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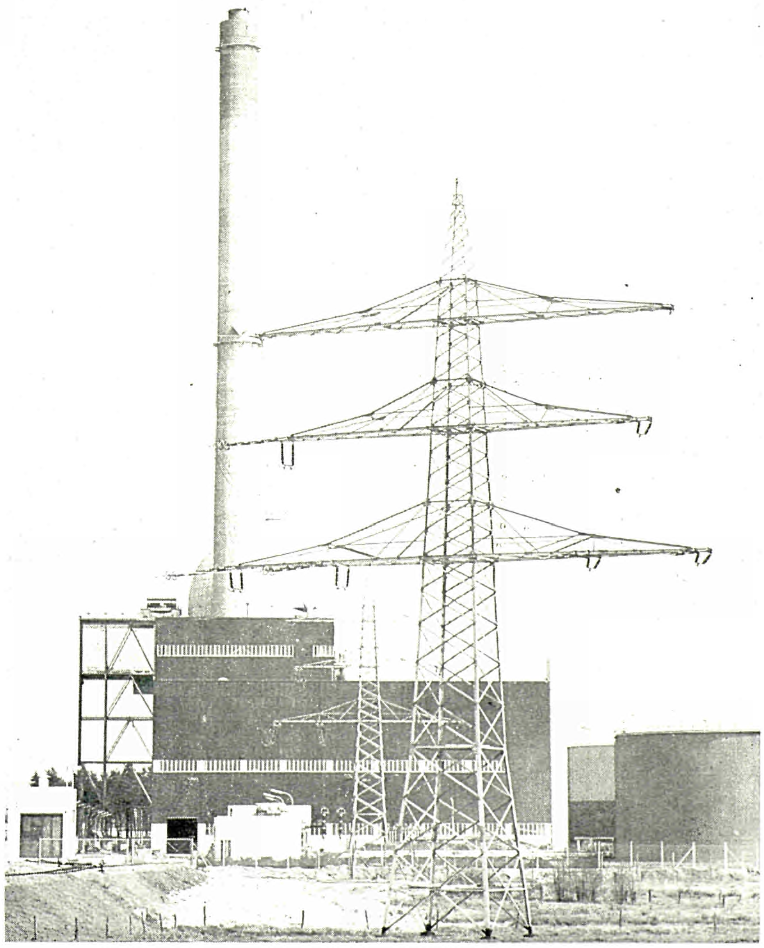

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NUCLEAR ENERGY

The present situation and future outlook

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In this issue, entirely devoted to the analysis of the various aspects of nuclear power generation, we have endeavoured to present a really comprehensive picture of the present situation and the prospects ahead, in which the pros and cons are supposed to be examined with equal objectivity.

We deliberately abstain from offering any comment. In recent times particularly, with the great energy crisis, the debate on the wisdom or otherwise of expanding the number of reactors can hardly be called dispassionate; vehement supporters and opponents of nuclear power have rushed to defend their respective opinions, some with excessive fervour and some with manifest bias. As was only to be expected, such contributions have helped rather to fuel the dispute than to dispel doubts.

Since such doubts still persist for many people, it is hoped that the study contained in this issue will clear them away, to some extent at any rate.

Nuclear energy

The present situation and future outlook *

WILLEM VINCK, JULIEN VAN CAENEGHEM

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* Abridged from *CEC* document No. 2438/III/72 „The present and future situation of nuclear energy", by W. Vinck, H. Boos, F. Luyckx, H. Maurer and J. Van Caeneghem, presented at the *International Colloquium on Nuclear Energy and the Environment*, Liège (Belgium), 22-25 January 1973.

Abbreviations

Ci	: curie (3.7 × 10^{10} disintegrations/s)
MCi	: megacurie (1 000 000 curies)
mCi	: millicurie (one thousandth of one curie)
µCi	: microcurie (one thousandth of one millicurie)
pCi	: picocurie (one millionth of one microcurie)
rad (D)	: unit of absorbed radiation dose (1 rad = 100 ergs per gram of irradiated substance)
mrad	: millirad (one thousandth of one rad)
rem (H)	: dose equivalent ("that dose of any radiation which, when delivered to man or mammal, is biologically equivalent to the dose of one rad of X- or gamma radiation")
mrem	: millirem (one thousandth of one rem)
man-rem	: the product of the dose and the number of persons affected
MPC	: Maximum Permissible Concentration
MWe	: megawatt, electrical (one million watt electrical)
MWth	: megawatt, thermal
BWR	: Boiling Water Reactor
PWR	: Pressurized Water Reactor
FBR	: Fast Breeder Reactor
LWR	: Light Water Reactor
HTGR	: High Temperature Gas-cooled Reactor
LMFBR	: Liquid Metal-cooled Fast Breeder Reactor
ICRP	: International Commission of Radiation Protection

PART I

Table of Contents

I. FORECASTS OF NUCLEAR POWER

The illustrative programme drawn up by the Commission of the European Communities *(CEC)* in 1972 set a target for nuclear power development and the related industry in the Community of six member states during the years to come.

On the basis of this programme and the published figures for the new member states, the foreseeable development of nuclear power in the European Community is summarized in Table I.

A forecast of the growth of nuclear power in the Western world was published by the United States Atomic Energy Commission *(USAEC)* in 1971 *(WASH-1139 Reactors; TID-4500)*. The data given in this forecast for the countries of the European Communities, the other European countries, the *USA*, Canada and Japan are summarized in Table II.

Both forecasts relate to the medium-term development (up to 1985) and largely arrive at the same conclusions. Discrepancies can be accounted for by differences in the scope of the forecasts and in the underlying assumption on which they were based.

As far as the long-term forecast is concerned, the figures in Table III are for the original Community of six countries and the present Community of nine.

The light water reactor *(LWR)* is the dominant reactor type in the United States accounting for almost 100% of all the nuclear energy produced.

Within the six countries of the Community, half of the power plants in existence in 1972 were equipped with light water reactors and it was expected that *LWRs* would provide 75% of total nuclear energy production by 1975. In the European Community of nine, gas-cooled reactors outnumber light water reactors. It is, however, estimated that over a number of years the light water reactor will also become the dominant type. Other reactor types, e.g., fast breeders *(FBR)* and high temperature gas cooled reactors *(HTGR)*, will start to make a significant contribution around 1985.

Attention here has been mainly devoted to light water reactors. This should give a reasonable estimate of the pollution potential of nuclear power in normal

Year	Installed nuclear capacity (thousands of MWe)	
	Community of Six	Community of Nine
1975	12	23
1980	45	55
1985	100	140

Table I: *Foreseeable development of nuclear power in the European Community as from the illustrative programme drawn up in 1972 and the published figures for the new member states.*

Year	Installed nuclear capacity (thousands of MWe)	
	Community of Six	Community of Nine
1990	210	315
1995	380	570
2000	620	930

Table III: *Long-term forecast of installed nuclear capacity (in thousands of MWe) for the original Community of six countries and the present Community of nine.*

Table II: *Estimated cumulative capacity of nuclear power plants: Western Europe*, USA, *Canada, Japan (thousands of MWe).*

	1970	1975	1980	1985
Belgium + Luxembourg	0.1	0.9	2.6	4.2
Denmark	—	—	0.8	2.6
France	1.5	3.7	8.8	16.9
Germany	0.8	5.9	15.6	31.3
Italy	0.6	0.6	4.3	9.6
Netherlands	0.1	0.5	2.3	5.2
United Kingdom + Ireland	4.7	10.6	24.8	48.3
Community of Nine Total	7.8	22.2	59.2	118.1
Austria	—	—	1.3	2.0
Finland	—	—	1.0	2.6
Norway	—	—	0.5	1.6
Portugal	—	—	0.3	0.6
Spain	0.6	1.0	6.0	13.7
Sweden	0.4	2.7	7.9	16.6
Switzerland	0.4	2.0	4.4	7.5
Other West European countries Total	1.4	5.7	21.4	44.6
USA	5.0	59.0	150.0	299.0
Canada	0.2	0.6	7.8	17.0
Japan	0.2	5.1	20.8	49.0

(routine) operating conditions and in the event of accidents, it being borne in mind that gas-cooled reactors, heavy water reactors and fast breeders may be more attractive in some aspects and less in others.

II. RADIOACTIVE EFFLUENTS FROM A GROWING NUCLEAR INDUSTRY

A. *Radioactive effluents produced during routine operation of nuclear power plants: radiation standards, current practice, medium- and long-term forecasts*

1. Radiation protection standards, e.g., Euratom Basic Standards

The purpose of the Euratom radiation standards is to protect man against the somatic and genetic effects of exposure to ionizing radiation. They cover the following three categories of persons:

a) persons professionally exposed, i.e., those who, by reason of the performance of their professional duties, are exposed to ionizing radiation;

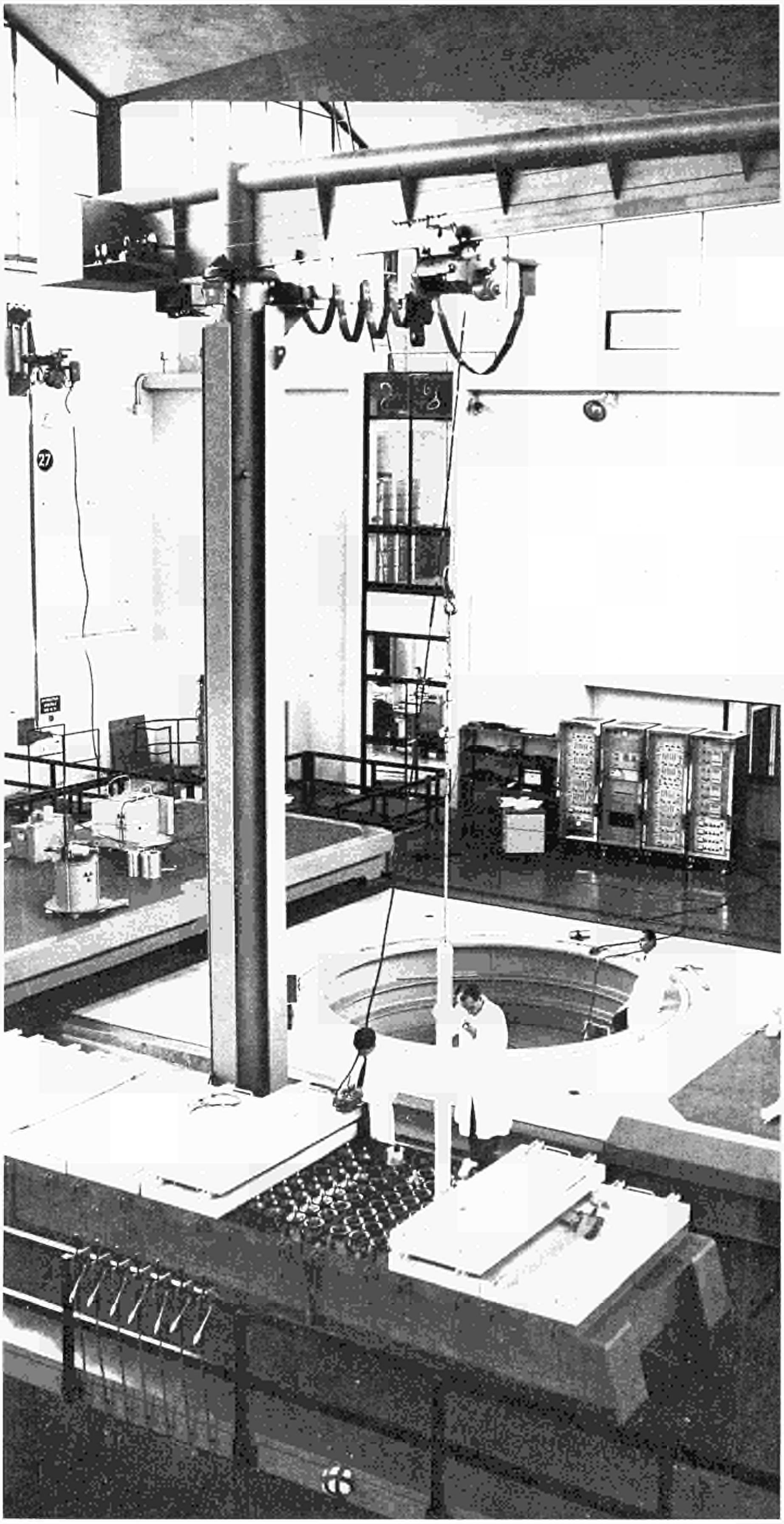
b) members of the population, who may be more sensitive to the effects of ionizing radiation, such as children or pregnant women and persons living adjacent to nuclear installations (sometimes called population "at risk"), for whom the standards will therefore have to be stricter;

c) the population at large [1], for whom the setting of standards is essentially determined by genetic considerations.

These standards, which were drawn up and are periodically revised on the basis of the recommandations of the *ICRP* (International Commission for Radiological Protection), are being used by the Member States of the Community in drawing up their own regulations.

The radiation protection standards consist of basic standards, i.e., those specifying the maximum permissible levels of exposure and the principles governing their application. They also include derived limits, such as the maximum permissible concentrations

1 The term "population at large" is taken here to mean the population as a whole i.e., the professionally exposed, the so-called population "at risk" (adjacent to nuclear installations) and the rest of the population.



(MPC) of radionuclides in the atmosphere or in drinking water.

2. Application of radiation protection standards in the generation of nuclear power—present experience of discharges from power plants

In applying these standards, a distinction must be made between those who are professionally exposed to radiation and the general public.

Persons professionally exposed to radiation: in the field of nuclear power generation, they are exposed both to radiation and contamination.

Experience has shown that the internal contamination of personnel occurs only rarely in nuclear power stations. On the other hand, external radiation often gives rise to problems with regard to the observance of the radiation protection standards, especially in the case of inspection, maintenance and repair. This is mainly because provision for access to various components and for maintenance purposes is not always made at the design stage.

It should be noted that, in order to eliminate these difficulties, the *CEC* has decided to have studies carried out of the measures which can be taken in nuclear power stations, both at the design stage and in practice, to reduce the extent to which maintenance and inspection staff are exposed to radiation.

For the sake of comparison it should be pointed out here that in fuel processing plants the occupational exposure hazard may be considered to be somewhat higher, but the number of installations is and will remain much lower.

The population living in the vicinity of nuclear power plants is subject to the hazard of exposure to the gaseous or liquid radioactive effluents discharged by them into the environment.

A distinction must be made here between the effluents from nuclear power stations and those from nuclear fuel processing plants (dealt with in section II.B below).

The gaseous effluents from nuclear power stations, particularly water-cooled reactors, contain radioactive rare gases, mainly xenon-133 (Xe^{133}) and krypton-85 (Kr^{85}), and sometimes also radioactive iodine and aerosols (particulates).

Table IV: *Comparative table of human exposure to radiation.*

1. *Mean annual dose-rates due to natural background radiation*		
in "normal" areas	gonads	93 mrad /year
	bone-lining cells	92 mrad /year
	bone-marrow	89 mrad /year
However, dose-rates vary with the geology and with altitude.		
a) *geology:* igneous rocks		up to 5 000 mrad /year
b) *altitude:*		
at 3 000 m altitude		approx. 90 mrad /year
transatlantic flight at 10 000 m		3-5 mrad /flight
2. *Diagnostic X-ray doses in medicine*		
genetically significant dose		6-60 mrem/year[1]
3. *Annual exposure due to gaseous effluent releases from nuclear power stations*		
a) individuals at site boundary:		
PWR		< 1 mrem/year
BWR[2]		< 5 mrem/year
b) average exposure of individuals within a 5 km radius:		
PWR		< 0.1 mrem/year
BWR		< 1 mrem/year
c) average population exposure due to all nuclear power applications		< 1 mrem/year
4. *Maximum permissible doses and dose limits as laid down by Euratom standards* (whole-body doses)		
persons professionally exposed		5 000 mrem/year
members of the public (critical groups)		500 mrem/year
population "at large"		5 rem in 30 years[3]
(genetically significant dose)		(170 mrem/year)

1 Values for Europe; in the United States about 100 mrem/year.

2 For *BWRs* two calculated peaks of 170 and 31 mrem respectively, have been reported at the site boundary.

3 Other, more severe values are recommended or laid down:

ICRP	2 rem in 30 years = 66 mrem/year
FRG, *USSR*	2 rem in 30 years
Sweden	1 rem in 30 years
UK	1 rad in 30 years

The liquid effluents may contain mixed fission products, corrosion products and tritium.

Examination of the figures for routine discharges indicates that they can be broken roughly as follows:

1. GASEOUS EFFLUENTS

a) *Noble and activation gases*

   typical values for a BWR: range between 10^4 and 10^6 Ci/year (10^2-10^4 Ci/MWe a year);

   typical values for a PWR: up to 10^4 Ci/year (up to approx. 10 Ci/MWe a year).

b) Aerosols 50-100 mCi/year.

c) Halogens generally less than approx. 0.5 Ci/year.

2. LIQUID EFFLUENTS

a) *Mixed fission and corrosion products* (tritium excluded). Less than 10 Ci/year to a few tens of Ci/year.

   Typical values: approx. 0.03 Ci/MWe a year.

b) *Tritium:* a few tens *(BWR)* to thousands *(PWR)* of curies/year.

   Typical values for BWR: approx. 0.05 Ci/MWe a year.

   Typical values for PWR: 10-20 Ci/MWe a year.

In order to obtain an idea of what these activity releases represent in terms of population exposure, one has to consider the different pathways through which the activity can reach man.

The (calculated) exposure values due to radioactive effluent releases from nuclear power plants can be broken down as follows:

	dose rate (mrem/year) at site boundary	*dose rate (mrem/year)* at 5 km
Noble gases		
PWR	< 1	< 0.1
BWR	< 5	< 1
Aerosols and iodine (inhalation)	< 0.5	< 0.05

With regard to the exposure due to the consumption of milk contaminated with iodine-131, an evaluation which is too imprecise to be of any practical significance, it can be noted that I^{131} has never been detected in milk produced near a nuclear power plant (detection limit about 10 pCi/liter, corresponding to a thyroid dose in a small child of about 40 mrem/year).

Dose rates due to the activity released with liquid effluents are conversely estimated to be less than 1 mrem/year.

3. **Comparison between the exposure of man to radiation originating from nuclear power plant operations and that from other radiation sources**

Section II.A.2 above shows that the only normal radioactive effluent releases from nuclear power plants which are worthwhile assessing in terms of the dose limits for the public are the gaseous effluents and more particularly the rare gases. At present, radioactive effluents from fuel processing plants play a minor role in the radiation exposure of the population.

By way of a comparison of human exposure to radiation from radioactive effluents released by nuclear plants with exposures from other sources of ionizing radiation, Table IV gives a survey of dose-rates to man from various radiation sources as well as the whole-body dose limits for different population groups. These data indicate that the doses to critical population groups in the vicinity of nuclear power stations do not exceed 1/100 of the dose limits for such populations as fixed by the Euratom standards and correspondingly no more than about 1/20 of the radiation dose due to the natural background.

4. **Application of "practical" discharge standards**

Until recently most discharge limits for radioactive effluents from nuclear power plants were fixed so as to ensure that the radiological dose limits to members of the population in the environment of the plants were not exceeded. Experience has shown that the actual releases in both gaseous and liquid form were always far below these "radiologically" acceptable limits.

On the basis of this experience and with the aim of reducing man's exposure to a practicable minimum, radiation protection authorities in several Member States and in other countries have recently recommended, or set, much more restrictive limits on discharges from nuclear power stations than those formerly accepted. In certain cases these "practical" limits may be exceeded (theoretically up to the ceiling of the radiological values), provided that adequate grounds are furnished and are accepted by the licensing authorities.

Such "as low as practicable" or "design objective" standards (corresponding to actual experience) serve in particular the purpose of long-term caution and forecasting.

5. **Medium- and long-term forecast for the long-lived nuclides krypton-85 (half-life = 10.4 years) and tritium (half-life = 12.4 years) from nuclear power plants**

a) *Influence of additional retention equipment*

First of all, it should be pointed out that such forecasts have to be handled with care, partly because of a probable increase in the use of additional retention equipment or methods.

Several techniques are already in use or under development for reducing the discharge of radioactive gaseous effluents from power stations (as well as from reprocessing plants), with the particular aim of storing the long-lived Kr^{85} (half-life = 10.4 years) and the short-lived Xe^{133} (5.3 days) on a temporary or permanent basis.

By way of an example, a large twin 1 600 MWe (total) *BWR* station with the usual short-time hold-up equipment could cause several tens of mrem/year individual average whole-body dose due to noble gases at the plant boundary. This would be reduced by a factor of 10 by a recombiner and charcoal delay system.

As regards tritium (H^3), there is no practical way of keeping it from being released into the environment. Total coolant recycling could be applied, but a build-up of tritium would merely shift the problem (e.g., containment contamination and purging) with most likely a greater hazard to the professionally exposed.

The general conclusion to be drawn concerning the additional retention equipment mentioned here is that the advantage of reduced releases to the environment has to be weighed against the greater potential hazards "in-plant" or during subsequent transportation and storage of the accumulated waste.

b) *Future estimated releases of krypton-85 and tritium from nuclear power plants*

Allowing for the reservations pointed out in the preceding section, a rough estimate can easily be made for releases from power plants up to 1985 for purposes of comparison with the expected releases from reprocessing plants under the same programme. The estimates summarized in Table V are based on:

1. the nuclear power forecast for the Community of Six (Chapter I, Table I),
2. typical *BWR* and *PWR* releases quoted in Section II.A.2 [2],
3. and assuming half the units of the *PWR* type and the other half of the *BWR* type.

There seems to be little point in making long-term forecasts as it is impossible to foresee the technological developments which will influence the release situation.

The estimates given here will be compared with those for releases of Kr^{85} and H^3 from fuel processing plants (see next section).

B. *Radioactive gaseous and liquid effluents produced during routine operation of fuel processing plants: medium- and long-range estimates*

A considerable quantity of radioactive material is produced during the operation of power reactors, most of it in the fuel elements where it is retained until they are reprocessed. Reprocessing is the operation in the fuel cycle in which the largest quantity of radioactivity is dealt with and is therefore the source of most of the different kinds of radioactive waste.

[2] *BWR* ~ 10^3 Ci/MWe — year } rare gases
PWR ~ 10 Ci/MWe — year
BWR ~ 5×10^{-2} Ci/MWe — year } tritium
PWR ~ 20 Ci/MWe — year

1. Radioactive waste produced as a result of the reprocessing of irradiated fuel

Categories and types of waste

Reprocessing is the operation in which the fissile materials are separated from the products contained within the irradiated fuel and the materials recovered then purified.

At present, the main process used on an industrial scale is aqueous reprocessing. All the operations involved in reprocessing lead to the production of radioactive wastes of different types and categories; these break down into solid waste consisting mainly of the fuel cladding material, gaseous waste (consisting of the gaseous fission products contained in irradiated fuel, such as Kr^{85}, Xe^{133}, I^{131} and H^3) and aqueous wastes made up chiefly of all the non-volatile fission products after the fissile material has been extracted.

2. Discharge of radioactive effluents into the environment

Since the reprocessing plant is the point at which a large proportion of the radioactive waste produced during the irradiation of the fuel in nuclear power stations is accumulated, it is also the main point of discharge in the entire fuel cycle (fuel fabrication, power plants; enrichment, reprocessing).

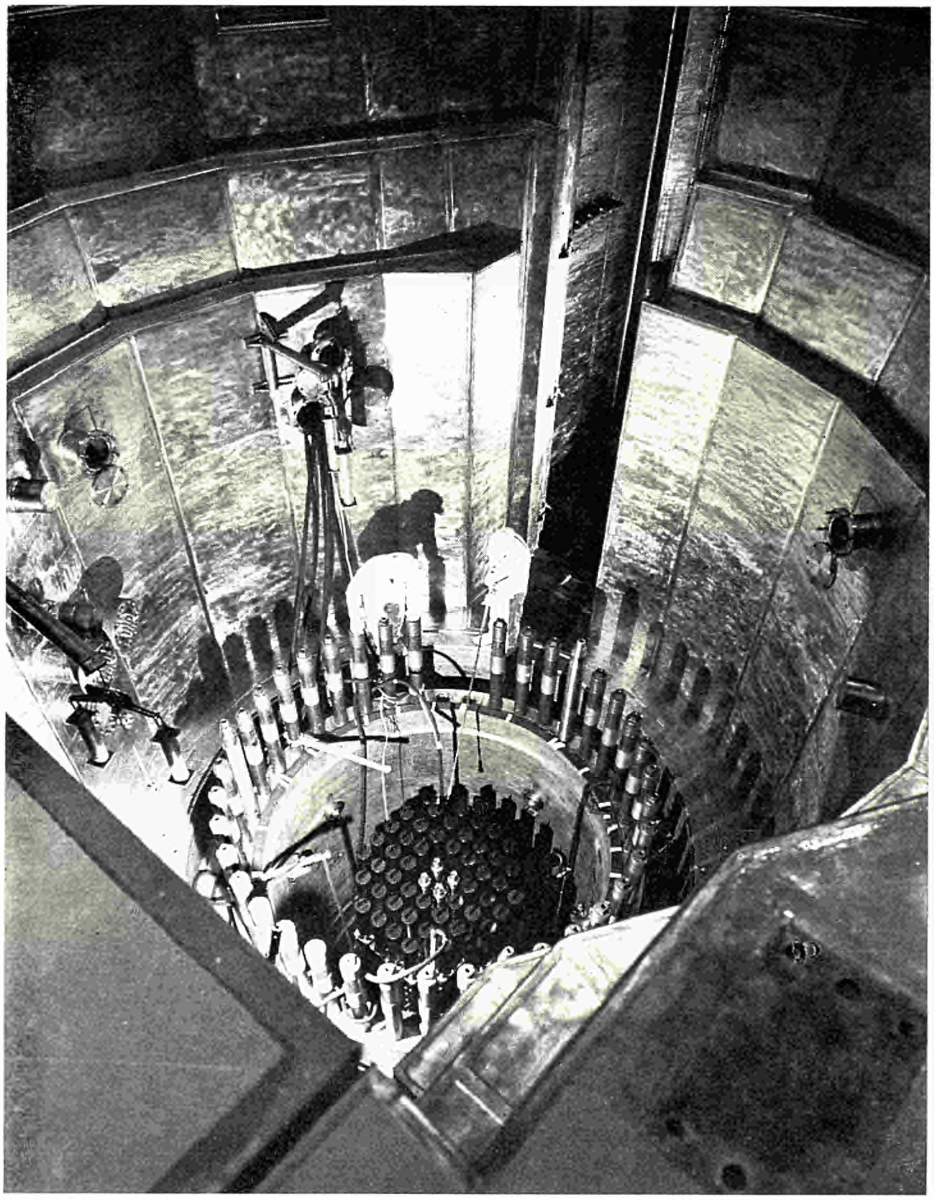
Table V: *Future estimated releases of krypton-85 and tritium from nuclear power plants.*

Year	Installed power	Kr^{85} *(PWR)*	Kr^{85} *(BWR)*	Kr^{85} (total)	H^3 *(PWR)*	H^3 *(BWR)*	H^3 (total)
	thousands MWe	(Ci/year)			(Ci/year)		
1972	5.5	10^4	1.35×10^4	2.35×10^4	5.4×10^4	135	5.4×10^4
1975	12	2×10^4	3×10^4	5×10^4	1.2×10^5	300	1.2×10^5
1980	45	7.4×10^4	1.1×10^5	1.85×10^5	4.5×10^5	1.13×10^3	4.5×10^5
1985	100	1.65×10^5	2.5×10^5	4.1×10^5	10^6	2.5×10^3	10^6

Table VI: *Quantities of* Kr^{85} *and* H^3 *produced in relation to the quantities of fuel to be reprocessed (Illustrative Programme of the Community of Six:* LWR *fuel).*

Year	Fuel to be reprocessed tonnes/year	Kr^{85} Ci/year	H^3 Ci/year
1975	110	1×10^6	8×10^4
1980	720	7×10^6	5×10^5
1982	1 200	1×10^7	8×10^5
1985	1 940	1.7×10^7	1.4×10^6
(2000	~ 9 000 [1]	~ 8×10^7	~ 6×10^6)

1 Quantity influenced largely by proportion of *LWR* uranium fuel and plutonium fuel (recycle and *FBR*).



Two types of effluents—gaseous and liquid—are discharged into the environment by reprocessing plants. The maximum discharge rates for both these effluents are also based on "radiological" protection standards such as those of the European Atomic Energy Community and the *ICRP* recommendations.

The discharge limits for a particular reprocessing installation are laid down in accordance with the basic and derived standards, a minimum dilution factor being taken into account for the point of discharge under consideration.

a) *Gaseous waste*

In the present state of the art, the gaseous waste from reprocessing installations is mainly made up of the rare gases Kr^{85}, Xe^{133} and some of the H^3 in the fuel. Fuel cooled for 150 days contains only a small quantity of Xe^{133} as a result of the decay of this isotope. Iodine-131 (half-life = 8.1 days) is almost completely eliminated by decay during the cooling of the fuel and by specific "off-gas" treatments to retain the iodine.

b) *Liquid waste*

The liquid radioactive wastes discharged by a reprocessing plant consist of liquids which have been subjected to decontamination treatment and therefore contain only very small quantities of radioactive nuclides.

c) *Forecasts for the discharge of radioactive effluents*

A natural consequence of the growing use of nuclear energy will be an increase in the quantities of radioactive waste from plants reprocessing irradiated fuel.

The quantities of nuclides from which no retention treatment is currently employed in reprocessing plants, particularly Kr^{85} and H^3, will therefore increase in proportion to the installed nuclear power.

It should be pointed out that the discharge of H^3 in gaseous form by reprocessing plants represents only a fraction (max 10%) of the total quantity of H^3 in the fuel. Most of the H^3 appears in liquid form (tritiated water) and is discharged with low-radioactivity liquids. The maximum permissible concentration of Kr^{85} and H^3 in the atmosphere is of the same order of magnitude.

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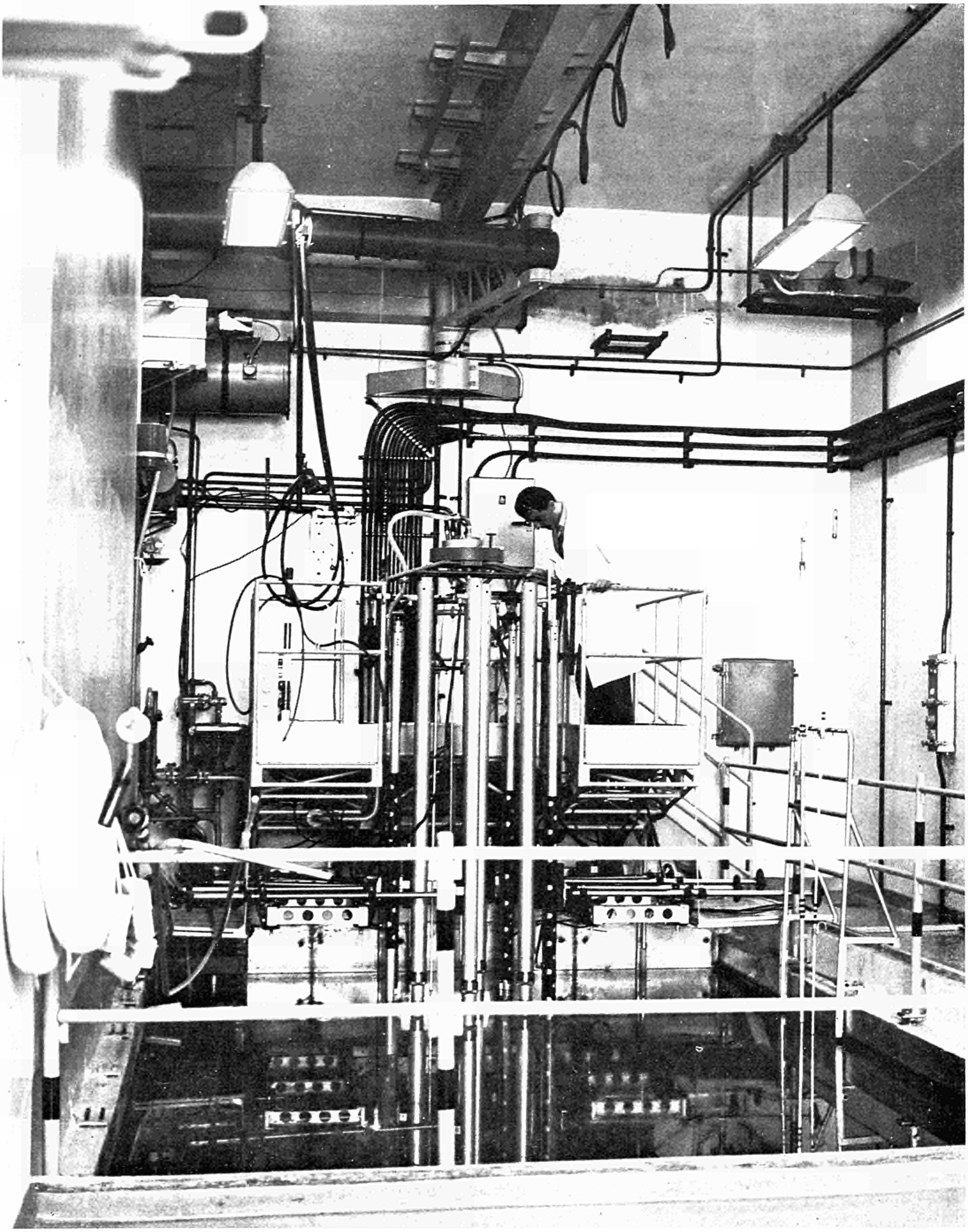


PART II

Table of Contents

C. *Long-range effects of exposure on mankind: medium- and long-range forecasts for the growth in the number of nuclear power plants and reprocessing plants*

1. Conservative nature of estimates

In the estimation of the long-range effects which the potential exposure due to nuclear power plant development will have on the population (general public), a conservative approach is mostly adopted, it being assumed that up to the year 1990 or 2000 there will be no significant contribution from advanced type reactors (especially *LMFBR*). The long-range effects of fuel reprocessing of plutonium thermal fuel (plutonium recycle) and plutonium fast breeder reactor *(FBR)* fuel may become significant around 1990-2000. The decay time before the processing operation starts may be short in the case of *FBR* fuel (30 days as compared to 150 days for *LWR* fuel) because of economies in the Pu cycle, which leads to a significantly higher amount of radioactivity at the time of reprocessing, especially for the relatively short-lived isotopes Xe^{133} and I^{131}. Generally speaking, estimates do not take into consideration further developments in hold-up and retention techniques, so that the conservative nature of the results arrived at is further enhanced.

2. Results

a) *Krypton-85 and tritium*

From Tables V and VI (Part I) it can be deduced that, as far as "global" releases of Kr^{85} into the atmosphere are concerned, reprocessing operations are determinant (by a factor of 100 over nuclear power plants), whilst tritium releases are practically equal for both types of nuclear installation. The same would apply on a world-wide basis.

b) *Global effects of krypton-85*

On the basis of the world-wide energy forecasts, and assuming that Kr^{85} is continuously released and accumulated in the atmosphere, its average annual contribution to the individual whole-body dose for the public could be at the most about 0.05 mrem by the year 2000 (corresponding to an accumulated 3.150 $\times$ 10^6 Ci), and about 0.15 mrem by 2050. As was mentioned previously, about 99% of this would be due to reprocessing operations. With respect to the "global" effects, the increase in the quantity of Kr^{85} in the environment attributable to the use of nuclear energy in the world would not give rise to a significant radiation hazard by the end of the 21st century, as compared with differences in the natural background from one region to another, although there might be some regional accumulations of Kr^{85}. It is most likely, however, that by that time retention equipment will be installed, mainly for "local" requirements, as outlined below.

c) *Global effects of tritium*

The cosmic tritium equilibrium is of the order of 100 $\times$ 10^6 curies (the natural background radiation due to tritium is about ten times this because of residual tritium from weapons tests fall-out). By the turn of the century, the tritium accumulated as a result of nuclear power generation in the world would also total 100 million curies (about 6% of the amount which was present in 1963, on account of weapon tests). After the year 2000, nuclear power will become the main source of tritium in the biosphere and troposphere unless retention means are developed. At present the estimated individual dose to a member of the population at large ranges from 0.04 to 0.06 mrem/year, the higher figure being mainly due to residual tritium from weapons tests. By the year 2000 the average dose would be only about 0.02-0.03 mrem/year. On a "global" basis tritium can be considered to present even less of a hazard than Kr^{85}.

d) *Local effects of krypton-85 and xenon-133 releases*

Rather than the "global" effects, it is the local consequences of Kr^{85} releases near large capacity (5-10 tons/day) reprocessing plants which may require caution (Kr^{85} releases of more than 10^7 Ci/year). Quite apart from the trend towards larger reprocessing plants, a gradual increase in the burn-up of the fuel to be processed also increases the production of Kr^{85}. Furthermore, in the case of Pu fuel processing the Xe^{133} contribution may become determinant (see Section II.C.1). A reprocessing plant with a capacity of 5-10 ton/day would give rise to a whole-body rare gas dose of the order of hundreds of mrem in the immediate vicinity of the plant, and several tens of mrem at a distance of 3 000 m (depending on the type of fuel treated, Xe^{133} being the determinant isotope for *FBR* fuel).

It should be borne in mind that a "park" (multi-unit site) of several (e.g., 10) nuclear power plants could give rise to equivalent local problems, mainly because of the "total" noble gas releases (the relatively short-lived Xe^{133} and Kr^{85}, the main contribution coming from Xe^{133} in this case, however). Such problems could be more serious for *BWRs* than for reprocessing plants if no additional hold-up or retention equipment were provided (e.g., 10 current *BWRs* are equivalent to about one 5 ton/day reprocessing plant from the standpoint of rare gas release).

Table 1: *Estimated average whole-body radiation (mrem/person-year from various sources).*

	1970	2000
Natural background	110	110
Medical	90	100
Global fall-out (weapons)	5	5
Miscellaneous [1]	3	1
Occupational [2]	0.8	0.8
Other environmental (nuclear power and associated industry)	0.07	0.6

1 Television, air transport, consumer goods.
2 The main contribution coming so far from medicine and dentistry.

These long-term considerations will probably also determine the development and use of additional equipment for the retention of rare gases in nuclear power and reprocessing plants. However, this raises the problem of the transportation and disposal of the accumulated Kr^{85}.

e) *Local effects of tritium release*

In the long term, the main aspect of the tritium hazard is "local" receptivity both for high capacity reprocessing plants (5-10 ton/day) and to a lesser extent for multi-unit groups of nuclear power plants (especially of the *PWR* type). For example, a 1 000 ton/year reprocessing plant would discharge about 7.5×10^5 Ci/year of tritium in its liquid effluents, whilst ten 1 000 MWe *PWR* units would discharge about 2×10^5 Ci/year. In Section II.A.2 it was shown that exposure rates could become significant in the immediate vicinity of such installations, if no large dilution capacity (such as fast-flowing rivers or the sea) were available. However, a practical technology might be developed some time in the future for removing tritium from liquid effluents. Also, the fraction of tritium released with the gaseous effluents (10-20%) might conceivably become determinant in a reprocessing plant. Estimates indicate that in the case of a plant treating about 5-10 t/day, the whole-body dose (mainly due to inhalation from plume passage) would still be of the order of "tens" of mrem/year at a distance of 3 000 m from the plant.

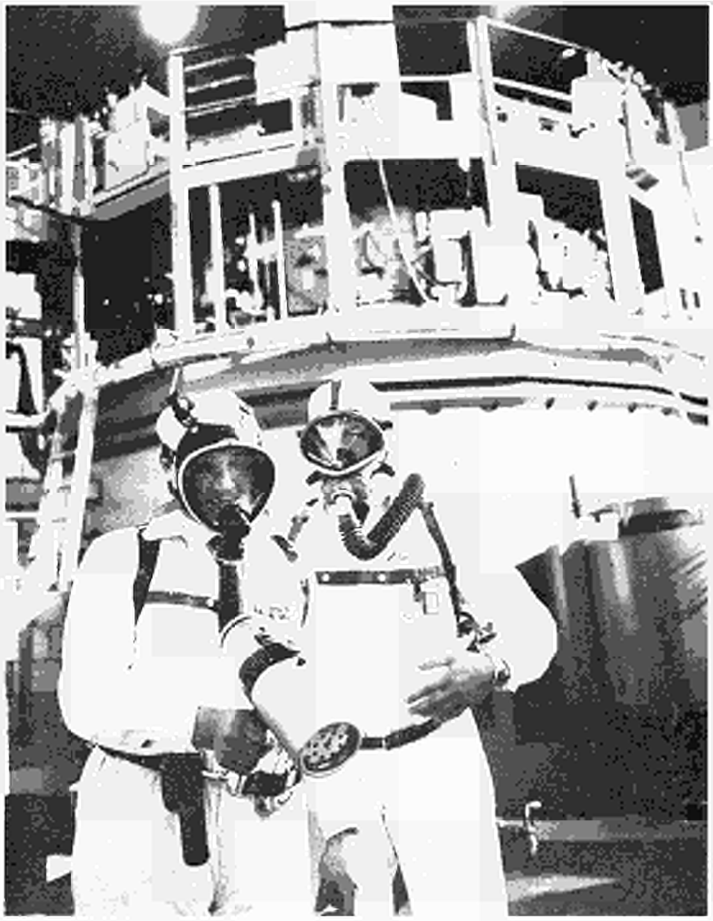
It therefore seems that on a long term basis, and considering "local" factors only, reprocessing plants will become determinant elements in the tritium hazard.

3. **Overall contribution of growth in nuclear power to exposure**

If consideration is given to the present and potential future contribution of nuclear energy towards mankind's total radiation burden, ample evidence is revealed to indicate that its situation in routine operation is a "healthy" one.

Various estimates suggest the trend in the exposure of an average member of the population at large in a highly developed country (e.g., *USA*); these are summarized in Table I.

It can be seen that at present the contribution of the whole field of nuclear energy is less (see also Table IV, Part I) than 1 mrem/person mean annual whole-body exposure (which is genetically significant).



D. *Illustrative comparison of importance of nuclear effluents and effluents due to conventional industry and consumer goods*

1. **General**

No attempt at a comparison of nuclear and conventional activities has been made here, some examples merely being given in order to provide—in addition to the more detailed data on nuclear activities dealt with in other sections—a general idea of the relative hazards to mankind.

Emphasis is placed here on a comparison of "gaseous" effluents because it was seen earlier that these effects are more important to the nuclear industry than discharges of liquid effluents. In later sections a similar illustrative comparison will be made of waste "storage" and accumulation and of thermal effects.

2. **Comparative examples**

The quantities of gaseous effluents discharged from conventional and nuclear power stations are indicated in Table II.

It will be seen that the volume of air required to dilute the quantity of effluents released from nuclear plants in order to meet the permissible radiation standards is much smaller than that needed for conventional coal and oil-fired stations to respect the standards governing conventional pollutants, which is an indication of the relative cleanliness of nuclear power stations.

Other data are available for industrialized countries which give an idea of the present and probable long-term situation as regards pollution.

In the *USA*, atmospheric pollutants from all conventional sources run at present about 125 million tonnes a year, 12.5% of this total being due to electric power (mostly sulphur oxides). They are alleged to cause 20 000 deaths a year; this means that the individual's risks of death from this cause are about 10^{-4} a year (cf Section VI, Part III), which is of the same order as the risks from all types of accident.

In an urban area, 60% by weight of the pollutants are at present emitted by motor vehicles.

The annual rate of pollutant discharge in West Germany is reported to be 1.5 × 10^6 tonnes of sulphur oxides and some 450 × 10^6 tonnes of CO_2 (excluding the CO_2 emitted by motor vehicles). On the world scale, it is estimated that a total of 1.3 × 10^{10} tonnes of CO_2 are at present discharged annually, compared with 10^{12} tonnes of natural origin.

Such examples of estimated effects may be set against the calculated long-term nuclear effects of the accumulation of Kr^{85} and tritium, for instance, the foreseeable consequences of which would, in the year 2000, be by no means worrying on a universal (or global) basis, even assuming that no special retention steps were taken.

III. RADIOACTIVE WASTE STORAGE AND ACCUMULATION

A. *Origin of industrial radioactive waste*

The main waste to be considered comes from nuclear power plants and fuel reprocessing plants.

1. Power plants

Because of the utmost precautions taken in order to keep the doses to the professionnally exposed within the standards laid down and those to the population as low as practicable, the various effluent streams are purified and decontaminated (ion exchange, filters, etc...). This leads to the production of medium-level waste in solid form which is temporarily stored at the plant sites. Moreover, special types of waste are also produced, such as radioactive pieces of equipment (e.g., pressure vessel internals), which have to be handled on an *ad hoc* basis.

2. Fuel reprocessing

Most of the waste (in terms of activity) stems from reprocessing operations and is produced:

a) during fuel decladding: compacted cladding is stored at the reprocessing plant site;

Table II: *Quantities of gaseous effluents discharged from conventional and nuclear power stations.*

Pollutant	*For a 1 000 MWe power station* — *Annual waste discharge* (in millions of pounds)			
	Coal [2]	*Fuel oil* [3]	*Gas* [3]	*Nuclear*
Sulphur oxides [1]	306	116	0.03	0
Nitrogen oxides	46	48	27	0
Carbon monoxide	1.15	0.02	—	0
Hydrocarbons	0.46	1.47	—	0
Aldehydes	0.12	0.26	0.07	0
Fly-ash (retention 97.5%)	9.9	1.6	1.0	0
Radioisotopes/half-life	*Annual discharge (curies)*			
Radium-226/1620 years	0.017	0.00015	—	0
Radium-228/5.7 years	0.011	0.00035	—	0
Krypton-85/10.4 years	0	0	0	10^3 *PWR*
Xenon-133/5.3 days	0	0	0	10^6 *BWR*
Iodine-131/8.1 days	0	0	0	0 *PWR* 0.5 *BWR*

1 Typical values for sulphur content of *USA* fuels.
2 Only fly-ash control.
3 Without pollution control equipment.

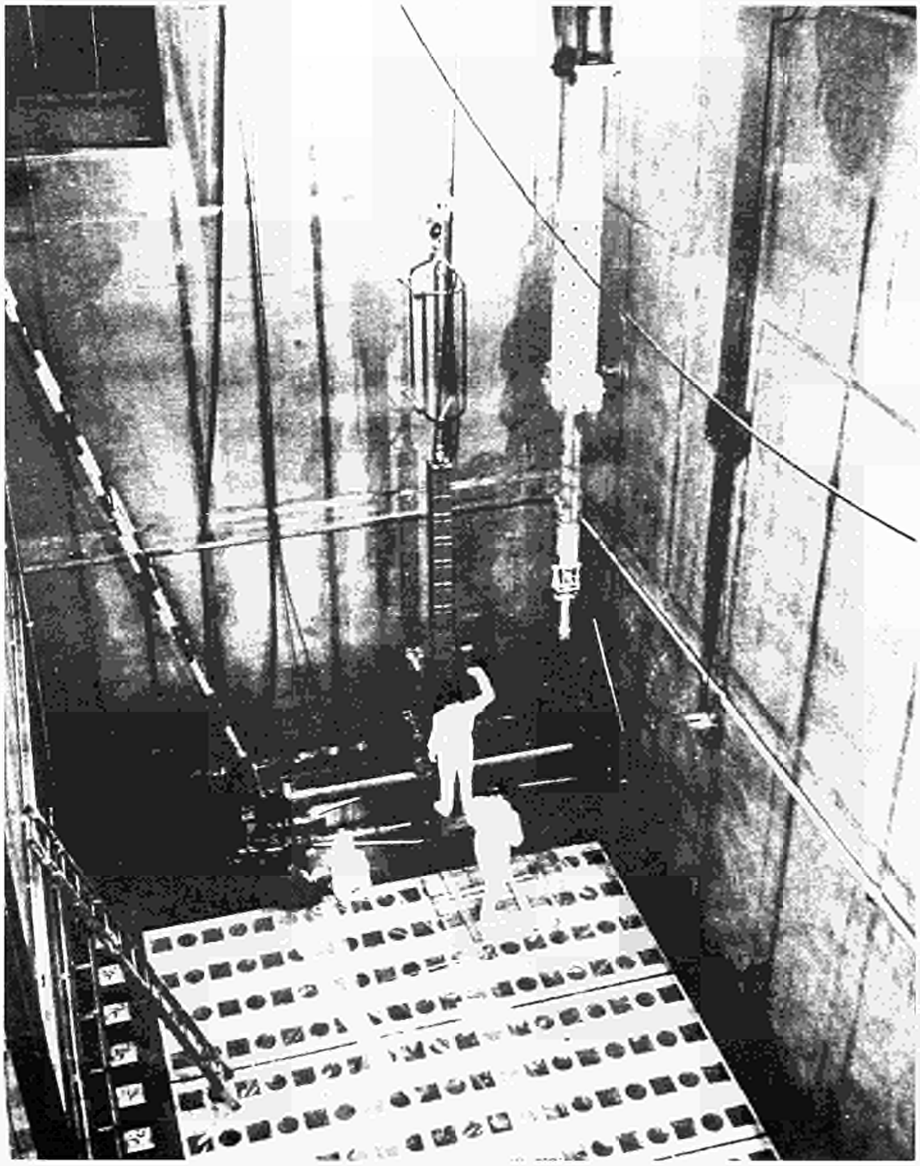
b) during extraction and purification operations.

This waste is treated prior to concentration and concentrated solutions then stored on site.

– The highly active liquid waste ($> 10^4$ Ci/m^3) is concentrated so that it occupies the minimum volume compatible with its temporary storage in liquid form, pending ultimate treatment and storage in solid form. Temporary storage in liquid form for three to five years is economically justified by the decrease in activity, which in turn allows the volume to be reduced further during final solidification.

– The liquid waste of medium (10^{-2}-10^4 Ci/m^3) and low ($< 10^{-2}$ Ci/m^3) activity produced in the plant is usually made up of widely differing types. The concentration processes to which they are subjected are selected depending on the chemical composition of the solutions and their activity. The most usual methods are evaporation, ion exchange and coprecipitation. During these processes the radioactivity is thus concentrated in the evaporation concentrates, the ion exchange resins or the precipitates, according to the treatment used. The residual solution comes into a low-activity class and, as the case may be, can be discharged into the environment if its activity is low enough, or, if its activity is still above permissible levels, must be subjected to further treatment before discharge. The concentrates produced by these treatments come into a higher-activity class and are then dealt with in the same way as other waste in this class or are temporarily stored pending solidification.

The solid or nearly solid waste produced by the treatment of low- or medium-activity liquid waste must later be further processed to facilitate handling and final storage or dumping. The processing systems most often used for this category are incorporation in bitumen or concrete.



B. *Storage, accumulation and ultimate disposal of radioactive waste from operation of fuel reprocessing and nuclear power plants: medium- and long-term forecasts*

1. **Reprocessing waste**

The radioactive waste produced during fuel reprocessing is temporarily stored at the reprocessing plant in liquid form pending its ultimate treatment and disposal in place where there is no possibility of its dispersing into the environment. The presence of long-lived nuclides amongst the fission products implies that the manner in which these products are stored and finally disposed of should ensure that dispersion into the environment can be ruled out for extremely long periods.

More than 95% of the volume of the waste to be stored is in the form of high- and medium-activity liquids. From the safety standpoint, their temporary storage in well-protected stainless steel tanks presents no special problem. The drawback of this method is that, because of corrosion, the tanks have to be replaced after a number of years (several decades). On account of the financial outlay involved and the need for spares to cope with possible accidental leakage, this type of storage is only a temporary solution.

Investigations are therefore being conducted into various ways of storing these wastes for long periods. The solid form of the waste combines the advantages of smaller volume and a limited risk of dispersion. The methods of solidifying solutions of high- and medium-activity waste mentioned earlier are currently being studied on a pilot scale. The choice of the solidified product and its containment depends on the depository requirements and on the medium in which it is to be ultimately disposed of. Final disposal should provide a guarantee that the

Table III[1]: *Waste produced by fuel processing (from LWRs).*

Year	Fuel to be reprocessed	Decladding waste	Unprocessed waste		
			High level waste (fission products)		Medium level (con-centrates)
	t/yr	m^3/yr [3]	m^3/yr	Ci [2]	m^3/yr
1975	110	11	110	0.7×10^9	204
1980	720	72	720	4.4×10^9	1 080
1985	1 940	195	1 940	12.0×10^9	2 915
(2000)	9 000	900	9 000	56.0×10^9	16 000 [2]
	t	m^3	m^3	Ci	m^3
accumulated quantities 1975-85	9 300	930	9 300	57.0×10^9	14 000

1 Cf assumptions in Table VI, Part I, Section II.B.

2 The activity is expressed at the time of reprocessing for fuel cooled for 150 days; as was already mentioned earlier (Part I, Section II.B), Pu fuel may be subjected to lower cooling times and this may become important from 1985 onwards.

3 Assuming mechanical or chop or leach decladding.

radioactive products will not find their way into the biosphere.

It may be pointed out that on a long-term basis a problem is posed not only by the high-level fission products, but also by the foreseeable growth in the quantity of transuranium elements, such as plutonium, with long half-lives and high toxicity.

One of the most promising methods at present under consideration for the final disposal of waste (especially high-level waste) consists of storing it in salt deposits.

It has been estimated that between 400 and 600 $\times$ 10^9 curie of high-level fission product wastes will have accumulated throughout the world by the year 2000, about one-tenth of them being due to strontium-90 and cesium-137.

Table III gives an estimate of the annual production of different classes of waste, related to the quantities of fuel to be reprocessed, on the basis of the Illustrative Nuclear Programme for the years 1975, 1980, 1985 (and tentatively 2000) for the Community of Six. On the basis of the forecasts outlined in Section I (Part I), the figures for the Community of Nine countries are about double.

The level of annual waste production is fairly low up to 1985. The cumulative volumes in the period 1975-1985, on the other hand, represent a considerable amount in terms of both volume and activity. The volume of decladding waste is fairly small in comparison with the total volume of waste. Moreover, since it consists of metal, its storage presents fewer problems.

2. Nuclear power plant waste

a) The solid or semi-solid low- or medium-activity waste resulting from the operation of these plants varies with the reactor type. Some data are given below for *LWRs* gas-cooled reactors and *LMFBRs*.

1. *LIGHTWATER REACTORS*

For *LWRs* the volumes of solid or semi-solid waste, *after dewatering but "before" treatment and packaging*, are reported to be those of Table IV.

It should be noted that the specific activity data are representative of plants of different sizes with different process systems; they are hence not cumulative for a particular reactor.

As an example of a typical two-unit *BWR* station (total about 1 600 MWe), 840 drums/year of demineralizer resins fixed in concrete and 280 m^3 of other packaged miscellaneous low-level waste have been estimated, giving a total yearly volume of 455 m^3.

As an example of a typical two-unit *PWR* station (total about 1 600 MWe) about 300-600 drums are expected, corresponding to a maximum annual volume of about *120 m^3*.

2. *GAS-COOLED REACTORS AND FAST BREEDERS*

These volumes may be compared with those expected for advanced types of gas-cooled reactors. For the Fort St. Vrain plant (330 MWe), for example, an annual solid waste volume production of only 11 m^3 is expected. The same order of magnitude is to be observed in the *GCRs* now in operation.

For the fast breeder reactor of the liquid metal type (e.g., a 300-500 MWe plant) between 14 and 28 m^3 of "packaged" (drums) solid waste a year are expected.

b) Relative importance of medium-level reprocessing waste and power reactor waste—accumulation of this waste

From the information given in Section 2.1 it can be deduced that the volume of medium- and low-level packaged waste from *LWRs* is approximately the following:

– for a *BWR*:

$$\frac{455\ \text{m}^3/\text{yr}}{1\,600\ \text{MWe}} = 0.28\ \text{m}^3/\text{MWe-yr}$$

– for a *PWR*:

$$\frac{120\ \text{m}^3/\text{yr}}{1\,600\ \text{MWe}} = 0.07\ \text{m}^3/\text{MWe-yr}$$

From the nuclear power forecasts outlined in Section I (Part I), the quantites produced yearly and accumulated quantities can be deduced. It will suffice here to indicate that in 1985, for instance, on the assumption of 50% *BWR* stations and 50% *PWR* stations, the yearly production would be (for 100 000 MWe installed):

– *BWR:*

$$50 \times 10^3\ \text{MWe} \times 0.28\ \text{m}^3/\text{MWe} = 14 \times 10^3\ \text{m}^3$$

– *PWR:*

$$50 \times 10^3\ \text{MWe} \times 0.07\ \text{m}^3/\text{MWe} = 3.5 \times 10^3\ \text{m}^3$$

$$\text{total} \quad 17.5 \times 10^3\ \text{m}^3$$

By rough comparison with the volume of medium-level waste produced by corresponding fuel reprocessing operations, the total accumulated wastes for the decade 1975-85 can be quickly estimated.

A 5 t/day reprocessing plant (about 1 500 t/year) would produce about 2 100 m^3/year of medium-level concentrates (unprocessed and unpackaged) (see Table III, column 5). Processed and packaged, this may correspond to about 4 000 m^3/year. Such a reprocessing plant could serve about 45 000 MWe, or 46 nuclear power units of the *LWR* type. These nuclear power units would produce yearly (assuming half *BWR* and half *PWR*) about 7 800 m^3 of low- and medium-level waste (processed and packaged). Broadly speaking, then, nuclear power (under these hypotheses) causes

Table IV

	Volume (m^3/yr)	*Activity (Ci/m^3)*
BWR		
spent resins	5.6 - 11.2	< 7
sludges (condensate) clean-up systems	22 - 64	from 3.5 to 70
PWR		
spent resins	4.8 - 7	175 - 3 500
evaporator bottoms	1.4 - 4.2	< 35

about twice the "processed" waste volume produced as a result of reprocessing operations.

In conclusion, the problem of the accumulation, packaging, transport and ultimate disposal of these wastes is considerable, but it is not more serious than the equivalent problems posed by conventional industrial waste of a hazardous nature.

IV. THERMAL WASTE FROM NUCLEAR POWER

1. General

This section contains a synthesis of the data and considerations relating to the thermal effects on the environment resulting from the operation of nuclear power stations and their relationship to other sources of thermal effects.

Most thermal power plants, whether conventional (fossil-fuelled) or nuclear, use a body of cold water, such as the sea, a river or a lake, to dissipate a considerable portion of the heat generated which, for fundamental thermodynamic reasons, cannot be transformed into electrical energy.

Over the past few years, the effects of the increase in the temperature of the waters close to these power plants have formed the subject of many studies and research programmes. The large quantity of literature on the subject shows in particular that these effects are still not well known, but also that so far directly harmful phenomena have not been identified. Nevertheless, certain problems connected with the release of waste calories are now often referred to as "thermal pollution", because the foreseeable increase in the use of electrical energy in the industrialized countries suggests that irreversible damage would be done to the environment in the next few decades if precautions were not taken.

These preoccupations are based on two main considerations, namely:

a) A relatively slight temperature rise can have a considerable effect on the natural balance of the ecological system of the river, lake, etc., into which the heat is dissipated. The resultant changes generally constitute an impoverishment as soon as certain temperature limits are exceeded.

b) In the extreme case, heat released into the rivers and inland waters could heat up some of them to the point at which their fauna or flora were in danger.

2. Main factors involved in the problems

- In a fossil-fuelled power plant, 38% of the thermal energy is converted into useful electrical energy, while 53% must be evacuated by the cooling water and 9% is dissipated through the stack.
- In a water-cooled nuclear power plant, about 32% (31-33%) of the heat is converted into electricity, while 68% must be dissipated into the aqueous heat sink (river, lake, sea).
- In *HTGR* power plants, the efficiency may amount to about 45%, so that 55% of the heat produced has to be discharged into the aqueous heat sink.
- In breeder reactors, the efficiency is about 40%, leaving 60% of the heat to be discharged.
- The condenser ratio "loss/useful power" is thus:

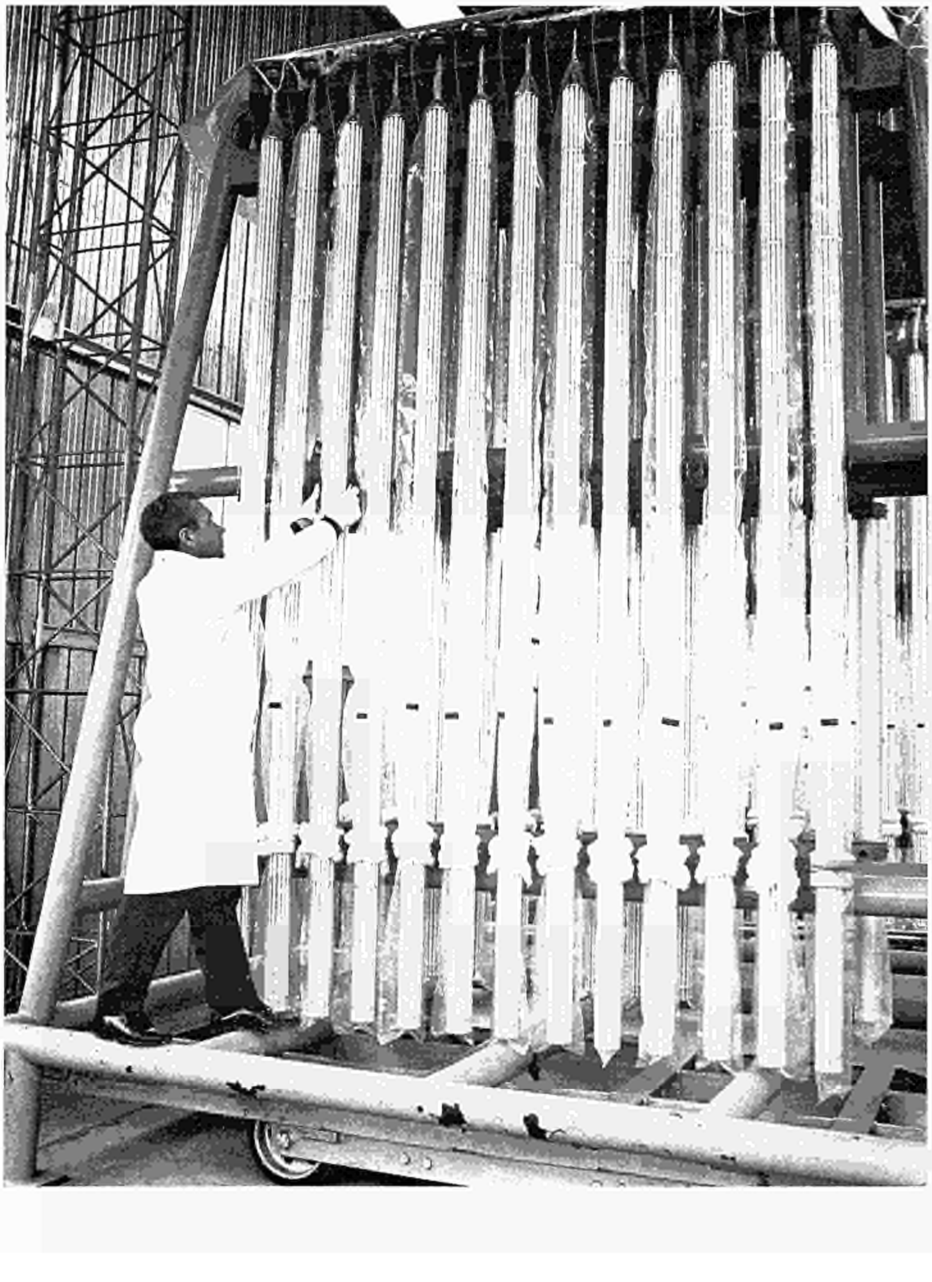
for a fossil-fuelled plant	= 1.4 (1.6 total ratio)
for a *LWR* plant	= 2.1
for a *HTGR* plant	= 1.2
for a breeder plant	= 1.5

Therefore, per kWh of electricity produced, *LWR* plants have to reject about 50% more heat to the aqueous environment than fossil-fuelled plants (*HTGR* 14% less and an *FBR* 7% more). For the same "thermal" energy produced, the *LWR* heat rejection to the aqueous environment is about 28% more than that from the fossil-fuelled plant.

- In the Community, it is estimated that, in the future, nuclear power plants will meet an increasing proportion of electricity requirement, namely, 22% in 1975, 26% in 1980, 40% in 1985 and almost 80% in 2000.
- In the industrialized countries, the need for electrical energy doubles about every 8-10 years; even if energy requirements in all their forms were to increase less rapidly in the future than in the recent past, it must be assumed that demand for electricity will still increase at an important rate.
- For economic reasons, greater power units are used; this results, however,

in local concentrations of thermal discharges.

- A degradation in the ecological system of the waters into which the excess heat is released is in itself a loss which is difficult to quantify. In addition, an increase in the temperature can lead to the destruction of the self-purification mechanism and endanger further use of the water body for drinking or industrial purposes.
- In spite of the enormous quantities involved, the possibilities for recovering the thermal waste from power stations are very limited, apart from a few special cases.



3. Survey of suitable heat dissipation methods

The main methods of obtaining cooling water for the condenser are the following:

a) *Once-through direct cycle*

The fresh water (from the lake or river) passes through the condenser before being discharged directly into the environment. This is the method normally used in the past, and is generally the most economical.

b) *Artificial lakes or reservoirs*

The fresh water passes through the condenser and is discharged to allow the heat to be dissipated by radiation and convection before being returned to the condenser. The trend is increasingly towards this method, sometimes combined with a spray-type cooling system.

c) *Salt-water cooling*

This type of cooling has also been used for many years. However, the corrosion problems usually entail higher costs than fresh water cooling systems.

d) *Cooling towers*

Wet cooling towers can be used if there are only limited supplies of cooling water. Here most of the heat is dissipated by the natural or forced evaporation of a small proportion of the water from the condenser. This type of tower is most frequently used for large power stations; the natural circulation type is employed in particular in Europe and the ventilated, forced-circulation type in the United States.

e) *Air cooling towers*

The principles on which dry cooling towers work is the direct transfer of heat to the ambient air via a tubular radiator. This type of tower costs more to build than wet towers.

4. Some advantages and drawbacks of certain commonly used cooling systems

Each of the methods given above has its advantages and drawbacks from the environmental point of view. First of all, with regard to costs, the situations in Europe and the United States, for example, are not necessarily comparable. Unlike the situation in the United States, the capital costs for the direct cycle fresh water system and for (wet) cooling towers in Europe are of the same order of magnitude. However, the cost of the power generated is about 5-6% higher in the case of cooling towers because of the reduction in the vacuum effect of the turbine. The increase in the cost of energy produced with the aid of air cooling towers ranges from 10% (natural circulation) to 15% (forced circulation).

In the case of light-water nuclear power plants (*PWR* and *BWR*) the use of (dry) air cooling towers is unsuitable (because of the saturated steam form the turbine). However, for high-temperature reactors, with the possible use of gas turbines in the future, these air cooling towers have an advantage because of the increased temperature difference in relation to the cooling air.

For water-cooled power plants, preference is given in Europe to natural-circulation (natural draft) wet cooling towers. Apart from economic considerations militating in favour of this choice, the water vapour from this type of cooling system disperses quickly and the likelihood of the formation of low mist is substantially reduced, thus rendering it acceptable from the microclimatological standpoint. In addition, the capital cost of natural-circulation wet towers is lower.

It is nevertheless quite possible that if construction costs in Europe rise and electricity generating costs continue to drop, "forced"-circulation (mechanical draft) wet towers will become economically interesting in the future (cf. United States [3]).

However, environmental problems could then arise, especially in the case of plants in urban areas, because of the presence of various dusts and the formation of low mist.

5. General criteria (standards)

Basically each site has its own particularities and has to be assessed from the standpoint of waste heat dissipation on its own merits, so that it is difficult to fix general criteria. Nevertheless, some general guidelines have recently been issued in West Germany, for instance. These can be summarized as follows:

- "after mixing", the cooling water released from power plants should never (in any season) heat up the aqueous medium by more than 3 °C, or in exceptional circumstances 5 °C;
- the temperature of the cooling water discharged should not exceed 30 °C, or in exceptional circumstances 35 °C;
- the temperature of the "mixed" water should not exceed the following limits:
  a) 25 °C for waters having summer mean temperatures between 17 and 20 °C (and peak temperatures of 23 °C);
  b) 28 °C for waters having summer peak temperatures of 25 °C.

Usually these guidelines can easily be met. A controversial point still to be settled is that of at what point downstream full "mixing" occurs.

Similar but rather more detailed criteria have been issued in the United States by the National Technical Advisory Committee on Water Quality Criteria.

6. Local and global long-term effects

a) *General*

As in the case of the routine releases of gaseous radioactive waste from a growing nuclear industry (Section II.B, Part I), the question of discharged waste heat can be considered from the standpoint of the local effects on the one hand and of the world-wide consequences on the other.

Notwithstanding the fact that *LWR* nuclear power stations have to dissipate more waste heat than conventional plants, the global and local problems to be examined are not significantly different in the two cases.

b) *Global effects*

Rough illustrative estimates have been made on the basis of electricity production forecasts.

It emerges that, on the assumption that the once-through direct cycle system only will be applied in certain highly industrialized countries, by the turn of the century a major proportion of the available inland waters will be used for cooling purposes (as much as two-thirds of all inland waters in the United States). The situation will probably be roughly the same in Europe. However, the ultimate heat sink for power plants is the atmosphere. If it is assumed that at the plant sites all the heat will be dissipated by the evaporation of water, the inland water thus consumed would represent a minor fraction of the water available (about 1% in the *USA*).

Another indicative example can be quoted. Solar radiation provides about 100 000 times as much heat as all the electrical energy currently produced in the world. Assuming no heat losses from the earth by radiation, the rise in the earth's temperature has been estimated to be about 3 °C annually. On the same assumption, all the heat released by conventional and nuclear power plants during the period from 1970 to 2000 would increase the temperature of the earth's surface by only 0.5 °C and only after a period of operations ranging from 10 000 to 100 000 years.

It would therefore seem that no global problem arises as long as the artifical addition of heat on a planetary scale remains negligible in relation to solar thermal energy.

A more serious long-term problem, relating to global thermal considerations and possible climatic changes, is that of the accumulation of CO_2 in the atmosphere (see Section II.D.2) because of the imbalance between its formation and reabsorption which may occur in the future.

c) *Local effects*

Locally, the thermal problems are by no means negligible. For example, the artificial residual heat to be dissipated in an urban area in the year 2000 has been estimated at about 1.4×10^6 cal/m^2 (500 Btu [4]/sq. ft), compared with 2.8×10^6 cal/m^2 (1 000 Btu/sq. ft) due to solar radiation. Again, however, this is not a specifically nuclear problem.

The standards drawn up in various countries—and applying to all methods of generating electricity and to other industries producing residual heat—have so far resulted only in fairly general guidelines. These could be rendered more specific by drawing on research results and by the developing dispersion models and correlated methods of calculation.

The possible local thermal effects can only be countered by a suitable choice of sites and/or by additional methods of protection (artificial cooling). This will become more and more imperative with the development of multi-unit sites. With this in mind, the use of sea-shore or off-shore sites may increase significantly in countries where such solutions are practicable.

The foreseeable medium and long-term outlook for the development of *HTGR* and breeder reactors is no doubt advantageous from the standpoint of thermal effects but should not be regarded as essential on this count.

Finally, it may be concluded that these problems are not peculiar to nuclear power production and therefore should be dealt with in the context of measures designed to combat excessive thermal pollution from industrial sources in general.

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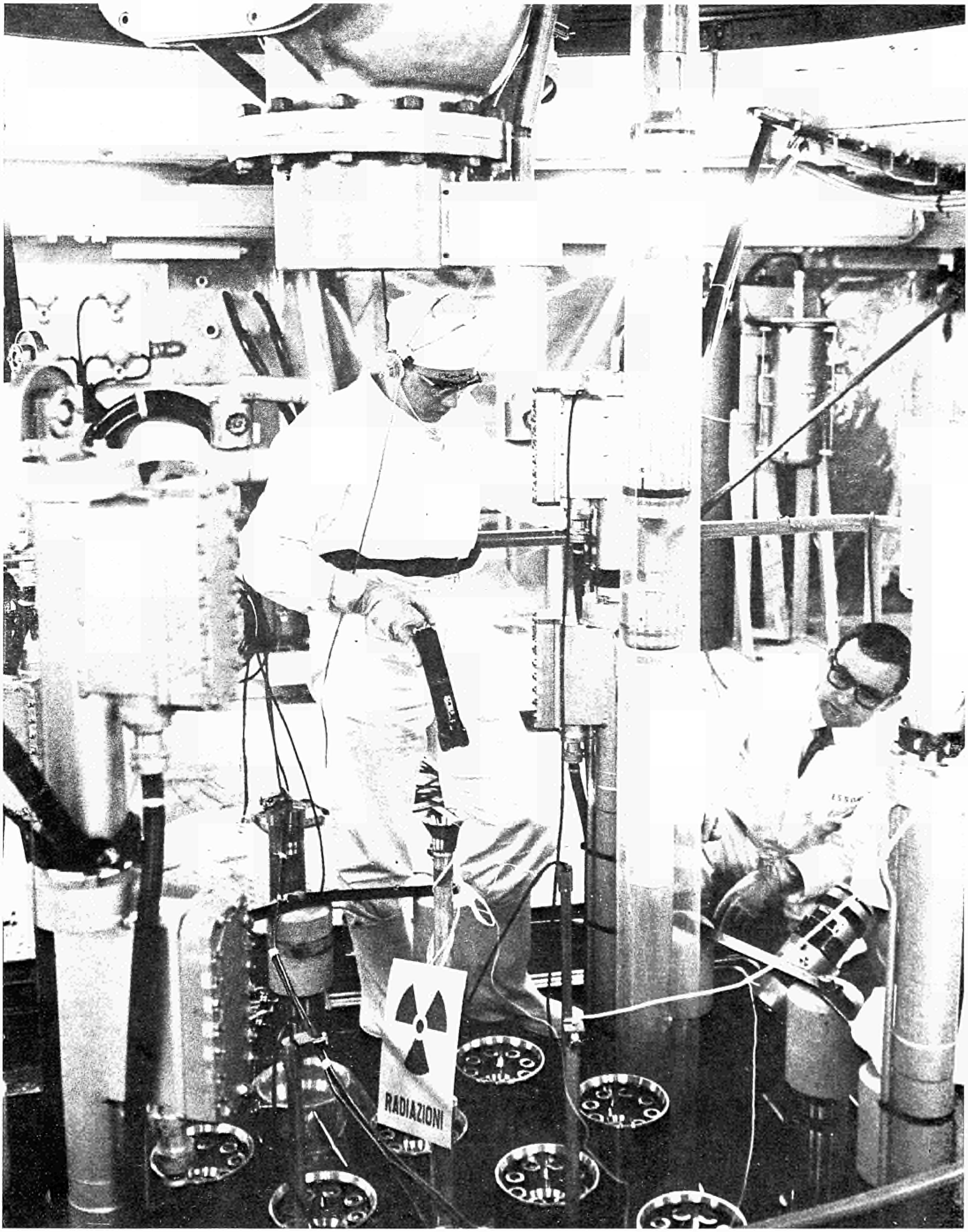
3 in the *USA*:
— wet mechanical draft tower: $ 10-12/kW;
— wet natural draft tower : $ 15-20/kW.

4 British thermal units.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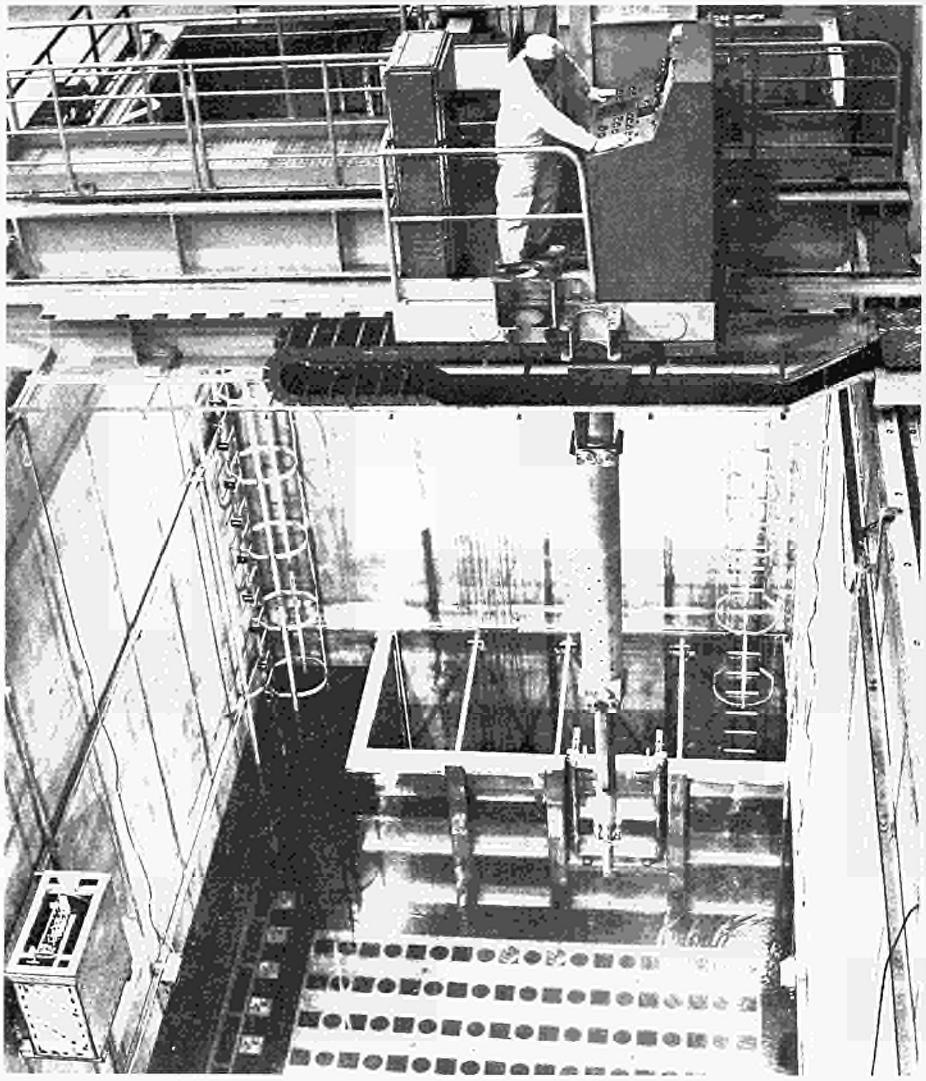


PART III

Table of Contents

V. ACCIDENT POTENTIAL, ACCIDENT PREVENTION AND LIMITATION OF POSSIBLE CONSEQUENCES

1. Precautions of an administrative nature

All nuclear installations are subject to strict control by various authorities responsible for granting building and operating licences according to the procedures and legal requirements in force in each country. Equally stringent precautions are seldom found in conventional (even hazardous) industrial activities or in the use of consumer goods (automobiles).

2. Precautions of a technical nature

The common factor inherent in this type of control consists in a detailed in-depths analysis of the technical safety features of the installation as a whole, and in particular of the systems and equipment designed to limit the radiological or other consequences of all conceivable failures and accidents.

The technical examination of the safety aspects is made first by the design group itself and then by the future operator of the installation. Subsequently, an independent investigation is carried out by the safety and control body delegated by the appropriate regulatory authorities. This independent investigation results in a comparison of the data and opinions supplied on the one hand by the promoters (designer and operator) and on the other by the safety and control bodies. A great deal of effort goes into this procedure.

It may also be considered that these efforts and the confrontation between promoters and "safety and control" bodies lead to a plant design and limited operating conditions of a severity and conservatism unequalled in the non-nuclear industries. It is only since recently that equally detailed safety analyses have sometimes been applied in hazardous conventional industries (e.g., the petroleum and chemical sectors).

Operators and control bodies must also constantly monitor the quality requirements imposed on equipment during manufacture, assembly and prolonged operation (30-40 years).

There is at present an increasingly marked tendency to develop—both nationally and internationally—"technological" standards [5] (criteria, codes and complementary requirements, guidelines, etc.) with the aim of standardizing the methods of design and construction used and the operating limits imposed. This tendency will undoubtedly increase further with the growth in the use of nuclear power and as the prospect for an international market for designs and equipment becomes brighter.

3. Conceivable serious radiological consequences

In the case of industrial nuclear installations, potentially the largest accident hazard capable of affecting the general public (population "at risk" and "at large") will, for many years to come, be that due to power reactors. Generally, several types of potential accident and the corresponding means of mitigating the consequences form the subject of detailed analyses. More and more frequently these analyses incorporate probabilistic considerations such as comparative analyses of the reliability of protection and emergency systems or classifications of accidents according to their severity.

Moreover, the probabilistic approach to the analysis of accidents, which tends to link the probability of events to the seriousness of the consequences, also seems promising in the long run. However, the systematic and quantitative use of nuclear accident probabilities which would be considered acceptable for the general public and the application of these acceptable (tolerable) risks as plant design "target criteria" still gives rise to psychological and practical problems (e.g., uncertainty of the statistical data used and taken as the basis of comparison).

For accidents of maximum conceivable seriousness, it is generally considered acceptable for an individual member of the population to receive 25 rem of whole-body irradiation (somatic and genetic effects) and 15-25 rem of irradiation of the thyroid gland (a distinction sometimes being made between adult and child doses). The precautions limiting the effects of serious accidents on the environment (e.g., containment, ventilation and associated filtering systems) vary to a certain extent with the location and features of a particular reactor. Broadly speaking, however, it may be said that, for a 1 000 MWe plant, the total integrated individual irradiation doses would lie between about 0.1 and 10 rem (mainly due to whole-body irradiation from noble gases), whereas the accident doses due to radioactive iodine would generally be between 0.01 and 0.1 rem.

With the development of *FB* reactors and thermal Pu recycling, the accident considerations will have to include the hazards involved with plutonium.

It is likely that in the future more attention will be directed at accidents which have their origin in "external" causes against which no explicit protection is factored into the design. Examples of such external causes are aircraft crashes or even sabotage.

In general terms, it should be pointed out, firstly, that conventional industries and activities are also fraught with comparable external hazards and, secondly, that the method of protecting man against such hazards can therefore only be the same for both conventional and nuclear activities, e.g., siting away from airports and aerodromes, administrative security measures, exceptional police measures, etc.

4. Present and future siting implications

Together with routine operating conditions (e.g., radioactive effluents, thermal effects, etc.), the accident analysis doses thus estimated determine the acceptability of the chosen sites with regard to the present and foreseeable population distribution around nuclear plants.

It should be noted that:

- "Site criteria" (if based on "accident" considerations) often lead to permissible "exceptions" being made to the "basic" requirements.
- Siting practices are fairly divergent from one country to another (and even within one country) and from the safety standpoint are mostly still dealt with on a "case by case" basis, account being taken of possible additional preventive or accident-mitigating safeguards.

In a growing nuclear industry there will be—in the next decade—a need for more generally applicable site selection requirements based on health and safety considerations both for "routine" and for "abnormal" conditions. It is also likely that, with a growing fraction of population "at risk" (as compared to the rest of the population not living in the vicinity of a nuclear plant), the concept of integrated man-rem dose will be more usefully applied in the future.

For fuel processing plants, the site selection will always remain a "case" study based essentially on "routine" operating conditions and certainly less related to potential accident conditions.

5. Present situation and future outlook

Up to 1970, ninety power plants distributed over the whole world had produced 250 million MWh of electricity and accumulated 650 years of experience without any significant accidents from the point of view of the population at large.

It may be hoped that, by the continued applications of strict standards and precautions and more stringent quality control, this positive balance can be maintained as the use of nuclear power continues to grow.

There is no doubt that, at present, the record of nuclear power as compared with, say, conventional hazardous industries (such as the petroleum and chemical industries) is extremely favourable in terms of the material damage, injury or death caused to the general public and from the point of view of professional accidents (see Section VI). With a growing nuclear industry and the development of higher ratings and new technologies (e.g., *FBRs*), the accident potential tends to increase and in this connection there is certainly every reason to devise methods for the quantitative assessment of future "risk-potential ranges". However, it should be emphasized:

a) that this trend is not limited to the nuclear industry, and applies certainly to a similar extent to conventional industry and hazardous consumer goods also;

b) that, alongside the increase in nuclear power production and the development of new technologies, increas-

5 As opposed to "radiation" protection standards, dealt with in Section II (Part I), for instance.

ingly expanded nuclear safety research programmes are under way which have barely any counterparts in the conventional field.

VI. NUCLEAR HAZARDS IN RELATION TO OTHER RISKS

The quantitative assessment of the hazards from normal operation or from accident conditions which could lead to material or bodily damage (to persons professionally exposed or to the population in general) must be regarded with some circumspection because of the relative value of the interpretations placed on statistical information. A few comparative figures are given here by way of example which show how the nuclear energy "hazard potential" is situated in relation to other industries and human activities the risk of which is generally accepted either by the individual or by the community.

1. Risk in occupational duties

a) *Accidents*

Statistics covering over 22 years' operation of various types of nuclear installation (laboratories, reactors, prototypes, etc.) and 2.5×10^9 man-hours reveal 7 693 individual cases of bodily injury causing disability, of which only 36 (0.5%) were actually due to radiation effects. This gives an accident rate of $\frac{7\,693}{2.5 \times 10^9} = 2.45$ accidents per million man-hours, corresponding to one-quarter of the national rate for all industrial activities for the country concerned.

Moreover, the rate for accidents due to radiation is only $\frac{36}{2.5 \times 10^9} = 0.01$ accident per million man-hours, which is clearly negligible compared with the national rate under consideration.

From the numerous statistics on accidents in various conventional industrial activities, it can be concluded that in industrialized countries accidents have an individual casualty (fatal injury) probability of about 10^{-4} per year of exposure and a permanent injury probability of about 10^{-2}.

In comparison, the nuclear industry is in a very favourable position because of the high level of the health and safety precautions taken since its birth a quarter of a century ago.

Table I: *Probabilities of individual fatal injury (casualty) due to conventional activities and*

Type of risk	Individual probability of fatal injury per year of exposure (orders of magnitude [1])	Remarks
Conventional (casualties only)		
– all diseases	10^{-2}	
– traffic accidents (automobiles)	10^{-4}	[10^{-2} (for light, serious *and* fatal injury)]
– total mortality risk	10^{-3} (men) 10^{-4} (women)	
– accidents of all types	5×10^{-4}	
– smoking	5×10^{-4}	
– traffic accidents (in general)	2.5×10^{-4}	
– suicide	2×10^{-4}	
– falls	10^{-4}	
– air pollution	10^{-4}	
– industrial accidents	10^{-4} (all ages) 10^{-5} (age 20)	
– drowning	3×10^{-5}	
– firearms	2×10^{-5}	
– electricity	2×10^{-5}	
– leukemia (natural causes)	10^{-5}	
– poisoning	10^{-5}	
– coal and oil-fired power stations (pollution)	4×10^{-6}	
– cancer of thyroid (natural causes)	10^{-6}	
– natural disasters	2×10^{-6}	
– lightning	5×10^{-7}	

1 Summary of data from various sources, with slight variations according to the country.

b) *Normal operation*

However, as has been outlined already in Section II.A.2 (Part I), the doses "normally" accumulated by the professionally exposed, expressed in integrated man-rem, although remaining within the permissible radiation standards, may nevertheless lead to a significant exposure. This will increasingly be the case with an expanding nuclear power production and associated industry. This problem has already been tentatively approached by assessing the social value *to the effects of radiation.*

Effects of nuclear radiation (individual injury, not necessarily *casualty*, except perhaps with a long latent time)		
1. *Radiation in accident conditions* (in the assumption that such an accident occurred; "frequency" considerations)	10^{-3}-10^{-4}	1. Based on a linear dose and risk relationship of 30×10^{-6}/person per rad for total irradiation and 1 (any age) to 50×10^{-6} (child) per person per rad for effects on the thyroid gland. 2. Doses generally considered acceptable in accident conditions of 25 rem (total irradiation) and 15-25 rem in the thyroid are taken into account for this risk.
2. *Radiation in normal operating conditions*	10^{-7}	1. At the rate of 1 to a few mrem/year, e.g., individual mean doses received around a nuclear installation. 2. Estimated on the basis of *ICRP* data; linear extrapolation with dose and decreasing dose rate.

of the man-rem concept and the economics involved in "risk" acceptance based on this concept and in supplementary protective measures. It comes down to the question of whether there is any biological difference between 50 occupational people receiving 3 rem/year or 150 persons receiving 1 rem/year (in both cases 150 man-rem), and how the total man-rem received compare with other industrial occupational risks.

These considerations may become a further justification (besides those mentioned earlier with regard to the general public in Sections II, Part I, and V) for introducing "man-rem criteria" also in addition to the individual dose limitations (basic standards).

2. Nuclear risk in general in relation to other risks

Table I shows the likelihood of harmful effects to the public which may result from the use of nuclear power in relation to other risks readily accepted by human society.

It is first of all important to recall that a casualty risk of 10^{-3} per person per year is generally considered unacceptable and means that steps must be taken to reduce it. At a figure of 10^{-4} per person per year we are prepared to spend money (generally public money) to eliminate the causes of accidents or mitigate their effects (e.g. traffic signals, publicity, police, fire precautions, etc.). Below a figure of 10^{-5} per person per year, risks are considered individual risks and are combatted by individual warnings (e.g., handling of firearms, swimming, etc.). Risks of the order of 10^{-6} and below cause no loss of sleep at all.

Remarks concerning Table I

1. Genetic hazards, which do not cause fatal injury in the true sense, have not been included in this comparative table. However, some indications can be given on the basis of mutation rates and the genetic equilibrium of the population. The normal mutation rate has been estimated at $200\,000 \times 10^{-6}$ per person/generation. For an irradiation of 1 rad, $7\,200 \times 10^{-6}$ induced mutations are considered possible, only 2.5% of which are expected in the first generation.
   This risk, of about 7×10^{-3}, corresponds to accidental conditions (dose of the order of 1-10 rem). For normal operating conditions, it would correspond (according to the conservative linearity hypothesis) to 7×10^{-7}, i.e., once again a negligible value.
2. The permissible doses or those actually received by persons professionally exposed to radiation and the general public may, for example, be compared with the genetic effects resulting from the consumption of coffee or alcohol. Thus, the continuous consumption of six cups of coffee a day would correspond to the potential genetic damage equivalent to 5 rem/year, whereas the consumption of 28 ml of alcohol per day would correspond to 50 rem/year. This shows that the individual risks resulting from nuclear energy in routine operation are negligible for the general public, as long as the

stringent radiation standards now applied are observed. As was shown above, there is no reason to believe that this would not be the case over the medium and long haul.

On the other hand, the most serious accident conditions conceivable lead to the class of risk where measures have to be taken. However, the structural precautions taken in nuclear installations are at present usually such that the radiological consequences of those accidents which are regarded as most serious and the least likely to occur still remain a factor of about 10 below the reference doses on the basis of which the risk value of 10^{-3}-10^{-4} is arrived at.

3. Medium- and long-term risk—probabilistic approaches

It is difficult to define accurately the quantified probabilities of the occurrence of various types of accident which would affect the environmental population and in particular to fix a clear-cut border-line between conceivable and inconceivable accidents. The very worst eventuality would be that of the release of radioactive material which exceeded, even by a large margin, the doses considered acceptable for emergency purposes (without evacuation requirements).

Attempts have been made to solve this problem semiquantitatively or quantitatively. The principle is to apply an arbitrary (but reasonably defined) inverse relationship between accident "frequency" and release magnitude and to apply this to urban, semi-ruban and remote sites, taking into account the severity of injury which could be caused by the respective releases, the population densities and the nuclear power production growth requirements.

In order to make a valid comparison between nuclear and non-nuclear risks, it has been suggested that the "crowd-type" of accident hazards be used for the latter. A comparison with estimates of random aircraft crashes around airfields (a risk of between 1 to 100 casualties per crash) would, for instance, indicate a risk two orders of magnitude less for a semi-urban sited nuclear power plant. The latter would be about the same as the risk of death from meteorites.

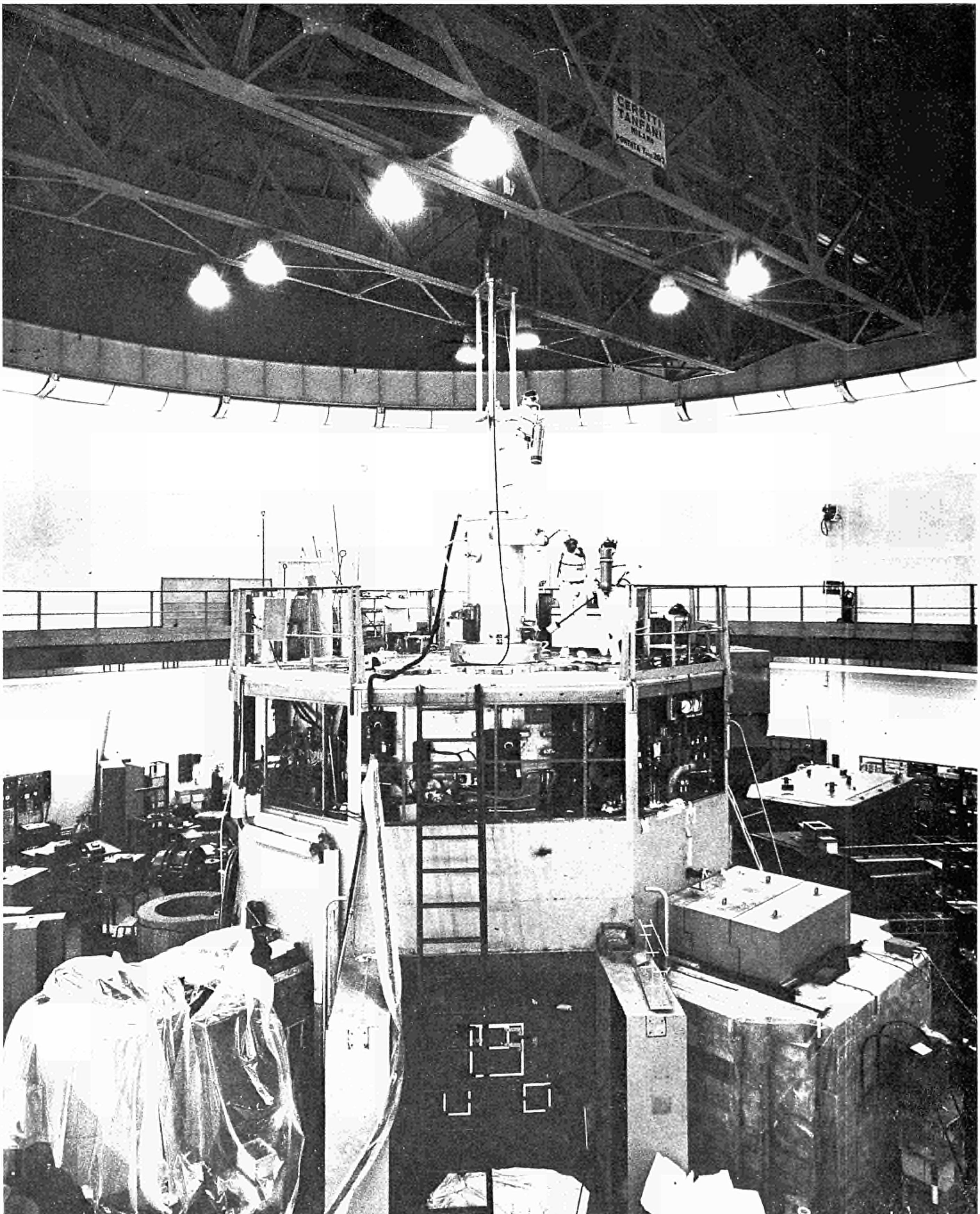
Perhaps the best approach to be adopted as regards "unlikely" (or inconceivable) accidents involving nuclear reactors is to compare them with man-made structures potentially affecting "large" groups of the population (e.g., dam-type developments) rather than with frequently recurring conventional hazards affecting a relatively small number of people. An interesting example which was recently quoted refers to the Netherlands "Schelde River Delta Plan", which is designed to protect about a million inhabitants; assuming an "inconceivable" flood happens once in 10 000 years and causes casualties totalling 0.1% of the population, then the risk is about 10^{-7}. This seventh order of risk is accepted by society with regard to the benefits. Catastrophe-type accidents involving nuclear power plants and causing casualties have risk factors of the same order of magniture or even less.

Another example of a risk can be quoted which is readily accepted in Western Europe and will never cause additional precautions to be taken (structural, warning systems, etc.). On 13 November 1972 a storm hit Western Europe with wind speeds of up to 125 m/s. This is about the most severe hurricane force on record over long periods of time in this moderate climate region.

The last time a storm of equivalent severity affected the British Isles and the Continent on a wide scale was in November 1940, with wind speeds up to 150 m/s. Both these storms caused casualties, injuries and material damage, but information is only available on the latest one; little attention was paid to the 1940 storm, presumably because of war conditions. If we confine ourselves to the "casualties" out of a population of roughly 200 000 affected, about 54 persons were killed. This risk, spread out over the period between 1940 and now (roughly 30 years) amounts to about 10^{-5} casualty risk/person-year. It should also be borne in mind that this is certainly a "non-benefit" type of risk, as opposed to the previous examples. The risk here is two orders of magnitude greater than that expected from catastrophe-type accidents involving nuclear power.

This type of estimation and comparison with nuclear power undoubtedly requires further examination but if offers inherent attractions on a long-term basis.

EUSPA 13-10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Technical Notes

The Commission's *technical notes* give descriptions of original results obtained under the Euratom research programmes. Their purpose is to enable firms to decide whether they should consider industrializing these results.

On the basis of article 12 of the Euratom Treaty, a non-exclusive licence may be granted on the results covered by patents, in so far as the licensee is in a position to make effective use of these results. The conditions of the licence, as well as the royalties for technical assistance, will, for each individual case, be fixed after joint consultation.

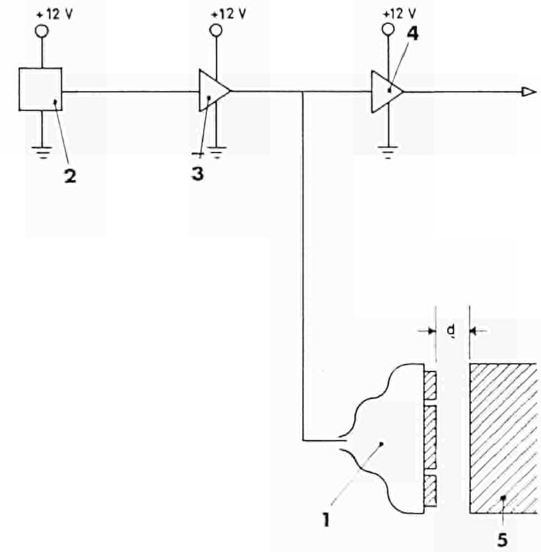
Requests for additional information should be sent to: Commission of the European Communities, D.G. XIII-A, 29, rue Aldringen, Luxembourg.

— 2014: Capacitive sensor for displacement measurement

This capacitive sensor is intended for measuring displacement or distance of conducting or non-conducting materials. The only condition of use is that the dielectric constant of the material to be measured must be different from that of the medium in which the measurement is being made.

The sensor (1) consists of two electrodes; their form, their arrangement and the distance between them depend on the kind of measurement to be made.

One sensor, with a gap of 1 mm between the electrodes and a total capacitance in air of 8 pF, has been tried out; it gives excellent results for rapid (1945 m/s) and slow (4.5 m/s) displacements of the object being measured (5). The electronic measuring circuit consists of an oscillator (60 MHz) (2), a preamplifier (3) and a demodulator-amplifier (4).



— 2015: Vapour concentration detector

This is a description of a dectector suitable for the continuous measurement of vapour concentrations in a carrier gas, such as air or any inert gas.

A light source emits a spectrum of light which becomes almost monochromatic after passing through a series of filters or a monochromator. Only the spectral band corresponding to the band spectrum of the type of vapour to be detected is allowed to pass through. A photomultiplier measures the amount of light absorbed, and this is related to the vapour concentration by an exponential law.

The vapour enters a tube through which the light passes. This tube contains two quartz windows welded to the metal by a process developed at the *JRC* in Ispra; the Société de Verrerie et de Thermométrie, Paris, France, has been granted the licence for this process (Technical Note 2008, *euro-spectra* 1973, vol. XII, No. 3, p. 87).

The tube through which the vapour passes may be heated to 500 °C in order to avoid any condensation on the quartz windows and to reduce the background noise.

A special detector has been developed for Gilotherm OM2 vapours and has also been used to measure the vapour concentrations of other refrigerating liquids, such as AKPM, HB4O, Thermip and OMPH. The concentrations measured vary between 0.02 mg/l and 100 mg/l.

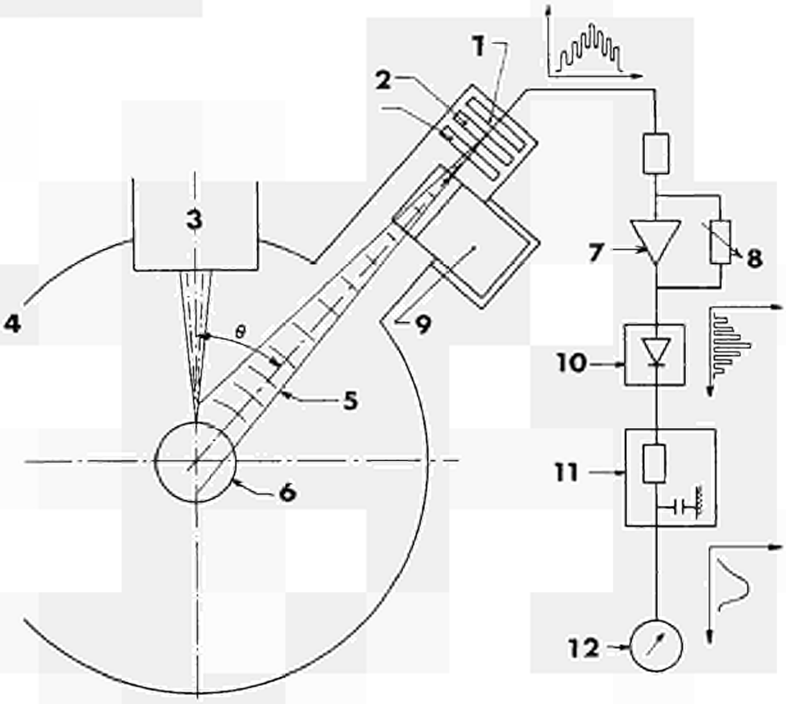
— **2016: Electron-beam welding: optimum focusing**

The *JRC* in Ispra has developed this method of achieving optimum beam focusing of a welding electron beam gun, by using the infrared radiation emitted at the beam's point of impact, and thus avoiding the use of additional fittings in the weldcasing at the impact point.

The infrared radiation is transformed into an electric signal, the amplitude and form of which provide information about the beam's optimum focusing point and about the form of the weld carried out by this means (surface weld or stud weld).

A mechanical chopper system protects the radiation sensor from the metal vapour released by the welding process. It is possible for the beam concentration to be automatically controlled by means of the electric signal.

1. Photovoltaic cell
2. Infrared filter
3. Gun
4. Welding chamber (∅ int. 800)
5. Measuring core
6. Welding workpiece
7. Amplifier
8. Gain control
9. Rotating cylinder
10. Rectifier
11. Integrator
12. Galvanometer



— **2017: Heavy leakproof tongs**

These tongs are designed for fitting to remote handling devices of type M8, M9 (or 8HD), D and F (Société La Calhène).

In tongs commercially available until now, the booting acted as the seal around the movable parts of the body of the tongs. The disadvantage of this method was that it often led to damage to the booting around these movable parts.

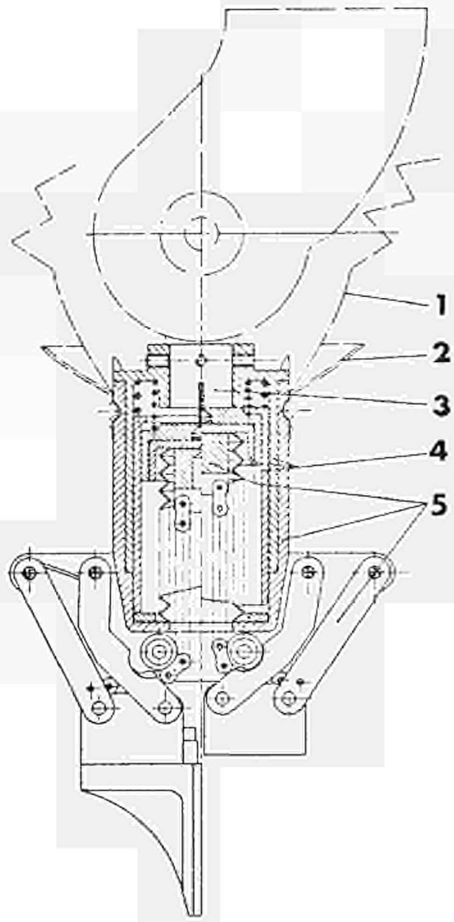
In tongs of the type described, sealing is ensured inside the tongs themselves by bellows.

Brief description

The tongs comprise two parts joined together by a ball-bearing interlock system controlled by the gripping action of the tongs.

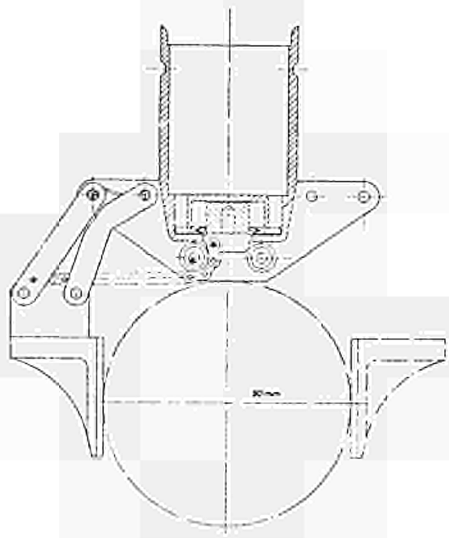
Part 1: a jointing unit, firmly fixed to the stump emerging from the wrist of the remote handling device.

Part 2: the tongs themselves, made from light alloy, consist of 4 arms controlled by a chain device for "traction holding". Gripper noses, which can be disconnected, are fixed to the extremities of the arms. The booting seal is fixed into a specially designed groove on the upper part of the body of the tongs.



Characteristics:

- Maximum opening between gripper noses: 90 mm.
- Load capacity of the tongs: 40 kg.



This equipment, sold under the trade name PLK 89, is manufactured under licence by the Société La Calhène, 5 rue Emile Zola, 95-Bezons, France.

1. Sleeve α
2. Protective spring guide
3. Bevel-gear stub shaft
4. Coupling mechanism
5. Body of leak-tight clamp — clamping mechanism

— 2018: 256 Channels portable pulse analyser for Gamma-rays

The necessity of a true portable analyser is apparent for many applications where the primary importance is related to the possibility to act independently of any external supply.

The analyser is assembled in a completely autonomous suitcase which contains:

— integral line with NaI crystal;

— NiCd battery for 10 hours of operation;

— battery charger;

— chart recorder for data output;

— quartz controlled timer;

— solid state display for the time in a preset-count operation.

An input is provided for an external solid state detector (GeLi or intrinsic-Ge). Energy ranges are selectable in five steps up to 2MeV, and the maximum sensibility is 0.5 KeV/channel. The core memory (256 channels $\times$ 2^{16} max. content) can store up to 4 spectra of 64 channels.

Two analog thresholds allow to select a restricted range of energy. Measurements can be performed for pulses whose amplitude is comprised within these thresholds both in single-channel and in multichannel analysis. The rate of these pulses is also measured by a ratemeter (10^3 pps f.s.).

Data out possibility are the following:

— point recording of spectra in internal chart recorder;

— print out on external printer (print driver not included);

— display on external oscilloscope.

The total power consumption is about 5 W.

— **2019: „AE-6" electrode spharpener**

The AE-6 was designed to sharpen the tungsten/thorium electrodes used in arc and inert gas *(TIG)* welding of the miniature thermocouples which form part of the instrumentation controlling nuclear fuel elements.

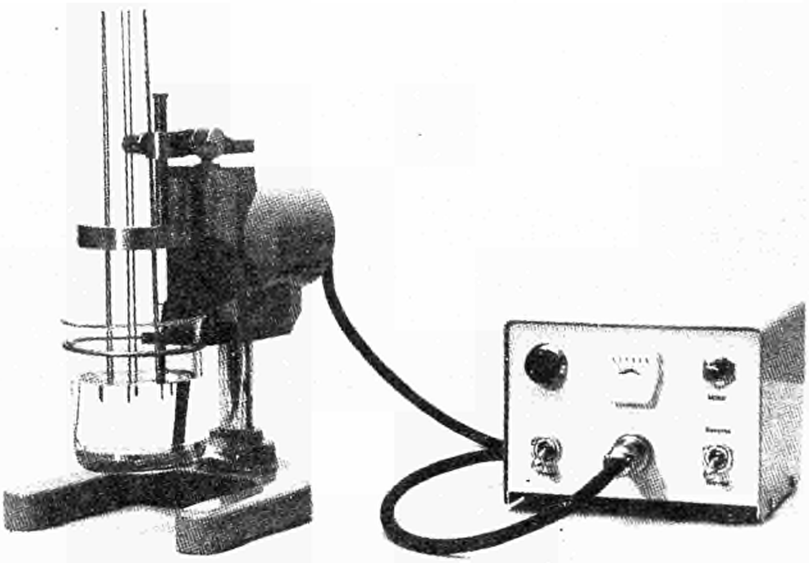
The sharp points required to weld, inside a sheath of 0.25 mm internal diameter, the hot junction of a thermocouple, i.e. two wires some hundredths of a mm in diameter, are hard to achieve by grinding or machining.

Mechanical sharpening in fact causes the point to break.

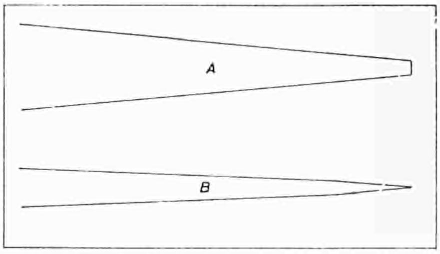
The AE-6 uses electroerosion in conjunction with an electrode mover device. The standard model makes 6 points simultaneously in less than 60 min. and requires no supervision.

The electrode holder, designed for a diameter of 1 mm, can on request be supplied for other dimensions, and the electrolytic tank for tungsten and its alloys can be supplied for other metals and their alloys.

This simple device permits best advantage to be obtained from the new types of *TIG* microwelding machines currently available on the market. It also provides extremely sharp metal points or electrodes which can be used in areas other than microwelding, e.g. biology, medicine, micro-manipulation, etc.



The AE-6 sharpener is manufactured under licence by Franco Corradi, Via Cornaggia 11, Casella Postale 98, I-20017 (Milan), Italy.



Reproduction of the picture given by a section projector (magnification × 100):
A) point made by conventional method;
B) point made by the AE-6.

Fifth symposium on microdosimetry

to be held on 22-26 September 1975 in Verbania Pallanza (Lake Maggiore), Italy.

In September 1975 the Commission of the European Communities will be holding the *Fifth Symposium on Microdosimetry* as part of the programme on "Biology and Health Protection". An initial announcement has just been published and is available to interested scientists on request.

Earlier symposia on microdosimetry have highlighted the rapid development of microdosimetry, which is of great practical importance for radiological protection, neutron therapy and an increased knowledge of the biological effects of radiation.

The purpose of the symposium is to facilitate a comprehensive exchange of information on the progress made in microdosimetry and to stimulate further research into the problems involved in the spectral and spatial distribution of the energy transferred to irradiated tissue and in radiation energy deposition in organs, cells, sub-cellular particles and biological macromolecules. The physical, chemical and biological aspects of microdosimetry will be examined during the symposium. In this connection special attention will be paid to energy transfer and energy deposition by monoenergetic ions, fast neutrons and their secondary particles, as well as to radiation at low *LET* values.

Further information can be obtained from:

Conference Secretariat for the
Fifth Symposium on Microdosimetry,
Commission of the European Communities,
DG XII - Biology Department,
Attention Dr. H. G. EBERT,
Rue de la Loi 200
B-1049 Brussels, Belgium
Tel.: 735.00.40
Telex: COMEURBRU 21877

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- **Convenzione Eurex Euratom-C.N.E.N. Relazione annuale 1971.** *EUR 5062 i.*
- **Radioaktive Kontamination von Lebensmitteln in den Gemeinschaftsländern im Jahre 1971.**
  Contamination radioactive des denrées alimentaires dans les pays de la Communauté en 1971.
  Contaminazione radioattiva delle derrate alimentari nei paesi della Comunità nel 1971.
  Radioactieve besmetting van voedingsmiddelen in de landen van de Gemeenschap in 1971. *EUR 5063 d, f, i, n.*
- **Irradiation of plutonium fuelled coated particles in FRJ 2.** *EUR 5066 e.*
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- **Environmental radioactivity. Ispra 1972.** *EUR 5118 e.*
- **Bibliography on heat transport in nuclear fuels and related phenomena.** *EUR 5119 e.*
- **Eine Neutronenquelle hoher Intensität mit definiertem Spektrum.** *EUR 5120 d.*
- **Interface zur Steuerung einer Multichopperanlage.** *EUR 5121 d.*
- **Fourth Symposium on microdosimetry. I.** *EUR 5122 d - e - f.*
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COMMISSION OF THE EUROPEAN COMMUNITIES

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euro abstracts — Section II — COAL AND STEEL

a new edition of euro abstracts

Starting January 1975, the Commission of the European Communities will regularly publish a second section of "euro abstracts" devoted entirely to reseach under the Treaty of Paris establishing the European Coal and Steel Community (ECSC).

The new section will be published by the Directorate-General for "Scientific and Technical Information and Information Management" in collaboration with the Directorates-General for "Industrial and Technological Affairs", "Social Affairs", and "Energy", and will be entitled "euro abstracts Coal and Steel".

Information on research supported by the ECSC budget pursuant to Article 55 of the Treaty of Paris will be arranged in four sections, as follows:

— medium-term research programmes, one-year or six-monthly research progress;

— research agreements;

— final reports and other publications arising from research;

— patents arising from research.

Within each section the information will be listed under one of the following three headings:

— technical research relating to coal;

— technical research relating to steel;

— social research (occupational hygiene, safety and medicine).

The information will be published in three languages (English, French and German) in the form of summaries preceded where appropriate by bibliographical references.

The Commission of the European Communities is convinced that this new section meets a real need and that "euro abstracts Coal and Steel" will arouse the interest of scientific and industrial circles by keeping them regularly informed of the progress of coal and steel research at Community level.

"euro abstracts - section II - Coal and Steel" will be distributed free of charge in 1975. Requests should be directed to the

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