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COMMISSION OF THE EUROPEAN COMMUNITIES

THE PRESENT AND FUTURE SITUATION OF NUCLEAR ENERGY PRODUCTION AND ITS ASSOCIATED INDUSTRY - NORMAL OPERATION, ACCIDENT PREVENTION AND MITIGATION, COMPARATIVE RISK ASSESSMENT

by

W. VINCK
with the cooperation of
H. BOOS, F. LUYKX, H. MAURER, J. VAN CAENEGHEM

1973

Directorate-General
for Industrial, Technological and Scientific Affairs, Brussels
and
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Paper presented at the International Meeting of the American Nuclear Society, Washington (USA), November 12-16, 1972
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ABSTRACT

The report gives first an outline of the present status and of the future trends in nuclear energy development in the six Member States of the European Community and in the frame of its future enlargement. Emphasis is placed upon the technical safety aspects, the siting and environmental considerations and the implications for mankind as compared to conventional hazardous industries.

The radioactive effluents routinely released at present by nuclear power and reprocessing plants are investigated and the future increase estimated on the basis of :

— the second Community Target Nuclear Programme;
— typical BWR and PWR releases, and
— the nuclear capacity installed assuming that half of it being of the PWR type and the other half of the BWR type.

It was found that only the noble gas releases in the gaseous effluent streams give rise to special supplementary precautions, justified not really by global effects through gradual build up in the biosphere and the atmosphere, but rather by local or regional effects in the vicinity of the plants.

Finally it is concluded that at present the record of the nuclear power industry is, compared to conventional hazardous industries, extremely favourable from the point of view of human injury from gaseous and effluent releases for the professionally exposed as well as for the general public.

KEYWORDS

REACTOR SAFETY
REACTOR SITES
ENVIRONMENT
RADIOACTIVE WASTES
POWER PLANTS
FUEL REPROCESSING PLANTS
EUROPEAN COMMUNITIES
BWR TYPE REACTORS
PWR TYPE REACTORS
RARE GASES
REACTOR OPERATION
REACTOR ACCIDENTS

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Preamble and General Conclusions

A - PRESENTATION

The present report gives first an outline of the current status of the future trends in nuclear power development in the Six Member States of the European Community and in the frame of its enlargement to Nine countries.

The quantified effects of the growth of nuclear production and associated industries from the standpoint of routine occupational exposure, effluent releases and waste accumulation and disposal—on the environmental issues of interest to mankind, are assessed with regard to the forecast development within the Community and extrapolated and examined on a world-wide scale.

With any human activity—be it on an individual or collective scale—corresponds a risk which can be qualitatively or to a certain degree of precision quantitatively assessed. The same applies to the peaceful applications of nuclear energy. This report summarizes some data and ideas which may help in appreciating the nuclear safety record and risk—now and in the future—under routine operation and in potential accident conditions, as compared to other risks to which modern society is exposed and most of which are readily accepted.

The topics dealt with have been grouped into 6 chapters, each forming an entity :

- Chapter I : Nuclear power forecasts
- Chapter II : Radioactive effluents from a growing nuclear industry
- Chapter III : Radioactive waste storage and accumulation — or processing wastes
- Chapter IV : Thermal waste from nuclear power production
- Chapter V : Accident potential, accident prevention and limitation of the possible consequences
- Chapter VI : Nuclear hazards in relation to other risks.

The references used are grouped per chapter, i.e. : according to their main application. Sometimes they apply to other chapters; in such a case this has been mentioned. Most of the references have been explicitly indicated at the relevant place, when they were applied; however, those of more widespread application have not been quoted explicitly in the text.

The opinions expressed by the authors of the present report do not necessarily engage the views of the Commission of the European Communities.

B - GENERAL CONCLUSIONS

On the basis of the quantitative data and findings outlined in the various chapters of the present report the following summarizing conclusions can be put forward.

1. *The standards of radiation protection* applied in nuclear industry for the professionally exposed as well as for the general public are fixed in a cautious and conservative manner. It is only recently that an analogous conservative approach starts to be applied for numerous other sources of man-made pollution (conventional power and industry, consumption goods, etc.).

2. *From all the radioactive effluents* routinely released at present by nuclear power and reprocessing plants only the noble gas releases in the gaseous effluent streams give rise to special supplementary precautions.

   With growing energy requirements and an expanding nuclear industry on a medium- and long-term basis, the only nuclides which will imply more of this special caution will be Kr^{85}, Xe^{133} (noble gases in the gaseous effluents and—to a lesser extent—H^3 in the liquid effluents (as well as gaseous effluents at reprocessing sites). This caution will be justified, not really by global (or overall planetary) effects through gradual build up in bio- and atmosphere, but rather by local or regional effects in the vicinity of the plants.

   Supplementary hold-up and retention equipment for noble gases is most likely to be further developed and applied in the following decades in multi-unit power production sites and larger-capacity reprocessing plants (5-10 t/day). The long-term prospects of using more plutonium-fuels may influence the problem near reprocessing sites, because of a foreseeable shorter decay time of the irradiated fuel before processing starts.

   For large-size reprocessing plants and multi-unit power plant sites, the tritium releases may also have to be reduced in the long run by appropriate retention means yet to be developed. It could be that the determining factor would be the Tritium released with the off-gas streams from such reprocessing plants rather than the liquid effluents.

   The application of such supplementary hold-up or retention equipment should after all be carefully weighed in general and in each particular case. Indeed, will the apparent supplementary direct gain in protection of the population justify a possible relaxing in the protection and dose-burden of the occupationally exposed and increases in medium level waste volumes to be treated, handled, stored, transported to burial grounds and possibly disposed of?

3. The global contribution of routine operation of nuclear power production facilities and associated industry in the *radiation burden to the population* at large (as a whole) is at present and will remain for the next century a small fraction (less than 1%) of the total natural and man-made radiation (mainly from medical purposes). This remains valid even assuming that within that period of time *no* special hold-up or retention equipment were developed for the noble gases and tritium.

4. *The waste heat* to be dissipated in growing quantity in the aquatic environment (thermal pollution) of thermal power plants (conventional as well as nuclear) does not seem to be a problem on the planetary scale and on a medium and long-term time scale. Locally however —in particular for multi-unit sites or for (semi-) urban areas—careful site selection and/or artificial cooling methods will become more and more necessary. This will not be facilitated because of landscaping considerations or fog (and smog) formation possibilities connected with the use of cooling towers.

   It is not excluded that—in connection with these problems—the use of sea-shore or offshore sites be increased especially in countries where this is feasible and economically attractive.

   With respect to the local effects, the standards proposed in various countries—for all types of thermal power plants and other industries producing residual heat—have resulted so far only in rather general guidelines which merit to become gradually more detailed, in particular as a result of research and by the development of dispersion models and correlated methods of calculation.

   No discrimination should be made, in connection with residual heat rejection, against the nuclear generation of power as more particularly worrying source of pollution as compared to conventional industries.

5. The utmost precautions taken to limit strictly the releases of radioactive effluents from nuclear energy applications so as to protect adequately the general public, have as a consequence *the treatment and accumulation of medium and high level wastes.*

The well-controlled presently applied *temporary storage* at the plant sites is not a cause of concern to the public.

With growing energy demand, the accumulated volumes of these hazardous industrial wastes will increase in importance, and *ultimate disposal methods* will have to be sought for. The *highly radioactive* fission products wastes from fuel reprocessing operation are the most difficult to handle although the foreseeable volume will not be exceedingly large.

Compacting, solidification and disposal methods are already in an advanced stage of development and will presumably be ready for large scale application within the next decade or two. The *medium level wastes*—although from the radioactivity standpoint less of a problem—may after all constitute a more complex problem as they are largely produced at the nuclear power production sites (rather than at the fuel reprocessing sites) and therefore the foreseeable accumulated volumes are much larger than in the case of the high level wastes. This implies also more frequent transportation requirements in connection with central cemetery storage and/or ultimate disposal.

This problem will also have to be solved within the next decades. It should be borne in mind however that the long-term problem of this type of waste is not exceptional for the nuclear industry. In fact, it is to be compared with the even more voluminous hazardous wastes produced and accumulated by conventional hazardous industries (e.g. chemicals such as cyanides), which are at present less controlled than their nuclear equivalent.

6. At present the record of the nuclear power industry is, compared e.g. to conventional hazardous industries, extremely favourable from the point of view of human injury or casualties resulting from the *accident situations.* This is true for the professionally exposed as well as for the general public.

This is due to a large extent to the stringent administrative and technical safety analysis and quality control requirements applied, of which seldom an equivalent counterpart is found in conventional hazardous activities.

Besides the severe 'radiation protection standards' referred to under paragraph 1 above, the gradual further development of 'technological safety standards' is also a requisite of maintaining this degree of safety in a growing nuclear industry.

New technological developments (e.g. FBR's) and higher power ratings will, with increased nuclear power requirements, necessitate amplification of associated research programmes and stimulate quantitative 'risk-potential' assessment (sec. par. 8 below). Again the present trend in this is already now of such importance that no equivalent counterpart is found in the conventional hazardous activities.

7. The growing number of *nuclear power production plants* will require within the next decade the development of more (than is presently the case) generally applicable *site selection requirements* based on 'routine' as well as 'abnormal conditions', health and safety considerations.

The growing fraction of population 'at risk' (living in the vicinity of a nuclear facility, as compared to the total population) is *one* of the factors which may contribute to applying in the future also the integrated 'man-rem' concept into 'radiation standard' requirements for siting purposes (see also par. 8 below).

8. An illustrative comparison of the present *'risk-potential' of nuclear energy* to the professionally exposed as well as to the general public—as compared to the risks generally accepted in modern society—confirms its very favourable position with regard to 'normal operation' conditions and indicates its comfortable position with regard to 'abnormal or accident' conditions.

With the future expansion of nuclear power production programmes this will readily remain so for 'normal' operation (see e.g. chapter II), although the professionally exposed will be subject to an increased integrated man-rem consumption. This may in the long run become another incentive of applying this integrated dose concept, besides the 'individual' requirements (see par. 7 above).

On a medium- and long-term basis also, the 'accident'-potential should normally remain within the present range because there is no reason to believe that the measures and precautions outlined under 6 above will in the future be relaxed. It is likely however that 'probabilistic approaches' including for instance also the 'man-rem' concept—will be particularly useful to define more precisely the total nuclear risk and to weigh it versus the benefit gained for mankind and versus other accepted risks which are *not* necessarily compensated by a benefit.

9. It may be pointed out that the stringent safety precautions and environmental protection measures which have always been standard practice in nuclear technology also have a beneficial effect through their gradually more frequent application in conventional activities, for example the use and intensive development of diffusion models and calculation programmes in the field of ecology, geology, hydrology and meteorology in general, or the systematic safety analyses in hazardous activities.

10. Now that human society is, quite rightly, devoting more and more attention to its environment, it is to nuclear energy that it *must* turn, since in the long term, the use of most of the other sources of power is likely to become prohibitive from this point of view. Any delay in development in this direction caused, for example, by strictly psychological and not objective considerations, can only *damage* the cause of environmental protection instead of furthering it.

CHAPTER I

Nuclear power forecasts

Nuclear power production forecasts have always to be dealt with applying a certain caution : the starting purpose of the predictions and the assumptions used can be different from one forecast to another.

Two examples of recent forecasts have been used here as a check basis for further considerations.

1. The second 'Target' Nuclear programme for the European Community (Ref. 1) - Forecast No 1

As pointed out by its title, this programme which was issued by the European Commission in 1972 sets out a 'target' on one hand taking into account the trend of nuclear power development versus coal, petroleum products, natural gas, geothermal and hydro power and assuming on the other hand that necessary measures be really taken to eliminate a number of barriers which hinder the requisite speeding up of nuclear plant constructions and operations.

In summary this Target programme makes the following forecast up to 1985 for the original Six Member States of the Community :

TABLE 1

Year	Power (Thousands of MWe)
1970	3.2
1972	5.5
1975	12.0
1980	45.0
1985	100.0

These estimates are based partly on projections from the units either already producing or firmly decided or under construction. No allowance is made in this case, for possible delays in the presently established scheme by the utilities and/or within a national energy programme.

2. **Forecast of growth of nuclear power - world-wide Forecast** (Without USSR and other Communist countries) - **Forecast No 2**

This forecast prepared by the USAEC Division of Operation Analysis and Forecasting (WASH — 1139 Reactors; TID — 4500) was issued in January 1971 (Ref 2).

It is in fact optimized towards the requirements for nuclear fuel (from ore and separative enrichment work, including assumptions on a certain amount of plutonium recycle).

From this forecast the data are summarized in Table II up to 1985 for the Community of Six Member States, the enlarged Community (9 Member States as of January 1st 1973) the other countries of Western Europe, the USA, Canada and Japan.

The indication 'enriched' refers generally to power plants using light water reactors, the indication 'natural' refers to power plants of the gas cooled or heavy water moderated type.

TABLE 2

Estimate of cumulative capacity of nuclear power plants in Western Europe as compared with forecast for USA, Canada and Japan

(Up to 1985 — thousands of MWe)

	1970	1975	1980	1985	1972	1973	1974
Belgium (Luxembourg)							
Enriched	0.1	0.9	2.6	4.2	0.1	0.1	0.5
Natural	—	—	—	—	—	—	—
Total	0.1	0.9	2.6	4.2	0.1	0.1	0.5
France							
Enriched	0.2	1.2	6.3	14.4	0.2	0.2	0.8
Natural	1.3	2.5	2.5	2.5	1.8	1.8	2.5
Total	1.5	3.7	8.8	16.9	2.0	2.0	3.3
Germany (West)							
Enriched	0.8	5.9	15.6	31.3	0.9	2.2	2.2
Natural	—	—	—	—	—	—	—
Total	0.8	5.9	15.6	31.3	0.9	2.2	2.2
Italy							
Enriched	0.4	0.4	4.1	9.4	0.4	0.4	0.4
Natural	0.2	0.2	0.2	0.2	0.2	0.2	0.2
Total	0.6	0.6	4.3	9.6	0.6	0.6	0.6
Netherlands							
Enriched	0.1	0.5	2.3	5.2	0.1	0.1	0.5
Natural	—	—	—	—	—	—	—
Total	0.1	0.5	2.3	5.2	0.1	0.1	0.5

TABLE 2 (continued)

	1970	1975	1980	1985	1972	1973	1974
Community of 6							
Enriched	1.6	8.9	30.9	64.5	1.7	3.0	4.4
Natural	1.5	2.7	2.7	2.7	2.0	2.0	2.7
Total	3.1	11.6	33.6	67.2	3.7	5.0	7.1
United Kingdom (Eire)							
Enriched	0.1	5.4	19.6	43.1	0.8	2.2	4.1
Natural	4.6	5.2	5.2	5.2	5.2	5.2	5.2
Total	4.7	10.6	24.8	48.3	6.0	7.4	9.3
Denmark							
Enriched	—	—	0.8	2.6	—	—	—
Natural	—	—	—	—	—	—	—
Total	—	—	0.8	2.6	—	—	—
Community of 9							
Enriched	1.7	14.3	51.3	110.2	2.5	5.2	8.5
Natural	6.1	7.9	7.9	7.9	7.2	7.2	7.9
Total	7.8	22.2	59.2	118.1	9.7	12.4	16.4
Austria							
Enriched	—	—	1.3	2.0	—	—	—
Natural	—	—	—	—	—	—	—
Total	—	—	1.3	2.0	—	—	—
Finland							
Enriched	—	—	1.0	2.6	—	—	—
Natural	—	—	—	—	—	—	—
Total	—	—	1.0	2.6	—	—	—
Norway							
Enriched	—	—	0.5	1.6	—	—	—
Natural	—	—	—	—	—	—	—
Total	—	—	0.5	1.6	—	—	—
Portugal							
Enriched	—	—	0.3	0.6	—	—	—
Natural	—	—	—	—	—	—	—
Total	—	—	0.3	0.6	—	—	—

TABLE 2 (continued)

	1970	1975	1980	1985	1972	1973	1974
Spain							
Enriched	0.6	0.6	5.6	13.3	0.6	0.6	0.6
Natural	—	0.4	0.4	0.4	—	0.4	0.4
Total	0.6	1.0	6.0	13.7	0.6	1.0	1.0
Sweden							
Enriched	0.4	2.7	7.9	16.6	0.4	0.4	1.2
Natural	—	—	—	—	—	—	—
Total	0.4	2.7	7.9	16.6	0.4	0.4	1.2
Switzerland							
Enriched	0.4	2.0	4.4	7.5	0.7	1.0	1.4
natural	—	—	—	—	—	—	—
Total	0.4	2.0	4.4	7.5	0.7	1.0	1.4
Other countries of Western Europe							
Enriched	1.4	5.3	17.9	37.4	1.7	2.0	3.2
Natural	—	0.4	0.4	0.4	—	0.4	0.4
Total	1.4	5.7	18.3	37.8	1.7	2.4	3.6
USA							
Total	5	59	150	299	19	32	46
Canada							
Enriched	—	—	—	—	—	—	—
Natural	0.2	2.6	7.8	17	1.6	2.1	2.6
Total	0.2	2.6	7.8	17	1.6	2.1	2.6
Japan							
Enriched	—	4.9	20.6	48.8	1.1	1.6	2.8
Natural	0.2	0.2	0.2	0.2	0.2	0.2	0.2
Total	0.2	5.1	20.8	49.0	1.3	1.8	3.0

Similar to the 'Community Target Programme' this forecast has been based on electrical generating capacity installed, firmly decided or under construction up to approximately 1977 and on further extrapolation (average percent increase per year) from 1978 onwards.

However in this forecast more conservatism has been built in. For instance, up to about 1975 (depending on the period for which information is available) the plants have been assumed to start commercial operation in the year *following* the one 'predicted' by the utilities, in order to take in account delays (e.g. less experience, construction and licensing delays). Subsequently the 'predicted' schedule has been assumed for the US forecast, whilst the one year delay has been maintained for all the other countries.

Furthermore the USAEC forecast does *not* take into account improvements in the matter of hindrances in nuclear power plant development, which the 'Community Target programme' presumes as a prerequisite for the validity of its forecast, such as subdivision of the market in national areas, discrepencies in technical and safety standards, spread of construction industries, lengthy procedures due to implementation of safety and environmental requirements.

The USAEC report also indicates that for instance taking account of the lessons learned from previous forecasts, it would be better to speak about 'ranges' of installed MWe especially the further one moves into the future. For instance this leads for the USA to a range of 130 thousands of MWe to 170 thousands of MWe at the end of 1980, and of 260 thousands of MWe to 330 thousands of MWe at the end of 1985 (as compared with respectively 150 000 and 299 000 MWe in 1980 and 1985). For the other forecasts, grouping several countries, the possible spread (or error) versus the total estimate figures arrived at in table 2 could be of the order — 20% and + 10% in 1980 and of the order of ± 15% in 1985.

3. Comparison of the recent Forecasts 1 and 2

It can be noted that up to 1975 forecasts 1 and 2 are largely corresponding. However the indications for 1980 and 1985 for the Community countries are in forecast 2 significantly lower than in forecast 1. The 'Community Target programme' estimates are even higher than the upper level range of error for the USAEC forecast. This can—as already indicated—be explained by the difference in scope and basic assumptions in the forecasts and by the inherently conservative approach in the USAEC estimate. Also the provisions for Italy, [1] France and Germany for the period 1980-1985 are in the USAEC forecast low by over 1/3 as compared to the national forecasts referred to in the 'Community Target programme'.[2] This leads by itself to a difference of as much as *29* thousands of MWe between the two forecasts for 1985.

4. Medium and long-term assumptions chosen for the safety and environmental implications

For the purpose of further considerations in this report, the estimations retained for the total nuclear energy forecasts up to *1985* (medium-term forecasts) will be those of the 'Community Target programme', as seen from the point of view of the safety and environmental implications—this implies the more 'conservative' assumptions.

In the same perspective for the period 1985-2000 (long-term forecasts), the following assumptions can be taken.

Within the Community of Six at present half the nuclear power units use the light water reactor type (LWR). It is expected that this type would cover 75% of nuclear energy production in 1975 to reach practically 100% in 1980 (cfr. also USAEC forecast).

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1 According to Ref. 3 : 1975 1 400 MWe
1980 6 500 MWe
1985 20 000 MWe

2 In 1985 : West Germany 45 000 MWe
France 27 000 MWe
Italy 16 000 MWe.

For the Community of Nine the cross-over will be slower and the ultimate proportion of light water reactors for that period will depend on the decisions to be made for the UK nuclear power policy (Ref. 4).

TABLE 3

(Thousands of MWe)

Year	Community of Six	Community of Nine [1]
1990	210	315
1995	380	570
2000	620	930

[1] Arrived at by projecting the USAEC forecasts for UK-Denmark-Eire with nuclear power growth factors applied for the Community of 6.

It is expected that other types of reactors (e.g. fast breeders HTGCR) will start contributing significantly within the Community of Six around 1985. Again this may occur earlier for the Community of Nine depending for instance on the UK programme.

With regard to the latter uncertainties, the further considerations in the present paper have been optimized towards reactors of the light water cooled type (LWR). This should anyhow give a reasonable estimate of the pollution potential from nuclear power in normal (routine) operation conditions, and in accident situations, bearing in mind that gas cooled reactors, heavy water reactors, or fast breeders may be more advantageous in some aspects and less in others.

CHAPTER II

Radioactive effluents from a growing nuclear industry

A - THE RADIOACTIVE EFFLUENTS PRODUCED IN ROUTINE OPERATION OF NUCLEAR POWER PLANTS: RADIATION STANDARDS, CURRENT PRACTICE, MEDIUM- AND LONG-TERM FORECASTS

1. Radiation protection standards - e.g. EURATOM standards

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

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

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

(3) 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 revised periodically on the basis of the recommendations of the International Commission for Radiological Protection (ICRP) have been and 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 of radionuclides in the atmosphere or in drinking water.

These radiological dose limits may not be regarded as threshold values below which there will be no somatic or genetic effects : for all such effects, the starting point used is a linear dose/effect relationship. From this is derived the essential principle of radiation protection : 'the exposure of persons, and the number of persons exposed to ionizing radiation must be kept as low as practicable (Ref. 1)'.

2. The application of radiation protection standards in the generation of nuclear power - actual release experience from power plants

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

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[1] By population 'at large' is meant here the population *as a whole*, i.e. : professionally exposed, so called population 'at risk' (adjacent to nuclear installations), and the rest of the population.

Persons professionally exposed to radiation in the field of the generation of nuclear power are exposed to both external and internal radiation.

Experience has shown that, in nuclear power stations, internal contamination of personnel occurs only rarely. On the other hand, external radiation often gives rise to problems with regard to the respect of the radiation protection standards, especially in the case of inspections, maintenance and repair (Ref. 2). This is mainly because provisions are not always made at the design stage for access to and the maintenance of various components. The result may be that the operator of the installation is forced to make use of outside labour in order to avoid over-exposing its own staff, and this, in turn, can raise problems relating to e.g. :

(i) the psychological preparation of such outside labour;

(ii) their training in matters of radiation protection;

(iii) the detailed planning of the scheduled work;

(iv) accounting for the time spent working in controlled areas and the doses received.

It should be noted that, in order to eliminate these difficulties, the Commission of the European Communities (CEC) has decided to have studies carried out of the measures which can be taken, both at the design stage and in practice, in nuclear power stations to reduce the degree of exposure to radiation of maintenance and inspection staff (Ref. 3).

For comparison's sake it may be pointed out this time that in fuel processing plants the occupational exposure hazard may be considered somewhat higher but the number of installations is and will remain much lower.

The population living in the vicinity of nuclear power installations are subject to the hazard of exposure to the gaseous or liquid radioactive effluents discharged into the environment by these installations.

Here, a distinction must be made between the effluents from nuclear power stations and that 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 noble (or rare) gases, mainly : Xe^{133} and Kr^{85}, and sometimes also radioactive iodine and aerosols (particulates).

In BWR's the gaseous effluents originate generally from the condenser air ejector, the gland seal, the turbine building and the mechanical vacuum pump. The major portion originates from the condensor air ejector and because of the relatively short hold-up time (about 30 min), the contribution of short-lived isotopes of Krypton such as Kr^{85} and Kr^{88} and of Xenon such as Xe^{133}, Xe^{135}, Xe^{138} is still significant at the time of release.

The application of supplementary retention by recombiner and charcoal delay systems for the condensor air ejector systems reduces the released activity by a factor 10-20.

In PWR's the gaseous effluents originate generally from the gas processing system, the steam-generator blow-down vent, the auxiliary building and containment purging. Usually 4 containment purges are assumed for the purpose of calculation; 0.25% leaking fuel and about 75 l leakage from primary to secondary are usually also considered. Because of the long hold-up time (around 45 days) of the radioactive gaseous effluents from the gas processing system the short-lived isotopes of Kr and Xe have mostly decayed, so only Kr^{85} and Xe^{133} contribute significantly in this case.

In the case of advanced gas cooled reactors the essential source of gaseous effluents is the primary coolant purification system, with almost all the activity due to Kr^{85} (expected : about 3 CI/MWe-year; Ref. 7, chapter III). For the more conventional types of gas cooled reactors mainly Ar41 contributes, formed by neutron activation of Ar40 present in CO_2.

For the liquid metal fast breeder also almost 3 Ci/MWe – year (Kr^{85} essentially) is *expected,* assuming 1% of failed fuel (Ref. 8, chapter III).

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

By way of example Table IV lists a survey of the discharges made the last years from a number of nuclear power plants within the Community. The data summarized here have been obtained from published information, specially references (4, 5, 6, 12, 13, 14, 54, 55, 56 and 57).

Other full and accurate data have recently been published more extensively, on the normal discharge rates of gaseous and liquid effluents from nuclear power stations in operation (especially in the US), mainly for the years 1968, 1969 and 1970 (Ref. 11 through 24).

Upon examination of the values reported, one can summarize these grossly as follows :

1. *Gaseous effluents*

(a) noble and activation gases

typical values for a BWR [1] : range between 10^4 and 10^6 Ci/year
(10^2 to 10^4 Ci/MWe – year)

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

(b) aerosols ca. 50-100 mCi/year ([2]).

(c) halogens generally less than ca. 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 : ca. 0.03 Ci/MWe – year.

(b) Tritium : a few tens (BWR) to thousands (PWR) of Curies.

Typical values for PWR : ca. 0.05 Ci/MWe – year.

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

To have an idea of what these activity releases represent in terms of population exposure, one has to know the different pathways through which the activity can reach man.

For the gaseous effluents the principal ways of exposure are :

— external irradiation by the radioactive cloud;

— inhalation of radioactive aerosols and iodine;

— consumption of milk contaminated with iodine-131 (through the chain air-grass-cow-milk).

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1 Lower limit for plants with supplementary retention equipment of gaseous effluents.

2 Discrepant figures are quoted in literature : values of expected activities up to the order of 10 Ci/year are also quoted (Ref. 14, 30).

TABLE 4

Gaseous and liquid radioactive waste disposals from nuclear power plants in Europe

Plant (net output MWe)	Year	Annual load factor (%)	Gaseous: Noble gases released /Ci/y/	Gaseous: Noble gases release limit /Ci/y/	Gaseous: Aerosols released /Ci/y/	Gaseous: Aerosols release limit /Ci/y/	Gaseous: Iodine 131 released /Ci/y/	Gaseous: Iodine 131 release limit /Ci/y/	Liquid: Mixed fission and corrosion products (without tritium) released /Ci/y/	Liquid: Mixed fission and corrosion products (without tritium) release limit /Ci/y/	Liquid: Tritium released /Ci/y/	Liquid: Tritium release limit /Ci/y/
KWO - Obrigheim PWR (328)	1969	77.9	5,560	80,000	$< 1.8 \cdot 10^{-2}$		$6.3 \cdot 10^{-2}$	15	10.5	18	328	
	1970	83.9	7,700	80,000	$< 1.7 \cdot 10^{-2}$		$4.4 \cdot 10^{-2}$	15	3.24	18	430	
	1971	76.9	1,456	7,600 (new)					4.4	18		
KRB - Gundremmingen BWR (237)	1967		8,800	$1.9 \cdot 10^{6}$ [1]	$9.6 \cdot 10^{-3}$	2 850	$1.75 \cdot 10^{-2}$	22				
	1968		8,800	$1.9 \cdot 10^{6}$ [1]	$4.1 \cdot 10^{-3}$	2 850	$3.2 \cdot 10^{-2}$	22				
	1969	74.1	11,400 (mean : 0.68 mCi/s) (peak : 9 mCi/s)	$1.9 \cdot 10^{6}$ [1]	$7.6 \cdot 10^{-3}$	2 850	$3.6 \cdot 10^{-1}$	22	1.65	14.4	17.8	432
	1970	84.3	7,350	$1.9 \cdot 10^{6}$ [1]	$7.4 \cdot 10^{-2}$	2 850	$2 \cdot 10^{-1}$	22	1.52	14.4	30	432
	1971	90.2	6,650	$1.9 \cdot 10^{6}$ [1]	$5 \cdot 10^{-2}$	2 850	$3.2 \cdot 10^{-1}$	22	1.89	14.4		

[1] Mean : 60 mCi/s
Peak : 600 mCi/s.

TABLE 4 (continuation)

Plant (net output MWe)	Year	Annual load factor (%)	Gaseous: Noble gases released /Ci/y/	Gaseous: Noble gases release limit /Ci/y/	Gaseous: Aerosols released /Ci/y/	Gaseous: Aerosols release limit /Ci/y/	Gaseous: Iodine 131 released /Ci/y/	Gaseous: Iodine 131 release limit /Ci/y/	Liquid: Mixed fission and corrosion products (without tritium) released /Ci/y/	Liquid: Mixed fission and corrosion products (without tritium) release limit /Ci/y/	Liquid: Tritium released /Ci/y/	Liquid: Tritium release limit /Ci/y/
KWL - Lingen (Germany) BWR (174)	1969	91.2	200,000 (peak : 25 mCi/s)	$3.1 \cdot 10^6$ [2]	$2.5 \cdot 10^{-1}$	15 800	—	16	0.64	5.4	26	
	1970	69.3	132,000 (peak : 30 mCi/s)	$3.1 \cdot 10^6$ [2]	$6.7 \cdot 10^{-1}$	15 800	$2.6 \cdot 10^{-1}$	16	0.6	5.4	31.7	
	1971	67	(peak : 40 mCi/s)	$3.1 \cdot 10^6$ [2]		15 800		16	0.3	5.4		
Garigliano - Italy [3] BWR (151.5)	1967		29,200	$3 \cdot 10^6$		—		—	3.4	$5 \cdot 10^3$	8	$5 \cdot 10^5$
	1968		82,000	$3 \cdot 10^6$					4.8	$5 \cdot 10^3$	8	$5 \cdot 10^5$
	1969		140,000	$3 \cdot 10^6$	$6.3 \cdot 10^{-2}$	$3 \cdot 10^3$	negl.	$1 \cdot 10^4$	9	$5 \cdot 10^3$	7	$5 \cdot 10^5$
	1970		275,000	$3 \cdot 10^6$	$6.3 \cdot 10^{-2}$	$3 \cdot 10^3$	$6 \cdot 10^{-2}$	$1 \cdot 10^4$	11.9	$5 \cdot 10^3$	5	$5 \cdot 10^5$
	1971		640,000	$3 \cdot 10^6$	$6.3 \cdot 10^{-2}$	$3 \cdot 10^3$	$1.3 \cdot 10^{-1}$	$1 \cdot 10^4$	19.1	$5 \cdot 10^3$	5	$5 \cdot 10^5$

[2] Mean : 100 mCi/s
Peak : 1 Ci/s.

[3] The release limits are actually being revised.

TABLE 4 (continuation)

Plant (net output MWe)	Year	Annual load factor (%)	Gaseous						Liquid			
			Noble gases		Aerosols		Iodine 131		Mixed fission and corrosion products (without tritium)		Tritium	
			released /Ci/y/	release limit /Ci/y/	released /Ci/y/	release limit /Ci/y/	released /Ci/y/	release limit /Ci/y/	released /Ci/y/	release limit /Ci/y/	released /Ci/y/	release limit /Ci/y/
Trino Vercellese (Italy) PWR (247)	1967		59.1	$5 \cdot 10^4$					8.97	21	600	$5 \cdot 10^3$
	1968	shut down	—	$5 \cdot 10^4$					5.51	21	—	$5 \cdot 10^3$
	1969	shut down	—	$5 \cdot 10^4$					3.09	21	—	$5 \cdot 10^3$
	1970	65.4	19.2	$5 \cdot 10^4$	$<1.2 \cdot 10^{-4}$	0.2	$5.9 \cdot 10^{-4}$	$5 \cdot 10^{-2}$	2.96	21	135	$5 \cdot 10^3$
	1971	68.5	585	$5 \cdot 10^4$	$<1.4 \cdot 10^{-4}$	0.2	$1 \cdot 10^{-3}$	$5 \cdot 10^{-2}$	19.07	21	1 117	$5 \cdot 10^3$
Latina [4] (Italy) GCR (153)	1967		2,500	$5 \cdot 10^5$	negl.	$5 \cdot 10^2$	negl.	$3 \cdot 10^3$	14.2	$1.6 \cdot 10^3$		$2.5 \cdot 10^5$
	1968		2,500	$5 \cdot 10^5$	negl.	$5 \cdot 10^2$	negl.	$3 \cdot 10^3$	72	$1.6 \cdot 10^3$	398	$2.5 \cdot 10^5$
	1969	39.8	1,500	$5 \cdot 10^5$	negl.	$5 \cdot 10^2$	negl.	$3 \cdot 10^3$	29	$1.6 \cdot 10^3$	25.2	$2.5 \cdot 10^5$
	1970	90.1	2,500	$5 \cdot 10^5$	negl.	$5 \cdot 10^2$	negl.	$3 \cdot 10^3$	10.2	$1.6 \cdot 10^3$	16.7	$2.5 \cdot 10^5$
	1971	73	2,470	$5 \cdot 10^5$	negl.	$5 \cdot 10^2$	negl.	$3 \cdot 10^3$	1.5	$1.6 \cdot 10^3$	13	$2.5 \cdot 10^5$
Sena-Chooz (France) PWR (270)	1970	62.5	3	$2.5 \cdot 10^6$	negl.	10^3			6.4	100	339	$7 \cdot 10^6$
	1971	78.8	4,500	$2.5 \cdot 10^6$	negl.	10^3			34.4	100	706	$7 \cdot 10^6$

[4] The release limits are actually being revised.

TABLE 4 (continuation)

Plant (net output MWe)	Year	Annual load factor (%)	Gaseous						Liquid			
			Noble gases		Aerosols		Iodine 131		Mixed fission and corrosion products (without tritium)		Tritium	
			released /Ci/y/	release limit /Ci/y/	released /Ci/y/	release limit /Ci/y/	released /Ci/y/	release limit /Ci/y/	released /Ci/y/	release limit /Ci/y/	released /Ci/y/	release limit /Ci/y/
Chinon EDF1 (70) EDF2 (200) EDF3 (480)	1969	68.8; 89.4; 55.3	12,300	$4 \cdot 10^5$	$<10^{-2}$	10^3			7.44	900		
	1970	79.7; 91.1; 72.7	8,085	$4 \cdot 10^5$	$<10^{-2}$	10^3			2.25	900		
	1971	41.5; 57.1; 79.2	4,225	$4 \cdot 10^5$	$1.8 \cdot 10^{-2}$	10^3			2	900		
St-Laurent-des-Eaux SL1 (480) SL2 (515)	1969	SL1 : 52.1	1,900	$4 \cdot 10^5$	<1	10^3			2.71	800		
	1970	SL1 : 8.6	305	$4 \cdot 10^5$	$<10^{-2}$	10^3			0.77	800		
	1971	SL1 : 73.9 SL2 : 77.2	3,425	$4 \cdot 10^5$	$4.7 \cdot 10^{-2}$	10^3			2.25	800		

The ways of exposure to the activity in liquid effluents depend on the utilization of the receiving river system for :

— drinking water;

— fishery;

— watering cattle;

— irrigation;

— recreation (external irradiation).

From the wide sources of information used here, the (calculated) exposure values due to radioactive effluent releases from nuclear power plants can be outlined as follows :

Dose rate (mrem/year)

	at site boundary	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 by consumption of milk contaminated with I-131, an evaluation being too imprecise to have any practical meaning, it can be noted that iodine-131 has never been detected in milk produced around any nuclear power plant (detection limit about 10 pCi/liter, corresponding to a thyroid dose of a small child of about 40 mrem/year).

Dose rates due to the activity released with liquid effluents are conservatively 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 sources of radiation

From section II.A.2. above can be seen that the only normal radioactive effluents releases from nuclear power plants which are worthwhile to assess, versus the dose limits for the public, are the gaseous effluents and more specifically the noble gases. *At present* radioactive effluents from fuel processing plants play a minor role in the radiation exposure of the population.

To allow a comparison of human exposure to radiation from radioactive effluents released by nuclear plants with exposure from other sources of ionizing radiation, table V gives a survey of dose-rates to man from various radiation sources as well as the whole body dose limits for the different groups of the population.

These data lead to the conclusion that the doses to critical population groups in the vicinity of nuclear power stations do not exceed 1/100th of the dose limits for such populations as fixed by the Euratom standards and correspondingly no more than about 1/20th of the radiation dose level from the natural background.

TABLE 5

Comparative table of human exposure to radiation
(Ref. 9, 25, 26, 30, 58 and others)

1. *Mean annual dose-rates due to natural background*		
In 'normal' areas	gonads	93 mrad/year
	bone-lining cells	92 mrad/year
	bone-marrow	89 mrad/year
Typical example of distribution :		
— cosmic radiation (at sea level)		28 mrad/year
— terrestrial radiation (including air)		44 mrad/year
— internal irradiation	gonads	21 mrad/year
	bone-lining cells	20 mrad/year
	bone marrow	17 mrad/year
N.B. : This dose-rate varies with the geology and with altitude.		
a) *geology :* igneous rocks		up to 5 000 mrad/year
b) *altitude :*		
at 3 000 m altitude		ca. 90 mrad/year
transatlantic flight at 10 000 m		3-5 mrad/flight
2. *Diagnostic X-ray dose 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 the 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 fixed by the 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 mrem in 30 years ([3])
(genetically significant dose)		(170 mrem/year)

[1] Values for Europe : in the United States presently about 100 mrem/year.

[2] For BWR's two *calculated* highs, respectively of 170 and 31 mrem, have been reported at the site boundary (ref. 48).

[3] Other more severe apportionments are recommended or applied :

ICRP	2 rem in 30 years = 6 mrem/year
FRG, USSR	2 rem in 30 years
Sweden	1 rem in 30 years
UK	1 rad in 30 years.

4. Application of 'practical' discharge standards

Until recently most discharge limits for radioactive effluents from nuclear power plants were so fixed 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 the *practicable minimum*, the authorities on radiation protection in several Member States and in other countries have recently recommended or set much more restrictive limits (Ref. 20, 30 and others) on discharges from nuclear power stations than those formerly accepted. These are, then, 'practical' limits, lower again than the 'radiologically' acceptable limits. In certain cases these more restrictive dose values may be exceeded (theoretically up to the ceiling of the radiological values), provided that suitable justification is given and accepted by the licensing authorities.

A few significant practical values can be quoted by way of illustration :

— *gaseous effluents :* (rare gases essentially)	10 [1] - 30 mrem/year
— *liquid effluents :*	30 mrem/year 5 Ci/year (excluding tritium)
Concentration in cooling water before discharge (less tritium)	20 pCi/l
Concentration of tritium in cooling water before discharge	5 000 pCi/l

Such 'as low as practicable' or 'design objective' standards (corresponding to actual experience) serve in particular the purpose of long-term caution and provisions. Some of these standards are still subject to controversy, in particular the validity of the integrated population dose (man-rem) concept advocated by some especially for long-term provisions (e.g. Ref. 49) and in fact already applied in recent forecasting studies (e.g. Ref. 50).

5. Medium- and long-term forecast for the long-lived nuclides Kr^{85} (10.4 years) and H^3 (T = 12.4 years) from the nuclear power plants

5.1. *Influence of supplementary retention equipment*

First of all, it should be pointed out that such forecasts have to be handled with care, amongst others, because of a probable increased use of supplementary retention equipment or procedures.

Several methods 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 special aim of temporarily or permanently storing the long-lived Kr^{85} (10.4 years) and the short-lived Xe^{133} (5.3 days). They include systems such as simple hold-up for Xe^{133} decay, adsorption on active carbon (cooled at near ambiant temperature); supplementary retention factor varying from 40 to 2 000 for BWR), cryogenics (very low temperature concentration followed by solid adsorption), or use of fluorocarbonated solvents at low temperature.

[1] i value proposed in the United States, but may be exceeded if suitably justified;
ii valid for a residential area, and not necessarily at the boundary of the plant site (Ref. 41);
iii recommended for total body irradiation, gonads and bone marrow (15 mrem for other organs) (Ref. 41);
iv *in practice* this would lead to a whole body dose of 5 mrem/year because of shielding from buildings, and limited periods of occupancy near site boundary (Ref. 47).

These rad-waste or 'mini-release' systems can be applied for use in fuel reprocessing plants, or they can be adapted for nuclear power plants with variants depending on the volume of the effluents to be processed (larger volumes for BWR for instance). For nuclear power plants the first aim is to facilitate construction in more densely populated or urban areas.

As an indicative example a large twin 1 600 MWe (total) BWR station with usual short time hold-up equipment, could give rise to several tens of mrem/year individual average total body dose due to noble gases at the plant boundary, This would be reduced by a factor 10 by a recombiner and charcoal delay system. At this occasion may be mentioned that the same would apply for the thyroid dose due to iodine release : several tens of mrems (inhalation and injection of milk) would be reduced by the rad-waste system to a few mrem.

The additional cost of such equipment is estimated to be about 1-3 million u.a. (units of account).

For liquid effluents in general improvements of the presently applied decontamination systems can be applied also, at a cost of about 0.5-1 million u.a.

For tritium there is no practical way to keep it from being released to the environment. Total coolant recycling could be applied but build-up of tritium would give rise to a displacement of the problem (e.g. containment contamination and purging) with most likely a higher hazard to the professionally exposed.

In general, one can say for all the supplementary retention equipment mentioned here that one has to weigh the advantage of reduced releases to the environment versus potential higher hazards 'in-plant' or during subsequent transportation and storage of the accumulated wastes.

5.2. *Future estimated releases of Kr^{85} and H^3 from nuclear power plants*

With the restrictions pointed out in the preceding section 5.1., a rough estimate can easily be made *up to 1985* for the sake of comparison with the expected curie-release values from reprocessing plants (section II.B.). The estimates summarized in table VI below are based on :

(a) the nuclear power forecast for the Community of Six (Chapter I, Table 1).

(b) typical BWR and PWR releases quoted under section II.A.2. [1].

(c) and assuming half the units of the PWR-type and the other half of the BWR-type.

It should moreover be reminded that the releases for noble gases (see II.A.2.) include *both* Xe^{133} (T = 5.3 days) and Kr^{85} (T = 10.4 years) and their short-lived isotopes.

Depending on the hold-up time either the Xe^{133} and other short-lived Xe-isotopes or the Kr^{85} and other short-lived Kr-isotopes activity contribution may be determining *at the time of release.*

For long term evaluations however the short-lived isotopes can be discarded and therefore Xe^{133} and Kr^{85} become determining. With these assumptions, one can reasonably assume for a PWR with a 40-50-day hold-up of the hydrogenated effluents (besides the permanently released venting air), Kr^{85} would constitute about 1/3 to 1/5 (e.g. Ref. 9, Chapter III) [2] of the total noble gas activity released. For a BWR, where most of the active effluents discharged stem from the gas-ejector at the condenser and from turbine leaks with generally a hold-up of not more than 0.5 hours, Kr^{85} constitutes about a fraction of $5 \cdot 10^{-3}$ of the total activity (e.g. Ref. 22 and 28) and Xe^{133} the rest (also Ref. 6, Chapter III).

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[1]

BWR	10^3	Ci/MWe-year	noble gases
PWR	10	Ci/MWe-year	noble gases
BWR	5.10^{-2}	Ci/MWe-year	tritium
PWR	20	Ci-MWe-year	tritium

[2] For calculation purposes the value of 1/3 has been retained (Table VI), i.e. the most conservative from the point of view of environmental effects.

TABLE 6

Year	Power	PWR-power	BWR-power	Kr^{85}-PWR	Kr^{85}-BWR	Kr^{85}-total	H^3-PWR	H^3-BWR	H^3-total
	(thousands of MWe)			(Ci/year)			(Ci/year)		
1972	5.5	2.7	2.7	10^4	$1.35 \cdot 10^4$	$2.35 \cdot 10^4$	$5.4 \cdot 10^4$	135	$5.4 \cdot 10^4$
1975	12	6	6	$2 \cdot 10^4$	$3 \cdot 10^4$	$5 \cdot 10^4$	$1.2 \cdot 10^5$	300	$1.2 \cdot 10^5$
1980	45	22.5	22.5	$7.4 \cdot 10^4$	$1.1 \cdot 10^5$	$1.85 \cdot 10^5$	$4.5 \cdot 10^5$	$1.13 \cdot 10^3$	$4.5 \cdot 10^5$
1985	100	50	50	$1.65 \cdot 10^5$	$2.5 \cdot 10^5$	$4.1 \cdot 10^5$	10^6	$2.5 \cdot 10^3$	10^6

There seems to be little point in making further detailed forecasts, as from 1985 onwards other types of reactors may contribute in the nuclear power production [1] and hold-up, and complementary retention equipment may be more developed and applied in all types of reactors than is presently the case.

Just in order to obtain a further indication — on a 'conservative basis' — of the power plant versus the reprocessing plant problem, using the same (LWR) assumptions, in the year 2000 could be expected $2.55 \cdot 10^6$ Ci/year of Kr^{85} and about $6.2 \cdot 10^6$ Ci/year of H^3.

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

B - THE RADIOACTIVE GASEOUS AND LIQUID EFFLUENTS PRODUCED IN 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 this material produced in the fuel elements are retained within them until they are reprocessed.

Reprocessing is the operation in the fuel cycle in which the largest quantity of radioactivity is dealt with. This operation is therefore also the source of most of the radioactive waste of different kinds.

1. Radioactive waste resulting from the reprocessing of irradiated fuel

Classes and types of waste

Reprocessing consists in separating the fissile materials from the products contained within the irradiated fuel and in purifying the recovered materials.

[1] As will be seen also in section II.C. e.g. LMFBR's can be operated with less radioactive gaseous effluents than LWR.

At present, the main process used on the industrial scale is aqueous reprocessing. All the operations involved in reprocessing result in the production of different types of radioactive waste. The first class includes the pieces of fuel cladding forming the solid radioactive waste produced by decanning operations. The gaseous fission products contained within the fuel elements, such as Kr^{85}, Xe^{133}, I^{131} and H^3, are liberated during sectioning and dissolving; they constitute the gaseous radioactive waste class.

Most of the radioactivity is contained in the non-volatile fission products present together with the fissile material in the dissolver solution. After the fissile materials have been extracted, this solution contains the fission products and forms the class of highly active waste.

The entrailment of certain fission products require the final purification of the fissile material and the cleaning of the solvent before it is reused. These operations produce solutions of radioactive waste in sometimes very large quantities, although its specific radioactivity is very much lower than that of high-activity solutions. It is sometimes necessary, for plant maintenance purposes, to decontaminate parts of the installation or certain items of equipment. This decontamination process produces also considerable quantities of medium and low-activity radioactive waste.

2. Discharge of radioactive effluents into the environment

Since a reprocessing plant is the point at which a large part of the radioactive waste produced during the irradiation of fuels in nuclear power stations is accumulated, it is also the main industrial 'unit' point of discharge in the whole of the fuel cycle operations (fuel fabrication, power plants, enrichment, reprocessing).

Two types of waste are discharged into the environment by reprocessing plants : gaseous and liquid waste. The maximum discharge rates for both these effluents are also based on 'radiological' protection standards such as those of European Atomic Energy Community and the recommendations of the ICRP (International Commission for Radiation Protection). Starting from the principle of the limitation of exposure doses to the levels considered permissible (basic standards) and an analysis of the critical pathways, maximum permissible concentrations (derived standards) in water and in the atmosphere have also been laid down for each radioactive nuclide. The discharge limits for a reprocessing installation concerned are prescribed on the basis of these basic and derived standards, bearing in mind a minimum dilution factor for the discharge point in question.

2.1. *Gaseous waste*

In the present state of the art, the gaseous waste from reprocessing installations is mainly made up of the noble gases Kr^{85}, Xe^{133} and part of the H^3 contained in the fuel. Fuel cooled for 150 days contains only a small quantity of Xe^{133} [1] as a result of the decay of this isotope. I^{131} ($T = 8.1$ days) is largely eliminated by decay during the cooling of the fuel and by specific 'off-gas' treatments to retain the iodine.

The gaseous discharge is thus essentially limited to the following radioactive products : all the Kr^{85} produced during irradiation, the (small) fractions of Xe^{133} still present at the time of reprocessing, the quantity of H^3 released in gas form [2], a very small proportion of I^{131} which escapes the retention treatment, and a very small fraction of the aerosols or solid particles.

[1] May be different for Pu-fuel (see section II.C).

[2] The largest fraction of H^3 is released in the liquid effluents (about 90 %).

2.2. *Liquid waste*

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

The decontamination treatments used retain almost all the fission and activation products with the exception of tritium, which is mainly found in the treated liquids in the form of tritiated water and is thus finally discharged into the environment, either into a river or into the sea. The radioactivity level of the liquids discharged by reprocessing plants is limited in a similar way to that used for gaseous radioactive waste. The limits on low-activity waste depend on the dilution achieved at the point of discharge and the specific use of the body of water into which the liquid is discharged.

3. Forecasts for the discharge of radioactive effluents

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

The quantities of nuclides for 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.

TABLE 7 [1]

Quantities of Kr^{85} and H^3 produced in relation to the quantities of fuel to be reprocessed (from LWR)

Year	Fuel to be reprocessed tons/year	Kr^{85} Ci/year	H^3 Ci/year
1975	110	$1 \cdot 10^6$	$8 \cdot 10^4$
1980	720	$7 \cdot 10^6$	$5 \cdot 10^5$
1982	1 200	$1 \cdot 10^7$	$8 \cdot 10^5$
1985	1 940	$1.7 \cdot 10^7$	$1.4 \cdot 10^6$
(2000	9 000	$8 \cdot 10^7$	$6 \cdot 10^6$) [2]

1 Chapter 1, Ref. (1).
2 Quantity influenced largely by proportion of LWR uranium fuel and Pu-fuel (Recycle and FBR).

It may 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 concentrations of Kr^{85} and H^3 in the atmosphere are of the same order of magnitude; it may thus be concluded that the discharge of about 10% of H^3 in gaseous form will not present any problems for relatively small capacity reprocessing plants (1 t/day).

C - LONG-RANGE EXPOSURE EFFECTS ON MANKIND - MEDIUM AND LONG-RANGE FORECASTS FOR NUCLEAR POWER PLANTS AND REPROCESSING PLANT GROWTH

1. Conservative nature of the estimates

In the long range effects estimates for the potential exposure from nuclear power plant development to the population (general public), a conservative approach is mostly taken by assuming up to the year 1990 or 2000 no significant contribution from advanced type reactors (especially LMFBR). This is justified by the fact that these reactors can operate with lower routine effluents than 'current' LWR's and by the uncertainties in the future extent of their power generation application (Ref. 50).

On the other hand for the long-range effects of fuel reprocessing operation Plutonium thermal fuel (Plutonium recycle) and Plutonium Fast Breeder Reactor (FBR) fuel may become significant around 1990-2000. The decay time before processing operation starts may be short for FBR-fuel (about 30 days [1] (because of economies in the Pu-cycle and this leads to a significantly higher amount of radioactivity at the time of reprocessing (especially for the meaningful relatively short lived isotopes Xe^{133}, I^{131}). This would in turn lead to about 3 times higher doses for the whole body (noble gases, due to Xe^{133}) and for the Thyroid (I^{131}) in comparison with LWR-fuel processing presuming present day technology now applied. In other words FBR development is — from the standpoint of routine environmental implications—advantageous at the power production stage and disadvantageous at the fuel-processing stage, i.e. as compared to LWR's.

As has been pointed out in a recent detailed national and world wide estimate study of ionizing radiation up to the year 2000 (Ref. 50), the estimates do generally not include further development of hold-up and retention techniques.

Therefore the results one arrives at are further enhanced in their conservative nature.

2. Results

2.1. *Kr^{85} and H^3*

From the tables 6 and 7 in the previous sections can be deduced that as far as the 'global' releases of the Kr^{85} to the atmosphere are concerned the reprocessing operations are determining, and this by a factor 100 over nuclear power plants whilst the Tritium releases are practically equal for both types of nuclear installations. The same would apply on a world-wide basis (Ref. 37, 38, 50).

2.2. *Global effects of Kr^{85}*

On the basis of the world-wide energy forecasts, and assuming that Kr^{85} is continuously released and accumulated in the atmosphere without any complementary retention equipment whatsoever, its permanent average annual contribution to the individual total body dose of the public could in the year 2000 at most be about $5 \cdot 10^{-2}$ mrem [2] (corresponding to an accumulated $3\,150 \cdot 10^6$ Ci [3]), about 1.15 mrem [2] by 2050 (Ref. 27, 38, 52). As already mentioned earlier about 99 % of this would result from the reprocessing operations. With respect to the 'global' effects, the increase of Kr^{85} in the environment from the use of nuclear energy in the world

[1] As compared to 150 days for LWR-fuel.

[2] Whole body dose and gonads; for 0.15 mrem this corresponds to about 7 mrem 'body surface' or 'skin' dose.

[3] $1000 \,.\, 10^6$ Ci in 1990.

would up to the end of the 21st century not give rise to a significant radiation hazard (as compared to natural background differences from region to region and with height, for instance). Some regional global accumulation of Kr^{85} might however occur (Ref. 43).

It is most likely however that nevertheless by that time retention equipment will be installed but mainly for 'local' requirements as outlined below.

2.3. *Global effects of Tritium*

The cosmic [1] Tritium equilibrium is of the order of $100 \cdot 10^6$ curies (presently the natural background of Tritium is about a factor 10 higher because of residual Tritium from Weapons tests fall-out). By the turn of the century, the Tritium accumulated by the nuclear power generation in the world would also reach 100 million curies, which is about 6% of amount which was present in 1963 (due to weapon tests). After the year 2000, Tritium production from nuclear power would become the main source of Tritium in bio- and troposphere unless retention means were developed. Presently the estimated individual dose to a member of the public is about 0.04 to 0.06 mrem/year, in the high range due mainly to residual Tritium from weapon tests. By the year 2000 this average dose would only be about 0.02 to 0.03 mrem/year (Ref. 28, 50). It can be considered that on a 'global' basis Tritium presents even less of a hazard than Kr^{85}.

2.4. *Local effects of Kr^{85}-release (and Xe^{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). Besides the trend in operating reprocessing plants of larger capacity, also a gradual increase in the burn-up of the fuel to be processed increases the needs of Kr^{85} releases. Furthermore for Pu-fuel processing the Xe^{133} contribution may become determining (see section II. C. 1). A reprocessing plant of 5-10 ton/day capacity would give rise to a whole body noble gas dose of the order of *hundreds of mrem* in the near vicinity of the plant and of *several or tens* of mrem at 3 000 m (based on Ref. 50) distance (depending on the type of fuel treated; with Xe^{133} determining for FBR-fuel).

For exemple (Ref. 39), if all the fuel expected to be available for reprocessing in West Germany during the year 1990 were reprocessed in a single plant, an annual individual total body dose in the vicinity of over 100 mrem could be attained (i.e. still below a 'radiological' limit value of 500 mrem but in excess of the recommended 'practical values' of e.g. 30 mrem/year for nuclear power plants in the Federal Republic of Germany). This would imply that if the stringent, 'practical standards' for nuclear power plants (for gaseous effluents) were applied also to reprocessing plants assuming that no complementary retention equipment were used, about one reprocessing plant would have to be available for every ten nuclear power stations (Ref. 40).

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, essentially because of the 'total' noble gas releases (the relatively short-lived Xe^{133} and Kr^{85}, with the main contribution from Xe^{133} however in this case). For BWR such problems could be more severe than for reprocessing plants, if no supplementary hold-up or retention equipment were provided, as demonstrated easily by comparing for instance Tables 4 and 7 (e.g. ten current BWR's are about equivalent to one 5 ton/day reprocessing plant from the point of view of noble gas release). For instance a twin BWR plant of the order of 1 600 MWe may give rise to several tens [2] of mrem/year average individual total body dose due to noble gases. However with improved rad-waste systems this can be reduced to a few mrem/year (see also section II. A. 5. 1).

[1] $4 \cdot 10^6$ to $8 \cdot 10^6$ annual production (Ref. 36).

[2] But less if spread out over cumulative population around the plant (derived from integrated man-rem).

These long-term considerations will probably also determine the development and application of supplementary retention equipment for rare gases in nuclear power and reprocessing plants (Ref. 51). This does however lead to the problem of transporting and disposing of the accumulated bottled Kr^{85} (e.g. in 50 l. cylinders).

2.5. *Local effects of Tritium release*

In the long term, the main aspect of the Tritium hazard is also 'local' receptivity for both high capacity reprocessing plants (5-10 ton/day) and to a lesser extent multi-unit siting of nuclear power plants (especially of the PWR-type). For example a 1 000 tons/year reprocessing plant (200 days load factor at 5 ton/day) would discharge about $7 \cdot 5 \cdot 10^5$ Ci of Tritium in the liquid effluents, whilst ten 1 000 MWe PWR units would discharge about $2 \cdot 10^5$ Ci of Tritium. From section II.A.2. can be deduced that exposure rates could become significant [1] in the near vicinity of such installations, if no large dilution capacity were available such as fast-flowing rivers or the sea. It may be however that in the future a practical technology be developed for removing Tritium from liquid effluents (see also section II.A.2.). Moreover it may be that the fraction of Tritium released with the gaseous effluents (10% to 20%) becomes determining in a reprocessing plant. Indeed estimates indicate (Ref. 50) that for a plant, treating about 5-10 ton/day, the whole body dose (mainly due to inhalation from plume passage) would still be of the order of tens of mrem/year [2] at 3 000 m distance from the plant.

It seems therefore that on a long-term basis due to 'local' considerations only, reprocessing plants will become determining in the Tritium hazard.

3. Overall exposure contribution of nuclear power generation development

If one looks at the present and future potential contribution of nuclear energy development in the total radiation burden to mankind a clear illustration is given of its 'healthy' situation in routine operations.

Various estimates indicate for instance for a highly developed country (e.g. USA) the trend in exposure for an average individual of the population at large. They are summarized in Table 8.

It can be seen that at present the whole field of nuclear energy (including power production, fuel cycle industry, research) would contribute for less (see also Table 5) than 1 mrem/person mean annual whole body exposure (which is genetically significant), a value which is in Europe probably also not reached (Ref. 8, 9). Only 1/10 of this value from exposure to the general public and at the turn of the century — assuming conservatively no retention equipment improvements by that time — the average individual fraction given to the general public may grow to about the same order of magnitude of that by the occupational contribution, due to population and nuclear industry growth. The total average individual dose would then be somewhat larger than 1 mrem/person-year. These figures can be compared with the 'most conservative' radiological allowances made for the genetically significant doses, e.g. 1 mrem/30 year equals 34 mrem/year (Ref. 49) [3].

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1 By significant is meant : overriding the 1.5 mrem/year apportionment for Tritium *of which 0.5 mrem/year via drinking water* and 1 mrem/year through inhalation and consumer goods (i.e. for the public).

2 According to Ref. (50), about 5 mrem/year for a 300 t/year plant at 3,000 m distance; according to Ref. (51) a ten times smaller dose is found (about 3.5 mrem/year for a 1 600 t/year plant, no distance specified). Anyhow the apportionment for Tritium to be inhaled (1 mrem/year) would be overridden in both cases.

3 Proposed apportionment of 0.5 rem/30 years to nuclear power generation plants and 0.5 rem/year to transportation, reprocessing and waste disposal.

TABLE 8

Estimated average Whole-body Radiation (mrem/person-year) from various sources

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

[1] After a peak of about 12 mrem in 1963.
[2] *Miscellaneous :* Television, air transport, consumer goods.
[3] With the main contribution so far from practice of medicine and dentistry (Ref. 50).

D - INDICATIVE COMPARISON BETWEEN IMPORTANCE OF NUCLEAR EFFLUENTS AND EFFLUENTS FROM CONVENTIONAL INDUSTRY AND CONSUMPTION GOODS

1. General

A detailed comparison between nuclear and conventional activities has not been performed here. Only some indicative examples are given in order to obtain — further to the more detailed data on nuclear activities dealt with in other sections of the present report — a crude idea of the relative hazards to mankind.

Emphasis is placed here on 'gaseous' effluents comparison, because it was seen earlier that on a short, medium and long-term basis these effects are—for the nuclear industry (from the standpoint of *radioactivity*)—overriding the importance of liquid (radioactive) effluents discharge. In later sections a similar indicative comparison will be made on waste *'storage'* and *accumulation* and on *thermal effects* which, for the nuclear industry is more of a routine operation problem than the liquid 'radioactive' effluents discharges to the environment.

2. Comparative examples

In the case of gaseous effluents from nuclear power plants, the contaminants discharged give the following comparative figures in relation to waste from conventional power stations (Ref. 11).

The fact that, in the nuclear case the volume of air required to dilute the quantity of effluents released in order to respect the permissible (radiation) standards is much smaller than the volume needed for conventional coal and oil-fired stations (Refs. 27 and 28) in order to respect the standards for conventional pollutants, also gives an indication of the cleanliness of nuclear power stations in comparison with other power sources. Other data from industrialized countries give an idea of the present and likely long-term situations as regards pollution.

TABLE 9
(partly taken from Ref. 11)

For a 1 000 MWe power station

Pollutant	Annual waste discharge (in millions of pounds)			
	coal [2]	fuel oil [3]	gas [3]	nuclear
Sulfurous 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-226/1 620 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 in U.S. fuels.
[2] Only fly-ash control.
[3] Without pollution control equipment.

In the USA, all the atmospheric pollutants from all conventional sources amount at present to about 125 million tons per year, 12.5% of which are due to electric power (mostly oxides of sulphur). They are alleged to cause 20 000 deaths a year (Ref. 34); this means that the individual's risk of death from this cause are about 10^{-4} a year (cfr. Chapter VI), which is of the same order as the risk 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 tons of oxides of sulphur and some 450×10^6 tons 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} tons of CO_2 are at present discharged annually, which is to be compared with the 10^{12} tons of natural origin (Ref. 45).

There seem to be two contradictory theories on the effects of the accumulation of CO_2 (see also Chapter IV) in the long term. One theory states that, by the year 2000, the exchange of heat between the earth and the atmosphere will have dropped, causing the temperature of our planet to increase by 2-4 ^0C. According to the other theory, the solar radiation will at the same time be absorbed by the CO_2 enriched atmosphere to such an extent that the temperature of the planet will drop.

Such examples of estimated effects may be compared with nuclear long-term effects, for example, calculated for the possible accumulation of Kr^{85} and Tritium (see Section II.C.2.1. and 2), the foreseeable consequences of which would, in the year 2000, be far from truly worrying on a universal (or global) basis, even assuming that no special retention measures were taken.

CHAPTER III

Radioactive waste storage and accumulation - ore processing wastes

A - ORIGIN OF INDUSTRIAL RADIOACTIVE WASTES

The main wastes to be considered stem from nuclear power production plants.

1. Power production plants

Because of the utmost precautions taken in order to maintain exposure to the professionally exposed within the fixed standards and to the population as low as practicable, purification and decontamination of the various effluent streams is applied (ion-exchange, filtres, etc.). This leads to medium-level wastes under solid form which are temporarily stored at the plant sites. Moreover special types of wastes originate at those plants, 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 wastes (from the activity standpoint, not necessarily from the volume standpoint) stem from the reprocessing operations.

They result

(a) from the fuel decladding type : compacted cladding is stored at the reprocessing plant site;

(b) from the extraction and purification operations. These wastes are treated in order to concentrate and store them.

(i) The highly active liquid wastes (activity level $> 10^4$ Ci/m^3) are concentrated so as to occupy the minimum volume compatible with their temporary storage in liquid form, pending ultimate treatment and storage in solid form. Temporary storage in liquid form for three to five years will always be economically justified by the decrease in fission products with a medium half-life. This decrease consists in a reduction in the heat released which, in turn, allows the volume to be more reduced, on final treatment. The processes for ultimate disposal under development are e.g. various calcination methods and vitrification.

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

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

B - STORAGE, ACCUMULATION AND ULTIMATE DISPOSAL OF RADIOACTIVE WASTES FROM FUEL REPROCESSING AND NUCLEAR POWER PLANT OPERATION - MEDIUM AND LONG TERM FORECASTS

1. Reprocessing wastes

The radioactive waste resulting from reprocessing of fuel is temporarily stored at the reprocessing plant in liquid form pending its ultimate treatment and disposal in places where dispersion into the environment can be excluded. The presence amongst the fission products of long-lived nuclides and the presence of long-lived alpha emitters imply that the manner in which these products are stored and finally disposed of should ensure that dispersion in the environment is prevented for extremely long periods.

More than 95% of the volume of the wastes to be stored are in the form of high and medium-activity liquids. From the safety point of view, 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 (several decades) of years' use. The importance of the investment required and the need for spares to cope with possible accidental leakage makes this type of storage only a temporary solution.

Investigations are therefore being conducted into various ways of dispositing and disposing of these wastes for long periods in solid form, which combines the advantages of smaller volume and a limited risk of dispersion. The earlier mentioned methods of solidifying solutions of high and medium-activity waste are currently being studied on a pilot scale. The choice of the solidified product and how it is to be enclosed depend 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 radioactive products will not be dispersed into the biosphere. The investigation and analysis of potential sites meeting this requirement must be encouraged at Community level so that all the countries of the Community will have the necessary facilities ready during the next decades.

It may be pointed out (Ref. 4) that on a long term basis not only the high-level fission products constitute in wastes a problem, but also the foreseeable growing accumulation of transuranium elements, such as Plutonium, with long half lives and high toxicity. It has e.g. been estimated that by the turn of the century between 400 to 600 t Pu per year will be produced. Assuming the presently allowed loss of 1.5% in the waste solutions this would lead to a yearly accumulation of 6 to 8 tons of Pu.

One of the most promising methods at present under consideration for finally storing waste (especially high-level waste) consists in depositing it in salt deposits, which are attractive because of their lack of contact with water-bearing strata and their good heat conductivity.

It has been estimated that in the world by the year 2000 between 400 and 600 $\times 10^9$ curie of high level fission product wastes will have accumulated, of which about 1/10 will be due to Sr 90 and Cs 137.

With existing methods under development, it can be forecast that, in the long term, about 77 sq.km of salt deposits would be necessary throughout the world every year for the storage of these wastes, the world's reserves of salt deposits being of the order of tens of millions of sq.km

(Ref. 1). Therefore the serious problem of these industrial wastes does not seem to be prohibitive.

Table 10 below gives an estimate of the annual production of different classes of wastes, related to the quantities of fuel to be reprocessed, on the basis of the Target nuclear programme for the years 1975, 1980, 1985 (and tentatively 2000) for the Community of Six. For the Community of 9 countries these figures should — on the basis of the forecasts outlined in Chapter I be multiplied by a factor 1.5.

TABLE 10 [1]

Wastes produced by fuel processing (from LWR's)

Year	Fuel to be reprocessed	Decladding waste	Unprocessed wastes		
			High level wastes (fission products)		Medium level (concentrates)
	t/year	m^3/year [3]	m^3/year	Ci [2]	m^3/year
1975	110	11	110	$0.7 \cdot 10^9$	204
1980	717	72	720	$4.4 \cdot 10^9$	1 080
1985	1 940	195	1 940	$12.0 \cdot 10^9$	2 915
(2000	9 000	900	9 000	$56.0 \cdot 10^9$	16 000) [2]
	Tons	m^3	m^3	Ci	m^3
Accumulated quantities 1975-1985	9 300	930	9 300	$57.0 \cdot 10^9$	14 000

1 Cfr. assumptions Table VII, Section II.B.

2 The activity is expressed at the time of reprocessing for fuel cooled for 150 days; as already mentioned earlier (e.g. section II.B) Pu-fuel may be subjected to smaller cooling times and this may become important from 1985 onwards.

3 Assuming mechanical or chop or leach decladding.

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 both volume and activity. The decladding wastes are of fairly small volume in comparison with all the total waste volume. Moreover, because it consists of metal, its storage presents fewer problems.

2. Nuclear power plants wastes

2.1. The solid or semi-solid low- or medium-activity wastes resulting from the operation of these plants vary with the reactor type. Below some summarizing data are furnished for LWR's, gas-cooled reactors and LMFBR's.

(1) *Light-water reactors*

For LWR's the volumes of solid or semi-solid waste, *after dewatering but 'before' treatment and packaging*, have been reported (Ref. 5) to be as follows :

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

It should be noted that the specific activity data are representative of plants of different sizes with varying process sytems, hence they are not additive for a particular reactor (also Ref. 5).

After on-site decay-storage, the BWR resins would generally be packaged in drums (about 55 gallon-drums in the US) and in ether types of packages (usually large casks for the low-level miscellaneous dry wastes.

As an exemple for a typical two-unit BWR-station (total about 1 600 BWe) (Ref. 6) 840 drums/year of demineraliser resins fixed in concrete, and 280 m^3 of other packaged miscellaneous low-level waste have been estimated. This leads to a total yearly volume of 455 m^3 to be transported to a central despository (or burial ground). This would correspond to about 45 truck-loads (Ref. 6).

For PWR reactors, after on-site storage and decay, the resins and evaporation concentrates would generally be packed in drums (30 or 55 gallon-drums in the US).

As an example for a typical two-unit PWR station (total about 1 600 MWe) (Ref. 9), about 300-600 drums are expected. This corresponds to a yearly maximum volume of about ***120m³*** to be transported to a central despository (burial ground). This amounts to about 12 shipments.

(2) *Gas-cooled reactors and Fast breeders*

These volumes may be compared with those expected for advanced types of gas-cooled reactors. For example for the Fort St. Vrainplant (330 MWe) a yearly solid waste volume production of only 11 m^3 is expected (approximately 1 shipment) (Ref. 7). The same order of magnitude of volume can be noted at the presently operating G.C.R.'s (eg. Ref. 5, Chapter II).

For the fast breeder reactors of the liquid metal type (e.g. a 300-500 MWe plant, Ref. 8) between 14 to 28 m^3 of 'packaged' (drums) solid wastes per year are expected. This amounts to about 2 shipments.

2.2. *Relative importance of medium-level reprocessing wastes and power reactor wastes — accumulation of the wastes:*

From the information under section 2.1 above can be deduced the following approximate volumes for medium- and low-level packaged wastes from LWR's :

$$\text{for a BWR} : \frac{455 \text{ m}^3/\text{year}}{1\ 600 \text{ MWe}} = 0.28 \text{ m}^3/\text{MWe-year}$$

$$\text{for a PWR} : \frac{120 \text{ m}^3/\text{year}}{1\ 600 \text{ MWe}} = 0.07 \text{ m}^3/\text{MWe-year}$$

For the nuclear power forecasts outlined in Chapter 1 yearly produced quantities and accumulated quantities can be deduced. It will suffice here to indicate that for instance in 1985 the yearly production would be (for 100 000 MWe installed; assuming half by BWR's and half by PWR's) :

— BWR : $50 \cdot 10^3$ MWe $\times$ 0.28 m^3/MWe = $14 \cdot 10^3$ m^3

— PWR : $50 \cdot 10^3$ MWe $\times$ 0.07 m^3/MWe = $3.5 \cdot 10^3$ m^3

total $17.5 \cdot 10^3$ m^3

By rough comparison with the medium-level waste volumes produced by corresponding fuel reprocessing operations, the total accumulated wastes for the decade 1975-1985 can be quickly estimated. A 5-10 t/day reprocessing plant (or about 1 500 t/year) would produce about 2,100 m^3/year of medium-level concentrates (*unprocessed* and *unpackaged*) (see Table 10, 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 wastes (processed and packaged). Grossly the nuclear power production operations (under these hypotheses) lead thus to about twice the 'processed' waste volume originated by the reprocessing operations.

The quantity of low- and medium-level wastes from both nuclear power plants and reprocessing operations *accumulated* between 1975 and 1985 in the Community of 9 would therefore amount to the maximum of about $1.25 \cdot 10^5$ m^3 [1], or about one-tenth of a hectometer. Assuming most of these wastes would be packed in drums, this represents about 600 000 drums ([2]).

It is considered that per drum a depository 'surface' area of 0.57 m^2 or $5.7 \cdot 10^{-7}$ km^2 is necessary. [3]

Hence the total area needed for this accumulated waste of 600 000 drums, would be about 0.34 km^2, i.e. about one third of km^2.

In conclusion, the problem of the cumulation, packaging, transport and ultimate disposal of these wastes is not negligible, but it is not more serious than the equivalent problems of conventional industrial wastes of a hazardous nature (e.g. cyanides).

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[1] 14 000 m^3 (table 10, column 5) $\times$ 2 (packaging factor) $\times$ (2 + 1) $\times$ 1.5 = $1.25 \cdot 10^5$ m^3.

[2] 55 gallon, or 0.21 m^3 drums.

[3] i.e. about 40 m^2 for 70 drums.

C - MISCELLANEOUS : LONG-TERM INDUSTRIAL WASTE FORECASTS FOR OTHER NUCLEAR FUEL CYCLE OPERATIONS

The present chapter essentially deals with these wastes from nuclear industry which may constitute a medium or long term problem. These are covered in the preceding sections.

To illustrate that at the other side of the fuel cycle (uranium mining) the problem is far from preoccupying, the following estimate may be quoted (Ref. 3). On the basis of the power production forecasts during the coming century and assuming coal would be its principle source, several *million* Cu km of mine slag would be produced. The volume of coal in relation to that of uranium ore needed for the generation of the same amount of power is in a ratio of the order of 10^3-10^4 as compared to uranium-rich ores and 10^2 compared to ore poor in uranium. In addition, the slag resulting from the operation of coal mines is 10^2-10^3 times as large in volume as that from uranium mines. This leads to a total volumetric ratio of coal mining wastes of 10^5-10^6 over uranium wastes.

The magnitude of uranium ore processing waste volumes would therefore be of the order of cu kms only.

CHAPTER IV

Thermal waste from nuclear power production

1. General

This chapter 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 source of cold water, such as the sea, a river or a lake, to dissipate a considerable proportion 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 been the subject of many studies and research programmes. The large quantity of literature on the subject shows in particular that these effects are still little known, but also that directly harmful phenomena have not so far 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 future needs for electrical energy in the industrialized countries makes it possible to predict 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 factors, namely :

(1) 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 — which are often crudely defined — are exceeded.

(2) In the extreme case, a considerable increase in the quantity of heat released into the rivers and inland waters would heat up at least some of them to the point at which their fauna or flora were in danger.

2. Main general factors involved in the problems

(i) In a fossil-fuelled power plant, 38% of the heat energy is converted into useful electrical energy, while 53% must be evacuated by the cooling water and 9% is dissipated through the stack.

(ii) In a water-cooled plant, about 32% (31 to 33%) of the heat is converted into electricity, while 68% must be dissipated into the aqueous heat sink (river, lake, sea).

(iii) In HTGCR power plants, the efficiency may reach about 45% so that 55% has to be rejected in the aqueous heat sink.

(iv) In breader reactors, the efficiency is about 40%, with 60% to be rejected.

(v) The condensator-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 LWR plants have to reject about 50% [1] more heat to the *aqueous* environment than fossil-fuelled plants (HTGCR 14% less and an FBR 7% more) *per kwh of electricity produced.* For the same *'thermal' energy* produced, the LWR heat rejections to the aqueous environment is about 28% more than the fossil-fuelled plant.

(vi) In the Community, it is estimated that, in the future, nuclear power plants will supply an increasing proportion of the electricity provided by thermal sources [2] rising as follows : 22% in 1975, 26% in 1980, 40% in 1985 and almost 80% [3] in 2000.

(vii) 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 needs for electricity production will increase (22% of the total in 1970, and about 45% at the end of the century for the countries of the Community).

(viii) For economic reasons, more and more powerful units are used in the construction of nuclear power stations, resulting in local concentrations of nuclear power stations, resulting in local concentrations of thermal discharges which are only slowly dissipated.

(ix) 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 from the point of view of aesthetics, recreation, fishing, agriculture, etc. In addition, an increase in the temperature of waters which are already chemically or biologically polluted accelerates their degradation (e.g. the excessive development of algae leading to eutrophication) and finally their use for drinking or industrial water could be jeopardized once the self-purification mechanism is destroyed.

(x) Likewise, the distribution of electricity from sites where the dissipation of heat presents no problems (the seaside or on estuaries) also affects the cost of electrical power at the point where it is used. The tendency towards industrial development in coastal regions, to the detriment of inland areas, may therefore be accelerated in the long run.

(xi) Since, in spite of its enormous quantities, the thermal waste from power stations constitutes low-grade energy (small temperature difference), the possibilities of recuperation are very limited except in a few special cases. There are, in particular, the experiments now on hand aimed at using it for agricultural purposes and for raising some species of fish where a limited rise in temperature can have a beneficial effect.

3. Survey of suitable methods of heat dissipation

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

(1) *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 one.

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[1] 30 % more waste heat as compared to the conventional plant reject heat to both atmosphere and aquous environment.

[2] Conventional competitive sector : coal, petroleum products and natural gas (fossil fuels).

[3] Perhaps lower in certain countries (60 % in the U.K.) (Ref. 6).

(2) *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. Attention is tending more and more to turn towards this method as a means of heat dissipation. It is sometimes combined with a spray-type cooling system.

(3) *Salt-water cooling*

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

(4) *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 often 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 (for economic reasons given below).

(5) *Air-cooling towers*

The principle on which dry cooling towers work is the direct transfer of heat to the ambient air via a tubular radiator. The construction of this type of tower is more costly than that of 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 in the United States, for example, are not necessarily comparable. Unlike the United States' situation, the capital costs for the direct cycle fresh water system and for (wet) cooling towers are in Europe 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 cost of energy produced lies with air cooling towers between 10% (natural circulation) and 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 from the turbine). However, for high-temperature reactors, with the possible future use of gas turbines, 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 in Europe is given to natural-circulation (natural draft) wet cooling towers. Apart from the economic considerations entering into this choice, from the micro-climatological viewpoint the water vapour from this type of cooling system disperses quickly and the likelihood of the formation of low mist is substantially reduced. 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 the cost of the generation of electricity continues to drop, 'forced'-circulation (mechanical draft) wet towers will become in the future economically interesting (cf. United States [1]).

[1] In the U.S.A. : — wet mechanical draft tower : $ 10-12/kw;
— wet natural draft tower : $ 15-20/kw.

However, environmental problems could then arise, especially for 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 merit. It is therefore difficult to fix general criteria. Nevertheless, recently, some general guidelines have been issued in Western Germany for instance (Ref. 5). They can be summarized as follows :

(i) the power plants' released cooling water should 'after mixing' *never* (in any season) heat up the aqueous medium by more than 3°C, exceptionally 5°C;

(ii) the rejected cooling water should not exceed 30°C, exceptionally 35°C;

(iii) the 'mixed' water should not exceed the following limits :

  (a) for waters having summer mean temperatures between 17°-20°C (and peak temperatures of 23°C) : 25°C.

  (b) for waters having summer peak temperatures of 25°C : 28°C.

Usually these guide lines can easily be met. A point of controversy remains at what point downstream full 'mixing' is reached.

Similar but rather more detailed criteria have been issued in the US (Ref. 28- Chapter II) by the National Technical Advisory Committee on Water Quality Criteria.

6. Local and global long-term effects

6.1. *General*

Like for the gaseous radioactive routine releases from a growing nuclear industry (Chapter II) the question of rejected waste heat can be considered from the point of view of the local effects on one hand and from the standpoint of global world-wide consequences.

Notwithstanding the fact that, e.g. LWR nuclear power stations have to dissipate more waste heat than conventional power i.e. thermal production units the global as well as the local problem to be examined is not significantly different in both cases.

6.2. *Global effects*

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

It seems that for some highly industrialized countries by the turn of the century a major portion of the available inland waters would be used for cooling purposes, *assuming once-through direct cycle only were applied* (about as high as 2/3 of all inland waters in the US; Ref. 28, Chapter II). The situation would probably be equivalent in Europe. However the ultimate heat sink for power generators is the atmosphere. If one assumes that at the plant sites the heat would *all* be rejected by water *evaporation only*, the inland water thus consumed would represent a minor fraction of the water available (about 1% in the US; same reference).

Another indicative example can be quoted. Solar radiation provides about 100 000 times as much heat as all the electric 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 assumptions, all the heat released by conventional and nuclear power plants during the period from 1970 to 2000 would increase the earth's surface temperature by only 0.5°C and only after 10 000-100 000 years of operation.

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

In fact, a more serious problem in the long term, on the basis of 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 disequilibrium between its formation and reabsorption, which may occur in future times.

6.3. *Local effects*

Locally, the thermal problems are not negligible at all. For example, the artificial residual heat to be dissipated in an urban area has been estimated in the year 2000 to be about $1.4 \cdot 10^6$ cal/m^2 (500 Btu/sq.ft) to be compared with the heat received by solar radiation of $2.8 \cdot 10^6$ cal/m^2 (1 000 Btu/sq.ft). However again this is not a specifically nuclear problem.

The standards drawn up in various countries — for all methods of generating electricity and other industries producing residual heat — have resulted so far only in fairly general guidelines which could be made more detailed, for instance on the basis of research results and by the development of dispersion models and the correllated methods of calculation.

The possible local thermal effects can only be countered by the 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. In this perspective, the development of sea-shore or off-shore sites may increase significantly for countries where such possibilities exist.

The foreseeable medium and long-term perspectives of the development of HTGR and breeder-reactors are no doubt advantageous from the standpoint of thermal effects but are not to be considered a prerequisite for this reason.

Finally it can be concluded that these problems are not a particularity of nuclear power production and therefore should be dealt with in the framework of any measure to protect against excessive thermal pollution from industrial sources in general.

CHAPTER V

Accident potential, accident prevention and limitation of the possible consequences

1. Precautions of administrative nature

All nuclear installations, especially industrial ones, are subjected 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. Equivalent stringent precautions are seldom found in conventional (even hazardous) industrial activities or in the use of consumer goods (automobiles).

2. Precautions of 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 especially 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 firstly 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 and by any other Committees (e.g. the Advisory Commissions on Safety). This independent investigation results in 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.

By way of illustration, can be noted, for an industrially advanced country :

(i) all the work (design studies, compilation of safety reports, technical discussions with safety and control bodies) involved in obtaining an operating licence requires, on the part of the designer and the operator of a nuclear installation, about 240 man-months spread over about two years (Ref. 1). The normal amount of work which the safety and control bodies have to carry out in connection with the granting of an operation licence comes to at least 50 man-years spread over three to five years.

There is at present a tendency of even significantly more effort-input and lenghtier procedures due to the environmental implications of nuclear power resulting from the so-called 'nuclear controversy'.

(ii) It is no simple matter to calculate the 'direct' costs involved in these licensing procedures, which include the development and engineering expenses forming an integral part of the design work. However, an estimate of the concomitant costs (Ref. 1) specifically connected with the licensing procedure yields the following breakdown :

compilation of safety report		200 000 u.a.	(unit of account)
meetings with safety and control bodies		100 000 u.a.	
technical support (design studies, analyses)		350 000 u.a.	
legal and administrative support		100 000 u.a.	
	Total	750 000 u.a.	

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 non-nuclear industries except for certain advanced technologies which are particularly costly and/or dangerous for professionally exposed employees (e.g., aeronautics and space travel). It is only since recently that equivalent detailed safety analyses are sometimes applied in hazardous conventional industry (e.g. petroleum and chemical industry).

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

There is at present an increasingly marked tendency to develop — both nationally and internationally—, technological' standards [1] (criteria, codes and complementary requirements, guidelines, etc.), by means of which the methods of design and construction used and the imposed operating limits can be standardized. This tendency will undoubtely increase further with the development of nuclear power generation and the outlook of an international market of designs and equipment.

In general terms, the safety analyses and efforts towards standardization connected therewith cover the following fields :

(a) design analyses, which are important for the protection of professionally exposed and for public health and safety (e.g. qualitative and quantitative evaluations of systems reliability and insurance that the quality of the equipments will be adequately maintained throughout their useful life). Some examples are quoted :

 the treatment of effluents in 'normal' operation; reactor control and shutdown systems and the associated electronic and electromechanical equipment; power supply requirements; the maintenance of the quality of large mechanical components such as pressure vessel and primary piping; criticality control in reprocessing plants; the effectiveness of heat removal in irradiated fuel transportation containers.

(b) analyses of 'abnormal' transients and major accidents, e.g. : reactivity transients and accidents; large leaks or breaks in the primary or secondary circuits; interactions between the primary heat transfer medium and the fuel; explosive occurrences in reprocessing plants, missile effects; drop and fire resistance of transportation containers.

(c) design analyses or analyses of control methods specifically aimed at *preventing* abnormal transients or serious accidents, e.g. : reactivity limitations; special equipment to prevent rod dropping, methods of continuously detecting abnormalities; periodic surveillance and inspection; the redundancy of protective systems, protection against criticality accidents in reprocessing plants.

(d) analyses of systems specifically intended to *limit* the consequences of abnormal transients or serious accidents (mitigating means, engineered safeguards); e.g. : emergency cooling systems (their reliability and effectiveness); secondary containments and engineered means of limiting the pressure, temperature or, possibly, explosive effects or shock waves; emergency ventilation and emergency electricity supplies.

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 presented by power reactors. Generally, several types of potential accidents and the corresponding mitigating means form the subject of detailed analyses, which more and more

[1] As opposed to 'radiation' standards dealt with in Chapter II, for instance.

often incorporate probabilistic considerations such as comparative analyses of the reliability of protection and emergency systems or classifications of accidents according to their degree of severity (e.g. Ref. 5).

Moreover, the probabilistic approach to the analysis of accidents, which tends to link the probability of events to the seriousness of the consequences (e.g., Ref. 2 and 3) 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 problems (e.g., uncertainty of the statistical data used and compared with).

For accidents of maximum conceivable seriousness [1], it is generally considered acceptable for an individual member of the population to receive 25 rem 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 the reactor selected. Broadly speaking, however, it may be seen that, in this serious case, for a 1000 MWe plant, the total integrated individual irradiation doses lie between about 0.1 and 10 rem (mainly due to whole-body irradiation by noble gases), whereas the accident doses due to radioactive iodine lie generally between 0.01 and 0.1 rem (Ref. 4, numerous others and safety reports).

With the medium and long-term development of FB reactors (e.g., sodium cooled — see Chapter I) the accident considerations will have to include more and more the Plutonium-hazard [2] (likewise for thermal Pu-recycle). Whilst the iodine retention will probably be improved by the the presence of Sodium-vapour in accident conditions, the Plutonium can only be volatilized under aerosol (particulate) form.

Preliminary calculations — assuming conservative release and dilution values (e.g. 0.5% containment leak rate, 100 m stack, $2 \cdot 10^7$ Ci m^{-3}/Ci s^{-1} dilution) indicate in the vicinity of the plant (500 m) concentrations by a factor 100-1 000 below the permissible concentrations for the public 'at risk' (Ref. 6).

It is likely that more attention will in the future be given to accidents, which find their origin in 'external' causes, against which design did *not* explicitly protect (as opponent for example to seismic or flooding design). Examples of such external causes are airplane crash or even sabotage. The assessment of these questions is certainly not easy to tackle. There is no doubt that here also probabilistic approaches will prove useful. A recent study (Ref. 9) has estimated for the particular BWR and PWR designs chosen in specific regional conditions (Switzerland), that the risk (frequency, activity release) would still be at least two orders of magnitudes below the proposed tolerable risk-limit (Ref. 2) (see also Chapter VI).

In general terms it should be pointed out, first that equivalent external hazards also exist for conventional industries and activities (chlorine tanks, explosives, hi-jacking, etc.), secondly that the way of protection of man against such hazards can therefore reasonably only be the same for both conventional and nuclear activities, e.g. location versus airplane landing strips, administrative security measures, exceptional police measures, etc.

4. Present and future siting implications

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

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[1] Based on engineering judgement taking into account the state of the art (deterministic approach).
For instance a 1 000 MWe FBR may contain an inventory of $8 \cdot 10^6$ CiPu and $80 \cdot 10^6$ Ci I-131.

It should be noted that :

(i) 'Site criteria' (if based on 'accident' considerations) often lead to allowable 'exceptions' to the 'basic' requirements.

(ii) Siting practices are rather divergent from one country to another (and even within one country) and are, from the safety standpoint mostly still dealt with on a 'case by case' basis, taking into account possible supplementary preventive or accident mitigating safeguards. This divergence is, for instance, demonstrated in a recent paper comparing population distributions around nuclear power plant sites in the U.K. and the Community of Six (Ref. 7).

In a growing nuclear industry there will be — in the next decade — need for *nuclear power plants* to develop more generally applicable site selection requirements based on 'routine' as well as 'abnormal' conditions'health and safety considerations. 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 become more usefully applied in the future.

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

5. Present situation and future outlook

Up to 1970, ninety power plants distributed over the whole world had produced 250 billion kWh of electricity and accumulated 650 years of experience without any significant accidents from the point of view of the population at large (Ref. 8).

It may be hoped that this positive balance can be maintained with the development of the nuclear generation of power as a result of the continued application of strict standards and precautions and more stringent quality control.

An assessment of the risk of damage involed in serious nuclear accidents is given in Chapter VI.

There is no doubt that, at present, the record of nuclear power as compared e.g. to conventional hazardous industries (such as petroleum, chemical) is extremely favourable from the point of view of material damage, injury or death caused to the general public and from the point of view of professional accidents (see Chapter VI). With a growing nuclear industry and the development of higher ratings and of new technology (e.g. FBR's), the potential of accidents tends to increase and there is certainly, in this connection, merit in developing methods to assess quantitatively future 'risk-potential ranges'. However, it may be emphasized :

(a) that this trend is not specific for the nuclear industry only, and applies certainly in similar porportion also for conventional industry and hazardous consumer goods;

(b) that concurrently with nuclear power production increase and development of new technologies, increasingly expanded nuclear safety research programs are developed, of which a counterpart is hardly to be found in the conventional field.

CHAPTER 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 considered with care because of the relative value of the interpretations placed on statistical information. A few comparative values are summarized here by way of example, showing how the nuclear energy 'hazard-potential' is situated in relation to other industries and human activities, the risk of which is generally accepted either individually or as a community.

1. Risk in occupational duties

1.1. *Accidents*

Statistics extending over 22 years' operation [1] of various types of nuclear installations (laboratories, reactors, prototypes, etc.) and $2.5 \cdot 10^9$ man-hours show, for bodily injury involving inability to work : 7 693 individual cases, of which only 36 (0.5%) were due to effects of radiation. This gives an accident frequency rate of

$$\frac{7693}{2.5 \cdot 10^9} = 2.45$$

accidents per million man-hours, corresponding for the country concerned, to one-quarter of the national frequency rate for all industrial activities.

Moreover, the partial frequency of accidents due to radiation is only

$$\frac{36}{2.5 \cdot 10^9} = 0.01$$

accident per million man-hours, which is clearly negligible in relation to the national frequency under consideration.

Numerous statistics on various conventional professional (mining, industries) activities have been issued, some of which have been reported in Ref. (1) and (2). Grossly speaking such accidents have in industrialized countries an individual casualty (fatal injury) probability of about 10^{-4} per year at exposure and a permanent injury probability of about 10^{-2}.

Compared to this the nuclear industry is in a very favourable position, because of the high degree of health and safety precautions taken since the onstart of development onwards a quarter of a century ago.

[1] Symposium on accidental irradiation at place of work (26-29 April 1966 : a review of criticality and reactor incidents at USAEC installations).

1.2. *Normal operation*

However — as has been outlined already in section II.A.2. the 'normally' accumulated doses by professionally exposed may — remaining within the permissible radiation standards — nevertheless in integrated man-rem, lead to a significant consumption. This will be more so with an expanding nuclear power production (especially for multi-unit sites) and associated industry. This problem has already been tentatively approached (Ref. 3 and 4) by assessing the social value of the man-rem concept and the economics involved in 'risk' acceptance based on this concept and in supplementary protective measures.It boils down to question for instance 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 to other industrial occupational risks.

These considerations may become a further justification (besides those mentioned earlier with regard to the general public chapters II and V) for introducing also 'man-rem criteria' besides the individual dose limitations (basic standards).

2. Nuclear risk in general in relation to other risks

Table 11 below 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 would be well first of all 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 firearms, swimming, etc.). Risks of the order of 10^{-6} and below do not worry the population.

Remarks to table 11

1. The genetic hazards, which do *not* cause fatal injury in the true sense have *not* been included in this comparative table. However some indications (e.g. expressed in natural = genetic death = rate — extinction of a gene lineage) can be given on the basis of mutation rates and the genetic equilibrium of the population (cfr. Ref. 7 and 8). 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, of which only 2.5% 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 more 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 (Ref. 10). 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 cm^3 of alcohol per day would correspond to 50 rem/year.

Based on this information it is found that the individual risks resulting from nuclear energy in routine operations are negligible for the general public, as long as the presently applied stringent radiation standards are reported. As shown in previous chapters of this papers there is no reason to believe that this would not be the case on a medium- and long-term basis.

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

TABLE 11

Probabilities of individual fatal injury (casualty) through conventional activities and causes and through the effects of radiation

Type of risk	Individual probability of fatal injury per year of exposure (orders of magnitude) [1]	Remarks
Conventional (casualties only)		
— all diseases	10^{-2}	
— motor accidents (automobiles)	10^{-4}	10^{-2} for light heavy *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.

TABLE 11 (continuation)

Type of risk	Individual probability of fatal injury per year of exposure (orders of magnitude) [1]	Remarks
Nuclear effects of radiation (individual injury, not necessarily *casualty* except perhaps with a long latent time)		
1. *Radiation in accident conditions* (in the hypothesis such accident occurred; 'frequency' considerations)	10^{-3}-10^{-4}	1. Bases 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 (carcinoma) (Ref. 7 and 8). 2. For 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.

[1] Summary of data from various sources; with slight variations according to the country.

3. Risk on a medium- and long-term basis - probabilistic approaches

It is difficult to define accurately the quantified probabilities of occurence of various types of accidents which would result in consequences to the environmental population and in particular to fix a clear-cut border-line between conceivable and inconceivable accidents. At worst one could speak about radioactive material releases exceeding even largely the doses considered acceptable for emergency purposes (without evacuation requirements) mentioned earlier.

Attempts have been made semi-quantitatively or quantitatively to solve this question. For instance in ref. (11) and ref. (2) - chapter V, 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-urban and remote sites taking into account the severity of injury which could be caused by the respective release magnitudes, the population densities and the nuclear power production growth requirements.

To compare with non-nuclear risks it has been suggested in such case (Ref. 11) to compare with 'crowd-type' of accident-hazards. A comparison with estimates on random crashes of aeroplanes around airfields (a risk 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 programme. The latter would be about the same as the risk of death from meteorites.

Perhaps the most appropriate approach to be made for 'unlikely' (or inconceivable) nuclear reactor accidents is — rather than to compare with rather frequent recurrent conventional hazards affecting a relatively small number of people — to compare with man-made constructions potentially affecting 'large' groups of the population (e.g. dam-type developments). An interesting example which was recently quoted (Ref. 4) refers to the Netherlands 'Schelde-river Delta plan' : this plan has to protect about 1 mns of inhabitants; assuming an 'inconceivable' flood happens once in 10 000 years and causes casualties in 0.1% of the population then the risk is about 10^{-7} [1]. This seventh order of risk is accepted by society with regard to the benefits. Such catastrophy-type of accidents of nuclear power plants leading to casualties are of the same order of magnitude or even less.

Another similar example of a risk, which is in Western Europe readily accepted and will never lead to supplementary precautions (structural, warning systems, etc.) can be put forward. On November 13th 1972 a storm hit Western Europe with wind speeds up to 125 m/sec on the continent. This is about the most severe hurricane force one can note over long periods of time in this moderate climate region. The last time a storm of equivalent severity affected widely the British Isles and the Continent was in November 1940 with wind spreads up to 150 m/sec.

Both these storms caused casualties, injuries and material damage, but only on the latest one information is available; as to the 1940 storm little attention was paid, presumably because of war conditions. Confining ourselves to the 'casualties' out of a population of roughly 200 mio affected, about 54 persons were killed. This risk spread out over the period between 1940 up to now (roughly 30 years) amounts to about 10^{-8} 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 more than that expected from catastrophy type of accidents from nuclear power.

This type of estimation and comparison with nuclear power requires undoubtedly further examination but they are inherently attractive on a long-term basis.

Once more in such probabilistic estimations, the man-rem concept may prove to become useful, as total body irradiation has to be considered, besides the thyroid — carcinoma effects which would result from important Iodine releases taken mainly as reference so far (Ci I-131 equivalent).

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[1] 10^{-4}/year $\times$ $10^{3}/10^{6}$ persons $= 10^{-7}$/person-year.

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