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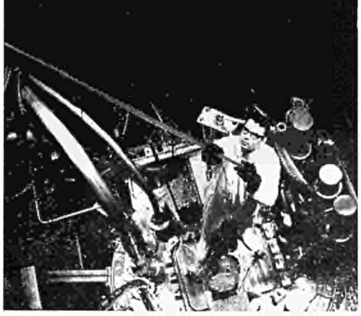

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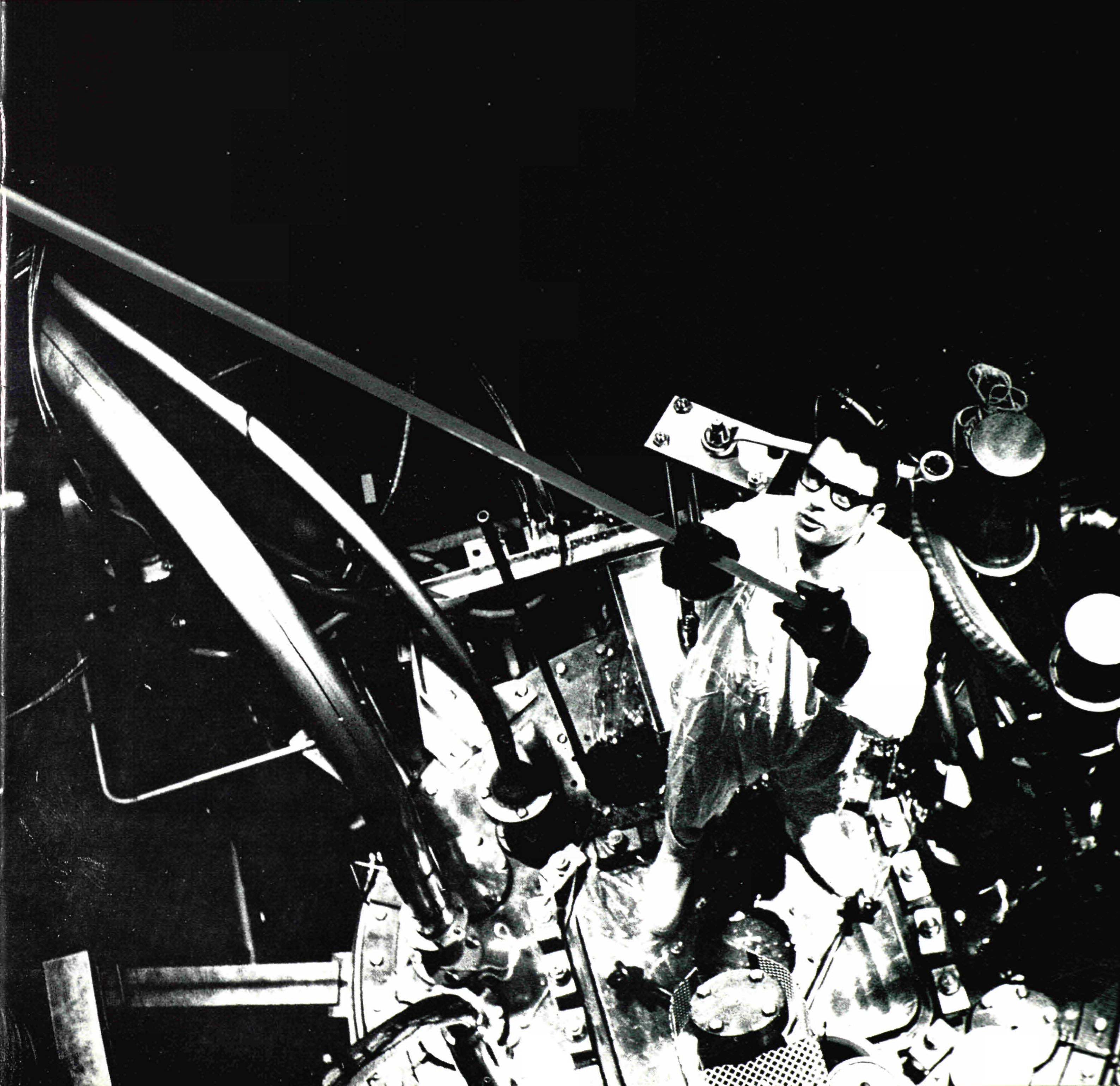

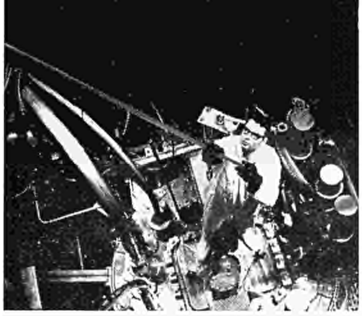

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1967-4

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

At a time when it is passing through a crisis, the Community's iron and steel industry has a clear duty, not only to undertake advanced technical research with the aim of improving its competitive position, but also to take over techniques already developed in other sectors, adapting them if need be to its special requirements.

In this context, it may be asked to what extent the technological knowhow acquired by the nuclear sector, for instance, can be turned to advantage by the steel industry.

Today, nuclear energy is already rendering material service, albeit modest, in the form of radioelement applications. But could it not make a more substantial contribution?

We believe it could, and more particularly on two levels.

On the one hand, attention must be focussed on the possibility of bringing about a spin-off of nuclear technology to the steel industry.

Let us take a case in point. Under programmes for the development of fast reactors, it has been necessary to devise liquid metal entrainment systems, notably of the electromagnetic type. Hence the proposal that similar systems should be employed for the transport of molten metals, in particular as regards continuous steel production by the counterflow process, or again for the vacuum degassing of steel, where the carrier gas could be replaced by an electromagnetic pump.

But the most attractive prospects for the iron and steel industry are doubtless those offered by the direct use of nuclear energy, as is demonstrated by Professor Wenzel in the article which he contributes to the present issue. Outstanding in this respect is the possibility of using the heat from a reactor for iron ore reduction, thus substituting it for two-thirds of the coke at present consumed for this purpose.

A radical change of this kind would undoubtedly call for considerable efforts in research and development; but, having due regard to the fact that its economic effect on the iron and steel industry might be comparable to that which nuclear energy is going to have on the electricity production sector, it is clear that we must contemplate it very seriously.

Manfred SIEBKER **Henri MARTIN**

Irradiations in materials-testing reactors

JOSEF BUGL, *Petten Establishment, Euratom Joint Nuclear Research Centre*

IF THE REQUIREMENTS imposed on certain components of a reactor are considered, it becomes obvious that material problems play a great part in reactor technology. Apart from optimum neutron physics properties, reactor materials must possess, among other things, good mechanical strength even under thermal cycling, dimensional stability, corrosion resistance and, especially in the case of fuel materials, good thermal properties. Although a large number of reactor materials more or less satisfy these requirements outside the radiation field, irradiation in the reactor causes damage which leads to changes in physical and chemical characteristics.

The fast neutrons and fission fragments released during nuclear fission may eject atoms from their lattice points in solids, and displace them more or less perma-

*Table 1: High-flux materials-testing reactors in the countries of the European Atomic Energy Community**

Reactor	**HFR**	**BR 2**	**FRJ 2 (DIDO)**	**FR 2**	**SILOE**	**OSIRIS**	**EL 3**
Location	Petten, North Holland	Mol, Belgium	Jülich, Germany	Karlsruhe, Germany	Grenoble, France	Saclay, France	Saclay, France
Type of reactor	MTR, light water cooled and moderated	MTR, light water cooled and moderated	DIDO, heavy water cooled and moderated	heavy water cooled and moderated	MTR, light water cooled and moderated	MTR, light water cooled and moderated	heavy water cooled and moderated
Power (MWth)	30	60	10	44	30	50	14
Number of fuel elements	37	28	25	160	30	34	280
Average fuel charge	180 g U 235	244 g U 235	115-140 g U 235	13.6 kg UO_2 1.5% enriched	280 g U 235	390 g U 235	UO_2 4.5% enriched
Maximum thermal neutron flux density (n/cm². sec)	1.3×10^{14}**	5×10^{14}**	1.1×10^{14}**	1.2×10^{14}**	3.2×10^{14}**	2.2×10^{14}**	0.8×10^{14}**
Maximum fast neutron flux density (n/cm². sec)	1.1×10^{14}**	4.5×10^{14}**	0.3×10^{14}**	0.6×10^{13}**	1.4×10^{14}**	2.7×10^{14}**	0.1×10^{14}**
Nuclear heating (W/g)	2-4.5	6-14	2-3	3	7	7-12	0.5
Irradiation positions	13-15 reflector positions 12 positions in the poolside facility 10 horizontal beam tubes	28 fuel element positions 17 reflector positions 5 vertical channels 9 horizontal or axial beam tubes	5-8 fuel element positions 28 vertical channels 30 horizontal channels	1 vertical central channel 2 vertical channels 41 isotope channel positions 4 horizontal beam tubes	58 fuel element and reflector positions 2 horizontal beam tubes	5 in core positions 50 positions in the periphery of the core	25 in-core positions 4 vertical channels in-core 16 beam tubes

* Special materials-testing reactors, such as ESSOR, MZFR, PEGASE, etc., are not mentioned.
** Disturbed values

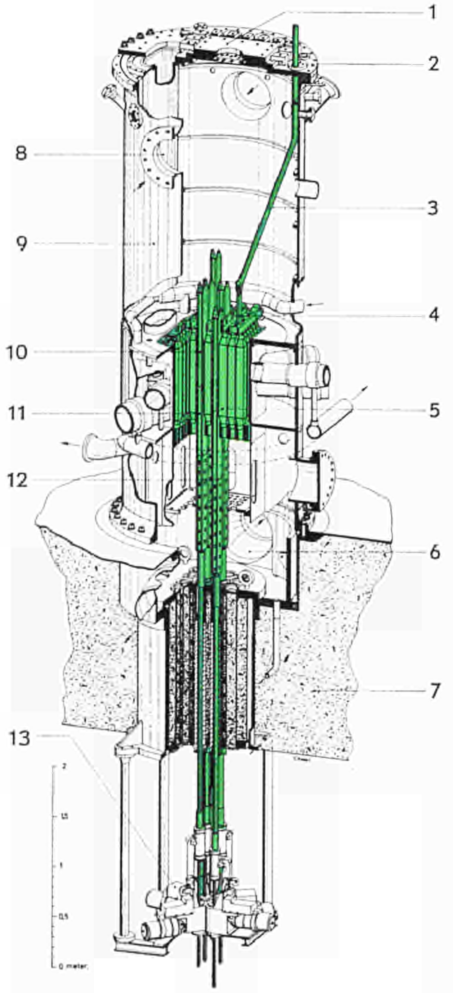


Fig. 1: The Petten high-flux reactor.

Output	*30 MW*
Av. thermal neutron flux density	*1 . 10^{14}n/cm².sec*
Max. thermal neutron flux density	*1.3 . 10^{14}n/cm².sec*
Av. fast neutron density	*0.8 . 10^{14}n/cm².sec*
Max. fast neutron flux density	*1.1 . 10^{14}n/cm².sec*

1 Cover; 2 Penetration; 3 Extension pipe to irradiation capsule; 4 Irradiation capsule; 5 Reactor core with fuel and reflector elements; 6 Cooling water outlet; 7 Concrete foundation; 8 Cooling water inlet; 9 Reactor vessel; 10 Control rod—absorber section; 11 Control rod—fuel section; 12 Control rod—connection to drive mechanism; 13 Control rod drive mechanism.

nently. In the simplest case, a vacancy is formed while the expelled atom comes to rest as an interstitial atom at a different point in the lattice. Should the kinetic energy of the primarily excited atom be greater than the Wigner energy, then this primary recoil atom can knock further atoms out of their lattice points and a displacement cascade results. In this way, for example, a 2 MeV neutron in graphite can displace 1,900 atoms.

Slow neutrons having an energy of about 20 eV and below cannot knock any atoms out of the lattice but they do cause ionisation and nuclear excitation. Thus foreign atoms are incorporated in the lattice as a result of nuclear transformation.

The precise effect of radiation damage on the physical characteristics of reactor materials depends to a large extent upon whether it is a question of a lattice effect or an ionisation effect. The following changes, in some cases quite pronounced, occur in the characteristics of solids:

— volume and density
— mechanical strength
— ductility
— electrical resistance
— thermal conductivity
— diffusion coefficients

Materials-testing reactors

Since the damage which radiation causes to many reactor materials must be known when a prototype reactor is built, this characteristic is studied in specially constructed materials-testing reactors. The main data on the high-flux materials-testing reactors built and operating in the countries of the European Atomic Energy Community are given in Table 1. With the exception of the *DIDO*, *EL 3* and *FR 2* reactors, they are all light-water reactors whose principal design features are based on the *MTR* (Materials-Testing Reactor).

Fig. 1 shows a longitudinal cross-section of Euratom's *HFR* high-flux materials-testing reactor at Petten. At present the core of this reactor consists of 37 fuel elements of the *MTR* type and six control rods. It is surrounded on three sides by beryllium reflector elements (Fig. 2). The fuel elements are composed of 19 vertically arranged plates of a 90%-enriched U-Al alloy (Fig. 3). The control rods consist of a cadmium absorber section which extends downwards into a fuel section. The control rod drive mechanism is situated beneath the reactor. The control rods move vertically so that the fraction of neutron-absorbing cadmium in the core can be varied. In this way power control is possible with increasing burn-up.

The total charge of an *HFR* core is 5-6 kg of U 235 and permits a maximum burn-up of 20-30%. The present thermal power of the reactor is 30 MW. The reactor core is mounted in an aluminium tank designed for a maximum overpressure of 2.6 atm. Demineralised water pumped through the reactor tank at a rate of 3,000 m³/hr acts as the coolant. The fission heat absorbed in the primary circuit is transferred to a secondary circuit and led off. The water temperature in the primary circuit is approx. 50°C. The reactor vessel is mounted in the lower part of a pool and is sunk into the concrete foundation.

Experimental facilities

The devices containing the irradiation samples are either inserted in the fuel and reflector elements in the core or are installed in their place in order to achieve the highest possible neutron flux density. In principle, therefore, any of the positions shown in Fig. 2 can be chosen in the *HFR*. However, for the time being only the positions immediately adjacent to the active core are of practical importance in this reactor. A longitudinal hole is bored in the beryllium-reflector elements to house the irradiation devices. Those holes not containing irradiation devices are fitted with cylindrical beryllium inserts. The reflector can also be replaced by aluminium filler elements for the installation of irradiation devices. These filler elements have the same outside dimensions as the fuel and reflector elements.

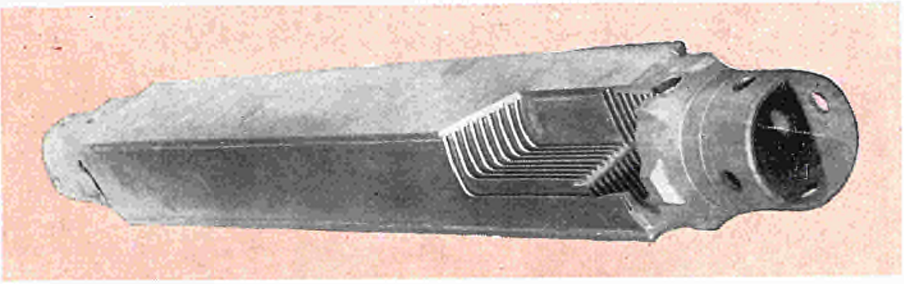


Fig. 3: HFR *fuel element.*

In other reactors, e.g. the *BR 2*, the fuel plates are arranged concentric in the fuel elements so that one or more of these cylinders can be removed and replaced by an irradiation device.

Both the reactor vessel cover and the pool wall contain penetrations for cable and pipe connections leading to the measurement and control station. The insertion of irradiation devices in pressure-vessel reactors and their removal is only possible when the reactor is shut-down.

As well as radiation facilities in the core, a number of materials-testing reactors are fitted with equipment for bringing samples from outside the pressure vessel to within varying distances from the core. The *HFR* vessel contains a rectangular cavity for this purpose at the same height as the reactor core. The irradiation device is situated in the reactor pool.

All materials-testing reactors have in addition several pneumatic or hydraulic transport systems for carrying out short-time exposures with irradiation times of seconds, minutes or hours. These facilities are mainly used for isotope production and activation analysis.

Many factors determine the choice of reactor for a particular irradiation experiment. From a purely technical point of view, the chief ones are as follows:

— neutron flux density,
— nuclear heating,
— size and layout of the irradiation device.

Neutron flux density

Since the neutron flux density is of special interest when planning irradiation experiments, particularly with regard to the choice of the reactor and irradiation sites, a great number of flux density measurement data for various core compositions are held by all reactor centres. These measurements give us the approximate density distribution of the thermal and fast neutrons.

Since the neutron flux density depends on the layout and charging of the reactor core, the number and position of the irradiation experiments, the burn-up of the fuel elements and the position of the control rods—all of these factors which can alter from cycle to cycle—the neutron flux density values given by the individual reactor plants can naturally only be approximations.

Table 2 shows the approximate values of the thermal and fast neutron flux density in the reflector positions of the *HFR* (reactor power 30 MW).

Fig. 4 shows as an example the vertical distribution of the neutron flux density in position G8 in the *HFR*. As the control rods are gradually extracted from the active zone with increasing burn-up during the reactor cycle, the peak of the vertical flux

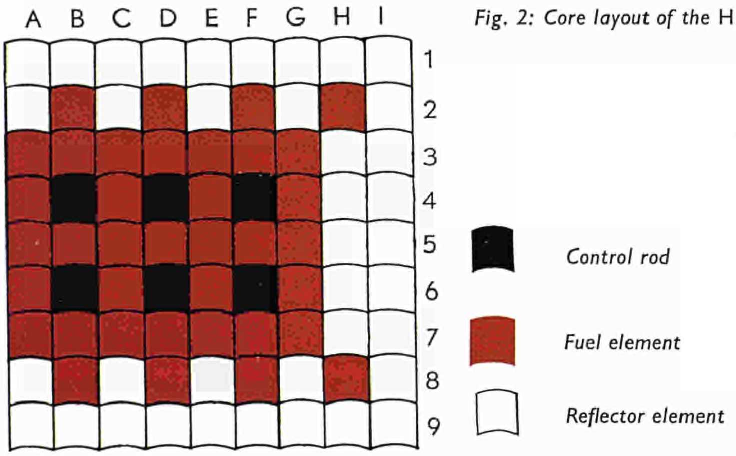


Fig. 2: Core layout of the HFR.

distribution, which is about 10 cm below the middle plane of the reactor core at the beginning of a reactor cycle, shifts slowly upwards.

Theoretically, it should be possible to ascertain from the neutron flux density values the integrated neutron flux densities for longer irradiation times. However, for reasons already outlined, it has been seen that these values are only approximate, so that no exact data can be obtained in this way on the integrated neutron flux densities, henceforward referred to as "fluence". For this reason, all irradiation experiments are provided with flux detectors. Ni 58, Ti 46 and Fe 54 are possible choices as probe materials for measuring the flux density of fast neutrons and Co 59 and Fe 58 for that of thermal neutrons. The given measurements are exact to ± 10%. This value is sufficient for the majority of irradiation experiments.

Table 2: Approximate values for the thermal and fast neutron flux density in the reflector positions of the HFR at 30 MW reactor output.

Reflector position (see Fig. 2)	Thermal neutron flux density *		Fast neutron flux density *	
	Average	Maximum	Average	Maximum
A2 A8	0,45**	0,6	0,3	0,5
C2 C8	0,9	1,2	0,7	1,0
E2 E8	0,9	1,2	0,7	1,0
G2 G8	0,7	1,0	0,5	0,7
H2 H8	0,5	0,7	0,35	0,55
H4 H6	0,55	0,8	0,4	0,6
H5	0,6	0,9	0,4	0,6

* Perturbed neutron flux density measured in the central axis of the so-called "standard experiment". This consists of an aluminium cylinder surrounded by a 2 mm thick stainless steel sheath. It is housed in the 52 mm hole in an aluminium filler element.

** 10^{14}n/cm^2.sec

Nuclear heating

Similarly to the neutron flux density, nuclear heating also depends upon the layout and charging of the reactor core, the number and position of irradiation experiments, the burn-up of the fuel elements and the position of the control rods. Since the nuclear heating can amount to up to 15 W/g, especially in positions with high neutron flux densities, the samples become extremely hot, with the result that this heat must be removed in the case of many irradiation devices. In other instances this effect is used to bring the irradiation samples to a desired temperature.

Irradiation devices

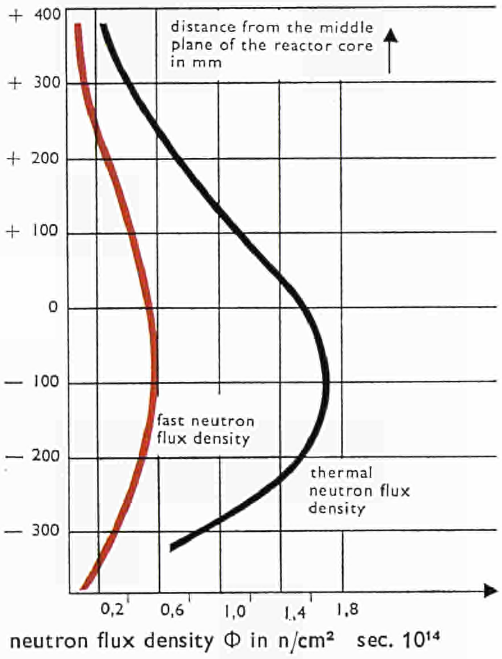
As it is frequently impossible, for purely geometrical reasons, to test whole reactor structural parts or components, attempts are made to irradiate the relevant materials in conditions which simulate as closely as possible those in the reactor in which the material is later to be used. Samples are inserted in irradiation devices for this purpose.

Fig. 5 shows a capsule for the irradiation of non-fissile material samples in a temperature range of 250-750°C. The irradiation temperature for a particular helium/neon gas mixture is determined by the width of the gas gap between the sample-holder and the capsule wall.

To make allowances for the vertical distribution of the nuclear heating, the gas gap is graduated by using a conical shape for the sample-holders.

Furthermore, the temperature changes brought about by nuclear heating are evened out by eight independently controllable heaters. If necessary the helium/neon gas mixture, and therefore the thermal conductivity in the gas gap, can be changed during irradiation. The samples are inserted in four graphite holders, each having an outer diameter of 53 mm. The Thermocoax heating coils are fitted in suitable spiral indentations on the outer circumference of the sample-holder. A stainless steel protective tube is placed over them. The sample-holders are held inside a cylindrical stainless steel capsule by means of centring cams and connecting pieces. The power supply for the heaters and thermocouples is led through holes drilled in the sample-holder. Continuous

Fig. 4: Vertical distribution of fast and thermal neutron flux density in position G8 of the HFR.



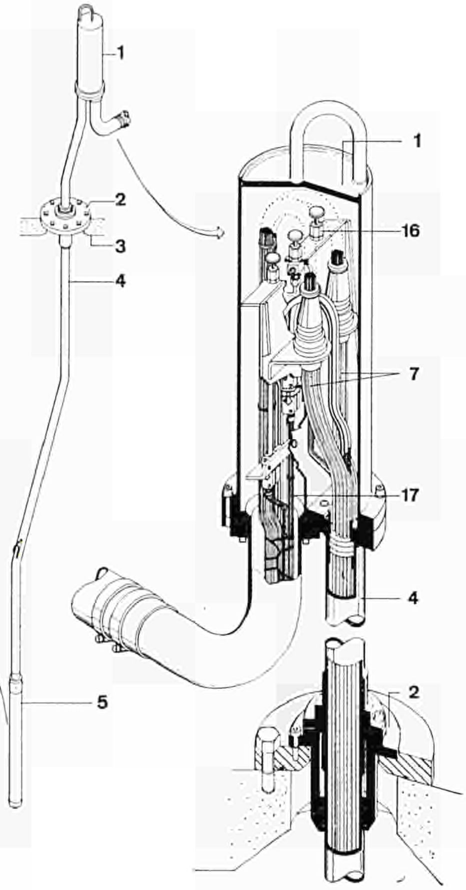


Fig. 5: Capsule for irradiating non-fissile material in the 250-750°C temperature range.

1 *Connecting lug*
2 *Penetration*
3 *Reactor vessel*
4 *Extension tube*
5 *Capsule*
6 *Thermocouples and gas line*
7 *Heater power supply*
8 *Heater connecting block*
9 *Sample-holder*
10 *Gas gap*
11 *Heating coil*
12 *Heater sheath*
13 *Samples*
14 *Centring cam*
15 *Connecting piece*
16 *Shut-off valve*
17 *Thermocouple compensating lines*

observation of the samples and the capsule temperature is ensured by a number of thermocouples, so arranged that even axial thermal gradients can be picked up. The capsule is inserted in the hole in a reflector element for irradiation purposes.

Fig. 6 shows a capsule for irradiating nuclear fuels. Rod powers of between 400 and 1,200 W/cm can be achieved with this irradiation device. As can be seen from the figure, the fuel rod is housed in a cylindrical stainless steel container, nine-tenths filled with NaK. The space above this level is filled with helium. A second cylindrical stainless steel container surrounds the first one and forms an annular gas gap.

The composition of the gas in this gap can be varied between 100% helium and 100% nitrogen, so that the thermal conductivity in the gap and hence the cladding temperature, which is measured with several thermocouples, can also be varied. Naturally, the resulting control range is dependent upon the rod power and reaches a maximum of 200-250°C. Capsules of this type have been used in various materials-testing reactors. In the *HFR*, as shown in Fig. 7, they are arranged on a movable mechanism in the reactor pool. In this way it is possible to keep the linear rod power constant during the irradiation period by varying the distance between the capsule and the core. Furthermore, thermal cycling can be easily simulated this way.

Once irradiation devices have been installed in the reactor, they become components of it for as long as irradiation continues. Such devices must therefore be closely examined as early as the development stage as to whether they satisfy the very stringent safety conditions inside the reactor. In addition, all irradiation devices are fitted with a safety system which gives off a signal should radiation conditions suddenly change and releases the reactor's control rods if critical conditions suddenly occur, thus causing shut-down.

Dismantling and post-irradiation studies

In almost all installations the experimental facilities offered by high-flux reactors are backed up by equipment for carrying out post-irradiation checks on irradiated material samples. In order to do this, the following procedure is adopted: after a decay time, which is largely dependent upon the activity of the irradiation device, the capsule is placed in a dismantling cell in which the irradiated samples can be cut open by means of remote-control tools. The actual examination is then performed in the hot laboratories.

It is by now evident how time-consuming and costly irradiation experiments in materials-testing reactors are. In general it can be said that, for the development of a new irradiation device, a period of at least one year must be allowed between the planning of the experiment and insertion in

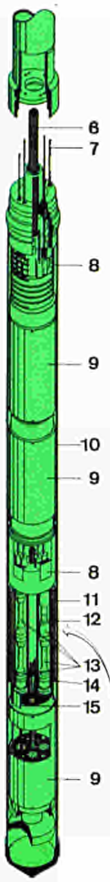


Fig. 6: Capsule for irradiating nuclear fuels.

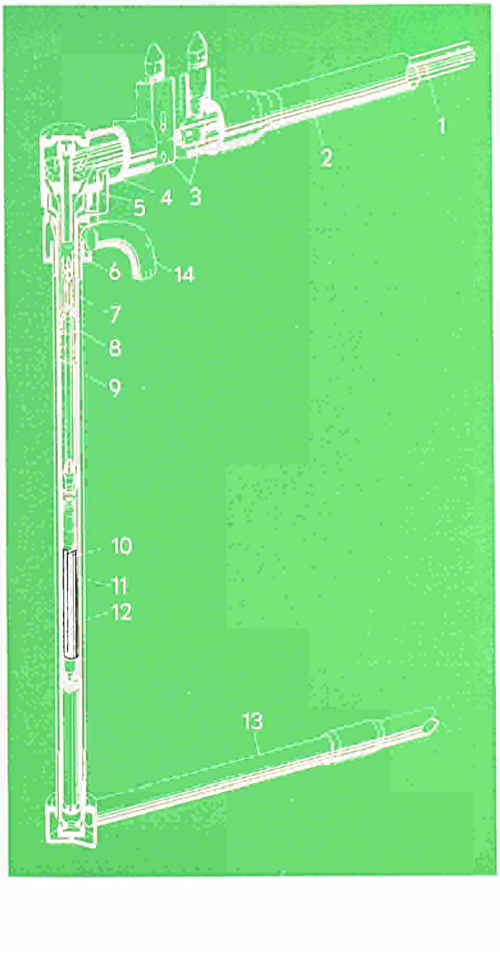
1 *Thermocouples*
2 *Extension tube*
3 *Mouldings for gas ducts*
4 *NaK filling nozzle*
5 *Gas-tight penetration*
6 *NaK level*
7 *Outer capsule*
8 *Gas gap*
9 *Inner capsule*
10 *Sample rod*
11 *Thermocouple*
12 *Cooling water sheath*
13 *Cooling water inlet*
14 *Cooling water outlet*

the reactor. However, in many cases rechargeable capsules already available can be used or a new irradiation device built from existing plans. The time between planning and insertion is then cut to approximately six months.

Between two and three years must be estimated for building irradiation loops, there being a risk that after such a long construction period the irradiation results may not prove so interesting to the experimenter as when the order was originally placed.

The time spent in the reactor naturally depends on the fluence required. Materials-testing reactors work continuously in so-called reactor cycles. Thus, for instance, irradiations are carried out for six-week periods in the *HFR*.

The reactor is switched off twice during this time. The first shutdown lasts one or two days and is mainly used for core charging and installation of simple irradiation devices. An irradiation period of 16-18 days follows this shutdown and during this time the reactor works at full power. There then follows another shutdown of four or five days when routine maintenance work on the reactor and its instrumentation are carried out, in addition to core charging and insertion of irradiation devices.



The maximum fast fluence which can be achieved in the *HFR* during a 16 day cycle is 10^{20}n/cm^2. Depending on the irradiation position adopted, this value is reached after three to four days in the *BR 2*.

The cost factor

The cost factor should not be underestimated. Although a considerable sum of money, generally speaking, is set apart for irradiation experiments in large reactor development programmes, many, and in some cases urgently required irradiation projects break down over this point. Simply instrumented irradiation devices cost in the region of $ 5,000-10,000. This price rises extremely quickly if the irradiation devices are instrumented so as to permit the measurement of certain physical characteristics, e.g. creep behaviour, during irradiation within the reactor. Loops used for simulating certain reactor conditions —a type in increasing demand today—can run to between $ 500,000 and one million.

The fact that maintenance costs are very high—$ 2.5 million in the case of the *HFR*—accounts for the high irradiation costs. For instance, a position with a fast neutron flux density of approximately 1 . 10^{14}n/cm^2.sec for a 16 day cycle costs about $ 11,000 in the *HFR*. This price is equivalent to a fluence of about 1 . 10^{20}n/cm^2.

It should also be mentioned that the performance of irradiation experiments requires teamwork, wide knowledge of the extremely varied fields of reactor technology, such as neutronics, technical thermodynamics and materials and, last but not least, considerable dexterity.

Even though there are in existence today prototype reactors in which whole components can be tested, high-flux materials-testing reactors will not lose their importance in the years to come.

This will be especially true if irradiation devices are developed with which changes in certain physical characteristics can be measured directly during irradiation.

(EUBU 6-16)

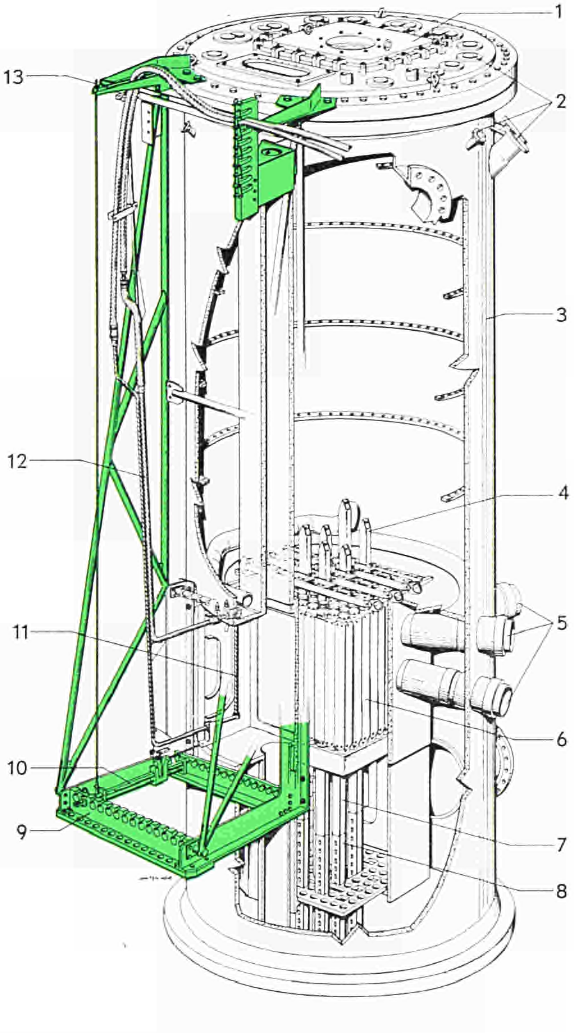


Fig. 7: HFR *irradiation device in the reactor pool with transport mechanism.*

1 Cover
2 Penetrations
3 Reactor vessel
4 Control rod—absorber section
5 Horizontal beam ports
6 Reactor core with fuel and reflector elements
7 Control rod—fuel section
8 Control rod—connection to drive mechanism
9 Irradiation device in the pool
10 Drive mechanism for EXOR*-type irradiation capsules*
11 EXOR *capsule*
12 Extension tube with supply ducts
13 Lug for the drive mechanism crank handle

Measuring burn-up in uranium fuel elements

MARCELLO BRESESTI, *Nuclear Chemistry Laboratory, Chemistry Department, Ispra Establishment of Euratom's Joint Research Centre*

A KNOWLEDGE OF burn-up, or the amount of heat generated by fission in a fuel element irradiated in a nuclear reactor, is of basic importance to any research programme on fuel elements. In fact, many of the phenomena which occur in a fuel element during irradiation—grain growth, swelling, emission of fission gas and migration of fission products—cannot be interpreted without a knowledge of the burn-up as well as data on the temperature. The former is also important in reactor economics, since a fuel element must produce a certain quantity of energy before it is replaced.

In the irradiation of natural or U 235-enriched uranium fuels, energy is generated by the fission of the two uranium isotopes U 235 and U 238, and of the plutonium isotopes Pu 239 and Pu 241 which are formed by successive captures of neutrons from U 238, as illustrated in Figure 1.

U 235 is fissionable with thermal neutrons, but U 238 only with neutrons of energy above 0.5 MeV. In thermal reactors the number of U 238 fissions is very small compared with that of U 235 fissions. The contribution of Pu 239 and Pu 241, which are fissioned by thermal neutrons, depends on the enrichment of the fuel and the burn-up of the element.

Burn-up of a fuel element can be expressed in various units, the most common being the MWd/ton, defined as the megawatt days of heat produced in 2,000 lbs (907.20 kg) of fissile and fertile materials, originally present.

A simpler method is that of expressing burn-up as a percentage of fuel atoms fissioned. However, to be usable, this latter expression for burn-up must be converted into heat units. The total average energy produced in a fission of U 235 is around 200 MeV, as shown in Figure 2.

The energy of the neutrino is lost, but this loss is compensated by the energy of the gammas produced by the neutrons captured in the reactor materials. An energy of some 200 MeV is also generated by the fission of a U 238 or Pu 239 atom.

Methods of measuring burn-up

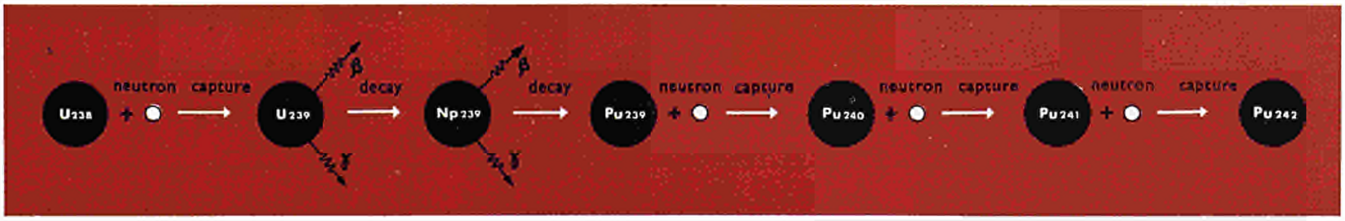
Depending on the required accuracy, the enrichment of the fuel, the irradiation characteristics, etc. there exists a number of choices for the method of measuring burn-up. The following methods are the most common:

- determining the radioactive or stable fission products;
- determining the isotopic ratio U 235/U 238;
- determining the plutonium isotopes;
- determining the integrated neutron flux on the surface of the element.

Determining fission products

When the number of atoms of an isotope formed by fission is known, the number of fissions can be obtained directly by using the known fision yield (number of atoms produced for each fission) of the isotope being measured. Fission products can be

Fig. 1: Formation of plutonium isotopes from U-238 by successive neutron captures.



divided into *radioactive* and *stable* fission products.

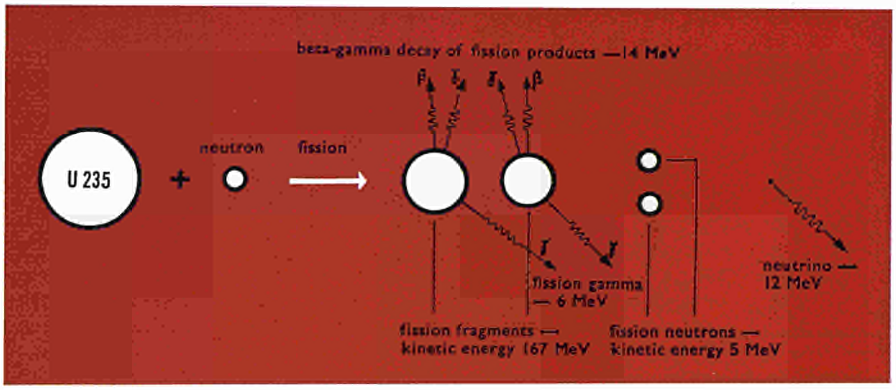


Fig. 2: Energy generated during fission of a U 235 atom.

Measuring radioactive fission products

Radioactive fission products are determined by measuring their activity. In order to reduce the corrections for decay during irradiation to a minimum, it is necessary to measure fission products whose half-life is long in comparison with the irradiation time.

The most frequently used isotopes for measuring burn-up are Cs 137 (half-life 30 years), which is in equilibrium with the daughter element Ba 137 m, and Ce 144 (half-life 384 days), which is in equilibrium with the daughter element Pr 144.

Another interesting isotope is Ru 106 (half-life 365 days), which is in equilibrium with the daughter element Rh 106. With this isotope there are marked differences in the fission yields for U 235 and Pu 239. This difference can be used to determine the individual contributions of U 235 and Pu 239 to the total number of fissions.

The simplest method of measuring radioactive fission products is by gamma-spectrometry using an NaI or germanium crystal spectrometer.

Germanium detectors have better resolving powers and enable the photo-electric peaks of various nuclide gammas to be pinpointed among the mixture of fission products. Figure 3 shows the gamma spectrum of a fission product mixture obtained from a germanium crystal spectrometer connected to a multi-channel pulse height analyser.

Gamma spectrometry offers the additional attraction of measuring burn-up without destroying the element, the gammas emitted from an irradiated fuel element being measured directly. A fuel element placed in a hot cell or behind a suitable screen can be moved in such a way that its various parts pass in turn in front of a collimator so that a thin gamma beam is sampled. Figure 4 shows an installation of this kind in which a germanium crystal spectrometer is used. By this method, which avoids the need to take fuel samples from the element and treat them radiochemically, it is easy to obtain information on the relative distribution of the fission products and hence the burn-up along a rod. The apparatus can also be calibrated so as to allow the burn-up to be measured absolutely. A simple calibration method consists in using other methods to determine the absolute burn-up value of a sample from the measured rod with respect to its various parts.

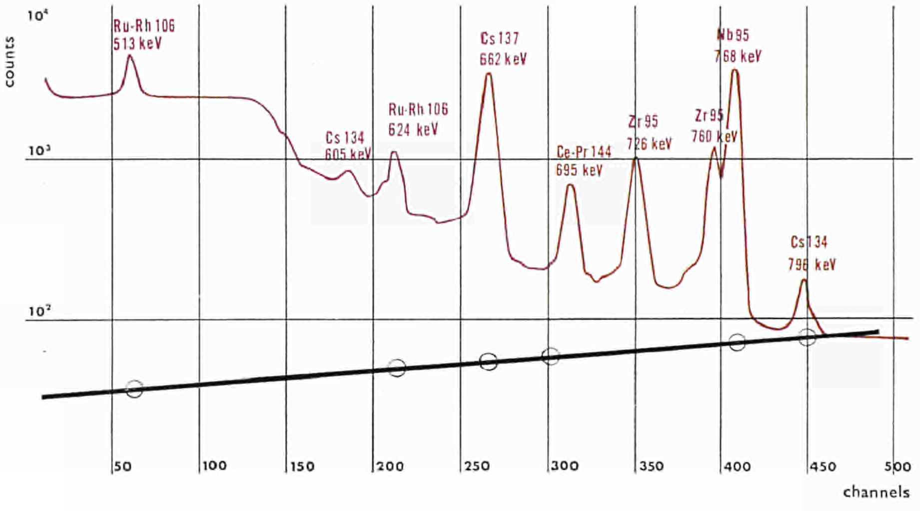
Measuring stable fission products

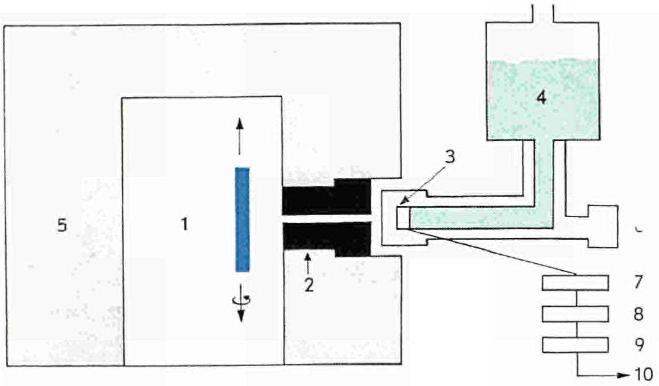
In measuring stable or very long-lived fission products it is possible to avoid the errors which arise from the uncertainties of the decay schemes of radioactive nuclides and the corrections to be introduced to allow for decay.

After purification these fission products can be assayed by radiochemical techniques of analytical chemistry (e.g. T 99—half-life 2 . 10^5 years—can be determined by colorimetry) or activation methods (e.g. La 139 can be determined by neutron activation). However, a greater degree of precision can be attained by mass spectrometry, using the isotopic dilution technique. The most interesting stable isotope for determining burn-up is Nd 148, which possesses very similar fission yields for both U 235 and Pu 239.

In this case the isotope dilution method consists of adding to the fuel solution under examination a known quantity of neody-

Fig. 3: Gamma spectrum, from a germanium detector, of a fission product mixture about 18 months after the end of the irradiation.





1 *Fuel element*
2 *Collimator*
3 *Detector*
4 *Liquid nitrogen*
5 *Shielding*
6 *Ion pump*
7 *Low noise preamplifier*
8 *Amplifier*
9 *Multichannel analyser*
10 *Spectrum (Fig. 3)*

Fig. 4: Apparatus for measuring gamma emissions from an irradiated fuel element—germanium detector.

mium enriched with Nd 150, carrying out suitable radiochemical purification of the neodymium and finally determining the quantity of Nd 148 present by measuring the ratio between the number of Nd 148 and Nd 150 atoms.

The isotopic ratio is determined by mass spectrometry, the working principle of which is illustrated in Figure 5. The neodymium is ionised and the ions accelerated and collimated through a slit. The ions thus formed are separated according to mass-to-charge ratio by the magnetic field set up by an electromagnet. These mass-analysed beams are separately captured on a collector at various intensities of the magnetic field. The ionic current at the collector, after amplification, is displayed on a recorder and from the peaks plotted it is possible to determine the ratio between the two isotopes.

Determining the isotopic ratio U 235/U 238

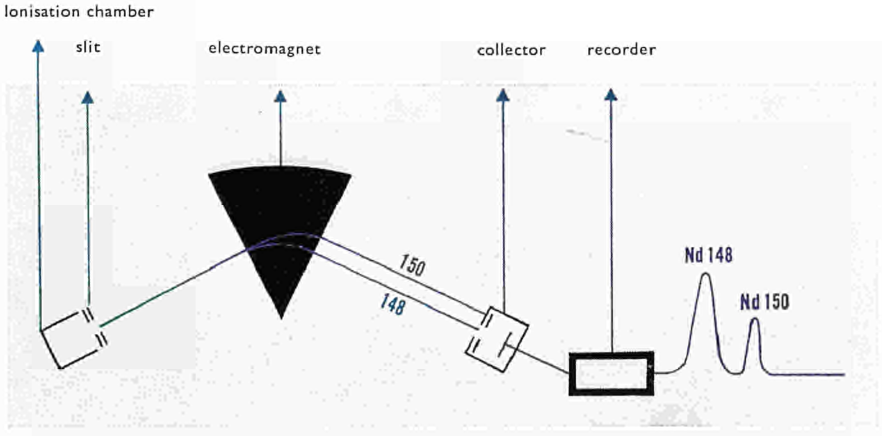
An immediate method of determining the number of U 235 fissions is to measure the amount of U 235 destroyed during irradiation. This can be determined from the variation in the isotopic ratio U 235/U 238 during irradiation. In calculating the number of fissions it should be remembered that U 235 can also be destroyed by neutron capture; the relation between captures and fissions for U 235 is well known, however.

During irradiation the quantity of U 238 destroyed is usually negligible. When the number of U 235 fissions is known, it only remains to determine, either experimentally or by calculation, the contribution of the Pu 239 and U 238 fissions.

The isotopic ratio can be measured by a mass spectrometer, the operating principle of which is shown in Figure 5.

It can also be measured by emission spectrometry. The working principle of an emission spectrometer is illustrated in Figure 6. The light from the excitation source enters the spectrometer through a slit, is reflected by a concave mirror acting as a collimator lens, and is then transmitted onto a grating. The light is broken into monochromatic radiations by the grating, returns to the mirror, and is again collimated onto a photographic plate in the form of spectral lines. From the wavelength of these lines it is possible to pinpoint the isotope emitting the light. The

Fig. 5: Diagram of a mass spectrometer.



intensity of the lines can be used to obtain a quantitative determination of the isotope.

Determining plutonium isotopes

The proportion of Pu 239 fissions to other fissions depends on the degree of enrichment in U 235 of the fuel and on the irradiation time.

In the case of long irradiations of natural uranium with burn-ups of more than 10,000 MWd/ton, more than 50% of the fissions can be due to plutonium. Plutonium assay can therefore be an important means of backing up measurements such as that of the isotopic ratio U 235/U 238, which supply information only on the fissions due to U 235. It should be remembered that knowledge of the plutonium content is also important for problems of reactivity variation in irradiated fuels.

This can be determined more precisely by isotopic dilution and mass spectrometry.

The total plutonium content can also be determined by ordinary chemical analysis.

The isotopic composition of plutonium can also be measured by alpha and gamma spectrometry. For example, a germanium detector can be used to measure gammas of Pu 239 and of the two daughter-elements of Pu 241 – U 237 and Am 241. The gamma spectrum shown in Figure 7 was obtained on a plutonium source separated from irradiated uranium.

Determining the integrated neutron flux on the element surface

A method which appears quite convenient, inasmuch as it does not entail sampling the fuel, is based on measuring the integrated neutron flux on the surface of an element, using an activation detector.

The detector normally used is cobalt, either pure or as an aluminium alloy with a low cobalt content. The cobalt is made of a single isotope, Co 59, which is converted into Co 60 by capture of a neutron; this last element decays emitting beta and gamma radiations, and has a half-life of 5.27 years. By measuring the activity of Co 60 in the detector, which is a wire fixed to the surface of the fuel element, the value for the integrated neutron flux versus time can be obtained if the capture cross-section is known.

The neutron flux to which the element is subjected, however, is different from the surface flux, since the thermal neutrons inside the element, which are the most important in the fission process, are heavily absorbed, with consequent flux-depression. But nuclear codes can be used to calculate the variation in the flux inside and near the element and, hence, the ratio between the surface flux and the average flux in the sample can be obtained. Figure 8 gives an example of such a calculation for a UC element to be irradiated in the central channel of the *ISPRA 1* reactor. When the average neutron flux is known, the number of fissions can be calculated from the fission cross-sections.

An uncertainty of this method is the cross-section values, since they depend on the energy of the neutrons, the distribution of which is not as a rule well known.

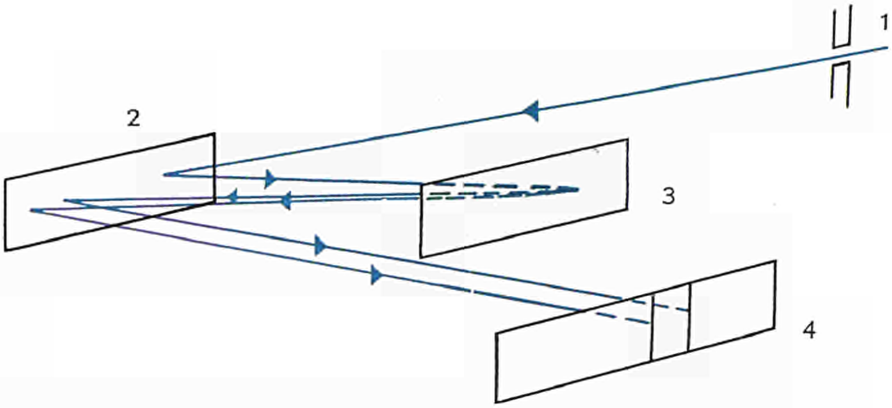
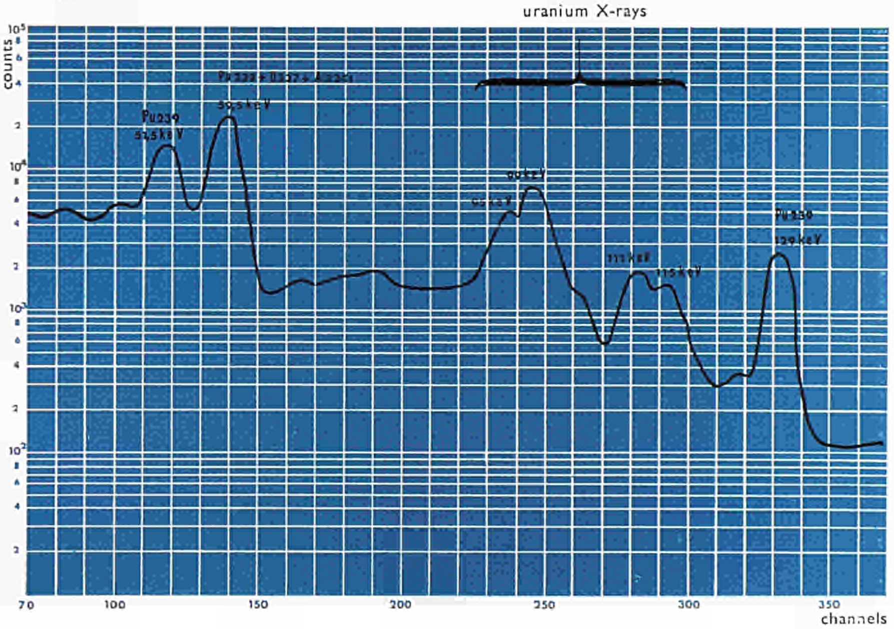


Fig. 6: Diagram of an emission spectrometer. 1 Slit 2 Mirror 3 Grating 4 Plate

Fig. 7: Gamma spectrum, with a germanium detector, of plutonium separated from irradiated uranium.



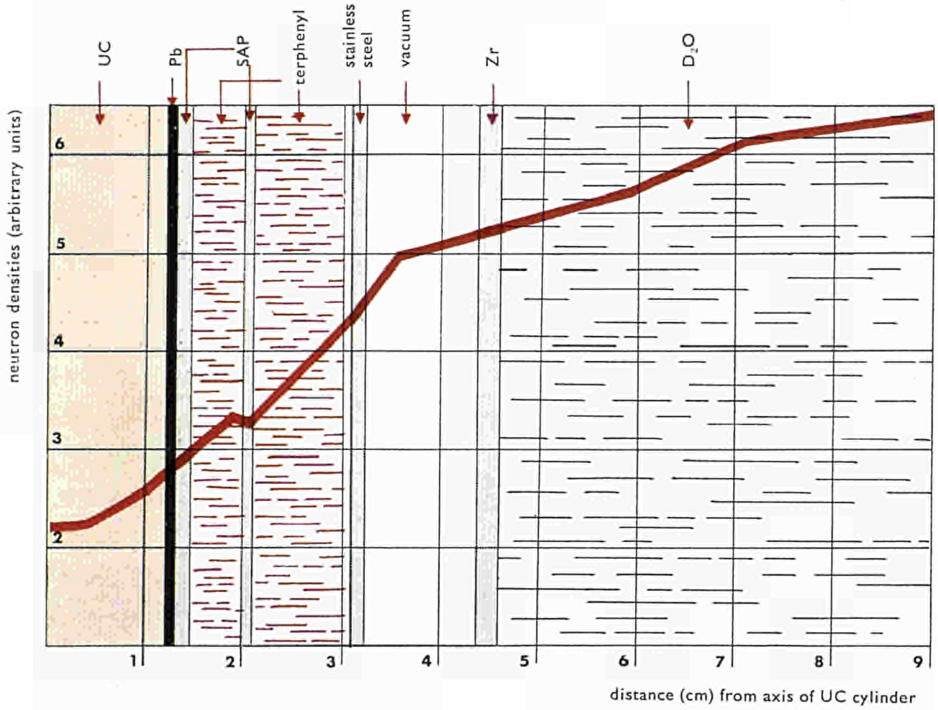
Conclusions

Final comparison is facilitated by dividing the methods into a non-destructive and a destructive group.

The first group includes the direct measurement of the gamma activity of an irradiated fuel element, as well as surface flux measurements. These methods have two major advantages. It is not necessary to take samples from the element or to carry out radiochemical processing, so there is much less work than with the other methods. In addition, the gamma measuring system shown in Figure 4, or suitably placed neutron detectors, can be used to measure burn-up along the entire axis of the fuel element.

In the case of destructive methods, since it is impractical to dissolve a whole fuel element and the number of samples tested must be limited, representative samples should be chosen, i.e. samples which yield information which can be applied to the whole element through calculations or the use of other available experimental data. When this choice is made, it should be remembered that diffusion of fission products and plutonium may lead to error in the determined burn-up values. The destructive methods are usually more accurate. Analyses by mass spectrometry, such as measuring the isotopic ratio U 235/U 238 and the Nd 148 content, give a high degree of accuracy.

Fig. 8: Calculation of neutron densities by nuclear codes.



In conclusion, it may be said that the non-destructive methods present the advantages of simpler and faster operation, while some destructive methods offer the attraction of greater accuracy.

The first determinations of burn-up on uranium carbide elements were performed at the Ispra Establishment by measuring the radioactive fission products Cs 137, Ce 144 and Ru 106. Ispra is at present developing non-destructive methods based on measurement of the neutron flux and direct measurement of gamma activity on the fuel element by means of a germanium spectrometer. (EUBU 6-17)

Calculating the world's uranium resources

JOHAN BRINCK, *Euratom Supply Agency*

The formation of uranium deposits (or of any mineral deposits for that matter) is not a purely random phenomenon, but one which can be described in mathematical terms. On the basis of this mathematical model it is possible to estimate, for any given grade of enrichment, the size of potential uranium ore reserves. The conclusion is that these are very large indeed.

IN 1963, EURATOM published a report on "the problem of uranium resources and the long-term supply question" (Report N° EUR 414e). There existed no short-term problem at that time, but from the predicted growth rate of nuclear energy, it could be foreseen that after 1975, the expected uranium requirements would exceed the economic production capacity of the reserves which would be left by then, it being assumed that reserves are economic only if they are exploitable at less than $ 8-10/lb U_3O_8.

Since 1963, the predicted growth rate has steadily been revised upward. One of the latest estimates of the *USAEC* has more than doubled the originally predicted growth rate until 1980. In view of technical improvements in the fuel cycle the consequent increase of uranium requirements has been estimated at 50% only, but with ore reserves at the same level of some 500,000 tonnes of uranium metal, it is clear that the long-term supply situation must have deteriorated during the four years between the two estimates.

The proposed solution of this problem, then and now, has been to launch a vigorous exploration programme for new uranium deposits exploitable at a similar price of $ 8-10/lb U_3O_8. On the basis of geological evidence it is generally believed that the existing uranium reserves could easily be doubled by this method, and possibly trebled.

Putting a price-tag to mineral resources

This recommendation, the validity of which has been demonstrated over and over again by the mining industry for almost all mineral commodities, brings us to the subject of this article, which links the purely factual concept of *mineral resources*, that is to say the total tonnages of an element available at different concentrations in the upper part of the earth's crust, with the economic concept of *ore reserves*, i.e. the tonnages extractable at a given cost. Through this link, ore reserves can be seen as mineral resources with a price-tag attached.

Any long-term mineral supply problem can therefore be divided into two main components:

— what is the natural availability, at different concentrations, of the mineral product?

— what is the price for extracting it from these resources?

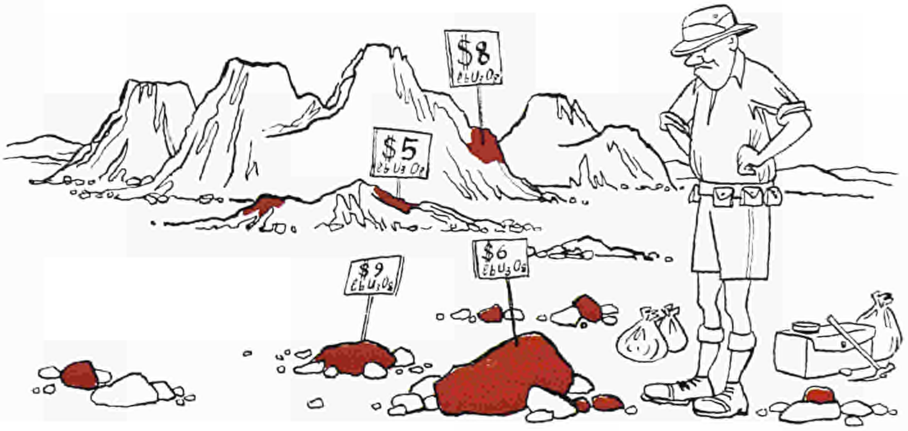


Fig. 1: Map showing the distribution of uranium provinces in the free world. The dark surfaces represent workable deposits (i.e. at a price of \$ 8-10/lb U_3O_8) and the lighter surfaces potential reserves.

Normally, the higher the concentration of the desired product in a mineral deposit, the higher are its chances of becoming exploitable as an ore. However, differences in prospection, mining, processing and marketing costs, resulting from differences in the size, type and location of mineral deposits, permit the profitable mining of a fairly wide range of concentrations at a more or less equal cost-price.

To stay with uranium as an example, most of the reserves known today, exploited or exploitable at a price of \$8-10/lb U_3O_8 or less, have an average content of some 1,500 g uranium per tonne of ore, but range in grade from some 900 to 3,000 g/tonne.

However, the very large South African reserves, owing to the fact that uranium is recovered here as a by-product of the gold-mining industry, can be exploited at this price from ores containing as little as 200 g uranium per tonne. Several very large uranium deposits are known with concentrations between these extremes, but at the present state of ore treatment technology and on the basis of current estimates on the future increase of uranium demand, these can be considered only as potential reserves at the very best. An example is the uranium deposit at Billingen-Mountain, Sweden (see Fig. 1), where an estimated geological reserve of one million tonnes of uranium in bituminous schists, with an average content of 300 g uranium per tonne, has been located. The present production costs (\$ 15-20/lb U_3O_8) place this deposit in the potential reserves.

Distribution and size of deposits . . .

This short description, which could be replaced with an equivalent for almost any mineral commodity, already shows some rather important trends in the distribution and size of mineral resources. At the higher enrichments, the individual mineral deposits are markedly smaller than at the lower enrichments. If for uranium ore deposits in the range of 900-3,000 g/tonne of uranium an individual reserve containing 10,000 tonnes of uranium metal may already be considered as a large deposit, at the lower enrichments we find individual concentrations for which the uranium metal content can be expressed in hundred thousands or millions of tonnes.

Taking the upper two-and-a-half kilometres of the emerged part of the earth's crust, with an estimated average uranium concentration of some 3 g of uranium per tonne of rock as the ultimate resource, the uranium metal contained therein would be some three million million (3×10^{12}) tonnes.

Another important observation which can be made with respect to the distribution of uranium ore deposits (and a similar observation could be made for other ores) is the fact that, although uranium deposits may be found almost anywhere, the real large contributions to the total uranium reserves are made by relatively few regions of the world. Such "uranium provinces" are for instance the Canadian Precambrian shield area, the Colorado-plateau province in the United States, the Rand province in South Africa, the Hercynian province of Europe. Looking to one uranium province in particular we find that its reserves are normally located in several, often geologically dissimilar mineral districts in which the ore deposits are of a very similar type. As a rule, ore deposits themselves are made up from several "ore shoots", surrounded by leaner or barren material, and even an individual ore shoot may show relatively high contrasts between mineral concentrations in its different parts. This indicates that the distribution of ore deposits (i.e. element concentrations of a given grade and size) in the earth's crust is not a completely random phenomenon.

. . . according to a pattern

Both mining exploration and geochemical

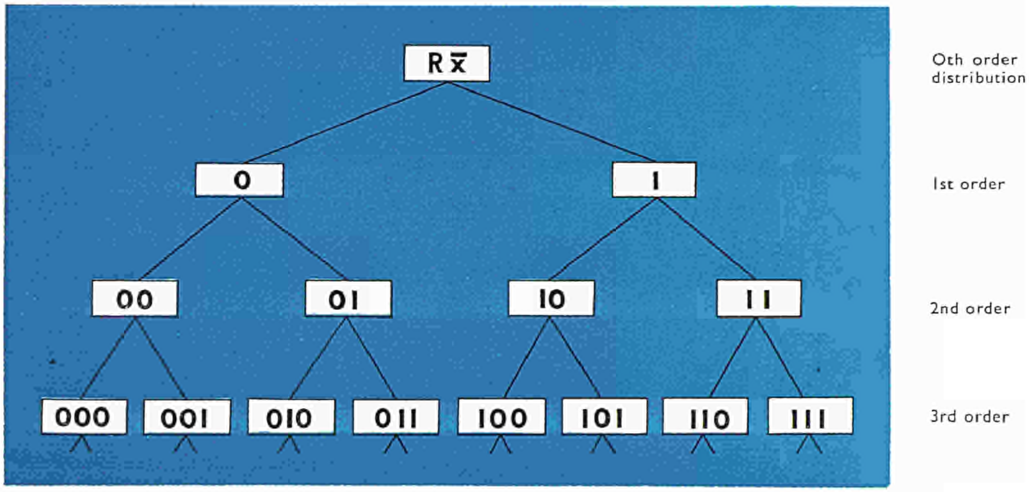


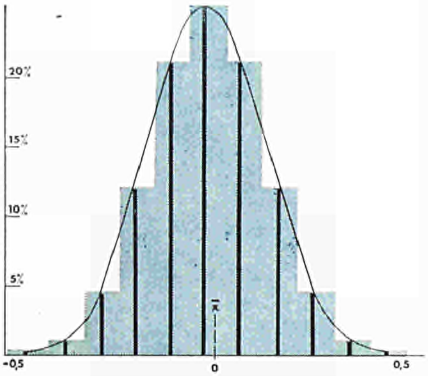
Fig. 3: Development of a log-binomial distribution down to the third order, showing the relation between terms of equal size and grade, such as $\boxed{011}$, $\boxed{101}$ *and* $\boxed{110}$. *Upon further subdivision, it can be shown that high concentrations, which will tend to group together, will occur over the whole length of the distribution.*

surveys have shown decisively that the concentrations of a given element in a large series of different samples from a given geological system show a very similar distribution pattern. Here again the relatively rare, highest grades tend to group together and are linked with the average grade through a range of increasingly abundant intermediate concentrations. Plotting the frequencies of the different concentrations, found from such a survey, against their logarithms, we find a curve which tends to fit the normal or Gaussian probability curve (see Fig. 2). This most universal observation appears to be independent of whether each sample represents a unit of millions of tonnes of rock, thousands of tonnes from a mineral deposit or only a few tonnes of material from an ore shoot. Therefore, at this scale we may say that element concentrations in the earth's crust tend to be *log-normally* distributed.

On the evidence of these observations, it seems that such element concentrations in the earth's crust result from a logarithmically-graded enrichment and depletion process (basically comparable to the isotopic separation process in a gas-diffusion plant, for instance). Through this process there exists a definite relationship between all possible concentrations of an element in a given geological environment.

Fig. 2: Tenth order log-binomial frequency disribution. If the order of the distribution is increased, the curve drawn through the plotted frequencies will tend to become the normal or Gaussian probability curve.

$q = 0.1 \quad \bar{x} = 1$



It is evident from geological observations that not all parts of the earth's crust could have been subjected to this process at the same time or to the same degree. Furthermore, it can easily be seen that the major geological processes, such as igneous activity, erosion, sedimentation and solution-precipitation, which on good evidence are held responsible for the formation of most ore deposits, with equal ease destroy formerly existing mineral concentrations.

The present distribution of element concentrations can therefore be seen as the net effect of the continuous rearrangement of the crust's material, after some three to four thousand million years of geological history. During this time, the material balance (i.e. the average concentrations of the elements) has remained essentially constant.

The range of concentrations of an element in a series of weighted samples is a measure of the net efficiency of this process in the sampled geological system; it depends on the physico-chemical characteristics of the element, as contrasted with those of all other elements in the system and their respective concentrations. For equal sample weight, a wide range indicates a high separation efficiency, a narrow range a low one.

A mathematical view of the pattern

An admittedly crude, but convenient way of visualising the interrelation between element concentrations in different parts of a geological system can be found in the *log-binomial* distribution, which becomes *log-normal* for an environment of infinite size.

For a given average concentration $\bar{x}$ of an element in a volume of rock R, the average net efficiency q ($0 \leq q \leq 1$) of the separation process, expressed as a fraction of $\bar{x}$, indicates the proportion of the element's available atoms in a volume 1/2 R, which as a result of one imaginary separation step will move from the depleted half of the volume to the enriched half.

Thus:

$$R . \bar{x} = \tfrac{1}{2} R . \bar{x} (1 - q) + \tfrac{1}{2} R . \bar{x} (1 + q)$$

and

$$R.\bar{x} =$$

$$\tfrac{1}{4} R.\bar{x} (1 - q)^2 + \tfrac{1}{4} R.\bar{x} (1 - q) (1 + q) + \tfrac{1}{4} R.\bar{x} (1 + q) (1 - q) + \tfrac{1}{4} R.\bar{x} (1 + q)^2$$

and so on;

or in general:

$$R.\bar{x} =$$

$$\sum_{k=o}^{n} \frac{{}^{n}C_{k}}{2^{n}} . R.\bar{x} \quad (1 - q)^{n-k} \; (1 + q)^{k}$$

n denotes the order of the log-binomial distribution, which becomes log-normal for $n = \infty$. Giving each depletion step $(1-q)$ the symbol 0 and each enrichment step $(1 + q)$ the symbol 1, the four terms found upon the second subdivision of R are completely defined by writing them respectively as:

$$\bar{x}R = \boxed{00} + \boxed{01} + \boxed{10} + \boxed{11}$$

These symbols are the consecutive numbers of the terms minus one, written in the binary system.

It can be seen from figure 3 that the equal size and grade concentrations 01 and 10 have been obtained respectively by an enrichment step in the depleted volume 0 (first term of the first subdivision of R) and a depletion step in the originally enriched volume 1 (second term of the first subdivision of R). Thus these equal concentrations are not related through a direct link between each other, but only through $R\bar{x}$.

	Total uranium resources in tonnes	Average grade in g/tonne U	Average metal content per deposit in tonnes U
Known reserves	500 000	1 500	4 000
Indicated resources	1 900 000	1 000	10 000
	7 500 000	670	25 000
	28 000 000	450	62 500
	100 000 000	300	150 000

Table I: Probable world uranium resources.

Through this numerical representation it can be shown that, upon division of R in ever smaller parts, high concentrations, with a tendency to group together, are generated over the whole scale. Translated to the three-dimensional geological environment, a process of this type would result in a regionalisation of high concentrations in mineral provinces, districts and ore deposits. Although not directly comparable, the grade contrasts between the consecutive terms found by this numerical representation show a remarkable resemblance to those found in some vein type ore deposits (see fig. 4).

It is also clear that, in a given environment, for each sample size only one sample with maximum enrichment could exist. Knowing this enrichment and the size of the sample, in respect to the average element concentration and the size of the environment, it would be possible to predict all other probable concentrations of lower grade enrichments.

Unfortunately, nature has not clearly marked the sample and environment boundaries of our model and, partly for this reason, statistical evaluation of mineral survey data can more conveniently be done by using the different formulae developed for the log-normal distribution (which is in fact the log-binomial distribution for an infinitely large environment). By choosing a sample size, which in respect to the sampled environment can be considered small (n is large), the results would be approximately the same.

Inferring the size and grade of reserves

With the log-normal distribution, knowing an element's average concentration $\bar{x}$ and its specific mineralisability α (essentially the log-normal equivalent of q), we can calculate the probability of occurrence of all possible concentrations of a given size and grade for any given geological environment.

Assuming that the distribution and individual size and grade of the uranium ore reserves are valid and significant mineral survey data, we can not only calculate the apparent specific mineralisability α of uranium in the accessible part of the earth's crust, but also all uranium resources which must have been generated by the formation of these reserves.

On the basis of measured total uranium reserves of 500,000 tonnes of uranium in exploitable and already exploited deposits (the low-grade South-African reserves being excluded), figure 5 shows all possible uranium metal resources as a function of their average metal content and concentration per mineral deposit.

We know that lower than average ore-grade deposits, to be of economic interest, should normally have a higher than average uranium metal content, as this makes it possible to carry out mining operations on a larger and therefore more economic scale. To illustrate this point we may again use the example of the Swedish uranium deposit in the Billingen District. At present, for an annual production of some 100 tonnes of uranium, concentrates can be produced from this deposit at a price of the order of \$ 15/lb U_3O_8. Increased production to several thousands of tonnes annually and further improvement of the extraction process could almost certainly bring this price down to our guiding price of \$ 8-10/lb U_3O^8. However, as long as sufficient higher-grade reserves could be found to sustain the demand, the competitivity of this deposit would remain doubtful.

Assuming arbitrarily that each lowering of the uranium concentration to 2/3 of the average current ore grade would have to be compensated by an increase of the metal content of a mineral deposit to 2.5 times the average metal content of the current ore deposits, we find the probable uranium resources, shown in Table I, which with increasing demand could largely become ore at concentrate prices of the order of \$ 8-10/lb U_3O_8.

It should be clear that this is only a minimum estimate, as only the *free world's* measured ore reserves have been used in the calculations, but for an environment comprising the *whole dry land surface* of the earth to a depth of 2.5 km. Furthermore each new ore discovery, increasing either the average grade or average size, would increase all other estimates proportionally.

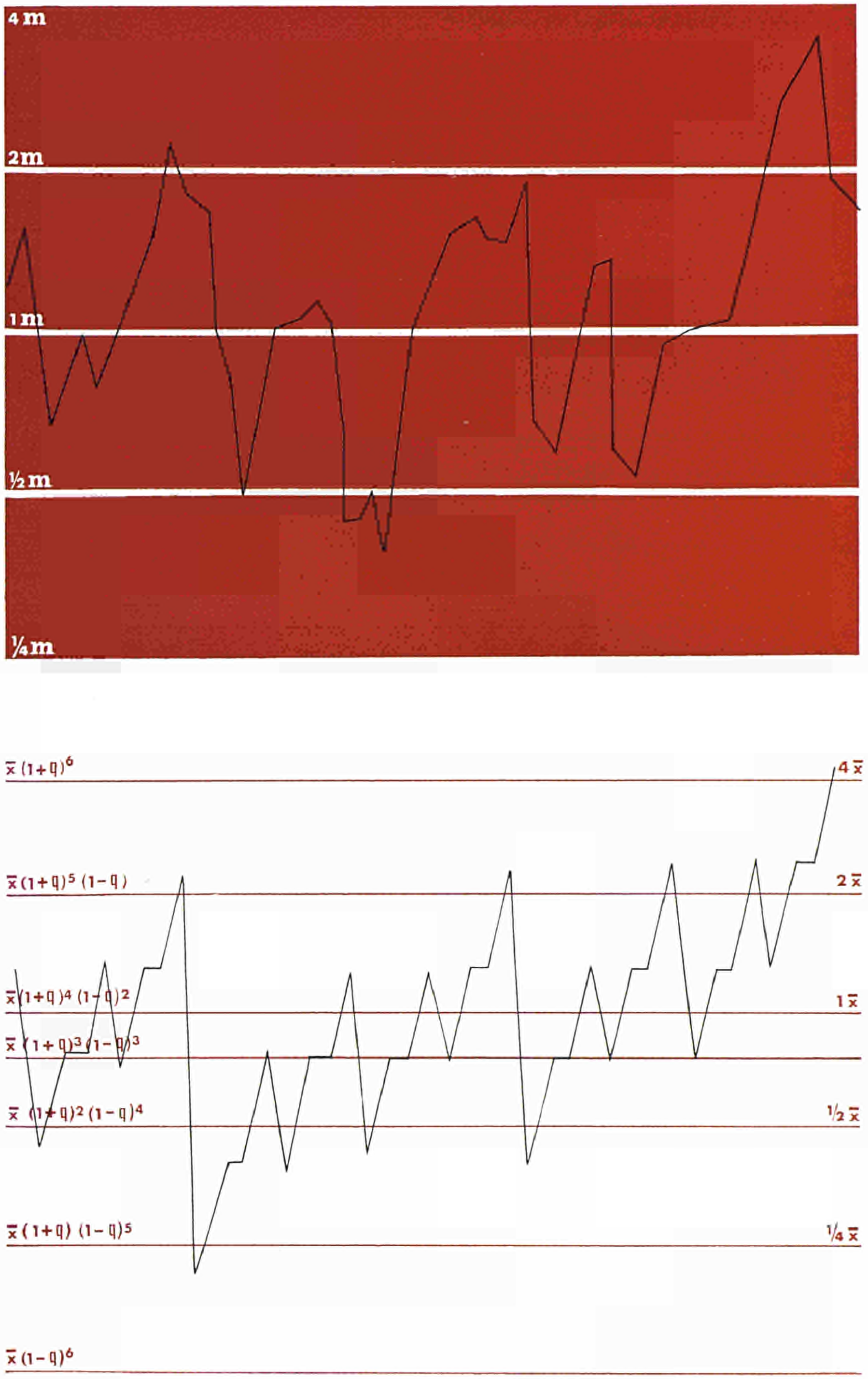


Fig. 4: The top curve shows the actual results of channel-sampling in a lead-bearing vein in Algeria (after Traité de géostatistique appliquée, *tome I, by Georges Matheron, p. 42). The values for lead-content are logarithmically plotted.*

The bottom curve is purely theoretical and shows the terms 12-64 of a 6th order log-binomial distribution for $q = \pm 0.30$.

It can be seen that there is a marked resemblance between the two curves.

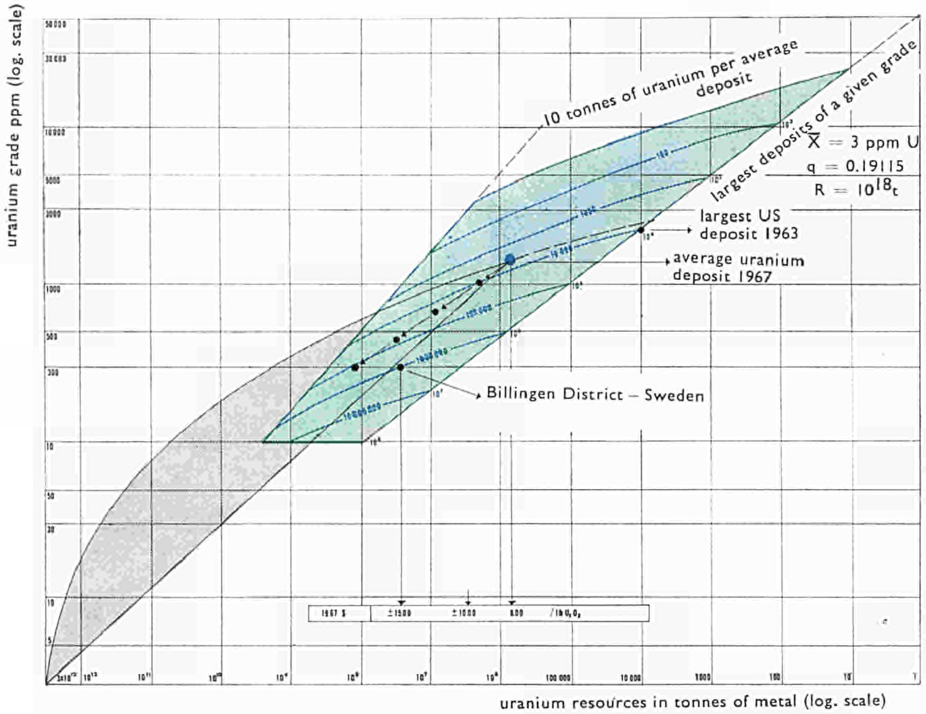


Fig. 5: Inferred minimum resources of uranium in the upper 2.5 km of the earth's crust.

Uranium resources of possible economic interest.

Reference distribution. The whole figure is based on the 500,000 tonnes of uranium metal reserves, known in 1967, occurring in ore deposits with an average metal content of 4,000 tonnes and an average concentration of 1,500 ppm.

Progression of resources by increasing average metal content 2.5 times and reducing average grade by 2/3.
Largest deposits of a given grade

From the order of magnitude of the thus inferred ore-grade and potential ore-grade uranium resources, as compared to all foreseeable uranium requirements, no shortage of uranium, nor excessively high prices may be expected from any natural insufficiency of uranium resources.

Therefore, the long-term supply question appears to be essentially a politico-economic problem, the solution of which could be directed entirely to the optimum conditions for the development of the nuclear industry. However, to have any uranium at all after the years 1975-1980 at a price of $ 8-10/lb U_3O_8, a vigorous programme aimed at locating and developing new ore reserves among these probable resources should start right from this moment. (EUBU 6-18)

Acknowledgement:

For the compilation of this short paper on the distribution and predictability of mineral resources the author has freely adapted the published geostatistical theories on mineral exploration and the evaluation of mineral deposits as developed by Prof. H. J. De Wijs (*Technische Hogeschool*, Delft) and Prof. G. Matheron (*Bureau de recherches géologiques et minières*, Paris). Most of the present calculations were made by Mr. H. I. De Wolde of the Joint Research Centre at Ispra (*CETIS*). The calculation method was published in Euratom Report N° EUR 3461e, 1967.

Using nuclear energy for making iron

PROF. WERNER WENZEL, *Iron and Steel Institute, Aachen Technical University*[1]

THANKS TO technological progress, nuclear energy is today becoming cheaper and cheaper. It is therefore essential to examine right now the repercussions of the availability of this cheap form of energy on the iron industry. It will thus be possible to draw in good time the necessary conclusions concerning the development of the processes which the adoption of nuclear energy involves.

Since, even if the cost of nuclear energy is low, the price to the consumer is made up of various energy sources, the iron industry can only derive real advantages from the production of cheap nuclear energy if the works is itself the owner or the direct beneficiary of a nuclear power plant. However, if a nuclear power plant is to form part of the basic equipment of such an integrated production unit, it is essential that the unit should be able to use all the energy thus produced, or at least the vast majority of it, for its own purposes, it being borne in mind that nuclear energy only offers economic advantages when very large units are employed.

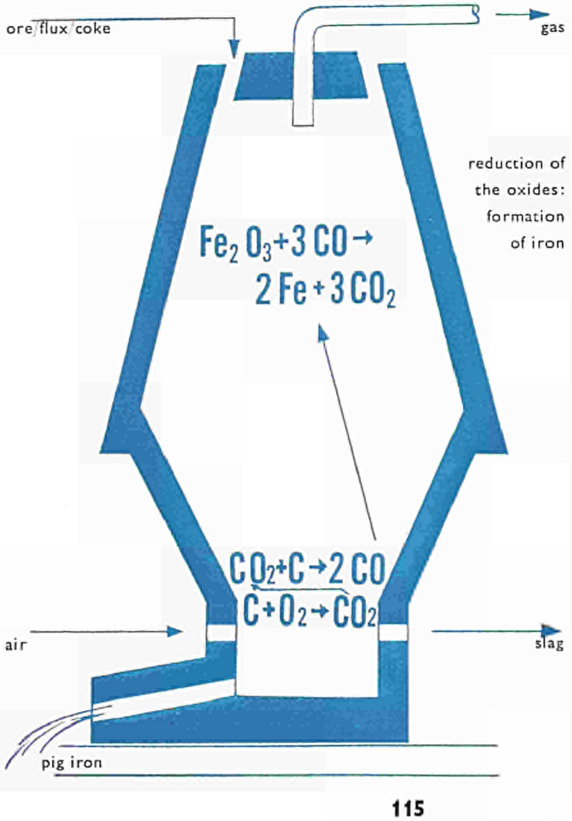
A closer examination of the situation reveals that, more than almost any other branch of industry consuming energy, the iron industry is one in which the energy requirements of one single plant are roughly of the same order of magnitude as the output of a nuclear power plant. It may be assumed that ironworks of the future will have to produce at least three million tonnes a year in order to remain competitive. The total energy requirements of such units can be estimated at 2,000 MWth and above if account is taken of all the fuel consumed, i.e. including the coal used for reduction purposes. If the requirements with regard to reduction coal are deducted, the amount of actual heat and mechanical energy needed—roughly 1,300 MWth and above—is still within the range of a viable nuclear power plant.

What methods are therefore available to enable the works of the future to adapt iron-production techniques to this energy situation?

Iron production using electrical energy

Let us first assume that the heat produced in a nuclear reactor will continue to be converted into electrical energy via mechanical energy, i.e. via turbo-alternators. Assess-

Simplified operating scheme of a blast-furnace.



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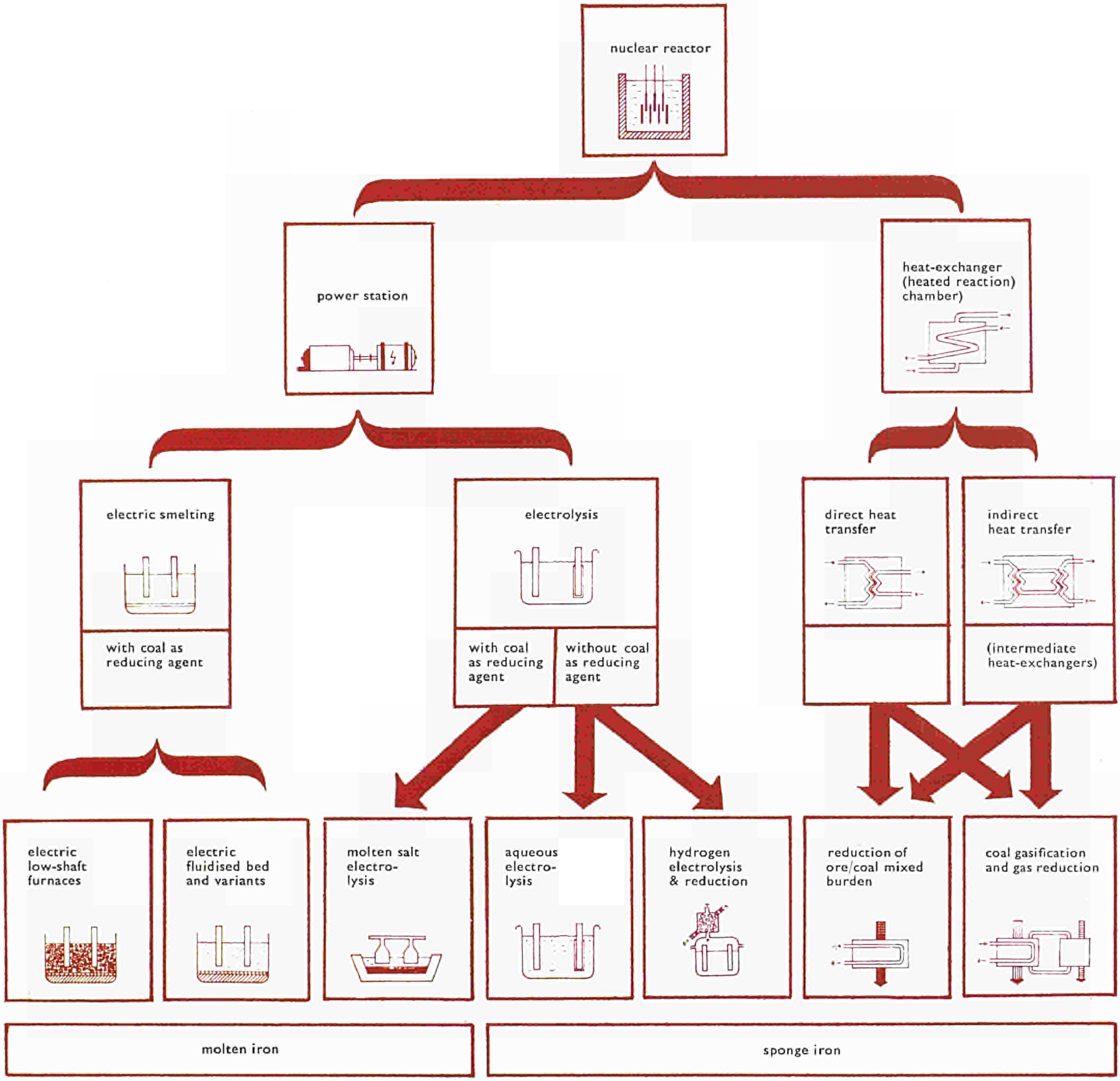
[1] After a lecture delivered in the "Haus der Technik", Essen on 10 April 1967.

Fig. 1: General scheme of iron production by means of nuclear energy.



ments of the prospects of nuclear electricity for the production of iron can take as their basis a kWh cost of 3.75 mills over the long term, meaning the next 25 years or so, while for the more immediate future a figure of 6.25 mills per kWh and above should be more realistic.

The significance of this 3.75 mills cost per kWh for the iron industry becomes clear when it is compared with the prices of the other forms of energy available. This cost of electrical energy is equivalent to a thermal cost of $ 4.35/million kcal. On the other hand the calorie prices for conventional primary energy sources, such as coal, oil and natural gas, are at present less than $ 2.50/million kcal and in some cases—as with natural gas—well below that figure.

Electrical energy as a source of power for use in the iron industry is, therefore, even in the nuclear age, only competitive with regard to fossil fuels if the latter are not available owing to reasons of a local nature or if special advantages are to be gained by using electrical energy rather than other sources of heat.

There is a wide selection of possible methods for producing iron with cheap power, in particular electrolysis and carboelectric techniques.

As things stand at present, neither aqueous electrolysis nor fusion electrolysis of iron compounds can be contemplated for reduction purposes in large-scale iron-smelting, the only conceivable method being via the electrolytic production of hydrogen.

Hydrogen electrolysis

From the cost angle, the basic feature of this technique is that, while an installation is required for obtaining hydrogen from water, all the units necessitated by the fuel in conventional iron-smelting are not needed.

The technical aspects of iron ore reduction using hydrogen can be regarded as thoroughly explored. The capital-dependent costs and the other operating costs do not differ much from those of other reduction methods, especially if it is borne in mind that there is a possibility of using attractive special designs. Here, too, the energy costs therefore constitute a decisive factor in assessing the potential usefulness of the technique. A figure of 3.75 mills/kWh gives about $ 15/tFe, while at 6.25 mills/kWh this rises to $ 25/tFe. Assuming the most favourable conditions possible, the energy costs involved in this method of iron production are thus at the upper limit of the figures obtainable by conventional techniques (blast furnace and LD process). This method can therefore only be contemplated if fossil reducing agents or fuels are either not available at all or only at abnormally high prices. It is obvious that as the population of this planet continues to grow, this situation may well occur sooner or later; in certain development areas such iron smelting methods will even probably have to be introduced very soon.

Carboelectric reduction

Assuming that fossil fuels can be used at least as reducing agents, the electrical reducing techniques which have the greatest edge are those by which liquid iron is produced directly. The electrical reduction —or, to be more precise, carboelectric reduction—techniques offer the advantage that they use coal, which is much cheaper than hydrogen, as a reducing agent and the more expensive electrical energy solely for heating purposes. In this way about 2.5 million kcal/tonne of iron produced are needed in the form of coal and—with the best methods at present available, using the waste gas for preheating and prereduction—about 1.2 million kcal in the form of electric current. If we take as a basis the above-mentioned optimum cost of electric heat ($ 4.35/million kcal) and reducing carbon ($ 2.5/million kcal), the total optimum cost, namely $ 11.5/t Fe, for producing iron by nuclear energy, is roughly the same as that of the net quantity of energy required in the operation of a modern blast furnace on a comparable site.

Even so, electroreduction techniques only stand a chance if they can compete with blast furnace plants with regard to total investments and from the operating angle.

To sum up, therefore, the use of electrical energy in the production of iron is only feasible in the nuclear age in special cases, when the quantity of fossil fuels available is limited. When cheap nuclear energy is available, however, such cases will certainly be more frequent than today.

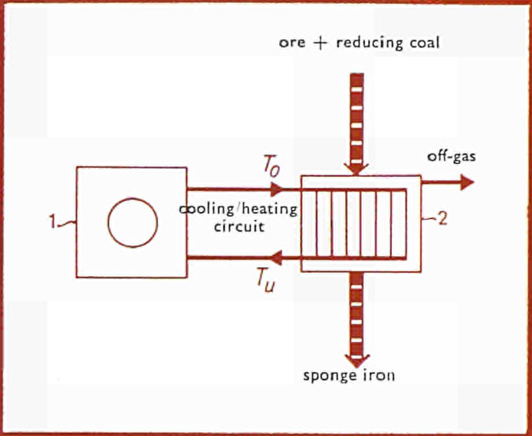


Fig. 2: Basic scheme of iron production by means of nuclear reactor heat.

1 Nuclear reactor
2 Reduction vessel

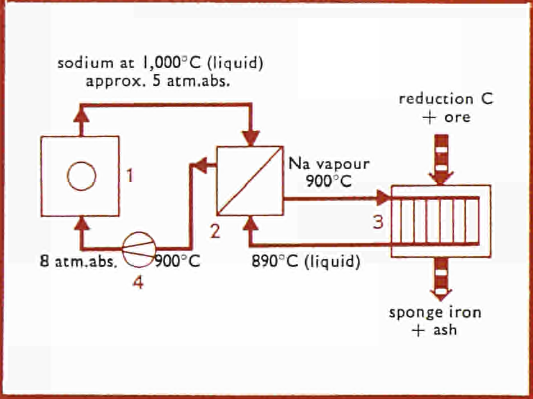


Fig. 3: Iron reduction by means of heat from a liquid-sodium-cooled fast breeder reactor (use of coal/ore mixed burden).

1 Nuclear reactor
2 Intermediate heat-exchanger
3 Indirectly heated reaction vessel
4 Pump (electromagnetic)

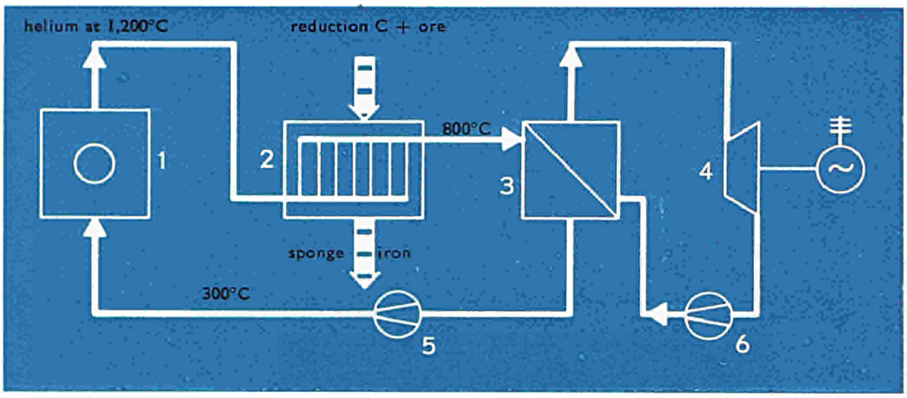


Fig. 4: Single chamber iron reduction by means of heat from a helium-cooled high-temperature reactor.

1 Nuclear reactor
2 Indirectly heated reaction vessel
3 Heat-exchanger
4 Power station
5 and 6 Blowers

Direct use of reactor heat for iron-smelting

From the above remarks it is clear that in spite of the low electricity costs obtainable with nuclear energy, iron-smelting technology would not be faced with a fundamentally new situation.

At this juncture mention should be made of some ideas which have been developing for several years at the Iron and Steel Institute of the Aachen Technical University and have prompted a good deal of research work at the Institute. Fairly substantial funds are made available for this work by the West German Ministry for Scientific Research.

The basic idea is that if fossil reducing agents such as coal are used, what is primarily required for iron production processes is not electrical energy but heat. At the same time it is basically heat that nuclear reactors produce rather than electrical energy. It is therefore worth while to consider applying the heat produced in a nuclear reactor directly to the iron production process, thus cutting out the use of electrical energy. If the attempt to transfer the heat from the nuclear reactor directly to the iron production process is successful this would mean that a reactor of given size could supply about three times as much heat to a smelting plant as if the process were effected via the use of electric current. This opens up the possibility of reducing the cost of iron production considerably; furthermore the expensive plant required for the generation of electrical energy can be dispensed with.

According to rough calculations the net cost of a million kcal of nuclear heat, supplied to the reduction installation, is between $ 0.50 and $ 1.00, depending on the size of the nuclear power unit feeding the works and on the state of nuclear development taken as the basis for calculations. In relation to the fossil fuels used at present, there is therefore a possibility of lowering heat costs by two-thirds or even three-quarters, depending on the site conditions of the works concerned.

The basic scheme of a nuclear-powered ironworks is shown in Fig. 2. The coupled system consists of a nuclear reactor (1) and a reduction installation (2). In the simplest case, the reactor coolant is also used to heat the reduction installation, but for reasons of plant safety and to ensure access to the reduction installation, it is assumed for the time being that an intermediate circuit would be needed.

Helium or sodium?

The main factor governing the type of fluid used for heat transfer is the temperature required in the reduction installation. Up to a maximum of about 800°C carbon dioxide/carbon monoxide mixtures could be used—theoretically at least. At 900-1,000°C only sodium and helium can be contemplated and over 1,000°C only helium to all intents and purposes. The nuclear reactors will of course be of different types depending on the coolant used, so it is particularly important, in assessing the reactor side of the operation, to know the minimum temperature which has to be maintained in the reduction installation.

Our theoretical and practical work in this field was at first concentrated on an attempt to solve this problem. It was first assumed that coal would be used as the reducing agent, since particular attention must be given to this commodity in Germany, as compared with oil and natural gas, for obvious reasons. However, similar technical solutions and appropriate results can also be obtained for other reducing agents.

In the interest of reactor technology the aim should be to operate economically at the lowest reducing temperatures possible. We therefore examined a large number of factors influencing the reduction of iron ores using coal, not only the obvious parameters such as temperature, cost, pressure, grain size, etc., but also less common aspects such as the generation of high pressures and the use of catalysts. It was found that economic use of the reaction space, which should not be below one tonne of iron produced per cubic metre per day, is only achieved with the types of coal considered if the reduction is carried out at temperatures of at least 850°C.

In the case of coal, incidentally, this lower temperature limit can only be observed by the trick of using a catalyst for regenerating the carbon dioxide into carbon monoxide from the carbon. It was generally found that temperatures between 900 and 1,000°C should be available in the installation in order to ensure an economic reduction process.

As reactor technology stands today and even taking foreseeable developments into account, heat at this temperature can only be removed from nuclear reactors by sodium or helium cooling. The existing helium-cooled high-temperature reactors have an outlet temperature of 750°C, an increase to 850°C being planned for the immediate future. There are no major technical obstacles to raising the temperature of the gas to over 1,000°C.

Sodium-cooled reactors are at the moment limited to a maximum coolant temperature of 600°C. There are admittedly no fundamental problems preventing the temperature being raised substantially, but new materials would have to be developed. Fig. 3 shows a flow scheme such as could be used for a nuclear reactor cooled with liquid sodium. Since, at a coolant outlet temperature of 1,000°C, the sodium is already in vapour form at normal pressure, the coolant circuit would have to be pressurised (about 5 atm.abs.) in order to prevent tricky vaporisation conditions in the reactor core.

The reactor coolant passes through a heat exchanger in which the heat is transferred to a secondary sodium circuit. It is best to keep this secondary circuit at normal pressure so that the heat exchanger acts as a sodium vaporiser for the secondary circuit. The sodium vapour is passed at a temperature of about 900-950°C to the reduction vessel, which is likewise designed as a heat exchanger. Here the sodium vapour condenses, giving off its condensation heat via dividing walls to the reduction mixture, which consists of ore and carbon. The condensed sodium is then recycled to the sodium vaporiser.

The advantage of such a vaporisation system is that, owing to the appreciable latent vaporisation heat of sodium, which is about 1,000 kcal/kg, the quantity of sodium in the intermediate circuit can be kept fairly small and in some cases a circulating pump dispensed with.

The use of helium as a reactor coolant and medium for transferring heat to the reduction vessel is shown in diagram form in Fig. 4. Owing to the high heat transfer capacity required, the helium is at a pressure of about 40 atm.abs. During normal operation the radioactivity of the coolant is low, so that it may even be possible to pass the helium from the reactor straight through the reduction vessel's heat exchange unit. The maximum temperatures obtainable using a high-temperature reactor coupled to the reduction vessel are, it is thought, higher than with sodium-cooled reactors, the reaction rates and specific throughputs being correspondingly higher.

There are two basic systems for smelting iron ores with solid reducing agents—single chamber and double chamber. The term "single chamber" is taken here to mean that a reduction mixture consisting of ore and reducing coal is heated directly. With the double chamber method, on the other hand, only the reduction of the carbon dioxide produced from carbon monoxide during the iron ore reduction process is carried out with the aid of the reduction carbon in the heated reaction chamber, the carbon monoxide thus obtained being passed to a separate chamber in which it is then reacted with the iron ore. The reduction gas therefore passes through a circuit between the heated coal gasifier and the unheated ore reduction vessel, in which an exothermic process takes place. Fig. 5 shows a diagram of such a double-chamber reduction installation coupled to a nuclear reactor.

In all cases it is advisable to fit a power source in the form of a boiler (or gas turbine) downstream of the reduction installation; this produces a particularly economical link-up between the reduction of iron ore at high temperatures and the recovery of energy in the medium temperature range of the heat-transfer fluid.

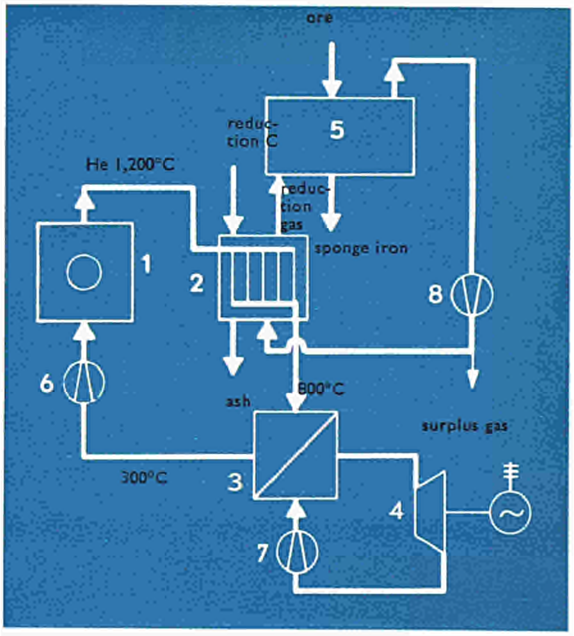


Fig. 5: Double chamber iron reduction by means of heat from a helium-cooled high-temperature reactor.

1 Nuclear reactor
2 Indirectly heated reaction vessel
3 Heat-exchanger
4 Power station
5 Ore reduction vessel
6, 7 and 8 Blowers

A blast-furnace at the moment of tapping - For how many years will this remain a common sight? (photo: Mannesmann/Stachelscheid)

While the future of sodium-cooled reactors is still fraught with numerous uncertainties as regards the temperature range of use to the iron and steel industry, the development of helium-cooled high-temperature reactors is largely complete (pebble-bed reactor in Germany, *Dragon* in Britain and *Peach Bottom* in the United States).

On the other hand—and quite apart from the undeniably appreciable difficulties still to be solved in the nuclear reactor field—it should be clearly realised that in the field of low-temperature iron ore reduction considerable work still has to be done before the large-scale use of nuclear energy for iron-making will be a feasible proposition. The greatest technical problem relates to the development of the heat exchangers or heat-exchange surfaces required in the reduction vessel in order to transfer the heat from the coolant (helium or sodium) to the reaction mixture.

The foremost problem here is that of the materials, since account has to be taken of the effects of temperature, pressure, wear and corrosive gases. Preliminary studies would seem to indicate that heat exchangers can be built to operate under such conditions, albeit at the cost of a considerable amount of development work and, possibly, the adoption of unusual designs. (EUBU 6-19)

An original sputtering process for surface-coating

MICHEL GIBB, *Editor*

AN ORIGINAL surface-coating process has recently been developed at the Ispra Establishment of Euratom's Joint Research Centre. This process enables a thin layer of metal, for example, or alloy to be deposited on a substrate of any kind. The process is particularly suitable for forming anti-corrosion coatings or diffusion barriers, but can also be used to apply electrically or thermally conductive coatings or even ornamental or optical coatings. It was developed at Ispra's High-Temperature Chemistry Department by Joseph Wurm, with the assistance of Pierre Beucherie and Michel Block. The patent on the process has been filed in 18 countries, including the USSR, and a licence on this patent was recently taken out by a French company.

The coatings intended for the above-mentioned purposes must possess a number of physical-chemical properties. It is required, for example, that the layers should adhere well to the surfaces they cover, in some cases even up to temperatures in the region of 1,000°C; that deposition should be uniform and homogeneous; that the layers should be impermeable to gases or liquids, in other words that they should have very low porosity; that they should have a very fine granular structure; that they should stand up well to corrosion, etc.

Several processes aiming at coatings of this quality are already in use, such as electrolytic deposition and evaporative deposition under vacuum. The present invention can be placed in the category of electrical discharge deposition techniques, which, in the present state of the art, embraces mainly cathodic sputtering: in a vacuum chamber, in most cases a glass dome, a low-pressure gas atmosphere of about one-tenth of a millimetre Hg is generated; two electrodes are arranged in this chamber, the cathode itself consisting of the material to be deposited, and the anode being formed in most cases by the casing of the apparatus; between these two electrodes a high voltage in the region of 2,000-4,000 volts is applied and the sample to be coated is placed opposite the cathode.

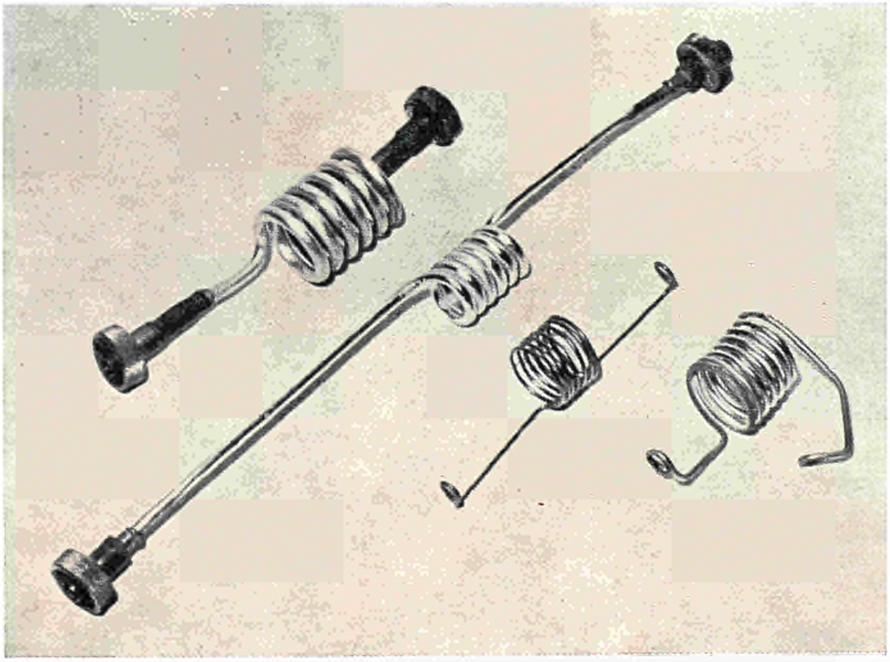
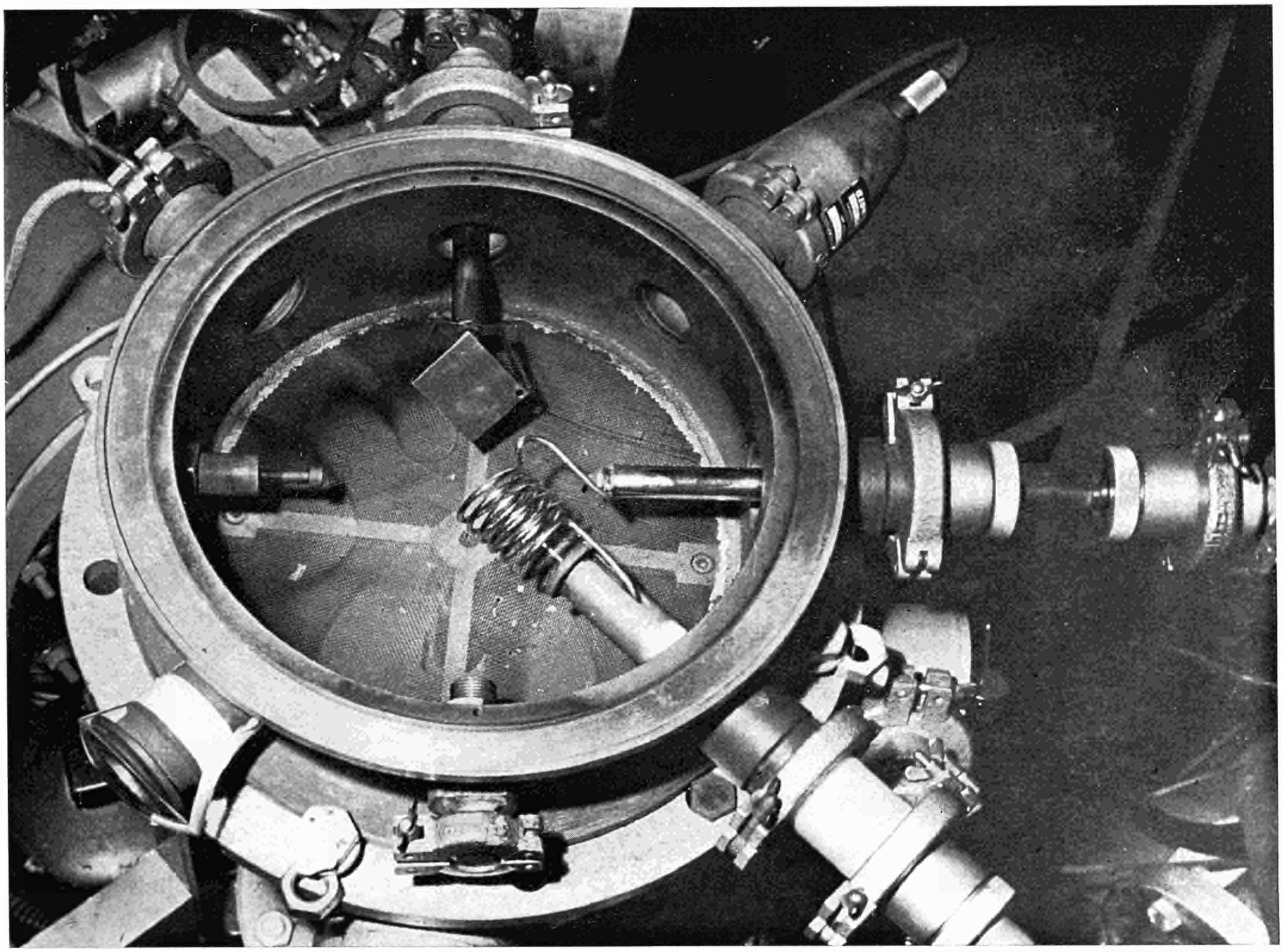


Fig. 1: Coils employed to generate the electromagnetic field of very high voltage and very high frequency which is essential to the process. These coils are made of the material to be deposited.

Fig. 2: The laboratory unit developed at Ispra. The photograph shows the base of the chamber in which the coating operation takes place.



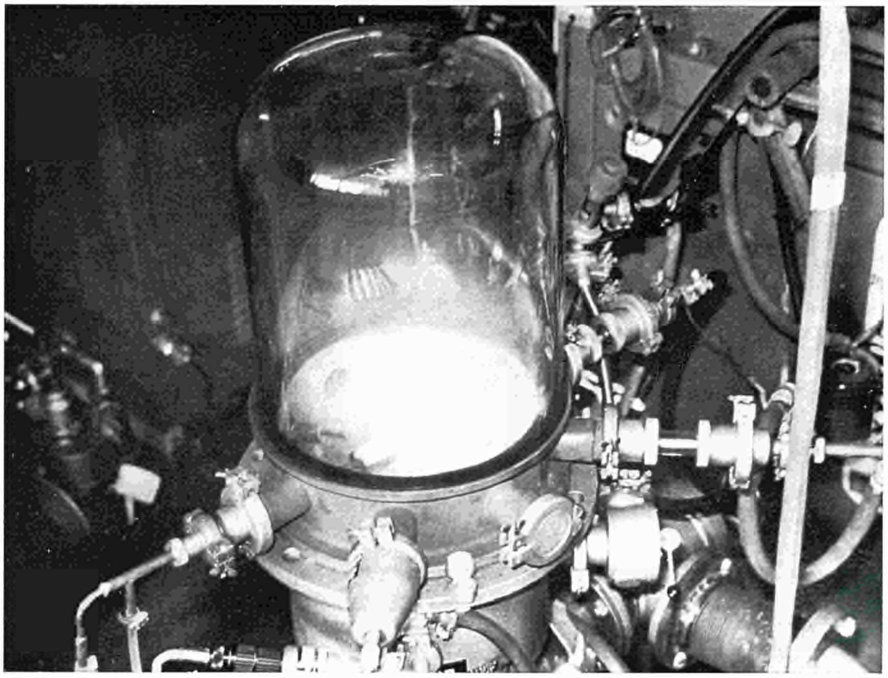


Fig. 3: The same apparatus as in Fig. 2 in operation. The flash is given off by the plasma which forms inside the coil.

Fig. 4: Appearance of a coil that has been in use for some time.

The exact mechanism of conventional cathodic sputtering is still not fully known, but the following description of it can be given.

The cathode emits electrons which, by colliding, ionise the gas, which can for example be argon. The argon ions thus produced are accelerated by the high-voltage field and bombard the cathode, so that micro-particles of the cathode are stripped off and projected all over the chamber and, at the same time, on to the surface of the object to be coated. The conventional cathodic sputtering techniques, which are satisfactory for the deposition of extremely thin layers, nevertheless present several disadvantages. Their yield is too low for deposits of any great thickness and the rate of deposition is too slow. Furthermore, the electrode system cannot be readily adapted for effective operation inside a tube, for example, in order to apply an internal coating. Altogether, the apparatus is not sufficiently mobile and is very unsuitable for effecting deposition in industrial equipment of any great size.

The process developed at Ispra enables these drawbacks to be overcome. It differs from the conventional sputtering methods essentially in that, instead of an electrostatic field, it uses an electromagnetic field of high voltage and very high frequency.

The coil (Fig. 1) by means of which this electromagnetic field is generated is made of the material to be deposited or of one of the constituents thereof; it is immersed in a gas atmosphere which is constantly renewed and maintained at a substantially constant pressure (of the order of one-thousandth of a millimetre Hg; see Figs. 2 and 3). Since it is exposed to an electromagnetic field of high voltage and high frequency, this gas is ionised and converted, at least partly, into a plasma. By the action of electromagnetic agitation the particles of this plasma are projected against the coil, from which they strip corpuscles of matter which spread throughout the chamber and are thus deposited on the object to be coated (Fig. 4). According to whether an inert or a reactive gas is employed, they are deposited either as such or after entering into chemical combination with the gas. For example, by using oxygen and an appropriate metal it is possible to obtain a coating of the oxide of that metal.

The new process has another distinctive feature: a small jet of gas at low pressure is introduced directly into the space inside the turns of the coil. It is due to this feature that optimum sputtering conditions are achieved.

The principal advantage of the process is that the rate of deposition is much faster than with conventional forms of sputtering. In addition, it is possible to connect several coils in parallel or in series, according to requirements, in order to secure higher

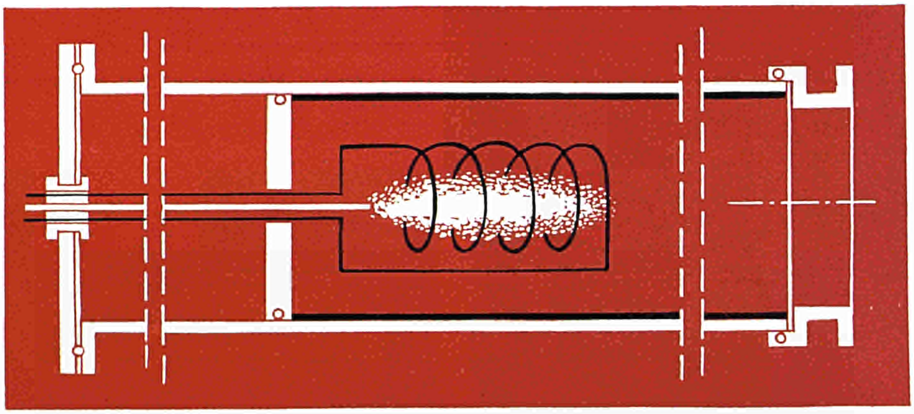


Fig. 5: A possible rig for the internal coating of a pipe.

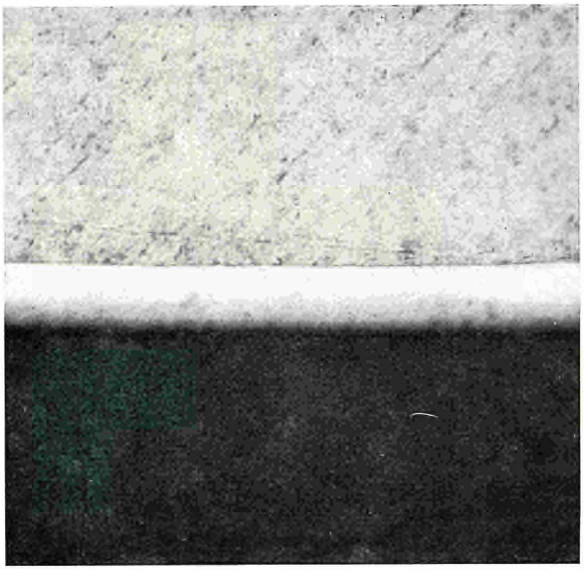


Fig. 6: Metallograph of a sample of zirconium coated with a layer of molybdenum about 16 microns thick (enlarged × 600). Note the remarkable evenness and firm adhesion of the deposit.

outputs. Furthermore, it was found experimentally that the plasma formed from the injected gas is largely confined inside the coil. It is thus possible, without any great difficulty, to maintain the low gas pressure which prevails outside the coil. This rarefaction of the gas has a beneficial effect on the structure of the deposit, since it reduces gas occlusions to a minimum.

The process has already been successfully used for applying 5 to 20 micron coatings of niobium, plutonium, tungsten, tantalum, zirconium, titanium, iron, cobalt, nickel, copper, aluminium, platinum, iridium, as well as nickel-chrome alloys and stainless steels, on substrates as diverse as metals, alloys, ceramics, glass, graphite, plastic and even cardboard.

Fig. 5 shows schematically how the process can be used for the internal coating of a pipe. In this case the pipe itself could constitute the vacuum chamber. All that has to be done is to seal the pipe, connect it to a battery of vacuum pumps and make the coil travel through it at a given speed. Anti-corrosive coatings have already been applied in this way to the interior of pressure tubes for nuclear reactors, and this has been done in the minimum of time.

In order to simplify the feed of the gas to be ionised, it is possible to construct a hollow coil, i.e., a coil consisting of a spirally wound tube provided with holes which point towards the inside of the space enclosed by the coil and through which the gas will flow (Fig. 7).

The examples of possible uses for the process could be greatly multiplied. At all events it has an extremely wide field of application: though invented in a nuclear research laboratory, it will undoubtedly find its way into numerous sectors of industry, even those which are far removed from the applications of nuclear energy. (EUBU 6-20)

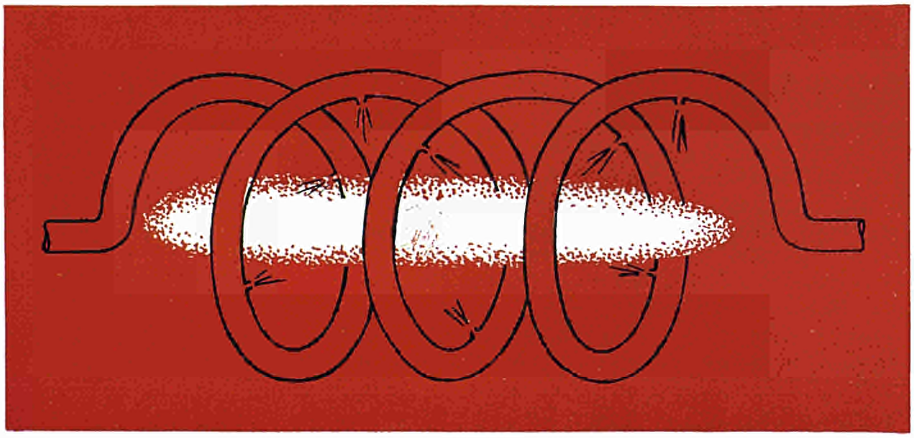
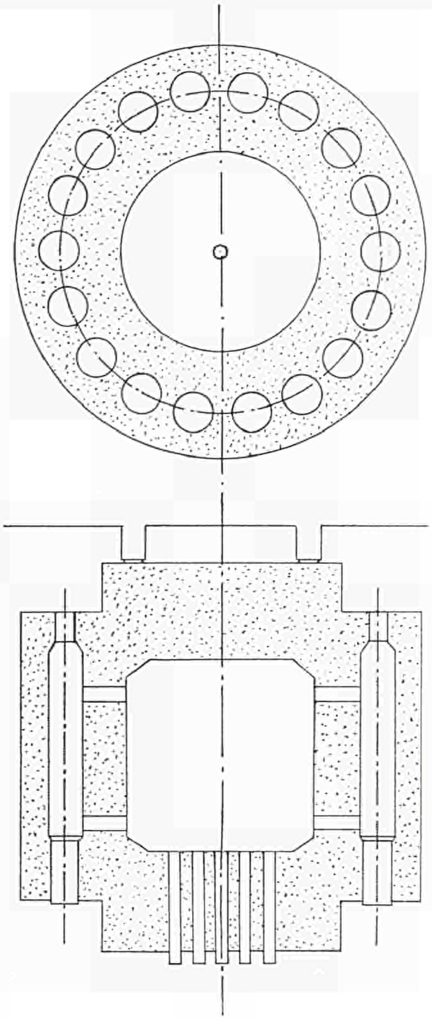


Fig. 7: A special hollow coil in which holes have been drilled for the supply of the gas to be ionised.

New design of prestressed concrete pressure vessel



The Italian institute ISMES (*Institute for experimental models and structures*) has been asked by Euratom to conduct research on a new reactor vessel concept in prestressed concrete, initially developed under the *Dragon* Project. In this system, suitable for gas-cooled reactors, the heat exchangers are housed in the actual walls of the concrete pressure vessel.

In traditional designs the exchangers are either installed in a special enclosure outside the vessel or else "integrated" inside the vessel.

The advantage of the first of these traditional solutions is that the exchangers are easily accessible for servicing, replacement, etc. A drawback is that a special leaktight enclosure has to be built for them.

In the second solution, the advantage of easy access is foregone and a concrete shield has to be erected to shield the exchangers against radiations from the reactor core. On the other hand, the complete unit is more compact.

The new design to be investigated by the *ISMES* to some extent combines both the foregoing advantages without the drawbacks:

- the concrete vessel wall serves at once as a structural material and as a shield for the exchangers;
- it should be possible to devise an arrangement that requires no more concrete than with the consolidated method, and certainly less concrete than in any design where the exchangers are housed separately;
- easy access to the exchangers can be provided.

A fourth advantage is that, since the layer formed by the ring of exchangers is less rigid than the concrete, it should be possible to distribute the circumferential stresses better than in the standard solutions.

The aim of the task entrusted to *ISMES* is to verify these theoretical advantages, particularly the last-named, by means of experiments on small-scale models.

Configuration of prestressed concrete pressure vessel for 1,250 MWth helium high temperature reactor.

Publication of a new series of technical notes

A new series of Technical Notes has recently been published by Euratom. It will be recalled that the aim of these notes is to draw the attention of industries within the Community to certain original devices or processes developed under the Euratom research programme.

In this new series, information will be found on a new process for preparing uranium carbide (No. 526), a handling arm, notably for reactor charging machines (No. 74), an anti-corrosion treatment process for zirconium alloys (No. 41), etc.

Not all the processes or devices described deal exclusively with nuclear technology. Note 60, for instance, describes a thermoelectric probe for measuring the temperature of fluids, Note No. 64 a galvanic cell for detecting oxygen, Note No. 726 a computer for determining the logarithm of binary numbers, etc.

For all information concerning these Technical Notes, write to Euratom, Industrial Property Directorate, Directorate-General for the Dissemination of Information, 51 rue Belliard, Brussels 4, Belgium.

EURATOM NEWS

Power reactors in operation, under construction and planned in the Community

(Status as at 31 October 1967)

1. The total net electric capacity of the nuclear power plants in operation, under construction or planned is 8401 MWe, broken down as follows:

	country	operational	critical	under construction	planned	total MWe
a) *Proven-type reactors*						
Gas/graphite						
Chinon 1 (EDF1)	F	70	—	—	—	70
Chinon 2 (EDF2)	F	200	—	—	—	200
Chinon 3 (EDF3)	F	480	—	—	—	480
St. Laurent 1 (EDF4)	F	—	—	480	—	480
St. Laurent 2	F	—	—	515	—	515
Bugey 1 (St Vulbas)	F	—	—	540	—	540
G 1 Marcoule	F	3	—	—	—	3
G 2 Marcoule	F	40	—	—	—	40
G 3 Marcoule	F	40	—	—	—	40
Fessenheim 1	F	—	—	—	650	650
ENEL (Latina)	I	200	—	—	—	200
Boiling-water						
KRB (Gundremmingen)	D	237	—	—	—	237
KWL (Lingen)[1]	D	—	—	173	—	173
VAK (Kahl)	D	15	—	—	—	15
ENEL (Garigliano)	I	150	—	—	—	150
GKN (Dodewaard)	N	—	—	52	—	52
Preussenelektra (Würgassen, Weser)	D	—	—	—	612	612
Pressurised-water						
KWO (Obrigheim)	D	—	—	300	—	300
SENA (Chooz)	F-B	266	—	—	—	266
ENEL (Trino Vercellese)	I	257	—	—	—	257
BR 3 (Mol)	B	10	—	—	—	10
NWK + HEW (Stadersand, Elbe)	D	—	—	—	660	660
b) *Advanced converters*						
Heavy-water						
MZFR (Karlsruhe)	D	50	—	—	—	50
KKN (Niederaichbach)	D	—	—	100	—	100
EL4 (Brennilis)	F	73	—	—	—	73
High temperature						
AVR (Jülich)	D	—	15	—	—	15
Sodium/zirconium hydride						
KNK (Karlsruhe)	D	—	—	20	—	20
Nuclear superheat						
HDR (Dettingen/Kahl)	D	—	—	25	—	25
c) *Fast breeders*						
Phenix (Marcoule)	F	—	—	—	250	250
d) *Type not yet decided*						
RWE Badenwerk/Elektrowatt (Hochrhein)	D/CH[2]	—	—	—	300	300
Geesthacht (Schl.-Holstein)	D	—	—	—	20	20
ENEL (Lombardia)	I	—	—	—	650	650
EBES (Doel)	B	—	—	—	730	730
INTERCOM (Tihange)	B	—	—	—	730	730
Total		2 091	15	2 205	4 602	8 913

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[1] Excluding 67 MWe conventional superheat. [2] Possibly a joint German/Swiss plant of 600 MWe.

2. Percentage breakdown of the reactors in operation and under construction, according to type

Gas/graphite	2568 MWe	i.e.	60%
Boiling water	627 MWe	i.e.	15%
Pressurised water	833 MWe	i.e.	19%
Heavy water	223 MWe	i.e.	5%
Other advanced converters .	60 MWe	i.e.	1%
	4311		100

3. Breakdown according to state of completion and by country (in MWe)

	Germany	France	Italy	Netherlands	Belgium	Community
Reactors in operation . . .	317	1 039	607	—	143	2 106
Reactors under construction	618	1 535	—	52	—	2 205
	935	2 574	607	52	143	4 311
Reactors planned	1 592	900	650	—	1 460	4 602
Total	2 527	3 474	1 257	52	1 603	8 913

4. Characteristics of reactors in operation (31 October 1967)

reactor and site	country	type[3]	critica-lity	connected to grid	output in MWe
KRB (Gundremmingen)	D	BWR	14. 8.66	12.11.66	237
AVR (Jülich)	D	HT	26. 8.66		15
VAK (Kahl)	D	BWR	13.11.60	. 6.61	15
MZFR (Karlsruhe)	D	EL	29. 9.65	. 8.66	50
Chinon 1 (EDF1)	F	GG	16. 9.62	14. 6.63	70
Chinon 2 (EDF2)	F	GG	18. 8.64	24. 2.65	200
Chinon 3 (EDF3)	F	GG	1. 3.66	4. 8.66	480
EL 4 (Brennilis)	F	EL	23.12.66	9. 7.67	73
G1 Marcoule	F	GG	7. 1.56	28. 9.56	3
G2 Marcoule	F	GG	21. 6.58	22. 4.59	40
G3 Marcoule	F	GG	11. 6.59	4. 4.60	40
SENA (Chooz)	F/B	PWR	18.10.66	3. 4.67	266
ENEL (Garigliano)	I	BWR	5. 6.63	23. 1.64	150
ENEL (Latina)	I	GG	27.12.62	12. 5.63	200
ENEL (Trino Vercellese)	I	PWR	21. 1.64	15.11.64	257
BR3 (Mol)	B	SSCR		6.12.66	10
Total					2 106

[3] GG = gas/graphite; PWR = pressurised light water; BWR = boiling light water; HT = high temperature; EL = heavy water; SSR = spectral shift reactor.

"Actinides Reviews"

The first issue of "Actinides Reviews" was published on 15th September 1967. The editors of this new Journal declare in their first editorial that the journal will be devoted to the study of the actinide elements in the widest sense. In other words it will concentrate on bringing together authoritative review articles on the state of the art in this field; moreover attention will be focussed on gaps in basic knowledge and on technological problems, with the aim of contributing to future developments. The journal is published by the Elsevier Publishing Company and its editors are A. H. W. Aten, Jr. of I.K.O., Amsterdam, and J. Kooi of Euratom.

Research on fast reactors in Luxembourg

Firms in the Grand Duchy of Luxembourg are henceforth to be associated with the work on fast reactor development which is already being carried out in other Community countries. A contract worth $ 200,000 was recently concluded between Euratom and the Luxembourg companies *ARBED*, *Gradel* and *Paul Wurth* for carrying out work under the fuel programmes run by the

Euratom-*Gesellschaft für Kernforschung* and Euratom-Belgium associations.

The three firms are to deal with problems arising from the dismantling of irradiated fuel assemblies. The principal work will be studying a complete dismantling installation under sodium adapted for a 1,000 MWe reactor. Solutions for dismantling in other media (water, gas, oils, etc.) will, however, be studied insofar as they allow a valid comparison to be made with the chosen reference solution.

Easy-to-peel tomatoes

A tomato plant mutant with easy-to-peel fruit was discovered in 1960 in the "Money Maker" variety at the Horticulture Department of the Agricultural University of Wageningen in plants grown from seeds which had been irradiated with thermal neutrons. A few years later, the Euratom/*ITAL* Association began research on possible uses for this mutant.

Genetic studies showed that easy peeling is a characteristic governed by one single recessive gene. This characteristic could be incorporated in other commercial varieties without their own essential properties being markedly altered.

An enquiry is being pursued into the commercial interest of this easy peeling characteristic from the industrial point of view (the "pelati" industry in Italy immediately springs to mind) as well as that of the private consumer.

Selected rows are at present being grown in Wageningen for purposes of observation and multiplication. Seeds from these selected rows are freely available for State institutions and universities. Requests should be addressed to the Euratom/*ITAL* Association, Post Box 48, Wageningen, Netherlands, by 1 January 1968 at the latest.

Mutation breeding produces high-yielding pea varieties

The mutation breeding work which the Institute for Genetics of the University of Bonn has been carrying out over the past few years under contract to the Euratom/*ITAL* Association has been concentrated on three pea mutants, obtained by irradiation, which are of direct interest for breeding on a commercial scale.

The first of these mutants is characterised by a forkening of the stem. The alteration

Right: Pea mutant no. 489 C, remarkable for its high yield,
Left: Control.

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leads to a higher number of pods per plant and therefore to increased yield.

A larger number of seeds per pod is the main characteristic of the second mutant. Unfortunately, there are less pods per plant, but it is hoped to improve this feature using conventional cross-breeding. The end result would then be, in this case as in the first, increased yield.

A similar claim is made for a third mutant, which is essentially of a "fasciated" type, in which the stem is broadened and the flowers are clustered in the top of the plant. In contrast to most other fasciated mutants it has an excellent seed production. Results of the 1966 breeding programme place the improvement in total yield of this strain at almost double that of the control line. This high yield is due to a large increase in the number of pods per plant as a consequence of the stem fasciation.

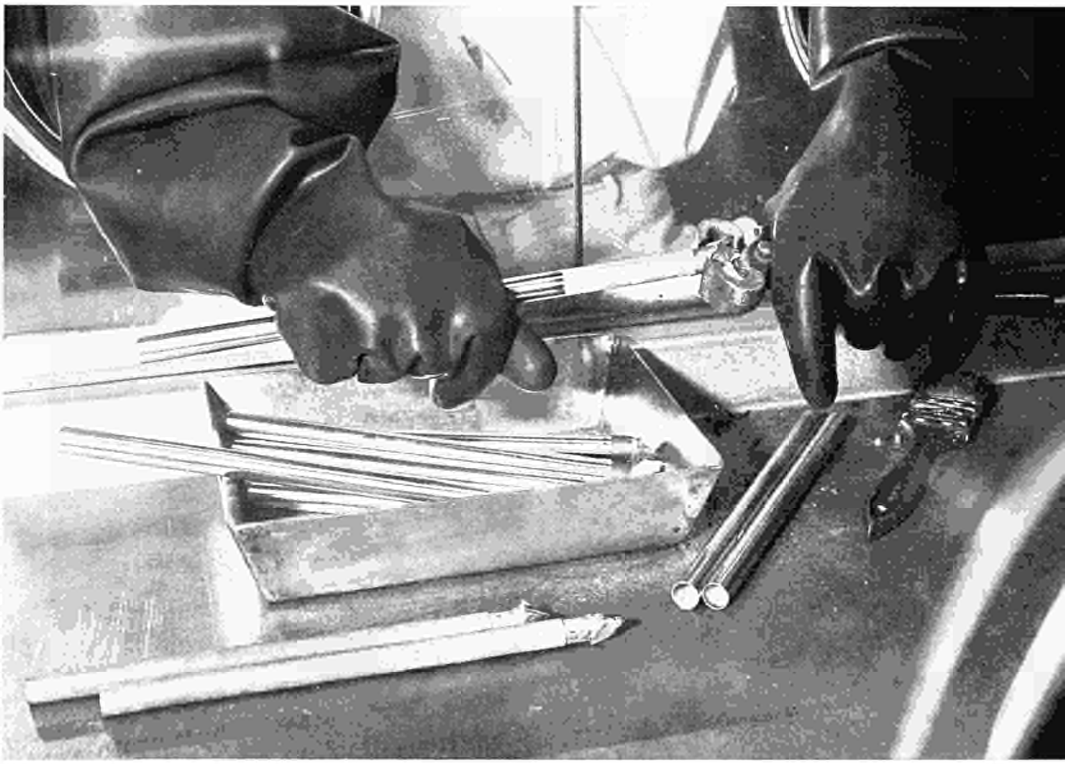
Study of the optimum use of plutonium

Interatom GmbH of Bensberg, near Cologne, is under contract to the Commission of the European Communities to carry out a study on the optimum use of plutonium in the power industry. In a series of as realistic models as possible the contractor is examining: (a) whether plutonium should be reserved for future use (in fast breeders, for instance); (b) whether plutonium should be first of all used in thermal reactors and which reactors are especially favourable for this purpose; (c) whether, and in which fast breeders, plutonium should be used.

The study involves thorough bibliographical research, such as that relating to the use of plutonium in thermal reactors. In connection with this, the reports of the Brussels Symposium of the International Atomic Energy Agency of March 1967 on the use of plutonium as a reactor fuel are to be evaluated in particular.

If necessary, nuclear data of several plutonium-enriched converters as well as thermal and fast advanced converters are also to be examined.

Finally, the problem of the price of plutonium for closed or partly open systems is to be examined and discussed.



A technology in its infancy: fabrication of plutonium-base fuel elements (European Transuranium Institute, Karlsruhe).

topi radioisotopen ship propulsion schiffsantrieb propulsion navale propulsione navale scheepsvoortstuwing biology biologie biologie biologia biologie medicine medizin médecine medicina geneeskunde health protection gesundheitsschutz protection sanitaire protezione sanitaria bescherming van de gezondheid automatic data processing automatische information information automatique informazione automatica automatische verwerking van gegevens insurance versicherungswesen assurances assicurazione verzekeringen economics wirtschaft économie economia economie education and training ausbildung enseignement insegnamento onderwijs en opleiding power reactors leistungsreaktoren réacteurs de puissance reattori di potenza energie reactoren nuclear fusion kernverschmelzung fusion nucléaire fusione nucleare kernversmelting radioisotopes radioisotope radioisotopes radioisotopi radioisotopen ship propulsion schiffsantrieb propulsion navale propulsione navale scheepsvoortstuwing biology biologie biologie biologia biologie medicine medizin médecine medicina geneeskunde health protection gesundheitsschutz protection sanitaire protezione sanitaria bescherming van de gezondheid automatic data processing automatische information information automatique informazione automatica automatische verwerking van gegevens insurance versicherungswesen assurances assicurazioni verzekeringen economics wirtschaft économie economia economie education and training ausbildung enseignement insegnamento onderwijs en opleiding power reactors leistungsreaktoren réacteurs de puissance reattori di potenza energie reactoren nuclear fusion kernverschmelzung fusion nucléaire fusione nucleare kernversmelting radioisotopes radioisotope radioisotopes radioisotopi radioisotopen ship propulsion schiffsantrieb propulsion navale leads to a higher number of pods per plant and therefore to increased yield.

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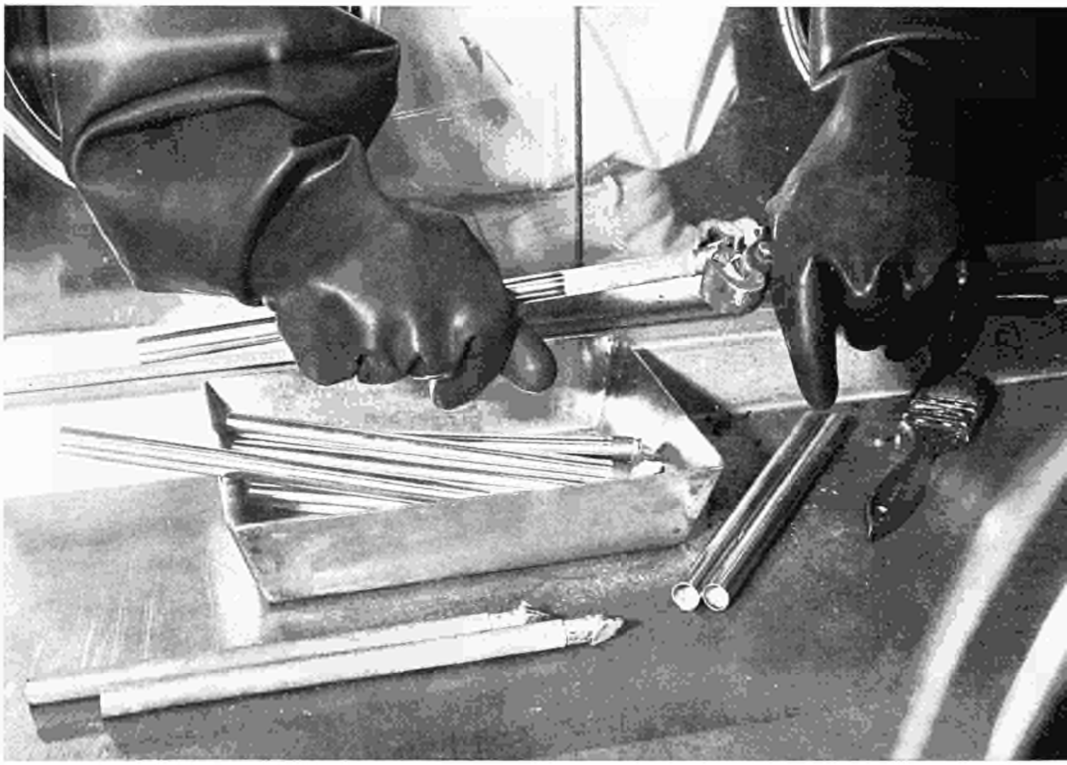

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