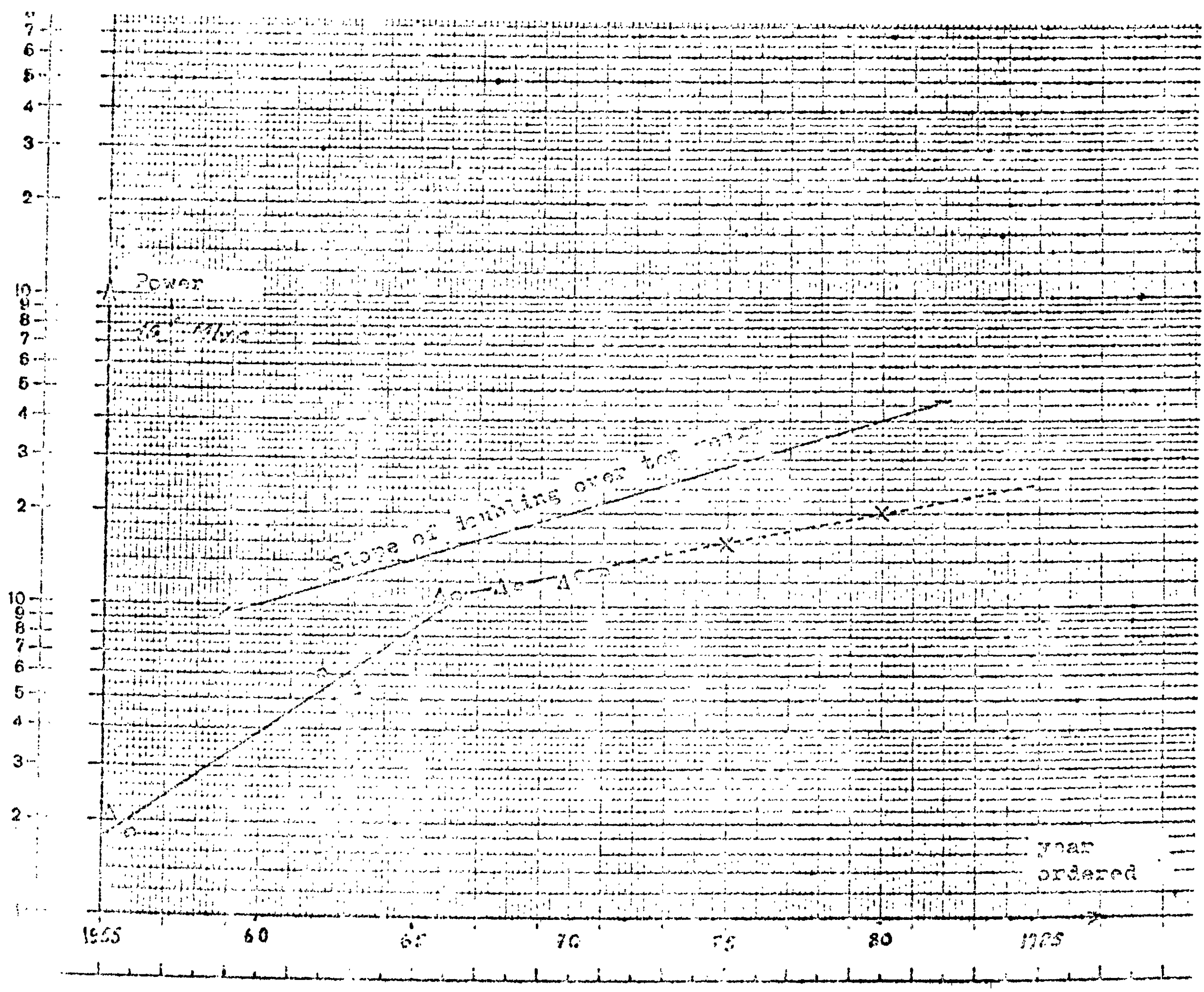
2.1.1 Size

Since their commercial introduction, the development of BWR and PWR plants has to a great extent kept pace with their size. The principal stages in this development in the United States are given in the table below and in Fig. 2.1 against the year ordered.

Year ordered		1955	1956	1962	1963	1965	1966	1968	1970
Net power MWe	BWR	200	-	-	515	715	1064	1115	1150
	PWR	-	175	575	-	873	1060	1124	1250

Fig. 2.1.

Trend of maximum size of light-water nuclear power plants



◦ PWR

△ BWR

x prevision

A similar development has taken place in the Community, but with a lag of a few years behind the United States, the 1,100 MWe stage not being reached until 1970. Two periods can be observed in the development, the first of which, up to 1966, corresponds to the industrial development of the light-water type of plant and marks the achievement of competitivity.

Though there has been a certain slowing down since 1966 in the growth of plant size, some constructors are already contemplating the possibility of offering much larger plants. Thus the opening of new departments at constructors' works and new methods of on-site assembly of large plant items (such as pressure vessels, steam generators, sets) might warrant the prediction of orders, from 1975, for units of 1,600 MWe, or even 2,000-3,000 MWe during the 1980's.

However, though no major technological problem appears to prevent the increase in plant size, the consequent increase in construction time might slow down this development, mainly because of the high cost of interest during construction. The development and linking-up of distribution networks would also have to keep pace with this growth in unit capacity.

2.1.2 Standardization

The large number of orders for plants received by American constructors has propted them to standardize production and offer plants of well-defined size, with the intention of:

(a) avoiding the drawing-up of new construction plans for each order;

(b) using their manufacturing facilities efficiently.

Thus, in the United States:

General Electric is offering four sizes of BWR plant, with approximate net capacities of 1,075, 762, 515 and 417 MWe;

Westinghouse is offering three sizes according to the number of cooling loops (a loop comprises a pump, a steam generator and the interconnecting circuits). The approximate net capacities are 590 MWe for two loops, 880 MWe for three loops and 1,140-1,250 MWe for four loops;

Babcock and Wilcox is offering two sizes of PWR plant, one of 800-840 MWe net with two loops and the other of 1,150 MWe with three loops (each loop comprises a steam generator, two pumps and the interconnecting circuits).

In the Community, the standardization of reactor size is less advanced than in the United States.

Siemens has now adopted three standard capacities, 470, 810 and 1,240 MWe, for reactors with two, three and four loops respectively. For power plants ordered after 1975, Siemens is planning to step up the capacity to 400 MWe per loop, with two pumps to a loop.

The present design of AEG boiling water reactors, using certain relatively small plant items of a modular type (fuel elements, steam separators, recirculation pumps, etc.), and the absence of external recirculation loops permit great flexibility in the choice of plant capacity.

Some plant items have been standardized, such as fuel elements, control rods and drives, in-core instrumentation, recirculation pumps, steam generators, steam driers and separators, and gaseous and liquid waste processing installations.

2.2. Performances

Since the first light-water plants were constructed, substantial technical progress has enabled their performance to be improved.

2.2.1 Boiling water reactors

The significant guarantees affecting the costs of the power plant and the fuel cycle are:

(a) the power developed per unit weight of uranium,

(b) the power developed per unit volume of core,

(c) the power extracted per unit volume of coolant,

(d) the steam quality at outlet from core,

(e) the fuel burnup,

(f) the.specific flow rate of the coolant.

The trend in these main parameters is given for power plants in the United States and the Community in Table 2.2.1.attached.

The increase in specific power, in particular, was made possible by:

an improved knowledge of the heat transfer correlations,

a reduction in the margins of uncertainty,

the optimization of fuelling programmes,

the use of various enrichments in the fuel element, leading to lower power peak factors.

The increase by a factor of more than 2 in the specific power (11.0 kW/kg U, or 28.3 kW/l, for the Garigliano power plant and 22.1 kW/kg U, or 50.6 kW/l, for the Brunsbüttel power plant) permitted a substantial reduction in:

- the specific investment in in-core fuel,
- dimensions (and hence weight) of reactor pressure vessels.

During the next few years, a further increase of the order of 20% in the specific power of boiling water reactors is to be anticipated. This increase might, for example, be achieved by allowing melting in the centre of UO_2 pellets during certain transient modes of operation, and by overstepping the critical heal flux. It was demonstrated by General Electric, under the Euratom/United States Research and Development Programme, that little or no damage is caused to the fuel as a result.

TABLE 2.2.1

BOILING WATER REACTORS

(project values)

		1st generation Dresden-1	2nd generation Gundremmingden	Present generation Philippsburg-1	Future generation
Year ordered		1955	1962	1970	1975
Net electric power	MWe	180	237	864	1,600
Net efficiency of plant	%	28.7	29.6	33.6	
Specific power	kW/kg U	11.5	17.2	22.4	30
Mean density of power	kW/l	28.9	40.9	51.1	70
Mean burnup of equilibrium	MWd/tU	11,000	16,500	27,500	27,500
Power per m^3 of coolant	MW/m^3	3.69	5.45		
Average steam quality at core outlet	% wt	5.17	8.36	13.6	
Specific coolant flow rate	t/h/MW		15.3	14.75	

More advanced designs, accepting central melting of UO_2 for the most heavily loaded fuel pins even during steady operation and using devices to encourage turbulent flow of the coolant in the fuel, are currently under study. They might lead in the 1980's to a new generation of BWR's with enhanced performances.

The coolant inventory per unit of power output from the core was reduced by a factor of about 2 by the adoption of:

- a water steam separator inside the reactor pressure vessel,
- a single steam cycle, and
- pumps incorporated in the reactor pressure vessel (jet pumps in the case of General Electric Co. and axial pumps in the case of AEG).

This led to an appreciable reduction in the leaktight containment volume and to the development of a new type of containment. This new pressure-suppression type was used for the first time at Humboldt Bay in the United States and at Dodewaard in the Community. Because the water volume in the primary circuit is so small, AEG was able to use a very compact design, in which the pressure-suppression system is housed inside the spherical containment.

The increase in average burnup from 11,000 to more than 25,000 MWd/t has substantially lowered the costs of the fuel cycle; however, at 27,500 MWd/t, this rate has now reached the economic optimum and makes further spectacular increases unlikely in the coming years.

2.2.2 Pressurized water reactors

The significant parameters affecting the costs of the power plant and fuel cycle are very similar to those already described for boiling water reactors:

(a) the power developed per unit weight of uranium,

(b) the power developed per unit volume of core,

(c) the power extracted per unit volume of coolant,

(d) the fuel burnup,

(e) the specific flow rate of the coolant.

The trend of these main parameters is given in Table 2.2.2, which also shows the principal steam characteristics on which the secondary part of pressurized water plants depends.

The economic significance of the trend of these main parameters is similar to that described for boiling water reactors. This trend, in particular, has been brought about mainly by the use of chemical control for the slow variations in reactivity and by the use of cluster-type control rods.

In this context, loading by multiple zones and an improved knowledge of power distribution in the core should also be mentioned.

As in the case of boiling water reactors, an increase in power density of the order of 15-20% seems possible without the need for new techniques.

Together with the above performance improvements, the increase in the reactor temperature differential permitted a reduction in steam generator surface and in coolant flow, giving rise to a substantial saving in the cost of steam generators, pumps and containment. In addition, the augmented mean coolant temperature led to higher steam pressure and hence to an improvement in the efficiency of the plant.

Tables 2.2.1 and 2.2.2 also give an idea of the basic characteristics of future generations of water reactors.

2.3 Design

In conjunction with the increase in light-water reactor performance, the design of power plants has improved, sometimes significantly. The following are the main developments.

2.3.1 Reactivity control

Ideas on reactivity control in boiling water reactors have progressed very little, although the number of control rods per MWe has decreased from 0.50 in the case of Dresden-1 to 0.17 in the case of comtemporary plants. However, in some versions now being built, there are plans to use burnable poisons as an additional means of reactivity control.

Control rod drive mechanisms are entirely hydraulic in the case of General Electric, whereas those developed by AEG are both hydraulic and mechanical.

TABLE 2.2.2

PRESSURIZED WATER REACTORS

(project values)

		1st generation Yankee	2nd generation Obrigheim	Present generation Tihange/Biblis*	Future generation
Year ordered		1956	1965	1969/70	1975
Net electric power	MWe	110	283	870/1,146	1,600
Net efficiency of plant	%	28.1	31.1	32.7/33.1	33.5**
Specific power	kW/kg U	18.8	23.3	38.1/34.9	46
Mean power density	kW/l	58.4	68	100/86.7	110
Mean burnup at equilibrium	MWd/t U	8,200	24,000	33,000/31,500	35,000
Power per m^3 of coolant	MW/m^3	5.35	5.85	10.3/8.30	-
Secondary steam Pressure	atm.abs	32	50	55/52	60
Reactor temperature differential	ºC	15	25.4	39/31.8	-
Specific coolant flow rate	t/h/MW	43.6	24.2	17.25/16.3	-

*This plant is designed to follow load variations on the grid.

**If cooling towers are used, the efficiency would be 32.0%

The control of pressurized water reactor core reactivity has made greater progress. First used in the San Onofre and Obrigheim plants, a new concept of cluster-type control rods has been developed.

This new concept has the following advantages over the old (cf. Trino Vercellese and Chooz plants) with cruciform rods:

- The power density peaks caused by water gaps are virtually eliminated.

- The efficiency of the control rods is improved per unit weight and volume of absorbing materials.

- The control rod fuel followers are no longer needed, permitting a reduction in overall height of the pressure vessel and the elimination of the cruciform openings in the lower core support plate. A reduction of about 2.1 m in the height of the pressure vessel was thus possible for a 500 MWe plant.

- The cost of the pressure vessel internals is reduced.

Moreover, the chemical control developed in the BR-3, Saxton and Yankee plants and adopted first in the Trino Vercellese, Chooz and Obrigheim plants has permitted:

- a sharp reduction in the number of control rods necessary;
- better use of fuel through lengthening of the neutron lifetime of the core;
- an increase in specific power through better distribution of power in the core.

2.3.2 Primary circuits

Main development features:

1. The adoption of shaft seal pumps, with the following advantages over the submerged rotor pumps in the Chooz (PWR) and Garigliano (BWR) plants:

   - higher efficiency and therefore lower operating cost;
   - greater reliability in the event of an electrical breakdown, as a large flywheel can be used if necessary;
   - lower maintenance costs and shorter outage times.

2. The elimination of isolating valves for the circulation loops.

3. An increase in power output per loop for PWR's (project values):

   210 MWt/loop for the Chooz plant,

   454 MWt/loop for the Obrigheim plant and

   nearly 900 MWt/loop for contemporary plants (Tihange/Biblis-A).

4. The adoption for BWR's of:

   - the simple direct steam cycle,
   - jet pumps or axial pumps and steam separators inside the pressure vessel. This concept eliminates the secondary steam generators and reduces the number of recirculation loops. Thus a BWR with a power of the order of 1,000 MWe now has only two

external loops (Dresden-2 and Browns Ferry, for example). The ultimate increase in void coefficients to be tolerated in BWR cores will probably permit external recirculation loops to be abolished completely. This stage has already been reached by AEG which has, without high void coefficients, developed an axial shaft seal pump, incorporated in the lower part of the pressure vessel. This design has recently been adopted for the Brunsbüttel and Philippsburg plants in Germany. The BWR pressure vessels are larger than those of the PWR because of the lower power density of the BWR's and the incorporation of recirculation pumps and steam separators.

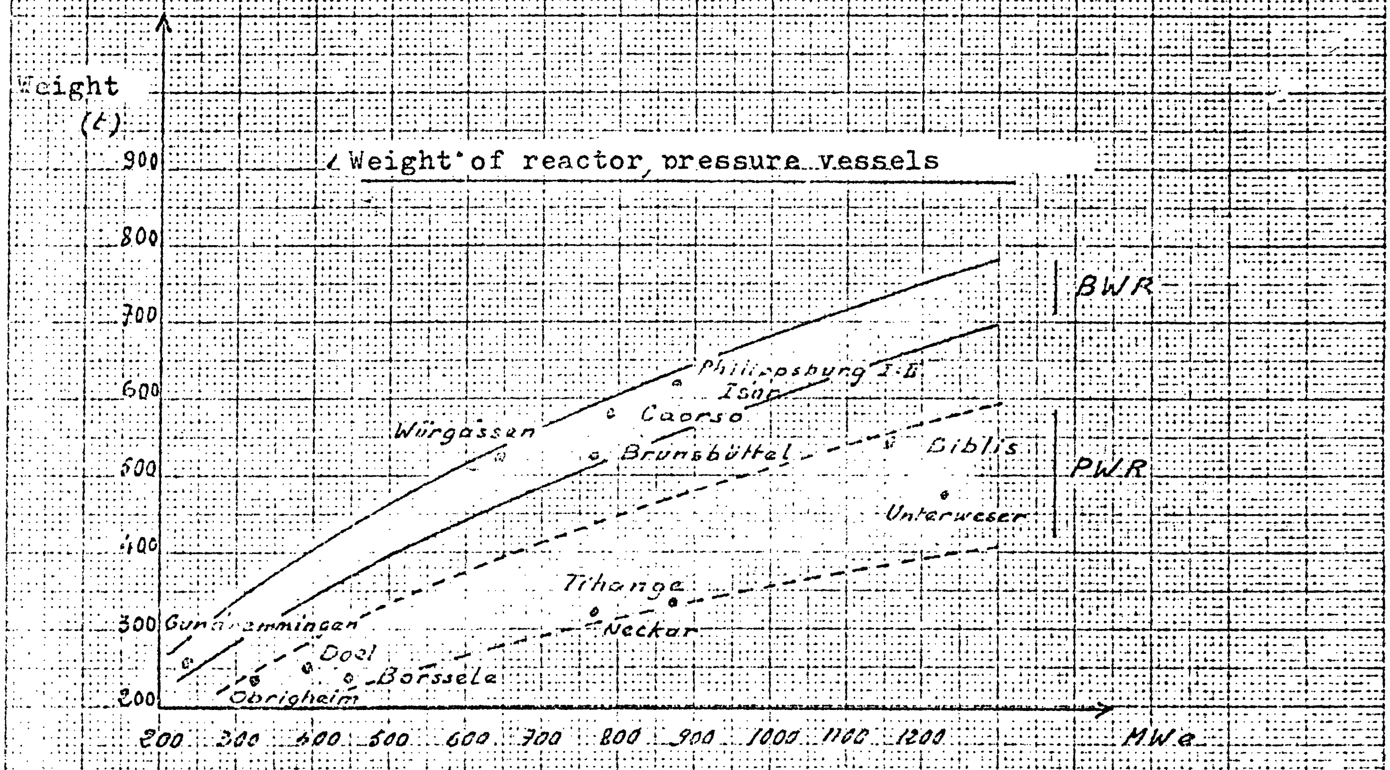
The graphs and drawings in Figs. 2.3.2-1 and 2.3.2-2 provide a comparison of the principal and relative dimensions of BWR and PWR pressure vessels. These graphs show the changes in weight and internal diameter of the pressure vessels as a function of reactor power.

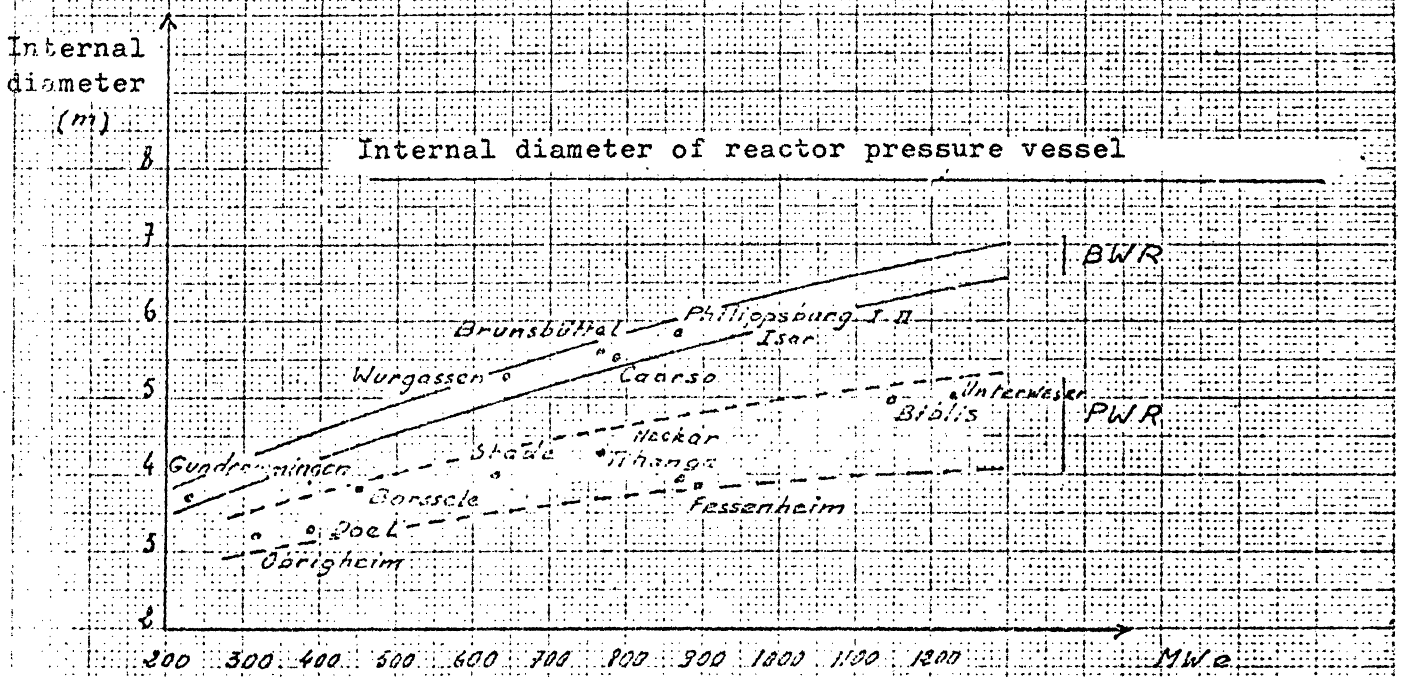
2.3.3 Containment

The relative compactness of BWR's led the promoters of this type of reactor to develop a containment known as the pressure-suppression primary containment. It was first constructed of steel for the Humboldt Bay, Oyster Creek, Dodewaard, Dresden-2 and Würgassen plants and has now also been constructed in concrete at Caorso, Italy. The advantages of the pressure-suppression system are as follows:

- Very compact design;
- zero leakage rate achieveable;
- the reactor containment and buildings can be constructed simultaneously on site.

Fig. 2.3.2-1


Weight (t)
Weight of reactor pressure vessels
900
800
700
600
500
400
300
200
BWR
PWR
Philippsburg I-II
Isar
Caorso
Würgassen
Brunsbüttel
Biblis
Unterweser
Tihange
Neckar
Gundremmingen
Doel
Borssele
Obrigheim
200 300 400 500 600 700 800 900 1000 1100 1200
MWe


Internal diameter (m)
Internal diameter of reactor pressure vessel
8
7
6
5
4
3
2
BWR
PWR
Brunsbüttel
Philippsburg I-II
Isar
Würgassen
Caorso
Unterweser
Biblis
Neckar
Stade
Tihange
Gundremmingen
Borssele
Fessenheim
Doel
Obrigheim
200 300 400 500 600 700 800 900 1000 1100 1200
MWe

PRESSURE VESSELS

Fig. 2.3.2-2

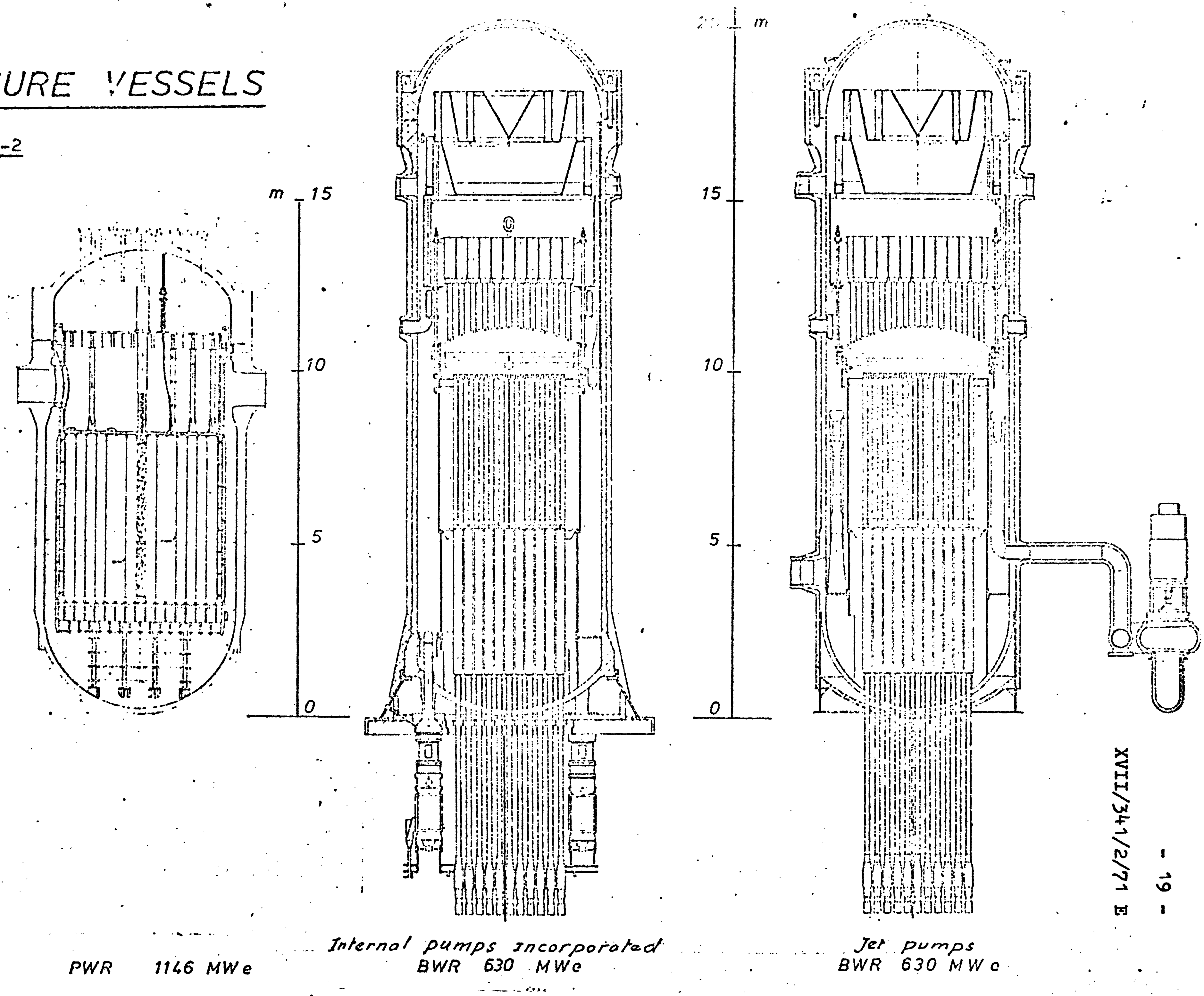
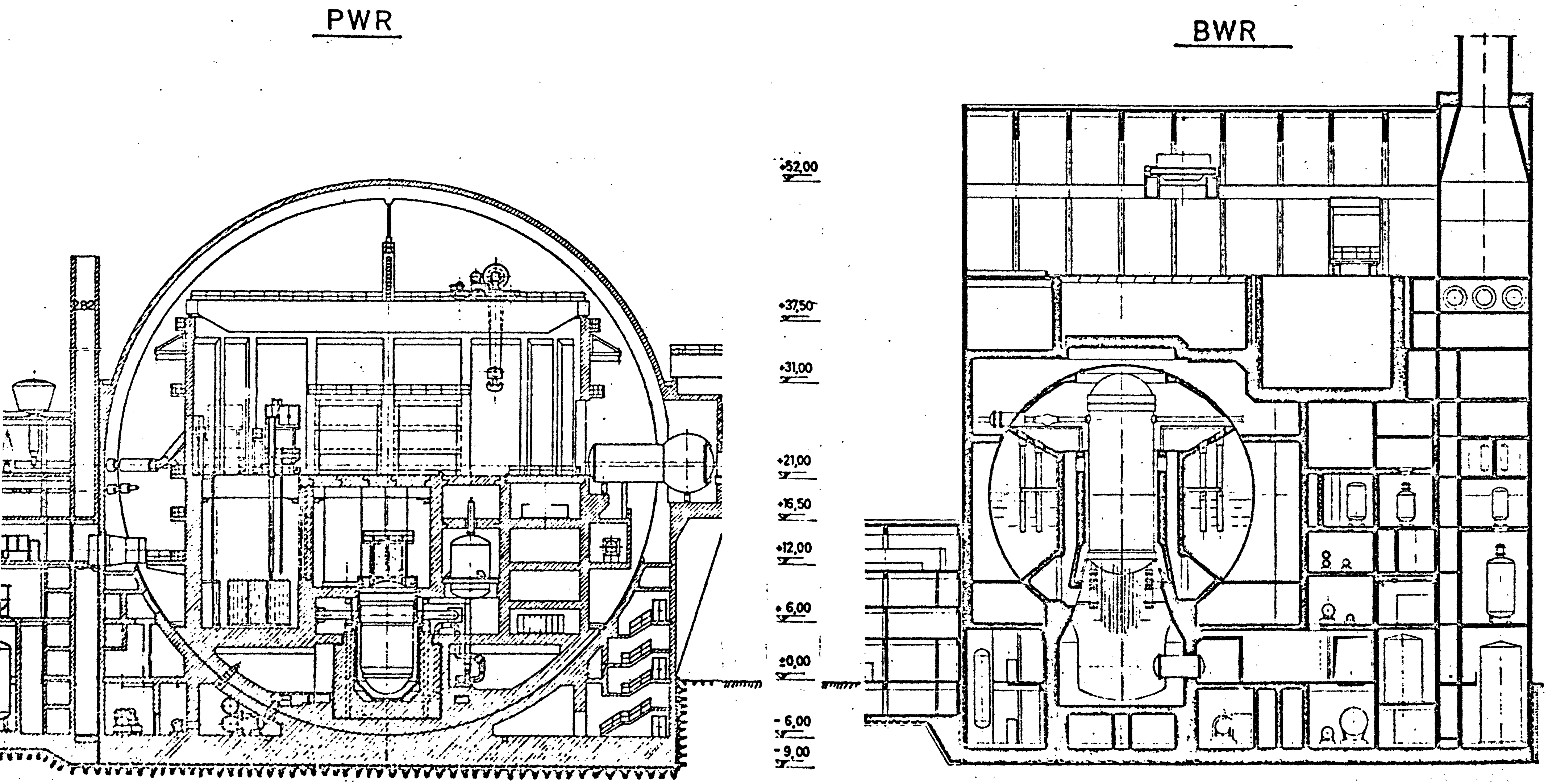


Fig. 2.3.3.

LEAKTIGHT CONTAINMENTS AND BUILDINGS

NUCLEAR POWER PLANTS OF 1200 MWe



The volume of PWR containments is much greater than that of BWR's; it has to house relatively large equipment: steam generators, pressurizer, etc. Unlike the system adopted for BWR's, the PWR containment is purely static and provides resistance to the maximum pressure following an incident. It consists of two parallel successive shells, one of steel and the other of concrete, with a ventilated and depressurized space in between. It is to be noted that at Fessenheim a single shell containment has been adopted.

However, a substantial reduction of post-accident pressure can be obtained by means of an ice condenser system developed by Westinghouse and commercially used for the first time at the Cook and Sequoyah power plants.

In Siemens power plant designs the irradiated fuel storage pool is inside the containment.

Fig. 2.3.3 gives a schematic comparison of the main dimensions and internal layouts of BWR and PWR reactor containments.

2.3.4 Power regulation

In the first BWR plants (Garigliano, Dresden-1, Gundremmingen) with dual cycle, the regulation of load between 70 and 100% of nominal power was obtained by varying the secondary steam flow, without movement of the control rods. Since Lingen and Oyster Creek, BWR plants have adopted regulation by coolant flow adjustment.

This new principle of reactor regulation developed in the Community by AEG at Lingen permits very flexible load variation (1%/sec) and, within a wide range (10-100% of rated power), instantaneous power variations exceeding 10%. The rapid power variations required by the grid can thus always be met.

In PWR's, three types of regulation characteristics have been adopted during the last decade. The first is that used at the Chooz power plant which operates at constant average coolant temperature, with a high steam pressure at low loads. Since the Obrigheim power plant, the regulating principle has been operation at constant steam pressure for low load and constant average coolant temperature for high load. This programme combines the advantage of favourable self-regulation, i.e., very low amplitude of control-rod movement at the higher steady loads which are best for operation, with that of a low steam pressure at low load.

A new regulation characteristic has been adopted for the Tihange power plant. Operation is based on a programme of variable average temperature which is linear with load. This characteristic permits a 21% variation in secondary steam pressure and a 17°C variation in the average in-core coolant temperature for a power variation of 0-100%.

The PWR system of power regulation in all cases enables at least the following load variations to be obtained (expressed as a percentage of nominal power):

- continuous variations of ±5%/min;
- instantaneous variations of ±10%.

For the Biblis-A power plant, higher values are planned (15%/min continuously and 15% instantaneously); in comparison with a power plant operated only according to a basic mode, the power density is slightly lower.

2.4 The fuel and its cycle

Only the fuel and fuel-cycle aspects connected with the operation of power plants are examined here.

BWR's and PWR's use fuel assemblies consisting of individual Zircaloy tubes containing sintered uranium oxide pellets. The technical characteristics of the fuel and its cycle are given in Table 2.4.1.

2.4.1 Fuel cans

Zircaloy is the only canning material used at present, because of its advantageous property of low neutron absorption: It should be noted that first-generation power plants temporarily used stainless steel, the behaviour of which in BWR's was not considered satisfactory. The adoption of Zircaloy also enabled the initial enrichment of the fuel elements to be considerably reduced.

Experience gained permitted a slight reduction in the thickness of the cans and the use of rods all of the same length. However, the increase in burnup to more than 25,000 MWd/t necessitated an increase in the volumes reserved for fission gases.

A technique applied recently consists in pressurizing the rods with helium before welding on the last plug, to improve the mechanical strength of the rods at the beginning of irradiation and obtain higher burnups.

In recent years fission product leaks have been recorded in under 1% of light-water reactor fuel pins. Though the mechanism of leakage is not yet fully explained, it appears certain now that most of the faults are due to excessive humidity of the uranium oxide pellets. These difficulties will probably be overcome by a suitable modification of fabrication processes and tighter quality control.

Termary zirconium-base alloys are currently under test. They show better behaviour at elevated temperature and may prove to be an alternative to Zircaloy as a canning material for light-water reactors.

2.4.2 Fuel

Sintered uranium oxide in pellet form is the only material with a well-developed technology and well-know stability under irradiation.

The density of the oxide used is optimized for each individual case, but is restricted to about 93% of the theoretical density to avoid too much swelling of the pellets when subjected to irradiation.

Sintered uranium oxide has good stability under irradiation up to about 50,000 MWd/t, a sufficient value to minimize the cost of the full cycle.

The present lack of large-scale experience of the irradiation behaviour of fuel elements fabricated by vipacking prevents any short term prospect of replacing the production methods by pelleting.

The difference in diameter of the uranium oxide pellets (12.4 mm for BWR's and 9.30 mm for PWR's) is caused partly by differing power densities due to the thermal characteristics of the coolant in one (PWR) and two (BWR) phases. Despite a lower production cost for the larger pellets, the fuel cycle costs are comparable for both types of reactor.

2.4.3 The assembly

PWR fuel assemblies are at present made without an outer can; the fuel rods are kept in position by regularly spaced Inconel grids welded to the 16-21 stainless steel or Zircaloy-4 guide tubes of the control rods. This arrangement allows significant saving in core structural materials, virtually rules out water gaps between the assemblies and hence eliminates radial power density peaks in the corner rods. However, the Babcock and Wilcox Co., has retained the perforated can in its designs.

In the case of BWR's, the control of coolant distribution when bubbles occur and the cruciform control rod guide system prevent any elimination of Zircaloy outer cans in the present design.

The fuel assemblies currently comprise 7 x 7 rods in the BWR's and 14 x 14-16, 15 x 15-20 or 16 x 16-20 rods in the PWR's, depending on the constructor and the reactor capacity.

2.4.4 The cycle

After a similar development, the two reactor types have arrived at the same fuelling procedure combining the chequerboard and outside-to-inside core fuel shuffling plans.

BWR's are started with only one type of fuel assembly with the same average enrichment. At the first refuelling, the boron steel poison curtains between the fuel assemblies are withdrawn from the core with 30% of the most irradiated assemblies. These assemblies are stored for reinsertion at the second refuelling when 50% of the oore is replaced. The subsequent loadings and unloadings take place each year by renewal of 1/4 of the fuel assemblies in the central zone of the core. However, the plan used so far may be greatly modified with the use of fuel elements containing burnable poisons.

PWR's use for their first charge three batches of fuel assemblies with three different enrichments, but with approximately the same number of assemlies. The first two batches, with a below-average enrichment, are uniformly mixed in the central core zone (2/3). At each annual renewal, 1/3 of the most irradiated assemblies of the central zone are withdrawn and replaced by the assemblies from the outer zone. It should be noted that Siemens uses a core with four fuel assembly zones for the Stade and Biblis power plants.

TABLE 2.4.1

TECHNICAL CHARACTERISTICS OF THE FUEL AND ITS CYCLE

American values for 1,000 MWe (the values for the Biblis plant with a new electric power of 1,146 MWe differ slightly and are given in brackets)

		BWR	PWR	
Core				
Total thermal power	MW	3293	3083	(3462)
Average power density	kW/l	50.8	93.1	(85.3)
Average specific power	kW/kgU	22.0	34.8	(34.9)
Total uranium load	T	149.8	88.6	(99.2)
Fuel rods (cold)				
Pellet diameter	mm	12.4	9.3	(9.08)
Pellet length	mm	17.8	15.2	
Pellet density	g/cm^3	10.22	10.19 - 10.3	(10.0 - 10.35)
Can thickness	mm	0.81	0.61	(0.72)
Can outer diameter	mm	14.3	10.72	(10.75)
Active length	cm	366	366	(390)
Fuel material		Sintered UO_2	Sintered UO_2	
Can material		Zircaloy 2	Zircaloy 2	
Fuel assembly				
Rod lattice		7 x 7	15 x 15	(16 x 16)
No. of rods		49	204	
Rod pitch	mm	18.7	14.3	
Channel material		Zircaloy 4	None	
No. of spacer grids		7	9	

Table 2.4.1 cont.

	BWR	PWR
Cycle		
1st core:		
Initial U^{235} enrichment g/kg initial U	21.9	(21.7 (21.8) (22.7 (23.8) (26.7 (25.3) (31.9)
Average burnup MWd/t U	20,900	21,800 (23,000)
Composition of unloaded fuel g/kg initial U		
U^{235}	6.87	7.58
U^{236}	2.61	2.79
Pu^{239}	4.35	4.75
Pu^{240}	1.70	1.91
Pu^{241}	0.71	0.84
Pu^{242}	0.22	0.25
Core at equilibrium		
Initial U^{235} enrichment g/kg initial U	25.6	33.0 (30.0)
Average burnup MWd/t	27,500	33,000 (31,500)
Composition of unloaded fuel g/kg initial U		
U^{235}	6.19	8.43
U^{236}	3.31	4.19
Pu^{239}	4.61	5.3
Pu^{240}	2.07	2.4
Pu^{241}	0.93	1.17
Pu^{242}	0.36	0.44
Loading/unloading		
Fraction withdrawn from core at first cycle and stored for reinsertion	0.30	-
Average fraction of core renewed at each cycle	1/4	1/3

2.5 Experience gained and main technological difficulties encountered

2.5.1 Reliability and results of operation of power plants

The energy availability and utilization reactors are representative of the reliability of nuclear power plants.

In the lists of operating characteristics of Community nuclear power plants published periodically by the Statistical Office of the European Communities the following factors in particular, are found:

$$\text{energy availability} = \frac{\text{electrical energy availability}}{\text{No. of hours in the period under review x max. poss. power}}$$

$$\text{energy utilization} = \frac{\text{electrical energy utilized}}{\text{No. of hours in the period under review x max. poss power}}$$

The reliability of a power plant can thus be validly represented by the energy availability factor. The energy utilization factor will not, however, be representative of the reliability of a power plant which required to keep pace with load variations on the grid.

In the United States, the same factors are normally used, principally the energy utilization factor (capacity factor), to define the reliability of power plants. The availability factor is, however, rarely published.

To compare the reliability of Community and US light-water plants, the availability factors have therefore been used whenever these were available.

The energy utilization and availability for Community and US power plants are given for comparison in the tables in Figs. 2.5.1-2.5.4, by years of operation.

Figs. 2.5.5 and 2.5.6 show the energy utilization and availability factors of Community and US power plants by years of operation in graph form. The number of plants in operation each year is also shown.

In order to compare the operation of Community and US power plants, the average values of utilization and availability factors by year of operation have been calculated. For the Community, the average values have also been calculated without taking into account the Trino Vercellese and Chooz plants. These PWR's have had to be almost entirely shut down for repair for three years, because of damage to reactor internals (Trino Vercellese was shut down from March 1967 to June 1970 and Chooz from January 1968 to May 1970). On the other hand, the four other Community plants at Garigliano, Gundremmingen, Lingen and Obrigheim have had good utilization and availability factors, despite some difficulties at start-up.

At the Garigliano power plant, the reactor had to be shut down for some months (from September 1965 to April 1966) for the replacement of fuel channels and cleaning of fuel elements and for various repairs to the reactor. The turbine caused a month's shutdown in 1964. In 1967 and 1968, five months' shutdown were necessary for refuelling and for certain work on the reactor and the conventional installations.

At the Gundremmingen nuclear power plant, a shutdown of about $5\frac{1}{2}$ months in 1968 was caused by four incidents on the low-pressure turbine blades. Between November 1968 and May 1969, the plant output had to be reduced to 60% of nominal because of the outage of the low-pressure part of the turbine; between June and August 1969, this plant had to be shut down to carry out turbine repairs and core reloading. How these shutdowns at the Gundremmingen plant during its third and fourth years of operation affected the availability and utilization coefficients (excluding the Trino Vercellese and Chooz plants) can clearly be seen in Figs. 2.5.5 and 2.5.6.

At the Lingen plant, connected to the grid for the first time in May 1968, two incidents were observed in 1970 on the blading of the low-pressure part of the turbine. Owing to missing blocks on the LP turbine stage, the plant output had to be reduced to 96% of nominal for eight months (September 1970-April 1971). A further power reduction to 95% was caused by the outage of two mobile ionization chambers in the core. Shutdowns due to turbine incidents, refuelling, inspections and repairs thus reduced the availability and utilization factors of the Lingen plant in 1970.

It is to be noted that the energy utilization factor of the Gundremmingen and Lingen plants was also reduced by the need for these plants to keep pace in part with load variations on the grid.

The lessons learned from experience with Trino Vercellese and Chooz enabled a high availability factor to be obtained for the Obrigheim plant as soon as it was connected to the grid (October 1968). By means of some design modifications to the Obrigheim reactor internals, the incidents which occurred on the Trino Vercellese and Chooz reactors were avoided. Thus the Obrigheim plant* achieved an annual utilization factor of 91% for the period 1 July 1969 to 30 June 1970.

The number of power plants in operation is, however, still too small to obtain an accurate statistical evaluation of the average utilization and availability factors. Nevertheless, a certain tendency to improvement of power plant reliability with years of operation may be noted. It can thus be estimated that, for nuclear power plants, a steady reliability value is reached after a relatively longer period (about five years) than for conventional plants.

It may also be noted that the average energy utilization and availability factors for Community light-water plants are of the same order of magnitude as those of US plants of the same type.

*ATW, March 1971, p.152

Fig. 2.5.1. : Energy utilization factors - Community power plants

Power plants	Type	MWe	Year connected to grid	Years of operation 1.	2.	3.	4.	5.	6.	7.	8.
Garigliano	BWR	150	1964	56	69	58	66	73	84	70 2)	83
Trino Vercellese	PWR	257	1964	37	55	67	27	0 1)	0 1)	52	60
Gundremmingen	BWR	237	1966	31	48	47	58	84	91		
Chooz	PWR	266	1967	30	3	0 2)	53	78			
Lingen	BWR	174 (254)	1968	43	86	64	64				
Obrigheim	PWR	328	1968	12	74	84	75				
Dodewaard	BWR	52	1968	68	79	84					

Réf. : Statistical Office of the European Communities

1) Shutdown for repair
2) Including a 3 months shutdown due to a strike

		1.	2.	3.	4.	5.	6.	7.	8.
No.of plants		7	7	7	6	4	3	2	2
Average value		40	59	58	57	58	58	61	71
No.of plants excl. Trino Vercellese and Chooz		5	5	5	4	2	2	1	1
Average value excl Trino Vercellese and Chooz		42	71	67	66	79	88	70	83

Fig. 2.5.2 Energy utilization factors - US powerplants

Power plants	Type	MWe	Year connected to grid	Years of operation											
				1	2	3	4	5	6	7	8	9	10	11	12
Shippingport	PWR	90	1958	37	34	45	59	62	66	-	62	67	61	47	39
Dresden 1	BWR	200	1960	23	33	74	54	56	55	80	46	52	48		
Yankee Rowe	PWR	175	1960	25	76	55	69	80	65	86	86	81	75		
Indian Point 1	PWR	265	1962	28	38	25	46	50	68	65	72				
Big Rock Point	BWR	72	1963	35	42	29	55	81	68	64					
Humboldt Bay	BWR	70	1963	78	83	60	37	75	82	68					
San Onofre	PWR	430	1967	21	34	69									
Haddam Neck	PWR	462	1967	30	73	75									
Oyster Creek 1	BWR	515	1969	93											
Ginna	PWR	420	1969	20											
No. of plants				10	8	8	6	6	6	5	4	3	3	1	1
Average value				39	52	54	53	67	67	73	67	67	61	47	39

Source: US Nuclear Power Reactors - Operating History - 1964 (Table 3)
- 1969 (Table 8)

Fig. 2.5.3.

Energy availability factors - Community power plants

Power plants	Type	MWe	Years of operation 1.	2.	3.	4.	5.	6.	7.	8.
Garigliano	BWR	150	53	68	58	66	88	95	74 3)	86
Trino Vercellese	PWR	257	- 1)	55	67	29	0 2)	0 2)	52	6
Gundremmingen	BWR	237	45	67	58	58	85	91		
Chooz	PWR	266	39	4	0 2)	53	78			
Lingen	BWR	174 (254)	- 1)	94	67	69				
Obrigheim	PWR	328	- 1)	75	84	75				
Dodewaard	BWR	52	67	78	85					

1) Data not available
2) Shutdown for repair
3) Including a 3 months shutdown due to a strike

			1.	2.	3.	4.	5.	6.	7.	8.
No. of plants			4	7	7	6	4	3	2	2
Average value			51	63	60	58	62	61	63	73
No.of plants excl. Trino Vercellese and Chooz			3	5	5	4	2	2	1	1
Average value excl. Trino Vercellese and Chooz			55	76	70	67	87	93	74	86

Fig. 2.5.4

Energy availability factors - US power plants

Power plants	Type	MWe	Years of operation											
			1	2	3	4	5	6	7	8	9	10	11	12
Shippingport	PWR	90	53	50	50	64	76	81	-	62	-	-	-	58
Dresden 1	BWR	200	33	38	79	76	81	79	-	-	-	61		
Yankee Rowe	PWR	175	-	89	64	78	88	73	-	-	-	83		
Indian Point 1	PWR	265	-	-	-	62	-	-	-	81				
Big Rock Point	BWR	72	39	43	30	-	-	-	90					
Humbold Bay	BWR	70	81	87	77	-	-	-	89					
San Onofre	PWR	430	-	-	76									
Haddem Neck	PWR	462	-	-	87									
Oyster Creek	BWR	515	95											
Ginna	PWR	420	82											
No. of plants			6	5	7	4	3	3	2	2	-	2	-	1
Average value			64	61	66	70	62	77	90	72	-	72	-	58

Source: US Nuclear Power Reactors - Operating History - 1964 (Table 3)
- 1969 (Table 8)

- Data not available

Fig. 2.5.5.

Energy utilization factor of light-water plants in the Community and the United States

No. of plants in operation

Community	6	6	6	4	3	2	2							•
	4	4	4	2	2	1	1	(Excl. Trino Vercellese et Chooz)						o
US	10	8	8	6	6	6	5	4	3	3	1	1		+

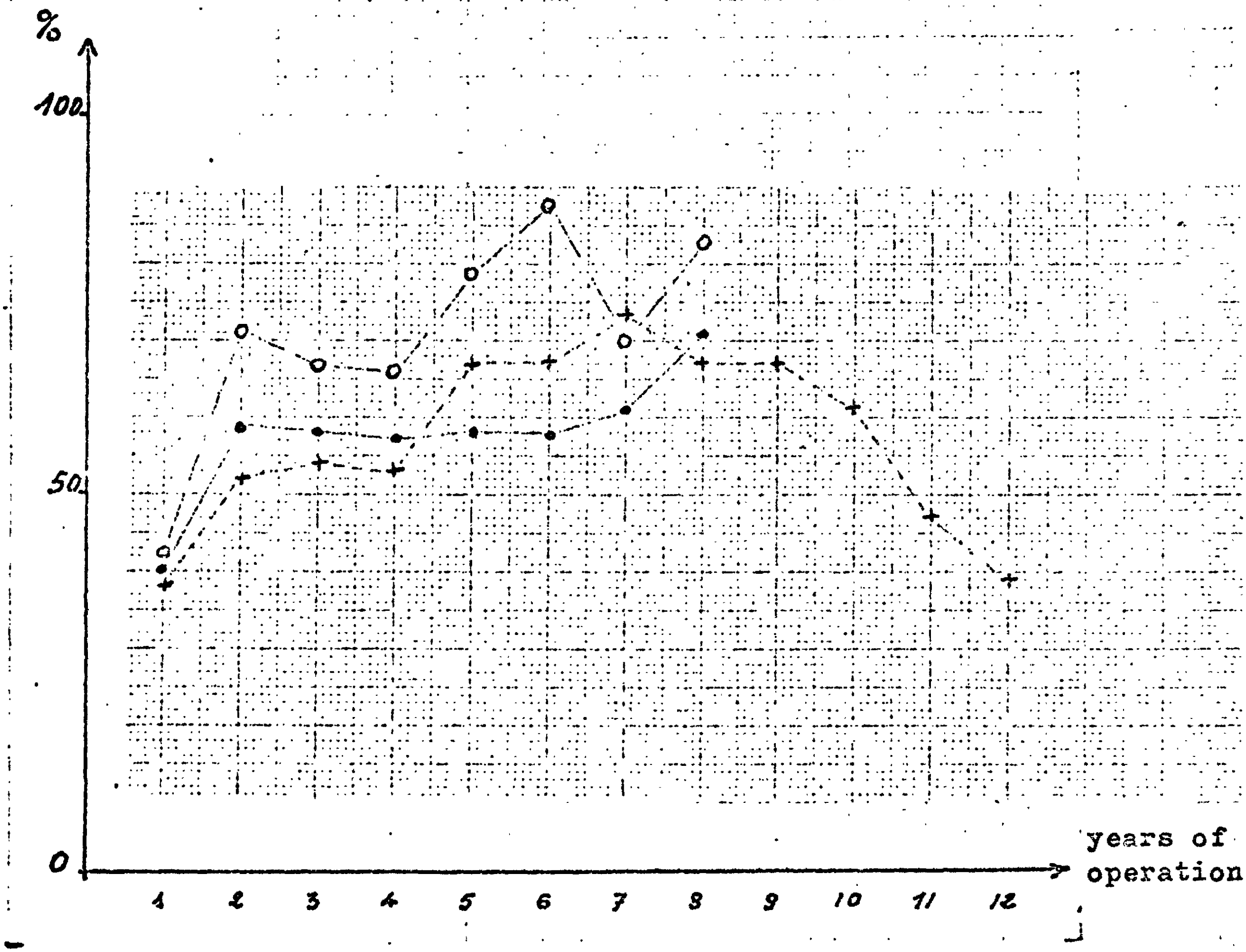
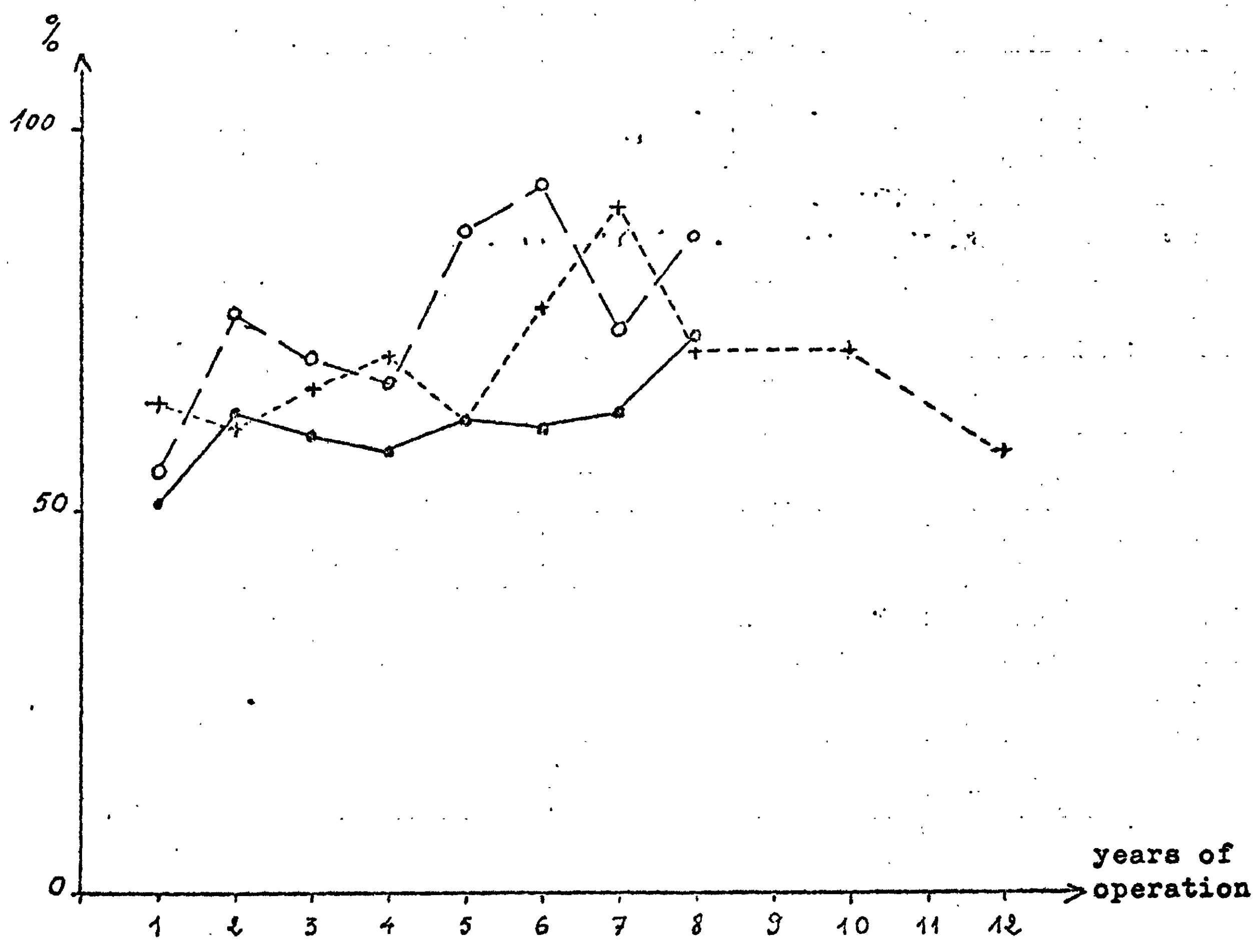


Fig. 2.5.6.

Energy availability factor of light-water plants in the Community and the United States

No. of plants in operation

Community	4	7	7	6	4	3	2	2						•
	3	5	5	4	2	2	1	1	Excl.	Trino Vercellese & Chooz				o
US	6	5	7	4	3	3	2	2	–	2	–	1		+



2.5.2 Main technological difficulties

The operation in the Community of five BWR and four PWR power plants has shown up the following principal technolgocial difficulties, all of which have been overcome:

A. Containments and associated equipment

The construction of large containments has not usually raised major problems. In two cases, however, the use of fine-grain steels sensitive to temperature conditions made it difficult, especially in winter, to meet the conditions necessary for the good performance of work (welding, etc.). In such cases, thoroughly preparatory tests are required, as well as meticulous compliance with welding procedures and conditions.

The specifications concerning maximum permissible leakage were met without difficulty in all cases. With regard to leaktightness, and especially periodic tests, an improvement in methods of measurement might yield a saving in time and hence better availability of power plants.

B. Reactor pressure vessels and internals

Here the particularly noteworthy incidents are those that occurred in two PWR's of identical design. In both cases the thermal shield (a forged ring 80-90 mm thick) began to rock on its supports as a result of the forces developed by the flow of recirculation water. In one of the plants these movements led to the rupture of the ball-and-socket joints linking the three segments of the shield, so that it comes completely apart. In the other plant the shield remained intact because the joints were welded. Nevertheless, as in the first plant, more than half the assembly bolts of the two core barrel rings were found to be broken. This is primarily because of the differential and oscillating pressures set up on the barrel by the flow distribution.

In both reactors the cruciform control rods were fitted with active followers. A cast plate was fixed to the core ring by 32 tie-rods, to take up part of the core weight and ensure the rigidity of the bottom grid by compressing the follower guide tubes. Some of these tie-rods were, however, found to be broken. These ruptures appear to result from vibrations caused by eddying behind the tie-rods in the perpendicular flow channel.

Apart from these ruptures, the debris in motion in the primary circuit also caused secondary damage, notably in the heat exchangers.

The modifications made to remedy the causes of these incidents were:

- the elimination of the thermal shield;
- the replacement of the core barrel assembly bolts by stronger bolts with mechanical locking;
- elemination of the tie-rods.

The Chooz power plant, which had been shut down since January 1968, was reconnected to the grid in May 1970 after two years of repairs.

The Trino Vercellese plant was restarted in June 1970, after three years of repair.

At Obrigheim, the same problem of tilting of the shield occurred during the start-up tests; consequently the necessary modification, in this case the fixing of the shield to its base, was made under better conditions. Even after one year's operation, no damage was observed.

Finally, a growing problem is that of the periodic pressure-vessel inspections and the monitoring of the reactor internals. A considerable effort is being made in this field; the development of efficient inspection methods should be continued.

C. Fuel elements and associated equipment

Non-scheduled shutdowns of a light-water reactor caused by defective fuel elements have not yet occurred. This does not, however, mean that there are no defects. These take the form of leaks and, in rare cases, can rupture. It seems that most defects are due to excess moisture in the uranium oxide pellets; other causes are friction by foreign bodies in the water (pieces of steel wire, etc.) and the ingress of water into the can, which results in secondary defects. The influence of a frequent change in mode of operation of the plant has also been recorded.

D. Instrumentation and control

The weaknesses encountered in this connection were usually of minor significance. They mainly concern the behaviour of the cabling and the induction of disturbances in insufficiently shielded cables and devices. Moreover, in view of the rapid technical and industrial development, the availability of spares on the market might become a problem in the medium term; the general adoption of standardized signals would be a help here. Numerous failures were also noted in the in-core instrumentation.

Lastly, experience has shown that for a satisfactory analysis of operating data, particularly in the event of a nuclear power plant breakdown, the conventional recording is no longer adequate; a computer is necessary.

E. Control rods and drives

The occasional, and in any case minor, troubles were mainly due to small design faults which did not come to light on the test rig because actual conditions were not adequately simulated, or, in the case of BWR's, were caused by impurities in the hydraulic control water during the start-up test period.

F. Heat exchangers, pipes, pumps, valves

In this connection a large number of difficulties were recorded, usually of secondary significance. The causes are unconnected with the "nuclear" use of the equipment in question. They fall mainly into the following groups:

- design faults (badly-chosen materials, thermal expansion not sufficiently taken into account, inadequate support of components, etc.);

- insufficient quality control.

G. Turbo sets

The number of occurrences of damage to the turbo sets in nuclear power plants is impressive 19 since 1963 in eight Community installations. Virtually all the undamaged installations also had to be modified in order to make them function satisfactorily. There are two root causes:

(a) It was necessary to review saturated-steam turbine technology and extrapolate it to large flow volumes;

(b) The development of increasingly large units was more repid in the nuclear than in the conventional field.

The damage observed in nuclear power plants is mainly due to fatigue rupture of blades. This is a result of operation close to resonance frequencies, which occurs when:

(1) the blade restraint conditions change during operation;

(2) the steam and its moisture are unevenly distributed.

Other problems, interdependent, were:

- speed sensing on loss of load,
- drainage of stages (often inadequate),
- drying of steam.

All these problems were ultimately solved by the constructors. However, the development of supervision systems permitting an immediate diagnosis of all deviations from normal operation is desired by the operators.

H. Processing of water, gas and waste

Apart from certain relatively minor troubles concerning, for example, the cleaning of systems before start-up, the decontamination of certain parts of the plant and the purification of the working pool water, the great problem encountered in this connection, by a Community power plant, was that of the deposition of corrosive products at preferred points, such as the inlet orfices of fuel elements.

Although the causes of the phenomenon are not fully understood, it now seems to be established that the sources of trouble were:

- copper from the Cu/Ni alloy feed-water heaters,
- an overhigh oxygen content in the water, resulting from the method of operating the water purification plant.

*

* *

To sum up, experience has shown that:

(a) most of the components causing trouble are conventional or paranuclear items;

(b) so far the control system, especially reactivity control, have not given cause for concern;

(c) the normal operation of a nuclear power plant can be ensured, even when there are a large number of defective fuel elements.

Accordingly, the following steps are necessary:

- More thorough quality control in manufacture.
- Greater care in the design of conventional and paranuclear items.
- Caution in modifications, even if minor at first sight, of already proven equipment when they have not been tested under all possible conditions of operation or malfunction of the system of which the equipment forms a part; in this connection, an appreciable increase in size is equivalent to a modification.
- Appreciation of the fact that during the lifetime of a power plant it may be necessary to dismantle and replace large units, such as the thermal shield or the heat exchanger tube bundles.

3. ECONOMIC DATA AND OUTLOOK

3.1 Capital cost*

The review of the recent trend in capital spending and electric power production in light-water reactor plants can be divided into distinct periods:

(1) An optimistic period, 1965-67
(2) A "waiting" period, up to 1970
(3) A period covering the seventies.

3.1.1 Optimistic period

After the ordering of the first nuclear power plants and their gradual commissioning from 1963 onwards, nuclear power plant constructors adopted an aggressive commercial policy to incite electricity producers to turn increasingly to the new nuclear installations for their production equipment.

Thus, while the installation cost of the first nuclear power plants, of a representative capacity of at least 200 MWe, was of the order of 400 u.a. per kW installed (Indian Point-1, 265 MWe - 404 u.a./kWe; Latina, 200 MWe - 474 u.a./kWe; Chooz, 266 MWe - 377 u.a./kWe), it underwent a very sharp reduction as from 1965.

This reduction in the specific capital cost is explained by:

(a) the determination of constructors to secure a good share of this new market;

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*The capital cost includes the items given in the table on p.56.

(b) the effect of size increase; contracts awarded as early as 1965 are mainly for plants of over 600 MWe;

(c) the hoped-for saving resulting from:
- experience gained in fabrication and construction;
- the effect of repetition and standardization of certain components;
- the technical improvements made after the first prototype power plants had been built.

As a result of the combined effect of these various factors, the specific capital cost of power plants ordered during this period in the United States was some 110-120 u.a./kWe.

(Indian Point-2 - 873 MWe - 121 u.a./kWe,
Turkey Point-3 - 722 MWe - 91 u.a./kWe,
Browns Ferry - 1,065 MWe - 113 u.a./kWe)

In the Community, two leading European constructors offerred nuclear power plants with a specific capital cost of the same order of magnitude (Wargassen and Stade plants).

3.1.2 "Waiting period"

United States

From the end of 1967, there was an appreciable increase in specific capital costs and orders for nuclear power plants fell off. This fact was due to:

(a) the need for constructors to make good the losses incurred to give them a foothold in the new market.

Because they wished to establish themselves in the nuclear field, constructors had been led into tendering at prices which did not make sufficient allowance for all the cost factors (underestimated actual costs, provision for unforeseeable expenditure, etc.).

(b) failure to meet delivery dates, as a result of:

(1) the saturation of the industrial capacity of constructors and their sub-contractors. Thus, following the avalanche of orders for nuclear power plants and the resulting bottlenecks the construction periods for nuclear plant turbines went up to nearly 60 months; a similar situation also occurred in pressure-vessel construction;

(2) the acute environmental problems raised by the installation of nuclear power plants and, consequently, the length of the procedures required to obtain licences from the competent authorities;

(3) the technical difficulties encountered when constructing and commissioning nuclear power plants previously ordered;

(4) the constructors' abandonment of turnkey contracts;

(5) social agitation and strikes affecting constructors and their sub-contractors.

Failure to meet delivery dates and delay in the commissioning of the plant as a result add considerably to the production cost of energy.

Besides the difficulties of supply confronting the electricity producer, it has been estimated that a delay of 12 months in the commissioning of a 500 MWe plant entails an additional capital cost of more than $36 \cdot 10^6$ u.a. (statement by J.H. Campbell, Chairman of Consumer Power Co., to the Joint Committee for the Nuclear Industry, April 1970).

(c) the appreciable increase in certain cost factors: high cost of money, increased wage-related charges, general rise in raw material costs, modification of initial plans to take into account increased safety requirements.

Moreover, safety requirements and the steps taken to prevent pollution are not without effect on the overall capital cost. Thus, to give only one example, the construction of cooling towers to prevent the heating of river water entailed, in the case of the Monticello power plant (545 MWe), an additional capital cost of $5 \cdot 10^6$ u.a. or 9.2 u.a./kWe (Eart Ewald, Chairman of NSP).

(d) in part also, the fact that the optimism consequent on the emergence of this new energy source represented by nuclear power and the urge to become familiar as soon as possible with the new techniques caused electricity producers to anticipate their equipment programme.

The combination of these various factors brought about a price revision entailing, for projects at the construction stage, an additional burden amounting to 50% and sometimes more of the initial cost. (For the Oyster Creek power plant, the specific capital cost, evaluated in 1964 as 104 u.a./kWe, increased to 132 u.a./kWe, while for the Tennessee Valley Authority power plant (Browns Ferry 1 and 2), this cost, evaluated in 1961 as 112 u.a./kWe, rose to 166 u.a./kWe.)

During this period (1967-1969), the specific capital cost of nuclear power plants of 800-900 MWe capacity increased from 120 u.a./kWe to about 200 u.a./kWe.

Community

In the Community, orders for nuclear power plants, from mid-1967 to mid-1969, were at an almost total standstill. Several reasons may be advanced for this:

(1) The fragmented structure of the European electricity grid makes it very difficult to introduce large power plants.

(2) The difficulties of construction, the somewhat disappointing results of operation of certain nuclear power plants installed in the Community and the persisting doubt as to the competitivity of the energy produced made the producers cautious.

(3) The Community coal available as a result of the maintenance in activity of mines subsidized by the authorities. Large-scale recourse to nuclear energy for electricity production would undoubtedly have caused an acceleration of mine closures and serious social unrest. Moveover, certain nationalized electricity production companies were obliged to obtain their supplies in part from the nationalized mines.

(4) The attraction of the prices charged by oil suppliers to combat the threat to oil outlets which nuclear energy represented (drop of the order of 40% in France between 1960 and 1969) caused the threshold of competitivity of nuclear power plants to recede.

(5) The hesitation of electricity producers when confronted with the variety of types proposed by the constructors (PWR, BWR, GG, AGR, CANDU, etc.).

(6) The industrial and energy policies of each of the Six also put a brake on the development of nuclear energy for electricity production.

The rise in prices in the Community, as stated by the KWU in its first annual report (published in April 1970), added 40% over the prices charged in Germany three years ago and, according to EdF, 14% for the period from 1 January 1969 to 1 April 1970.

The specific capital cost for power plants ordered in 1969 and 1970 in the Community, the commissioning of which will be staggered over 1973-75, is in a range of roughly 140-260 u.a./kWe (1970 values). The width of this range can be explained by the various industrial situations in the Community countries and by the <u>de facto</u> maintenance of a certain national fragmentation of the market.

3.1.3 <u>Period covering the seventies</u>

For power plants ordered during the first half-decade, there is every reason to expect, at constant value, a stabilization of prices at the 1969-70 levels. Towards the end of the decade, prices should again take a downward turn.

There is no doubt that the above-mentioned factors causing the high increase in capital costs in recent years will gradually be brought under control and numerous uncertainties will thus cease to exist. The consequent savings should enable the effects of the general increase in material and manpower costs to be offset.

The tendency of electricity producers to install a number of units of equal capacity in the same place should not only permit more advanced standardization and accordingly a reduction in costs be repetition, but also diminish the effects of certain cost factors.

Thus, the capital costs of a power plant equipped with two or more units will be favourably influenced by the savings derived from:

(a) site preparation;

(b) the use of common installations (cooling water supply line, energy offtake lines, control room fuel handling machine, cooling plant, irradiated fuel storage installation, decontamination equipment, etc.);

(c) reduction of the stock of spares;

(d) appreciable reduction of operating and maintenance personnel.

The saving resulting from the construction of two similar reactors on the same site, as against a single reactor, is estimated to be of the order of 10%.

The unwillingness of electricity producers to put up with the prices and delivery periods imposed by the major constructors is a further price-restraining factor and may even lead to a certain lowering. The electricity producers continually play off the "giants" against one another and are increasingly applying to other constructors for the supply of nuclear power plants. (In the United States, of 16 power plants ordered in 1970, four were ordered from Combustion Engineering and two from Babcock and Wilcox.)

In the United States, the constructors who overequipped in an effort to catch up on the accumulated delay will probably decide to charge more attractive prices, or at least to maintain the present prices in the face of inflation.

The estimates on the trend of capital costs in the coming years made in 1969 by electricity producers agree with those by constructors and the USAEC*.

For power plants of over 1,000 MWe, they arrive at the values shown in the table below.

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*Edison Electric Institute - Detroit Edison Service - Wright Senior Consultant, Westinghouse IAEA/SM 126 (Istanbul Symposium) - Wash 1150 - Nuclear Industry 1969 (USAEC).

Estimates of the trend in the United States of the specific capital cost of light-water power plants (in u.a./kWe, constant value 1970*)

Year commissioned	Unit capacity 800 MWe	1,000 MWe
1975	200-210	180-220
1980		155-190
1985		145-175

*The original values expressed in constant 1975 u.a. have been reduced to a 1970 value by the introduction of an arbitrarily estimated 10% inflation.

For the Community, the present situation is still far from uniform, because of:

(a) the unequal degree of maturity attained by the industry,

(b) the existing industrial structure in the various countries,

(c) the absence of any real interpenetration of markets.

It features construction work in progress which may be divided into two groups costwise:

140-160 and 220-260 u.a./kWe (expressed in 1970 u.a.),

In the near future the average cost can be expected to settle at roughly 170-220 u.a./kWe for 800-1,000 MWe capacity, thus levelling with the values given for the United States.

There are grounds for believing that the industry will in a few years achieve an equal degree of maturity throughout the Community and that it will have access, as the Community market opens up, to a much wider sales field in which competition will be one of the decisive factors.

According to the recent estimates published by the USAEC (end of 1969), the capital cost of a 1,000 MWe light-water plant is broken down as follows:

<u>Estimate of the breakdown as %* of capital cost for a 1,000 MWe installation in the United States</u>

Nuclear boiler (NSSS)		
Principal components		
Reactor pressure vessel	3.6	
Steam generator (PWR)	6.9	
Pressurizer (PWR)	0.5	
Primary pumps	1.5	
Instrumentation, control rods	6.0	
Valves, circuits, storage tanks	4.1	
Miscellaneous	7.0	
		29.6
Turboset	20.0	
Condenser, feed-water heat exchanger, feed-water pumps, etc.	2.8	
		22.8
Leaktight containment		4.1
Project superintendent, project engineering and construction		26.9
Interest during construction**		16.6
		100%

*Estimated on the basis of 60% PWR and 40% BWR

**Calculated on the basis of an 8% interest rate.

For the Community, on the basis of the information possessed by the Commission, the capital cost at the beginning of 1970, according to the breakdown used in the "Euratom Economic Handbook" (published 1966) for a power plant of the order of 1,000 MWe, is as follows:

<u>Estimate of the breakdown as % of capital cost for a power plant of the order of 1,000 MWe in the Community</u>

(a) Direct cost, comprising		70.8%
(1) Acquisition and survey of site	0.6	
(2) Site preparation	0.2	
(3) Civil engineering	10.4	
(4) Reactor equipment	26.8	
(5) Turboset	24.9	
(6) Electrical equipment	5.7	
(7) Auxiliary equipment	1.9	
(8) Initial spares	1.2	
(b) Indirect cost, comprising		29.2%
(1) Studies, design and inspection	6.3	
(2) General expenses during construction	3.5	
(3) Sundry expenses - unforeseens	2.4	
(4) Plant operation expenses during tests	2.3	
(5) Interest during construction	14.7	
Total, direct and indirect costs		100%

3.2 Cost of fuel cycle

The cost of the fuel cycle depends in particular on the following factors:

- cost of natural uranium
- cost of conversion of natural uranium into UF_6
- cost of enrichment
- cost of fabrication of fuel elements
- cost of transport of irradiated fuel
- cost of reprocessing of irradiated fuel
- credit for recovery of fissile materials.

Of the cost factors, only fuel element fabrication presents features specific to the light-water type of reactor.

The other cost factors, expressed as 1970 constant values, are taken from the Annex "Fuel Cycle" and summarized in the table below:

Table 3.2.1

Year commissioned		1975	1980	1985
Cost of nat. U in form of U_3O_8	u.a./lb U_3O_8	6.0	6.0	6.0
Cost of conversion of U_3O_8 into UF_6	u.a./kg U	2.3	2.3	2.3
Transport of irradiated fuel	u.a./kg U	5	5	5
Cost of reprocessing irradiated fuel (incl.storage radioactive waste)	u.a./kg U	35	35	35
Reconversion into UF_6	u.a./kg U	4.5	3.0	2.8
Value of recovered fissile Pu	u.a./g Pu	7.0	7.0	7.0

For the enrichment cost, the value adopted for the years in question in the above table is 32 u.a./kg USW, set by the USAEC as from September 1971.

Cost of fabrication of fuel elements

Fuel element fabrication comprises a number of stages, viz.:

Conversion of enriched UF_6 into UO_2
Fabrication of pellets
Fabrication of fuel element cans and structural components
Canning of pellets
Final inspection
Packing and transport to power plant.

The cost of each of these operations are difficult to determine, since they depend in particular on:

quality control
plant size
volume of orders
specifications and warranties required
licence agreements, etc.

Table 3.2.2 gives recent forecasts by Community and US manufacturers on the trend in fabrication costs for light-water reactor fuel elements.

An AEG study dating from 1969 gives the sensitivity of fabrication costs to the works' annual production capacity and forecasts a reduction of the order of 10% when the capacity increases from 100 to 200 t U/year.

A further study* indicates a reduction of 18% where the increase is from 100 to 500 t U/year.

The AEG values appear low in comparison with the American figures, which may, however, be based on a market three times larger than the Community market.

A 1968** study gives the fuel fabrication costs as 70-120 u.a./kg U, depending on the type of services included in the contracts and the supplier considered.

The maximum value of the range will probably be lowered appreciably in the future, through:

Standardization of fuel elements,
increase in the size of production units,
technical improvements,
easing of quality control,
increased competition.

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* Gupta et al.: "Expected fuel fabrication costs in an expanding nuclear economy - Proceedings IAEA/SM-105/28 1968.

** Current status and future technical and economic potential of light water reactors, USAEC, Wash 1082, March 1968.

A more recent study published jointly at the Foratom Congress 1970 by the German makers of light-water reactor fuel elements gives a breakdown of fabrication costs for BWR and PWR assemblies, as a function of works production capacities (cf. Table 3.2.3).

It is interesting to note that the quality of fabrication places a considerable economic role. By way of example (assuming a replacement energy cost of 3 mills/kWh), a single 10-day shutdown of an 800 MWe power plant caused by defective fuel elements would ential an additional expenditure of 600,000 u.a. This is equivalent to an extra cost of 300 u.a./kg U for the replacement fuel (1/3 of core), of improved quality, to avoid the above-mentioned shutdown.

TABLE 3.2.2

FORECASTS OF FUEL ELEMENT FABRICATION COSTS (in u.a./kg U)

Year	1970	1975	1980	1985
Community:				
AEG (October 1969)[1]	82.5-85[4]		70[4]	
Jahrbuch ATW 1972	90-110		50	
United States:				
Edison Electric Institute (1969)		70	70	70
Westinghouse (October 1969)[2]	135-160[4]	105	94-74[4]	70
Westinghouse (1968)[3]	135-145[4]		86[4]	

[1] IAEA/SM-126/27 (values based on an annual production capacity of 100 t U/year, 1969 u.a.)

[2] IAEA/SM-126-23 (values for reloading fuel, 1969 u.a.)

[3] IAEA/SM-105/36

[4] Including conversion of U_3O_8 into UF_6

TABLE 3.2.3

BREAKDOWN OF BWR AND PWR FUEL ELEMENT FABRICATION COSTS

Fabrication cost factors	100 t U/year		250 t U/year		500 t U/year	
	PWR %	BWR %	PWR %	BWR %	PWR %	BWR %
Conversion	8.2	8.2	7.1	7.1	6.4	6.4
Pellet fabrication	11.8	9.6	10.4	8.6	8.6	7.1
Rod fabrication	16.1	13.6	14.3	12.1	12.9	10.7
Rod assembly	4.6	3.6	3.9	2.9	3.2	2.5
Inspection	9.3	7.1	8.2	6.4	7.1	5.4
Analysis, waste recovery, packing	3.2	2.5	2.9	2.1	2.1	1.8
Can material and assembly components	46.8	30.4	42.5	27.6	37.9	25.7
	100.0	75.0	89.3	66.8	78.2	59.6

These factors are evaluated on the reference basis of an installation producing 100 t/year of PWR fuel elements.

It thus appears that, in the long term, the standard of quality achieved by the various manufacturers will exercise a significant influence on their competitive position.

*

*   *

On the basis of construction work currently in progress in the Community, the range of fabrication costs is approximately 100 u.a./kg U for BWR's and 140 u.a./kg U for PWR's.

In the future, a reduction should gradually become apparent and these costs might reach values of the order of 75-95 u.a./kg U in 1980 and 50-70 u.a./kg U in 1985.

*

*   *

The cost of the fuel cycle for power plants under construction in the Community is between 1.6 and 2.0 mills/kWh.

On the basis of the values adopted in the foregoing Table 3.2.1 and taking into consideration an average fabrication cost of 120 u.a./kg U, 85 u.a./kg U and 70 u.a./kg U for power plants commissioned in 1975, 1980 and 1985 respectively, the breakdown of the fuel cycle cost at equilibrium would be as follows (in mills/kWh) for a 1,000 MWe power plant operating with a load factor of 75%:

Year commissioned:	1975	1980	1985
Fabrication	0.51	0.36	0.27
Fissile material burnup	0.86	0.86	0.86
Transport	0.02	0.02	0.02
Reprocessing	0.14	0.14	0.13
Reconversion of U and conversion of Pu	0.06	0.05	0.05
Pu credit	-0.19	-0.19	-0.19
Fuel cycle fixed capital charges (interest at 8.5%)	0.30	0.27	0.26
Total Mills/kWh	1.70	1.51	1.40

The American forecasts by the Edison Electric Institute are from 1.7 to 1.9 mills/kWh for 1975, 1.5 to 1.7 mills/kWh for 1980, and 1.4 to 1.6 mills/kWh for 1985. In these forecasts, plutonium recycling is allowed for from 1975 onwards.

An examination of the calculation hypotheses and the various fuel cycle cost forecasts leads to the assumption that the fuel cycle could reasonably be established in the vicinity of the following values for the Community:

Year commissioned	
1975	1.6-2.0 mills/kWh
1980	1.5-1.7 " "
1985	1.4-1.6 " "

3.3 Operating, maintenance and insurance costs

The operating, maintenance and insurance costs assumed by Community electricity producers for estimating the cost of energy produced by power plants to be commissioned in 1973-75 are between 4 and 5 u.a./kWe/year.

A tendency to drop towards 3 u.a./kWe/year should make its appearance for power plants industrially commissioned in 1980, and towards 2.5 u.a./kWe/year for those commissioned in 1985, as a result of the increased automation of the various operations required for running the plant and the creation, in the case of a number of power plant operators, of joint teams specializing in maintenance operations.

It is to be noted that the Edison Electric Institute predicts a constant value of 2.1 u.a./kWe/year from 1975.

* * *

3.4 Production cost per nuclear kWh

According to the economic data which can be gathered from recent contracts awarded for nuclear power plants and the tendencies taking shape for the future, as indicated above (secs. 3.1-3.3), the cost of energy produced in light-water reactor power plants fluctuates between 4.9 and 7.1 mills/kWh in 1975, between 4.3 and 6.0 mills/kWh in 1980 and between 4.0 and 5.5 mills/kWh in 1985 (Table 3.4.1).

The breakdown of the production price per kWh given in Table 3.4.1 below is based on the following hypotheses, according to the commissioning date of the power plants:

Date Commissioned		1975	1980	1985
Specific capital cost	u.a./kWe	170-220	155-190	145-175
Cost of fuel cycle	mills/kWh	1.6-2.0	1.5-1.7	1.4-1.6
Operating and maintenance costs	u.a./kWe/year	4.5	3.0	2.5

Hours of utilization
per year: 6,500 h, whatever the commissioning date.

Annual amortization instalments: 10 and 13% for all power plants

TABLE 3.4.1

ESTIMATE OF PRODUCTION COST PER NUCLEAR kWh FOR POWER PLANTS COMMISSIONED IN 1975 1980 AND 1985 (in mills, 1970 value

	1975				1980				1985			
Installation cost ua/kWe	170		220		155		190		145		175	
Hours of utilization/year	6500		6500		6500		6500		6500		6500	
Annual repayment %	10	13	10	13	10	13	10	13	10	13	10	13
Fixed capital charges mills/kWh	2.62	3.40	3.38	4.40	2.38	3.10	2.92	3.80	2.23	2.90	2.69	3.50
Fuel cycle mills/kWh	1.6	2.0	1.6	2.0	1.5	1.7	1.5	1.7	1.4	1.6	1.4	1.6
	4.5 ua/kWe/year				3 ua/kWe/year				2.5 ua/kWe/year			
Operation and maintenance mills/kWh	0.69		0.69		0.46		0.46		0.38		0.38	
Production costs, mills/kWh	4.91 / 5.31	5.69 / 6.09	5.67 / 6.07	6.69 / 7.09	4.34 / 4.54	5.06 / 5.26	4.88 / 5.08	5.76 / 5.96	4.01 / 4.21	4.68 / 4.88	4.47 / 4.67	5.28 / 5.48

For comparison purposes, the Edison Electric Institute forecasts made for the United States at the end of 1969 are summarized in Table 3.4.2 below. The basic hypotheses for this estimate are as follows:

- The year given is the date of industrial commissioning;

- The values are in constant 1970 units of account and are assumed to include all the anticipated effects on costs (with the exception of inflation after 1970) such as optimum size, recycling, project superintendent's costs, interest during construction;

- The fuel cycle costs levelled out over 10 years and based on fixed prices (8 u.a./lb U_3O_8, 26 u.a./kg U SW, 7.50 u.a./g fissile Pw, 70 u.a./kg U for fabrication of fuel elements, 45 u.a./kg U for transport and reprocessing;

- Annual utilization of the power plant assumed to be 7,000 h;

- An interest rate of 7% and an amortization rate of 14% per annum;

- Unit capacity of the plant, 2 x 1,150 MWe.

TABLE 3.4.2

ESTIMATES OF PRODUCTION COST FOR ELECTRIC POWER PRODUCED IN THE UNITED STATES

(in mills/kWh)

Year Commissioned	Specific Capital costs (u.a./kWe)	Fixed Capital charges	Fuel cycle cost	Operation, maintenance, insurance	Production cost of electric power
1975	180-220	3.6-4.4	1.7-1.9	0.3	5.6-6.6
1980	155-190	3.1-3.8	1.5-1.7	0.3	4.9-5.8
1985	145-175	2.9-3.5	1.4-1.6	0.3	4.6-5.4
1990	140-170	2.8-3.4	1.4-1.6	0.3	4.5-5.3
2000	135-165	2.7-3.3	1.4-1.6	0.3	4.4-5.2

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SECOND TARGET NUCLEAR PROGRAMME FOR THE COMMUNITY

ANNEX IV

THE FUEL CYCLE

(1 JULY 1972)

THE FUEL CYCLE

1. Introduction

1.1 General observations

The growth of nuclear energy production in the years ahead means that the fuel cycle will become an increasingly important field. According to the estimates of the Second Illustrative Programme, the net production of electricity from nuclear power plants will rise to nearly $235 \cdot 10^9$ kWh in 1980 and to $550 \cdot 10^9$ in 1985, whilst the net installed electric capacity in the Community power plants will go from 3200 MW in 1970 to 45,000 MW in 1980 and to 100,000 MW in 1985. This large increase in installed nuclear power plant capacity, and even more so in the nuclear share of electric power production - in 1985, 25% of the installed capacity will provide 35% of the electricity generated - will require a parallel growth of the fuel cycle industry alongside the development of the reactor construction industry.

This industry covers all the activities concerning reactor fuels, from uranium prospecting, ore processing, production of concentrate, conversion, enrichment, reconversion, manufacture of fuel assemblies, down to the transport and reprocessing of irradiated fuels and waste storage.

The fuel cycle is not confined to uranium, but also covers other fissile and fertile materials, namely, thorium and plutonium.

This document deals in detail with the hypothesis that fast reactors and high-temperature reactors (HTR) will begin to come in after 1985, the total electric energy production being provided by light water reactors until that time.

The question of the plutonium produced in thermal reactors will only be dealt with from the standpoint of recycling in thermal reactors, where it can take the place of enriched uranium. This case is dealt with in Section 4.

1.2 Economic aspects

Although the annual volume of expenditure for the LWR fuel cycle only amounts to 11,000 or 12,000 u.a. per MWe installed, with the result that the fuel cycle industry's turnover is at present appreciably lower than that of the reactor construction industry, the total annual expenditure for the fuel cycle (first cores and reloads) will build up in the coming years so that the fuel cycle industry's turnover will rise from about 190 million u.a. a year in 1975 to 560 million a year in 1980 and 1100 million a year in 1985 (see Section 5).

These figures alone clearly show the considerable importance that the fuel cycle industry will assume in the future.

1.3 Quantitative aims

In the following sections each stage of the fuel cycle is dealt with in detail, particularly from the angle of the dynamic development of nuclear power production.

By comparing the fissile materials requirements resulting from this growth with the existing resources, it is possible to define the resources that will be needed at each stage as a function of time. Similarly the existing industrial capacity can be compared with the capacity that must be installed at future stages.

As a general guide the table below gives some indicative overall figures which give an idea of the growth of the market for light water reactors:

	1975	1980	1985	Period 1975-85
Requirements in natural U[1] (tons)	4,200	10,800	20,500	126,900
Requirements in 10^3 kg separative work units[2]	1,640	5,920	12,600	70,500
Requirements in enriched U (tons) for fabrication - first core	370	1,030	1,850	11,820
- refuelling	210	980	2,700	12,500
Tonnage of irradiated fuels to be reprocessed, in tons U	110	720	1,940	9,360
Total plutonium in kg[1]	820	5,450	14,900	71,500

[1] The gas/graphite reactors, of which the installed capacity in 1975 is 2500 MWe, need 540 tons a year; their plutonium production is 1.3 tons a year.

[2] Wastage 0.25%.

1.4 International aspect

Apart from the question of resources, requirements and capacities, other aspects of the final cycle deserve careful examination.

The fuel cycle industry's market has an international character, with the result that the problems it poses must be considered in a wider context than that of the Community countries. This international character, is due to several factors: the historical evolution of the nuclear industry, the geographical distribution of raw materials, the ease of transport of the various intermediate products. In any discussion of the industrial capacity yet to be created in order to meet the demands imposed by the growth of nuclear energy, due consideration must therefore be paid both to the Community's industrial situation and to the world market and its probable development in the future.

On account of the frequently low level of investment in certain sectors of the fuel cycle, and because of the policies pursued in the various countries in the past and delays in nuclear development, this sector of the industry is now highly developed, so that there is often excess production capacity, except, of course, as regards uranium enrichment in the Community.

2. Basic data and calculation results

The illustrative programme expresses the desired development of the production of nuclear energy in terms of installed MWe and TWh to be produced in future years. In order to be able to draw conclusions on the effects which these developments will have in the fuel cycle field, these figures have been converted by means of a computer program into quantities of fissile materials needed in the various stages of the cycle, and into fabrication capacity.

In accordance with the program hypothesis, only the light-water reactors will be involved during the period of the programme (1975-85). The main characteristics of these reactors, as used in the calculations, are given below in Table 2.2, namely, for a boiling-water reactor (BWR), a pressurized-water reactor (PWR) and a light-water reactor of intermediate characteristics ($\frac{1}{2}$ BWR + $\frac{1}{2}$ PWR), each of 1000 MWe.

These characteristics were adjusted in keeping with the most recent data. A typical light-water reactor, having the above-mentioned intermediate characteristics, was taken into consideration for the calculations.

The energy programme used as a basis for the calculations is given in Table 2.1.

The operating hours shown in this table are mean values calculated on the supposition that the hours of operation of a nuclear power plant (as in normal power plants) are equal to:

3000 hours in the first year,
5000 hours in the second year,
6500 hours from the third year onwards.

In the light of the assumptions regarding the growth of installed capacity and energy production, an attempt was made to evaluate:

- the net requirements in natural uranium at the level of the mine (Table 2.5);
- the requirements in enriched uranium, a distinction being made between the needs for the first cores (including reserve) and the reload cores at the level of the fabrication plant (Table 2.7);
- the requirements in separative work units at the level of the enrichment plant (Table 2.6);
- the amounts of uranium recovered from irradiated fuels and the amounts of plutonium produced, at the reprocessing level (Tables 2.8 and 2.9).

As regards the moment when the requirements become evident, the timescale involved in the various operations of the fuel cycle was taken into account (Table 2.3).

The losses inherent in these operations are also counted (Table 2.4).

Table 2.1

Energy programme

Year	Net installed capacity[2] GWE	Production in net TWh	Average utilization total capacity
1975	11.2	58.9	5,130[1]
1976	15.5	74.6	4,550
1977	21.6	98.6	4,620
1978	27.5	130.8	4,740
1979	36.1	172.4	5,000
1980	45.0	221.9	5,220
1981	54.0	277.0	5,240
1982	63.8	336.0	5,280
1983	75.0	400.3	5,300
1984	87.5	471.0	5,400
1985	1000.0	547.9	5,500

[1] Abnormal figure owing to very limited amount of new plant in 1972-75.
[2] Including reactors other than LWR in existence in 1975.

Table 2.2 - Characteristics of the reactors

Type of reactor Size MWe	BWR 1000	PWR 1000	LWR $\frac{1}{2}$ BWR, $\frac{1}{2}$ PWR 1000
Net efficiency MWe/MWth	0.33	0.32	0.33
Specific power MWe/t	7.21	11.31	8.93
Mean burnup MWd/tg			
First core	20.9	21.8	21.35
Equilibrium Core	27.5	33.0	29.69
Initial enrich. w/o			
First core	2.19	2.37	2.27
Equilibrium core	2.56	3.30	2.90
Final content at equilibrium			
- U^{235} w/o	0.61	0.83	0.71
- Pu^{239}	0.45	0.52	0.48
- Pu^{241}	0.09	0.12	0.10
- Pu^{240}	0.20	0.24	0.22
- Pu^{242}	0.04	0.04	0.04
Inventory			
First cores kg/MWe (incl. reserve)	141.4	90.2	114
Operating requirements at equilibrium t.enr. U 10^9kWeh	4.62	3.89	4.25
Recovery t.irr. U 10^9 kWeh	4.53	3.81	4.17
Irradiation time			
- fuel - years			
- first core	2.60	1.72	2.18
- equilibrium core	3.42	2.60	3.30
Time required to balance first core - years	4.25	3.00	3.75
Tails assay: w/o	0.25%		
Quantity of nat. U needed to produce 1 kg enr. U (kg nat U/kg enr. U)			
- at level required by the			
- first core	4.21	4.60	4.38
- equilibrium core	5.01	6.62	5.75
- at level attained in irradiated fuel	0.77	1.25	1.00
Amount of separative work units needed to produce 1 kg of enr. U (kg SWU/kg enr. U)			
- at level required by the			
- first core	2.22	2.57	2.38
- equilibrium core	2.93	4.37	3.57
- at level attained in irradiated fuel	0.08	0.12	0.00
Total Pu production kg/10^9 kWeh	35.35	34.73	35.09
Fissile Pu production kg/10^9 kWeh	24.58	24.14	24.38

Table 2.3

Time required for various operations

Type of reactor Size MWe	LWR $\frac{1}{2}$ BWR, $\frac{1}{2}$ PWR 1000
	Years
1. Concentration, transport	0.50
2. Coolings, transport, reprocessing	0.75
3. Conversion, enrichment	0.50
4. Reconversion of UF_6 into UO_2	
4.1 First core	0.15
4.2 Equilibrium core	0.10
5. Fabrication, transport On-site storage	
5.1 First core	0.35
5.2 Equilibrium core	0.25
6. Time between start of loading and industrial utilization	0.30

Table 2.4

Losses inherent to the various operations

	%
1. Purification of nat. U_3O_8 and conversion of nat. U_3O_8 into UF_6	0.50
2. After enrichment, reconversion enr. UF_6 into UO_2	0.50
3. Fabrication	0.50
4. Reprocessing	1.00
5. Reconversion of enr. UNH into UO_2	0.30

Table 2.5

Annual requirements of natural uranium at the mine level (tons)

(a) Tails assay 0.2%

Year	LWR				GGR
	First cores Reserves incl.	Consumption	Recovery	Net requirements	
1975	2630	1400	200	3830	540
1976	2770	2030	320	4440	540
1977	3690	2780	380	6080	540
1978	4100	3980	550	7530	540
1979	4170	5200	770	8600	540
1980	4460	6530	1050	9940	540
1981	5050	8030	1330	11750	540
1982	5660	9380	1600	13440	540
1983	5810	11050	1880	14980	540
1984	6450	13000	2160	17290	540
1985	6700	14600	2340	18870	540
1975-85	51460	77980	12670	116750	5940

(b) Tails assay: 0.25%

1975	2860	1530	220	4175	540
1976	3020	2210	350	4880	540
1977	4020	3030	410	6630	540
1978	4480	4330	600	8200	540
1979	4550	5670	840	9380	540
1980	4870	7120	1150	10840	540
1981	5500	8750	1450	12800	540
1982	6170	10230	1750	14650	540
1983	6340	12040	2050	16330	540
1984	7030	14170	2350	18850	540
1985	7300	15920	2650	20570	540
1975-85	56140	85000	13820	126900	5940

Table 2.6

Annual requirements in separative work units
(Kg/year)

(a) Tails assay: 0.2%

Year	First cores	Consumption	Recovery	Net requirements
1975	1.01	0.87	0.04	1.84
1976	1.76	1.08	0.04	2.80
1977	1.86	1.56	0.06	3.35
1978	2.47	2.13	0.07	4.53
1979	2.75	3.05	0.11	5.70
1980	2.80	4.00	0.14	6.65
1981	2.99	5.01	0.20	7.80
1982	3.38	6.17	0.25	9.30
1983	3.80	7.21	0.28	10.72
1984	4.20	8.40	0.32	12.28
1985	4.63	9.40	0.37	13.66
1975-85	31.65	48.88	1.89	78.65

(b) Tails assay: 0.25%

1975	0.90	0.78	0.03	1.64
1976	1.57	0.96	0.04	2.49
1977	1.65	1.39	0.06	2.98
1978	2.20	1.90	0.06	4.03
1979	2.45	2.72	0.10	5.07
1980	2.49	3.56	0.13	5.92
1981	2.66	4.46	0.18	6.95
1982	3.01	5.49	0.22	8.28
1983	3.38	6.41	0.25	9.54
1984	3.74	7.48	0.28	10.93
1985	4.12	8.37	0.33	12.60
1975-85	28.17	43.50	1.68	70.50

Table 2.7

Annual requirements in enriched uranium at the fabrication plant level (tons)

Year	First cores	Consumption
1975	370	210
1976	650	260
1977	680	380
1978	910	520
1979	1010	740
1980	1030	980
1981	1100	1220
1982	1240	1500
1983	1400	1750
1984	1550	2200
1985	1850	2650
1975-85	11820	12410

Table 2.8

Annual quantities of uranium to be reprocessed (tons)

Year	LWR	Gas/graphite
1975	110	540
1976	200	540
1977	250	540
1978	370	540
1979	500	540
1980	720	540
1981	940	540
1982	1180	540
1983	1450	540
1984	1700	540
1985	1940	540
1975-85	9360	5940

Table 2.9

Annual quantities of plutonium recovered from fuels (tons)

Year	LWR Fissile Pu	LWR Total Pu	Gas/graphite Fissile Pu	Gas/graphite Total Pu
1971	0.20	0.29	0.53	0.75
1972	0.24	0.33	0.62	0.90
1973	0.38	0.52	0.78	1.10
1974	0.56	0.78	0.85	1.25
1975	0.58	0.82	0.87	1.30
1976	1.10	1.54	0.87	1.30
1977	1.36	1.91	0.87	1.30
1978	1.97	2.76	0.87	1.30
1979	2.71	3.80	0.87	1.30
1980	3.88	5.44	0.87	1.30
1981	5.07	7.11	0.87	1.30
1982	6.40	9.00	0.87	1.30
1983	7.92	11.15	0.87	1.30
1984	9.35	13.10	0.87	1.30
1985	10.70	14.90	0.87	1.30
1975-75	51.05	71.52	9.60	14.30

3. Stages in the Fuel Cycle

3.1 Supplies of natural uranium

3.1.1 Requirements

On the basis of the predicted construction of nuclear power plants in the Community, and according to the hypotheses as to their utilization rate, Table 2.5 shows the requirements in natural uranium needed, at the mines' level, to fulfil this programme.

3.1.2 Uranium resources

3.1.2.1 Definitions and terminology

"Reserves" and "resources" are terms ordinarily used for the quantitative determination of some of the various geological, technical and economic aspects of the mineral potential, using the "available product weight" as the common denominator, in a given macro-economic situation.

For this reason, all estimates of mineral reserves or resources must be regarded in the context of the limits and hypotheses formulated (by the estimator) for the specific purposes of that estimate at the time at which it was made.

In this terminology, the term "reserves" is only applicable to the estimated quantities of ores considered as "usable" in present conditions, whereas "resources" means "reserves" plus all ores likely to become usable in the widest sense of the work and in the most favourable conditions.

Thus:

resources = reserves + marginal resources + sub-marginal resources + latent resources

or, in other words:

resources = reserves + potential reserves

The terminology used in estimating these resources and reserves is based partly on our present-day - incomplete - knowledge of the quantity of useful material in the ore deposits (quantitative aspect) and partly on the cost of their extraction, concentration or refining, transport and commercial processing, compared with a given market price of the mineral potential (quantitative aspect) (Fig. 1).

3.1.2.2 Community resources

(a) On Member States' territory

Since the publication of the First Illustrative Programme, the quantity of reasonably available and usable resources at a price of less than 8-10 u.a. per pound of U_3O_8 has risen from 30,600 tons to 36,200 tons of uranium.

These reserves are mainly located in France in the three mining districts of Forez-Grury, Crouzille and Vendeé, all operated by the CEA, and also in favourable areas in Brittany and the Massif Central. Lastly, there is a deposit in the Lodève (Hérault) Permian basin, which for the moment is not being mined.

In addition, there is a reserve of about 1,200 tons of U at Novazza in Northern Italy.

The total quantity of possible additional uranium resources at this price has remained at a constant level of 20,000 tons.

Over the past five years, production has risen to about 8,000 tons of uranium.

The average uranium content in the ore has risen from 0.14% to 0.185%.

This illustrates the fact that these estimates are based on the cut-off contents (i.e. those below which operation is no longer profitable), thus reflecting the present-day economic situation of the uranium mining industry.

It is for this reason that these resources represent reserves in the sense defined above.

(b) Resources controlled by the Community's mining industry in non-member countries

Important discoveries, resulting from prospection by the European mining industry in non-member countries have increased the resources controlled mainly by the CEA and French industry to the following amounts:

	Reasonably assured	Possible additions	Production capacity for 1974
Gaboon	15,000	-	700
CAR	8,000	8,000	500
Niger	20,000*	30,000	1,500
	43,000	38,000	2,700

The growth of production capacity beyond this level is linked to market developments.

*Participation of the ENI (Italy) and the Uraupesellschaft (Germany).

(c) Resources situated in non-member countries, mined in collaboration with Community firms

A large deposit in Canada - Rabbit Lake - is to be mined with the participation of the Uranerzbeigban and is hoped to yield quantities of 1000-2000 tons a year.

(d) Prospecting

Prospecting activities over the past seven years, which have been mainly French, have resulted in the discovery of about 70,000 tons.

Capital expenditure on prospecting in the Community and non-member countries by the Community's mining industry can be estimated at 8-10 million u.a. in 1969, and 14-16 u.a. in 1970.

This large increase is due to greater participation by Germany and Italy in prospecting activities.

In order to guarantee a secure and regular supply from the resources controlled by the Community's industry, these investments should be gradually stepped up to 20-25 million u.a. during the period 1985-2000.

Although the prospecting costs should ultimately be recovered from the sale of the product and normally be reinvested to replace the amounts extracted, this aim cannot be fully achieved during the term of the illustrative programme, and government aid appears to be necessary.

These developments might even ensure that the Community can meet all its requirements from its own mining industry.

If present prospecting activities continued to be as rewarding as those carried out recently, the Community's mining industry could produce enough uranium to cover its own needs and even more.

3.1.2.3 World uranium resources

Table 3.1. gives the latest estimate of uranium resources in the world, with the exception of the USSR, Eastern Europe and China, made in April 1970 (ENEA/IAEA, September 1970) and revised according to the new data published at the fourth international UNO Conference on the use of atomic energy for peaceful purposes, Geneva 1971.

The resources are divided into two price-categories: those that can be extracted at less than 10 u.a./lb U_3O_8 and those costing between 10 and 15 u.a./lb U_3O_8.

The reasonably assured resources and possible additional resources were separated in each of these groups.

For the United States and Canada at any rate, the estimated resources costing less than 10 u.a./lb U_3O_8 also include resources which cannot be profitably extracted under present market conditions.

Table 3.1
WORLD RESOURCES
in thousands of metric tons of uranium

Country	<10 u.a./lb U_3O_8			10-15 u.a./lb U_3O_8	
	Reasonably assured resources	Possible additions	Estimated content % U	Reasonably assured resources	Possible additions
United States	228 70	520 -	0.144 by-product	130 15	275 -
South Africa + recent discoveries	154 75	11.5 -	by-product (0.02) unknown	49 -	27 -
Canada	178	177	1.00	99	129
France	32	19	1.85	7	12
Italy	1.2	10	-	-	-
Germany	-	10	-	-	-
Niger CAR Gabon	20 8 15	30) 8) -)	2.8	10 - -	10 - 6.5
Australia + recent discoveries	16.7 100	5.1 -	0.6 - 7.6 >2	7 -	5 -
Spain	8.5	-	1.7	8	-
Argentina	7.7	17	0.9 - 1.3	8	25
Portugal (Europe) (Angola)	7.4 -	6 -	1.7 -	- -	11 11
Japan	2.1	-	-	3.4	-
Mexico	1.0	-	-	1.2	-
Brazil	0.8	0.8	8.5	-	-
Sweden	-	-	-	266	38
Denmark	-	-	-	4	-
India	-	-	-	2.3	0.8
Round total	930			600	

The Community's commercial position might therefore be better than it appears from a simple comparison of the figures.

Table 3.2 compares the supply and demand for natural uranium. It also shows that reasonably assured world resources cover forecast requirements for the next ten years almost twice over.

Similarly, the Community's reserves easily cover the requirements of the next ten years and approximately 50% of the planned requirements up to the end of 1985.

3.1.3 Production capacity

3.1.3.1 Community

Installed production capacity for the Community's reserves could enable 1,800 tons of uranium to be produced annually in concentrate form. Present output is of the order of 1,300 tons of uranium a year, this being principally on three plants in France, at Ecarpière (Vendée, 300,000 tons of ore a year), Bessines (Limonsin, 600,000 tons a year) and St Priest (Forez, 180,000 tons a year).

Table 3.2 - Uranium resources and requirements for the period 1970-85

	COMMUNITY							WORLD			
	Resources* 78,200 tons of uranium							930,000 tons of uranium			
Year	Cumulative net installed nuclear capacity GWe	% World	Requirements in 10^3 tons U Wastage rate 0.2%		0.25%		Production Capacity 10^3 tons U	Cumulative net installed nuclear capacity GWe	Requirements in 10^3 tons U		Production capacity 10^3 tons U
			year	Cum.	year	Cum.			year	Cum.	
1970	3.2	17.8	1.1	-	1.1	-	1.0	18 -	9	-	16.4
1971	3.3	14.6	1.2	2.3	1.3	2.4	2.5	26	12	21	
1972	5.7	13.6	2.8	5.1	3.0	5.4	-	42	16	37	
1973	6.7	11.5	1.8	6.9	1.9	7.3	3.9	59	20	57	28.4
1974	10.4	13.3	3.0	9.9	3.3	10.6	4.7	85	23	80	
1975	11.2	10.3	4.4	14.3	4.7	15.3		118	29	109	39
1976	15.5	12.2	5.0	19.3	5.4	20.7		148	34	143	
1977	21.6	13.3	6.6	25.9	7.2	27.9		180	38	181	
1978	27.5	14.1	8.1	30.4	8.7	36.6		220	44	225	
1979	36.1	14.6	9.1	43.1	9.9	46.5		260	50	275	
1980	45.0	15.0	10.5	53.6	11.4	57.9		300	56	331	
1981	54.0	-	12.3	65.9	13.3	71.2		-	-	-	
1982	63.8	-	14.0	79.9	15.2	86.4		-	-	-	
1983	75.0	-	15.5	95.0	16.9	103.3		-	-	-	
1984	87.5	-	17.8	112.8	19.4	122.7		-	-	-	
1985	100.0	(16.7)	19.4	132.2	21.1	143.8		(500)	(100)	-	

*Reasonably assured at less than 8-10 u.a./lb U_3O_8 and situated on Community territory or under Community control.

The installation of further capacity would depend on large new reserves being discovered, as the potential of the known reserves will be exploited at a relatively low rate, so as to make them last 30 years or more.

3.1.3.2 World

At the moment there is a temporary over-production of uranium, the demand being well below the production capacity of the existing mines. In 1973 this production capacity will be 29,000 tons.

It is not until 1977-78 that demand will exceed the 39,000 tons production capacity which could be installed on the basis of known resources. New resources will have to be found between now and then in order to match the supply to the demand.

3.1.4 Technical processes

Hydrometallurgy has been developed, first in America and subsequently in the other producer countries, as a technique for processing ores.

This method enables ores with a low metal content to be used and is characterized by high rates of concentration and yields.

Without going into the details or past history of preconcentration methods, it will be recalled that the most commonly used method for removing the deads found in uranium ore is based on radioactivity.

The processes involving the chemical attack of the preconcentrated ores make use of the fact that uranium is in general easily soluble in an acid or alkaline medium.

Once dissolved, the uranium can either be chemically precipitated or it can be extracted by ion exchange on resius or by organic solvents which permit an initial purification, a very high concentration rate and a recovery of close on 100%.

Various methods of attack are employed to dissolve the uranium contained in the ores.

The principal factor to be considered when determining what type of acid or alkaline attack should be used is the nature of the mineral gangue accompanying the uranium ores.

3.1.5 Investments

The total capital investment needed to work a known deposit and to install an ore processing plant for the production of U_3O_8 depends on the type of deposit (open-cast or below ground) and on the content and nature of the ore. From past experience, this amount may be estimated to be in the range of 25,000-35,000 dollars per ton of uranium per annum. It does not include prospecting costs, which can be averaged at about 2,000 u.a./ton for deposits with a mean content of 0.185% U and a mean overall content of 10,000 tons U.

The economies of scale which apply to these capital costs manifest themselves to a much greater extent as regards chemical ore processing plants.

The capital costs versus size for American acid attack plants are given below by way of illustration.

Daily capacity in tons of ore	Capital costs in u.a. per ton of ore processed daily
450	6,800 - 8,300
900	4,700 - 5,800
1,800	4,100 - 5,100
5,000	3,000 - 3,800

3.1.6 World uranium market and long-term prospects

Recent discoveries of major reserves with a very high content in Australia (Nabarlek, Ranger I) and in Canada (Wollaston Lake) and very large low-content reserves in South-West Africa (Rossing, Swankopmund) mean that the conclusions which might have been drawn from the ENEA-IAEA report of 1970, must be revised.

It is likely that these very high content reserves can be worked in the near future, production costs being appreciably lower than 6 u.a./lb U_3O_8. The same will undoubtedly apply to the South-West African reserves, in the event of a substantial growth in the market.

It is therefore likely that for the duration of the Illustrative Programme, the uranium market will remain a buyer's market, with supply exceeding demand. Prices will remain well below the figure of $10/lb U_3O_8 adopted by the ENEA in its estimate of available resources.

A mathematical calculation model[1] was made for the statistical evaluation of the reserves and for a long-term estimate of the operating costs (Fig. 2).

[1] Eurospectra, June 1971, Vol. X, No. 10.

On the basis of the estimated reserves for 1967, 1969 and 1971, illustrative operating costs were calculated by means of this code.

Year	Total reserves (tU)	Average tonnage of deposit (t U)	Average content (%)	Average working costs (u.a.)
1967	500,000	4,000	0.15	7.32
1969	717,000	4,000	0.165	6.78
1971	430,000	10,000	0.185	5.19

Whereas world demand for uranium up to the year 2000 is estimated at between 2.5 and 4.5 million tons, this mathematical model shows the existence of potential reserves of several tens of millions of tons at mining costs of less than 6 u.a./lb U_3O_8. Of course, these potential reserves have yet to be discovered.

In view of this situation and the trend of the market over the past-few years, a steady price of 6 u.a. (1970)/lb U_3O_8 has been assumed, this being considered the probable average value within a range extending from 4.5 to 7.5 u.a.(1970)/lb U_3O_8.

This hypothesis presupposes the continued parallel development of two quite distinct markets, namely:

- that of the United States, and
- that of the rest of the Western world*,

development in the US market being characterized by relatively stable and high prices. These circumstances must not be allowed to discourage exploration for new uranium finds outside the United States, and thus to create a shortage of discoveries in relation to the rapidly growing demand for uranium.

*The first, the United States market, is well protected and represents about 60% of the free world's short-term requirements up to 1975 and 50% of its production capacity. In the second, which covers the rest of the Western world, all uranium producers, including those in the United States, will be competing for the remaining 40% of the requirements.

3.2 Conversion of U_3O_8 (yellowcake) into UF_6 and reconversion of enriched UF_6 into UO_2

The conversion of the classical concentrate (U_3O_8) into uranium hexafluoride at the same time as it is purified, and the reconversion of enriched UF_6 into UO_2, are two essential stages in the fuel cycle of enriched uranium reactors. Although the relative cost is low, these operations nevertheless constitute a fairly well-developed special market. They will be dealt with here, in view of certain similarities they present, and despite the enrichment stage which separates them.

3.2.1 Conversion

On the assumption that a large Community enrichment capacity is set up, it may be attractive to carry out the conversion of the concentrate into UF_6 in the vicinity of the isotope enrichment facility. The relative ease with which products such as natural UF_4 and enriched UF_6 can be transported and the heterogeneous development of the fuel cycle industry have led to the geographical dispersal of capital investment in the Community. However, conversion is more economical in a large plant.

Two conversion processes are used industrially, namely the dry and the aqueous methods.

The former is used by Allied Chemical Corporation in its plant at Metropolis, Illinois. The conversion of the impure concentrate is performed by direct attack and all the operations are carried out on fluidized beds.

This latter technique is used in France, Britain and Canada. It consists in dissolving the impure concentrate, refining it by extraction, then precipitating and/or calcining the refined product in order to obtain pure UO_2 through reduction. This is then converted into UF_4 and finally UF_6.

The structure and capacity of the Community's conversion industry are as follows:

In France, the COMURHEX company (Société pour la Conversion de l'Uranium en Métal et Hexafluorure), which was set up on 1 January 1971, embraces the activities of the Malveri and Pierrelatte plants (conversion of UF_4 into UF_6). The present announced capacity is 6,000 t/a of uranium for conversion of concentrates into UF_6; for the time being, however, this capacity is limited in the fluorine production sector to a level corresponding to 3,000 t/a of uranium. Consideration is being given to extending this capacity to 10,000 t/a.

In Belgium, a plant with a capacity of about 600 t/a of contained uranium is owned by Métallurgie Hoboken. This plant has been shut down for a long time.

In Germany, Nukem owns plants for the conversion of concentrates into UF_4, the capacity being of the order of 100 t/a.

In Italy, a pilot plant for the conversion of U_3O_8 into UF_6 with a capacity of about 20 t/a of contained uranium is in the process of being set up, and its development is linked to the programme which this country has begun in the field of uranium enrichment.

In the UK, the conversion capacity is about 3000 t/a of contained uranium. The plant, situated at Springfield, is owned by British Nuclear Fuel Limited*.

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*Formerly the UKAEA production group.

As regards the Community, the total capacity is therefore at present about 3000 t/a of contained uranium, or 6000 t/a if the Springfield plant is included.

From the programme under consideration and the preceding technical studies, it is clear that the present capacities for converting concentrates into UF_6 are sufficient to cover Community needs until 1975.

In view of the fact that, according to French statements, the COMURHEX capacity for converting U_3O_8 into UF_6 could easily be stepped up to 6000 tons of uranium a year, the setting-up of new conversion capacities in the Community may not be necessary before 1977-78.

As regards the specific cost of conversion, Allied Chemical's basic price is £1.25/lb contained uranium, i.e. $2.76/kg U.

In Europe, the ruling prices are thoroughly competitive with those mentioned above.

Nevertheless, owing to the increasing vertical integration of the mining and conversion industries, there is a tendency to offer clients an all-in price for UF_6, including the price of the natural uranium.

3.2.2 Reconversion of UF_6 into UO_2

In this field, the Community possesses one large plant (400 t/year UF_6) in Germany owned by REG, and smaller plants in France and Belgium. In addition, the UK has one reconversion plant with an annual capacity of about 250 tons of contained uranium.

As regards capacity for the reconversion of slightly enriched UF_6 into UO_2, the existing plants linked with the fuel element fabrication facilities are also able to meet requirements up till 1975.

Regarding prices, the present trend is also to propose "package deals" covering the transportation of the UF_6, the reconversion into sinterable UO_2 and the manufacture of the fuel.

Capital investment

3.2.3 Conversion

The first section of the Allied Chemical Corporation's plant, which had a capacity of about 4500/t year of contained uranium, required an investment of about $11 million (1955). At present the plant has a capacity of 9000 tons of contained uranium a year. The investment required to effect this doubling of output was around 10 million u.a.

The Ma/vesi facility, which only goes as far as the UF_4 stage but also converts UF_4 into uranium metal, cost a total initial investment of nine million u.a. (1959) for a capacity of 1000 tons a year.

According to American data, the investment costs for a capacity of 5000 t/year, considered as the minimum profitable size, would be of the order of $\$20 \cdot 10^6$ for the dry method.

The investment cost for a European plant using the fluidized bed technique may be estimated at 5000 u.a. t/year for a capacity of the order of 4000-5000 t/year.

3.2.4 Reconversion

In the conversion of low-enriched UF_6 into UO_2, where the amounts to be processed are much smaller, the capital investment is less than in the case of conversion, despite the restriction imposed by the need to limit the dimensions of the equipment for reasons of criticality.

It is estimated that the investment cost for a plant of about 200 t/year of low-enriched uranium would be of the order of $\$2 \cdot 10^6$.

3.3 Uranium enrichment

3.3.1 Future separative work requirements*

3.3.1.1 Hypotheses relating to the evaluation of requirements

In view of the probable emergence of an international enrichment market, it is necessary to evaluate world requirements in order to define the Community's position; in particular, these requirements are dependent on the nuclear programmes planned (installed capacities, types of reactor). The year 1985 was chosen as the horizon for this calculation, as it is possible to describe the probable development of requirements up to this date reasonably accurately.

The evaluation of world separative work requirements are based on the nuclear power plant programmes given in recent documents drawn up by various countries, or presented in this report as regards the Community. These programmes are summarized in Table 3.3.

*Excluding USSR, East European countries and Communist China.

Table 3.3

Nuclear power plants' capacity in enriched uranium

(net GWe; at the end of the year)

Year	USA	Western Europe	Remainder of the "free world"	"free world"
1975	59	17	8	84
1978	108	49	23	180
1980	150	79	38	267
1982	199	109	57	365
1985	299	167	93	559

References:

- For Japan: Fourteenth JAEC Annual Report and paper p/298, Geneva Conference 1971.
- For the United Kingdom: values given at the Washington Conference on Enrichment, 16 November 1971.
- For the other countries: USAEC WASH-1139 report, January 1971.

They concern the installation of enriched uranium reactors, mainly of the light-water type, it being assumed that fast (or possibly advanced) reactors will go on the market after 1985.

The evaluation of the quantities of separative work, i.e., again enriched uranium, needed to achieve the LWR power plant programmes mentioned above necessitate the formulation of hypotheses concerning the enriched uranium consumption of the reactors with which they will be equipped.

This consumption, which is not the same for pressurized and boiling-water reactors, depends on the proportion of each variant making up the total installed capacity. It also depends on future possibilities of improving reactor performance, on their utilization factor and on the use which can be made of plutonium in light-water reactors.

For these reasons, the forecasts made are based on a collection of average hypotheses*, in addition, a certain amount of plutonium recycling has been assumed from 1975 onwards, which would not endanger the advent of fast breeder reactors during the period under consideration. Should the economic conditions for this period make recycling not worthwhile, the forecasts in Table 3.4 would have to be increased by 6-10% in 1980 and 10-15% in 1985**.

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*Given in Tables 2.2, 2.3 and 2.4.

** See notes on Table 3.4

Table 3.4

(a) Annual separative work requirements (millions Kg SWU)

year	Power Plants			Other requirements	Total requirements
	USA	Western Europe*	Remainder of the free world	free world	free world
1975	9.4	3.3	1.6	2.7	17.0
1978	14.2	7.3	3.4	1.15	26.2
1980	18.2	10.0	4.7	1.95	34.9
1982	23.6	13.7	6.8	1.35	45.4
1985	33.3	19.6	10.9	1.5	65.3

(b) Cumulative separative work requirements (millions Kg SWU from 1 January 1971 to end of year)

year	Power Plants			Other requirements	Total requirements
	USA	Western Europe*	Remainder of the free world	free world	free world
1975	30.1	9.7	4.3	7.6	52
1978	67.2	27.8	12.5	13.2	121
1980	102.5	46.8	21.5	16.8	188
1982	146.5	72.2	34.8	19.8	273
1985	236	124	64	24	448

Hypotheses:

1. technology and performances of second-generation light-water reactors (commissioned in 1975);

2. PWR/BWR distribution 50/50 outside US and 66/33 in US;

3. plutonium recycling is carried out, except in the UK; this recycling is assumed to lead to reductions of 5% in 1975, 10% in 1980 and 15% in 1985 on the annual separative work requirements in the US; these reductions would be 2, 6 and 10% respectively in the other countries;

4. the other requirements represent USAEC evaluations of government requirements (research reactors, submarine reactors, etc.).

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*Requirements at the Community level are given in Table 2.6.

The enriched uranium requirements have been converted into separative work requirements, a 0.25%*tails assay being assumed during the entire period under consideration.

3.3.1.2 Growth rate and breakdown of requirements

Table 3.4 shows the estimated development of separative work requirements over the years and their breakdown among the United States, Western Europe and the rest of the free world.

Whatever the accuracy of the hypotheses and the forecasts made, it seems that world requirements will double from 35-38 million Kg SWU in 1980 to 65-74 million Kg SWU (according to the hypotheses adopted regarding plutonium recycling).

An examination of the breakdown of these requirements shows that Western Europe's demand will increase more rapidly during the next 15 years than that of the United States, of which it will equal about 35% in 1975, 55% in 1980 and nearly 60% in 1985**; as regards the rest of the "free" world, Japan's share will be appreciably more than half of the total requirements during the period under consideration.

Table 2.6 gives the Community's separative work requirements for an intermediate-type light-water reactor as defined in Section 2.

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*This is a plausible hypothesis; in the field of enrichment required for light-water reactor fuels, a variation of 0.25 - 0.30% or of 0.25 - 0.20% in the tails assay causes a reduction or an increase of about 10% in the separative work.

** These percentages correspond to the following quantities; (Table 3.4):
(a) with plutonium recycling: 3.3 million kg SW/year in 1975; 10 million kg SW/year in 1980; 19.6 million kg SW/year in 1985;
(b) without plutonium recycling: 3.4 million kg SW/year in 1975; 10.6 million kg SW/year in 1980; 21.5 million kg SW/year in 1985.

3.3.2 Uranium enrichment processes

Among the numerous processes which may be used to separate uranium isotopes, two are most commonly employed today:

- the gaseous diffusion process, on which are based the Western world's enrichment plants, situated in the United States, the UK and France; the American plants have up to now covered the entire civilian requirements of the Western world with the exception of the UK,
- the ultracentrifuging process, in which such technological strides have been made that Germany, the UK and the Netherlands have signed a tripartite agreement for its development and utilization. Pilot plants are being constructed or commissioned in the UK and the Netherlands.

Mention should also be made of:

- the supersonic nozzle process, mainly developed in Germany, which, however, still consumes too much electrical energy to be able to compete with the two other processes, despite certain advantages relating to ease of construction of the apparatus, and service life,
- the entirely new technique perfected by South African scientists, on which no details have yet been published, however.

3.3.2.1 Gaseous diffusion

A. Characteristics and future outlook

This enrichment process is essentially characterized by:

- a theoretical enrichment factor limited to 1.0043 per stage which in practice means a large number of stages in series;
- high unit capacity stages - several thousand SWU/year - so as to reduce specific investment costs and consequently the cost per SWU;

- the existence of a relatively high capacity threshold, of the order of several million SWU a year, in order to be economical;
- a high electricity consumption, of the order of 2500 kWh/SWU.

Development prospects mainly rely on the hope of improving the effectiveness of the diffusion barriers, the compressors and the aerodynamics of the diffusers. It seems unlikely that the consumption of electric power will be much reduced.

The specific investment for a 7.8 million SWU/year plant is estimated at about 100 u.a./SWU year. Maintenance and operating costs are low, and about half of the cost of the SWU is for electricity.

B. Existing facilities

US

The American government possesses an enrichment complex comprising three gaseous diffusion plants with a capacity totalling 17,200,000 kg SWU/year. These are run by Union Carbide Corporation (Oak Ridge and Paducah plants) and Goodyear Atomic Corporation (Portsmouth plant) on integrated lines; the Paducah plant supplies UF_6 enriched to about 1% which simultaneously feeds the two plants at Oak Ridge (enrichment limited to 4%) and at Portsmouth.

At the present time, this enrichment complex is operating below capacity (out of an installed electric capacity of 6000 MWe needed at full load, one-third was employed during the financial years 1970 and 1971) and produces roughly seven million kg SWU/year; talks are in progress with the electricity producers gradually to raise output to the above-mentioned value by the middle of the decade.

In addition to this gradual restoration of the nominal capacity initially installed, there are the following increases, obtained by modifying the existing plants:

(a) Under the cascade improvement programme (CIP):
a gradual increase in capacity as from 1976 which should raise the overall capacity to 22,235,000* Kg SWU/year during the financial year 1981; this expanded capacity will not require an increase in electric power, as it will be obtained by improving the plant operating characteristics (in particular, the use of new compressors, diffusers and barriers); authorizations credit totalling $61 million have so far been given (fiscal years 1971-72) on an estimated total of $500 million.

(b) Under the Cascade Power Uprating Program (CUP):
a new gradual increase in capacity as from 1978 which will bring the total capacity of the three existing plants to 26,787,000* Kg (SWU/year during the financial year 1981, by stepping up the UF_6 pressure level and the compressor power, thus raising the installed electric capacity from 6000 to 7400 MWe.

No official decision has yet been taken regarding the implementation of this programme.

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*For a 0.25% tails assay (see note at foot of page 33).

UK

The gaseous diffusion plant at Capenhurst, which was originally built for military purposes and was recommissioned for civilian purposes in 1967, reached in 1970 a capacity of about 400,000 SWU/year, i.e., a little more than the British requirements for that time, estimated at 300,000 SWU/year.

France

The gaseous diffusion plant at Pierrelatte was built for military purposes; geared towards the production of high-enriched uranium, it is not suitable in its present state to supply the low enrichments required for civil purposes.

C. Availability of knowhow

This process now seems ready for use in large-scale industrial projects on the basis of French knowhow or the American offers, made in July 1971, to share gaseous diffusion technology. The progress achieved by the French studies and tests would in fact enable work to be commenced on a plant with a capacity of 6-10 million SWU/year in 1973; production could begin in 1978, full capacity being reached in 1980.

As regards the US-made gaseous diffusion technology, the amount of technological knowhow and operating experience is considerable; simply from the technical maturity aspect, this technology could be available in a very short time. However, the conditions of the American offer might be such as to affect this availability and hence the completion dates of the plants to be built.

D. Industrial situation in the Community

Various French companies have acquired experience in the field of gaseous diffusion technology, particularly under the aegis of the CEA in the construction of the Pierrelatte plant and pilot plants for the civil plant project.

In the field of design, at the beginning of 1971 the CEA carried out a siting study on a civil plant in collaboration with the companies of Technip and Bechtel.

Another design study will be done at the European level under a Study Association, for which the agreement was signed on 25 February 1972. This Association comprises:

- le Syndicat belge de séparation isotopique
- le Commissariat à l'Energie Atomique
- la Studiengesellschaft für Uranisotopentrennverfahren
- le Comitato Nazionale per l'Energia Nucleare
- l'Agip Nucleare
- Ultra-centrifuge Nederland nv
- British Nuclear Fuels Limited.

Its aim is to study the economic prospects accompanying the setting-up in Europe of a gaseous diffusion isotope separation plant which would be competitive on a world scale.

The work will deal with the technologies of gaseous diffusion for which the necessary data will be available, whatever their origin.

3.3.2.2 Ultracentrifugation

A. Characteristics and future outlook

This process, developed under the tripartite agreement concluded between Germany, the United Kingdom and the Netherlands, is characterized by:

- A high enrichment factor per stage, of up to almost one hundred times that of gaseous diffusion; this suggests, for 3-5% enrichment, only 10-15 stages in series, compared with the 1200-1500 corresponding stages in gaseous diffusion.

- An enrichment capacity per machine limited to a few SWU/year, which suggests the need for the simultaneous installation of a very large number of centrifuges in parallel - several million for a capacity of 10,000,000 SWU/year.

- A sufficiently low capacity threshold of several hundred thousand SWU/year in order to benefit from the savings possible with series production of the apparatus and to obtain profitable investment costs.

- A low specific electricity consumption, of the order of 250 kWh/SWU, or about one-tenth of that for gaseous diffusion.

The development prospects, according to the pilot plants being built at Capenhurst and Almelo, lie in an increase in the unit separation capacity of the apparatus and an improved estimation of their service life. A five-year lifetime is considered as the minimum to make this process attractive.

The specific capital investment, for large capacities requiring 2-3 million machines, is estimated at about 150 u.a./SWU/year.

This process could, however, lead to the most attractive separative work costs owing to its low electricity consumption, but these hopes will have to be confirmed at the industrial level when the pilot plants are in operation in 1972-73 and also building a 300,000 kg SWU/year prototype plant, to be commissioned in 1976.

B. Existing facilities

Under the cooperation agreement concluded on 4 March 1970 between Germany, Great Britain and the Netherlands, the following prototype plants are being constructed:

(a) the Netherlands plant, located at Almelo, with a capacity of 25,000 kg SWU/year; commissioning is planned for 1972;

(b) the German plant, located at Almelo, with a capacity of 25,000 kg SWU/year; it is being constructed in two stages so as to enable various types of centrifuge and cascade arrangements to be tested in accordance with the latest state of the art; it is due to go into service at the beginning of 1973;

(c) The United Kingdom plant, sited at Capenhurst, with a capacity of 15,000 kg SWU/year; it forms the first part of a group totalling 40,000 kg SWU/year, acceptance testing should take place at the beginning of 1973.

Each of the demonstration plants must comprise several thousand machines. They will provide experience regarding centrifuge interaction, maintenance and replacement operations, etc.

Workshops are already in existence for the manufacture and installation of about 10,000 centrifuges a year.

C. Availability of technology

Under the above-mentioned cooperation agreement, it is planned to reach a total capacity of 350,000 kg SWU/year in 1975. If the experience acquired with the prototype plants now being built confirms the hopes of the owners of this process, this capacity could be increased by several hundred thousand kg SWU/year by 1976, and by a million kg in each succeeding year.

D. Industrial situation in the Community

The following two companies were created under the tripartite agreement on ultracentrifugation:

- Urenco Ltd at Marlow, England, which is to manage the enrichment plants; Urenco is owned equally by UCN (Ultra-Centrifuge Nederland NV), Uranit (Uran-Isotopentrennungs-Gesselschaft mbH) and BNFL (British Nuclear Fuels Ltd).

- Centec GmbH at Bensberg, Germany, which is to handle the design and construction of enrichment plants, particularly on behalf of Urenco; Centec is owned equally by UCN, GNV (Gesellschaft für Nukleare Verfahrenstechnik mbH) and BNFL.

3.3.3 Evaluation of American and other contributions to the coverage of requirements

The covering of world requirements, as defined and evaluated in Section 3.3.1, depends at present on the existing means of production, represented by the above-mentioned plants, i.e., essentially on the existing American enrichment capability. During this period, due to end in 1975, American production will exceed world requirements, thus enabling the US government to increase its stock of enriched uranium (preproduction).

The American contribution to the coverage of non-US requirements will continue to predominate and will be made under the cooperation agreements which link certain countries and the European Atomic Energy Community to the United States government, and within the limit of the "authorized qunatities" stipulated in the agreements.

The USSR contribution to the coverage of the free world's requirements would probably be limited to specific deliveries negotiated case by case with the parties concerned.

The UK's means of production will enable it to meet its own requirements until 1976, no contributions from outside the United Kingdom being planned.

Contributions from various existing or potential producers to the coverage of world needs after 1975 will depend on each producer's assessment of future market trends and on the capital investment policy which he decided to pursue in a national or international context.

As regards the American contribution, the following factors will determine its size:

- timescale for exhaustive of preproduction stock;
- annual appropriations earmarked for implementation of the CIP (submitted to Congress for approval);
- decision to implement the CUP;
- and, of course, the construction of new plants.

The contribution from the rest of the world, excluding the USSR, will be limited until 1977-78 to the British national production mentioned above and to the annual output of the plants to be built by the signatories to the agreement for cooperation on uranium enrichment (Germany, United Kingdom, Netherlands), which should amount to 350,000 kg SWU by 1975. The time needed to build any type of high-capacity plant, for which the decision to build could not be taken before 1973, is such that the first kilogrammes of enriched uranium could not be delivered before the period given above.

The potential USSR contribution is not known.

On the assumption that the CIP and CUP programmes in the USA and the ultracentrifugation demonstration programme in Europe are carried out, then cumulative separative work output of the free world capacities would be:

End of 1975: about 74 million kg SWU in the USA and 2.4 million kg SWU in Western Europe, i.e., about 76 million kg SWU for the free world.

End of 1980: about 183 million kg SWU in the USA and 6 million kg SWU in Western Europe, i.e., about 189 million kg SWU for the free world.

3.3.4 Comparison of requirements and availabilities

The cumulative separative work requirements for different years have been compared (Fig. 3), with the cumulative world availabilities (USA and UK means of production, the tripartite agreement and American preproduction stocks).

This figure shows that world requirements will no longer be covered after the beginning of 1981 if plutonium is recycled in light-water reactors, and after the beginning of 1980 if it is not.

These estimates fit well with estimates from other sources.

It can thus be seen that world requirements will no longer be covered after about 1980.

This situation must not mean that, in the Community, supplies to power plants fuelled on enriched uranium are cut off around 1980. It thus seems necessary for the Community to have its own enrichment capacity. This should be decided upon in good time in order to ensure that the operators of plants to be commissioned after 1974-75 are guaranteed dependable supplies of enriched uranium.

These conditions appear to be essential if the aims of the present programme are to be achieved.

As regards the development of the Community's industrial and commercial potential in the nuclear sector, a lack of enrichment capacity in the Community would constitute a serious handicap both to the fuel cycle industry and to the nuclear reactor construction industry.

The operation of an enrichment capability would enable all the stages of the enriched uranium fuel cycle to be performed, integration of the various stages thus improving the management.

Moreover, the availability of complete fuel cycle services in the Community would mean that the market could expand in the necessary atmosphere of security, and its industry could assert its presence on foreign markets, where guarantees which could be given concerning the fuel cycle are a major asset in the competition between reactor manufacturers.

3.4 Manufacture of fuel elements

In the case of light-water reactors, this stage of the cycle comprises the processes which, starting with the enriched uranium oxide powder obtained by the reconversion of UF_6, end in the delivery of fuel elements ready for use at the power plant.

Manufacture is simpler in the case of gas/graphite natural uranium reactors, for which there is already excess capacity in France (SICN and CERCA) and in the UK, now that this reactor type has been abandoned.

We therefore do not propose to deal with this case, as its share will decrease over the period covered by the programme and no new capital investment will be required.

Fuel for high-temperature reactors is a more complex problem. This consists of particles, coated with uranium oxide or even carbide, which are then inserted into spheres or prismatic graphite elements. Fabrication, which is still in the experimental stage and limited to the development of the necessary technology, is carried out by Nukem. For the next few years, it will only meet the needs of the prototype reactor due to be built in Germany.

For fast reactors, which use a mixed UO_2/PuO_2 fuel in stainless steel cladding, the situation is similar to that of the HTR in that the requirements to be met are those of the Phénix and SNR prototypes reactors, pending the subsequent stage of the 600-1000 MWe prototype(s). The pilot plants of the CEA at Cadarache, Alkem at Wolfgang and Belgonucléaire at Dessel are the manufacturers of the corresponding fuel elements.

During the period covered by the Illustrative Programme the only industrial-scale fabrication is that of light-water reactor fuel elements.

The only case that will therefore be examined is that of the fuel elements for light-water reactors.

The fuel is enriched uranium oxide, in the form of sintered pellets of a density of the order of 94%, inserted into thin tubes about 4 m long made from Zircaloy 2 or 4, according to whether it is a BWR or PWR. These pins are seated at each end with welded plugs and are grouped in assemblies of 49 rods in the case of BWRs and 180, 205 or 235 rods for PWRs. The fabrication process is sufficiently well-known and standardized to make further description unnecessary.

3.4.1 Cladding tubes - Production capacity

Cladding tube requirements break down as follows:

Year	BWR: dia_e. = 14.3 mm; e = 0.81mm				PWR: dia_e. = 10.72 mm; e = 0.61 mm			
	No. of rods (thousands)		Length Km	Tonnage 1 Km = 0.250 t	No. of rods (thousands)		Length Km	Tonnage 1 Km = 0.250 t
	1st core	Reload			1st core	Reload		
1975	50	24	300	75	58	40	390	55
1976	89	29	470	120	100	50	600	85
1977	92	44	550	140	102	73	700	100
1978	124	60	730	180	137	99	950	130
1979	137	86	900	225	152	140	1180	165
1980	142	113	1020	255	154	185	1360	190
1981	153	142	1180	295	167	229	1590	220
1982	171	175	1380	345	189	282	1880	260
1983	193	205	1590	395	213	323	2140	300
1984	213	259	1890	470	236	413	2600	360
1985	252	312	2250	565	283	494	3200	440

The tube manufacturers in the Community are:

- UDM Zirconium GmbH, of the Vereingte Deutsche Metallwerke AG
  production capacity: 300 km/year
- CEFILAC, of the Vallourec group: capacity: 150 km/year
- Mannesmann Raehrenwerke GmbH: capacity: 150 Km/year
- Sandvik Universal Tube (SUT): Capacity: 300 Km/year.

The Community manufacturers' capacity therefore appears to be amply sufficient to cover requirements until 1975. It is, however, advisable to point out that the same plants are being used for the manufacture of stainless steel cladding tubes for prototype fast reactors and that the Community manufacturers are also aiming at the export markets.

3.4.2 LWR fuel elements - Production capacity

The requirements in enriched uranium for first cores and reloads are given in Table 2.7 for intermediate-type LWRs.

With the exception of CICAF (Compagnie Industrielle de Combustibles Atomiques Frittés) at Boltene, who only manufacture sintered UO_2 pellets with a capacity of around 100 t/year, all the other manufacturers are equipped to supply complete fuel elements. They are listed in the table below:

Country	Firm	Site	Production capacity t/year	
			present	planned for 1972
Germany	RBG	Wolfgang	100	300
	KRT	Grosswelzheim	100	185
France	CERCA	Romans	100	100
Italy	Fabbricazioni Nucleari	Busalla	120	120
	COREN	Saluggia	50	50
Belgium	MMN	Dessel	80	200

Plutonium bearing fuels used in Pu recycling in light-water reactors are dealt with in Section 4.

3.4.3 Capital investment

The capital investment in fuel-element manufacturing plants is small compared with other stages of the cycle. An initial plant with a capacity of 100-150 t/year only requires an investment of 5-6 million u.a. and extensions are possible at a lower cost merely by duplicating certain items. Moreover construction time is short and the capital investment can be decided upon after the order has been placed for the nuclear power plants themselves which are to be supplied with fuel by the plant.

3.4.4 Manufacturing costs

This aspect is dealt with in detail in the document "Centrales nucléaires à eau légers", (LWR power plants"). Hence we shall do no more than mention the cost trend, which has taken a pronounced downward turn owing to the development of the market and the standardization of the fabrication process, which will enable the present very substantial proportion spent in controls to be reduced.

The cost of fuel for an intermediate-type reactor may be estimated at approximately 110 u.a./kg contained uranium in 1975, 85 u.a. in 1980 and 70 u.a. in 1985.

3.4.5 Structure of the market

With the aid of a 10% customs duty on complete fuel elements, which thus provides the manufacturers with a protective cushion of 30%, the Community industry has an amply sufficient capacity to satisfy the market.

With the exception of the Netherlands, which closed down its fabrication plant several years ago but plans to enter the race again, there is a distinct partitioning of the market, each country distributing or planning the capacity to cover its own national market. Owing to the existence of the two types of BWR and PWR light-water reactors (except in Belgium, which, at least for a few years, opted for PWRs only), this situation is leading to an increase in the number of fabrication plants, so that economies of scale are not possible and the desirable regrouping of manufacturers is prevented.

In addition, the market is characterized by a vertical integration of the cycle services, and particularly of fabrication, within power plant manufacturing consortia.

3.4.6 Companies engaged in the manufacture of fuel elements

The companies manufacturing fuel elements are:

In Germany: KRT, RBG, Nukem, Alkem

France: CERCA, SICN

Italy: COREN, Fabbricazioni Nucleari

Belgium: MMN, Belgonucléaire.

- KRT Kernreaktorteile. Company founded in 1966 by AEG (55%) and the General Electric Co. (45%). Located at Grosswelzheim, the plant manufactures BWR fuel elements which are marketed by the parent companies.

- RBG: Reaktor Brennelemente GmbH. Company founded in 1969 by the Siemens (60%) and Nukem (40%) Association. Located at Wolfgang, the company mainly handles fuel elements for research and high-temperature reactors.

- Alkem: founded by AEG, Robert Bosch, Nukem and Siemens. Located at Wolfgang, after transfer of the laboratories and the pilot plant from leopoldshafen, the plant specializes in plutonium-bearing fuels for light-water and fast reactors.

- CERCA: Compagnie pour l'Etude et la Réalisation de Combustibles Atomiques. Founded in 1962 by St Gobain Techniques Nouvelles (24%), SFAC (Schneider Group) (26%), Pechiney (25%) and Sylior USA (25%). Located at Tomans, the company manufactures elements for gas/graphite reactors (capacity 800 t/year), light-water reactors with CICAF sintered pellets and research reactors.

- SICN: Societé Industrielle de Combustibles Nucléaires. Established by Trefimétaux (10%), the Société Lyonnaise des Eaux et de l'Eclairage (20%) Sté Alsacienne de Participations Industrielles (35%), Lille Bonnière Colombus (20%) and Ugine Kuhlmann (15%). Located at Annecy, the company manufacturers elements for gas/graphite reactors. (capacity 1,000 t/year) and is participating in the manufacture of fuel for the Phênix prototype.

- COREN: Combustibili per Reattori Nucleari SpA. Established in 1967 by Fiat (25%), Westinghouse (50%) and Ernesto Breda of the EFIM Group (25%). Located at Saluggia, the company specializes in BWR elements.

- MMN: Metallurgie et Mécanique Nucléaires. Founded in 1958. The present participation in the capital is as follows: Union Minière 34%, Metallurgie Hoboken 20%, Fabrique Nationale d'Armes de Guerre FN 20%, Ski Générale de Belgique 9%, ACEC 6%, Rio Tinto Zinc 10%, Belgonucléaire 1%. Located at Dessel, the plant produces PWR elements and fuel for research reactors.

- Belgonucléaire: Composed of a number of Belgian firms. Located at Mol and Dessel, the company produces plutonium-bearing fuels.

3.5 The transportation of irradiated fuel elements

The cost of transporting irradiated fuel from the nuclear power plant to the reprocessing plants represents a small part of the total fuel cycle cost. It amounts, according to various sources, to 1.2% of the total cost. Although this percentage seems extremely small, the expenditure on transport in absolute figures is quite considerable.

According to Table 2.8, the quantities of irradiated uranium to be transported to the reprocessing plants will total, for light-water reactors alone:

110 t in 1975
720 t in 1980
1940 t in 1985

From 1975 to 1985, a total of about 10,000 t of irradiated uranium will have to be transported in the Community countries. On the basis of an average cost of 5 u.a./kg U, these consignments represent a total turnover of about 50 million u.a. for the period 1975-85.

The transportation casks now used can generally hold 2 - 2.5 tons of uranium. If a cask can complete about 20 trips a year, 30-35 casks would be enough to meet transportation requirements in the Community in 1985.

At present there are only two large groups of transport companies in Western Europe, one in the Community and one in the UK.

As regards irradiated LWR fuel elements, the number of transport operations carried out up to now, and their volume, are such that there is no real transport market.

For this reason, the transportation prices of 5-12 or even 20 u.a./kg U in force up to now are not representative. Various aspects have to be taken into account in order to achieve more economic transportation. The determining factor governing profitability is the frequency with which the casks are used. Transportation costs diminish as the number of journeys per cask increases and as the weight of the fuel transported increases in relation to the weight of the cask, fuel element dimensions, safety aspects (criticality control) and irradiation aspects, cooling.

In view of the likely growth of the irradiated fuels transportation market, it seems justifiable to develop special medium-sized casks for the transportation of LWR fuel elements. Towards the end of the period covered by the Illustrative Programme, the use of large casks, of the order of 100 t, would appear to be economically viable, provided that not only the reprocessing plants but also the nuclear power plants are linked up to the railway networks.

In order to ensure the most economic utilization of the casks, coordination is essential between the plant operators, the transport companies and the reprocessing plant operators.

The technical problems which are now apparent, and whose satisfactory solution will also affect transport costs, concern the cooling of large-capacity casks and, above all, the neutron radiation from the high-burnup fuel elements planned for use in large light-water reactors.

Because the transportation of fuel elements comprises a series of operations (packaging, transportation proper, administrative formalities, insurance, etc.), improvements to the laws and regulations can also help to lower the cost of transportation, e.g., by drawing up regulations which take into account both safety considerations and economic aspects.

3.6 Reprocessing of UO_2 fuels

Reprocessing plants

There are several plants for reprocessing oxide-type fuels in the Community, in the form of a small-capacity plant (Eurochemic) and various pilot plants (WAK-Eurex) designed as test beds before industrial installations are built.

Eurochemic, Mol, Belgium

Designed within the framework of the OECD, the Eurochemic plant has been in operation since 1966. Its present capacity is of the order of 100 t/year of low-enriched oxide fuel. It is also able to reprocess metallic-type fuel elements (MGR), as well as all the MTR fuels discharged from research reactors in the Community. The plant is due to be closed down at the end of 1974 by decision of the shareholders.

WAK, Karlsruhe, Germany

This plant has been in operation since the end of 1971. It is a 40-50 t/year plant designed solely for low-enriched uranium-oxide-base fuels; it will be used for reprocessing fuel elements from various German light-water reactors and from the FR-2 up to 1974, after which it will become a pilot plant for fast reactor fuels.

Eurex, Saluggia, Italy

This plant is designed as an industrial pilot plant for reprocessing MTR-type high-enriched uranium elements. It is also able to reprocess power reactor fuels (natural uranium metal and low-enriched uranium oxide), for which it has a maximum capacity of 25 t/year. The plant is the subject of a ten-year agreement between the Community and the CNEN covering both operation of the plant and the implementation of a research programme covering the development of aqueous reprocessing methods.

Cap de la Hague, France

Designed to reprocess fuels from the various gas/graphite/natural uranium power reactors with a 1200 t/year capacity, it will be modified to reprocess enriched uranium oxide fuels. Its capacity for this type of oxide fuel element will be 400 t/year in 1975 and 800 t/year a few years later. The reprocessing of gas/graphite reactor fuels will at all events be guaranteed.

Windscale, England

Mention should be made of the existence of this plant which, since 1970, has had a reprocessing capacity of 300 t/year for oxide fuels and has exerted a major influence on the Community market. In the second half of the seventies this plant will have a total capacity of 800 t/year for oxide-type fuels.

Table 3.5
In Europe
Existing reprocessing capacity and planned extensions for oxide fuels (t/year)

	1970	1975	1980
Eurochemic (closing end 1974)	100	-	-
WAK	-	50	-
Eurex	-	25	-
Cap de la Hague	-	400	800
Community total (rounded off)	100	500	800
Windscale	300	300	800
Grand total	400	800	1600

Table 3.6 lists, by way of comparison, the existing or planned American plants, which also have a large excess capacity.

Table 3.6

American capacity (t/year)

Firms	Capacity	Date of Commissioning
Nuclear Fuel Services	300 t/year 600 t/year	1966 1973
General Electric	300 t/year	1972
Allied Chemical Nuclear Products	1500 t/year	1974

3.6.2 Trend of reprocessing capacity requirements and corresponding capital investment

Table 3.7 gives estimates for the quantity of irradiated fuels annually discharged by Community power reactors, according to the present programme.

Table 3.7

Fuel elements at the reprocessing plant level (t/year)

Year	LWR	GGR
1975	110	540
1976	200	540
1977	250	540
1978	370	540
1979	500	540
1980	720	540
1981	940	540
1982	1180	540
1983	1450	540
1984	1700	540
1985	1940	540

Reprocessing capacity in the Community and the increase planned (cf Table 3.5) show that from a theoretical point of view the Community reprocessing requirements are covered until 1981 as regards LWR fuels.

From a practical point of view, it must be remembered that the large plants at Cap de la Hague and Windscale will also be used to cope with part of the market outside the Community and the UK. New plants would therefore be justified shortly before the end of the decade.

The forecasts for Europe as a whole prepared by a Foratom working party indicate that for 1980 the quantity of oxide fuel to be reprocessed will exceed the reprocessing capacity planned for this date by 400 t. This study is based, for the Community, on larger reprocessing estimates than those used in the present programme. According to this study, an additional capacity of about 2000 tons should become available between 1980 and 1985. A major factor governing this choice of the size of the plant will be the desire for a high load factor in the first few years of operation. This will probably mean a plant capacity of between 2 and 5 tons per day.

Capital investments needed by the reprocessing plants may be estimated in 1970 dollars at:

- 45 million u.a. for a plant with a capacity of 1 t/d (300 t/year)
- 85 million u.a. for a plant with a capacity of 5 t/d (1500 t/year).

3.6.3 Trend of reprocessing costs taking into account the size/cost ratio and load factor

Present commercial situation

The commercial reprocessing market is seriously upset by the existence of a large excess capacity. In Europe, the UKAEA* has up to now pursued a commercial policy based on abnormally low prices (18-20 u.a./kg U) in the hope of conquering numerous markets. In these conditions, the companies situated in the Community (Eurochemic and WAK) have been forced to adapt to this policy as a result of which they have been running at a loss.

True cost of reprocessing

On the basis of the Foratom study mentioned above, it appears that the price which ought to be charged by companies operating without government aid for reprocessing facilities is far higher than the present price.

The cost of reprocessing is a function of the size of the plant and the load factor. The minimum size considered in the report is a capacity of 1 t/d. The table below shows the results of this study.

Plant capacity	1 t/day	5 t/day			
Load (%)	100	20	50	80	100
Cost of reprocessing ($/kg U)	55	93	40	26	22

*Actually the UKAEA production group, renamed the British Nuclear Fuel Limited after 1 April 1971.

These data show that the load factor has an appreciable effect on the cost and that the optimum size of a plant must be greater than 1 ton/day. It should be noted that large plants commissioned with an eye to the future market will only have a relatively low load factor at the time they go into operation. According to the "Foratom Report 1970", the mean cost for a 5 ton/day plant which increases its load factor from 0-100% in five years would be 35 u.a./kg U, whereas the optimum cost at 100% load is 22 u.a./kg U.

Needless to say, the commissioning of high-capacity plants must be carefully planned and coordinated at the European level in order to ensure that they operate at a profit.

In the light of this situation, British Nuclear Fuel Limited, the Commissariat Francais à l'Energie Atomique and the German KEWA* company set up on 12 October 1971 a services company, United Reprocessors GmbH, to market the services of the reprocessing facilities owned by its shareholders and to plan their new capital investment.

The creation of this company was the subject of a statement to the Commission of the European Communities under Council Regulation 17 relating to rules of competition.

3.6.4 Future trend of quantities of radioactive waste

The nuclear fuel cycle produces different categories of radioactive waste. From the radioactivity aspect, consideration will be given to three categories of waste produced in fuel reprocessing:

- fission products, which represent most of the activity and are in liquid form;
- waste due to fuel stripping;
- concentrates obtained by concentrating low- or medium-activity liquids.

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*KEWA (Kernbrennstoff - Wiederaufarbeitung GmbH) made up of Farbenfabriken Bayer AG -Farbwerke Hoechst AG, Gelsenberg AG and NUKEM GmbH.

Table 3.8 shows the quantities of radioactive waste in these three categories.

The activities of the fission products contained in the irradiated fuels were evaluated as after 150 days of fuel cooling time.

In the evaluation of the stripping waste it was assumed that the fuel is stripped mechanically, a technique used in most reprocessing plants.

In this assessment no account was taken of the waste produced by plant decontamination or the low-activity waste (liquid and solid) produced in the plant's analytical laboratories.

Table 3.8

Year	Fission products Act. (Ci)	Fission products Liq.vol.(m^3)	Compact stripping waste Vol. (m^3)	Concentrates Vol. (m^3)
1975	$6.84 \cdot 10^8$	110	11	204
1976	$12.4 \cdot 10^8$	200	20	300
1977	$15.6 \cdot 10^8$	250	25	375
1978	$23.0 \cdot 10^8$	370	37	551
1979	$31.1 \cdot 10^8$	500	50	746
1980	$44.8 \cdot 10^8$	720	72	1080
1981	$58.5 \cdot 10^8$	940	94	1410
1982	$73.4 \cdot 10^8$	1180	118	1770
1983	$90.2 \cdot 10^8$	1450	145	2175
1984	$105.7 \cdot 10^8$	1700	170	2550
1985	$120.7 \cdot 10^8$	1940	194	2915

3.6.5 Technical problems, cost of conversion and storage of radwaste

The present practice of storing fission products and concentrates in liquid form in steel tanks cannot go on indefinitely, owing to the need to replace the storage tanks after a certain number of years and in view of the safety problems posed by the prolonged storage of such large amounts of activity in liquid form. The present trend is to limit storage in this form to a few years, after which the substances are stored permanently in solid form. Another advantage of solidification is that it reduces the volume to be stored by a factor of about 10.

Several processes for solidifying fission product solutions have been developed in the United States. In the Community, a vitrification process is being tested at the industrial pilot stage in France. Another process, calcination-vitrification, is being studied in West Germany. The storage of high-activity waste on-site at the reprocessing plant is generally estimated to cost about 5 u.a. per kg reprocessed uranium; this is included in the reprocessing cost. Insolubilization and permanent storage in the solid state could entail an incremental cost of 4-5 u.a. per kg uranium.

4. Plutonium recycling in light water reactors

Introduction

The basic technological problems associated with the recycling of plutonium in thermal light water reactors have been solved by now; as the use of this fissile material in light water reactors is not impeded by any technological barrier, it will be regulated by the laws of market economics, under the international treaties governing the use of fissionable materials.

Plutonium is a fissile material which can be transported easily; in order to assess the likelihood of its being recycled in the thermal reactors, one should study the world supply and demand[1] and then examine the evolution of reprocessing capacity, plutonium fuel-element manufacturing capacity and the advent of the fast reactors all in conjunction with one another.

4.1 Fissile plutonium supply

The world supply of fissile plutonium is obtained from the fissile material generated by plutonium-producing reactors and power reactors. By the end of 1966, about 1,200 kg of fissile plutonium had already been produced by nuclear power plants; however, this was only quite a small fraction of the total quantity produced in the world at that time. The plutonium-producing reactors at Hanford and Savannah River in the United States, Calder Hall and Chapel Cross in the United Kingdom, and Marcoule in France, had generated far larger amounts (probably over 25 tons), some of which could be assigned to civil uses if necessary[2]. As the plutonium output from these plants is now decreasing, it has been disregarded in this study. On the other hand, output from the power plants has increased: between 1967 and 1970 they produced roughly 7,500 kg of fissile plutonium.

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[1] Apart from USSR and China.

[2] The Pu for the Sneak and Masurca plants was supplied by Hanford.

The plutonium production from the USA nuclear power plants and the rest of the free world in 1971-85 is shown in Table 4.1; Table 2.9 shows the Community output.

Plutonium output from enriched uranium reactors should be considered separately from that of natural uranium reactors, since the production cost if very different in the two cases.

Technically speaking, the plutonium in fuel elements irradiated in natural uranium reactors can be recovered just as well as it can from elements irradiated in enriched uranium reactors; but since it alone has to bear the cost of recovery (the residual U^{235} concentration being too low to have any market value), its production cost is high. For example, this cost is about \$12/g (\$6/g) when the cost of reprocessing, transporting spent fuel from a graphite/gas reactor and final disposal of the waste runs to about \$30/kg (\$15/kg).

In contrast, the cost of the plutonium[1] in spent fuel from enriched uranium (light water) reactors is low; this is because of the high residual concentration (0.6-0.8%) of U^{235} in these fuels, which consequently has an appreciable residual value. Under present market conditions (cost of reprocessing, transport of spent fuel and final disposal of waste about \$30/kg, price of 0.8%-enriched uranium \$20/kg), the cost of this plutonium is of the order of a few (1-2) u.a. per gram.

[1]Where plutonium is not specified as being fissile, total Pu is to be understood.

Table 4.1 Fissile plutonium recovered from USA and free world power reactors

Year	Fissile Pu recovered (tons)			
	USA		Rest of free world	
	Annual	Cumulative	Annual	Cumulative
1971	0.4	0.4	3.5	3.5
1972	0.5	0.9	3.8	7.3
1973	0.9	1.8	4.0	11.3
1974	2.1	3.9	4.5	15.8
1975	4.0	7.9	5.5	21.3
1976	6.4	14.3	6.7	28.0
1977	8.9	23.2	8.4	36.4
1978	10.8	34.0	10.6	47.0
1979	12.4	46.4	12.9	59.9
1980	15.6	62.0	16.2	76.1
1981	19.3	81.3	19.8	95.9
1982	22.5	103.8	23.8	119.7
1983	27.4	131.2	27.6	147.3
1984	32.3	163.5	32.2	179.5
1985	37.1	200.6	37.4	216.9

Ref: Wash-1139 Forecast of Growth of Nuclear Power - January 1971

Under these conditions, the plutonium produced in natural uranium reactors will only supply the plutonium market during shortages, such as now, when the market value (selling price) of Pu is very high, or if the reprocessing plants are operating at marginal cost ($15/kg).

4.2 Privileged requirements

Plutonium will primarily be used for research and development programmes and to fuel prototype breeder reactors and the master model breeders.

4.2.1 Requirements for R&D programmes

This category comprises the requirements for:

(a) plutonium fuel studies and experimental irradiations of fuel batches or partial or even complete reactor cores;
(b) critical assemblies and experimental reactors;
(c) physics and metallurgy studies using plutonium, fabrication of sources, etc.

The figures taken as the basis (in tons of fissile plutonium) are given in Table 4.2. The figures for 1967-70 and 1971-75 were estimated from the published details of the programmes carried out in the various countries. For the period 1976-80, since this is not a very destructive category of user, it was arbitrarily assumed that requirements would be the same as for 1971-75.

Table 4.2. Plutonium requirements for R&D programmes

Country	1967-70	1971-75	1976-80
USA	5	3	3
UK	1	0.5	0.5
Germany, France, Others	1.5	0.75	0.75
Total	7.5	4.25	4.25

Thus in 1980 over 15 t of fissile plutonium would be immoblized for R&D programmes - a substantial figure.

4.2.2 Fuelling of prototypes and master models

Table 4.3 lists the prototypes and master models of fast reactors that will be in service in the Community of the Six during 1970-85.

Table 4.3. Prototype and master model fast reactors

Name	Location	Owner	Commissioning date	Capacity (MWe)
Phenix	Marcoule	CEA/EDF	73	250
SNR	Kalkar	Germany/Benelux	78	300
?	France	EDF/RWE/ENEL	79	1000
?	Germany	"	82/83	1000
-	-	-	84/85	1000

To calculate the amount of fissile plutonium tied up in the prototypes, it was assumed that 3 kg fissile plutonium per MWe will be needed for the core of each reactor[1] and that 8 kg fissile Pu per MWe will have to be gradually added to this to allow for immobilization throughout the fuel cycle[2].

To calculate the amount of fissile Pu tied up in the master models, it was assumed that 2 kg fissile Pu per MWe will be needed for the core of each reactor[3] and that 2 kg fissile Pu per MWe will gradually have to be added to this to allow for immobilization throughout the fuel cycle[4].

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[1] Immobilized 2-3 years before the plant starts operation.

[2] Immobilized during the first year's operation of the plant.

[3] Immobilized 2-3 years before the plant starts operation.

[4] Immobilized during the first year's operation of the plant.

Table 4.4 shows the total privileged immobilizations in the Community.

Table 4.4. Total investment in fissile plutonium in the Community (tons)

Year	71	72	73	74	75	76	77	78	79	80	81	82	83
Annual	0.5	0.5	2.2	0.2	0.2	0.6	1.6	3.6	2.2	1.2	1.2	3.2	1.2
Cumulative	0.5	1.0	3.2	3.4	3.6	4.2	5.8	9.4	11.6	12.8	14.0	17.2	18.4

4.3 Comparison of supply and demand (privileged requirements)[1]

The world supply of and demand for fissile plutonium is compared in the following table.

Comparison between supply of and demand for fissile plutonium (kg of fissile Pu)		
Period	Supply	Demand (privileged requirements)
1967-70	7,500	8,000
1971-75	30,000	12,000
1976-80	110,000	20,000

This table shows three distinct periods:

(a) Before 1970, the fissile plutonium supply is low compared with the demand, which comes mainly from the R&D programmes. Plutonium is a rare substance and its price is high.

(b) After 1971 and up till about 1975, supplies will exceed demand. From 1971 to 1975 the cumulative output of fissile material from the USA power plants alone will be eight tons, while the world demand for the same period will be greater than this figure.

[1]Apart from USSR, China and East European countries.

(c) The most significant gaps between supply and demand will not appear until later, however, in the period 1976-80, when the world supply will be some 110 tons whereas the demand for privileged requirements will only amount to about 20 tons.

Thus the world supply of "cheap" plutonium will be in excess of privileged requirements from 1973-75 onwards.

In order to see what Community imports or exports of plutonium will be, one must compare the Pu supply and demand inside the Community. This is shown in the following table.

Comparison between the Community supply of and demand for fissile plutonium (kg of fissile plutonium)

Period	Pu from reactors		Demand (privileged requirements)
	LWR	graphite/gas	
1971-75	1,700	3,700	3,500
1976-80	10,000	4,400	8,000
1981-85	35,000	4,400	10,000

On the internal market of the European Community the supply of plutonium from light-water reactors will be lower than the privileged demand during the first years of the decade 1971-80. This is one of the reasons why, even if it is not very economical, the plutonium content will be recovered from the graphite/gas reactor fuel elements. During this period the Community will probably still import small amounts of plutonium.

The supply will not equal the privileged demand until about 1975, but will nevertheless be in excess of it around 1980.

. . . / . . .

The cumulative excess over the period 1976-80 will be of the order of seven tons of fissile plutonium, if it is assumed that the market is supplied with plutonium from both enriched and natural uranium reactors; this excess will only amount to two tons if it should prove more worthwhile to give up, at any rate temporarily, the reprocessing of natural uranium fuels.

4.4 Use of excess plutonium on the world market after 1973-75

On the world market, where fissile plutonium will be in excess from the years 1973-75 onwards, various possibilities will be open to plutonium producers:

1. If the production cost of plutonium is higher than its use value in light water reactors (generally taken to be \$7-8/g Pu, it will probably be more profitable to stockpile the non-reprocessed fuel elements (possible case of plutonium to be extracted from natural uranium elements) and wait until the technical and economic conditions have improved.

2. If the cost of plutonium is lower than its use value in light water reactors (general case of plutonium extracted from LWR fuels), this plutonium could be either recycled in the thermal reactors operating at this time, or stored for later use in fast reactors (use value generally taken to be \$14-16/g Pu).

This last hypothesis is not absurd, because if it is hoped to obtain a price of \$15/g plutonium on the potential fast reactor market in 1985, and if the cost of money is taken to be 8% a year, the present-worth value of this gram of plutonium in 1975 is at most \$7/g[1], which fits well with the use values quoted above.

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[1] The doubling time for the value of plutonium is a good deal less than 10 years, if storage costs and a money cost of over 8% a year are taken into consideration.

It must nevertheless be remarked that:

(a) electricity producers will be inclined to market their stock of plutonium immediately rather than wait for some ten years;

(b) the industry will wish to tackle the problem of plutonium storage;

(c) if all the available plutonium were stockpiled, there would be some 150-200 t of fissile plutonium available by 1985, when the fast reactors will probably appear. Even now it can be predicted that the potential fast reactor market is not big enough to absorb all this plutonium, for this stockpile would be sufficient for a fast reactor capacity of roughly 45,000 MWe to be installed in the world at this time.

All the evidence indicates that some of the plutonium will be recycled in light water reactors; nevertheless the recycling market will be incomparably larger in the USA than in Europe.

For the Community of the Six it can be predicted that, if the excess plutonium produced up to 1985 were recycled in light water reactors, the cumulative requirements for natural uranium and separative work in the period 1975-85 would be reduced by about 6 and 8% respectively.

Thus the market value of plutonium will settle down, on the world market, at a value somewhere between its production price and its equivalence value for recycling in light water reactors.

4.5 Present state of the art of Pu recycling in LWRs

Technically, the recycling of plutonium in light water reactors appears perfectly feasible.

4.5.1 Extraction of plutonium from irradiated fuels

Numerous studies have been carried out on the extraction of plutonium by the aqueous route from ceramic fuels irradiated in water-cooled reactors. This technique is now proven, and numerous plants use it as a basis (see Section 3.6).

4.5.2 Technology of UO_2/PuO_2 cores for light water reactors

The development of this technology has been the subject of numerous R&D studies carried out in the USA and the Community countries over the last 15 years.

These related on the one hand to methods of fabrication (vibro-compacted fuels, sintered fuels) and in-pile behaviour (irradiation and post-irradiation examination) of these fuels, and on the other hand to the analysis of the behaviour of LWR cores containing plutonium (physics studies, power density, temperature coefficients).

In the Community countries, these studies were started and carried out, largely under agreements between Euratom and the USAEC, by the CEA, Belgonucléaire and Alkem, with the assistance of various electricity producers, and in particular the Italian ENEL.

At present there are several pilot or preindustrial plants available for the fabrication of UO_2/PuO_2 fuels, e.g., those of Alkem, Belgonucléaire and the CEA. The principal irradiations are now being done in the Garigliano, Kahl[1], MZFR and BR-3 reactors; some are to be carried out shortly in the Dodewaard reactor, while others are scheduled for the KWO reactor (1973-74).

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[1]It is planned to load a whole plutonium-enriched core into the Kahl reactor in 1972 and into the Garigliano reactor at a later date.

The calculation of the behaviour of UO_2/PuO_2 cores for water-cooled reactors has reached a stage of maturity almost comparable with that of uranium-fuelled water reactors; the theoretical calculations still need to be confirmed by power density measurements on high burnup cores.

The technological development of mixed UO_2/PuO_2 fuels in sintered pellet form is equal to that of UO_2 fuels.

The development of vibrocompacted fuels is slightly behind that of sintered fuels; they offer a greater economic potential than the latter, provided that a high-capacity production line is installed. Nevertheless, owing to the mixed character of plutonium fuel fabrication plants (fuel for water reactors and fast reactors) and the electricity producers' preference for sintered fuels, it is foreseeable that sintered-fuel technology will predominate.

The cost of fabricating sintered UO_2/PuO_2 fuel elements is estimated to be 15-25% higher than that of UO_2 fuels, given the same production capacity of about a ton a day; both costs are very sensitive to the production capacity (see Fig. 4 for evaluation of partial costs), at least for outputs of less than 1 t/d.

This additional cost that plutonium fuels have to bear by comparison with uranium fuels lowers the "theoretical" (or nominal) equivalence value of plutonium as against uranium; the true equivalence value of plutonium, expressed as a percentage of its nominal value, is given in Fig. 5 vs the plutonium production capacity, the reference uranium production capacity being taken to be 1 t/d.

Table 4.5 lists the firms in the Community concerned in the fabrication of plutonium fuels.

Table 4.5. Firms concerned in the fabrication of plutonium-containing fuel elements

Country	Works/firm	Production capacity	Year of commissioning
Germany	Wolfgang/Alkem	40 t/year of Pu-containing fuel for LWRs	1972
Belgium	Mol/Belgo-nucléaire	Pilot unit	in operation
Belgium	Mol/Belgo-nucléaire	3.5 t/yr of Pu-containing fuel for fast reactors or 30 t/yr of Pu-containing fuel for LWRs	1973
France	CEA	Pilot unit, mainly for fast reactors	in operation
Italy	CNEN	Fuel laboratory	in operation
UK	UKAEA	Pilot unit, mainly for fast reactors	in operation

4.6 Trend of industrial maturity of plutonium recycling in the Community

Most of the plutonium produced in the Community power plants will be absorbed up to 1975 by the fast reactor programme. This plutonium will supply the preindustrial UO_2/PuO_2 fuel fabrication plants now under construction or operating in the Community. As these are dual-purpose plants, where plutonium fuels can be made for fast reactors and for thermal reactors, there is every reason to believe that the first Pu reloads for light water reactors will also be made there. This will constitute the preindustrial introduction of plutonium recycling in light water reactors.

This process will be speeded up between 1976 and 1980, during which period the excess of the Community plutonium supply over the Community demand for the fast reactor programmes will amount to several tons.

Towards the end of this period it should be possible to supply a Pu fuel production capacity of 100 t/year in the Community; this stage might establish the industrial character of plutonium recycling in light water reactors.

In the United States this development stage will be reached about five years earlier.

In the Community there is no hope of accelerating this industrial development process unless the Community obtains its plutonium from the external market (e.g., instead of enriched uranium) or the Community industry succeeds in cornering part of the American market for recycling plutonium in light water reactors.

5. Economic aspects of the fuel cycle

The nuclear programme involves heavy capital investment for new power plant construction and it also entails substantial expenditure for the fuel cycle.

In the preceding sections numerous details have already been given on the cost of uranium in its different forms and the specific costs of the fuel cycle operations. These data are brought together in this section to provide a consistent evaluation of the turnovers and sums involved in the execution of the present nuclear energy programme. Although this expenditure is well below the turnover of the power plant construction industry, its total sum, which will increase rapidly during the coming decades, nevertheless amounts to a considerable figure.

5.1 Review of costs of the various materials and operations

The rapid growth of the industrial nuclear market is accompanied by a lively movement of the prices for the various products and operations in the cycle. For this reason, any forecast of price trends can only be based on rough estimates.

5.1.1 Cost of natural uranium U_3O_8

The various estimates of the cost of U_3O_8 reflect market trends over the past few years; at present there is overcapacity. Owing to setbacks in the nuclear programmes, demand on the world market has risen more slowly than was expected. Recent discoveries of very rich deposits have also helped to keep prices down.

The USAEC report "Current status and future technical and economic potential of light water reactors", WASH 1082, March 1968, contains the following estimate:

	$/lb U_3O_8	$/kg U
1970	7.00	18.2
1975	7.75	20.2
1980	7.75	20.2
1985	8.25	21.4

Another USAEC estimate of January 1969 gives the following figures:

	$/lb U_3O_8	$/kg U
1971	7.20	18.7
1972	7.47	19.4
1973	7.71	20.0
1974	7.95	20.7

The NUEXCO estimate (October 1969) predicts the following trend:

	$/lb U_3O_8	$/kg U
1969	6.15	16.0
1970	6.20	16.2
1971	6.50	16.9
1972	6.75	17.6
1973	7.05	18.3
1974	7.50	19.5
1975	8.00	20.8

The analysis of the trend of cycle costs in the USA, given in a Westinghouse report, "Projection of nuclear fuel cost trends", December 1969, leads to the conclusion that the cost of uranium will tend to stabilize in the long term, between 1975 and 1985, at

$7/lb U_3O_8 (in 1970 currency)
= 18.2 $/kg U.

According to the papers presented at the Foratom Congress in Stockholm in September 1970, the present cost of U_3O_8 can be expected to remain steady until demand equals production capacity, and to rise in the long run.

The following trend was indicated:

	$/lb U_3O_8	$/kg U
1970-76 or 1978	6.5	16.9
long term	8.0	20.8

These estimates do not take into account the recent discoveries in Australia, Canada and South-West Africa.

In view of the present prospecting activities and the trend of uranium production, it seems fair to adopt the following cost for the evaluation of the total cycle cost, even in the long term[1]:

6 $/lb U_3O_8 = 15.6 $/kg U
in 1970 currency.

5.1.2 Cost of conversion into UF_6

The various information sources estimate this cost as follows:

The Nuclear Industry 1971 (USAEC):
$2.76/kg U (i.e., about 2.4 u.a. (1970)/kg U)

Jahresbericht der Atomwirtschaft 1971:
$2.2-2.9/kg U

[1]cf Section 3.1.6.

Westinghouse:
long-term 1975-85: \$2.30/kg U

For the Community, the following cost will be adopted:
2.40 u.a./kg U (basic cost; 1970 values).

5.1.3 Enrichment costs

The USAEC rate for uranium (UF_6) enrichment, which had long stood at \$26/kg SWU, was raised to \$28.7/kg SWU on 22 February 1971 and was again increased on 14 November 1971 to \$32/kg SWU.

This figure of \$32/kg SWU will be adopted for the present estimate.

5.1.4 Fuel element fabrication cost

The manufacture of LWR fuel elements comprises the following stages:

- reconversion of enriched UF_6 into UO_2,
- fabrication of pellets,
- fabrication of clads,
- cladding,
- assembly of elements.

Fabrication costs are changing quickly and depend closely on circumstances in the various countries (size of plant, capacity in use, standardization, manufacturer's experience, quality and guarantees demanded, etc.).

In the light of report A/Conf 49/P/062, published on the occasion of the Third Geneva Conference, the cost evolution can be estimated as follows*:

*Dollars at 1970 value; fixed charges at 30%; plant operating 260 days per annum.

Year	Average plant capacity t/day	Fabrication cost, BWR	$/kg U PWR
1970	0.6	90	100
1975	1.3	70	78
1980	5.4	43	50

"The Nuclear Industry 1971", estimating the total turnover of the United States fuel-element industry for the period 1970-85, bases its calculations on an average cost of $70/kg U.

The "Jahresbericht der Atomwirtschaft 1970" indicates that fabrication costs might drop from today's level of $90-110 to about $50/kg U in 1980.

The following average costs were adopted for the estimate of the total turnover in the Community:

Year	$/kg U
1975	110
1980	85
1985	70

5.1.5 Irradiated fuel transport costs

The ruling price in the USA for the transport of irradiated fuel elements is $8/kg U. The forecasts (see WASH 1082) point to a decrease to $4/kg U around 1980. According to the Westinghouse estimates, transport costs will go down from $8/kg U in 1975 to $7.50/kg U in 1980 and $7/kg U in 1985.

The prices in force for road transport in Germany vary from $3 to 5/kg U for short distances and are as much as $7.5/kg U for long distances.

For the estimate of the overall turnover an average cost of \$5/kg U was adopted for the period 1975-85.

5.1.6 Cost of reprocessing irradiated fuel

The reprocessing cost depends very closely on the plant capacity and the load factor.

Because of the excess capacity in the USA and Europe, the present cost of reprocessing does not correspond to the true cost. The cost based on a reprocessing capacity of 1-2 t/day is at present roughly \$32/kg U, not including the reconversion into UF_6, whereas the price to the customer is about \$20/kg U. With the building of large-scale plants (5 t/day) for a bigger market in the long run, this cost can be expected to come down.

This optimum size would bring the reprocessing cost to around \$25/kg U.

The Westinghouse estimates predict that the reprocessing cost will move as follows:

\$30/kg U in 1975
\$24/kg U in 1980
\$19/kg U in 1985.

The figures given at the Foratom Congress (Stockholm, September 1970) vary according to the plant capacity and the load factor and are as low as \$22/kg U.

To the cost of reprocessing the irradiated fuel must be added the cost of processing and storing the radioactive waste, which may rise from \$4/kg U to \$10/kg U (Foratom Congress). For the purpose of estimating the total cost of the radwaste reprocessing and storage operations, the following average values were used:

1975	$35/kg U
1980	$35/kg U
1985	$30/kg U.

5.1.7 Cost of reconversion into UF_6

In the Westinghouse study the reconversion cost is estimated as follows:

1975	$5/kg U
1980	$3/kg U
1985	$2.8/kg U.

The figures given at the Foratom Congress are $4-5/kg U.

The average cost adopted here for the estimate of turnover is:

Year	$/kg U
1975	4.5
1980	3
1985	2.8

5.1.8 Value of plutonium

The present estimates predict that the fissile Pu recovered from LWRs will have a use value stemming from its energy value and the excess cost of fabricating plutonium-bearing fuel elements. This value may be around $6-7/g Pu. For the purpose of calculating the turnover, a value of $7/g fissile Pu is adopted here.

5.1.9 Summary of costs of the various materials and operations

Table 5.1 lists the costs adopted in the preceding sections for estimating the fuel cycle turnover.

Table 5.1. Review of costs of the various materials and operations

	1975	1980	1985	average 1975–85
Natural uranium at mine				
$/lb U_3O_8	6	6	6	6
$/kg U	15.6	15.6	15.6	15.6
Conversion of U_3O_8 into UF_6				
$/kg U	2.4	2.4	2.4	2.4
Enrichment				
$/kg SWU	32	32	32	32
Fuel-element fabrication				
$/kg U	110	85	70	85
Transport of irradiated fuel				
$/kg U	5	5	5	5
Reprocessing of irradiated fuel including radwaste storage				
$/kg U	35	35	30	30
Reconversion to UF_6				
$/kg U	4.5	3	2.8	3
Value of fissile Pu recovered from LWRs				
$/g Pu	7	7	7	7

5.2 Estimate of expenditure on the different stages of the fuel cycle

On the basis of the costs shown in Table 5.1 and the requirements in Tables 2.5–2.9, one can estimate the expenditure on the different steps in the fuel cycle. The figures in Table 5.2 represent an overall estimate only, but they can be used to establish the order of magnitude of the fuel cycle turnover and its trend over the years. A tails assay of 0.25% at the enrichment plant was assumed for the calculations.

Table 5.2. Total approximate expenditure for the fuel cycle (10^6 u.a.)

	1975	1980	1985	1975–85
Natural uranium at mine	65	170	320	2000
Conversion of U_3O_8	10	25	50	300
Enrichment	50	190	400	2300
Fuel-element fabrication	65	170	320	2100
Transport of irradiated fuel	low	5	10	50
Reprocessing of irradiated fuel, including storage of radwaste	5	25	60	300
Reconversion into UF_6	low	low	5	30
Value of fissile Pu recovered from LWRs	4	25	75	350
Total net expenditure for the cycle after deduction of Pu credits	190	560	1100	6700

The expenditure for capital investment in the nuclear power plants and in the fuel cycle are compared in Table 5.3. To calculate the former, the following construction costs were adopted, decreasing from

200 u.a./kWe in 1975
to 170 u.a./kWe in 1980
to 160 u.a./kWe in 1985.

For each power plant the costs were distributed uniformly over the five-year period preceding its commissioning.

Table 5.3. Comparison between the expenditures for capital investment in the power plants and in the fuel cycle (10^6 u.a.)

	1975	1980	1985	1975-85
Total net expenditure for the cycle after deduction of Pu credits	190	560	1100	6700
Expenditure for capital investment	975	1650	2430	18600
Ratio	0.20	0.35	0.45	0.36

It is immediately noticeable in the last line of this table that the ratio between expenditure on the fuel cycle and on investment rises rapidly, reaching about 45% in 1985. The reason for this trend becomes clear if it is borne in mind that at the start of the period, when the existing plant capacity is very small, the cycle expenditure mainly relates to the first charges, which are proportional to the capital investments. From 1978-79 onwards, consumption overtakes the first-core requirements and continues to increase rapidly owing to the combined effect of the growth of installed nuclear capacity and the relative reduction in the nuclear power plant growth rate.

5.3 Estimates of the capital investment required for the various fuel cycle sectors

The indications given in the preceding sections can be used to estimate the capital investments that will be needed for the fuel cycle in the

Community between 1975 and 1985 in order to meet the requirements of the present programme. Table 5.4 shows the order of magnitude of these investments sector by sector.

Table 5.4. Cumulative amount of additional capital investments needed in the Community (10^6 u.a.)

Cumulative amount up to	1975	1980	1985
U_3O_8 supply (prospecting and production)	~100	~350	~750
Conversion/reconversion	low	~50	~150
Enrichment	-	~700	~1400
Fuel-element fabrication	-	~50	~100
Reprocessing	low	low	~100
Total	~100	~1200	~2500

6. Calculation of the fuel fraction in the cost per kWh

The fuel fraction in the cost per kWh can be calculated from the reactor and fuel cycle characteristics shown in Tables 2.2-2.4 and the economic data given in the preceding sections. Using the Euratom Handbook method to evaluate the price per kWh, the cycle cost per kWh has been computed for a 1000 MWe light water reactor commissioned in 1975. The reactor service life is assumed to be 30 years.

The principle of the method consists in adding together the expenditure at present-worth value for all the fuel and all the operations in the cycle during the reactor lifetime and relating this sum to the total energy, also on a present-worth basis, produced by the reactor.

The basic assumptions used for the calculation are given in the Handbook. The four-zone core exchange model is also adopted, the reactor being at equilibrium when the fourth fuel batch is replaced.

During the start-up period the reactor is operated as follows:

3000 hours in the first year
5000 hours in the second year
6500 hours from the third year onwards.

Refuelling is done once a year. The time allowed for fabricating the fuel elements is four months.

In accordance with the thirty-year life, the last batches are Nos. 30, 31 and 32. They are discharged part-used and the U and Pu value is credited on the basis of the average content.

The annual interest rate was taken to be 8%.

Table 6.1 below gives the principal data used in the calculations.

Table 6.1

Technical data

Weight of first charge (enrichment 2.27%)	112,000 kg
Reserve (enrichment 2.9%)	2,000 kg
Weight of individual batch (enrichment 2.9%)	28,000 kg
Uranium loss during irradiation	2%
Uranium loss during reprocessing	1.3%
Plutonium loss during reprocessing	1.3%
Uranium loss during reconversion	0.5%
Plutonium loss during conversion	0.5%
Specific plutonium production	5.2 g Pu/kg U
Specific plutonium production (four first batches)	5.83 g Pu/kg U
Specific plutonium production (batches 30, 31 and 32)	4.28 g Pu/kg U

Specific expenditure

Uranium in UF_6 form, losses included		
enriched to 2.27%		$158/kg U
enriched to 2.9%		$219/kg U
Conversion Sintering of pellets Fuel element fabrication Transport of fresh fuel	1975–79 1980–84 1985–89 1990–	$110/kg U $85/kg U $70/kg U
Transport of irradiated fuel		$5/kg U
Reprocessing	1975–84 1985–	$35/kg U $30/kg U
Reconversion into UF_6	1975–79 1980–84 1985–	$4.5/kg U $3/kg U $2/kg U

Specific credits

Uranium in UF_6 form	
batches 1–4 enriched to 1%	$39.3/kg U
batches 5–29 enriched to 0.711%	$17.9/kg U
batches 30, 31 and 32 enriched to 1.32%	$66.2/kg U
Plutonium	$7/g Pu

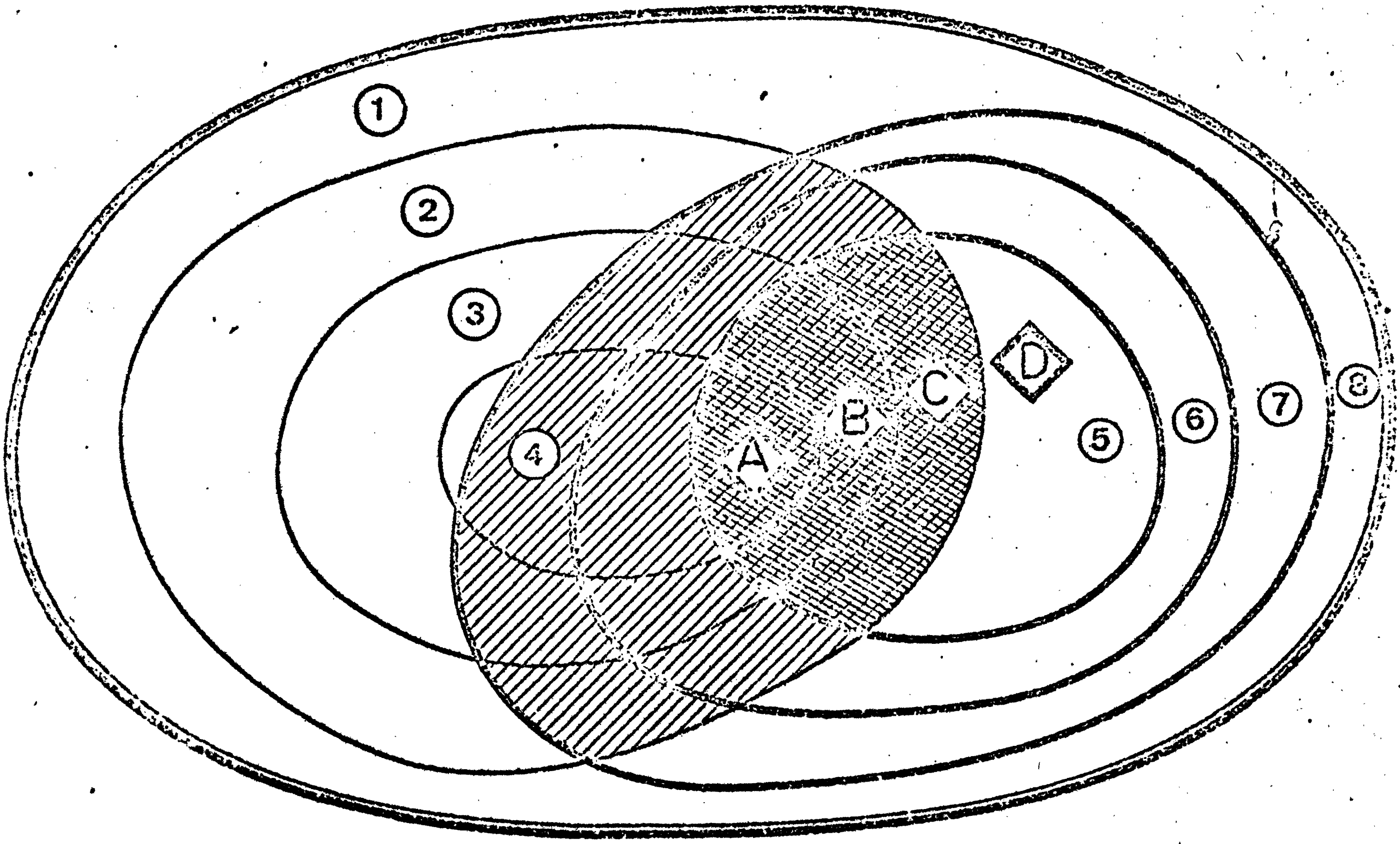
These data lead to the following values (Table 6.2 below) for each operation in the cycle. During the reactor lifetime the expenditure and credits listed above total a net overall value of 116.7 million u.a. on a present-worth basis.

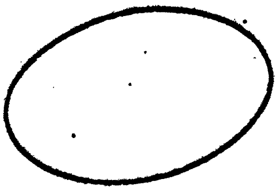
Total present-worth electricity production over the 30 years is $68.7 \cdot 10^{9}$ kWh. The specific cost of the fuel cycle is therefore $1.7 \cdot 10^{-3}$ u.a./kWh, a value which lies well within the range of the estimates given elsewhere.

Table 6.2

Operation	Weight kg	Specific cost of operation $/kg U	Overall cost of operation $
UF_6 purchase			
First charge	112,000	158	17,650,000
Individual batches	28,000	219	6,132,000
Reserve	2,000	219	438,000
Fabrication			
First charge	112,000	110	12,320,000
Reserve	2,000	110	220,000
Individual batches 1975-79	28,000	110	3,080,000
1980-84	28,000	85	2,380,000
1985-89	28,000	70	1,960,000
1990	28,000	50	1,400,000
Transport	28,000	5	140,000
Chemical reprocessing			
Individual batches 1975-84	27,440	35	960,400
1985-	27,440	30	823,200
Credits for each batch			
Credits for U: Nos. 1-4	26,950	39.3	1,061,000
Nos. 5-29	26,950	17.9	485,000
Nos. 30, 31, 32 together	80,840	66.4	537,000
Credit for Pu			
Nos. 1-4	140,900 g	$7/g Pu	985,000
Nos. 5-29	157,900 g	-	1,105,000
Nos. 30, 31, 32 together	349,500 g	-	2,447,000
Reconversion of U			
Individual batches 1975-79	27,083	4.5	121,875
1980-84	27,083	3	81,250
1985-	27,083	2	54,170

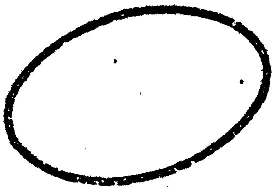
RESOURCES





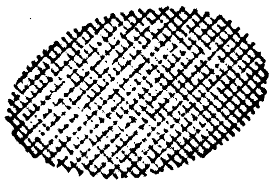
QUANTITATIVE ASPECTS

(1) not assessed (unknown)
(2) potential
(3) indicated } demonstrated
(4) measured }



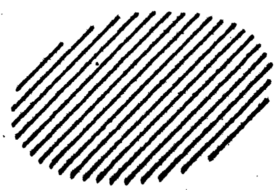
QUALITATIVE ASPECTS

(5) minable
(6) marginal
(7) submarginal
(8) latent



RESERVES

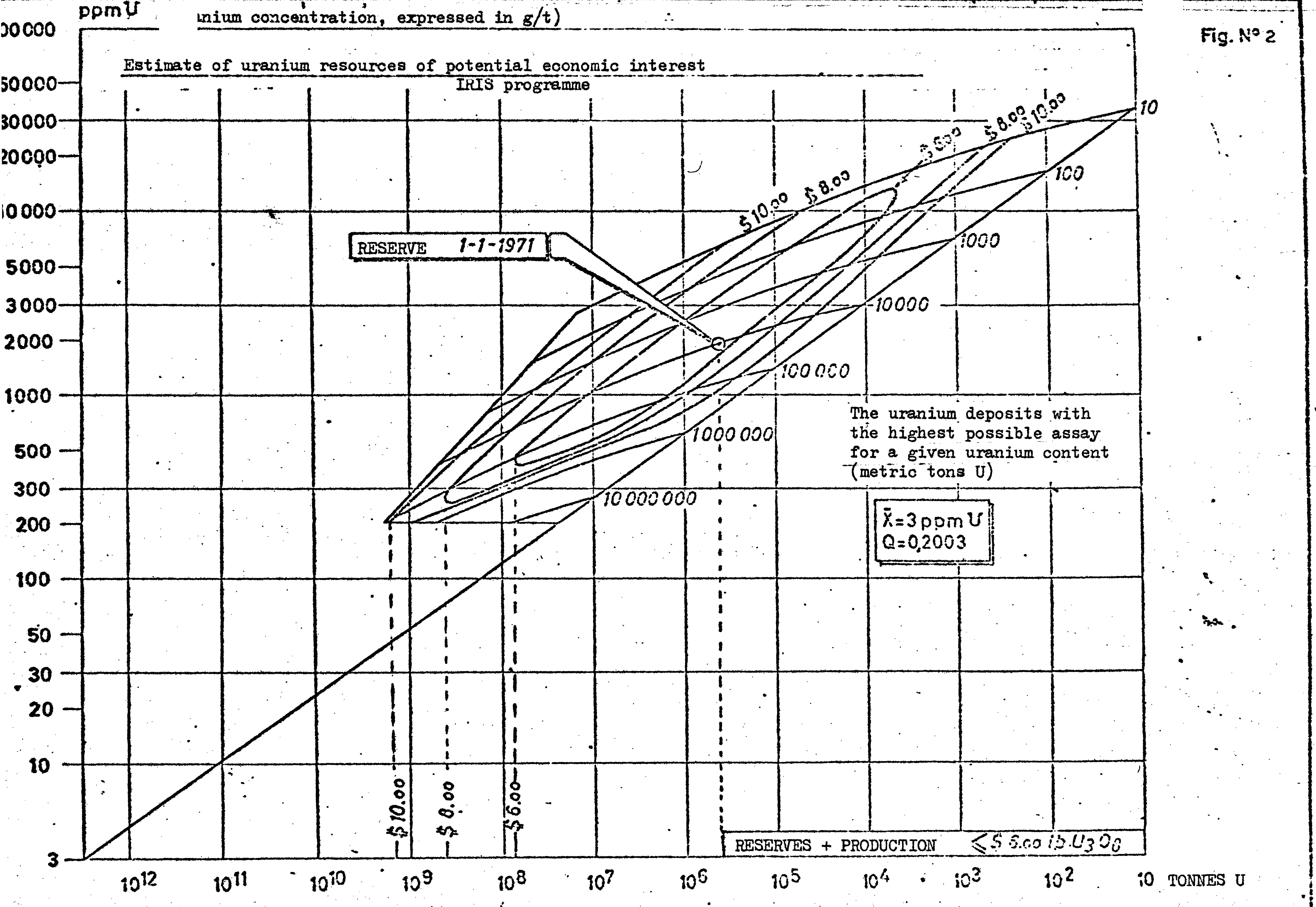
A = certain } reasonably assured
B = probable }
C = possible
D = not assessed (unknown)



POTENTIAL RESERVES

Fig. 1

Fig. N° 2


ppmU
…nium concentration, expressed in g/t)
Estimate of uranium resources of potential economic interest
IRIS programme
RESERVE 1-1-1971
$ 10.00
$ 8.00
$ 6.00
$ 8.00
$ 10.00
10
100
1000
10000
100 000
1000 000
10 000 000
The uranium deposits with the highest possible assay for a given uranium content (metric tons U)
$\bar{X}$=3 ppm U
Q=0,2003
RESERVES + PRODUCTION ≤ $ 6.00 lb U_3O_8
TONNES U

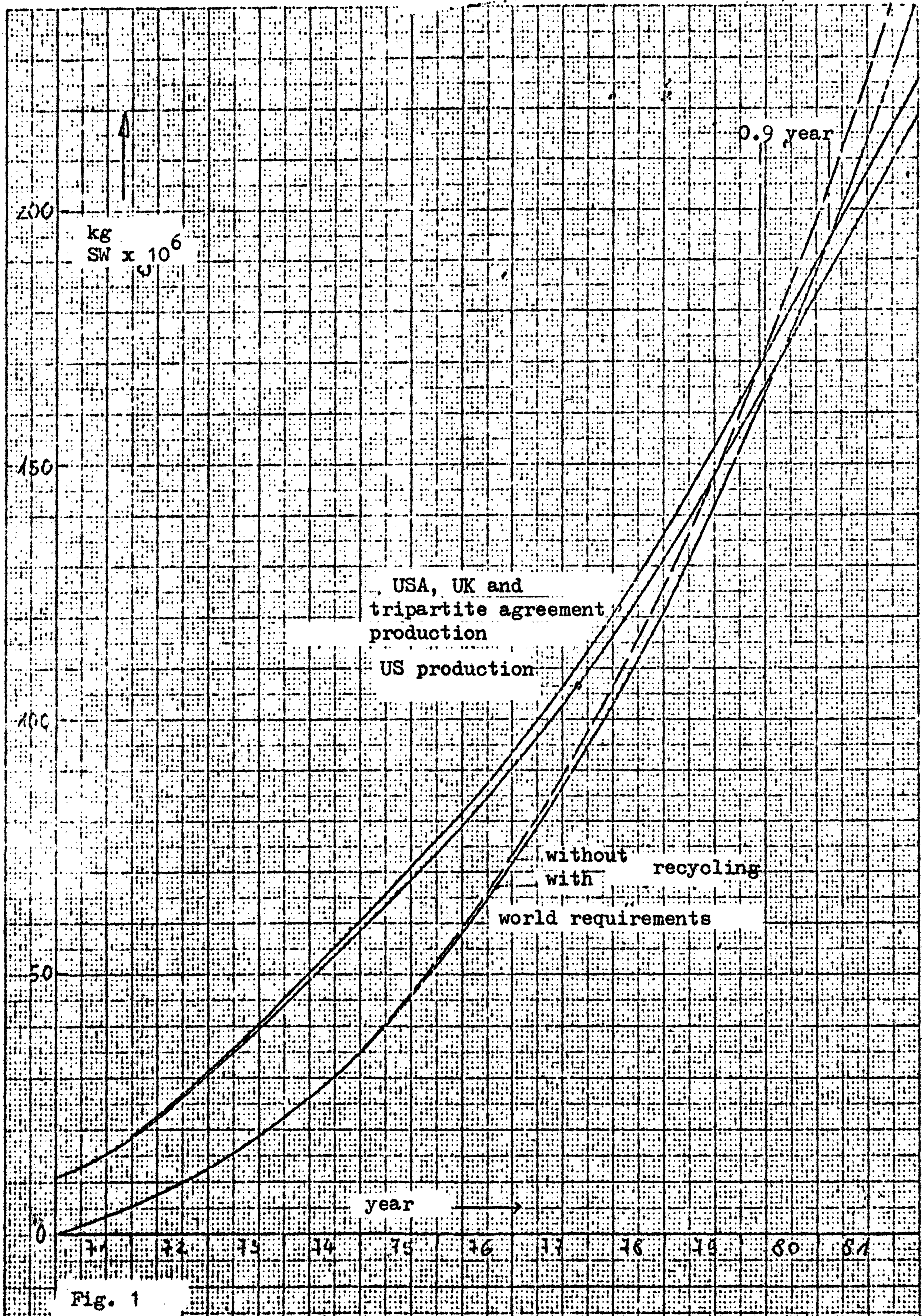
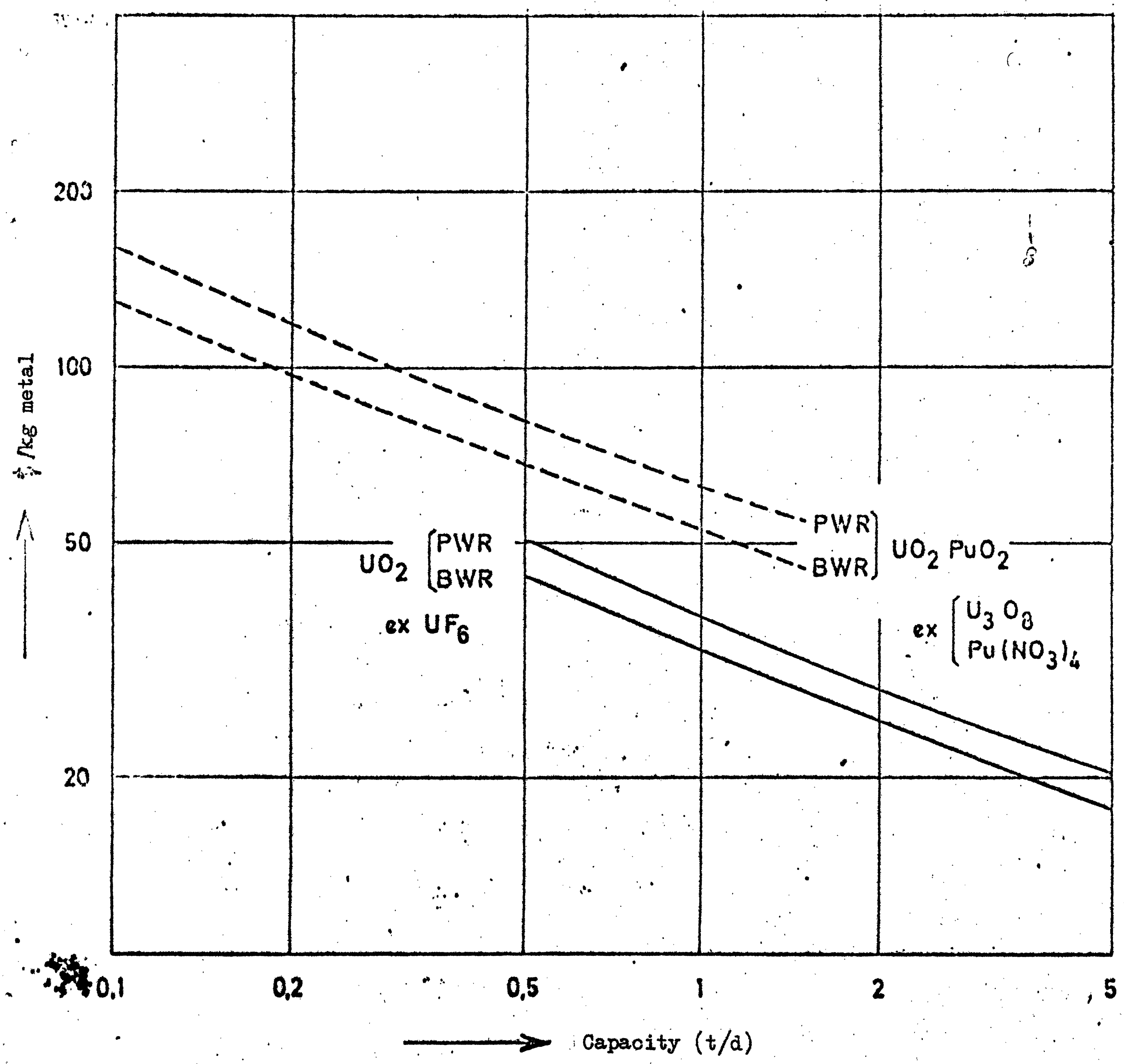


Fig. 1

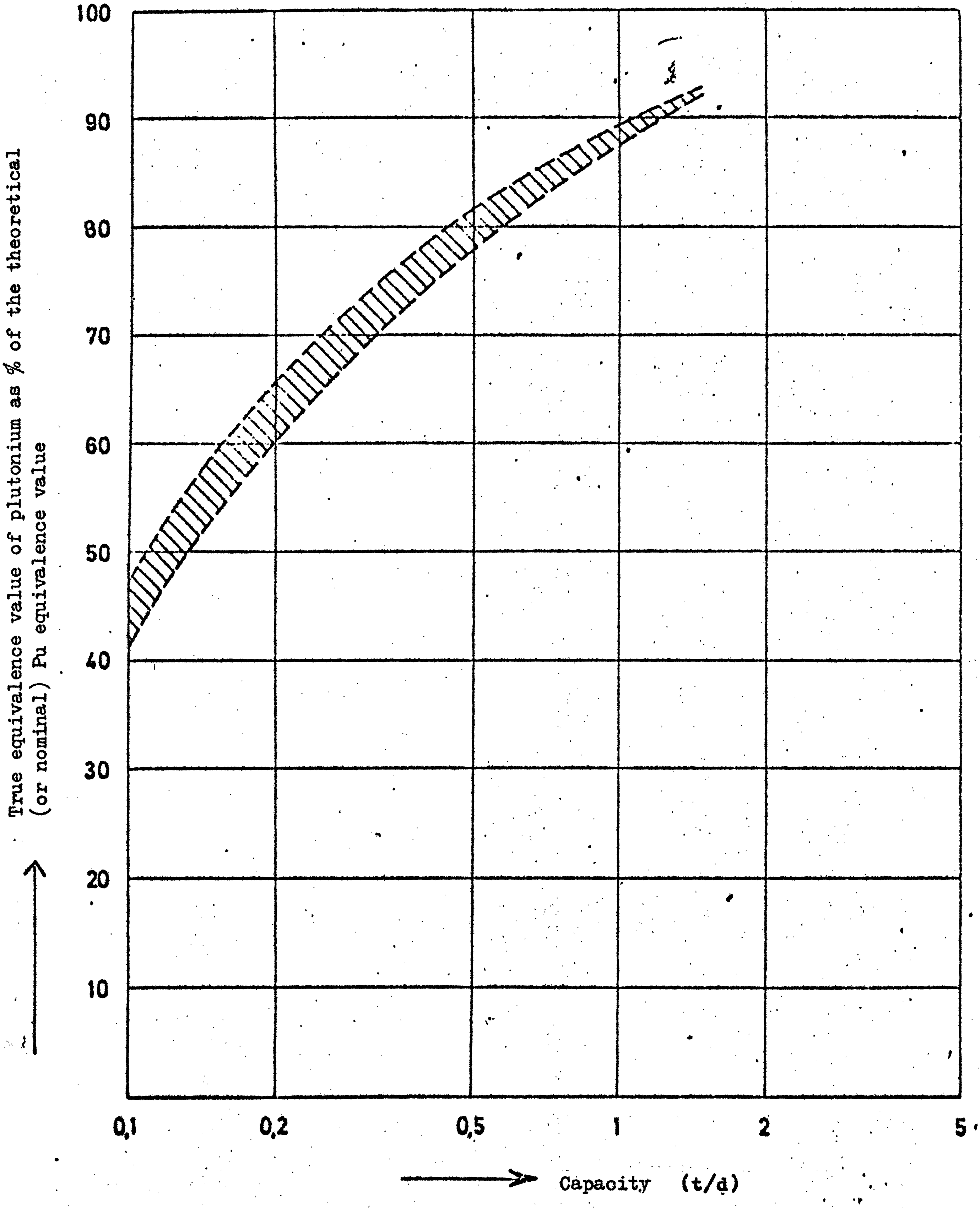
Separative work: requirements and output

Cumulative values as from 1 January 1971, including US reserves at that date.



Cost of preparation, pelletizing and assembly vs size of fabrication plants

FIG. 4



Relative value of fissile plutonium vs size of the Pu plant

FIG. 5

XVII/341/2/71-E

COMMISSION

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SECOND ILLUSTRATIVE NUCLEAR PROGRAMME

FOR THE COMMUNITY

ANNEX V

FAST BREEDER REACTORS

DRAFT

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FAST BREEDER REACTORS

I. INTRODUCTION

Breeding

The thermal energy liberated in a nuclear reactor results essentially from the fission of "fissile" material such as the uranium isotope U 235, the only one occuring naturally (0,7 % in natural uranium). Each fission liberates an average of more than two fast (high energy) neutrons. One of these neutrons is needed to trigger another fission to maintain the nuclear chain reaction, some are lost as they either leak out of the core or are absorbed non-productively, and the remainder are available to transform "fertile" isotopes of heavy elements into new fissile isotopes, i.e., to "breed" new fissile material. The fertile raw materials for such breeder reactions are thorium 232 which is transmuted to uranium 233 and uranium 238 which is transmuted to plutonium 239 by way of the following reactions:

$$n_0^1 + Th_{90}^{232} \longrightarrow Th_{90}^{233} \xrightarrow[23\ min]{\beta^-} Pa_{91}^{233} \xrightarrow[27.4d]{\beta^-} U_{92}^{233};$$

$$n^1 + U_{92}^{238} \longrightarrow U_{92}^{239} \xrightarrow[23\ min]{\beta^-} Np_{93}^{239} \xrightarrow[2.3d]{} Pu_{94}^{239};$$

Breeding occurs when more fissionable material is produced than is consumed. A convenient measure of this condition is the breeding ratio, BR > 1, or the breeding gain, (BR - 1).

If η is the number of neutrons produced per fission, the breeding ratio is

$$BR = \eta - 1 - \text{losses} \geq 1$$

Two different breeder systems are available. The "thermal breeder", employing slow (thermal) neutrons operates best on the thorium cycle achieving breeding ratios of about 1.1 at the most. The "fast breeder" employing high energy neutrons operates best on the plutonium cycle, and having both a higher neutron yield (η) and less non-productive absorption of neutrons, achieves breeding ratios of 1.4 and more. In this context it should be noted that it brings additional benefits to use the fissionable Pu-239 in fast reactors rather than thermal reactors due to the higher neutron yield than U-235 in the fast energy spectrum

A somewhat more significant parameter than the breeding ratio (BR) is the rate at which breeding occurs, i.e. the time required for a reactor to double its original inventory of fissile material called the "doubling time" (DT).

In first approximation

$$DT = \frac{3.2^{*}}{S\ (BR-1)} \qquad [\text{years}]$$

where S is the specific fuel rating in MW/kg fissile. Using for example a fuel rating of ~1 MW/kg, and BR = 1.4, a doubling time of 8 years is calculated.

Power Density

For reasons of general economics (inventory charges, doubling time) the fast reactor requires a fuel rating in the order of 1 MWt/kg which incidentally is similar to current ratings of thermal reactors. For thermal reactors such ratings cause no particular problem in regard to cooling as the volumetric fissile material concentrations are in the order of several grams per liter only. In a fast breeder reactor, however, this leads to power densities of 0.5 to 1.0 MWt/liter and more as such a dilution is not possible because:

- a dilution with moderating material would change the energy spectrum such as to reduce the neutron yield (η) and thus reduce the breeding gain;
- a dilution with structural material or coolant would have the same effect, at the same time increasing parasitic absorption thus again reduce the possibility of breeding;

---

* /1 + = 1; (where : ε = fast fission effect of U-238)
(α = capture oo fission ratio of Pu)

Load factor = 0.8;

- a dilution with breeding material, though desirable and necessary for the purpose of breeding, is only possible within close limits for criticality reasons (largely size dependent!).

This high power density has two direct consequences:

- the division of the fuel into very small diameter pins to avoid center fuel melting;
- the need for a good heat transfer medium leading to the choice of liquid metals as coolant from the very beginning of FBR development (although other coolants have been and are still being considered as indicated below).

Burn-up and Neutron Dose

If, for practical reasons, one includes in the above mentioned doubling time the out of pile fuel inventory, the doubling time becomes

$$DT^* = \frac{3.2}{S(BR-1)} \cdot \frac{Ft}{Fc} \quad ; \text{ or}$$

$$DT^* \quad \frac{3.2}{S(BR-1)} \cdot \left(1 + \frac{to}{tc}\right);$$

where Ft = total fuel inventory,
Fc = in pile fuel inventory,
tc = residence time of the fuel in pile,
to = out of pile residence time.

Or, expressing the in-pile residence time as

$$t_c = \frac{\text{burn-up (BU)}}{\text{fuel rating (S)}}$$

one obtains

$$DT^* = \frac{3.2}{BR - 1} \cdot \left(\frac{1}{S} + \frac{to}{BU}\right) \quad ;$$

which demonstrates the requirement of a high burnup in addition to the high fuel rating. For a commercial fast reactor a burnup of about 100,000 MWd/t is required for this and general economics reasons (reprocessing and fabrication) as compared to figures of about 30 000 MWd/t currently obtained in light water reactors.

While this already represents a factor of approximately 3, the actual consequences in regard to neutron exposure are much more servere due to the low fission cross-section in a fast neutron spectrum. As a direct consequence of this the neutron flux in a fast reactor is higher by the factor

$$\frac{\phi \text{ fast}}{\phi \text{ thermal}} \cdot \frac{\Sigma_f \text{ thermal}}{\Sigma_f \text{ fast}} = 200$$

This leads to a total damage dose of $2 \div 3 \cdot 10^{23}$ nvt for a fast reactor as compared to about 10^{21} nvt in a thermal reactor, a fact which is of particular concern for structural materials (swelling).

Dynamics and Safety

The fast neutron flux of an FBR also leads to some peculiar consequences in regard to safety for the following reasons:

- the neutron life time is in the order to 10^{-7} sec as compared to $10^{-3} \div 10^{-4}$ sec for thermal reactors; using plutonium as reactor fuel the delayed neutron fraction is about half that a U-235 fuelled reactor. While both thermal and fast reactors will become rather similar in the range of delayed criticality the period of a fast reactor drops to about $10^{-4} \div 10^{-5}$ as compared to $10^{-1} \div 10^{-2}$ in a thermal reactor when exceeding prompt criticality;
- due to the high power density temperature and power (reactivity) are closely coupled;
- temperature coefficients are small and depend strongly on the lay-out of the core both in magnitude and sign, as contrary to thermal reactors the geometrical arrangement does not correspond to the configuration of greatest reactivity.

Interest in Fast Reactors and Problems

This rather short and semiquantitative description already points out the potential advantages of fast reactors on the one hand, i.e.

- the capability to generate fissile material at a rate greater than thermal reactors and therefore use the available material resources to a much larger extent;
- when achieving a doubling time in the order of magnitude of the increase in electricity demand a fast reactor system is potentially self-supporting, i.e., creating its own fuel it is independent of enrichment capability. Only small amounts of natural uranium are needed to feed the system;
- the surplus of fissile material generated constitutes a credit item on the balance sheet of the fuel cycle cost, whereas the consumption of fertile material has only a negligible effect - leading to a low fuel cycle cost and a potentially lower total generating cost;
- for the same reasons the FBR should be practically unaffected by a rise in the cost of fissile material or natural uranium and thereby
    - permit the use of high cost uranium ores with little consequences on the total generating cost;
    - act as a stabilizer for fuel cost trends to the benefit of all reactor systems;
- provide a premium market for the plutonium produced by light water reactors;

and the principal problems faced by their development on the other hand, i.e.

- new coolant technology by using liquid metals as coolant or extremely high coolant pressures and velocities;
- development of a high burn-up plutonium bearing fuel;

- development of new materials required by the new coolant on one hand and the different neutron environment on the other;
- the resolution of safety problems due to the neutronic characteristics both from the point of view of understanding as well as engineered safeguard in order to avoid excessive penalty on the capital cost.

Early Development

The concept of fast breeder reactors is almost as old as the development of nuclear reactors. Realizing the high η - value for fast neutron induced fission, Fermi and Zinn began to design a fast breeder reactor as early as 1944. Development began soon thereafter in Great-Britain, Russia and France.

Consistent with the general approach to reactor technology of those early years, uranium metal was used as fuel, and, for reasons outlined above, liquid metal, i.e. sodium or NaK was used as coolant. The cores of this first round of fast reactors were small, the coolant temperatures were moderate and breeding was mostly externally, i.e. in the reflecting blanket and not so much in the core itself. The main purpose of the first experiments was to prove the general feasibility of the concept of breeding and the safety of the system. The economic aspects related to core inventory, burn-up and fuel cycle costs received little attention in this early development.

Table I summarizes the principal characteristics of this first round of fast reactors.

As a whole the experience obtained from this first development was very favorable. Indeed, EBR II, DFR and Rapsodie are still serving as irradiation facilities. EBR I suffered a major core meltdown in 1955 and demonstrated the importance of temperature related geometry effects on reactivity (fuel element bowing) but was subsequently repaired and then decommissioned in 1963 after satisfactory operation. DFR experienced

TABLE I

First Generation Fast Breeder Reactors

(EFK – 1167)

	U. S. A.				USSR			UK	France
	Clementine	EBR-I	EBR-II	EFFBR	BR-1	BR-2	BR-5	DFR	RAPSODIE
Reactor power									
Thermal MWt	0.025	1.2	62.5	200	0	0.1	5	72	20
Electrical MWe	0	0.2	20	66	0	0	0	15	0
Core									
Fuel	Pu metal	U metal	U metal	U metal	Pu metal	Pu metal	PuO_2	U metal	PuO_2/UO_2
Core volume liters	2.5	6	65	420	1.7	1.7	17	120	54
Fuel rating av MWt/kg fiss	0.0016	0.02	0.3	0.37	0	0.008	0.1	0.24	0.14
Power density av MWt/liter	0.01	0.17	0.8	0.45	0	0.06	0.3	0.5	0.32
Linear rod power max W/cm	(av 50)	300	450	250	0	150	200	(av 320)	(av 210)
Neutron flux max n/cm^2 sec	(av 5.10^{12})	$1.1.10^{14}$	$3.7.10^{15}$	$4.7.10^{15}$	5.10^{10}	1.10^{14}	1.10^{15}	$2.5.10^{15}$	$1.8.10^{15}$
Primary heat-transfer system									
Coolant	Hg	NaK	Na	Na	–	Hg	Na	NaK	Na
Coolant temperature									
Core inlet °C	40	230	370	290	–	30	375	200	410(450)
Core outlet °C	120	320	470	430	–	60	450(500)	350	500(540)
Coolant mass flow m^3/h	0.6	80	2200	5500	–	6	240	1800	800
Number of coolant loops	1	1	2	3	–	1	2	24	[illegible]
Time schedule :									
Design	1945	1945					1956		1958
Construction	9/1946	1949	1957	8/1956			1957	3/1955	1962
First criticality	11/1946	8/1951	10/1961	8/1963			6/1958	11/1959	1/1967
Full operation	3/1949	12/1951	4/1965	8/1966	1955	1956	7/1959	7/1963	3/1967
Shutdown	6/1953	1963	–	–	1956	1957	–	–	–
Remarks	First fast reactor. First Pu-fueled reactor	First nuclear electricity generation; Pu-core since 1962	Reactor plant [illegible] with integral fuel processing facility	Partial melt down oct.1966. Again on power since Aug. 1970			UC-core since 1965		Since 1970 transferred to Fortissimo version 40 MWt

early difficulties in removing impurities from the NaK coolant and suffered damage in primary piping in 1967, due to a design error leading to high thermal stress, which could also be repaired.

The EFFBR suffered a local melt down accident in 1966 due to subassembly blockage but has since been repaired and is currently operating at power levels up to 200 MWt. This reactor encountered repeated difficulties with the steam generators due to tube vibration and leaks in the tube to tube-sheet welds.

Present Programs

After this first round of reactors, and the relatively modest efforts associated with their construction, had proven the general principle of breeding some safety aspects, and the feasibility of cooling with a liquid metal, attention shifted to the economic aspects of the fuel cycle. Against a background of commercial light water reactors it was soon recognized that, at least initially, UO_2 - UO_2/PuO_2 could offer the best chances to obtain the high burnup of about 100,000 MWd/t required for fast reactors. At the same time it became clear that at the likely date of commercial introduction of fast breeder reactors it would be desirable to have power plants of 1000 MWe and more.

Technically this had the following consequences:

1) It was recognized that an intermediate step would be required, i.e. a non-commercial prototype in the 300 MWe range.

2) Reactivity coefficients of these reactors having a much softer neutron spectrum (moderation by oxygen atoms in UO_2/PuO_2) received renewed attention; in particular the Doppler effect, leading to the design and construction of the SEFOR (see table III) experimental reactor, and the void coefficient and related phenomena such as sodium boiling, superheat, fuel coolant interactions, etc......

3) Large programs to study material properties in sodium and fast reactor environment, and to develop commercial components, in particular large pumps and steam generators.

4) The power density in a large fast reactor with oxide (ceramic) fuel is lower by a factor or about 4 compared to early reactor concepts. This is due to the lower enrichment (10 to 12% instead of 20 to 25%) as a direct consequence of the size, and the lower density of PuO_2/UO_2 compared to metal fuel. It therefore became possible to consider other coolants than sodium, in particular He and dry steam.

When in 1966/1967 the first results of high burnup pin irradiations became available most fast reactor groups decided to take steps of building a prototype in the 300 MWe range. The first irradiations were carried out in thermal reactors such as BR-2 in Belgium, GETR at Vallecitos (USA) but others, in particular in DFR and EBR II, followed, indicating an initial performance capability of about 50,000 MWd/t, considered to be a good starting value for the first core of a prototype.

The UK was the first in the West to go ahead with the 250 MWe plutonium fuelled fast reactor, PFR, at Dounreay. France followed soon thereafter with the 250 MWe prototype Phenix and Germany together with Belgium, the Netherlands and Luxemburg somewhat later with their 300 MWe SNR. Itlay has decided to build a somewhat small but versatile test Reactor PEC (130 MWt) as a starting point for a possible future evolution, originally aiming at the vented fuel concept.

In Russia the design and construction of the BN-350 was going on and, at least timewise, this Russian group was and still is in the lead for this class of prototypes. Table II summarizes the principal characteristics of this class of reactors, i.e., the prototypes or demonstration plants (US-terminology).

In more recent years results on the behaviour of stainless steels used for cladding and other in core components having been exposed to fast fluences in excess of 10^{22} nvt and high fluxes ($> 10^{15}$ n/cm^2 sec) indicated reasons for concern as unexpected swelling phenomena occured. Mainly for this reason but also for more long range fuel and materials development as well as the development of incore instrumentation the US has given priority to the design and construction of a large fast test reactor FFTF, to the extent that construction of a demonstration plant can start in 1972 at the earliest (a decision, which at least initially was not leaked upon favorably by american industry, in particular General Electric, Westinghouse and Atomics International).

When the large size fast reactor provided the above indicated additional degree of freedom in regard to coolant technology, a number of groups considered dry steam as a coolant hoping to be able to extrapolate LWR technology to fast breeders. The doubling time was of course much larger (30 years and more) but the breeding ratio was still clearly above one. The concept was ultimately dropped because of difficulties in the fuel pin design and the need of start a rather broad development program (external dry steam corrosion attack, tight lattices, high temperatures), and to provide ultimately a test bed for such fuel with dry steam as coolant.

On the other hand the interest in using helium as a fast reactor coolant has been increasing in recent years. Again, the primary motivation is to use HTGR technology for the general engineering, components and perhaps fuel (coated particles). Contrary to the dry steam concept the GCFR has a potentially lower doubling time than the LMFBR and, in the long range could possibly be improved by the adoption of a direct He turbine cycle. The future of this concept depends largely on the commercial development of the HTGR concept. Adequate fuel testing capability, which, for the present LMFBR-line using UO_2/PuO_2 has been provided by DFR, EBR II, BOR-5 and Rapsodie, will also be required.

TABLE II

Second Generation Fast Breeder Reactors.

(EPK – 1167)

		USA			URSS		UK	France	Germany
		GE	Westinghouse	AI	BN-350	BN-600	PFR	PHENIX	SNR
Reactor power									
[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]
[illegible]	[illegible]	[illegible]10	300	500	350	600	250(275)	250	300
Core (reference)									
Fuel		PuO_2/UO_2	PuO_2/UO_2	PuO_2/UO_2	PuO_2/UO_2 or UO_2	PuO_2/UO_2	PuO_2/UO_2	PuO_2/UO_2	PuO_2/UO_2
Core volume		2000	1960	3000	1900	2300	1320	1150	1600
Fuel rating av	MWt/kg fiss	0.82	0.85	0.9	0.96		0.7	0.8	0.8
Power density av	MWt/liter	0.31	0.39	0.37	0.5	0.6	0.4	0.42	0.4
Linear rod power	max W/cm	500	440	490	470		450	430	400
Breeding ratio		1.2	1.22	1.3	1.5		1.2	1.16	1.29
Burnup	MWd/ton	100,000	75,000	75,000	50,000	100,000	70,000	50,000	55,000
Primary heat-transfer system									
Type		Pool	Loop	Loop	Loop	Pool	Pool	Pool	Loop
Coolant		Na	Na	Na	Na	Na	Na	Na	Na
Number of coolant loops		3	2	3	5	3	3	3	3
Pump capacity	m^3/h	5600	8500	8850	3200	9300	5000	4000	5100
Coolant temperature									
Core inlet	°C	425	400	405	300	380	400–425	400(420)	380
Core outlet	°C	590	550	570	500	530	560–585	560(580)	550
Steam conditions									
Temperature	°C	510	480	480	435	505	510–540	510	505
Pressure	at	160	170	163	50	140	162	167	165
Date of operation		?	?	?	1971/72	1975/76	1972	1973	1977

Finally it should be mentioned that also in Europe, particularly in Germany, it is felt necessary or at least desirable by some groups to have available a large fast reactor for fuel and materials development, not to achieve the goal of immediate commerciality as in the case of FFTF but to assure full long range exploitation of the commercial potential of the FBR beyond the present UO_2/PuO_2 concept. The Karlsruhe group (GfK) has just completed a preliminary design study for a fast test reactor FR-3 with Na-cooling and a number of test loops capable of handling - if desirable - other coolants such as He, a reactor which could suitably complement the Italian project PEC. Indeed the Commission has proposed in 1971 to examine the need of such a project within the Community and if desirable to press on with its construction as a joint project.

In conclusion the overall picture shows in most cases a line of development which leads to a prototype in the 300 MWe class. BN 350 is scheduled for operation in 1971/72, PFR and Phenix should go into operation in 1972 and 1973, respectively, followed by BN 600 in 1975/76 and, with some delay, by SNR-300 in 1977. Indeed as can be seen from recent developments on the side of both manufacturers and electricity producers, western Europe is already preparing for the next step - a close to commercial 1000 - 1300 MWe demonstration plant.

In the US the FFTF has become the major milestone in LMFBR development while both industry and utilities as well as the USAEC are trying to find the means and define the modalities for the construction of the first prototype (demonstration plant) and for the continued operation and exploitation of the Fermi project.

To complete the picture it should be mentioned that Germany has decided to convert the 58 MWt sodium cooled thermal reactor KNK I into a fast test facility (KNK II) scheduled for operation with a fast core in 1974.
Japan has decided to make the development of a sodium cooled fast reactor a major national project. This development is several years behind the European programs but the experimental reactor JEFR should be in operation in 1973 and the operation of a prototype reactor of the 300 MWe-class is planned by 1978/79.

Table III

SECOND GENERATION EXPERIMENTAL FAST REACTORS

		USA		USSR	France	Germany	Italy	Japan
		SEFOR	FFTF	BOR-60	RAPSODIE fortissimo	KNK-II	PEC	JEFR
Reactor power								
Thermal	MWt	20	400	60	40	58	130	100
Electrical	MWe	0	0	12	0	20	0	0
Core								
Fuel		PuO_2/UO_2	PuO_2/UO_2	PuO_2/UO_2 or UO_2	PUO_2/UO_2	PUO_2/UO_2 + UO_2	UO_2	PUO_2/UO_2
Core volume	liter	500	1030	53	45	320	420	280
Linear rod power max	W/cm	650	500	590	400	430	400	430
Neutron flux max	$n/cm^2 sec$	6.10^{14}	$7.2.10^{15}$		3.10^{15}	$2.3.10^{15}$	$2.8.10^{15}$	4.10^{15}
Primary heat-transfer system								
Type		Loop	Loop		Loop	Loop	Loop	Loop
Coolant		Na	Na	Na	Na	Na	Na	Na
Number of coolant loops		1		2	2	2	2	2
Coolant temperature								
Core inlet	°C	370	320	360-450	410	360	375	370
Core outlet	°C	430	480	600	530	550	525	500
Date of operation		1969	1974	1969	1970	1974		1973/74

Table III summarizes the principal characteristics of the above mentioned (second generation) test reactors including BOR-60, Rapsodie-Fortissimo, PEC, and KNK II.

It should be mentioned here that an alternative approach to breeding in a thermal reactor is being pursued at ORNL where the molten salt breeder reactor concept using the thorium cycle is under development. The concept uses, similar to MSRE*, a fuel which is dissolved in molten fluoride salts and reprocessed in a small on-line plant. Although the concept - still in its early stages of development - might become attractive by the turn of the century its principal disadvantage is a total lack of interest in this concept by industry.

2. PRESENT STATE OF TECHNOLOGY

The number of fast reactors in the 300 MWe range presently under construction or definitely planned demonstrates the degree of confidence in the know-how already obtained. The performance criteria adopted for these reactors are, with the exception of power and consequently size, relatively close to those expected for the commercial fast breeder reactors of the first generation. While these plants "demonstrate" the extent of present achievements, they only "prototype" what could be a commercially competitive plant, i.e.

- the performance is not guaranteed: some uncertainty surrounds the burnup capability of the fuel, especially in view of the fast neutron damage phenomena to cladding and other in core components, and in regard to the steam generator reliability;

* Molten Salt Reactor Experiment/First critical in 1965; operating since October 1969 on U^{233}, but closed now.

- the construction costs do not have direct commercial significance, although the actual costs will be an important factor in establishing the cost of future commercial plants. In fact, the R&D program in direct support of these plants is in the same order of magnitude, if not higher in some cases, as the actual cost for the construction of these plants. In addition, many ~~technological~~ problems have to be resolved on the industrial level due to the "first-of-a-kind" nature of most major components and the stringent quality assurance techniques to be applied for the first time. Excessive safety features are incorporated as the satisfactory performance of engineered safeguards will only be proof tested in these plants;

- the operation will be uneconomic, since apart from the perhaps limited fuel performance and the elevated capital cost, the fuel cycle capability, essential for implementing the principal direct, i.e. economical, advantage of the fast breeder reactor, will not exist.

These factors already indicate the principal areas of future R&D work and the supporting role these demonstration-(prototype) plants and in particular the special purpose reactors PEC and FFTF and other large facilities will have to play. The following will review in some more detail the main disciplines of fast reactor technology.

Core and Plant Design

Apart from some early concepts and some of the first generation fast reactors the liquid metal cooled fast reactor concept (LMFBR) now emerging from all fast reactor programs presents the same basic design characteristics, identified by:

- the use of sodium as reactor coolant;
- the use of the Rankine cycle for power generation;
- the use of an intermediate sodium circuit in order to avoid activation of the steam system and to maintain the integrity of the primary system in case of a steam generator leak;
- the use of free surface centrifugal pumps rather than electromagnetic pumps in the main heat transfer circuits.

Nonetheless there remain a number of design features which permit different solutions. The principal design choice lies in the arrangement of the primary system where a spread-out loop type or piped system is chosen by some and a pool or pot-type system with pumps and intermediate heat exchangers (IHX) in the primary vessel ("pot") by others. The principal advantages of the pool concept are:

- containment of the primary coolant in one vessel;
- lower thermal shock potential;
- reduced leak tightness requirement for primary system components, and less wetted surface contamined by primary sodium;
- smaller reactor building;
- lower capital cost.

The piped system on the other hand presents the following principal advantages:

- easier containment of a major excursion;
- easier access to components which facilitates maintenance work and could lead to increased plant availability;
- smaller and perhaps shop-built reactor vessel; facilitating fabrication and control;
- less potential to activate secondary sodium in intermediate heat exchangers;
- more potential in regard to reactor size.

Hybrid systems with pumps and intermediate heat exchangers in separate vessels have also been conceived and it is very likely that none of these concepts or subconcepts will be inherently superior to the other.

The second design choice is concerned with the fuel handling scheme where the two main concepts are:

- under plug fuel handling, and
- hot cell arrangement.

Again, many subsystems exist in particular in combination with the plant arrangement (pool or loop) chosen and in regard to cooling arrangements during handling, and intermediate storage previsions.

A number of other design choices do exist at this time but these are dictated by safety considerations and ist is very likely that the choice will no longer exist once common safety criteria (design basis accident, engineered safeguards, etc...) emerge. These are design options related to:

- containment; choice of a leaktight secondary containment or a low leakage conventional building;
- single or double wall primary piping;
- emergency cooling;
- core catcher;
- plug catcher.

The prototypes presently under construction or planned will play a major role in testing out some of these design options and in the evolution of acceptable safety criteria through development and proof testing of instrumentation and other engineered safeguards. They will not be representative of future plants in regard to safety measures and the special purpose reactors such as PEC and FFTF will need additional precautions related to their special function. One principal difficulty facing the designers of these plants are the uncertainties related to the stainless steel swelling phenomena requiring in core components to be exchangeable and fuel handling equipment to manipulate distorted subassemblies.

Sodium Technology and Components

When the first experimental liquid metal cooled reactors were designed it was not immediately clear that a whole new technology had to be mastered. Most of the early components were designed and built on a one-of-a-kind basis by adapting existing designs and techniques to the special needs and the requirements imposed by the use of sodium (or NaK). Sodium technology was treated very summarily, and in general the provision of an inert covergas (helium or argon) and the assurance of reasonable purity were thought to suffice. Even when problems with carbon transfer and mass transfer in general became known, the basic knowledge in sodium chemistry, methods of analysis and on-line instrumentation was inadequate to assure meaningful experimentation. Sodium chemistry and the interactions of cladding and structural materials with sodium have been studied extensively in recent years and it should be possible to design the present prototypes and facilities with reasonable confidence from this point of view. Analytical methods are well in hand and instrumentation and purification systems can be expected to give the necessary assurance to the operator for the maintenance of an adequate sodium purity. The same is true in regard to the necessary inert gas systems.

Present efforts are concentrated on the further development of on-line instrumentation for reliable long-term service in commercial systems. The behaviour and control of fission products in sodium and a number of special effects such as material compatibility in sodium environment, self-welding, wear, fretting corrosion, sodium water reaction and the relative equilibria of impurities in both covergas and sodium need further work. The design of purification systems is still lacking the scientific basis for a rational design of its components.

While most of the early components operated satisfactorily and much was learned from their mulfunctions, systematic development techniques had to be applied when economic considerations gained in importance. It became clear that in order to develop safe, reliable and economic components it would be necessary to provide major test facilities and to involve the component manufacturers. Table IV gives a summary of facilities built for testing of

SODIUM COMPONENT TEST FACILITIES

	Facility	Purpose	Technical data	Time schedule
[illegible]	[illegible] sodium component test installation	Testing of different steam generators and intermediate heat exchanger	Na-Na-steam system Na: max 650°C(700°C) Steam: 560°C/170 atm	1965 Preoperational t 1966 Operation
	SPM sodium pump test facility	Testing of pumps	Pump capacity up to 15000 m^3/h, temp max 650°C (intially only 5000 m^3/h)	1971 Construction 1973 Operation
[illegible]	3 MW sodium test loop	Investigation of steam generator and intermediate heat exchanger models		1960 Operation
	Sodium pump test facility	Testing of BN-350 pumps		1966 Construction
K	Sodium pump test facility	Testing of sodium pumps	Pump capacity 1620 m^3/h	1964/65 Operation
rance	5 MW grand Quevilly	Investigation of steam generator and intermediate heat exchanger models	Na-NaK-steam system Na: max 600°C(625°C) Steam: 545°C(565°C)/130 atm	1964 Operation
	50 MW EDF test facility	Testing of steam generator	Na-steam system Na: max 650°C	1967 Construction 1970 Operation
ermany-[illegible]	5 MW INTERATOM test facility	Investigation of special aspects of steam generators	Na-Na-steam system Na: max 560°C Steam: 500-540°C/200 atm	1963 Construction 1965 Operation for KN 1969 Operation for KN
	INTERATOM sodium pump test facility	Testing of pumps	Pump capacity 5000 m^3/h (15,000 m^3/h)	1967 Construction 1970 Operation
	50 MW NERATOOM sodium component test facility	Testing of 50 MW steam generator and 70 MW intermediate heat exchanger	Na-Na-steam system Na: max 650°C Steam: 600°C/215 atm	1968 Construction 1971 Operation
apan	2 MW sodium test facility	Investigation of various characteristics of sodium components	Na system, max 650°C	1969 Operation

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the principal components of main concern, i.e. the steam generator and the pumps. Industry has become involved increasingly as it is engaging in the design and construction of the prototype (demonstration) plants and is learning to design and fabricate components to tight tolerances needed and to apply the rigid quality assurance requested by the designer. The prototype reactors under construction or presently planned will allow the first large scale application of this industrial know-how. It is to be expected that minor difficulties will arise for most of these plants, especially in areas where first-of-a-kind industrial activities are involved, such as:

- the reactor roof required in the pot designs; the difficulties experienced at PFR are a good example of this;
- the large diameter thin-walled piping needed for the loop designs;
- the rotating plug required for most designs.

Mechanical free surface pumps used in the main sodium circuits as well as intermediate heat exchangers, tanks and most of the minor equipment and components pose little problems at this stage and for the reactor power levels under considerations.

The component of principal concern over the years has been and still is the steam generator due to the exothermic and violent sodium/water reaction phenomenon which would occur in case of a leak. There is little doubt that single wall solutions are imposed for economic reasons. While all of the demonstration plants will be provided with such single wall solutions, none of these proposals is truly representative of commercial designs envisaged for larger plants. Probably the steam generators of the prototype reactors will operate reasonably well, but may be subjected to frequent repairs which would reduce the availability of the plant. The extent of the work remaining to reach a commercial design will not be clear until these prototypes are operated.

FAST CRITICAL ASSEMBLIES

	Location	Year first critical	Short description	Fissile material	Typical core size, liters (for average reflector thickness)
ZPR-3	Argonne, Idaho	1955	Horizontal, split-table machine	U235, Pu239 (600 kg)	600
ECEL	Atomics International, California	1960	Horizontal, split-table thermal driver	U235, 25 kg of U233 were used in some assemblies	100 (test zone)
VERA	Aldermaston UK	1961	Vertical, split-table	U235, Pu239 (40 kg)	400
BFS	Obninsk, USSR	1961	Vertical, fixed	U235	1800
ZEBRA	Winfrith, UK	1962	Vertical, fixed	U235, Pu239 (400 kg)	3000
ZPR-6	Argonne, Illinois	1963	Horizontal, split-table	U235, Pu239	3000
ZPR-9	Argonne, Illinois	1964	Horizontal, split-table	U235, Pu239	3000
FRO	Studsvik, Sweden	1964	Vertical, split-table	U235	65
MASURCA	Cadarache, France	1966	Vertical, fixed	U235, Pu239 (200 kg)	3000
SNEAK	Karlsruhe, F.R. of Germany	1966	Vertical, fixed	U235, Pu239 (200 kg)	3000
FCA	Tokai-Mura, Japan	1967	Horizontal, split-table	U235, Pu239 planned	3000
ZPPR	Argonne, Idaho	1968	Horizontal, split-table	U235, about 3000 kg of Pu239 planned	3000
STEK	Petten, the Netherlands	1969	Vertical, fixed, thermal driver	U239	250

Physics and Safety

Fast reactor safety has been an important subject for exploration from the very beginning. Originally it was the short neutron lifetime that caused concern until it became clear that it is actually an advantage provided that the prompt power coefficient is negative.

The second concern for fast reactor safety stemmed from the EBR I melt-down due to the thermal bowing effect of the fuel element resulting in a positive partial and instantaneous power coefficient. The problem is well understood now and, using support and restraint mechanisms, cores are now designed such as to make these bowing effects negative.

However the ceramic fuelled fast reactors cannot rely on the thermal fuel expansion as inherent stability feature which lead to the Doppler coefficient becoming the center of attention. In fact the role of the Doppler effect is twofold: it terminates the first power peak of a fast excursion and decreases the total energy released thus providing the necessary time for shut-off systems to react; by the same token it helps to establish inherent operational stability. Second it strongly influences the energy release figures of Bethe-Tait calculations.

By now the theory is well understood and theoretical values agree well with those determined by Doppler measurements in critical facilities and SEFOR, using an entirely different approach to measure the Doppler coefficient by use of power excursions.

The sodium void coefficient, negative in the first reactors with metal fuels and small cores, became of major concern when it was discovered that it might be positive in a large reactor with UO_2/PuO_2 fuel. Both theoretical and experimental treatment was particularly difficult as the effect is governed by differences of major effects (absorption, spectrum hardening, and leakage) involving both space and energy dependency. The fast critical facility is the principal tool for the experimental investigation of the sodium void coefficient (Table V). With recent advances in the area of microscopic cross-sections, the determination of group constants and in particular calculational methods permitting three dimensional calculations (two space and one energy dimension) the theoretical and experimental treatment of the Na-void coefficient is now reasonably well in hand.

From the safety point of view the partial positive void coefficient plays a major role in the sequence of events following local incidents involving the voiding of zones having a positive contribution. The most likely incident leading to such voiding is the blockage of a subassembly followed by Na-boiling (superheat) and Na-ejection. Fuel failure and the possibility of subsequent propagation of such fuel failure have become of major concern and has lead in recent years to the extensive study of all hydraulic, mechanical, thermal and chemical phenomena involved in this sequence of events. While it is probably not possible to prove that such a sequence of events will under no circumstances lead to a major accident, the situation is further complicated by the difficulty in predicting with confidence the total energy release of a Bethe-Tait event and to prove the removability of the after meltdown decay heat of a large core following such a hypothetical accident. This leads to the concept of engineered safeguard measures which in the case of such local incidents are aimed at avoiding Na-ejection: instrumentation of each assembly, independent shut-off system, avoidance of superheat and avoidance of damage propagation by proper design. In such a context it is likely that work on post accident phenomena such as aerosols, sodium fires, equation of state, etc. will play a secondary role in future R&D.

Consistent with this approach of engineered safeguards is the new trend to use reliability data (failure rates) for the establishment of fault trees and statistical methods to evaluate the probability of a given accident. While the mathematical methods for handling the fault tree are well advanced the collection of input date is a major problem. The principal difficulty in this approach to safety and in particular fast reactor safety is the quantification of the risk in the framework of a system analysis approach.

In the reactor physics area future work will be directed towards improving and standardizing both data librairies and calculational methods. In particular fission product cross sections will be needed in view of the soft spectrum and high burnups of future reactors. As the present extent of physics calculations in particular the Na-void coefficient just fits the capability of todays computers, new and improved methods will be developped for reactor analysis and design as the new large 3rd generation computers are becoming available.

Fuel Element and Fuel Cycle

The fuel element is the central component of greatest concern. Here higher temperature and high burnup requirements create particular difficulties and the sodium and fast neutron environment and the use of plutonium require a whole new technology.

The first generation of fast reactors used metallic fuels which were abandoned when their limited burnup potential was realized. The success of oxide fuel for water reactors led to their acceptance for fast reactors, so much so that in the last 10 years the accent has been on the development of UO_2/PuO_2 fuels. Originally the principal concern in extending light water reactor fuel technology to burnups of 50 - 100,000 MWd/t was the swelling of highly irradiated fuel due to solid and in particular gaseous fission products. It is only in recent years, as the experimental testing of these fuel elements reached higher burnups, that a new phenomenon was observed - the swelling of the stainless steel which is used as cladding material and for the wrapper cans.

In fact there are three types of irradiation damage in structural materials :

- first, the well known lattice displacement by incident neutrons becoming increasingly severe with decreasing temperature and therefore called low temperature embrittlement; above about 400° C the annealing rate of this type of irradiation damage is sufficient to repair the lattice;
- second, the formation of the bubbles by (n,α) reactions. As only at temperatures above 500° C the He-atoms have sufficient mobility to

form large bubbles the phenomenon is called high temperature embrittlement leading to a reduction of applicable strains. As the high temperature embrittlem. becomes important only at fluences above $\sim 10^{22}$ nvt, this phenomenon has not received much attention in thermal reactor technology where fluences of $\sim 10^{21}$ nvt are known compared to $> 10^{23}$ nvt envisaged already for the prototype fast reactors.

- the third type is the void formation by vacancy condensation which has become obvious in the last few years at fluences above 10^{22} nvt reached in fast reactor irradiations, when swelling rates were observed that were larger than could be accounted for by the above He-formation. This phenomenon is both flux and temperature dependent and becomes noticeable at fluences of about 10^{23} nvt. It has been observed particularly in the temperature range 350 to 650° C and present theories predict a maximum at about 500 - 550° C. Local volume increases of up to 10 % have been observed.

Recent experiments and theoretical investigations have confirmed He-formation itself, i.e. the second phenomenon is interrelated with the formation of voids by vacancy condensation, which complicates the establishment of theoretical models although the individual phenomena involved are reasonably well understood. Fig. 1 and 2 summarize the most recent results for SS 304 at different temperatures together with reasonable working formulae. While swelling per se is one problem it is more difficult for the designer to cope with differential swelling due to the flux and temperature gradients in the core giving rise to fairly strong bowing effects. Indeed, it is to be expected that the higher flux of the reactors under construction and especially the future large plants will result in an increased swelling rate, but only the confirmation of models by further irradiation testing will permit confident predictions. The limitations of existing reactors compared to future plants present a particular difficulty in this area as illustrated in table VI.

On the other hand, it is known that for example certain impurities (noibium, carbon, etc.) and material treatment (cold work) have considerable influence on the swelling behaviour.

TABLE VI

VACANCY AND GAS GENERATION RATES IN SS-316

	EFFBR CORE A-	EBR-11	DFR	RAPSODIE (Fortissimo)	PHENIX PFR SNR	1000 Mwe
ACTOR POWER, Mwt	200	42	60	20 (40)	530(730)	2500
PORTED NEUTRON FLUX 10^{15} n/cm^2 sec	4.73	2.07	2.50	1.9 (3.0)	7.0(8.7)	11.0
eutron flux above 0.11 Mev 10^{15} n/cm^2 sec	3.51	1.77	2.22	1.6	4.1(4.8)	--
ACANCY PRODUCTION RATE n SS-316 x 10^{17} v/cm3 sec	2.46	1.34	1.68	1.3	2.8	4.5
IYDROGEN PRODUCTION RATE in SS-316 x 10^{11} p/cm3 sec	6.0	3.8	4.58	4.2	6.2	9.9
HELIUM PRODUCTION RATE in SS-316 x 10^{10} alpha/cm3 sec	2.55	1.60	1.90	1.9	2.9	8.9

APDA - communication

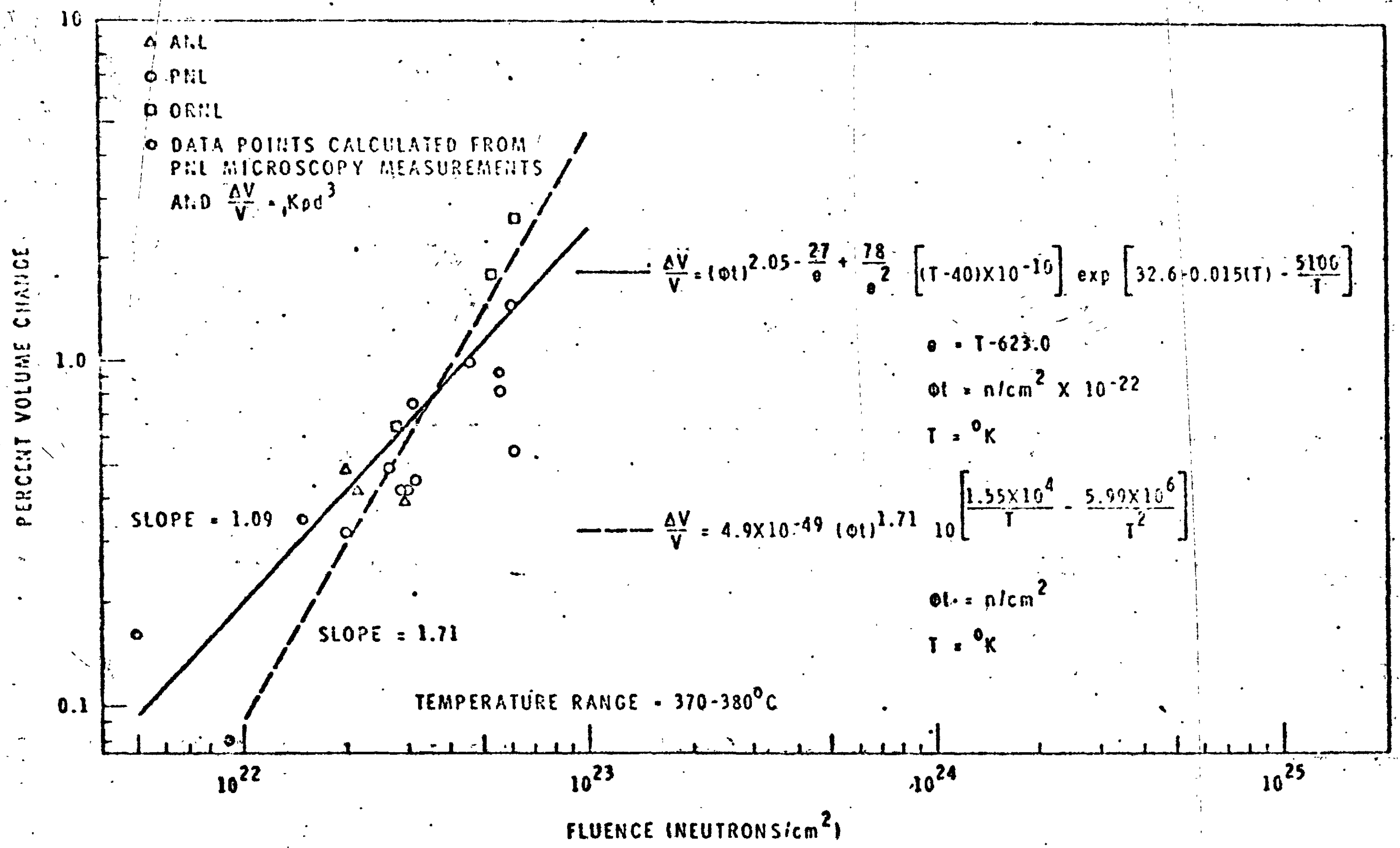

FIGURE 2. Swelling in AISI 304 SS

(WHAN-FR-15)

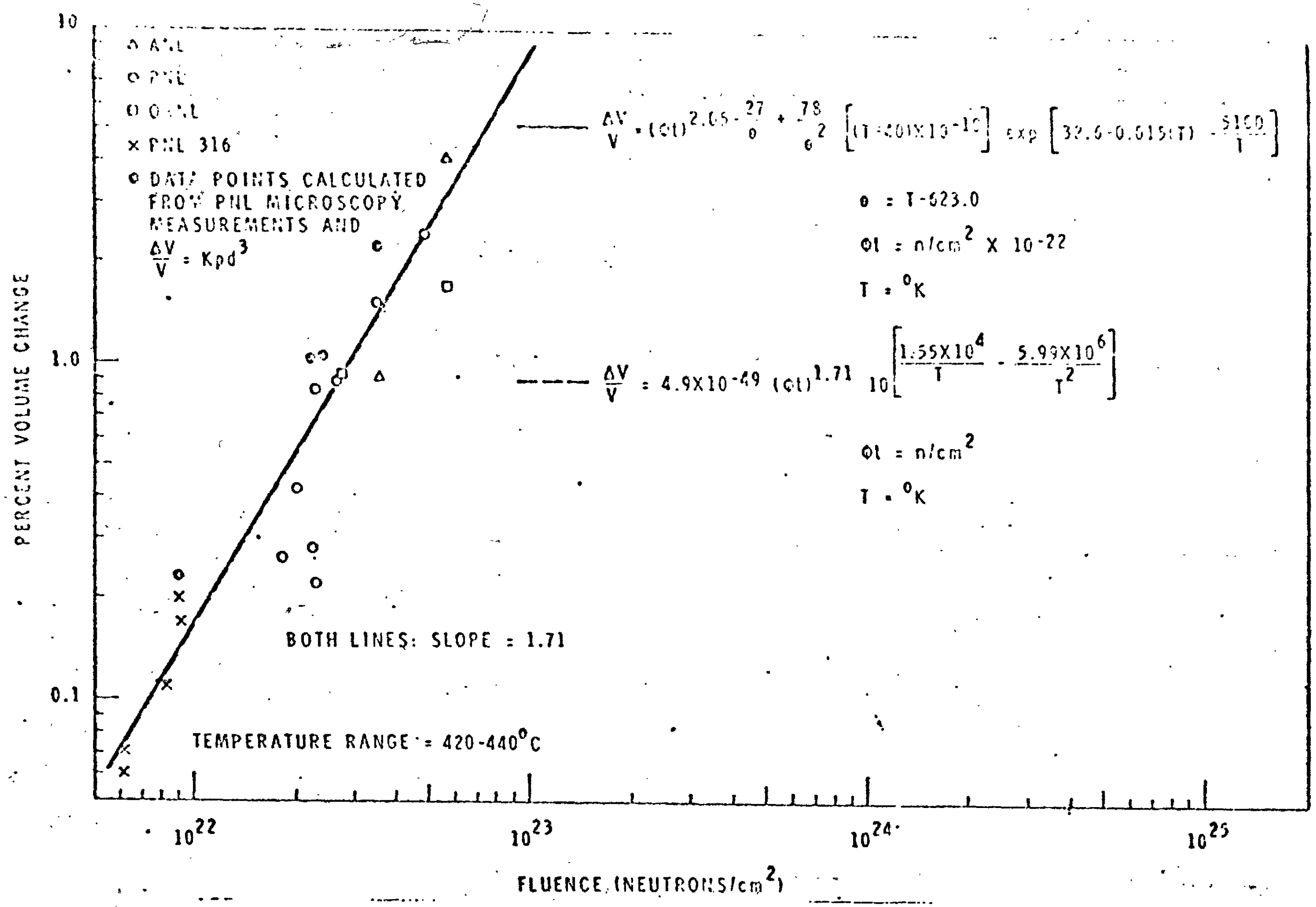

FIGURE 1. Swelling in AISI 304 SS
(WHAN-FR-15)

The other more general feature of the fast neutron irradiation damage is a reduction of the applicable creep-rupture strengths along with a general reduction of ductility as high temperature embrittlement comes into the picture. To complicate things further, the influence of radiation damage on ductility depends on the applied stress configurations.

The corrosion by Na poses no particular problem in fuel cladding as long as the oxygen contents can be kept reasonably low ($<$ 50ppm) by the purification systems. The most important limiting factor in fuel element design is the maximum hot spot cladding temperature (700° C in SS) leading to the study of vanadium base alloys for some advanced designs.

Apart from the development of cladding materials, future work will concentrate on the understanding of swelling mechanismus in fuel, the establishment of physical properties before, during, and after irradiation and on the development of advanced fuels.

While in former years the specifications of fuel pins were largely a matter of experience and judgement a more systematic approach has come to the forefront in recent years. Indeed, it has been possible to establish mathematical models which are constantly improved and refined as irradiation results are becoming available in increasing numbers, mainly from DFR, EBR II and Rapsodie.

In conclusion it can be said that on the basis of irradiation experience with UO_2/PuO_2 fuel, the fuel elements for the present prototype reactors can be and are being designed and indeed fabricated with sufficient confidence.

The emphasis is now on the development of high performance fuel; mostly still oxide bot some groups concentrate or carbide (UC/PuC), although early experience was not always encouraging. Development of the vented fuel element concept originally favoured mainly by the Italian group has largely been abandoned (with the exception of Gas Cooled Fast Reactors).

As the fuel cycle cost is becoming a key issue in the justification for initial acceptance of fast reactors it is clear that the industrial development of the entire fuel cycle covering fabrication, reprocessing, shipping, waste disposal, etc... has to coincide reasonably with the introduction of commercial plants. While recycle patterns for fast reactor fuel will be strongly influenced by the recycle technology for light water reactor fuels there are some major differences to be considered. LMFBR fuels will operate to high burnups at high specific powers. At the same time there will be a strong incentive to discharge fuel and recycle it as rapidly as possible, in order to reduce inventory charges, meaning short cooling times.

The short cooling time combined with high burnups and high specific power leads to radiation intensities, decay heat generation, and concentration of short lived fission products which are much larger than for thermal reactor fuels. In addition the high fissile plutonium concentration will result in stringent measures for critically control. Reprocessing of blanket fuel poses a separate problem giving rise to different options.

LMFBR fuels also will differ markedly in design from thermal reactor fuels. Mixed uranium and plutonium oxides clad in small-diameter stainless-stell tubing with an internal helium bond will be used initially. Considered for later use are mixed carbides, nitrides, and perhaps even (again) uranium-plutonium alloys. Advanced fuels may contain an internal sodium bond Cladding materials for these fuels have not been selected. The more exacting specifications require that considerable attention be given to quality assurance in fabricating both fuel and cladding material - in particular to the development of quality assurance methods and non destructive testing and inspection procedures.

In this overall development which is expected to take another 15 to 20 years, the major problems will be overcome by standard industrial procedures, that is their resolution will depend to a large extent on the ability of industry to organise and build large plants using advanced

T A B L E V I I

CHARACTERISTICS OF 1000 MWe REFERENCE DESIGN

		AI	BW	GE	W	C-E
Plant configuration						
Pool/loop		Loop	Pool	Pool	Loop	Loop
Hot cell/under plug refueling		Plug	Plug	Hot cell	Hot cell	Plug
Δ P Core (nozzle-nozzle)	kg/cm²	6.1	2.42	4.5	6.2	5.27
Δ T Core	°C	182	149	171	110	148
Core diameter	cm	260	392	466	417	348
Active core height	cm	109	83	76	100	61
Fuel material		Oxide	Oxide	Oxide	Carbide	Carbide
Core enrichment (inner/outer)	%	10.5/13.14	11.6 av.	11.05/13.94	15.9 av.	9.7/11.29
Fuel pin clad O.D.	mm	7.62	7.12	6.30	7.66	12.20
Clad thickness	mm	0.445	0.25	0.27	0.31	0.26
Fuel-clad bond gap	mm	0.13	Vibro-compacted	0.075	0.4	0.35
Pellet O.D.	mm	6.6	6.6	5.6 core / 6.15 5.7 axial		8.9
Fuel Bond		He	Na	He	Na	Na
Gas reservoir	cm	95.5	2.5	53.5	63.5	20.3
Over-all pin length	cm	281	168	209	240	203
Lineal power, Av.	W/cm	330	270	315	605	940
Lineal power, Max.	W/cm	525	525	585	1290	1425

T A B L E V I I (continued)

		AI	BW	GE	W	C-E
Temperature						
Core inlet	°C	415	406	432	410	475
Clad peak	°C	660	654	704	759	658
Fuel peak	°C	2310	2450	2870	1450	1430
Core outlet	°C	615	558	620	537	582
Breeding ratio		1.27	1.36	1.46	1.25	1.441
Doubling time	yrs	13	10.4	7.12	13.8	7.99
Sodium void worth (100% core voided)	% Δ k	-2.7	2.44	2.15	n.a.	1.36
Maximum reactivity (Central Na void)	% Δ k	+2.76	2.8	5.20	-0.12 with one module void	n.a.
Doppler coefficient	$-T \frac{dk}{dT} \times 10^{+3}$					
Na in		8.0	4.74	6.7	2.4	6.4
Na out		n.a.	1.76	4.1	n.a.	4.2

technology to the rigid quality assurance specifications required for reliability and safety. Therefore there is no need to expect any major road-blocks on the path to commercialization, with the possible exception of the fuel cycle (swelling, reprocessing) and the steam generator.

It can be seen that these possible long-range road-blocks are identical with the major unknowns surrounding the present prototype plants. One might therefore expect that the construction and operation of these plants will allow much more confident prediction of the long range problems to be faced on the way to commercialization. Table VII summarizes the main characteristics of the US 1000 MWe LMFBR designs (as of Dec. 1970).

The Gas-Cooled Fast Reactor does not necessarily require an entirely new technology. Indeed, a GCFR that could be introduced with a reasonable effort and on a time-scale not too far removed from that of the LMFBR would have to use rod type core geometry and would rely essentially on :

- LMFBR, technology in regard to fuel element development, including if possible a gas-loop in e.g. PEC, and
- HTGR technology for general layout, containment and components.

A major effort would be necessary in regard to safety as, in addition to most problems facing the LMFBR, the GCFR must still show that rapid depressurization is not credible. Until more experience is available for concrete pressure vessels under high internal pressure (100atm), this problem may retard its development. Assuming a leaktight concrete pressure vessel is available, the GCFR may well dispense with an outer pressure-tight containment. Fuel work would concentrate then on compatibility problems and a reasonable effort will be necessary to adapt HTGR experience in component and plant design in general to GCFR requirements.

The economics of the GCFR would on a long term basis benefit by taking advantage of the high temperature direct cycle and the coated particle. While much is being learned about coated particles for thermal reactors, a special coated particle would have to be developped for a GCFR including the support structure for service in the fast flux environment. There is little doubt that irradiation capacity, not now existing, will be necessary.

III. ECONOMICS CONSIDERATIONS

It is obvious that the principal motivations for the development of fast breeder reactors are the various economic considerations related to the fulfillment of energy needs on a long term basis, i.e., their development is aimed at commercial availibility before a system relying entirely on conventional plants including thermal reactors and converters would run into the problem of increasing fuel costs. While this critical period is slowly receding from 1990 to the year 2000 with the discovery of new resources it is unlikely that one can rely indefenitely on new discoveries. It is impossible to wait until the need arises considering the 20 - 30 year time span required to develop a new reactor line, which can be seen when looking at the timetable of fast reactor development.

Considering that systematic development started in the early 60's the following picture emerges :

- Commissioning of first prototypes in the 300 MWe range	1972/1976
- Commissioning of first large reactors still requiring public support	1976/1980
- Commissioning of first commercial plants	1985/1990

Without going in depth into the "resources" argument and all its implications it appears that this timetable would adequately respond to the needs.

Besides, a more sophisticated analysis of the parameters involved is impossible in the absence of all the economic boundary conditions. In addition there are political considerations, which are barely amenable to such a system analysis approach.

In fact, it appears that the fast reactor will - at least initially - have to defend its place in an established light water reactor market with low cost uranium ores available for another decade or two. It is partly for this reason that the projected cost of energy production from fast reactors has received particular attention in recent years. Cost benefit studies have been carried out in all countries with major fast reactor programs, some with a fixed set of assumption and others - perhaps more useful - keeping a number of variables, mainly:

- the evolution of the uranium ore cost;
- the time of introduction;
- the rate of penetration;
- the plutonium credit, etc.

The other principal reason for this exercise is the public scrunity to which most programs became exposed in recent years as it became obvious that most programs will cost in excess of one billion $ or even several times that (in the U.S) in public funds before the fast reactor will be commercial. The results of most of these studies are known and without going into details, or examining the validity of such calculations, it should suffice here to mention that all of them show considerable benefits for almost all sets of reasonable assumptions.

Obviously the principal difficulty in such calculations is the establishment of reliable cost figures for both capital and fuel cycle cost. Recent developments in the evolution of light water reactor capital costs have shown how futile any such projection might be for an already established reactor line not to mention fast reactors where the commercial phase is still 10 -15 years in the future. Future developments in the field of safety and in regard to environmental aspects might still further complicate the matter.

For this reason, most of these cost benefit studies do, in fact, not answer the essential and immediate question, i.e., how does the fast reactor present itself compared to its most immediate competitor : the light water reactor. In order to answer this question and to provide an additional element of judgement, some recent studies do not so much concentrate on calculating the benefits for a given set of assumption but on analysing the actual cost structure of the FBR compared to, e.g., the LWR. From these studies the following picture emerges :

Capital Cost

Most estimates agree that - at least initially - the capital cost of the LMFBR will exceed that of a LWR. The figures quoted for this differential vary from an optimistic 5 % to more than 20 %. Considering such a large difference one might easily conclude : why not 50 % and how would that affect the future of the LMFBR? The question can be answered by looking at the details of some of these estimates.

Atomics International published in 1969 a set of comparative figures for a hypothetical 1000 MWe plant given in table VIII.

Assuming that items 1 and 2 account for the principal differences between the two reactor types and thereby contain the largest uncertainty it can be seen that the estimate allows for the reactor plant of a LMFBR to be 1.3 times more expensive than the equivalent of a LWR. In fact, referring to thermal power, the actual NSSS item 1) allows for almost double the specific cost (20 $/KWt compared to 11 $/KWt). In order to provide a further element of judgement A.I. compiles a comparative list of items influencing these costs given in table IX.

TABLE VIII

CAPITAL COST OF TYPICAL 1000 MWE PLANT

COMPONENT	LMFBR		LWR	
	$/kWe	% of total	$ kWe	% of total
1. Nuclear steem supply system	48($20/kWt)	23.7	35($11/kWt)	18.2
2. Balance of nuclear plant*	35	17.3	29	15.1
3. Turbo generator equipment and installation	42	20.8	55	28.7
4. Structures	12	5.9	10	5.2
5. Electrical accessory	8	4.0	8	4.2
Total direct construction cost	145	71.7	137	71.4
6. Indirect construction cost	21	10.4	20	10.4
7. Owner's contingency	10	5.0	10	5.2
8. Interest during construction	26	12.9	25	13.0
Grand total	202	100	192	100

* includes containment and engineered safeguards;

TABLE IX

FACTORS INFLUENCING CAPITAL COST

	LMFBR	LWR
Thermal power, MWt/MWe	2.4	3.1
Heat rejected, MWt/MWe	1.4	2.1
Containment building pressure, kg/cm2	0.7	3.5
Reactor vessel and internals, kg/MWe	320	900
Primary piping, kg/cm	3	12
Control rods, numbers per 1000 MWe	15	85
Heat exchanges surface, m2/MWe	21	186
Reactor coolant pumping power/MWe	0,03*	0,02

* includes primary and secondary pumping power.

.../...

In a similar study the following figures were published by the SNR-Consortium in December 1970 (table X), also for a typical 1000 MWe plant.

TABLE X

CAPITAL COST OF TYPICAL 1000 MWE PLANT

COMPONENT	LMFBR		LWR	
	UC /KWe*	%	UC/KWe *	%
1. Nuclear system	67	32.1	43.4***	25.0
2. Steam system	34.5	16.5	36.4	21.0
3. Electrical plant	30	13.6	26.1	15.0
4. Conventional construction	28.3**	12.3	26.9	15.5
5. Engineering	12.8	6.1	10.6	6.1
6. Owner costs and interest during construction	36.1	17.4	30.3	17.4
Total	208.7	100	173.7	100

* assumed 3.60 DM = 1 UC

** includes concrete containment building.

*** includes steel containment.

.../...

This comparison uses a somewhat different breakdown combining all elements typical for fast reactors containing the largest uncertainties in item 1, (eliminating even the concrete containment for which estimates may be considered more reliable). Allowing thus a cost differential of over 55 % for the nuclear part, the total capital cost differential is 20 % or ~ 35 UC/KWe. Even if one would allow for a 100 % cost differential for the nuclear system of a LMFBR, i.e., ~ 85 UC/KWe the total capital cost would exceed that of the LWR by less than 30 %. It can easily be seen that a cost differential of about 25 % might be retained as an upper limit, a figure which at this point does not depend on the precision of the total estimate.

On the other hand, it should be mentioned here that any future development in the direction of larger plants works in favour of the LMFBR which, due to its low-pressure system, has a larger extrapolation potential than the LWR.

Fuel Cycle Cost

The same SNR-Consortium study calculates for the reference cases given in table X fuel cycle costs of a 0,66 mills/KWh for the LMFBR and 1.7 mills/KWh for the LWR using the input date listed in table XI.

It then analyses the actual cost structure of the fuel cycle cost and its sensitivity to various changes in the various input data assuming an 85 % load factor (expecting LMFBR's to be used for base load exclusively).

The results were summarized in table XII.

It can be seen that only the burnup and fabrication costs have a major effect on the fuel cycle cost, whereas the effect of changes in all other contributions is relatively minor. One should add that fuel cycle costs for carbide fuels are expected to be 0.14 to 0.28 mills/KWh lower than those given here due to the higher breeding gain.

TABLE XI

REFERENCE DATA FOR SNR-CONSORTIUM STUDY*

	LMFBR	LWR
Annuity, %/a	12	12
O & M cost, UC/kWe.a	2.8	2.8
Efficiency, %	42	33
FE fabrication, UC/kgHM		
LWR		78
FBR core (inclusive Pu-conversion) core + axial blanket rad. blanket	330 220 83	
Reprocessing, UC/kgHM	56	56
Average burnup, MWtd/tHM (core only)	100,000	30,000
Specific power, MWt/tHM (core only)	115	33
Pu-generation, kg/MWe.a	0.294	0.213
Pu-inventory, kg/MWe (Pu-fissile including 5% stock but no excore inventory)	2.98	

Natural uranium cost: 8 UC/lbU_3O_8

Separative work : 28.7 UC/kg S.W

Plutonium worth : 5 UC/g fissile

Plant factor : 80 %

* Costs converted 3.60 DM = 1 UC

T A B L E   X I I

FBR – Fuel Cycle Costs *

(varying one parameter at a time)

COST FACTORS	Cost contributions (mills/KWh)						
	Reference case (Pu-price 5 UC/g	Pu-price 10 UC/g	Burnup -33%	Inventory -33%	Out of core time + 100%	Fabrication cost core + 100%	Transportation and reprocessing cost + 100%
FE – fabrication	0.469	0.469	0.658	0.469	0.478	0.858	0.469
FE – transportation and reprocessing	0.094	0.094	0.147	0.094	0.089	0.094	0.189
Pu – inventory	0.267	0.534	0.253	0.178	0.267	0.267	0.267
Fuel consumption	0.172	0.345	0.142	0.117	0.122	0.172	0.172
Depleted U – cost	0.003	0.003	0.003	0.003	0.003	0.003	0.003
Total fuel cycle cost	0.661	0.755	0.920	0.627	0.715	1.050	0.756
Difference from reference case	–	0.094	0.259	-0.034	0.044	0.389	0.095

* Converted 3.60 DM = 1 UC

Total Generating Cost

Using again a plant factor of 80 % the following total generating cost picture emerges for the above reference case :

	Cost (mills/KWh)	
	LMFBR	LWR
Capital	3.58	2.97
Fuel Cycle	0.66	1.69
O & M	0.39	0.39
Total	4.63	5.05

This demonstrates that for the case presented here the LMFBR could have excess capital costs of 60 UC/KWe or 35 % as compared to the LWR before total generating costs would be equal. Or, going back to table X, that the cost estimates for item 1, the nuclear system, could be in error by 26 UC/KWe or about 40 % before the cost advantage could be lost. At this point the cost allowance for the LMFBR nuclear system (item 1) would be 2.15 times that for the LWR a fact which would be difficult to explain. On the other hand, assuming an upper limit ofa 25% for the capital cost penalty of the LMFBR as reasonable, the situation could easily absorb either a burnup penalty of 33 %, or most of an increase in fabrication cost by 100 %, or part of both.

Again when considering these uncertainties one should also keep in mind the possible improvements both in regard to capital cost due to the larger size potential and the fuel cycle cost due to possibly carbide fuel.

Indeed, in regard to capital costs there is reason to believe that on the long run there will be little difference between a LWR and fast breeder reactors.

The GCFR may very well present an even more favorable picture although in the more distant future.

XVII/341/2/71-E

COMMISSION
OF THE
EUROPEAN COMMUNITIES

SECOND ILLUSTRATIVE NUCLEAR PROGRAMME
FOR THE COMMUNITY

ANNEX VI
HIGH-TEMPERATURE REACTORS

DRAFT
(1 March 1972)

High-temperature gas reactors

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HIGH-TEMPERATURE GAS REACTORS

I. INTRODUCTION

The development of the high-temperature gas reactor has been based on characteristics constituting refinements of the features of other gas-cooled reactors:

1. By integration of the fissile and fertile components of the fuel in the moderator material (in this case graphite), it is possible to use a much larger fraction of the core to produce heat. For this reason, high thermal power densities can be used, resulting in compact reactor configurations and pressure vessels.

2. The use of helium as the coolant enables considerably higher outlet temperatures to be reached than would have been possible with other cooling media. These temperatures make it possible to utilize indirect high-efficiency steam cycles, which thus have less thermal effect on the environment, by means of compact steam generators and modern turbines. The helium coolant can also be used as the working fluid in gas-turbine energy conversion systems or as a heat source for high-temperature industrial processes.

3. Metal fuel canning is eliminated and the graphite is used both as structural material and for the cladding of the fuel for the confinement of radioactive fission products; this provides good neutron economy and enables high temperatures to be reached.

In the original conception of the HTR, the core consisted of a very homogeneous assembly comprising only fuel elements of low-permeability graphite (with a vent circuit for prismatic elements), containing fuel compacts of a mixture of powdered graphite and uranium and thorium carbides. Today, all fuel elements are based on the use of coated-particle fuels, in which the oxides or carbides of the fissile and fertile materials take the form of small spheres known as kernels, each coated with layers of pyrocarbon, sometimes combined with a layer of silicon carbide (the diameter of these coated particles is of the order of one mm). These fuel elements may be either prismatic or spherical in form.

Steel pressure vessels were used in the first experimental gas-cooled high-temperature reactors (AVR, Dragon, Peach Bottom), but prestressed concrete pressure vessels have been adopted for power reactors. Apart from their intrinsic operational safety, the use of these vessels allows large reactors to be run at considerably higher helium pressures, of up to 50-60 atm. This high pressure, combined with the adoption of a number of cooling loops in parallel, helps to keep down the size of the steam generators and circulators so that they may be incorporated inside, or in the walls of, the concrete vessel.

These reactors are suitable for U/Th (93%-enriched U) or U/Pu (approx. 5%-enriched U) fuel cycles. Each of these two cycles makes better use of natural resources than the fuel cycle of the light water reactors at present available, and should enable electricity to be generated at relatively low cost (see Section VII).

Characteristics of HTRs

	PRISMATIC ELEMENTS					SPHERES	
	UK		USA			W. Germany	
	Dragon	Oldbury	Peach Bottom	Fort St. Vain	Tender 1100	AVR	THTR
Type	experim.	master model	experim.	prototype	commerc.	exper.	prototype
Site	Winfrith	Oldbury	P. Bottom	Fort.S.Vain	-	Jülich	Schmeh.
Main contractor	Dragon	-	GGA	GGA	GGA	BBK	HRB
Fuel element manufacturer	Dragon	UKAEA	GGA	GGA	GGA	Nukem	Nukem
Customer	OECD	CEGB	Philadelphia El.	Pub.Serv. Colorado	Philadelphia El.	AVR	HKG
Commencement of construction	1960	(72)	1962	1968	(74)	61	1971
Criticality	1964	(77)	1966	1972	(79)	66	1976
Thermal power (MWth)	20/25	approx.4550	155	842	2804	45	750
Net electrical power (MWe)	-	2 x 909	40	330	1100	15	307.5
Net efficiency (%)	-	40	34.6	39.2	39.0	33.3	40.5
He circuit							
Pressure (atm)	20	44.8 bar/abs	22.8	47.6	46.5	10	40
Inlet temperature (°C)	335	288	343	400	399	175	26.2
Outlet temperature (°C)	850	729	750	775	765	850	750
Steam circuit							
Live steam (atm/°C)	-	159.6 bar/~~538~~	100/538	175/538	175/510	75/505	181/530
Reheat steam (atm/°C)	-	37.4 bar/538	-	42/538	42/538	-	47.2/530
Core							
Effective diameter (m)	1.08	-	2.8	5.95	8.3	3	5.6
Active height (m)	1.60		2.3	4.75	6.3	3	5.8
Power density (MW/m³)	14	8.3	8.3	6.3	8.2	2.2	6
Fuel cycle	misc.	Low-	U-Th	U-Th	U-Th	U-Th	U-Th
Average burnup MWd/tonne)	misc.	enriched U	60,000	100,000	92,000	80,000	135,000
Fuel							
Element geometry	long tube	Block with 24 tubes	cylinder	hexagonal prism	hexagonal prism	sphere	sphere

Characteristics of HTRs (cont.)

	PRISMATIC ELEMENTS					SPHERES	
	UK		USA			W. Germany	
	Dragon	Oldbury	Peach Bottom	Fort St. Vain	Tender 1100	AVR	THTR
Diameter (mm)	misc.	65/2/30.1	89	-	-	60	60
Width across flats (cm)	mics.	50	-	36.0	36.0	-	-
Height (cm)	mics.	100(2x50 for tubes)	366	79.2	79.2	-	-
Lifetime (years)	misc.		3	6	4	2-3	3.5
<u>Geometry</u>							
Initial charge (kg)	U 14		Th 1450	Th 19,500	Th approx. 40,700	Th 150	
	Th 26		U 220	U 870	U approx.2000	U 30	
Number of fuel elements	37	5 x 235	804	6 x 247 = 1482	8x475 = 3800	100,000	675,000
Pressure vessel	steel	concrete	steel	concrete	concrete	steel	concrete

II. DEVELOPMENT OF EXPERIMENTAL REACTORS

1. AVR reactor (Germany, Ref. 1)

The German HTR development programme began in 1956. The European Atomic Energy Community (Euratom) was associated with it from 1963 to 1968 under the THTR agreement.

The completion of the first important stage of the German programme was marked by the construction of the AVR (Arbeitsgemeinschaft Versuchsreaktor) experimental reactor (see Table 1).

The main objective of the construction of the AVR experimental nuclear power station was:

(a) to demonstrate the safety and reliability of a high-temperature reactor with movable fuel elements;

(b) to gain experience in construction and operation which would be of value for future development.

The AVR went critical in August 1966. After successful criticality and zero-power tests, in air and with helium, running up began in September 1967. The intrinsic safety of the AVR reactor was demonstrated by several sets of experiments under transient conditions.

The full power of 15 MWe was reached for the first time on 15 February 1968. On this occasion the temperature of the gas coolant reached a maximum of 870°C, this being the highest coolant temperature ever attained in a nuclear power station. In September 1970, the behaviour of the reactor at maximum power with the two circulators shut down (i.e., heat evacuation being interrupted) was studied, with all the shutdown rods fully extracted. The experiment showed that the reactor returned to criticality after 23.5 hours and that the reactor power did not exceed 1.8 MWth, or 4.1% of the original power.

The reliability of the AVR reactor was proved by an availability of 71% in 1969 and 84.5% in 1970. Up to 1 March 1971, 63,235 fuel elements had been added to the core. Burnup of the most exhausted fuel elements exceeds 124,000 MWd/tonne of heavy metal, or 13.5% fima. The activity of the coolant at present is approximately 10^{-1} Ci/m^3 STP, or about 200 Ci in the complete gas circuit. The activity is primarily due to inert gases. No other fission product has hitherto been detected in the circuit. Most of the accumulation of inert gases is probably attributable to contamination by the uranium of the coated particles during the fabrication process. The radioactivity due to the initial uranium charge has declined progressively during operation and is now levelling out. It may therefore be confidently assumed that with the arrival of new batches of fuel elements, the value will continue to fall, especially as most of the elements now in the AVR reactor were fabricated five years ago. Nevertheless, even with these elements, the cumulative activity is so low that a total loss of coolant through the ventilation stack would not exceed the safety standards, even in a larger station.

2. Dragon reactor

It is often difficult to predict the operational behaviour of large power stations from technical office designs, and the main value of experience is to establish the practical limits which can be reached. In this context, Dragon has demonstrated the perfect validity of the gas-cooled HTR concept and made it possible to lay down criteria for power reactor projects.

The agreement which launched Project Dragon under the auspices of the ENEA came into force on 1 April 1959 between Austria, Denmark, Euratom, Norway, Sweden, Switzerland and the UK, the aim being to design and construct an experimental gas-cooled high-temperature reactor at Winfrith Heath (UKAEA research centre).

Design and construction work began as soon as the agreement came into force, and the reactor went critical in August 1964, full power (20 MWth) being reached in April 1966.

The original form of the Dragon fuel element comprised seven hexagonal graphite tubes suspended from a reinforced graphite block which also served to lift the element. Today, the fuel takes the form of uranium oxide particles coated with pyrolytic carbon and silicon carbide (diameter 1 mm) compacted in cylindrical form in a graphite matrix. The length of the part containing the fuel is 160 cm. The helium coolant gas circulates upwards.

These early types of fuel elements consisted of seven identical vented rods. More recent types have only one central vented rod, the six outer rods, which can be detached from the central rod, not being vented. Using elements of this type, it is possible to irradiate experimental fuels for long periods, and the makeup fuel can be charged at intervals of approximately 300 days.

Because of the growing interest in power reactors of this type, Dragon was induced to construct a number of special fuel elements to irradiate, for example, the spheres of the THTR reactor, tubes of the "teledial" or tubular interacting types, etc., and, more recently, block-type elements.

At first the U/Th cycle was studied in the Dragon reactor, but by virtue of the flexibility of this reactor it was possible gradually to shift the emphasis to the study of the low-enriched uranium cycle.

Since the reactor reached full power, there have been four main periods in Dragon's operating history. Charge I featured a core used for the study of the thorium cycle with 10 experimental elements located in the central region and surrounded by the 27 makeup elements. In Charge II, the experimental fuel was distributed in the central rods of each of the 37 elements, the outer rods being the makeup fuel containing 93%-enriched U^{235}.

With the first two charges, the thermal flux increased as the makeup fuel was burnt, in order to maintain constant power. In consequence, the temperature of the elements containing fertile material rose. Charge III was designed to minimize this temperature rise by the introduction of a medium-enriched makeup fuel element with the production of plutonium. The cycles of this charge were reduced to about 50 days.

Charge IV featured the use of highly enriched uranium in the makeup fuel, to allow the testing of the low-enriched fuels proposed for power reactors. Cycles of 90-100 days were introduced to allow maximum utilization of the reactor. The total activity of the primary circuit is very low, less than 1 curie, consisting primarily of noble gases. This shows that the coated particles are capable of retaining a very high proportion of the fission products.

Over more than 1000 days' operation, the Dragon reactor worked exceptionally well. The only serious problem was corrosion of the primary heat exchangers on the water side; this problem was overcome by more stringent control of the pH of the water.

3. Peach Bottom reactor (USA, Refs. 1 and 2)

The first American project in this field was a low-power (40 MWe) reactor, built by General Atomic for the Philadelphia Electric Company.

The core was charged and the low-power tests carried out in 1966, but the power operation phase did not begin until early 1967, because the heat exchangers had to be replaced owing to corrosion under stress caused by chlorides. Normal operation commenced on 1 June 1967. The operating programme gave the following results:

	availability	circuit activity.
Period 0-150 equivalent full power days	77.3%	3 Ci
Period 150-300 " " "	83.3%	27 Ci
Period 300-400 " " "	85.4%	278 Ci at end of period

The increase in activity of the primary circuit (still low when compared with the specification of 4225 Ci, and having no effect on the operation of the reactor) was due, in the first core, to breakages of the graphite tubes (78 in all) containing the fuel compacts. These breakages were due to swelling of the compacts. The coated particles used had somewhat primitive coatings, of anisotropic structure, which fractured under the influence of irradiation. The coatings then became highly deformed, resulting in swelling of the compacts. When the defective elements were discharged, one of them fell into the core and jammed in it. However, the difficult operation of releasing and removing the element was carried out without additional damage to the core and without excessive exposure of personnel.

The second core, using industrial-grade coated particles and containing 17 experimental elements, was charged in June 1970 and power operation was resumed on 14 July 1970; since then, Charge 2 has operated for more than one-third of the 900 days laid down in the specification, resulting in primary circuit activity of well below 0.4 Ci. The reactor availability was 93%.

III. DEVELOPMENT OF PROTOTYPE REACTORS

1. The Fort St. Vrain reactor (USA, Refs 1, 3, 4 and 7)

The Fort St. Vrain reactor incorporates a number of characteristics not featuring in stations previously built in the United States, in particular: the use of a prestressed concrete pressure vessel, steam-driven axial circulators with water-lubricated bearings, and hexagonal block fuel elements. Construction began in 1968 and completion is scheduled for 1972 (see Table 1). Charging of the fuel elements is to be completed and the reactor is to go critical in 1972, so that commercial operation by the Public Service Company of Colorado can commence in the same year. The R&D programmes may be summarized as follows:

1. Development of helical-bundle heat exchangers of the once-through type by analyses and tests of helium flow, pressure drops, heat transfer, vibration of the tubes and bundles and boiling stability. The final stage consisted of a systematic series of tests of a complete module for heat transfer and vibrations, at GGA's plant at San Diego.

2. The circulators first underwent tests to confirm the validity of the water-lubricated bearing concept and were then subjected to a systematic series of tests on a prototype. Each production circulator has operated for at least 100 hours.

3. An extensive programme for the development of the prestressed concrete pressure vessel included the establishment of codes, and the study of the materials used and different configurations; the programme also incorporated basic work on concrete and on models carried out on a national basis under an ORNL development programme.

4. Other components were also subjected to thoroughgoing tests: the control rods and their mechanisms, the charge machine, the thermal shield, the reflector and the core support.

5. The fuel elements were developed and tested under the programme described in Ref. 7. The main emphasis of the work was the development and irradiation of two types of coated particles, BISO and TRISO, and fuel pins for charging in graphite blocks. A section of the complete fuel element was introduced into the Peach Bottom reactor as early as mid-1970.

2. <u>THTR reactor</u> (Germany, Refs. 1, 3 and 4)

Following the successful operation of the AVR reactor, the Federal Government decided to continue to subsidise, with the status of a programme of particular value, the construction of a 300 MWe thorium cycle pebble-bed reactor (THTR), as a first step in the marketing of the HTR system in West Germany. The design of this station is a result of the work carried out from 1963 to 1968 by the THTR Association, in which the Community participated. The power level of the reactor was chosen so as to permit extrapolation to 600 and 1200 MWe stations.

This reactor is under construction at Schmehausen (Westphalia) by the consortium of Brown Boveri, Hochtemperatur Reaktorbau GmbH, replacing the Brown Boveri Krupp Company (in which Krupp recently sold its shares to BBC and HKG), and Nukem, on behalf of Hochtemperatur-Kernkraftwerk GmbH (HKG), formed by six German electricity producers. Construction began in 1971 and is scheduled for completion in 1974; commercial operation is planned to commence in May 1976. The contract was signed at the end of October 1971.

The HKG company comprises the following six associates:

1. Gemeinschaftskraftwerk Weser GmbH
2. Kommunales Elektrizitätswerk Mark AG
3. Vereinigte Elektrizitätswerke Westfalen AG (VEW)
4. Gemeinschaft Hattingen GmbH
5. Stadtwerke Aachen AG, Aachen
6. Stadtwerke Bremen AG, Bremen

VEW was also responsible for the initiative of proposing the setting-up of a "Euro-HKG" company with the aim of the acquisition and sharing of technical and economic knowhow in the field of high-temperature reactors and exchanges of staff for training purposes. This new company was formed on 13 December 1971 in Essen by the CEGB, EdF, HKG and RWE. ENEL reserves the right to join at a later stage. This agreement is bound to make a positive contribution to the rationalization of decisions by Europe's electricity producers on the use of HTRs.

IV. THE PRESENT SITUATION OF STEAM-CYCLE POWER REACTORS

1. In the United States (USA, Refs. 1, 5 and 6), Gulf General Atomic in September 1971 received a letter of intent from the Philadelphia Electric Company regarding an order for two power plants each of 1160 MWe net, for commissioning in 1979 and 1981 respectively (see Table 1), and in December 1971

an order for two 770 MWe reactors, to be available in 1979 and 1982, for the Delmarva Power and Light Company. The interest of the American electricity companies (and this argument also applies to Europe) is justified in particular by the smaller thermal effect of HTRs on the environment. Heat disposal is a serious problem in connection with the forecast expansion of electricity generation for the future. For a given electrical power generated, the heat to be discharged varies in the proportion of 1-n/n (where n is the thermal efficiency); this factor increases sharply as n falls. For example, stations with an efficiency of 33% release one-third more heat than a modern superheat plant (40% efficiency). This is important when dry air cooling of a steam plant has to be employed because heat rejection to air from condensing steam is inherently more expensive than once-through water cooling.

The experience gained in the operation of the Peach Bottom reactor and the design and construction of the Fort St. Vrain reactor, together with all the accompanying licensing procedures, has enabled GGA to design and offer high-power reactors of around 1100 MWe. The use of a secondary containment and auxiliary cooling loops will facilitate the discussions with the licensing authorities. Much of the technology developed for the Fort St. Vrain reactor is applicable directly to the 1100 MWe reactor. However, supplementary programmes have been under way since 1968. Most of the development work for this project has now been completed. The remaining work relates primarily to component proof tests, long-term metallurgical tests and various demonstrations of the validity of design improvements (USA, Ref. 1). The pod-type PCRV has already been tested by GGA on a 1:20 scale model. A machine for the assembly of the circumferential prestressing tendons has been designed, built and tested in preparation for use in a commercial project.

A prototype of the control rod drive mechanisms was constructed and thoroughly tested in various operating modes, although the design closely resembles that of the Fort St. Vrain reactor. Long-term proof tests in helium began in 1971 in the installation built by GGA. Tests of the pressure vessel liner, the thermal insulation and the materials used for the heat exchangers have been completed or are in progress.

2. In Germany, the first objective relates to the development of the HTR plus steam generator type of station (indirect cycle). The intention is to use the experience gained during the construction of the THTR together with the experience of foreign companies on a basis of international cooperation and to embark on the construction of additional demonstration stations on this basis about 1975.

With these reactors, the commercial phase of the HTR will have begun.

3. In the UK, in response to an invitation to tender by the CEGB, the two consortia TNPG and BNDC have submitted a preliminary bid accompanied by a proposal for a £32 million two-year research programme for a low-enriched prismatic element reactor with a power of 750 MWe or more (UK Ref. 1). Some technical details of the characteristics of the station proposed by BNDC (UK, Ref. 2) have been published, and are included in Table 1.

The whole of British nuclear policy is now in the process of revision, with a view to defining the respective priorities for the AGR (Mark II), the HTR (Mark III), the SGHWR, fast reactors and light water reactors.

The UKAEA has embarked upon a large-scale research programme (approximately 24 million u.a. over three years), covering in particular industrial development of the fuel, to be tested mainly in the Dragon reactor (UK, Ref. 5).

4. In France, until very recently, the programme was based on the natural uranium/graphite/carbon dioxide system (installed power approximately 2500 MWe with eight reactors) (France, Ref. 1). In view of the present energy situation, the Government recently decided on the principle of speeding up the construction of LWR power stations (8000 MW programme for the Sixth Plan (1971-75)).

In spite of the highly promising outlook for breeders, France is devoting increasing attention to the HTR, whose technique is an extension of that of the graphite/gas family. Efforts are being concentrated on overcoming the specific technical problems of the HTR.

An association was formed in 1970 for the construction of power stations. Continuation of a development programme will depend on the results of studies in hand, the international development prospects of the system, and the part that France could play in this development (France, Ref.1). If the studies in progress confirm the hopes placed in this type of reactor, the launching of a preliminary project might be decided upon, followed by a construction project, probably in cooperation with foreign partners, in accordance with the recommendations of the Commission Consultative pour la Production d'Electricité d'origine nucléaire (Commission PEON).

V. DEVELOPMENT POTENTIAL OF THE SYSTEM

1. In Germany, the future development envisaged for the HTR system was recently defined (Germany, Ref. 1). The HTR possesses the valuable property of generating nuclear heat at high temperature, and this is to be exploited in two directions:

1.1 Adoption of the direct cycle

The possibility of developing a high-temperature reactor directly linked to a gas turbine (direct cycle station) is to be studied in greater detail, to permit the construction and operation of an experimental station and the starting up of a demonstration station, leading to the commencement of construction of the first commercial direct cycle stations in 1980.

The additional advantages of using a helium turbine are:

1. Lower specific capital cost.
2. Lower cooling water requirements.
3. Higher efficiency even in part-load operation.

These advantages may place the helium turbine HTR in a favourable position in the field of energy production.

The following studies and projects are subsidized by the Federal Government under a cooperation contract between German private industry and the KFA:

1. Fuel and graphite development programmes for high temperatures (up to 1000°C at core outlet).

2. Construction of in-helium test benches, including a test loop in which a fluid flow machine on the scale required for a 300 MWe plant circulates 220 kg/s of helium at a maximum temperature of 1000°C for tests on insulating and structural materials, valves, etc.

3. Construction of an experimental nuclear station with gas turbine; the type of reactor has not yet been finally settled (spherical or block type fuel elements).

4. Preparation of complete construction documents and a tender for a 600 MWe station.

The development programme will be completed in 1977.

Study work for the 600 MWe station project is based on the construction of a single horizontal shaft gas turbine. Single-shaft construction will probably have the advantage of lower turbogenerator set costs and more favourable behaviour of the control system.

In view of the large number of constructional and layout possibilities for the components of a gas turbine HTR, systematic studies will be necessary in order to select the optimum configuration.

On the basis of an analysis of all the possible configurations of the station, the most favourable solutions have been studied from the point of view of availability and economic viability.

The cooling water requirement is much less than with steam turbines. The use of dry cooling towers is more profitable than with steam turbines, since the heat is evacuated at 80-100°C instead of approximately 25°C, thus substantially reducing the cooling area needed.

1.2 Process-heat reactors

The possibility of producing cheap nuclear heat at temperatures in excess of 900°C suggests that this form of energy could be used on an industrial scale for purposes other than the generation of electricity. Several analyses at the Jülich Nuclear Research Centre have shown that nuclear heat can be used for the conversion of fossil raw material into refined products by known processes, in particular the gasification of lignite and coal by a process of hydrogenation. It is probably also possible to use nuclear heat for the current hydrocracking processes. The simplest form of application, however, appears to be the conversion of natural gas into hydrogen and carbon monoxide in tube-still heaters (steam reforming).

The products of these processes can be used on an industrial scale for different technical applications. For example, the gasification of lignite will probably be advantageous. The hydrogen produced by this process can be used for steel production by direct reduction of iron ore. Furthermore, there is at present, and will be in the future, a big market for hydrogen. The products obtained can also be converted into a number of basic chemicals by known processes.

It is difficult to describe all the possible applications at this stage, but the studies have shown that there is a market for the forms of application described above, in addition to the use of atomic energy to generate electricity, and this market should not be ignored, since it represents an energy demand amounting to a third or half of the amount of primary energy used for electricity generation.

It must, however, be noted that the development of this source of heat requires the concomitant development of exchangers or cracking tubes able to meet nuclear safety requirements and withstand chemical reactions. It seems that, so far as the reactor is concerned, the only new problem which will arise in connection with the techniques to be developed for the direct cycle will relate to the presence of larger quantities of hydrogen in the primary circuit, as a result of diffusion through the exchanger walls from the secondary circuit.

2. <u>In the United States</u>, the firm of Gulf General Atomic in 1971 was awarded a study contract by the State of Oklahoma relating to the application of the HTR system for the gasification of coal, which would require helium temperatures at the reactor outlet in the range 870-1070°C. GGA is also interested in the development of direct cycle HTRs with air cooling towers and helium-cooled fast reactors (see Annex on fast reactors).

3. <u>Japan</u>

Interest centres primarily on the ability of this type of reactor to supply high-temperature heat at low cost, which might permit the development of new steel-making techniques.

The electricity producers are also interested in the HTR system. The JAERI (Japanese Atomic Energy Research Institute) is at present studying a 40-50 MWth experimental reactor project, on which construction work might begin in 1972. The original feature of this reactor, to be used to study a range of applications, is that it is designed to reach a helium temperature of 1000°C (possibly 1200°C at a later stage).

4. France

Although the present French HTR effort is concentrated on the indirect cycle, with steam generator, since only this cycle appears capable of yielding concrete short-term results, work is continuing on the direct cycle (gas turbine), which seems promising in the medium run. The French studies are not at present directed towards the use of very high temperatures, which pose difficult problems, but are based on gas temperatures in the 850°C range, for which existing fuels appear to be suitable.

The construction of the actual turbine does not raise any insuperable problems. A layout similar to that of the intermediate pressure section of a steam turbine is suitable, using alloys and cooling devices as developed for conventional turbines. A life of 100,000-200,000 hours can be assured, although some components may need to be replaced after 50,000 hours. Some improvements to the compressors are necessary, to reduce the number of stages and shaft lengths, especially if one single turbine is employed. The decision with the most far-reaching consequences will be whether to use one or more sets. The arrangement of the main circuit and the emergency cooling loops differs considerably in the two cases.

VI. TECHNIQUES USED IN HTRs

The construction of HTR reactors employs techniques developed at different stages in the evolution of graphite-moderated, gas-cooled reactors. The present state of the art is summarized below, starting with a description of the techniques common to prismatic and spherical fuel element types, followed by an examination of the particular techniques used in the two cases. Note that to ensure maximum utilization of the high capital cost of a reactor, components must be replaceable after commissioning: heat exchangers, circulators, control rods and their mechanisms, certain monitoring instruments, etc.

A. Techniques used in all HTRs

1. Coated particle fuels and their reprocessing

All fuel elements are today based on the use of coated particle fuels (a concept dating from about 1960), in which the oxides or carbides of the fissile and fertile materials take the form of small spheres (kernels), each coated with layers of pyrocarbon, sometimes combined with a layer of silicon carbide. These fuel elements may be either prismatic or spherical in shape. The coated particles are of different types[1] (see Table 2).

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[1] The particles used in the first Peach Bottom core were not optimized for the retention of fission products, the coatings serving principally to prevent hydrolysis of the uranium and thorium carbides during fabrication of the fuel. The second core, now in operation, has particles of the BISO and TRISO type, which have been tested intensively in MTR reactors.

(a) Thorium cycle reactor: In this case, the particle contains U^{235}, 93% enriched. There are two main possibilities for the U/Th combination:

a.1 Homogeneous U/Th mixture, proportions between 1:5 and 1:10:
This is the system currently used in Germany (AVR and THTR reactors).

a.2 Thorium and uranium incorporated in different particles:
Separation of the fertile (breed) and fissile (feed) materials into different particles is important because it gives better fuel utilization, for three reasons:

1. There is no criticality limit for thorium-based particles, so that large quantities of thorium kernels can be coated in a single operation, resulting in low-cost fabrication for 90-95% of the fuel.

2. Most of the U^{236} produced in the reactor is contained in the fissile particle, which can be recovered separately in the head-end operation (see below). U^{236} formed by non-fissile neutron absorption in U^{235} is a parasitic absorbent and therefore must not be accumulated in the system.

3. Most of the fission products are produced in the fissile particles. More complex coatings can easily be deposited on the fissile kernels, if necessary, with only a slight economic penalty.

The U^{235} enrichment of the fissile particles after a four-year cycle is approximately 28%. Although the U^{236} content is important (of the order of 55% of the uranium in the fissile particles), the residual U^{235} value is still substantial (USA Ref. 7); this U^{235} would be recycled in both an HTR and an LWR. The concentration of this recycled U^{235} in the smallest possible number of fuel elements reduces the resonance of U^{236}, so that the associated U^{235} has an economic value equivalent to 70% of that of high-enriched U^{235}. For this reason, it is advantageous to use U^{235} for two four-year cycles before removing it from the system at zero value.

The uranium in the fertile particle (thorium) is mainly U^{233}, the quantity of uranium produced representing approximately one-quarter of the initial fissile charge.

With U^{233} recycle, the U^{235} requirements are reduced to about half what would be needed for a non-recycle charge. The isotope ratio of the uranium isotopes in the discharged fuel as compared with the initial quantity of U^{235} (non-recycle operation) is as follows:

	fissile particle		fertile particle
	after four years	after eight years	after four years
U^{233}/U^{235}	-	-	0.27
U^{234}/U^{235}	-	-	0.06
U^{235}/U^{235}	0.08	0.01	0.01
U^{236}/U^{235}_{o}	0.14	0.13	
U^{238}/U^{235}	0.05	0.04	

Continuous recycling of the U^{233} produced, results in an increase in the contents of U^{234}, U^{235} and U^{236} in the recycled uranium, the values for the cycle at equilibrium being 54% U^{233}, 29% U^{234}, 10% U^{235} and 7% U^{236}. The recycling of this uranium does not excessively complicate the calculation of the reactor core.

(b) U/Pu cycle reactors

The particle contains 3-5% enriched U^{235}. The plutonium is formed in the reactor itself. All the particles are of the same type, but with diameters of 600 to 800 μm so as to increase the heavy metal density.

The coatings are either of pyrocarbon or a combination of PyC and SiC, for better retention of solid fission products.

The following properties of these particles must be maintained even under maximum operating conditions:

1. Mechanical integrity of the coating (during fabrication and in the reactor, not more than one particle in 10^4-10^5 may be damaged).

2. Retention of fissile heavy metals (release $<10^{-5}$ of the total).

3. Retention of gaseous fission products (release of Kr^{88} $<10^{-5}$ of the total).

4. Retention of solid fission products. This retention must be specified taking account of the

geometry of the fuel element.

The values shown in Table 2 are those currently attained, but it will probably be possible to achieve better performance.

Calculation codes have been drafted for predicting the behaviour of the coated particles under irradiation. The mathematical models allow for the composition of the kernel, the nature and thickness of the different coatings, temperatures, the accumulation of fission products and the variation of the mechanical characteristics of the materials with fluence and temperature. The many experiments carried out bear out current forecasts of the behaviour of new types of fuels at high fluences.

The European techniques of coated particle fabrication and incorporation in fuel elements are being studied mainly in the context of the Dragon project (THTR until 1968). The results are thus widely available for the whole of the Community.

For longer-term applications (direct cycle and process heat reactors), the work begun must be continued in order to achieve higher burnups and integrated neutron fluences, with even higher operating temperatures than those currently employed.

TABLE 2 - Data on coated particle

	Germany (ref. 5) pebbles		USA				Europe
	AVR	THTR 300	Fort St.Vrain Fissile	Fort St.Vrain Fertile	1100 MWe Fissile	1100 MWe Fertile	Low-enriched
Kernel							
Chemical composition	(U,Th) C_2	$(U,Th)O_2$	$(U,Th)O_2$	Th O_2	U O_2	Th O_2	U O_2
Average diameter (µm)	400	400	200	450	200	450	600 or 800
Ratio Th/U	5/1	10/1	4.25/1	1/0	0/1	1/0	-
Coating							
Type	BISO	BISO	TRISO	TRISO	TRISO	BISO	TRISO
Porous layer (µm)	50	70	50	50			60
PyC high-density	120	110	20	20			30-40
SiC layer	-		20	20			35-40
PyC high-density	-		30	40			30-40
Total	170	180	120	130	130	130	160-190
Burnup (% fima)							
Specified	9	14	20	7			12-15
Actual	10	16	27	27	27	27	12
Fast neutron dose							
10^{21} nvt (E > 0.1 MeV)							
Maximum specified	2	6.8	8	8			7-9
Actual	2.2	7.2	8.7	8.4	8.7	8.4	7
Irradiation temperature (°C)	1250	1250	1150	1150	1150	1150	1100-1200

<u>The reprocessing and refabrication of HTR fuels</u> are a vital economic objective, as stated in Section VII, particularly for thorium cycle reactors, in which recycling permits effective utilization of U^{233}. However, reprocessing is also a possibility for the U/Pu cycle, on the basis, except for the first head-end operation, of the experience gained in fuel reprocessing in light water reactors. Work on U^{233} recycle is in hand in the USA at the Oak Ridge National Laboratory (ORNL) and GGA. Work in Europe has been carried out under the Dragon Project in Britain and in Italy, but the main centre of action at present is in Germany. R&D is proceeding on three fronts:

1. Head-end operation for the selective recovery of fissile and fertile particles.
2. Dissolving of uranium and thorium followed by solvent extraction.
3. Remote refabrication of particles, compact and fuel elements containing U^{233}.

Different techniques are being investigated; for example, some details are given here of the methods being studied in the United States (USA, Ref. 7).

These methods will be tested in the Thorium-Uranium Recycle Facility (TURF), a pilot plant under construction at ORNL, which will become operational in 1976-77. This plant will have a daily reprocessing capacity of 8-10 HTR fuel elements (Fort St. Vrain type block) and will be capable of refabricating two or three blocks a day incorporating the recovered U^{233}. This production rate is equivalent to the equilibrium requirements of a recycling facility designed for a capacity of some 3000 MWe, and corresponds to 5-10% of the scale of a commercial recycling plant.

(1) The head-end comprises various stages, in particular:

(a) Crushing followed by separation of the fuel particles from the graphite pieces.

(b) Combustion

(i) of the graphite pieces without fuel

(ii) of the compacts, graphite pieces with fuel and PyC coatings of the coated particles.

Combustion takes place in fluidized beds of different types depending on the composition and size of the parts to be burnt. The part of the crushed fuel elements containing the coated particles will be burnt in two stages at the rate of approximately 10 kg/h, in a fluidized bed containing, in addition to the particles, alumina to ensure uniform combustion; the TRISO particles containing U^{235} and U^{236} are partially burnt by virtue of the protection of the SiC layer and are screened out after the first combustion.

(2) Dissolving and solvent extraction

After separation of the alumina, the ash obtained will be attacked by an acid solution, to dissolve the oxides of Th and U. After dissolving, solvent extraction takes place for decontamination and purification of the U^{233} and thorium and for their separation. The solution of U^{233} nitrate is sent to the refabrication plant, whilst the thorium, partially decontaminated, is concentrated and stored. The basic technology of the chemical operations, based on the Thorex acid process, has already been developed at ORNL for other thorium applications.

(3) Refabrication: The coated particles containing the recycled uranium incorporate approximately 20% uranium oxide (mainly U^{233}) and 80% thorium oxide (ThO_2). The fuel kernels will be fabricated by a sol-gel process using equipment housed in shielded cells, and then coated in a 12.5 cm diameter fluidized bed, which is also remote-controlled. The particles will be of the BISO type, similar to the original fertile particles. Hot-cell tests have shown that the apparatus works satisfactorily. The particles will then be remote-agglomerated using injection-moulding techniques similar to those employed for the fuel of the Fort St. Vrain reactor (as described later in this document). Experiments in this field are in progress, and work is also in hand on the techniques of introducing and fixing the compacts in holes machined in the graphite blocks.

Fuel elements incorporating U^{233} must, of course, reach the same standards of those of the initial fuel. The development programmes completed and in progress indicate that the production of recycled fuel elements is possible, thus confirming the economic value of the thorium cycle.

2. Helium technology

The principal advantage of helium, which has led to its substitution for the carbon dioxide originally used in gas-cooled reactors, is its chemical inertia, permitting graphite surface temperatures in the 1000°C range for the fuel elements. Furthermore, helium absorbs hardly any neutrons and has no appreciable moderating effect on them, so that it does not influence the reactivity of the system. Helium is and will remain easily available for the requirements of HTR reactors.

Helium technology has been perfected for HTR reactors. The main problems that had to be solved related to the sealing of the primary circuit, helium pumping, lubrication of surfaces subject to friction, helium purification, heat transfer from the fuel to the helium, corrosion of the graphite (since the carbon plate-out may give rise to mass transfer) and thermal insulation of the primary circuit. All these problems have been solved, and the solution satisfactorily tested, in the operation of the AVR, Dragon and Peach Bottom reactors.

The general view is that the adaptation by the industry of these solutions to large-scale reactors does not raise insuperable problems. There are, however, specific problems associated with the use of prestressed concrete pressure vessels, which have to remain leaktight and maintain their thermal insulation for the lifetime of the reactor (30 years). It will also be necessary to be able to guarantee the leaktightness of the heat exchangers (in this connection, a detailed examination of the composition of the gas from the Hinkley Point A reactor carried out by the CEGB showed that the exchangers had no leaks) and of the large helium circulators. In this sphere too, the basic knowhow and experience gained in the operation of the Dragon and AVR reactors are accessible to all manufacturers in the Community.

3. <u>Heat exchangers (steam generators)</u> (France, Ref. 1)

By virtue of the gas temperature (750°C as against 675°C in the AGR), the nature of the gas (helium) and its pressure, the heat exchangers are smaller than in graphite/gas reactors. For the same reasons, the tubes in the bundles are subjected to higher temperatures and steeper thermal gradients in their walls (at least in the superheater and reheater regions). The problem therefore arises of the choice of materials for these tubes, and for their supporting structures, as well as the behaviour of these materials and assemblies in a helium environment at high temperature, with traces of moisture.

Small-bore tubes are favourable for several reasons (exchange surface per unit volume, wall thickness, safety in the event of tube failure). To limit any accidental introduction of water into the helium, the installation of large headers within the primary circuit is avoided. The tubes must therefore pass through the penetrations virtually individually, so as to enable a very small part of the exchanger to be blocked off and taken out of service from outside the pressure vessel in the event of a tube failure.

Because of the direction of circulation adopted for the core cooling, it is necessary, where exchangers are installed in pods in the vessel wall, to use downward boiling in the exchangers, which results in operating constraints and difficulties (in particular on startup, shutdown and part-load operation), or to adapt the heat exchanger bundle configuration or modify the direction of helium circulation in the exchangers, at the cost of some degree of mechanical complication.

4. Circulators (France, Ref. 1)

Compared with the circulators of earlier graphite/gas reactors, the greater specific work requires higher speeds than for CO_2, but the higher speed of sound in helium results in low Mach numbers. The machines are of the axial or centrifugal types, the final solution depending on the specific characteristics of each design and on the drive system.

By virtue of the lower ratings (3-5 MW, as compared with 10 MW or more for graphite/gas reactors), electric motive power is a possibility, and in particular the use of a submerged motor, which avoids the difficulty of dynamic shaft sealing. With oil-bearing circulators, it is necessary to provide an effective barrier between the lubricating oil and the primary helium, and this appears to be a difficult problem with certain types of installation (vertical-shaft circulators with bottom-mounted impellers). The use of gas bearings eliminates this risk of pollution, but necessitates a horizontal layout and special arrangements to ensure correct working of the machines under various operating conditions (in particular, operation at atmospheric pressure after depressurization of the core).

5. Prestressed concrete pressure vessel and integrated primary circuit

These techniques were developed for CO_2/graphite reactors. PCRVs were used for the first time in the construction, in 1956, of the Marcoule G2 and G3 reactors. The integrated primary circuit technique was used in the AVR in 1961, but with a steel pressure vessel. These have been standard techniques for many years in gas/graphite reactors in France and the UK (Magnox and AGR).

However, compared with the vessels of earlier graphite/gas reactors, the HTR vessels have a smaller internal cavity diameter but have to withstand a higher pressure (approx. 55 bar as against 40 at most). The wall and head thicknesses, assuming the same architecture, thus remain on the same scale as previously. A variety of geometries were used for the first prototype reactors. However, most recent reactor designs employ a new type of PCRV concept, in which the exchangers are accommodated in pods in the side wall of the vessel. This is a system patented by Dragon and adopted by BNDC (British Nuclear Design Corporation) for the Hartlepool station, so it will be partially tested before the construction of large HTRs. All the experience in the construction of large PCRVs for nuclear reactors is concentrated in France and the UK, but GGA has also gone in for PCRVs by building the Fort St. Vrain reactors and by adopting the new type of vessel for its 1100 MWe HTR tender. The circumferential prestressing of this new type of vessel may be applied by wire winding, but prestressing by tendons is also possible.

The pod closures (diameter approximately 3.5 m) must be so designed as not to prejudice the intrinsic safety of the PCRV. Further, with multiple pod refuelling (prismatic elements), the top head has a large number of penetrations. For this reason, transverse tendons cannot be used to prestress the top head.

B. Techniques specific to homogeneous prismatic-element HTRs

1. Fuel elements and reflectors

In Europe the manufacturers and electricity producers interested in prismatic HTRs have concentrated mainly on the low-enriched uranium cycle. The design of the fuel and core is largely based on the fundamental study presented by the Dragon project in 1967.

The whole of the moderator is incorporated in the fuel elements, which occupy the entire volume of the core. The coated particles are inserted in graphite tubes whose outside diameter is similar to that of the fuel tubes used in the Dragon reactor. The behaviour of these tubes under irradiation can therefore easily be studied in Dragon. The particles are incorporated in cartridges (compacts), allowing maximum heavy metal densities in the range 0.8-1.0 g/cm^3 to be reached with particles of kernel diameter 600-800 μ. Accelerated testing of these particles is under way in the experimental reactors.

The type of rod favoured by the manufacturers is of the tubular interacting type, in which the compacts are clad internally and externally with graphite in contact with the coolant gas; the fuel, since it may expand under irradiation, sets up a stress in the outer graphite tube. The fuel tubes containing the compacts are inserted in channels in the hexagonal blocks (about 40 cm across flats) of graphite (isotropic pressed or fine-grain anisotropic drawn) to form the fuel elements. These graphite blocks will reach a maximum temperature of 700-900°C in the reactor and will remain in it for about three years. The fast neutron fluences will be lower than the fluences specified after 20 years in the isotropic graphite blocks used in AGR reactors. A large-scale high-temperature and high-fluence graphite irradiation programme has been put in hand in Europe and in the United States (see below), and there are regular exchanges on the results obtained. The European tests have been concentrated on gilsonite coke-based graphite and have shown the excellent stability under irradiation of this graphite at the required fluences.

The variants of this type of tubular fuel are now being studied. In the "teledial" type, the compacts are inserted in the wall of a single graphite tube in an arrangement resembling that of a telephone dial. With directly cooled compacts, a fuel region is linked directly to a fuelless region, the latter being in direct contact with the coolant gas. The properties of the two regions, with and without fuel, must in this case be adjusted in such a way as to avoid cracking under irradiation (Germany, Ref. 5). Whichever type of fuel tube is used, account must be taken of the risks of vibration of these tubes and stabilization systems must be provided for. The graphite blocks used in HTRs cannot be irradiated in materials test reactors because of their size. The dimensional behaviour and internal stress situation must be predicted from complex calculation codes involving the block dimensions, variations in temperatures and flux with time, and variations of the graphite characteristics with fluence and temperature; in addition, account must be taken throughout of creep phenomena. It will be possible to verify the mathematical models on small-size fuel blocks irradiated in Dragon.

In the USA, GGA has adopted for Fort St. Vrain, and is proposing for its 1100 MWe reactor, a different fuel element from the type described above. The cycle in both cases is based on $U^{235}/Th/U^{233}$. The hexagonal non-isotropic graphite blocks are perforated with 108 cooling holes 15.8 mm in diameter alternating with 210 holes filled with 12.4 mm diameter fuel pins. The coated particles, having a kernel diameter of 200 µ for the feed and 450 µ for the breed, are incorporated in the fuel pins in the form of compacts. These pins are fabricated by a hot-injection technique: a viscous mixture of a binder and powdered natural or isotropic graphite is injected into a mould previously filled with coated particles. The material is baked and treated at 1800°C to stabilize the pin dimensions and to ensure partial graphitization of the matrix, so as to improve the irradiation behaviour.

The dimensions of the coated particles used by GGA are such that most of the available experimental results are applicable to them, and no difficulty is to be anticipated on this account.

The pins were irradiation-tested and the results were satisfactory: burnup 20% fima, fast neutron fluence $7.0 \cdot 10^{21}$ nvt (E >0.1 MeV) temperature 1400°C.

The graphite for the Fort St.Vrain reactor is nuclear-grade needle coke manufactured by the Great Lakes Corporation, Grade H-327.

An extensive irradiation programme has covered a temperature range from 650 to 1250°C with maximum fast neutron fluences of 1.10^{22} nvt(E>0.1MeV) and has given excellent results. All the values obtained were better than those specified for the reactor. It should also be noted that the fuel elements reaching fluences of 8.10^{21} nvt will have only a relatively constant temperature below 1050°C, whilst the ones at 1260°C will reach a fluence of only about 2.10^{21} nvt.

2. Core and internal structures

Because the fuel is integral with the moderator, handling facilities must be provided for the core of a prismatic-type high-temperature reactor, and this affects the design. The integrity of the core must be ensured by the supporting system, and the lateral and head restraints guarantee earthquake resistance and column holding during unloading.

In view of the core outlet temperature of the helium (750°C), two types of support are being studied: a cooled metal floor, or a system with ceramic extensions of the core columns down to the bottom cap (the latter is well-suited to the annular architecture, and is most frequently used).

3. Charge machine and reactor control

Because the fuel and moderator are integral, fuel handling consists of manipulation of the blocks of the core. This problem is difficult both in itself and as regards its consequences for the core (and vice versa) and for the top cap of the reactor vessel.

Two types of handling are at present being investigated, off-load and on-load, the machine in both cases being outside and above the vessel.

On-load refuelling, as opposed to off-load refueling, reduces fuel cycle costs and anti-reactivity requirements, and permits greater freedom in shutdown scheduling. In view of the organization of the primary circuit (downward cooling), it is carried out in gas at 300°C, whereas some equipment operates in gas at 400°C in CO_2-cooled reactors. However, the core is surrounded by a gas flow at a pressure of approximately 55 bar, and a failure would have more serious consequences than a similar failure with the reactor shut down. The European manufacturers are at present still studying the two charging systems, whilst in the USA, GGA so far favours off-load refuelling.

Control is effected by means of a large number of control rods, operated by mechanisms inside the top cap of the reactor vessel. The Xe oscillation control techniques used for the Magnox and AGR reactors may be employed. This problem is, however, less critical in HTR reactors, because of the smaller size of the core.

C. Techniques specific to spherical-element HTRs

1. Fuel elements and core of the THTR reactor

The reactor core consists of a pebble bed. The pebbles, having a diameter of 6 cm, are introduced at the top of the reactor through different fuelling tubes, allowing the pebbles to be fed either to the centre or to the edge of the core.

The pebbles are removed at the bottom of the reactor. This type of fuel was designed for the AVR and was also adopted for the THTR reactor. The retention of the same shape for the fuel element is an invaluable advantage for testing the mechanical properties of the fuel (irradiation behaviour, resistance to corrosion, erosion, shock, etc.). Development was concentrated on the following:

1. The fabrication of particles with heavy metal kernels diameter 400 µm) coated with pyrocarbon layers.

2. Appropriate composition of the graphite matrix, with particular reference to problems of compatibility and irradiation behaviour.

3. Fabrication of complete fuel elements consisting of a kernel containing the fuel and a fuelless graphite sheath.

Extensive irradiation programmes in Dragon and in materials-testing reactors were carried out in parallel with the development of fabrication techniques. Finally, computer programs were written to analyse the behaviour of the material under thermal stress and expansion.

Table 3, which shows the results obtained in demonstration tests for the prototype reactor, gives the breaking force and drop test figures. A breaking force of 97.5% means that 97.5% of the fuel elements must have breaking forces in excess of 1800 kgf measured between steel plates. The drop test figure is the number of times a fuel element can withstand dropping onto the pebble bed from a height of 4 m without being damaged; 99.99% of the fuel elements must reach these values.

TABLE 3

THTR fuel element test results

(Germany, Refs. 1 and 5)

	Specification	Results
Heavy metal content	0.96 g U^{235} 10.2 g Th	0.96 g U^{235} 10.2 g Th
Breaking force	97.5% 1800 kgf	over 97.5% 1800 kgf
Drop test	99.99%	over 99.99%
(Standard test)	50 times	50 times
Matrix graphite anisotropy	1.3	1.15
Thermal conductivity of matrix graphite at 1000°C (non-irradiated element) (cal/cm.sec. °C)	0.07-0.08	0.07-0.08
Corrosion rate at 1000°C (1 v/o H_2O in 1 atm He, 10 h) ($mg/cm^2.h$)	1.5	<1.5
Contamination in uranium U/U total	$5 \cdot 10^{-4}$	$5 \cdot 10^{-5}$
Specification test		
Average burnup % fima	12	-
Average fast neutron fluence (E >0.1 MeV)	$4.8 \cdot 10^{21}$ nvt	
Maximum burnup % fima	14	16
Maximum fast neutron fluence (E >0.1 Mev)	$6.8 \cdot 10^{21}$ nvt	$7.2 \cdot 10^{21}$ nvt
Xe^{133} release rate		$5 \cdot 10^{-5}$
Dimensional stability under irradiation	≤2%	≤2%

2. Manipulation of fuel and high-speed measurement of burnup

Compared with the AVR reactor, it was necessary for the THTR to increase the circulation rate from 50 to 500 spheres per hour and the number of positions for charging the fuel above the pebble bed from 5 to 15; for this reason the fuel element circulation system requires a larger number of functional parts.

The spheres are transferred from the core to the fuel circulator by gravity (angle of inclination usually 10^{o}); for this reason the circulator must be located underneath the core and the PCRV. In order to ensure simplicity, moderate construction costs, avoidance of helium leaks and a high degree of safety, the functional parts of the fuel circulation system, and also of the charging and discharging system, were built in the form of individually replaceable modules.

The fuelling room is not normally accessible. The maintenance of drives and gearing can be carried out in a radiation-free workshop.

On leaving the core, the spheres go first to a separator which separates the individual spheres, eliminates fragments and removes spheres whose dimensions are out of tolerance, and thence to a measuring reactor having a power of approximately 100 W, which was developed as an instrument to measure the burnup of the fuel elements leaving the prototype reactor. The spheres pass through this reactor at a maximum frequency of one every 7 sec. Computer analysis of the change in the reactor neutron flux, measured during the passage of the spheres, yields information as to the fissile and fertile material and fission product composition.

The spheres are pneumatically recirculated in the core by elevator tubes. They are decelerated on exit from the fuel charging tube by gas reflux. For this purpose a section of perforated tube is connected to the intake of the fuel circulation fan.

Because of the large number of junctions, which are highly exposed to wear, the fuel circulation system must be easily accessible and repairs must be carried out on-load. The integral fuel circulation system can be separated from the primary circuit and independently depressurized. The reactor is designed to be able to operate at full power for several weeks with the fuel element circulation system shut down. Normally, the replacement of small units takes no longer than 24 hours. For this reason, even more serious faults in the fuel circulation system have little effect on the availability of the reactor.

The development of the THTR fuel element circulation system included extensive functional tests. In addition to the tests on prototypes of components of the final machine (under conditions simulating reactor operation), lifting tests were carried out with three of the 15 reactor elevator tubes, at full scale.

The correlations between the sphere and gas speeds were evaluated with this test arrangement.

3. Reflector

No part of the reflector is replaceable, and some of the isotropic graphite blocks of which it is composed will be exposed to high fast neutron fluences during the life of the reactor. This, together with the risk of corrosion and erosion of the bottom reflector, makes it difficult to predict the long-term mechanical behaviour of this structure. However, the experimental results available on gilsonite coke-based graphite give reason for optimism.

4. Neutron physics and thermal design of the reactor.

These aspects of the reactor are complex. The composition of the core must be calculated from the sphere charging programme, taking account of the information on the laws of sphere motion gained from models. Allowance must be made for the effect of helium and temperature on the coefficients of friction between spheres, between spheres and the vessel walls, and between spheres and the control rods. The calculation of the maximum temperatures must take into consideration the distribution of the gas flow, which tends to avoid the hot parts of the core, the composition of the core and the probability of fresh fuel spheres bunching.

The neutron flux calculation must allow for the continuous movement of the fuel elements and the gap between the irregular top of the core and the upper reflector.

It is necessary to determine at every instant the method of reecharging - central or peripheral - of the spheres removed from the bottom of the reactor depending on their composition - graphite, fuel, poison - and, in the case of the fuel, their burnup. These calculations must be carried out regularly during the operation of the reactor, but they do not raise any fundamental problem. They require the use of a computer, but this is already needed for measurement of the burnup.

5. Reactor control

Control of the THTR reactor will be effected partly by means of 36 rods located in the lateral reflector and partly by means of 42 rods to be introduced directly into the pebble bed by a pneumatic drive system. Extensive model studies indicate that the direct introduction of rods into the bed is unlikely to raise problems.

The drive consists of a double-acting piston (stepping piston) with a motion mechanism and a stopping mechanism. The drive is helium-actuated.

For a scram, the rods are lowered to a depth of 2-3 m by a long-stroke piston system operated by an independent helium circuit, at high speed (~ 30 cm/sec); thereafter, the descent continues if necessary at low speed down to the bottom of the reactor.

The end of the absorption rod is concave, so a sphere is grasped during the descent of the absorption rod and the load exerted on the spheres is reduced, since the geometry of the graphite/steel contact is favourable. The rods are cooled by the helium circulation due to the pressure gradient in the core.

The individual components of the drives, e.g., the rod linkages, the piston/cylinder system, the motion and stopping mechanism, the rod piston position indicator, the valves, etc., have been thoroughly tested on individual test beds. In this way a high degree of reliability was attained and demonstrated for the individual components before construction of the prototype drive.

The constructional specifications of the drive are to be tested in an extensive programme using the prototype drive, mounted in a test tower, starting in March 1972.

6. Prestressed concrete reactor vessel

For the development of the prestressed concrete reactor vessel, it was necessary to carry out stress analysis studies to evaluate the breaking strain for different load conditions. Three-dimensional computer programs were available for the stress analyses. An approximation technique developed by a British firm was used to determine the breaking strain. The breaking strain of the upper chamber of the vessel was tested by loading a 1:20 scale model to failure. The influence of the penetrations in the top cap was also studied in these tests.

The precalculated stress distributions were checked by pressure and temperature tests on a 1:5 scale model. A deformation test on the liner showed that the concrete tendons satisfied the static requirements. The calculation methods were checked by tests on a 1:47 scale cast resin model with accurately defined material conditions.

VII Economics of the HTR family

1. General data

1.1 The real prospects of high-temperature thermal neutron gas reactors becoming commercially competitive with LWR reactors progressed in 1971, when Gulf General Atomic received two large orders representing 15% of the nuclear electric capacity ordered in the USA that year. This indicates that the transition point between subsidized R&D programmes and genuinely commercial sales has been reached in the United States. However, the contacts initiated by GGA with various European manufacturers could lead to fast progress on the European nuclear energy market.

1.2 In the United States, the share of the cost of the Fort St. Vrain station for Public Service of Colorado amounts to 61 million u.a., or 15 million more than the original estimate. For its part, the USAEC spent 55 million u.a. to back a "first-of-a-kind" project, in particular in the financing of an R&D programme conducted partly by Gulf and partly at Oak Ridge. Furthermore, GGA spent an unspecified sum on the industrial development of the fuel elements and a complementary research programme. GGA is the owner of the fuel for the first eight years and sells electricity to PSC at a rate equivalent to 1.7 mills per kWh (as compared with 2.6 mills per kWh for a recent coal-fired station belonging to the same company).

In response to a call for bids issued in 1969 by Eugene Water & Electric Board (Oregon), Gulf General Atomic submitted a project for an 1100 MWe HTR station on terms competitive with those for LWR plants. However, following a campaign against the construction of a nuclear power station, the Eugene project has been deferred.

In 1971, Gulf General Atomic received two orders for large power stations:

(a) A letter of intent from the Philadelphia Electric Company for the construction of two 1160 MWe power stations, to be commissioned in 1979 and 1981 respectively. This order, which was placed without any competing bids being submitted, will represent at $200-300 million contract for GGA, the total installation cost being $700 million. The thorium fuel cycle has been chosen, since the low-enriched uranium (or U+Pu) cycle is more expensive and not worthwhile in the USA (see also USA, Ref. 7). The AEC has confirmed its principle of guaranteed buy-back of U^{233} pending the availability of a commercial reprocessing facility for the thorium cycle.

(b) A letter of intent for two 770 MWe reactors from the Delmarva Power and Light Company, the nuclear part amounting to $200 million, out of a total station cost of $680 million. The first reactor is to go on line in 1979, which implies a start on construction in 1974, the same year as for the Philadelphia Electric station. The second reactor would be operational three years later.

GGA is also in contact with other American electricity producers.

1.3 In Europe, the total cost of the THTR power station programme so far is about 190 million u.a. This includes 22 million u.a. for the direct backing R&D programme, 108 million for the construction and commissioning of the station on a turnkey basis, 9.6 million for the first fuel charge, 25.2 million for customer costs (including interest during construction, taxes and cost indexing) and up to 25.2 million for additional capital expenditure which might be necessary for modifications deriving from results of the research programme.

Construction is being financed by the HKG company with bank credits guaranteed by the Federal Government and the Government of North Rhine-Westphalia, and by Federal and Land subsidies. Further, operating risks will be covered up to a total amount of DM 150 million (37.5 million u.a.).

2. Fuel cycle

2.1 Thorium cycle

The HTR was originally designed for the thorium cycle, which gives an improved energy yield from the natural resources of uranium; the neutron economy is superior because U^{233} has a higher η value (number of neutrons produced per neutron absorbed) than U^{235}.

This cycle requires the availability of high-enriched (93%) uranium. Several alternatives are possible:

1. Th reference cycle with U^{235} and U^{233} recycling: lowest cycle cost, around 1 mill/kWh (USA, Ref. 8).

2. Thorium cycle without recycling: more expensive than the reference cycle by about 0.25 mill/kWh.

3. Thorium cycle with recycling and makeup plutonium: cycle cost the same as for the reference cycle.

GGA's economic calculation (USA, Refs. 11 and 12) thus show that the Th cycle with U^{233} recycle is the optimum for HTRs, but requires the development of thorium fuel reprocessing facilities.

The basic economic hypotheses chosen are as follows:

Cost of U_3O_8 ($/lb)	8
Cost of separative work ($/kg U)	26
Ratio U^{233}/U^{235}	14/12
Ratio fissile Pu/U^{235}	10/12
Cost of machined graphite ($/block) (approx. 3.6 blocks per MWe)	1000
Coated particle fabrication cost ($/kg)	60
Transport cost ($/block)	230
Reprocessing cost ($/kg)	65
Storage cost ($/block)	230

For the present, GGA have chosen cycle No. 2, U^{235}/Th^{232}, for the period up to the beginning of the eighties. By then the HTR station installed capacity will make the reprocessing of fuel elements and the recycling of U^{233} economic. Meanwhile, fuels discharged from HTR stations will be stored. At least until 1977, and possibly also subsequently, the USAEC had adopted a fuel reprocessing policy. On the basis of a conceptual study completed in 1970, of a reprocessing plant for an installed capacity of 25,000 MWe, the USAEC has fixed a reprocessing charge of $125/kg of metal (corresponding to a daily output of 1040 kg Th+U).

In addition, the USAEC has established a guaranteed price for U^{233} up till 31 December 1975 as follows:

(a) $13.79/g on the basis of a isotopic separation cost of $28.70/SWU, or

(b) $14.76/g, corresponding to $32/SWU.

The economy of the fuel cycle depends on a number of reactor design and operating characteristics, in particular the fuel charge, the power density, the in-pile dwell time of the fuel.

For GGA, the reference cycle is based on a power density of 8.2 MW/m^3, a dwell time of four years and annual recharging. The minimum cost of the fuel cycle is obtained for a C/Th ratio of 250.

The divergences from the reference cycle are + 0.5 mill/kWh for the cycle with deferred recycling (after eight years' storage) and approximately 0.25 mill/kWh for the non-recycling cycle (USA, Ref. 9). The fabrication costs of HTR fuel elements are given in USA, Ref. 10.

A German document (Germany, Ref. 6) quotes a fuel cycle cost of between 1.40 and 1.59 mills/kWh, depending on the assumptions made. Some difference of opinion exists between the American authors (USA, Ref. 8) and these German authors (Germany, Ref. 6) as to the value of the Th cycle as compared with the low-enriched uranium cycle. However, this difference is apparently resolved by a detailed comparison of the Th cycles adopted: different thorium charge, recycling or non-recycling of the U^{235} discharged after irradiation, different size of power station markets, choice of coated particles for U^{235} recycling, size of reprocessing market.

A study has also been carried out in the United States by the Edison Electric Institute on the use of plutonium in the HTR as a makeup fuel to replace U^{235}. The use of plutonium from water reactors is justifiable when its price lies between $9 and 11 per fissile gram (USA, Ref. 9). The fabrication of coated particles containing a kernel of plutonium oxide with different degrees of porosity has been demonstrated by Belgonucléaire in Belgium (Belgium, Ref. 1) under a Dragon contract, and also in contact with the THTR project and the Karlsruhe JRC. Irradiation tests have proved the validity of the concept of separation of the fissile Pu particle from the fertile particle.

2.2 Low-enriched uranium cycle

The countries (in particular, the UK and France) which have developed an industrial infrastructure for gas/graphite reactors (Magnox and AGR respectively) have realized that HTRs could be introduced using a fuel cycle similar to that of earlier reactors and making use of existing techniques. Hence the appearance of the low-enriched uranium cycle $U^{235}/U^{238}/Pu^{239}$. This cycle is conceivable with or without Pu recycling, but under present-day economic conditions the costs of these two variants of the low-enriched U cycle are similar and are 0.20 - 0.25 mill/kWh higher than the thorium reference cycle.

With a view to reducing the enrichment to 3-4%, manufacturers have designed HTR reactors of the "heterogeneous" type, where the fuel was concentrated in rods using large particles 800 μm in diameter, located in a large-pitch moderator lattice. However, with this type of fuel, the temperatures and fast neutron fluences in the vicinity of the fuel rods became prohibitive. On the other hand, subsequent physics tests showed that a sufficient degree of heterogeneity (to reduce resonance absorptions) was reached with coated particles dispersed in graphite matrices; this has permitted a return to a low-enriched U homogeneous reactor concept.

The enrichments in this case are 6-7%. The economic incidence of such a variation in enrichment is negligible.

(Table 4 gives, as an example, the figures obtained in the USA for different types of fuel cycle (USA, Ref. 11).)

Table 4. <u>Characteristics of different types of HTR fuel for a 1000 MWe reactor at steady state (USA, Ref. 11)</u>

Type of fuel cycle	Thorium reference cycle	Th cycle without recycling	U/Pu cycle
C/Th or C/U^{238} ratio	225	200	(325)
Fuel lifetime	4	5	3
Conversion ratio	0.74	0.69	0.53
Average specific power (MW/kg fissile)	1.9	1.6	2.7
Fissions per initial fissile atom	1.5	1.5	1.2
Age peaking factor	1.41	1.47	1.42
Thorium loading (kg/year)	9130	8200	-
Makeup uraniun (kg/year)	273	557	8920
Enrichment of makeup U	93.5	93.5	6.95
Recycled uranium (kg/year)	388	-	-
Enrichment of recycled U	64	-	-
Thorium discharged (kg/year)	8550	7600	-
Uranium discharged (kg/year)	71	374	8030
Enrichment of discharged U	29	62	0.90
Plutonium discharged (kg/year)	-	-	60
Enrichment of discharged Pu	-	-	55.4
Cost of fuel cycle (mills/kWh)			
With reprocessing	1.05		1.30
Without reprocessing		1.32	

3. Energy cost

The only comparative study of the economic prospects in the United States of high-temperature gas reactors as compared with light water reactors is that of the Edison Electric Institute, carried out in 1969 (USA, Ref. 13). The figures given were as follows (expressed in 1975 dollars):

	1975	1980	1985	1990	2000
1. HTR with recycling					
Capital cost ($/kWe)	230-270	180-220	160-190	150-180	135-165
Cycle (mills/kWh)	1.2-1.4	1.2-1.4	1.1-1.3	1.0-1.2	1.0-1.2
Operation (mills /kWh)	0.3	0.3	0.3	0.3	0.3
Energy cost (mills/kWh)	6.1-7.1	5.1-6.1	4.6-5.4	4.3-5.1	4.0-4.8
2. LWR					
Capital cost ($/kWe)	200-240	170-210	150-180	155-185	150-180
Cycle (mills/kWh)	1.7-1.9	1.5-1.7	1.4-1.6	1.4-1.6	1.4-1.6
Operation (mills/kWh)	0.3	0.3	0.3	0.3	0.3
Energy cost (mills kWh)	6.0-7.0	5.2-6.2	4.9-5.7	4.8-5.6	4.7-5.5

The following assumptions underlie these figures:

1. The electricity producers must build at least three large-scale HTR stations in the next few years in order to provide, in the opinion of Gulf General Atomic, sufficient business to ensure economic construction of the reactor and development of fuel fabrication and reprocessing.

2. The cycle cost of an HTR reactor in 1975 assumes that a viable industry has in fact been set up.

3. The reduction in the cycle cost in 1985-90 is due to improvements in fuel fabrication and reprocessing techniques.

4. The 1975 capital cost is based on the technology of Fort St. Vrain and is taken from Gulf General Atomic tenders submitted to American electricity producers.

5. The reductions in the capital costs are due to technological improvements (similar to those experienced with LWRs), improved construction techniques and the use of gas turbines.

According to the Edison Electric Institute, the HTR concept may prove to be an economic reactor system before breeder reactors go onto the market. A relatively modest R&D programme in the field of fuel reprocessing is still necessary. However, until fuel reprocessing and refabrication are on a commercial footing, there is some risk that the projected fuel cycle costs may not be attainable. For this reason it is important for three large-scale HTR stations to be ordered quickly, in order to provide an adequate industrial foundation. This is also the reason why the Edison Electric Institute considered that the USAEC should continue its HTR reactor development effort.

VIII. CONCLUSIONS

HTR reactors have, at least in the United States, reached the break-through point between subsidized R&D programmes and genuinely commercial sales. Only U^{233} recycling still requires government backing, both for an R&D programme and for a policy of U^{233} buy-back and irradiated fuel reprocessing. Thanks to an intensive and wide-ranging programme of tests on the main components, GGA has reached the industrial stage for reactors in the 1100 MWe region.

American studies (USA, Ref.11) indicate that, by comparison with the light water reactor, only the reference cycle $U^{235}/Th^{232}/U^{233}$ permits significant gains as regards both the fuel cycle and natural uranium requirements.

In Europe, HTR technology has been extensively demonstrated and tested in the AVR and Dragon reactors; in addition, the technology developed for CO_2 reactors has yielded extremely valuable practical experience. However, the only demonstration station under construction is the 300 MWe THTR reactor based on the thorium cycle, scheduled to commence operation in 1976 on the Schmehausen site in Germany. The other European countries have not yet taken a decision as to the introduction of the HTR.

Since little precise economic information is available on HTR reactors, either in the USA or in Europe, it is difficult to make a categorical statement as to the chances of the large-scale introduction of this reactor family in electricity generation systems. But the first orders placed in the USA, the initial contacts made by GGA with various European manufacturers and the development potential of the HTR family could lead to rapid growth on the European energy market.

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