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Experience in the application of exemption principles

Proceedings of a specialists meeting held in Vienna, 2–4 November 1993

INTERNATIONAL ATOMIC ENERGY AGENCY

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FOREWORD

This publication contains the proceedings of a Specialists Meeting on Experience in the Application of Exemption Principles held from 2 to 4 November 1993 in Vienna.

The aim of the meeting was to provide a forum for the presentation of national and international experiences in applying principles for the exemption of radiation sources and practices from regulatory control. The meeting offered an opportunity to review the progress made in applying the principles which were the subject of an international consensus in 1988 and were summarized in IAEA Safety Series No. 89.

The meeting was held as part of an ongoing IAEA programme on the application of exemption principles to radioactive waste management.

EDITORIAL NOTE

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INTRODUCTION

Radiological principles for exempting sources from regulatory control were established in the course of a series of international meetings in 1987 and 1988 and published in IAEA Safety Series No. 89[1]. There are several areas, particularly in the field of radioactive waste management, in which exemption principles can be usefully applied. For example, much of the low level radioactive waste which is generated both in the nuclear fuel cycle and in the application of radioisotopes in medicine, research and industry presents a very small hazard to health and may be considered as material that can potentially be exempted. Similarly, a significant fraction of the materials recovered from the decommissioning of nuclear facilities contains very low levels of radionuclides and may be reused or recycled with negligible hazard to members of the public.

Since the principles were published, efforts have been directed towards their conversion into practical and measurable quantities in various application areas. The value of the principles can now be more properly assessed by examining the derived exempt values and their usefulness in terms of the types of sources and the quantities of material which can be released under the exemption option. The IAEA has published guidance on the derivation of exempt quantities in the context of disposal[2] and recycle and reuse[3].

Publication of the principles prompted reviews of national arrangements for exemption from regulatory control in several countries and the introduction of new or revised regulations. At the international level the principles have formed the basis for revised guidance on exemption in international standards.

Some problems have arisen in applying the principles, for example, in the methodologies used for the derivation of the practical quantities, in their translation into regulations, and in relation to their public acceptance.

The Specialists Meeting on Experience in the Application of Exemption Principles held from 2 to 4 November 1993 in Vienna was organized with the intent of providing a forum at which experience in applying the principles could be presented and exchanged.

The papers presented at the meeting covered different aspects of the subject and were categorized as follows:

- International developments on principles and standards,
- Exemption principles in national regulations,
- Derived exemption and clearance levels,
- Practical application of exemption and clearance concepts.

This publication is organized as follows: the first part contains summaries of each of the four sessions of the meeting as provided by the respective chairpersons and an overall summary of the meeting by the scientific secretary. This is followed by the papers presented at the meeting organized according to the session in which they were presented.

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[1] INTERNATIONAL ATOMIC ENERGY AGENCY, Principles for the Exemption of Radiation Sources and Practices from Regulatory Control, Safety Series No. 89, IAEA, Vienna (1988).

[2] INTERNATIONAL ATOMIC ENERGY AGENCY, Exemption of Radiation Sources and Practices from Regulatory Control-Interim Report, Progress at the IAEA on the Development of Radiological Principles for Exemption and the Application of the Principles to Low-Level Radioactive Waste Disposal in the Terrestrial Environment, Parts I and II, IAEA-TECDOC-401, IAEA, Vienna (1987).

[3] INTERNATIONAL ATOMIC ENERGY AGENCY, Application of Exemption Principles to the Recycle and Reuse of Materials from Nuclear Facilities, Safety Series No. 111-P1-1, IAEA, Vienna (1992).

SUMMARIES OF SESSIONS AND DISCUSSIONS AT THE SPECIALISTS MEETING

INTERNATIONAL DEVELOPMENTS ON PRINCIPLES AND STANDARDS (Session 1)

S. Mobbs
United Kingdom

Three main points were raised in the discussion which followed the session. The first concerned the way in which the exempt levels for total activity and activity concentration were used in the draft Euratom Basic Safety Standards (EUR BSS) and the draft FAO/IAEA/ILO/NEA(OECD)/PAHO/WHO Basic Safety Standards (IAEA BSS). The draft EUR BSS states that small quantities of radioactive substances are exempt from the requirements of reporting if either the total activity or the activity concentration are below the relevant exempt levels. The exempt levels calculated for the EUR BSS were derived on this basis. The IAEA BSS, however, adopts the same exempt levels as the EUR BSS but states that both the total activity and the activity concentration must be below the relevant exempt levels, It was not clear whether this was a deliberate modification or not. (Note - in the 5th and 6th versions of the draft "IAEA BSS" the change to "or" was made. Ed.).

The second point concerns the issue of skin doses. The exempt levels for the EUR BSS included calculations of skin doses and used a skin dose criterion of 50 $mSva^{-1}$. It can be shown that for some scenarios, for example, those involving holding objects contaminated with beta-emitters, the skin exposure pathway may be critical. IAEA Safety Series No. 89 does not prescribe a separate dose limit for skin but refers only to "effective dose". This deficiency has to be borne in mind when applying the advice in Safety Series No. 89.

The third and most important question was the definition of the terms "exemption" and "clearance". This was discussed extensively as it was obviously a source of confusion. The contents of the viewgraphs provided by Mmes Sugier and Chapuis to clarify the distinction is shown in Figure 1. Exemption as used in the context of the EUR BSS only refers to small quantities of radioactive materials. It means that materials at or below the exempt concentration or activity need never enter the regulatory system. The use and subsequent disposal of such material is therefore outside the system of surveillance by the authorities. If, however, the activity or concentration of the material is above the exempt level, or the quantity of material is very large, then the material is subject to the system of surveillance and authorization. Clearance levels apply at the point of exit from a practice that is already under regulatory control. Radioactive material below the clearance level can be released for recycling or disposal in an unrestricted manner. (This is of particular interest for materials arising from the decommissioning of nuclear installations.) The receiver may or may not know that the material is radioactive. Radioactive material above the clearance level requires an authorisation with the derivation of discharge limits under the conventional procedures of optimisation under a constraint.

It was generally agreed that the dose criteria for exemption and clearance should be the same. However, unconditional clearance levels (levels at which radioactive material can be released without restriction on the fate of the material) will generally be less than or equal to exemption levels because of the larger volumes involved, and hence the possibility of extra exposure scenarios or pathways. Conditional clearance levels are levels that are derived for material that is released for a particular destination, e.g. disposal at a landfill site. These will be higher than unconditional clearance levels.

There was some discussion on the need for clearance levels at all. It seems inconsistent to treat the release to the environment of solid radioactive materials arising from practices subject to the regulatory system differently to the release of liquids and gases, i.e. via a clearance system rather than an authorisation procedure. However, from the decommissioning viewpoint, there seemed to be some need for conditional clearance levels, for example, for recycling. The US concept of "general licensed materials" was described as an alternative approach.

CLEARANCE LEVELS	EXEMPTION LEVELS
Provided by national authorities	EC Basic Safety Standards
Authorisation of release	Exemption from reporting
Material from licensed plants	Small quantities of radioactive material
Unconditional and conditional disposal or recycling	Use, and subsequent disposal
The receiver may know, or not, the radioactivity of the material	Radioactivity of the material is known by the user but not after disposal

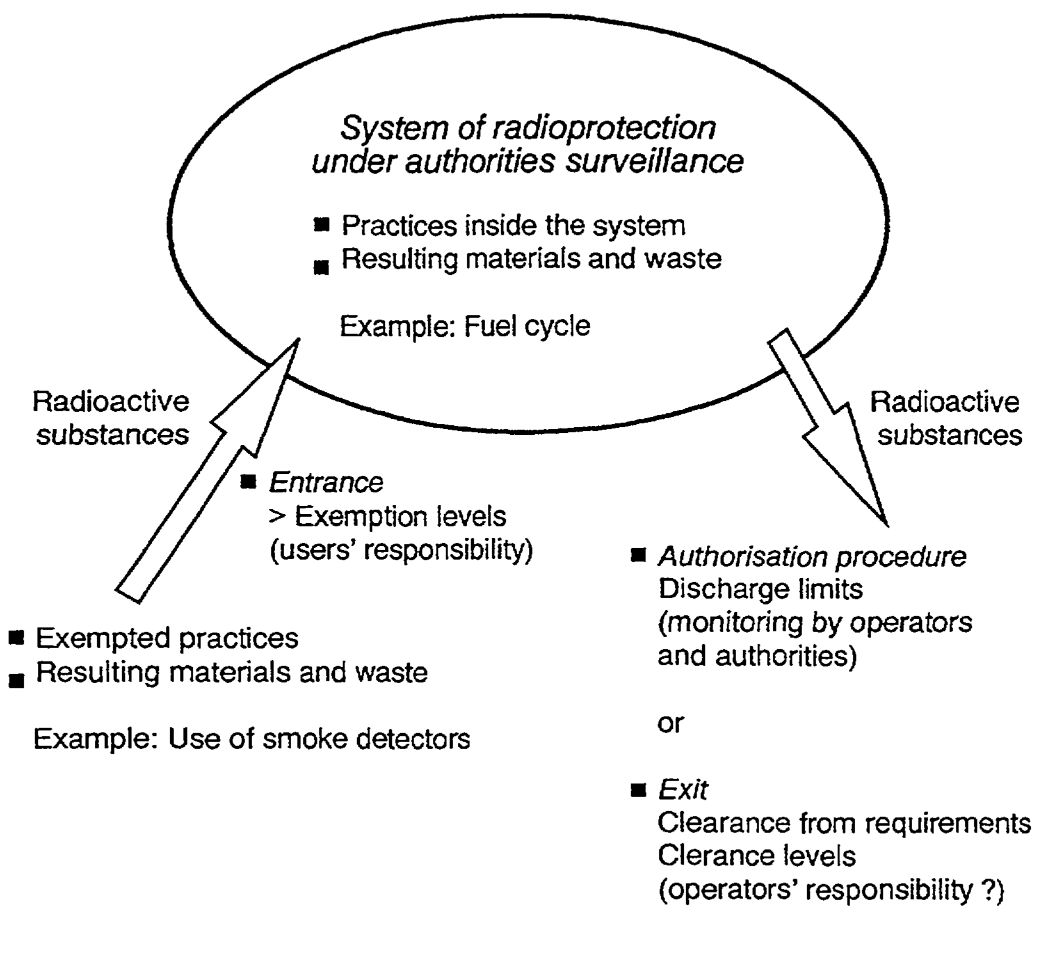

Illustration of the first articles of the EC directive

FIG. 1.

EXEMPTION PRINCIPLES IN NATIONAL REGULATIONS
(Session II)

L. Baekelandt
Belgium

Exemption/clearance is practised in most countries represented at the meeting, sometimes on a case-by-case basis, sometimes on the basis of general arrangements within national regulations. In some countries studies are being carried out with a view to develop such regulations.

The radiological basis being used by the countries represented at the meeting appears to be consistent with the general principles laid down in IAEA Safety Series No. 89 (except for one country).

For surface contamination, the limiting values from the IAEA Transport Regulations (Safety Series No. 6) are commonly used.

Many countries seek advice from international bodies and hope that an international consensus will be reached on exemption/clearance levels. However, it is recognized that country specific levels may also be needed.

From the discussion it seems that difficulties may be expected in implementing the low exemption levels, as proposed by CEC and IAEA, for non-nuclear fuel cycle industries where large amounts of material containing naturally occurring radionuclides are involved.

Many questions were raised about the proposals for new Euratom Basic Safety Standards, in particular about the provisions dealing with exemption and clearance.

The change of terminology that was introduced recently gives rise to difficulties in understanding the difference between exemption and clearance (see International Developments on Principles and Standards (Session 1) S. Mobbs, UK).

Some countries seek guidance from international bodies on criteria for the clean-up of large surface areas, for example, contaminated land and soils.

DERIVED EXEMPTION AND CLEARANCE LEVELS
(Session III)

A.M. Chapuis
France

Among the six presentations, only one was really dealing with exemption; the others were concerned with clearance. The presentation concerning exemption described the methodology used for calculating activity levels for exemption from the requirement of reporting in the Euratom Basic Safety Standards, presented by S. Mobbs. This exemption concerns only small users of radioactive materials, it does not apply to material or waste from nuclear installations.

The other papers presented numerical values for the *clearance* of materials. They were:

- IAEA proposal for unconditional clearance of all types of material or waste,
- US study on scrap recycle and items or building reuse,
- German recommendation for the recycling or reuse of metal scrap,
- Belgian and French study for the disposal of waste to landfills,
- Spanish examples of specific clearance and studies related to later release.

The presentations and the discussions which took place during this session brought out several ideas:

- When the calculation of the activity limits is done for a very general application, including all types of disposal, recycle and reuse, there are still many questions concerning the validity of the scenarios and the choice of the parameter values. However, a large number of derivations have now been carried out and it seems that the limiting values do not change by more than an order of magnitude, even with improved knowledge of parameters.
- The problem concerning low probability events remains. It is not clear that the ICRP 60 recommendations on the use of risk instead of dose are helpful in the case of exemption, release or clearance.
- The omnipresence of natural radioactivity, at levels leading to doses much higher than 10 μSv per year, may point to a need to reconsider the criteria used for clearance that could be different from those used in the case of materials contaminated with artificial radionuclides.
- Obtaining a clearance for release does not mean that we can relinquish our responsibilities. Clearance or release applications require an in-depth knowledge of the practice, controls at the point of release, records of adequate documentation and a very high degree of technical and administrative rigour throughout the process. This is the only way in which recycling or disposal outside the nuclear field can become accepted.

PRACTICAL APPLICATION OF EXEMPTION AND CLEARANCE CONCEPTS

(Session IV)

W.E. Kennedy, Jr.
United States of America

This session consisted of eight technical papers covering practical clearance experience and current regulatory activities. Four papers considered various aspects of technology related to decontamination methods, experience with smelter operations, and radionuclide characterization methods. One paper provided a case history of two situations where radioactively contaminated metals well above the IAEA draft clearance levels had been accidently released for recycle and another paper provided a case history of an actual decontamination and decommissioning of a former nuclear fuel reprocessing facility. Finally, two papers provided an overview of the efforts of national authorities to develop new regulations regarding clearance by metal recycle and license termination after decommissioning.

The first technology-related paper was entitled "Decontamination of Metal Components to Clearance Levels by Means of Abrasive Blasting" by L. Teunckens et al. from Belgium. This paper showed that favorable metal decontamination results can be obtained using both wet and dry methods that are routinely used during facility decontamination. Application of these methods prior to smelting could reduce levels sufficiently to allow recycling; however, clear international criteria for clearance are a necessity. The second technology-related paper was entitled "Radiological Characterization of Metal Components in View of Melting" by L. Teunckens and W. Blommaert from Belgium. This paper described problems encountered in attempting to characterize radioactive contamination in metal components with difficult to monitor surfaces. The problems were made more difficult because of a lack of records about the history and operation of the components. After melting, the overall inventory associated with the components was determined to be between a factor of 2 to 2.5 greater than the initial monitoring results. The third technology-related paper was entitled "Recycling of Radioactive Contaminated Metals from Nuclear Installations" by M. Sappok from Germany. This paper provided an overview of the German experience with metal recycle. The experience shows that the recycle criteria imposed in Germany can be met and that, for selected radionuclides, multiple melts can purify the metal ingots. Partition factors in the metal ingot, slag, and offgases for various radionuclides were provided. The final technology-related paper was entitled "Decontamination and Recycling of Contaminated Metals in Sweden" by D. Aronsson. This paper provided information from three case histories associated with the recycle of various metal components from nuclear reactors. Information on worker dose during smelting and the potential costs and cost savings of smelting were presented.

Two papers provided case history information. The first was entitled "Radioactive Contamination of Metal Scraps in Italy" by A. Susanna and S. Piermattei. This paper reported on two situations where contaminated metal was brought into Italy for recycle. The first situation involved ^{137}Cs contaminated aluminum and was detected through environmental samples only after smelting. The second involved ^{60}Co contamination detected in steel scrap being transported on a rail car. These events prompted recommendations for revised regulations in Italy and resulted in proposals for international actions associated with potential control or monitoring of metal scrap crossing international borders. The second paper was entitled "Decommissioning of Final Product Storage Buildings at the Former EUROCHEMIC Reprocessing Plant" by L. Teunckens, et al. from Belgium. This paper provided detailed information about an actual decommissioning project regarding technologies, costs, manpower, decontamination factors, survey methods, and worker doses.

Finally, two papers were presented providing an overview of the efforts of national authorities in the United States to develop new regulations regarding the clearance of materials covered by license conditions. The first paper was entitled "A Preliminary Economic Analysis of Recycling" by G. Durman from the U.S. Environmental Protection Agency (EPA). The paper provided the elements under consideration by the EPA including collecting information on existing and potential inventories, potential technologies, environmental impacts, health risk estimates, options for the fate of materials and cost/benefits. Since there is a high degree of public concern about the risks from exposure to radiation in the U.S., the paper asked the question "How does recycling resolve the political problems of low-level radioactive waste disposal?" Although recycle of metals makes environmental sense, will the public accept it as a practice? The second paper was entitled "U.S. Nuclear Regulation with Enhanced Public Participation" by R. Meck et al. from the U.S. Nuclear Regulatory Commission (NRC). This paper provided an historical account of the birth and death of the "Below Regulatory Concern (BRC)" policy and current efforts by the NRC to develop policy associated with decommissioning. The NRC was forced to abandon the BRC policy because of issues with the States about when the States may adopt more restrictive regulations than those developed by the NRC. The lessons learned from the death of the BRC policy included the fact that building consensus after the fact does not work; the public and interested parties must be involved through the policy development. It was also noted that the BRC policy was too broadly based and that a more specific set of regulations is needed, each focusing on a specific area like decommissioning. Finally, information about the NRC enhanced participatory rulemaking effort, with public workshops and involvement, was presented.

(Note by Editor: the presentations by D. Aronsson, by A. Susanna and S. Piermattei, and by G. Durman were not accompanied by papers and are not therefore included in this document.)

OVERALL REVIEW OF THE MEETING AND GENERAL CONCLUSIONS

G. Linsley
International Atomic Energy Agency

The basis and starting point for much of the work presented at the meeting was the exemption principles set out in Safety Series No. 89. Although some modifications and refinements were suggested during the meeting to cover issues such as skin dose limitation and treatment of probabilistic events, the basic guidance in the document was not seriously challenged. Indeed, it was noted that the principles had been endorsed in ICRP publication 60 and that they have been used as the basis for deriving the exempt levels in the FAO/IAEA/ILO/NEA (OECD)/PAHO/WHO Basic Safety Standards and in the Euratom Basic Safety Standards.

An important issue for many at the meeting was the exemption levels established in the two draft BSS documents and their interpretation. In particular, the implication that many premises in which naturally occurring radionuclides are used could now be subject to regulation (under the Euratom system) concerned many people. Another issue discussed at length were the distinctions between exclusion, exemption and clearance. The discussions served to reveal and sometimes resolve some of the issues, but it is evident that some problems remain.

The session on exemption principles in national regulations showed that while there is great interest in countries to apply the exemption concept, there is still comparatively little experience of exemption in use - outside of a handful of countries where exemption has been practised for many years. As a result, there was only a limited amount of feedback on related problems and issues.

The view of those at the meeting engaged in translating the exemption principles into practical working quantities using modelling techniques is that sufficient work has now been done in the areas of disposal, recycle and reuse for there to be confidence in the reliability of predictions and therefore in the derived quantities. It was felt that there are no real gaps in knowledge. The same confidence was not shared by representatives of those involved in the practical application of exempt levels. They felt that too much conservatism is present in many of the calculations. They noted that the opportunity to test modelling results in real industrial situations now exists and should be taken.

The issues of economics and public opinion in the context of exemption are important and may be expected to have a strong influence on the long-term success or failure of the exemption option. They were touched upon in the final session in a presentation from the USA but were not broadly debated. Again, this is a reflection of insufficient experience having been obtained on these subjects.

In summary, some useful experience has been gained in the application of exemption principles but much more is needed. It may therefore be valuable to hold a similar meeting on this subject in a few years time.

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INTERNATIONAL DEVELOPMENTS ON PRINCIPLES AND STANDARDS

(Session I)

Chairman

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CRITERIA FOR EXEMPTION FROM THE REQUIREMENT OF REPORTING IN THE EURATOM BASIC SAFETY STANDARDS

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Abstract

The general requirement of reporting practices involving radioactive substances is waived in a number of cases, in particular if the total activity or activity concentration of the substances is below nuclide specific levels. The scope of application of such exemption levels is not straightforward because it is e.g. overruled by the requirement of prior authorisation for practices belonging to the nuclear fuel cycle and for waste disposal or recycling.

The radiological criteria applied to the setting of exemption levels are to a large extent inspired by the recommendations in IAEA-SS 89. However, further considerations emerged during implementation of these principles which shed new light on the issue.

The methodology developed for establishing exemption levels is the subject of another paper.

1. Introduction

The Basic Safety Standards for the health protection of the general public and workers against the dangers of ionising radiation (1) incorporate values of activities not to be exceeded so that the requirements for reporting and obtaining prior authorisation of activities involving a hazard arising from ionising radiation need not be applied (Article 4 of Council Directive 80/836). For this purpose all relevant nuclides have been classified in four groups, according to their relative radiotoxicity. Also radioactive substances of a concentration of less than 100 Bq g^{-1} are exempted from this requirement, this limit being increased to 500 Bq g^{-1} for solid natural radioactive substances. These exemptiôns, while allowing competent authorities in Member States to disregard a multitude of trivial practices, have so far not given rise to any situations where the health of the general public of of workers were put at risk. However, the opportunity of a major revision of the Basic Safety Standards, to bring these in line with the recommendations of ICRP (publication 60), was taken to introduce a more transparent and consistent methodology for establishing exemption levels on a nuclide-specific basis.

This paper sets out the overall framework within which such exemption levels have been established, including the regulatory context and international guidance on exemption criteria. The methodology applied for the actual calculation of the levels is described in a second paper (2).

2. Framework

2.1 Exemption in the current EC-Directive

In the current EC Directive 80/836/EURATOM (1) all principal radionuclides are classified in one of four radiotoxicity classes. The number of nuclides was extended in the 1984 amendment to the Directive (3). The classification was still performed on the basis of a grid with maximum permissible intakes expressed in activity (Ci) and mass (g) as co-ordinates, as proposed in an IAEA report of 1963 (4). Exemption values, below which the requirements for reporting and obtaining prior authorisation need not be applied (Art. 4(a) of the Directive), relate to the four radiotoxicity classes in a range 5.10^3–5.10^6 Bq (total quantities involved). As for the concentration exemption values, there are merely two figures: 500 Bq g^{-1} for solid natural radioactive substances and 100 Bq g^1 for the other. Exemption is granted if either one of the exemption values, total activity or concentration, is not exceeded.

The Directive does not specify the type of practices that can be exempted. This gave rise to different interpretations. In some cases the exempt concentration values have been used in the context of solid waste disposal. In general, however, authorities have introduced controls at much lower levels.

2.2 <u>IAEA-NEA guidance</u>

Guidance on principles governing exemption from regulatory control was given by IAEA-NEA in 1988 and published as IAEA Safety Series no 89 (5). This document proved to be a major breakthrough in international consensus on this issue. Well-known are at least the radiological criteria that were proposed: an annual effective dose of 10 μSv to individuals in the critical group, for each exempted practice, and a collective dose of 1 man.Sv per year of practice. For a better understanding it is worthwhile, however, to draw attention to a few other conditions that need to be fulfilled in order to exempt a practice.

A first consideration is that regulatory authorities can always exempt practices from different levels of regulatory control on the basis of an impact assessment. In some cases however practices may be exempted from all sorts of controls and hence from the overall system of notification, registration and licensing. The guidance refers to the latter situation. It is important to note that exemption from controls does not imply exemption from the radiation protection requirements. It is e.g. explicitly stated that justification is not overruled.

It is also a prerequisite that the practice considered for exemption be well defined. In particular it is necessary to be able to identify easily the sources and coordinated activities constituting the practice, and one or more critical groups concerned specifically by the practice. Similar practices may be considered separately only to the extent that the critical groups are different or doses are not significantly affected by taking into account more than one practice.

The individual dose criterion brought forward in SS89 was derived in terms of the concept of trivial or negligible risk and of variations in the natural radiation environment. It was concluded that doses of a few tens of microsieverts would satisfy this criterion, which for practical applications and for a single exempted practice is taken to be 10 μSv. The collective dose criterion is based on the assumed cost of an optimisation assessment. This cost would be in excess of the possible reduction of the collective dose if the latter is lower than 1 man.Sv.

The IAEA-NEA document further gives examples of practices liable to exemption, ranging from the placing on the market of consumer goods, disposal, recycling or reuse of solid waste or scrap material, and even to the discharge of low level liquid or gaseous effluents. Attention is drawn to cases where sources or practices have been subject to controls and are exempted at a later stage. This change of status remains as such under control and can be subject to revision on the basis of information on the exempted practices that continues to be forwarded to the competent authorities.

2.3 ICRP-recommendations

ICRP-publication 60 (6) also devotes a chapter to the concept of exemption from regulatory control. While referring to the advice issued by IAEA and NEA, ICRP points to the difficulty in establishing a basis for exemption on grounds of trivial dose, and to the underlying problem that exemption is a source-related process while the triviality of dose is individual-related (par. 288). If a source is exempted, it may be necessary to distinguish between different practices to which a class of devices, may belong (par. 289). It may e.g. be justified to exempt the sale and use of the devices, but not their manufacture and large scale storage. If the use of a source is exempted, it is necessary to be able to exempt also its eventual disposal.

Exemption on the basis of collective doses is considered a possibility, however on grounds of an exercise similar to the optimisation of protection. Contrary to the IAEA-NEA advice, this is thought to provide a logical basis for exemption of sources that cannot be exempted solely on the grounds of trivial doses (290). It is generally understood that the SS89 criteria for individual and collective dose are meant to be fulfilled simultaneously. ICRP gives little emphasis to collective dose as a quantitative basis for exemption, but rather to an assessment whether regulation on any reasonable scale will produce significant improvement.

3. Reporting and prior authorisation

3.1 Regulatory control

The Basic Safety Standards Directive addresses Member States and imposes the incorporation of its provisions into national radiation protection legislation. It sets out the overall framework or the goal to be achieved and leaves it to the legislator to work out the regulatory structure that is most suitable for the purpose in the national context.

Administrative procedures may vary from one country to another because of different attribution of competences or different legislative tradition.

There is also a descretionary power of the competent authorities at the level of the individual practice or undertaking. This power is exercised most clearly with practices involving significant hazards and therefore requiring prior authorisation.

3.2 Reporting

The Directive makes it compulsory to report "for each person who or undertaking which carries out the practices ...". The Directive does not say which administrative procedure will be applied subsequently.

The requirement of reporting is nevertheless the starting point of any administrative procedures submitting individual practices to regulatory control. Practices requiring prior authorisation are a fortiori bound to be reported at the stage of planning. In general, however, reporting does not necessarily imply advance notification. Since the requirement applies not only to the use of radioactive substances, but also to their production and placing on the market, the administrative structure may be set up in such a way that the authorities are nevertheless aware of the planned use of a source before initiation of the practice. The practical arrangements, e.g. content, form, timeliness of reporting will be specified in national legislation. The information provided should allow the authorities to verify the practice at any moment. In some cases an assessment, simple or elaborate, will be required before an authorisation is granted.

3.3 Exemption from the requirement of reporting

3.3.1 Exempt practices

In the proposed Basic Safety Standards Directive, exemption from the requirement applies only to the use of radioactive substances and their subsequent disposal.

The aim of exempting the "use" is to avoid overwhelming the authorities with a large number of small users reporting the holding of sources with a low amount or concentration of radioactivity. The processing, production, and placing on the market of the same sources will need to be reported or be subject to prior authorisation.

Exemption, in these terms, should be distinguished from the release of radioactive substances already under the system of prior authorisation. An example is the release of steel, contaminated to very low levels, into the general scrap pool following the decommissioning of a nuclear site. In such cases it is better to speak in terms of "release authorisations" or "clearance levels".

3.3.2 Radioactive substances

The definition of "radioactive substances" does not in itself exclude any low levels of radioactivity.

Hence there is never exemption from regulatory control as such, since the overall system of radiation protection, which applies to practices rather than to individual sources, continues to apply. Regulatory authorities should e.g. satisfy themselves that the uses of radioactivity are justified.

Despite the fact that the "or" logic between exemption on the basis of quantities involved or concentration values is maintained, the concept of "small user" introduced in the previous paragraph is particularly important with regard to setting exempt concentration values. It is understood that exemption on this basis can only be granted if "moderate" amounts (of the order of a tonne) are involved.

The exemption of radioactive substances from the requirement of reporting is based solely on its radioactivity content (in terms of total activity or activity concentration) with disregard of the physico-chemical form of the substances.

3.3.3 Apparatus containing sealed sources

The use of apparatus containing radioactive substances beyond the exemption quantities and concentration values can be exempted from reporting provided it is of a type approved by the competent authority. The approval must ensure that the structure of the source guarantees effective protection against any contact with the radioactive substances and against their leakage or dispersion into the environment. Even though it is understood that this protection should be guaranteed under "normal conditions of use", the authorities will surely take into account the possibility of misuse of the apparatus or of accidental dispersion.

The protection against leakage or dispersion will not be secured indefinitely in case of disposal. The type approval of the source may indicate e.g. whether the source may be disposed of without precautions or will need to be returned to the producer. In general therefore exemption applies only to the use of sealed sources and not to their subsequent disposal, as opposed to radioactive substances below the exemption values.

4. Criteria for exemption

4.1 Risk considerations

This criterion applies to the possible exposure of an individual upon normal use of the source. Higher doses could result from deliberate or accidental misuse of the source, but in view of the overall risk concept one may assign a probability function to such situations. According to the concept of "potential exposures" introduced by ICRP, this can be taken into account rightaway by multiplying doses with their probability of occurrence.

As mentioned before, a further consideration in the setting of exemption levels is on the collective dose. As long as the individual doses are as low as 10 μSv, however, the collective dose will be orders of magnitude smaller than 1 man.Sv, unless the practice is liable to affect a substantial fraction of the population. This would be contrary to the idea of "small user". In the Euratom Basic Safety Standards, exemption is granted rather on the basis of the superfluous character of administrative controls, in line with the ICRP recommendations (see par. 2.3).

It was found that even with an effective dose criterion of 10 μSv significant skin exposure could not be disregarded. Thus a further criterion of 50 mSv to the skin (averaged over 1 cm^2) has been introduced. This equals the dose limit for members of the public, which is well below the threshold for deterministic effects.

4.2 Scenarios

The use and subsequent disposal of radioactive substances cannot be exempted directly on the basis of the radiological criteria. Operational quantities are defined as in the past for (a) the total activity (Bq) and for (b) the activity concentration (Bq/g) of the radioactive substances involved in the practice.

The choice of scenarios for calculating doses for comparison with the dose criteria is a crucial point. They need to be generic, i.e. allow for any possible situation, and still conserve some degree of realism. This is very difficult to achieve in a hypothetical environment.

This is particularly true for the potential exposures. The introduction of probability factors however allows to incorporate these adequately, rather than dismissing such scenarios as being unrealistic. The probability factors which have been introduced are either 1 (normal situation) or 10^{-2} (accidental situation). This guarantees that even in the "worst case" actual doses will be below 1 mSv i.e. the dose limit to *the public and the lower boundary to classification as exposed* worker.

The scenarios pertaining to concentrations of activity per unit mass are particularly difficult to manage: the mass of materials containing radioactive substances can in principle be infinite. This is obviously unrealistic and adequate geometries of the material must be considered for external exposure.

These difficulties imply that the choice of scenarios should grow on the basis of consensus rather than being a strict scientific issue. The dose criteria and the degree of conservatism built into the scenarios may indeed compensate each other to some extent. The methodology which was developed in order to define the exemption levels in the Euratom Basic Safety Standards (7), is the result of thorough collaboration between different institutes and has been extensively debated by the Art. 31 Group of Experts. This interaction ensures that the resulting values are fairly robust. The occurrence of significant exposures as a result of the use of radioactive substances below the exempt activity or concentration values is therefore precluded.

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INTERNATIONAL EXEMPTION PRINCIPLES AND THEIR APPLICATION — SOME ISSUES

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Abstract

Since the issue of IAEA Safety Series No. 89, "Principles for the Exemption of Radiation Sources and Practices from Regulatory Control" in 1988, experience has been gained in the application of the principles to practical problems. One result is that issues requiring further clarification and interpretation have been raised. Also, during this period the International Commission on Radiological Protection has issued new recommendations, some parts of which are relevant to exemption. This paper reviews some of the issues relevant to exemption principles which have emerged as a result of these developments.

1. *Introduction*

Radiological principles for exempting sources and practices from regulatory control were established in the course of a series of international meetings in 1987 and 1988 and published in IAEA Safety Series No. 89 (1). Since Safety Series No. 89 was issued there have been changes in the recommendations of the International Commission on Radiological Protection (2) of relevance to the exemption principles. The process of developing derived exempt quantities based on the principles has raised certain issues on which clarification has been needed and the incorporation of exemption principles into international regulatory texts has also necessitated some additional interpretation.

In this paper the international exemption principles are briefly summarised and then a number of the issues which have arisen since their publication are discussed.

2. *Principles for Exemption*

In IAEA Safety Series No. 89, two basic criteria are specified for determining, from a radiation standpoint, whether or not a source can be exempted from regulatory control (1)

- individual risks must be sufficiently low as not to warrant regulatory concern; and

- radiation protection, including the cost of regulatory control, must be optimised.

The individual risk is addressed by defining a level of individual dose[1] that can be regarded as "trivial": firstly, to choose a level of risk and the corresponding dose which is of no significance to individuals; secondly, to use the exposure to the natural background, to the extent that it is normal and unavoidable, as a relevant reference level.

After evaluating these approaches, it was concluded that for the purpose of exemption, a level of individual dose of some tens of microsieverts in a year could reasonably be regarded as trivial by competent authorities (1).

Because an individual may be exposed to radiation doses from several exempted practices, it is necessary to ensure that the total dose does not rise above the trivial dose level. Safety Series No. 89 therefore recommends that each exempt practice should contribute only a part of the identified 'trivial dose'. The apportionment suggested in Safety Series No. 89 could lead to individual does to average members of the critical group in order of 10 μSv in a year from each exempt practice(1).

In relation to the optimisation of protection, Safety Series No. 89 recommends that each practice should be assessed as if it were to be subjected to a formal optimisation procedure. A study of the available options (including various kinds of regulatory action) should be made and the conclusion reached that exemption is the option that optimises radiation protection. If, however, a preliminary analysis shows that the practice gives a collective dose commitment of less than about 1 man-Sv per year of practice, then the total detriment is low enough to permit exemption without more detailed examination of other options.

3. *Issues Relevant to the Exemption Principles*

3.1 *Individual Dose Criteria*

In Safety Series No. 89 two approaches were used for deciding upon a trivial risk or dose level. Firstly, a level of risk was chosen, and the corresponding dose, which is of no significance to individuals. Secondly, the exposure to natural background radiation was used, to the extent that it is normal and unavoidable, as a relevant reference level.

It was noted that most authors proposing values of trivial dose have set the level of annual risk of death which is held to be of no concern to the individual the at 10^{-6} to 10^{-7}. Taking a rounded risk factor of 10^{-2} Sv^{-1} for whole body exposure as a broad average over age and sex, the level of trivial individual dose would be in the range of 10 - 100 μSv per year.

The reviews during the late 1980's of the dose/risk relation were summarised in ICRP Publication No. 60 (2). A new rounded dose risk factor for fatal cancer of about 5 x 10^{-2} Sv^{-1} was recommended. The level of trivial individual

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[1]The term 'dose' refers to the sum of the effective dose from external exposure in a given period and the committed effective dose from radionuclides taken into the body in the same period.

dose, revised on this basis, would be in the range 2 - 20 μSv per year. It might therefore be argued that the conclusions of Safety Series No. 89, that "an individual radiation dose, regardless of its origin is likely to be regarded as trivial if it is of the order of some tens of microsieverts per year" should be revised accordingly and that the partitioning of the trivial dose to "10 μSv in a year from each exempt practice" might also be reconsidered. However, rationale for trivial dose in Safety Series No. 89 also considered arguments based on natural background radiation. It was suggested that a level of dose which is small in comparison to the variation in natural background radiation can be regarded as trivial (one to a few per cent ie 20 - 100 μSv per year). This argument remains unaffected by the changes in the risk/dose relationship.

The issue has recently been considered by the US NCRP in the context of the review of its definition of "negligible individual dose"(3). The revision in the dose/risk relationship was considered along with the other reasons originally considered in defining the negligible individual dose, including "difficulty in detection and measurement of dose and health effects" and "estimated risk for the mean and variance of natural background radiation exposure levels". The NCRP concluded that their original recommendation for a negligible individual dose of 10 μSv per source or practice should be retained.

3.2 *Definition of Practice*

In Safety Series No. 89, a definition is given of practice as

'a set of co-ordinated and continuing activities involving radiation exposure which are aimed at a given purpose, or the combination of a number of similar such sets.'

The need to define 'practice' with some degree of accuracy in the context of exemption comes from the recommended approach to optimisation of protection in which the need for further consideration of optimisation can be eliminated if it can be shown that the practice gives a collective dose commitment of less than about 1 man . Sv per year of practice (see Section 2).

Defining "practice" in a consistent way in all of the different contexts in which exemption principles have been applied has proved to be difficult. The way in which practice is defined affects the size of the collective dose. One example is from the area of recycle, in which the practice could be defined as the material coming from the decommissioning of a single nuclear power plant, from all of the nuclear power plants in a region, or from all of the nuclear power plants in a country. Clearly the collective dose would increase with each successive definition. There are similar examples which can be taken from other areas of radiation protection and waste management.

While the issue of practice definition is not resolved, most assessment studies conducted to date on the exemption of sources, wastes and recycled materials have shown collective doses to be small when individual doses are kept below 10 μSv per year. In these studies various different assumptions have been made relating to the nature and size of the practice considered.

3.3. *Probabilistic Scenarios*

A concern of the experts who prepared Safety Series No. 89 was that exemption should not be used for large and potentially hazardous sources which give rise to trivial radiation dose rates only because of the level of protection provided, eg. shielding or filtration. It was perhaps for this reason that the statement "Exemption must not be granted if there is a possibility of scenarios leading to doses in excess of those specified in granting the exemption" (Section 5.3) was included.

The statement is misleading since even for comparatively innocuous sources of radiation, scenarios will exist which could give doses greater than the individual exempt dose criterion.

Another statement later in the text gives a more realistic view. "The national authorities will have to exercise judgement in considering situations, associated with a low probability of occurrence, in which the chosen radiological protection criteria may be exceeded". (Section 6.1)

In making judgements on the exemption of sources it is clear that the potential for large radiation doses, even with low probability of occurrence, should not be allowed. One approach is to select some dose level say, at an order of magnitude higher than the exemption criterion, which could serve as a constraint for the scenarios judged to be unlikely. ICRP has proposed a more formalised approach to this problem, in the context of its advice on potential exposures, in which the product of the expected dose and its probability of occurrence may be used as if this were a dose that is certain to occur (2). However, the problem of determining the probability of occurrence means that, in this approach too, a strong element of judgement is needed.

3.4 *Unconditional and Conditional Exemption*

The concept exemption from regulatory control implies a removal of restrictions so that the exempted materials can be treated without any consideration of their radiological properties. However, the removal of restrictions may not always be complete; there is also the possibility to exempt material under specified conditions.

The full and complete exemption of a source requires that all reasonably possible exposure routes are examined and taken into account in the derivation of the exemption levels, irrespective of how that material is used and to where it may be directed. Such exemptions are called "unconditional exemptions".

Considerations in deriving unconditional exemption levels

As discussed, the derivation of unconditional exemption levels must take into account radiation exposure during all of the reasonably possible uses and movements of the material intended for exemption. For a given radionuclide, the derived quantity will be determined by the scenario and exposure pathway which give rise to the highest radiation dose. Since the value of the exemption level must be acceptable everywhere, it must be based on consideration of generic scenarios and data. In order to be sure that the derived values are widely applicable it will usually be necessary to err on the conservative side of the range of observed data in the choice of assumptions and parameter values. For these reasons, the values derived for use in unconditional exemptions will tend to be conservative, that is, in most cases the actual doses received will be well below the individual dose criterion.

The individual dose criterion of 10 μSv/a is appropriate for use in cases where there is no or insufficient knowledge about other exposures of the critical group.

Considerations in deriving conditional exemption levels

When the practice which is a candidate for exemption is well defined, such as the use of a device containing a specific type of source, or disposal of a particular waste type to a landfill, it will usually be possible to take account of the known features of the practice. The likelihood of critical group exposure due to overlapping practices may be taken into account in interpreting the recommendations of Safety Series No. 89. If it is clear that the likelihood of accumulating doses from more than one exempted practice is small then a more liberal apportionment of the 'trivial dose' (a few tens of microsieverts) may be considered. Also, in deriving the exemption levels, if there is good knowledge of the practice being considered, it may be possible to limit the number of exposure scenarios which need to be considered and to introduce practice specific data into the dose calculation.

These considerations may be expected, in general, to lead to higher exemption levels as compared to the unconditional exemption case. The values will, of course, be conditional on certain assurances being given over the fate or use of the radioactive material. If the assurances cannot be given or guaranteed then values of the exemption levels determined by unconditional exemption considerations will be more appropriate.

3.5 *Different Exemption Situations*

Recently, in work related to incorporating guidance on exemption principles into international regulatory documents it has been found to be convenient to identify two types of situations:

1) radiation sources which never enter the regulatory control regime, that is control is not imposed, and
2) radiation sources which are released from regulatory control, that is, control is removed.

The first situation typically includes the small sources of radiation such as tracers used in research, calibration sources and some consumer products containing small sources or low levels of activity per unit mass which are not normally regulated. The corresponding levels of activity levels of activity or activity concentration are being called exemption levels (sometimes also reporting levels) in the international regulatory documents (4,5). The second situation includes materials and scrap for recycle and wastes containing low levels of radioactivity from within the nuclear fuel cycle or from other regulated facilities such as hospitals, research laboratories and industry. When regulatory controls are removed, materials are said to be "cleared" from regulatory control (4,5).

Because of the larger volumes and surface areas of the sources considered in the context of "clearance", the exposure scenarios considered in deriving clearance levels tend to give higher radiation doses for the same activity concentration as compared with the exemption case. Clearance levels will, therefore, usually be lower than exemption levels for the same radionuclide.

References

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EXEMPTION IN THE CONTEXT OF THE IAEA TRANSPORT REGULATIONS

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Abstract

This paper highlights the issue of exemption within the framework of parallel Agency activities to revise both the Transport Regulations and the Basic Safety Standards. Specifically, the paper examines the current regulatory approach to exemption as contained in the 1985 Edition of the Regulations. It then discusses, with respect to exemption, the progress made to date in the revision of the Regulations, which are due for publication in 1996. Finally, the paper draws some conclusions from a review of the exemption values and supporting text that are presented in the fourth draft of the Basic Safety Standards.

The Transport Regulations call upon the Basic Safety Standards as a general provision for radiological protection, and exemption is a major aspect of the Standards. It is therefore important that the two publications are consistent with each other. There are recognized benefits in using the same values for exemption in transport as in other practices. However, adopting the values for exemption presented in the draft Standards will be a major change to the Regulations. Nevertheless, there is a general willingness to keep the Regulations abreast of developments in radiological protection and in line with any international consensus on exemption principles and values.

The paper calls for more information to be made available on the exposure scenarios and methods used to derive the exemption values. It also recognizes the importance of easy implementation for both operators and regulators. Indeed, the unique nature of transport as a practice may justify a different method of implementing the exemption values that are presented in Appendix II of the draft Standards. The paper explores the possibility of adopting a relatively simple method of implementation and draws conclusions which are mainly favourable.

Any changes introduced to the Regulations will have a greater impact on some transport operations than others, but this will only become known when the exemption values, together with the methods of implementation, are given wider review during the forthcoming months.

1. Introduction

101. To keep the IAEA's Transport Regulations[1] abreast of new scientific and technological developments, a major review of the Regulations takes place approximately every ten years. This time interval allows international organizations and Member States to schedule rule making activities regarding the new Regulations. The next Edition of the Regulations is due to be published in 1996.

102. One of the major topics being considered within the revision process is the incorporation of the new Basic Safety Standards[2], which embrace the subject of exemption. The revision process for the Regulations

is an activity that is taking place in parallel with the review of the Basic Safety Standards. However, it will be important for the Standards to be published first because the Regulations call upon the Standards as an important general provision for radiological protection. The planned completion date of 1994 for the Basic Safety Standards is timely in view of the plans to produce a 1996 Edition of the Regulations.

2. Exemption and Exclusion in the 1985 Edition of the Regulations

201. This Section comments on the way in which the concepts of exemption and exclusion are treated in the current Regulations. The key paragraphs in respect of exemption and exclusion can all be found in the Introduction to the Regulations. They are paras 104, 122, 139 and 143. For ease of reference they are reproduced in Annex I of this paper.

202. For the purposes of the 1985 Edition of the Regulations, para. 139 defines radioactive material as any material having a specific activity greater than 70 kBq/kg. Specific activity of the material is defined later in para. 143 as the mass activity concentration in which the activity is essentially uniformly distributed throughout the material. It should be noted that the value 70 kBq/kg has not changed since the first edition of the Regulations in the 1960's. This definition of radioactive material has been interpreted as an exemption value. In principle, material with lower mass activity concentrations could be regulated, but national competent authorities have considered regulatory controls for the transport of such material to be unnecessary.

203. It should be noted that the Regulations are declared, in para. 104, as not applying:

(a) within establishments where the radioactive material is produced, used, or stored other than in the course of transport, and in respect of which other appropriate safety regulations are in force, or

(b) to human beings who have been implanted with radioisotopic cardiac pacemakers or other devices, or who have been treated with radiopharmaceutical.

This paragraph of the Regulations excludes certain practices from the Transport Regulations.

204. In passing, another aspect of exemption is surface contamination on packages, overpacks, conveyances and their equipment. Paragraph 122 defines contamination as the presence of a radioactive substance on a surface in quantities in excess of 0.4 Bq cm^{-2} for beta and gamma emitters and low toxicity alpha emitters or 0.04 Bq cm^{-2} for all other alpha emitters. Low toxicity alpha emitters are: natural uranium; depleted uranium; natural thorium; ^{235}U or ^{238}U; ^{232}Th; ^{228}Th and ^{230}Th when contained in ores or physical or chemical concentrates; or alpha emitters with a half-life of less than 10 days.

3. Relevant Consultant Service Meetings held previously by the IAEA

301. On the recommendation of Director General's Standing Advisory Group on the Safe Transport of Radioactive Material, the Agency has convened two Consultant Service meetings dedicated to exclusion and exemption in the context of transport. The following text draws heavily upon the unpublished

reports of those meetings. It is therefore important to acknowledge the following individuals;

Ms. C. Fasten	Germany	
Mr. G. Kafka	Austria	(first meeting only)
Mr. K. B. Shaw	UK	(second meeting only)
Mr. L. Sztanyik	Hungary	

302. The latter of the two meetings took place in November 1992. At that time the values for exemption were 'in the melting pot' and the second draft of the Basic Safety Standards contained values which were clearly subject to review and revision. This review paved the way for the values contained in Appendix II of the fourth draft of the Basic Safety Standards. Despite this changing background, many of the observations and recommendations made by the Consultants remain valid. The participation of Mr. Sztanyik ensured that the international consensus on exemption principles as published in Safety Series No. 89[3] was given proper consideration. Mr. Sztanyik was a major contributor to that publication.

303. There is an awareness of the benefits of harmonization between regulations. In particular the importance of consistency between the Basic Safety Standards and the Transport Regulations should be stressed. The Consultants recommended that exemption values for transport be as consistent as possible with requirements of other practices. This would be aided if the derivation of general values for exemption took account of transport situations, including the potential for accidents during transport. There was an expectation among the Consultants that transport operations do not provide more limiting scenarios than those already considered in the derivation of exemption values. Nevertheless, it is important to check this supposition.

304. The Consultants noted that when radionuclide specific calculations of derived values are made, the values usually span several orders of magnitude. Furthermore, it was predicted that exemption values derived from reasonable exposure scenarios using the international consensus on exemption principles would span the existing threshold 70 kBq/kg used to define radioactive material for the purposes of the Regulations. The Consultants forecast that some radionuclides, such as ^{60}Co, ^{22}Na, ^{24}Na and ^{232}Th might have very low exemption values. This might lead to a large overall increase in the number of packages coming under the scope of the Regulations. The focus of the concern being that increased costs, disproportionate to any risks averted, could be incurred. On the other hand, concern was expressed that some radionuclides might have rather high exemption values, in particular, the noble gases, ^{85}Kr and ^{133}Xe. The ultimate concern being an illogical outcome whereby packagings requiring competent authority approval are needed to transport exempted material. The Consultants recommended that these concerns should be investigated when the exemption values next appeared in the draft Basic Safety Standards.

305. The Consultants made a number of observations during discussions that are worth noting:

(a) Exemption from the Regulations, rather than exemption from the requirements of notification, registration or licensing, is the preferred regulatory option.

(b) The exemption values can be expressed as total activity, activity concentration or the surface contamination below which the presence of radioactive material is considered to be of no regulatory concern.

(c) Activity concentration should over-ride the total activity, if the activity concentration is sufficiently low. This is because of self-shielding by the material and the impossibility of inhaling or ingesting a radiologically significant quantity of such material.

(d) Similarly, total activity should over-ride the activity concentration, if the total activity is sufficiently low;

(e) Any exemption values should be straightforward to apply both for operators and regulators.

(f) Due consideration will need to be given to how the exemption values apply to packages, consignments of multiple numbers of packages and multiple consignments of packages on board conveyances.

4. The Revision Process to date

401. Within the frame of revising the Regulations, a Revision Panel instructs the Secretariat on the drafting of the revised Regulations and their supporting documents[4],[5],[6]. Four meetings of the Revision Panel are planned leading to 1996. To date, the Revision Panel has met twice and instructed the Secretariat to produce the first draft 1996 Edition of the Regulations. Only topics which are sufficiently mature are tabled at meetings of the Revision Panel. In the case of major changes to the Regulations, this requires the preparation of detailed proposals for amendment that have been debated previously. The second Revision Panel met in May 1993, and at that time it was impossible to provide detailed proposals for amendment concerning exemption owing to the absence of definitive values and text for exemption in the Basic Safety Standards. It is hoped that the Standards, especially Appendix II, will be finalized before June 1994 when the final meeting of the Technical Committee on the impact of the revised Basic Safety Standards on the Regulations is planned. To promote good discussions, in June, on how to incorporate exemption into the Regulations, it will be important that the values in Appendix II are supplemented by explanatory text describing the criteria, exposure scenarios and method of modelling. A detailed set of proposed amendments should emerge from these discussions for tabling at the third Revision Panel in October 1994.

402. Under direction from the Revision Panel, the first draft of the 1996 Edition of the Regulations carries amendments to both paras 104 and 139 that indicate the extent of progress that has been made on the subject of exemption. A revised definition of radioactive material has been introduced in which radioactive material shall mean any material which spontaneously emits ionizing radiation. Thus the definition of radioactive material loses the function of exempting certain radioactive material. However the concept of exemption as a function of specific activity is moved to para. 104. With regard to the changes proposed to para. 104, the draft is very tentative, but gives some insight into the likely approach towards exemption;

(a) Consumer product items, that have received regulatory approval, following their sale to the end user are added to the list under which the Regulations do not apply.

(b) The single exemption value of 70 kBq/kg is indicated to be under pressure to change.

(c) The total quantity of material is cited as a condition for exemption, but as yet no values are given.

(d) A table of radionuclide specific values (to be developed) is cited.

403. No substantive change to para. 122 has been recommended by the Revision Panel. However a radionuclide specific approach towards the derivation of contamination limits has been rejected on the grounds that it would be impracticable and unnecessary.

5. The fourth draft of the Basic Safety Standards

501. In July 1993, the fourth draft of the Standards emerged. Appendix II of the draft contains text on exemptions followed by a tabulation of exempt activity (Bq) and exempt activity concentrations (Bq/g) for a wide selection of radionuclides. At the time of writing, no Agency meeting has tabled exemption in the context of transport since the appearance of the fourth draft. The opinions expressed in the remainder of this paper are those held by the authors, they are not positions taken by the Secretariat under advice from transport experts representing Member States.

5.1. Remarks on the text of Appendix II of the draft Standards

502. The text of Appendix II adequately explains the exemption criteria, but gives no information on either the exposure scenarios or the method of modelling. While it may be inappropriate to detail the methods used, reference to available source material should be provided as a minimum. Even though transport operations are not expected to provide more limiting exposure scenarios, the current draft precludes a proper analysis.

503. Appendix II advises that sources within practices are automatically exempted from the requirements for notification, registration or licensing where the total activity used in the practice in a year and the activity concentration do not exceed the exemption levels given in the tabulation. The Standards provide definitions for several of the terms used in this advice which are reproduced in Annex II to this paper. At the moment it is only possible to speculate on how the advice will be applied to transport. However, recalling the recommendations made by the Consultants for the exemption values to be as consistent as possible with the exemption values for other practices and straightforward to apply some ideas can be explored.

504. To be of practical value to transport operations, the advice contained in Appendix II will need to be interpreted. For example, the total activity used, or transported, by the practice in a year is not a useful quantity in the context of exemption in transport. At its simplest;

(a) transport is a practice;
(b) a single package is a unit source;
(c) exemption means exemption from the Transport Regulations; and
(d) exemption values of total activity or activity concentration apply to the contents of a single package or the load on a conveyance in the case of unpackaged material.

The problem with this approach is the extent of the deviation from the advice given in the Standards and the potential for exempting accumulated quantities of radioactivity in consignments of multiple packages aboard conveyances. Alternative methods of implementation could lead to a closer alignment with the advice given in Appendix II, but these will be less straightforward for operators and regulators. A further deviation from the advice given in Appendix II may be the exemption of packages containing more than an exempt total activity in cases where the activity concentration is less than the exemption level. Similarly, there may be exemption from the activity concentration provided that the total activity limit is not exceeded. Lastly, no exemption values for surface contamination are provided in the Standards, whereas the Regulations for transport are expected to retain the current definition of surface contamination in para. 122.

505. The concept of conditional exemptions being granted subject to conditions specified by regulatory authorities, i.e., competent authorities in the context of transport, may prove to be helpful. Conditional exemptions citing physical or chemical form of the radioactive material, or restricting

the scope of the exemption solely to transport might provide an appropriate degree of regulatory flexibility to cover special circumstances.

5.2. Remarks on the values in Appendix II of the draft Standards.

506. The numbers in Appendix II need to be assessed as administrative controls for exemption in the context of transport. In this exercise the most important aspects are whether the values provide sufficient protection and whether they are practicable.

5.2.1. Exemption values for activity concentration (Bq/g)

507. The exemption values for activity concentration range from 10^{-1} to 10^{6} Bq/g. This range can be compared with the current value of 70 Bq/g which is used in the definition of radioactive material in para. 139 of the Regulations. There is potential for an increase in the number of radioactive shipments for some radionuclides of interest. For example, the exempt activity concentration for ^{241}Am, ^{239}Pu and ^{232}Th is only 1 Bq/g, while for ^{60}Co, $^{22\&24}$Na and ^{235}U the value is 10 Bq/g. The potential increase in radioactive shipments will be offset by more relaxed exempt activity concentrations for other radionuclides. For example the value for ^{3}H is 10^{6} Bq/g, for ^{14}C it is 10^{4} Bq/g, and for ^{51}Cr, ^{125}I and ^{32}P it is 1000 Bq/g. There may be a reluctance on the part of some competent authorities to a 'relaxation' of the Regulations.

508. A range of exempt activity concentrations will be more complicated to apply than a single value, but the problems should not be insurmountable. A rule for application to mixtures of radionuclides can be formulated and a procedure established for determining the exempt activity concentration in the event that radionuclides are not listed. As a preliminary conclusion, the exemption values for activity concentration appear to be practicable. The range of values is wide, but this must be expected of a radionuclide specific approach, given the variation in radiotoxicity. Clearly, the changes introduced by Appendix II will have a greater impact on some practices than others, but this will only become known when the exemption values, together with the method of their implementation, are given a wider review.

5.2.2 Exemption values for activity (Bq)

509. The exemption values for activity range from 10^{2} to 10^{9} Bq. Within the Regulations for transport there are no numbers for exemption for direct comparison. However, the Regulations do incorporate radionuclide specific quantities as content limits in various types of package. It is useful to compare the exemption values with package content limits to determine whether any anomalies arise. This exercise results in the logical outcome that exempted quantities can be transported in untested and non-approved packages for two reasons;

(a) exempt levels are very much less than quantities requiring packaging that is designed to withstand accidents and needs competent authority design approval; and

(b) exempt levels are below the quantities requiring packaging that is tested to withstand incidents associated with normal transport.

5010. For many radionuclides, a very large number of exempt quantities would have to accumulate in a consignment such that, had the total activity been declared to be contained in a single package, the package design would have been required to withstand tests to simulate the stresses of normal transport. However, for other radionuclides, such as ^{239}Pu and ^{241}Am, only about twenty exempt quantities would need to accumulate before a tested package would be required. The conundrum of radiologically significant quantities being divided into small payload packages that meet the requirements for their individual content, rather than the total content of the consignment, is not newly created by the introduction of exempt quantities. For example, the Regulations do not prescribe restrictions on the number of Type A packages transported on a conveyance. In theory, it becomes possible for quantities to accumulate to the extent that a Type B package would be required if the radioactive contents were contained within a single package.

5011. As for the values for activity concentrations, a rule for application to mixtures of radionuclides should be formulated and a procedure established for determining the exempt quantity in the event that radionuclides are not listed.

5.3 Clearance levels

5012. The concept of clearance values has been introduced to the fourth draft of the Standards. It is understood that material and objects already subject regulatory control may be released without restriction subject to satisfying clearance levels. Furthermore, the logic that clearance levels will not exceed exemption values is acknowledged. Currently, it is difficult to see an interaction between clearance levels and the Transport Regulations unless, for any reason, lower values for exemption are unilaterally adopted for transport. Unrestricted clearance must embrace transport and the decisions will be taken at sites where the material and objects are produced or used. Clearance levels are not likely to be a matter for transport authorities.

6. Conclusions

601. The revision of the Regulations and the Basic Safety Standards are parallel Agency activities. It is important that the two publications are consistent with each other and that the Standards are published first. The Regulations call upon the Standards as a general provision for radiological protection and exemption is an important aspect of the Standards. There is a recognized benefit in using the same values for exemption in transport as in other practices. To this end, it is encouraging that the international consensus on exemption principles has been carried through to the draft Standards.

602. Consensus on the exemption values within the 'transport community' will be more easily achieved if information on the exposure scenarios and method of modelling is provided. With this detail, it is possible to check the supposition that transport is not a limiting scenario in the derivation of exemption values.

603. Another important aspect is the ease of implementation for both operators and regulators. In this regard, there may be a need for a different implementation of the exemption values in the case of transport. In particular, the exemption value may be best applied to the contents of a single package as being exempt from the Transport Regulations as a whole. Arguments to justify this approach will have to be carefully constructed in cases where multiple exempt quantities could accumulate in consignments. Again, an analysis of the exposure scenarios and modelling methods may be useful in the construction of such arguments.

604. This paper raises the question of whether the exemption values for activity concentration and for total activity must both be satisfied before exemption can be granted? At present the advice given in Appendix II of the draft Standards indicates that both exemption conditions have to be met. This may be unnecessarily onerous, at least in the context of transport and the matter warrants further attention.

605. Another aspect requiring further consideration is whether surface contamination belongs in the frame of exemption within the Standards? For the purposes of the Transport Regulations, a definition of contamination is likely to be retained. This will play a role in the exemption of very low-level contaminated objects from the requirements of the Regulations.

606. Adopting the exemption values contained in Appendix II of the draft Standards will be a major change to the Transport Regulations. It is important that the Regulations stay abreast of developments in the field of radiological protection. Therefore, there is a general willingness to be aligned with any international consensus on exemption principles and values. Clearly, any changes introduced to the Regulations will have a greater impact on some practices than others, but this will only become known when the exemption values, together with the method of their implementation, are given a wider review. This wider review will take place in the coming months under the auspices of the Agency's programmes of work on Standards, safe transport and waste management.

Annex I

Key paragraphs from the Regulations

Note: Paragraph 104 appears in the Introduction to the Regulations under the sub-title Purpose and Scope.

104. These Regulations do not apply:

(a) within establishments where the **radioactive material** is produced, used, or stored other than in the course of transport, and in respect of which other appropriate safety regulations are in force, or
(b) to human beings who have been implanted with cardiac pacemakers or other devices, or who have been treated with radiopharmaceuticals.

Note: Paragraphs 122, 139 and 143 appear in the Introduction to the Regulations under the sub-title Definitions for the Purposes of these Regulations.

122. **Contamination** shall mean the presence of a radioactive substance on a surface in quantities in excess of 0.4 Bq/cm^2 (10^{-5} $\mu Ci/cm^2$) for beta and gamma emitters and low toxicity alpha emitters or 0.04 Bq/cm^2 (10^{-6} $\mu Ci/cm^2$) for all other alpha emitters. Low toxicity alpha emitters are: **natural uranium**[1]; **depleted uranium**[2]; natural thorium; uranium-235 or uranium-238; thorium-228 and thorium 230 when contained in ores or physical concentrates; or alpha emitters with a half-life of less than 10 days.

139. **Radioactive material** shall mean any material having a **specific activity** greater than 70 kBq/kg (2 nCi/g).

143. **Specific activity** shall mean the activity of a radionuclide per unit mass of that nuclide. The **specific activity** of a material in which the radionuclide is essentially uniformly distributed is the activity per unit mass of the material.

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[1]The following definition of natural uranium is contained in para. 150; **natural uranium** shall mean chemically separated uranium containing the naturally occurring distribution of uranium isotopes (approximately 99.28% uranium-238, and 0.72% uranium-235 by mass).

[2]The following definition of depleted uranium is contained in para. 150; **depleted uranium** shall mean uranium containing a lesser mass percentage of uranium 235 than in **natural uranium**.

Annex II

Relevant defined terms from the glossary of the Standards

Source:

Any physical entity that may cause radiation exposure, e.g. by emitting ionizing radiation or releasing radioactive substances. Sources can be existing or introduced, or used as elements within a practice. For example, radon emitting material are sources existing in the environment, a sterilization gamma irradiation unit is a source for the practice of radiation preservation of food, an X ray unit may be a source for the practice of radiodiagnosis, and a nuclear power plant is a source for the practice of generating electricity by nuclear power. A complex or multiple installation situated in a same location or site is considered as a source for the purposes of the application of the standards.

Practice:

Any human activity that introduces additional sources of exposure or exposure pathways or extends exposure to additional people or modifies the network of pathways from existing sources, so as to increase the exposure or the likelihood of exposure of people, or the number of people exposed.

Notification:

A document submitted to the Regulatory Authority by a legal person to notify his intention to carry out a practice or any other action described in the General Obligations of the Standards when the normal exposures associated with the practice or action are unlikely to exceed a small fraction, specified by the Regulatory Authority, of the relevant limits and the likelihood and expected amount of potential exposures and any other detrimental consequences are insignificant. When the notification is received by the Regulatory Authority, no other action pursuant to the requirements of the Standards is warranted.

Registration:

A simplified form of authorization whereby the description of the proposed practice or action submitted by the applicant is registered by the Regulatory Authority and the practice or action is authorized without being accompanied by specific prescriptions or conditions.

Licensing:

An authorization which is granted by the Regulatory Authority on the basis of a safety assessment and accompanied by specific prescriptions and conditions to be fulfilled by the licensee.

Activity:

For an amount of radionuclide in a particular energy state at a given time, the activity, A, is:

$$A=\frac{dN}{dt}$$

where dN is the expectation value of the number of spontaneous nuclear transformations from that energy state in the time interval dt. The SI unit of activity is the reciprocal second, s-1, with the special name becquerel (Bq).

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EXEMPTION PRINCIPLES IN NATIONAL REGULATIONS

(Session II)

Chairman

L. BAEKELANDT
Belgium

APPLICATION OF EXEMPTION PRINCIPLES IN NATIONAL REGULATIONS

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Abstract

The use of consumer products containing radioactive substances, recycling and reuse of materials and items containing low levels of radioactivity, disposal of wastes by research laboratories and release of patients with residual radioactivity in their bodies are some practices exempted in India from the application of regulatory controls. The criteria for exemption include small scale of operations involving exempted sources, triviality of individual dose and collective dose, and avoiding wastage of effort and cost of control. However, the use of radioactivity in unjustified consumer products is prohibited and authorised products are assessed for safety. Compliance with tests prescribed in standards is mandatory. Recycling of materials and items containing radioactivity is regulated in accordance with prescribed levels of average concentration and surface contamination. Clearance levels and procedures for compliance are specified in regulations. The levels are based on assessment of dose to workers in recycling industry arising due to (a) external exposure from scrap piles, (b) external exposure from repair of reusable items, (c) dust inhalation in steel foundry/cement plant as well as dose to public, arising due to use of recycled steel in dwellings, furniture etc. Also exposure pathways in disposing of such items are also considered. Small quantities of radioactive wastes are permitted to be disposed of in public domain based on considerations of dilution, decay and exposure pathways.

1. Current Regulations

Radiation Protection Rules (RPR) issued in 1971 under the Atomic Energy Act, 1962 of the Government of India, exempts the import, export, transport, possession, sale and use of time pieces or other devices containing radioactive luminous compounds [1]. The intent may be to exempt any or all consumer products containing small quantities of radioactive materials but the only products containing radioactivity and used widely in the country at the time of framing the rules, were luminous painted dials in watches, clocks and instruments. The main consideration for exemption is the triviality of risk and the large scale of operations that would otherwise be needed for regulating the widespread use, misuse, disposal etc. of these products.

2. Revised Regulations

The RPR is currently undergoing revision in the light of experience gained in its enforcement, and also to take into consideration the current philosophy and concepts of radiation protection contained in ICRP-60 and related publications. In the intervening years, there has been several fold growth in production and use of radioactive sources and related scenario in the country. The regulatory system in the country is required to cope with large number of small sources. In order to ensure that regulatory emphasis is commensurate with the hazard potential of a source or a practice, the practical approach is to grade the system of controls in terms of hazard potential, individual and collective doses, potential exposures and intervention needs and scale in case of accidents. Extensive widespread sources with very low hazard potential do not require as much regulatory attention as for high consequence scenario associated with intense sources. Thus a graded system of regulatory control is being formalised in the new regulations. Sources and practices that contribute to less than 1 μSv per year to any individual and less than 1 man Sv collectively in a year from a practice are considered to be trivial because the risk associated with it is not recognisable in comparison with other risks commonly accepted by society as normal.

3. The Use and disposal of consumer products

The exemption provision applies to use of consumer products (ICSDs, luminous dials, GTLSs, Fluoroscent lamp starters, static eliminators, incandescent gas mantles etc.) A prerequisite for exempting the use and disposal of these items from the application of regulations, is that the items have to be evaluated and assessed in terms of inherent safety and compliance with standard specifications. The general guidelines for exempting consumer products containing radioactive substances were contained in a report prepared by the Task Group set up by the competent authority in 1990 [2]. The main recommendations of the Task Group are as follows :

3.1 The first step in establishing regulatory control of consumer products is to ensure adequacy of inherent safety in the design, manufacture, quality assurance and performance. The Task Group therefore recommended that AERB should issue Safety Standard Specifications stipulating the requirements for design, manufacture, quality assurance performance testing and labelling, so that the designers/ manufacturers and sellers will then be required to demonstrate compliance with the Standards. Competent Authority should ensure that only those items complying with the Standards will be approved for release to public and any substandard or unsafe products will be prohibited from supply.

3.2 The Task Group recommended that the users of consumer products should be exempted from the provisions of Licensing, provided that each product type is pre-assessed and approval granted by Competent Authority prior to manufacture and marketing. Such prior approval will ensure that only those products meeting the safety stipulations are released to the public.

3.3 The radiation safety of personnel involved in the manufacture and distribution of the consumer products should be controlled in the same manner as other radiation workers. The practice of granting NOCs to manufacturers for procuring radioactive materials and safety surveillance of work premises, work practices and personnel monitoring including internal dose assessment where relevant, should be continued. At present, the regulatory controls are limited only to the manufacture and it is desirable to extend the same to cover post-manufacture storage, and distribution.

3.4 The Task Group recommended that AERB should assess the impact of each practice in terms of maximum individual doses to members of critical groups as well as collective dose to users and non-users and decide on each practice by a formal cost-benefit analysis and optimization.

3.5 The Task Group recommended that products like ICSDs serve a safety function and hence should be classified as safety products. The dose limits in respect of safety products can be relaxed as compared with products not intended to serve any safety function. This is in line with the procedures recommended by IAEA and followed in many countries.

3.6 With regard to products already in the market with or without previous authorisation, the Task Group recommended appropriate procedures for post-facto approval of acceptable products and phased withdrawal of unacceptable products.

3.7 The Task Group recommended that each consumer product containing radioactive substance should be marked or labelled as appropriate so that the users become aware of the presence of radioactivity in the product and are advised to follow the procedures for safe use and safe disposal. The marking should also prevent the user from possible misuse and unacceptable disposal.

Based on the above guide lines, the standard specifications were issued [3] giving specific requirements for the design, construction and testing of each type of product. A notification under the existing provision of the rules to enable the competent authority to exempt the use, misuse and disposal of all pre-assessed products released in the market is being issued. Several brands of ICSDs, Fluoroscent lamp starters, Gas mantles etc. have so far been assessed and approved for supply to users. The manufacturers are required to inform the competent authority on an annual basis, total number of items released and the total radioactivity in them. This information is used to assess the radiological impact of introducing radioactive products in consumer domain.

4. Transport of Excepted Packages Containing Radioactive Material

The exemption principles are also applied in regulatory frame work for transport of small quantities of radioactive materials in excepted type of packages. The maximum activity content of excepted packages are specified in accordance with IAEA SS-6 [4]. For postal transport the total activity in each package is limited to only one tenth of the content allowed in excepted packages. Transport of empty packages is also treated as excepted packages. Regulations governing the transport of radioactive materials are issued as Radiation Surveillance Procedures under the Radiation Protection Rules.

5. Recycling of materials containing radioactivity

The steel scrap and concrete wastes from nuclear facilities and radiation installations when these are decommissioned and dismantled has reuse potential and economic value. These materials contain radioactivity as activation products in bulk materials or as surface contamination. Common activation products are Co^{60}, Fe^{55}, Mn^{54}, Ni^{63} etc. Tools, and items such as pumps, valves, motors, etc. with residual radioactivity on their surfaces after decontamination, may also be reused. Based on the same criteria for exemption of sources and practices from regulatory control, (i.e) 1 µSv / year individual dose and 1 man Sievert annual collective dose, and considering most probable scenario of recycling and end use of the materials, the clearance levels for release of the materials in public domain are worked out. Individual dose to workers involved in recycling of steel and concrete, and dose to members of the public living in buildings constructed with recycled materials, driving a car made with recycled steel, use of furniture made out of recycled steel and activities concerning the disposal of recycled items were considered in determining at the clearance levels. The clearance levels (Table 1) are specified in terms of mass activity concentration (Bq/kg) for beta-gamma nuclides, Uranium, Thorium and Plutonium in steel and concrete.

TABLE - 1

Clearance levels for release in the public domain

A Clearance levels for Mass activity concentration (Bq/kg)

Beta-Gamma (Steel and Concrete)	Uranium		Thorium		Plutonium	
	Steel	Concrete	Steel	Concrete	Steel	Concrete
4000	1500	15,000	100	1000	300	3000

Modifying factors on clearance levels

1)	Unrestricted release of ferrous scrap such as pipes, tools, etc. can be released to garbage, landfill, municipal dumps etc.)	x 0.1
2)	Release of ferrous scrap for melting without specific consent from the competent authority.	x 1
3)	Release of ferrous scrap for melting with specific consent from the competent authority (ferrous scrap to be released with a dilution of 10 at source)	x 10

Note : If beta-gamma and alpha activities are present, the rate of mixtures should be applied.

The modifying factors to the clearance levels provide for the release of ferrous scrap arising from a wide range of decommissioned facilities such as nuclear medicine laboratories or gas mantle manufacturing industries to the nuclear fuel cycle facilities.

Clearance levels (Table 2) are specified for surface activity concentration (Bq/cm^2) for the same radio-nuclides in steel and concrete [5].

TABLE - 2

Clearance levels for surface activity concentration (Bq/cm^2)

Beta-Gamma (Steel and Concrete)	Uranium		Thorium		Plutonium	
	Steel	Concrete	Steel	Concrete	Steel	Concrete
0.4	0.4	0.4	0.4	0.4	0.04	0.04

Note: a) The surface activity concentration should be averaged over 300 cm^2 area of the accessible surface and should be non-fixed.

b) If doubt exists for non-accessible surfaces, the activity should be assumed to be higher than the clearance level.

c) For fixed contamination, clearance levels for mass activity concentration should apply.

The exposure scenario considered for dose estimation consists of external dose to workers from scrap piles, pump repair and internal dose due to dust arising in various stages of recycling process.

Modifying factors are also prescribed for clearance levels for ferrous scrap for purposes other than for melting without specific consent from competent authority. The modifying factor for clearance levels is 0.1 for unrestricted release of ferrous scrap such as pipes, tools, etc. and a factor of 10 is specified if the scrap is remelted with a ten fold dilution.

6. Concluding Remarks

Exemption of sources used for invitro studies in biomedical field and certain tracer experiments are also exempted from regulatory controls. These are specified in Codes and Guides issued by AERB from time to time. These are formalised in the new regulations and a graded system of regulatory control is proposed.

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CHINA'S NATIONAL STANDARD ON EXEMPTION AND ITS APPLICATIONS

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Abstract

With the growth of the nuclear applications and with reaching decommissioning stage of the first generation of nuclear facilities of China, the application of exemption principles is more and more important. In early period, some exemption items was included in the National Basic Safety Standard (NBSS). Recently, a China's national standard on exemption was issued in 1992, mainly refering to some documents of IAEA and some other Organizations. The most basic criteria for generic exemption from notification, registration and licensing are: (a) the annual risk is not greater than $10^{-7}a^{-1}$ or (b) the annual effective dose equivalent to individuals of the critical group at no time exceeds 10 μSv and the collective effective dose equivalent commitment from the exempted source or practice shall not be greater than 1 man · Sv. This standard involved exemption for low level solid and liquid radioactive wastes and for recycling or reuse of materials. Another specified standard for controlling "consumer products" is to be published in near furture. Some issues related are mentioned.

1. INTRODUCTION

The first generation of nuclear facilities of China were built about thirty years ago, and most of them are reaching the end of their useful life. At the same time the applications of nuclear technology in various fields of economy and the nuclear power production are increasingly expanded in recent period.

With the gowth of the nuclear applications and the reaching of decommissioning stage of first generation of nuclear facilities, it was increasingly recognized that the application of exemption priciples is of great importance from the point of view of regulation and resources saving. Thus there was an increased need to consider the criteria for transferring very low level radioactive sources or practices from being fully controlled and licensed to being exempted.

Although there were some items related to exemption were included in several national standards on radiation protection issued in earlier period, a national standard specified for exemption application was just issued in 1992. Its title is "Principles for the exemption of radiation sources and practices from regulatory control (GB 13367—92)". In addition, the "Radiological health protection standard for controlling consumer products containing radioactive substances" is to be finished in the near furture.

2. MAIN CONTENTS OF NATIONAL STANDARD ON EXEMPTION

The China's national standand on exemption mainly includes the following parts.

2.1. PRINCIPLES FOR EXEMPTION

2.1.1 Any single practice or source, if a generic assessment, in its early stages, indicates that any one of the fallowing items is complied, the authorities may well decide to grant the exemtion directely:

(1) Its related risk of radiation induced lethal cancers and serious hereditary effects is not greater than $1\times10^{-7}a^{-1}$;

(2) The individual effective dose equivalent to the critical group shall not be greater than 10μSv in a year (500μSv to the skin), the collective dose commitment from the practice in a year shall not be greater than 1 man・Sv.

(3) Radiation machinery producing radiation of quantum energy not higher than 5keV;

(4) Radioactive substances in the form in which they occur in nature without preparation intended to increase the concentration of radioactive nuclides, except for uranium and thorium mine;

2.1.2 If simplified procedure can not indicate that doses are below the criteria above, some assessments will be required, permission for generic exemption can only be given after the assessment indicats the requirement listed in item 2.1.1 is complied. The assessment should be carried out using calculational models, which take account of both characteristics of the practice to be exempted and the characteristics of the sources involved in the practice, and all important exposure scenarios and pathways should be considered.

The selection of assessment models and necessary perfectness will be determined depending on particular situation of the practice so as to avoid the waste of man power and resouces.

2.2 EXEMPTION TO MULTI-PRACTICES

In cases where an individual may be exposed to radiation from more than one practice, the multi-practices can still be exempted if any single practice among them has been judged exempt.

2.3 REMARKS

The formulation of an exemption should not allow the circumvention of control that would otherwise be applicable, should not allow misuse and abuse of the principles for exemption, by such means as deliberate dilution of material of fractionation of the practice, the requirement given in 2.1.1 should be complied.

Table I EXEMPTION SPECIFIC ACTIVITY FOR SOLID RADIOACTIVE WASTES (Bq/g)

Radionuclide Group	Nuclides	Generic exempt range
High-Energy $\beta-\gamma$ Emitters	Na-22, Co-60, Cs-137	0.1−1.0
Mid-Energy $\beta-\gamma$ Emitters	Mn-54, Ru-106, I-131	$1.0-10^{2}$
Alpha Emitters	Pu-239, Am-241	0.1～1.0
Carbon	C-14	$10^{2}\sim10^{3}$
Short-Lived β Emitters	P-32, S-35, Ca-45	$10^{3}\sim10^{4}$

2.4 DERIVED "EXEMPT" LEVEL

2.4.1 Low level solid radioactive wastes

a) if the specific activity is below the lower boundry of the specific activity range given in table I, then it can be exempted from regulatory control directely; if the specific activity is within the range, then assessments using approriate models and parameters will be required, it can be exempted only after study shows that the requirement of 2.1.1 is complied. The exempted very low level solid radioactive wastes can be disposed at municipal landfills or incineration facilities.

b) To determine if a mixture of radionuclides is at or below an "exempt" specific activity, a simple ratio expression can be used. The expression is :

$$\sum_{i=1}^{n} \frac{C_i}{C_{Li}} \leqslant 1.0$$

where

C_i = the specific activity of radionuclide i in the waste, Bq/g

C_{Li} = the "exempt" specific activity of radionuclide i in wastes, Bq/g

n = number of radionuclides in the mixture.

2.4.2 Low level liquid radioactive wastes

a) The total activity discharged from each installation into the sewer per month shall not be greater than the ALI_{min} value; the discharged activity from each installation each time shall not be greater than 0.1 ALI_{min} and 10MBq (The ALI_{min} is the smaller one between annual limit of intake for inhalation and ingestion) .

b) Each installation should discharge their very low level liquid wastes at some fixed points, and should wash the discharge points after each discharge so as to avoid the accumulation of contamination. Some sign should be set up at such points.

c) If there are several radionuclides in the mixture, it should be controlled according to the following formulas:

The discharged activity per month should be complied:

$$\sum_{k} \frac{A_k}{ALI_{min,k}} \leqslant 1.0$$

The discharged activity each time should be complied:

$$\sum_{k} \frac{A_K}{ALI_{min,k}} \leqslant 0.1$$

where:

A_K——activity of radionuclide k;

$ALI_{min,k}$——ALI_{min} value for radionuclide k.

d) If a scintillation liquid is complied with both of the following points, it can be exempted from regulatory control:

i) there is no any α emitters in it;

ii) concentration of low energy β emitters is less than 10 Bq/ml, or there is only ^3H or ^{14}C and their concentration is less than 100 Bq/ml.

2.4.3 Recycling or reuse of materials

The steel from decommissioned facilities can be recycled, provided that the recycle can be carried out within the exemption requirements. The exemption limits are shown in table II and table III.

3. APPLICATIONS AND SOME RELATED ISSUES

In recent years refering to the international documents some efforts were made in China with the object of developing guildance on the principles for exemption of radiation sources and practices from regulatory control and on the application of the principles to practical problems. The most important application is for unrestricted release of materials or sites from decommissioned facilities, and the uncontrolled disposal of very low level radioactive waste.

Some response and issues are following:

Table Ⅱ EXEMPTION LIMITS FOR RECYLING OR REUSE STEEL

Radionuclide Group	Radionuclides	Exemption specific activity Bq/g
a) Long-lived activation products	Ni-59, Ni-63	$10^4 \sim 10^5$ 1)
b) Activation products	Mn-54, Co-60, Zn-65, Fe-55	1~10 1)
c) Fission products	Sr-90, Cs-134, Cs-137	1~10 1)
d) Uranium Plutoniucm	U-235, U-238, Pu-239	0.1~1.0 1)
e) Unknown β − γ emitters		1.0
f) Unknown α emitters		not recommended

1) If concentration is below the lower limit, can be exempted directely; if is within the range it can be exempted only after assessment using appropriate model and method indicates that requirements of 2.1.1 are complied. No single item may exceed 10 times of the average concentration in one t of steel.

Table Ⅲ EXEMPTION CONTAMINATION LEVEL FOR RECYLING AND REUSE STEEL AND EQUIPMENT

Type of emitters	Exemption surface contamination 1), Bq/cm^2
β、γ emitters	0.4
α emitters	0.04

1) Refer to the average value of surface contamination in any 300 cm^2 area of the surface.

a) The criterion of 10 μSv individual dose for generic exemption was considered to be unduly restrictive. Usually the regulatory bodies in China exempts a site depending upon the result of concret optmization analysis, not on generic criterion. For example, they exempted a decommissioned site depending on a residual activity level of uranium in soil which is corresponding to exposure to individual of public of about 24 μSv/a.

b) The criterion of collective dose was considered to be less or more ambiguous and difficult to judge. For example, how to choose the cut-off level of individual dose or the diameter of assessment area in the calculation of collective dose related to a candidate of exempt source or practice? Obviously, the collective dose depends on the scale at which "practice" is defined, e.g.for distribution of smoke detectors, the longer of producing time of the factory, the more of the products. How long period of production of the factory is considered?

c) There are some gaps even contradictories existed between the Standand on exemption and the related items in NBSS or standard for Classification of Radioactive Waste (For instance, exempted specific activity is 70 Bq/g in NBSS but the minium concentration which can be defined a solid waste is 7.4×10^{4}Bq/kg in the latter. Both of them look much higher than that given by the Standard on exemption, although there were some difference in their definitions) . Moreover, sometimes there are some difficulties to distincte the artificial sources and the enhanced exposures due to naturally occuring radionuclides.

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THE ROMANIAN SYSTEM OF EXEMPTION FROM REGULATORY CONTROL

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Abstract

The first regulation, which established an authorization system for the users of radioactive materials, was issued in Romania in 1962.

As a result of developing radionuclides utilization in industry, medicine and research, a complex regulatory system, based upon a law adopted by parliament, which was later completed with a set of norms binding all users of radioactive materials, was established in 1974-1976. These norms are still valid.

As part of this system, some criteria of exemption from authorization, especially those concerning the use of nuclear installations and radioactive sources, that are fulfilling a minimum of technical requirements, were settled.

The main criterion of exemption is that the exposure of an individual members of the critical group should not exceed the limit of exposure permitted for individual members of public (which is 5 mSv/year).

This paper presents the Romanian system of exemption from authorization. Application of this system is exemplified in the case of smoke detectors and electronic microscopes.

1. INTRODUCTION

The Romanian nuclear regulatory and authorization system has been developed in parallel with the spreading of activities using atomic energy in various fields of activity on one hand and with the increasing number of users, on the other hand.

Even from the beginning, the Romanian authorization system has shown some particular aspects, mainly generated by the centralized planning of economy -with no allowance for the use of atomic energy elsewhere than in governmental institutions and enterprises- and also because of some supplementary rights given to occupationally exposed workers.

In Romania, regulating and licensing are two functions of the same governmental institution, to whom we further address as the regulatory authority.

For quite a while, the regulatory authority was also in charge of nuclear research and development. Consumer products (products intended for individual use like TV sets) are licensed by the Ministry of Health.

As the number of radioactive sources and nuclear installations increased, the regulatory authority was forced to establish a discriminating authorization system, which stipulated some exemptions from authorization. This was done because the regulatory authority was insufficiently staffed, and couldn't handle all cases.

2. BRIEF HISTORY OF THE ROMANIAN AUTHORIZATION SYSTEM

The first official paper that established obligatory licensing for radioactive sources and nuclear installations was HCM 741 (HCM = a decree of the government at the time) issued in 1962 [1].

The authorization system was made effective by the Committee for Nuclear Energy, through the Commission for the Guidance and Control of Nuclear Units.

In order to put it into practice, the Commission issued a set of rules [2] concerning the licensing procedure, in 1962.

The authorization system was structured at two levels and took into account only the use of radioactive sources and nuclear installations.

The first level covered the use of radioactive sources or X ray generators contained in devices designed and manufactured for the purpose of detecting, measuring, gauging or controlling thickness, density, level, interface location, radiation, leakage or qualitative or quantitative chemical composition, or for producing light or an ionized atmosphere, and provided simplified licensing procedure. The second level, with a complete licensing procedure, was divided into four categories in accordance with the activity and with the type of sources and nuclear installations utilized.

The user was obliged to organize his activity in a specified area, adequately equipped and fitted out, called nuclear units. The nuclear unit's staff was considered as having occupational exposure to radiation, and received supplementary rights (such as additional salary, shorter working time, longer vacations).

At that time, no exemption principles had been established. The governmental decree did not require authorization proceedings for the use of radioactive materials whose specific activity was lower than 0.002 μCi/g (74 Bq/g) and for X ray diagnostic installations.

In 1974, as the law concerning the carrying out of nuclear activities [3] was adopted, a new authorization system was created. The law stipulated that nuclear activities could be carried out only by legally authorized persons.

According to the requirements of the law, activities such as owning, using, producing, transporting, transferring, transitting, supplying, repairing, importing and exporting of radioactive sources and nuclear installations had to be authorized.

The Law was completed with a set of standards issued by the regulatory authority, as follows :

- Norms for Radiation Protection (1975) [4], which established basic standards, and
- Work Rules with Nuclear Radioactive Sources (1976) [5], which settles the administrative and technical requirements for authorization. These Norms are still valid.

After the political changes that occurred in Romania, the regulatory authority changed its name to National Commission for Nuclear Activities Control (NCNAC) and it functions as a part of the Ministry of Waters, Forests and Environmental Protection. Our agency is only in charge of regulations and licensing and has no competence in such fields as research and development in nuclear energy.

3. THE SYSTEM OF EXEMPTION FROM AUTHORIZATION IN ROMANIA.

After 1970, the number of applicants for licences increased due to strong development in the use of radionuclides in industry and medicine. This obliged the regulatory authority, which was not correspondingly staffed, to exempt some activities from authorization, in order to concentrate on important matters of nuclear safety.

One should mention that the legally authorized persons whose activities had been exempted tried hard to regain their previous status, under the pressure of their own employees, who had lost their supplementary rights.

One should mention that the legally authorized persons whose activities had been exempted tried hard to regain their previous status, under the pressure of their own employees, who had lost their supplementary rights.

The exemptions criteria were established so that the exposure to radiation of the workers who developed their activities by using exempted radioactive sources or nuclear installations should be lower than 0.5 rem (5 mSv) per year (which represents the total dose admitted for individual members of population in Romania).

Neither collective dose estimations nor any considerations of accident circumstances have been made until now.

In these terms, the following activities are exempted from authorization [6]:

(i) professionals who work with materials having a specific activity lower than 0.002 μCi/g (74 Bq/g),

(ii) professionals whose activities with materials for which the total activity does not exceed the following limits shown in Table 1 (only the most utilized ones are mentioned) :

TABLE 1. SOME EXAMPLES OF EXEMPTED QUANTITIES

Symbol	Element and atomic number	Activity			
		Unsealed source μCi (MBq)		Sealed source μCi (MBq)	
^{241}Am	Americium (95)	.01	($4{\cdot}10^{-4}$)	8	(0.3)
^{198}Au	Gold (79)	100	(4.0)	100	(4.0)
^{60}Co	Cobalt (27)	100	(0.04)	7	(0.26)
^{131}I	Iodine (53)	1	0.04)	40	(1.48)
^{194}Ir	Iridium (77)	10	(0.4)	100	(4.0)
^{90}Sr	Strontium (38)	0.1	(0.004)	10	(0.4)
^{99m}Tc	Technetium (43)	100	(4.0)	100	(4.0)
^{3}T	Tritium (1)	1000	(40.0)	1000	(40.0)

Note :

(a) If many radionuclides or mixtures are used, the sum of the activities of the utilized nuclides divided by the upper limits of activity indicated in the table for the respective nuclide, must be lower than the unit.

(b) For the radionuclides not mentioned above, or for mixtures with unknown content, the total activity must not exceed:
- 0.01 μCi (0.4 kBq) for unsealed sources with radioactive nuclides that emit alpha rays.
- 0.1 μCi (4 kBq) for unsealed sources with no such radioactive nuclide
- 1 μCi (40 kBq) for sealed sources.

(c) In owning and/or storing cases the quantity will be limited so that the total activity would not exceed ten times the table values

(d) For supplying, selling, transferring and transitting, the annual quantity will be limited so that the total activity would not exceed 100 times the table values.

(iii) Professional activities at nuclear installations which accelerate electrons to an energy up to 5 keV.

(iv) Professional activities at installations, devices or other objects that contain radioactive sources with an activity level up to ten times the table values, and also at radiation generators with accelerating energies up to 50 keV, for which the dose rate does not exceed 0.1 mrem/hour at 0.1 m near the surface of the installation.

The exemptions do not apply for :

(1) production, import and export of radioactive materials and/or nuclear installations

(2) medical use of radioactive material

(3) intended contamination of foods, fertilizers, drugs, cosmetics or any other objects for public use.

4. APPLICATION OF SYSTEM OF EXEMPTION FROM AUTHORIZATION

4.1 Use of smoke detectors

There are many types of smoke detectors utilized in Romania. Some of them, produced before 1970, contain relatively large quantities of radioactive materials (e.g. CERBERUS type; KI type, made in the former USSR, containing plutonium).

After 1970, the production of DICI-73 type, which contains a radioactive source of ^{241}Am with an activity of about 16 μCi (0.6 MBq) was started in Romania. The production of DICI-76 type for which the radioactive source of ^{241}Am had an activity around 4 μCi (0.15 MBq) succeeded only several years later.

According to the exemption principles [6], the use of DICI-76 smoke detector is exempted from authorization, but its production, repair and distribution have to be submitted for authorization.

The organizations authorized for smoke detector repair are obliged to retain the detectors that cannot be repaired and to hand them over to an authorized organization specialized in radioactive waste collection and treatment.

Having this procedure, only a small part of the smoke detectors exempted from authorization are disposed of at municipal landfills.

This exemption method does not entirely correspond to IAEA Safety Guides No. 89 [7] recommendations, because it does not exempt the whole specific practice; some of the old types of smoke detectors are still in use and their users have to obtain authorization.

4.2 Use of electronic microscopes

The electronic microscopes are devices that accelerate electrons up to energy levels higher than exclusion limit [5] and therefore are subject to authorization. The types of electronic microscopes utilized in Romania belong to the class of nuclear installations for which the dose rate does not exceed 0.1 mrem/hour at 0.1 m near surface, and thus, according to the exemption criteria [6], their utilization is exempted from authorization.

REFERENCES

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DEVELOPMENT OF EXEMPTION PRACTICES IN FINLAND AND IN NORDIC COLLABORATION

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Abstract

The regulations for the exemption of radioactive wastes have recently been revised in Finland. The general authorizations for exemption are defined in nuclear energy and radiation legislation and the detailed requirements, including clearance levels, are given in two guides issued by the Finnish Centre for Radiation and Nuclear Safety. The adopted radiation protection principles are consistent with those recommended in the IAEA Safety Series No. 89. A Nordic project providing practical guidance for exemption, e.g. on radiation impact analysis and activity monitoring, will be completed and reported by the end of 1993.

1. REGULATIONS

1.1. Radiation protection principles

The radiation protection principles adopted in the Finnish exemption regulations are consistent with those recommended in the IAEA Safety Series No. 89. Accordingly exemption shall not give rise to radiation exposure of the public or workers at the waste treatment facility or disposal site, exceeding

(1) An effective dose of 0.01 mSv in a year to the most exposed individuals (members of the critical group), or

(2) A collective dose commitment of 1 manSv per year of practice.

For nuclear wastes, the practice refers to exemption of wastes from one nuclear power plant or a nuclear facility of other kind. Regarding exemption of wastes from the use of radioisotopes, the constraints are applied separately for all gaseous and liquid discharges and for all disposal of solid wastes.

1.2. Guide for exemption of waste from the use of radioisotopes

The principles followed in the exemption of radioactive wastes from hospitals and research institutes are extensively based on the recommendations [1] issued in 1986 by the Nordic radiation protection authorities. The Finnish Centre for Radiation and Nuclear Safety has issued a guide on exemption of such waste [2]. According to that guide, solid waste may be disposed of via the public waste treatment systems, if

(1) The total disposed activity from one laboratory is below 25 ALI_{min} per month and the activity disposed of annually is below 100 GBq,

(2) No single package contains more than 2,5 ALI_{min} of radioactive material and the surface dose rate on individual waste packages does not exceed 5 μSv/h.

There is also a limit of 100 kBq for the disposal in a landfill of sealed sources containing mainly beta–gamma emitters.

1.3. Guide for exemption of nuclear waste

According to the Finnish Nuclear Energy Decree, nuclear waste can be transferred to another holder having no authorization pursuant to nuclear energy legislation (i.e. the waste is exempted from regulatory control) provided that

(1) The average activity concentration in the waste is less than 10 kBq/kg, and the total activity in the possession of a transferee is less than 1 GBq and alpha activity less than 10 MBq, and
(2) Due to the exempt waste, the estimated annual effective dose to any individual is less than 0.01 mSv and the total radiation exposure is as low as reasonably achievable.

The criteria given above are applied when the Finnish Centre of Radiation and Nuclear Safety makes case–by–case decisions on the exemption of nuclear wastes (so called conditional exemption).

There is also an option for unconditional exemption of waste. In a guide [3] issued by the Finnish Centre for Radiation and Nuclear Safety, the following activity constraints are given for such exemption (excluding natural radionuclides):

(1) The total activity concentration, averaged over a maximum amount of 1000 kg of waste, shall not exceed 1 kBq/kg of beta – gamma activity or 0,1 kBq/kg of alpha activity. In addition, no single item or waste package weighing less than 100 kg may contain more than 100 kBq of beta – gamma activity or 10 kBq of alpha activity.
(2) The total surface contamination of non–fixed radioactive substances, averaged over a maximum area 0,1 m^2 for accessible surfaces, shall not exceed 4 kBq/m^2 of beta – gamma activity or 0,4 kBq/m^2 of alpha activity.

Unconditional exemption is not applicable to waste that may easily cause radiation exposure (e.g. highly volatile or flammable materials).

2. EXEMPTION PRACTICES

2.1. Wastes from the use of radioisotopes

In Finland, the amount of radioactive wastes from hospitals, industry and research institutes is estimated to be about 10 TBq per year, if radionuclides with a short half-life (not more than 6 hours) and sealed sources are excluded. About 80 % of this activity is composed of old Mo–99/Tc–99m generators and I–131 used in nuclear medicine and in the production of radiopharmaceuticals. Radionuclides with a half-life exceeding one month constitute some percent of the total activity in the wastes. Radioactive wastes mentioned above are generally disposed into the sewer systems and to landfill sites after a sufficient decay period in accordance with the instructions given in the regulatory guide [2].

Tritium waste consists of gaseous tritium light devices and radioluminescent paint. The production rate is 3 TBq per year and the total amount in the national waste storage is about 30 TBq. Since 1992, it is allowable to dispose separate ionization chamber smoke detectors (maximum activity 40 kBq of Americium–241) in municipal landfills.

2.2. Nuclear wastes

At the Finnish nuclear power plants, typically 10 m^3 of contaminated oil per reactor unit is annually exempted for incineration or reuse. The activity concentration in oil has been about 100 Bq/liter on the average.

The typical amount of exempt scrap metal from a NPP is around ten tonnes per year. Occasionally the amounts can be much higher: when the copper–aluminium condensers of the Olkiluoto NPP were replaced by titanium ones, a total of more than 300 tonnes of scrap metal was exempted for use as raw material by the metal industry. The average activity concentration in that scrap metal was some hundred Bq/kg.

Annually about 15 tonnes of very low level trash waste from the Olkiluoto NPP is exempted for burial in an on–site landfill. The clearance levels are e.g. 2 kBq/kg for Cs–137 and 10 kBq/kg for Co–60. From the Loviisa NPP, trash waste is exempted for disposal in a municipal landfill in accordance the unconditional clearance levels given in Chapter 1.3.

3. A NORDIC PROJECT ON EXEMPTION

The Nordic Committee for Nuclear Safety Research (NKS) organizes pluriannually joint research programmes financed by regulatory and some other organizations in Denmark, Finland, Iceland, Norway and Sweden. During the programme period 1990 – 1993, one project addresses exemption of radioactive materials. The project aims at providing practical guidance for exemption; the following issues are dealt with in the final report [4]:

(1) Estimates of types and quantities of potential exempt wastes in the Nordic countries
(2) Radiation impact analysis for different exemption options; models and parameter values for such analyses are given
(3) Monitoring to demonstrate compliance with clearance levels; e.g. determination of the difficult–to–measure nuclides and estimation of their concentrations by means of scaling factors is discussed
(4) Regulatory control in context with exemption
(5) Status of regulations concerning exemption in the Nordic Countries.

The information included in the document is mainly extracted from various reports about exemption published by the IAEA, CEC and some other organizations. The applications are focused on conditions that prevail in the Nordic Countries. It is expected that the document will become a useful tool for both the implementors who make exemption applications and other documents and the regulators who review them.

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EXEMPTION OF MINOR QUANTITIES OF RADIOACTIVE MATERIALS AND SOME MACHINES AND DEVICES EMITTING IONIZING RADIATIONS FROM THE REQUIREMENTS OF REGISTRATION AND/OR LICENSING IN ISRAEL

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Abstract

The use of radioactive materials and of machines or devices emitting ionizing radiations in Israel is subject to requirements of licensing or registration.

These requirements are specified in several regulations related to radiation protection and to the use of radioactive materials.

Based on the exemption principles formulated by the IAEA a few years ago (IAEA Safety Series 89), a group of radiation protection experts at the Soreq NRC submitted in 1992 a draft regulation proposal related to the exemption of certain radiation sources and practices from these regulations.

This draft was reviewed and ratified by the licensing division of the Israel AEC and is now in the process of rewriting and being formulised by the legal staff of the Ministry of the Environment responsible for the enforcement of the above mentioned regulations.

Among the radiation sources and devices recommended for complete or partial exemptions are:
Naturally radioactive ores; machines and devices emitting very low energy X-rays (<5 keV); certain consumer products (such as radioactive smoke detectors and gaseous tritium light devices); television sets and VDU's; certain industrial nuclear gauges; low activity sealed radioactive test sources of portable radiation survey meters; radioimmunoassay kits with limited activity; and depleted uranium blocks used as counterweights in aircraft.

1. INTRODUCTION

The use of radioactive materials and machines and devices emitting ionizing radiations is subject, in Israel, to requirements of registration and licensing.

The radiation protection measures and the administrative requirements relating to the procedures of registration and licensing are detailed in specific regulations and decrees under the authority of the Ministries of Labour and Social Affairs (MOLSA) and the Ministry of the Environment (MOE). The licenses are issued by the Chief Radiation Executive (CRE) appointed by the Minister of the Environment. Registration of radiation sources used in industry, agriculture, medical facilities and research institutes is carried out by the Chief Labour Inspector appointed by the MOLSA.

The present radiation protection regulations include some very general clauses related to the exemption of minor quantities of radioactivity from the requirements of registration and licensing. However, these clauses do not include any specifications and quantitative directions (i.e. list of specific radioisotopes, or devices or minimal quantities to be exempted) and the authorities find these clauses difficult to implement. This causes, in many cases, unnecessary bureaucratic difficulties and unjustified expenses to the users.

At the request of CRE and of the Israel AEC, a group of radiation protection experts from the Soreq Nuclear Research Centre, Yavne prepared in 1991 a draft regulation proposal specifying quantities and concentrations of certain radioactive materials which are to be regarded as being below regulatory concern and should therefore be exempted from the requirements of all radiation protection legislation. It refers also to some machines, devices and sealed radioactive sources with minimal radiation risks that should be exempted.

The draft regulations were recently reviewed and ratified by the licensing division of the Israel AEC and by the National Advisory Board on Radiation Protection and are now in the process of being rewritten and legally formulised in the legal division of the MOE. It is expected that they will be signed by the Minister of the Environment and published, as an amendment to the present regulations, during 1994.

The draft regulations are based on the general principles laid down a few years ago by an IAEA expert group and published in Safety Series no 89 (1). Consideration was given in the proposal to the practical conditions in Israel relating to the specific practices and the types of machines and radioactive materials used and to the economic and social priorities.

The following is a concise list of the radiation sources and devices recommended for exemption.

2. EXEMPTIONS

2.1 Exemption from All Requirements of Registration and Licensing of Radioactive materials and sources

2.1.1 Naturally radioactive ores which have not been subject to physical or chemical processes causing an increase in the concentration of any of the radioactive elements they contain and provided that the total activity concentration in the ores does not exceed 500 Bq/g.

2.1.2 Any solid radioactive material (excluding radioactive waste) with a total activity concentration that does not exceed 100 Bq/g and provided that this activity is distributed fairly homogeneously throughout the solid material.

2.1.3 Radioactive materials and radiation sources with an activity concentration that exceeds 500 Bq/g but with a total activity that does not exceed the following:

a) $5x10^3$ Bq for radionuclides classified as having a very high radiotoxicity (group 1 as listed in the Official Journal of the European Communities 246, 17.9.80).
b) $5x10^4$ Bq for radionuclides having high radiotoxicity (group 2 in the above list).
c) $5x10^5$ Bq for radionuclides having moderate radiotoxicity (group 3 in the above list).
d) $5x10^6$ Bq for radionuclides having low radiotoxicity (group 4 in the above list).

and provided that the dose rate at 5 cm from any accessible surface of the material does not exceed 10 microSievert per hour.

Note:

(The exemptions detailed in 2.1.1 and 2.1.3 does not allow the introduction, by human activity of any radioactive element into food stuffs, beverages or cosmetic products).

Machines and Devices

2.1.4 Machines and devices emitting ionizing electromagnetic radiations with photon exceeding 5keV.

2.2 General Licenses

Some sources and devices with very low radiation risks will be subject to minimal regulatory requirements. These will be exempted from many of the administrative requirements by granting to these sources and devices as a group general licenses without requiring the submission of applications for license by the individual user. The following is a list of these sources and devices and the radiation protection requirements which are a condition for the use of the general license.

Consumer products

2.2.1 Smoke detectors utilizing Am-241 as the alpha radiation source, with an activity not exceeding 37 kBq, and provided the source and the smoke detection assembly comply with IS (Israeli standard) 1220.

2.2.2 GTLS's (Gaseous Tritium Light Sources) and GTLD's (Gaseous Tritium Light Devices) with an activity not exceeding 7.4×10^9 Bq per source and 1.85×10^{10} Bq per device.

(The general license will be granted for the individual user but not for the production and/or maintenance activities related to these sources).

2.2.3 Machines and devices emitting electromagnetic ionizing radiation provided that the dose rate in air at a distance of 10 cm from any accessible surface of the machine and/or device does not exceed 1 Sv/h.

2.2.4 VDU's and television receivers provided that the radiation dose rate in air at a distance of 5 cm from any accessible surface of these instruments does not exceed 5Sv/h.

Sealed radioactive sources

Sealed radioactive sources which are part of certain industrial nuclear gauges provided that:

2.2.5 The sources comply with the ISO requirements for sealed radioactive sources (2-3)

2.2.6 The gauges were imported/installed by a qualified person with the appropriate license to carry out this activity.

2.2.7 The total activity in the gauge does not exceed 3.7×10^8 Bq.

Radiopharmaceuticals for in-vitro use with limited activity

2.2.8 A general license will be granted for the use of radiopharmaceuticals for in-vitro applications labelled with H-3, C-14, Fe-55, Se-75, I-125 and I-131 provided that the radioactive materials are part of a commercial kit and the activity of a single kit does not exceed 1.85×10^8 Bq for H-3 and 1.85×10^7 Bq for C-14, Fe-55, Se-75, I-125 or I-131.

(The general license will apply only to medical laboratories which use at one time not more than 10 times the activity limit for a single kit and only if the annual turnover in the laboratory or clinic does not exceed 50 times the single kit limit).

Miscellaneous

2.2.9 Depleted uranium used as counterweights in aircraft, provided it is appropriately labelled and is coated with a protective paint or other cover, to prevent dispersal.

References

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PRACTICAL PROBLEMS OF VERY LOW LEVEL RADIOACTIVE MATERIALS MANAGEMENT IN THE RUSSIAN FEDERATION

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Abstract

The principles and national standards applied in the Russian Federation in the management of very low radioactive materials have been described in the Report. It has to be noted that the current standards are as a rule temporary ones.

The standards are based on the use of the dose of exposure as the main criteria, while the recommended unconditional clearance levels contain values which could be easily measured in practice.

Temporary standards are developed and used as a rule for a concrete type of work and contaminated materials with known radionuclides content.

At present these standards are applied to territories of Russia contaminated as a result of the Chernobyl Accident as well as other contaminated areas.

The experience gained as a result of the utilization of temporary standards, as well as international recommendations, form a basis for the adoption in the near future of national regulations of a permanent character.

1. Regulations for very low level radioactively contaminated wastes.

The "Basic Sanitary Regulations for radioactive materials and other sources of radiation BSR-72/87" [1] is the main document, regulating the basic requirements for radiation safety in Russia.

This document sets the lower limit of referring radioactive materials to radioactive wastes. This level for solid wastes is $2*10^{-7}$ Ci/kg for sources of alfa-radiation (for transuranium elements $1*10^{-8}$ Ci/kg), $2*10^{-6}$ Ci/kg for beta-radiation , $1*10^{-7}$ gramm-equivalent of radium/kg for gamma radiation sources.

Respective criteria of the lower limit of radioactive wastes from NPP and wastes from enterprises and institutions, not connected with the nuclear fuel cycle, had been worked out on the basis of this regulation. The criteria rest on the specificity of the radionuclide composition and the simplicity of determining the levels of radionuclide content on practice.

Different approaches are applied to setting regulations of materials , containing radionuclides, depending on: the dose exposure, the specific activity, the particle flux. These values do not correlate easily for different radionuclide contents, so the primacy is given to the most rigid criterion. For example, the lower limit of radioactive wastes for NPP is set the following way [2]: solid wastes are considered to be radioactive, if the dose rate of gamma-radiation 0,1 m way from the surface exceeds 1 mkSv/h (100 mkrem/h) or the specific activity for beta-radiation sources exceeds 7,4 $*10^4$ Bk/kg (2 mk Ci/kg), and for alfa-radiation sources exceeds 7,4 $* 10^3$ Bk/kg (0,2 mkCi/kg). For soluble radioactive salts (chlorides, nitrates, borates etc.) the maximum activity is lessened 10 - fold.

The lower limit for surface contamination has also been set. Materials with the particle flux exceeding 500 particles/sq.cm.min. for beta-radiation sources and 5 particles/sq.cm.min. for alfa-radiation sources are considered to be radioactive.

In 1988 after specifying the calculation of the radionuclide content typical for NPP the lower limit for gamma-radiation sources for NPP wastes was changed from the value of 0,3 to the value of 1 mkSv/h (30 to 100 mkrem/h) [3].

In accordance with the Sanitary Regulations for the design and the exploitation of NPP (SR NP-38) [2] non-radioactive wastes (< 100 mkrem/h) from NPP are buried at special sites within the NPP sanitary - protective zone accordind to the general regulations concerning the burial of industrial wastes.

The Sanitary Regulation for managing radioactive wastes in other branches of industry and not regulating the management of radioactive wastes of nuclear fuel cycle facilities have set only the upper limit for low level wastes - 0,3 mSv/h (30 mrem/h). The lower limit has not been set. Practically, materials with the contamination level exceeding 3 mkSv/h were brought to burial sites for radioactive wastes.

Materials, with radioactive contamination levels lower than the above mentioned, were treated as industrial wastes and were brought to municipal landfills.

After 1986 large-scale studies were performed to reveal radioactive anomalies in towns, on industrial sites, in areas of nuclear fuel cycle facilities. The problem of setting the lower limit of solid radioactive wastes and of setting the criteria for very low level radioactive materials appeared inmediately in view of the task of the radioactive anomalies decontamination. The lower limit for carrying out decontamination activities in urban conditions was set at 1,2 mkSv/h (120 mkrem/h) [5].

By that time there was established the practice of carrying out decontamination activies in big towns practically to the background level and organising a centralized system of burying the removed soil and construction materials in accordance with the demands of the population and the local municipal authorities.

In 1992 temporary criteria for decision-making in managing soils, solid construction materials, industrial and other wastes, containing gamma-radiation radionuclides, were proposed to regulate these activities.

According to the established criteria, no interference is required, if the dose exposure (DE) does not exceed the background rate more than 0,1 - 0,3 mkSv/h.

In the DE range of 0,3 to 1 mkSv/h decontamination activities are carried out to the level of 0,1 - 0,3 mkSv/h. Removed materials are used to fill-up hole-pits and ravins, to build roads outside settlements with the consequent recultivation of these sites to the level not exceeding DE of 0,1 - 0,3 mkSv/h.

In the DE range of 1 to 3 mkSv/h decontamination activities are also carried out to the level of 0,1 - 0,3 mkSv/h. Removed materials are brought to the industrial and municipal landfills , where special sections for them are organized with the consequent recultivation to 0,1 - 0,3 mkSv/h. Materials with DE over 3 mkSv/h are removed to special storages of radioactive wastes.

The above mentioned criteria calculated for defining the individual effective dose equivalent for the population in different dose estimation models usually don't exceed 0,1-0,2 mSv per year.

These criteria have been worked out for and applicable to local radioactive anomalies, contaminated mostly by gamma-radiation sources. Limits, set by the Basic Sanitary Regulations BSR-72/87 are applicable to alfa-radiation sources and beta-radiation sources [1].

2. Practice of of managing very low level radioactive wastes in contaminated areas.

Another approach is applied for areas with large-scale contamination as a result of radiation accidents on nuclear facilities.

The accidents on the Chernobyl NPP and the activities of the "Mayak" nuclear plant in the South Urals had the most grave environment contamination consequences for the territory of Russia [6].

In Russia the total square of contamination in the result of the Chernobyl NPP accident over 1 Ci/sq.km (37 kBk/sq.m.) for cesium-137 is approximately 56000 sq.km or 1,4% of the European part of Russia. The square with the contamination 15 - 40 Ci/sq.km (555 - 1480 kBk/sq.m.) is 2130 sq.km, with contamination over 40 Ci/sq.km (1430 kBk/sq.m.) 310 sq.km.

In the result of the "Mayak" NP (in the South Urals) activities 25000 sq.km were contaminated by long-living radionuclides, mostly by cesium-137 and stroncium-90.

The protective activities dose criterium for territories in Russia, contaminated in the result of large-scale radiation accidents is set at 1 mSv per year.

Decontamination and the improvement of the settlements are the measures taken to lessen the dose of external exposure of the population affected by the Chernobyl accident. About 300 settlements have been decontaminated by now. During the decontamination activities the soil, road cover and the roof-cover were removed.

In the first years after the accident temporary burial sites of the trench-type were constructed in the most contaminated (> 40 Ci/sq.km) areas of the Bryansk region.

Data of the samples collection of the buried materials in the burial-site near the most contaminated villages showed that the buried materials mostly belong to the very low level radioactive materials.

At the moment local decontamination activities are continued in the settlement with the contamination density of 15 - 40 Ci/sq.km.

The Chernobyl contamination is characterized by spotting. The decontamination of the radiation anomalies in towns and viligies mostly near the infant institutions is being carried out now.

The removed soil belongs to the category of very low level radioactive wastes, so special burial-sites are not organized. The contaminated soil is brought to areas with a higher level of contamination, and is used for road maintenance, filling-up ravins etc. Special regulations for these activities are not established.

The external exposure doses of the population were lowered 2-fold due to measures carried out. Apart from direct decontamination activities, the contribution into lessening the dose by the improvement measures is great: the road construction and covering, house-building, communication lines improving the municipal economy requires removing the soil and filling up the hole-pits. The contribution to lessening the total exposure dose of the population while decontamination and municipal economy improvement is approximately 30%.

Temporary criteria for levels of radioactive contamination of construction materials in view of their secondary utilization after decontaminattion are set for contaminated areas (eviction zone > 40 Ci/sq.km (1480 kBk/sq.m.) and relocation zone > 15 Ci/sq.km (555 kBk/sq.m.)) in the Bryansk region. The limit of 10 beta - particles/sq.cm.min. is set for house-building and nussery-building; the limit of 20 beta-particles/sq.cm.min. is set for other types of construction. In the range of 20 - 50 beta-particles/sq.cm.min. construction materials are buried as general urban wastes. If the level of contamination is over 50 beta- particles/sq.cm.min. then the construction materials in these zones are to be buried as radioactive wastes [7].

Old buildings in settlements from which the population was relocated are buried now. The roofs of the buildings are most contaminated. The levels of their contamination reach ~ 50% of the respective indicator for the soil. The contamination of the walls of the buildings is below 10% of the respective contamination of the soil. These materials also mostly belong to very low level radioactive and are buried mainly out of fire-protection and sanitary considerations.

To normalize the radiation situation decontamination of the radiation anomalies on the contaminated territories is carried out also in areas where the DE is below 1 mSv per years.

3. Regulation perspectives.

The normative base is being improved due to the accumulated experience of managing very low level radioactive wastes. Special regulation and instructions are being worked out on the basis of dose limits recommended by international organizations. For instance, proposals concerning normatives for cleaning and recycling contaminated metals and metal products are being aprobated. While elaborating regulations we strive to make then simple in exploitation and optimal from the point of view of radiation control and supply of necessary equipment.

The draft " Exemption from regulatory control: recommended unconditional clearance levels for solid materials incorporating radionuclides" being worked out under the auspecies of IAEI are very timely and will facilitate the elaboration ana adoption of respective national regulations.

The problem of defining the normative base of managing very low level radioactive materials in contaminated areas consitutes the specificity of the situation in Russia. Unfortunately, this problem is excluded from the concern of the IAEI Regulations under discussion. We think it worth to be included in the respective section of the Regulations or as a supplement to it. Respective proposals can be worked out by the Russian, Ukranian and Belorussian experts.

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A SURVEY OF POTENTIAL PROBLEMS FOR NON-NUCLEAR INDUSTRIES POSED BY IMPLEMENTATION OF NEW EC STANDARDS FOR NATURAL RADIOACTIVITY

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Abstract

A working group of experts according to Article 31 of the Euratom Treaty prepared a draft "Council Directive" for handling radioactive materials. Particularly for naturally occurring radioactive materials this new Directive allegedly signifies a considerable tightening of the limits below which radioactivity is allowed to be present in the environment. The Dutch Ministry of Housing, Physical Planning and Environment commissioned an inquiry to be held among a limited number of companies in separate branches of the non-nuclear industries in the Netherlands to inventorise the possible consequences for these companies if the published draft Directive were to be implemented unabridged in its present form.

This paper provides an analysis of the inquiry. It shows that after implementation of the Directive in the Dutch legal system a number of companies will be under official duty to report or obtain licenses under the Nuclear Energy Act without advantages for radiological protection. For these companies this means a great deal of administrative work and considerable additional costs. If the proposed exemption levels for the naturally occurring radioactive materials, are liberalized by a factor of ten, the problems will be much smaller and controllable.

1 INTRODUCTION

Generally speaking, the average concentrations of naturally occurring radioactive materials in the earth's crust are not high. Geological processes may have, however, brought about great variations in concentration, as a result of which some ores possess higher levels of natural radioactivity. The products and waste produced from these ores can have similar high concentrations.

The existence of natural radioactivity in ores and their products has long been recognized. National governments have imposed limits, above which the use of radioactive materials is restricted. In the Netherlands, the limits are laid down in the Implementing Orders of the Nuclear Energy Act. In general, the present limits are such that they are not exceeded when the primordial radioactive materials are present only in the form of pollutants. However, the concentration of radioactive materials is sometimes such that somewhere in the raw material-product-waste chain or at releases these limits might be exceeded. Under such circumstances, the legal restrictions and inspections apply.

2 THE DRAFT COUNCIL DIRECTIVE

The activities of the European Community in the area of radiological protection are based on the treaty by which the European Atomic Energy Community (Euratom) was founded. By virtue of Article 2b of this treaty the Community must "set up uniform safety standards to protect public safety and the safety of workers and monitor compliance with these standards."

The new recommendations of the ICRP [1] prompted an expert working group of the Commission of the European Communities, established under the terms of Article 31 of the Euratom Treaty, to submit to the Commission proposals for the amendment of the regulations. While the basic structure of the existing

Directive has been retained, the following aims have been pursued in this revision:

- To provide radiation protection based on the most up-to-date scientific knowledge, which should be utilised to the benefit of workers and to the general public.

- To provide a sound technical and scientific basis and a uniform approach to radiation protection, and to ensure technical consistency with the recommendations of other international organisations.

- With a view to the completion of the Single Market to preserve a high degree of harmonisation in the radiation protection measures under the Euratom Treaty.

- To strenghten the provisions on control of radioactive materials in accordance with the undertaking made to the Council in 1992, at the time of the adoption of Directive 92/3/EURATOM on the supervision and control of shipment of radioactive waste between Member States and into and out the Community.

On the basis of these proposals, the CEC prepared a draft for a new "Council Directive" in July 1992 [2] in which the basic standards were amended. The most important changes in the draft Council Directive are as follows:

- The inclusion of more restrictive dose limits taken from the latest ICRP recommendations, which take into account of the most recent estimates of cancer risk associated with exposure to ionizing radiation, together with the complex concept of health detriment.

- The introduction of provisions concerning radiation protection in certain cases of occupational exposure to natural radiation sources.

- The prohibition of certain unjustified uses of radioactivity.

- Expansion of the provisions concerning protective measures to be taken in the event of a radiological accident.

- Introduction of the "dose constraint" concept in relation to a given source.

- Changes to the radioactivity levels associated with the authorisation/reporting provisions laid down in the Directive.

- The exemption levels set for radionuclides have been altered. This aspect is of particular importance in connection with the survey reported here.

The system of reporting and prior licences for handling radioactive materials has been altered: the requirements regarding the application of this system have been made more explicit than in the existing Directive, with in particular the circumstances under which these requirements do not have to met being extensively altered. The intention is that this should contribute to further harmonization of the licensing procedures within the Community, a move which has implications for the completion of the single market.

After completion of this study a revised draft was published in which the applicability of the exemption levels was restricted to the system of reporting [3]. Also, the exemption levels were only to be applied to small amounts of materials. However, the conclusions of the study are in general also valid for the revised draft.

The Directive specifies exemption levels for each type of radioactive material. Exemption levels are set for both specific activity and absolute activity. A material is <u>not</u> exempt if <u>both</u> levels are exceeded. In industry, where natural radioactivity plays a role, it is the levels of specific activity which are in practice critical, since the quantities of material present mean that the absolute activity levels are almost always exceeded.

Table 1 Exemption levels for some important natural radionuclides according to the draft and current Council Directive

Nuclide	Activity (Bq)		Spec. Act. (Bq/g)	
	Draft	Current	Draft	Current
^{238}U sec	10^3	5.10^6	1	5.10^2
^{226}Ra +	10^3	5.10^3	1	10^2
^{210}Pb +	10^3	5.10^3	10^1	10^2
^{210}Po	10^3	5.10^3	10^1	10^2
^{232}Th sec	10^2	5.10^4	1	5.10^2

For anthropogenic materials, the new levels are in some cases more restrictive and in others more liberal than the existing standards. For a few anthropogenic materials the new levels are even a great deal more liberal. On the other hand, however, the levels set for primordial radionuclides are considerably more restrictive. Table 1 provides a summary of the exemption levels specified in the draft Council Directive for naturally radioactive materials.

The exemption levels for naturally radioactive materials are such that they are likely to impose obligations on various industries. If a company is obliged to apply for a licence to hold, process, store or trade materials that have not in the past been subject to regulation, it will incur considerable inspection and administration expenses. Registration will in many cases also involve inspection and administration. Moreover, there are fears of a considerable volume of waste henceforth being designated as radioactive. If this leads to a better situation of radioactive health protection all well and good; but there are doubts as to whether such waste in many cases really does constitute a hazard.

The above considerations were a ground for the Dutch Ministry of Housing, Physical Planning and Environment to ask KEMA to make an assessment of the problems which might be expected in non-nuclear industries in the Netherlands if the proposed exemption levels for naturally radioactive materials were to be adopted in full. The results of this study [4] were to be for use in discussions within the EC on the consequences of implementing the Directive, without however losing sight of the intended degree of safety.

2 INQUIRY

The non-nuclear industry in the Netherlands has a wide range. It includes by definition all Dutch industries except the nuclear industry. It is obviously impossible, certainly within a short period, to fully examine all the problems involved. It was therefore decided in consultation with the commissioning authority to conduct an inquiry by means of a questionnaire among a number of selected companies from various industrial sectors, with the emphasis on those making use of materials in which higher levels of specific natural radioactivity might be expected. Given the limited scope of the survey, it cannot be expected to yield a representative picture of every industrial sector; nevertheless, the examples given can be used to give an insight in the possible consequences of the implementation of the Directive for the non-nuclear industry.

The questionnaire was distributed amongst companies invited to attend an information meeting. During this meeting the objectives of the questionnaire were clarified and an explanation of the potential problems was given. The inquiry was intended primarily to get insight in the radioactivity balance of companies in various industrial sectors. This balance can determine whether the introduction of the proposed Council Directive would be likely to cause these companies problems in relation to the exemption levels regarding radioactivity in materials.

Since the Directive relates both to environmental protection and to occupational health and safety, and given the fact that the new exemption levels could have implications for Dutch regulations in these areas, the questionnaire was divided into three chapters, each of which was in turn subdivided. The content of the questionnaire is summarized below.

GENERAL

What materials and products enter the company and circulate within the company, and how do they leave?

What levels of specific activity from natural nuclides are associated with those materials, products, residues and wastes?

What is the nature of the processes carried out?

These data were obtained to indicate whether and where radioactivity limits are exceeded. It was also to serve to facilitate evaluation of the environmental and occupational hazards.

ENVIRONMENT

Stock survey: stock data are required to estimate direct radiation beyond the premises and to determine whether such radiation was still in line with the specified (Dutch) criteria.

Use of the surrounding area.

Emissions into the air, discharge into water and solid waste.

OCCUPATIONAL

Warehouse stocks: considerably increased radiation levels can exist in warehouses and storage sheds.

Materials during production: with some types of processes workers come into close contact with raw materials or products and are therefore externally exposed to radiation.

Extent to which workers are exposed to dust: with dispersion-sensitive materials active components are inhaled, which may lead to exposure to the lungs to radiation.

Protective measures.

Radiological workers.

In a workshop, held a few months later on, the respondents were given the opportunity to provide more information regarding their answers.

4 RAW MATERIALS

4.1 General

As was to be expected, Dutch industry uses a broad range of raw materials from many different locations all over the world. In general, little is known about contamination with natural radioactivity. A few companies were able to provide an accurate radioactivity balance, but the majority had no information or could at best provide only sketchy information. Consequently, it has been impossible to obtain a complete picture. One thing that did become evident from the inquiry is that there are some groups of materials which frequently display higher levels of radioactivity. These groups will be discussed below. Wherever possible, the proposed limits are given between braced brackets { }.

4.2 Zircon (cf. § 5.1, 5.3, 5.5)

Zircon is the common name for zirconium silicate ($ZrSiO_4$). It is a clear white material, mined in various places around the world. It is always contaminated with radioactive materials. Typical activity levels in Zircon imported to the Netherlands are:

U-series	4 Bq/g	{1 Bq/g}
Th-series	0.6 Bq/g	{1 Bq/g}

It is used:

- in fine ceramics; technically and commercially, it is virtually irreplaceable as a raw material for fine ceramics (cf. § 5.2).
- for high-temperature applications:
  - in steel and iron foundries (cf. § 5.5);
  - as "ZAC-stones" for furnace walls (cf. § 5.3 and § 5.5).
- as an additive in special types of glass (cf. § 5.3).

Under the new regulations, since the exemption levels for both specific and absolute activity would be exceeded, it would always be mandatory to register or obtain a licence to use this raw material. However, zircon containing products will remain under the exemption level for specific activity, because of the zircon in them being diluted; nevertheless, a certain amount of processing is required before this is so.

4.3 Thorium (cf. § 5.5)

Both as a metal and as an oxide, thorium possesses properties which make it pre-eminently suitable for certain applications.

- It can withstand very high temperatures (3000 K), making it suitable for
  - lighting (special high-pressure gas discharge lamps, gas mantles)
  - welding electrodes.
- It is an excellent alloying element for hard and wear-resistant material:
  - cemented carbide
  - drawing punches, moulds.

The specific activity of metallic thorium and thorium oxide is approx. 4000 Bq/g. A material with a thorium content as low as 0.025% would therefore exceed the specific activity limit {1 Bq/g}. At most 25 milligramme of thorium may be present in 100 gramme of material to remain under the absolute activity limit. If the thorium content is higher, the exempted amount of material would be proportionately lower than 100 gram.

4.4 Pigments (cf. § 5.1)

A great many pigments are in use and their origins are extremely diverse. It was therefore impossible to determine the natural radioactivity of a representative cross-section of those in use. It was possible, however, to test a supplier's warehouse stock using a simple hand-held measuring device. With the majority of pigments, measured radiation levels were no higher than the background levels. For a number of samples, however, a considerable increase was found. Although no exact measurements were made, it can be stated beyond doubt that these pigments exceed the levels of both specific and absolute activity above which registration or licensing would be compulsory.

As only relatively small quantities of pigments ever find their way into products, it is not anticipated that such products will be subject to compulsory registration or licensing. Pigment waste material does, however, require attention.

Because of the many combinations of colours and hues that can be made, it may be possible to avoid radioactive pigments; at worst this could mean that some very subtle hues cannot be created. However, this would require an extensive checking system to determine whether purchased pigments required registration or licensing.

4.5 Fossil Fuels (cf. § 5.8)

a Coal

The 1988 UNSCEAR report gives the respective concentrations of U-238 and Th-232 found in coal from a number of countries [5]. Table 2 contains a summary of the data for the major coal-producing countries.

Table 2 Summary of activity concentrations measured in coal from various countries.

Country	Global prod. (%)	Dutch import (%)	Activity concentration (Bq/kg)	
			U-238	Th-232
China	26		7	16
USA	24	49.0	18 (1-540)	21 (2-230)
USSR	16	1.0	28	25
Poland	6	11.1	38 (2-140)	30 (7-110)
S-Africa	6		30	20
India	5		24 (10-70)	38 (20-90)
Australia	4	28.8	35 (30-48)	30
Germany	3	7.0	20	<20
UK	3	1.5	15 (7-94)	13 (2.4-19)
Canada	1		12	7
weighted av. (global)			20 (5-350)	22 (5-230)
weighted av. (Netherlands)			25 (10-325)	24 (3-275)

The figures in the third column indicate the percentage of the Dutch total coal imports accounted for by the type of coal in question. The weighted average concentrations of radionuclides in coal used in the Netherlands does not differ significantly from weighted average concentrations world-wide. Carbon combustion increases the activity of the fly ash to a degree dependent on the ash content. The ash content varies from 3 to 30%, the average being 10%. Thus, the activity of the fly ash is typically about ten times as high as that of the coal, i.e. approximately 0.5 Bq/g.

Because of the large volumes of coal involved, the absolute activity limit {10^3 Bq} would always be exceeded. Where the more active types of fly ash, bottom ash and the soon to be produced coal gasification slag are concerned, the sum of the uranium and thorium contents could easily exceed the specific activity limit {1 Bq/g}. The possibility cannot therefore be excluded that the measurement of radioactivity concentration would have to be incorporated into the certification procedure for all coal residues. It is worth noting that the standstill principle applied in housing and construction policy in the Netherlands already means that the more active ash types cannot easily be reused as it is.

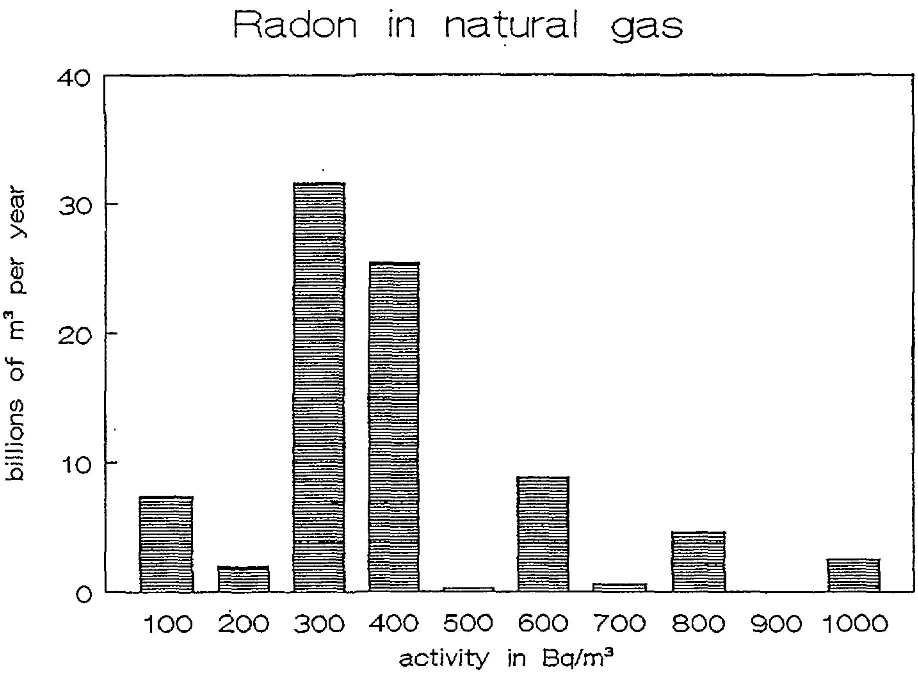

Figure 1 Distribution of radon activity in Dutch natural gas, in proportion to production

b Fuel oil

There are little published data on radioactivity in fuel oil. The study reported here failed to provide any additional information on the subject. However, oil contains less than one hundredth of the amount of inorganic material contained in coal. If the activity concentrations in fuel-oil fly ash, insofar as oil-fired power stations emit fly ash, do not differ significantly from those in coal fly ash, the same conclusions can be drawn regarding the two materials.

c Natural gas

The radioactive noble gas radon is extracted along with natural gas. Other pollutants such as dust and water which accompany the gas are separated close to the source. The separated watery and solid products often contain decay products in considerable quantities, so that production problems involving radiation and waste arise at the wells and treatment plants, even under the present regulations. Radon cannot be separated and is thus pumped into the piping system. When radon decays, the piping system is consequently contaminated with decay products.

The radon content of natural gas produced in the Netherlands and on the continental shelf varies greatly. Considerable differences occur even within a single concession. The lowest content reported is 12 Bq/m^3 (0.015 Bq/g) and the highest content is 4,600 Bq/m^3 (5.8 Bq/g) [1]. The distribution between these two extreme values is represented in Figure 1. This figure shows the values from 46 Dutch locations grouped into 10 classes and according to annual production in billions of cubic metres. The number of values > 1000 Bq/m^3 (1.2 Bq/g) is minimal. The average production weighted concentration is 337 Bq/m^3 (0.4 Bq/g).

The limits for the Rn-222 series are 10^8 Bq and 1 Bq/g. The occasional well does exceed the specific activity limit, but none exceed the absolute value because of the limited volumes of gas on site.

[1] A cubic metre of natural gas weighs approx. 0.8 kg.

Natural gas is also frequently used as a raw material in organic chemistry. The possibility of radon decay products finding their way into finished products cannot be excluded. If accumulation occurs, then the limits might easily be exceeded

4.6 Phosphates (cf. § 5.6)

Phosphate ore is transported in bulk to the Netherlands for extraction and production of pure phosphor, fertilizer and phosphoric acid.

The draft Council Directive stipulates that for a mixture of nuclides or a series that is not in equilibrium, total activity should be weighted by dividing the activity of the individual nuclides by their numeric values. If the sum of the values thus obtained, referred to here as the Σ value, should exceed 1, activity would be considered to be above the exemption level. Table 3 shows that the series are not in equilibrium. Therefore the weighted aggregate activity must be calculated to obtain the Σ value. However, there is insufficient information to do so, since the activity of only a few nuclides is known. In other words, there is also a practical problem in the determination of weighted activity. Nevertheless, the likely Σ values are given in the last column.

In spite of the paucity of information, Table 3 shows that a few types of phosphate ores exceed the specific activity exemption level. Because of the large quantities of material involved, the absolute activity limit would always be exceeded. There is considerable variation in the levels of specific activity. South African and Russian ores are below the limit, but are not easily available.

Table 3 Average levels of radioactivity [in Bq/g] in phosphate ores.

	U-238+ (1 Bq/g)	Ra-226 (1 Bq/g)	Po-210 (10 Bq/g)	Σ value
Jordan	0.5	0.75	1.02	≈ 1
'IMC'	1.33	1.58	2.68	> 1
Mixed	0.94	1.17	1.77	> 1
Morocco	1.40	1.40		> 1
Florida	1.80			> 1
South Africa	0.14			?
Russia	0.03			?

In fertilizer, Po-210 is even enriched by a factor 1.5 to 2 during production. During phosphor production, considerable enrichment occurs in the residues (up to 800 Bq/g Pb-210 in calcine).

4.7 Other Ores/Minerals (cf. § 5.7)

In addition to the above-mentioned ores, a large number of other ores are processed. Table 4 lists a number of ores whose aggregate specific activity levels have been reported to be in the region of the exemption level of 1 Bq/g.

Here again, it is impossible to calculate the weighted aggregate activity in the proper manner. However, the last column shows that in certain cases, the specific activity exemption level is certainly exceeded. Because of the large quantities of material involved, the absolute activity limit would always be exceeded.

Table 4 Specific activity levels [in Bq/g] of several ores near to the exemption level.

	U-238+ (1 Bq/g)	Ra-226 (1 Bq/g)	$Th-232_{nat}$ (1 Bq/g)	Σ value
bauxite from China	0.46	0.31	0.37	≈ 1
bauxite from Guyana	0.08	0.05	0.23	< 1
ilmenite	1.50	2.30	1.20	> 1
rutile	0.71	0.54	0.23	≈ 1
rutile	0.15	0.13	0.16	< 1

5 PROBLEMS BY INDUSTRIAL SECTOR

In this chapter, consideration is given to the question of whether the individual industrial sectors could face problems in the event of the draft Council Directive being implemented, relating to the above-mentioned raw materials themselves, or to products made from them, or waste associated with them. It is worth repeating at this point that registration or licensing is only compulsory where both the specific activity exemption level and the absolute activity exemption level are exceeded. Where products are concerned, a situation could arise whereby – depending on the mass of the product – a single product unit has an absolute level of activity beneath the limit (and could therefore be freely traded under the draft Council Directive), but a stock of such products would fall within the scope of the Council Directive. It should be noted that such a situation could also arise under the present Dutch regulations.

5.1 Ceramics

Fine ceramics (cf. § 4.2, 4.4, 4.5)

Fine ceramic products include: crockery, tiles, sanitary fittings etc. Zircon is both a technically and commercially indispensable component of quality products, typically constituting 5% of the finished product. The combined concentration of uranium and thorium in the finished products is therefore of the order of 0.2-0.3 Bq/g, which is well below the specific activity limit of 1 Bq/g. Pigments are added to the glaze to colour the products. Finished products have typically 0.2-0.3% pigment. This small quantity sometimes causes an increase in specific activity of the total product.

In fine ceramics industry, therefore, the raw materials do represent a problem, but the activity levels of the products generally remain below the specific activity limits. Whether the absolute activity limits are exceeded depends on the mass of the product. Attention should be given to waste products (residues of zircon, pigments).

Coarse ceramics (cf. § 4.4, 4.5)

Coarse ceramic products include bricks, roof tiles etc. Most raw materials are extracted locally. The specific activity levels of the raw materials are generally < 0.1 Bq/g. The addition of pigments, particularly common in the manufacture of roof tiles, might increase the levels of activity. In the coarse ceramics industry, therefore, raw and waste pigment materials could present problems. The products would generally possess levels of activity too low to require registration or licensing.

5.2 Construction Industry

No raw materials are processed in the construction industry. The work is restricted to the construction of large functional structures from ready-made products. During construction, no radioactivity concentration effects occur in products or waste. It is therefore mainly the suppliers who have to face problems involving radioactivity.

ceramics (cf. § 4.4, 4.5, 5.1)

The bricks and roof tiles used in the Netherlands are generally within the limits. In some cases, fly ash is mixed with the clay from which bricks are made, causing higher levels of specific activity in the product, which can be restrictive in connection with the Dutch 'standstill' principle. Occasionally the glaze on tiles contains excessive specific natural radioactivity. This no longer happens very often in the construction of new buildings, but demolition can occasionally give rise to unexpected problems, with the debris having to be disposed of as radioactive waste. Decorative tiles sometimes contain considerable concentrations of radioactivity, well in excess of the level at which reporting or licensing becomes compulsory.

cement (cf. § 4.5)

The basic materials for cement are marl and/or blast-furnace slag. After mixing in additives and grinding, these materials are burnt at high temperature (1400°C). Volatile components condense in the furnace dust which is recirculated in the process. The clinker thus obtained is then mixed with gypsum and also with fly ash. Because of the large amounts of cement produced, a considerable proportion of the fly ash produced in the Netherlands has traditionally been recycled in this way.

The sum of the specific activities of Ra-226 and Th-232 in the raw materials varies between 0.01 and 0.2 Bq/g. Only the specific activity of the fly ash is considerably higher, i.e. up to 0.5 Bq/g. Turning to the products, it has been calculated that the specific Ra-226 and Th-232 activity levels are between 0.07 and 0.17 Bq/g. Hence, the levels of radioactivity so far calculated for or measured in raw materials or products do not exceed the limit of 1.0 Bq/g.

Not all raw materials' specific activity levels are known. There are, however, no indications that any of the raw materials commonly used in the Netherlands possess levels of activity substantially in excess of those which are known. The draft Council Directive is not therefore likely to present problems where cement is concerned.

fly ash

As well as being added to bricks and cement, fly ash can also be used in the construction industry in the form of artificial gravels, such as Lytag, for making concrete. Apart from saving on gravel, this has the advantage of making the concrete lighter in weight.

The disadvantages associated with the resulting higher radioactivity are limited in the construction industry. The Dutch 'standstill' principle presently allows fly ash to be used as an additive in materials for houses and office buildings, but for the longer term the use of fly ash is under threat. However, the possibility cannot be excluded that the total radiation hazard will prove to be lower if fly ash is used. This is possible because the radon exhalation of fly ash is lower than that of the raw materials substituted for it. The additional direct radiation resulting from the higher specific activity might thus be offset by the lower radon exhalation. However, both the present regulations and the proposed Council Directive only take into account the quantities and concentrations of radioactive materials.

As was stated in the discussion of fly ash as a product (cf. § 4.5), a product specification based on the concentration of radioactivity would be almost inevitable if the new Council Directive were introduced. The more active fly ashes could easily possess levels of specific activity high enough to make registration or licensing obligatory, necessitating constant testing.

5.3 Glass (cf. § 4.2)

The activity of the basic raw materials is low (< 0.1 Bq/g) and therefore presents no problems. A few special types of glass do, however, contain additives with higher activity levels. Levels as high as 0.7 Bq/g – only just below the exemption level {1 Bq/g} – have been reported in finished products. The walling of glass furnaces consists of ZAC bricks (zircon-alumina castings). This material typically has a specific activity of 3 Bq/g. It would no longer be possible to use this material without reporting or licensing. Glass furnaces are decommissioned after ten years or less, generating waste which of course includes the ZAC bricks.

The glass industry would not experience problems, on any significant scale, with regard to its products if the draft Council Directive were implemented. However, some additives and the furnace materials would require reporting or licensing, both in use and as waste.

5.4 Chemicals

The chemicals industry uses a multitude of raw materials and produces an even greater variety of materials. It is not impossible – certainly in the heavy chemicals industry where primary raw materials are processed – that problems relating to the raw materials used or to concentration effects during processing could arise if the new limits were imposed. However, the inquiry brought no such problems to light, because little information was made available by this particular industry.

5.5 Metal (cf. § 4.7, 4.2, 4.3, 4.5)

Iron is extracted from ore in blast furnaces. The ore is first pre-processed in sintering or pelleting plants. Both in sintering and in pelleting, Pb-210 and Po-210 escape into the waste gases, with concentrations of 10 Bq/m^3 (approx. 10 mBq/g) Po-210 sometimes being reached. Coal is converted into coke. The Pb-210 and Po-210 activity finds its way mainly into the tar formed as a residual product of the process. The coke oven gas formed is used for underfiring.

A blast furnace yields three products: pig iron, slag and blast furnace ash. Both pig iron and slag are drained from the blast furnace. The level of activity in the pig iron is low, and during metal preparation this material is purified even further. The slag possesses about 0.15 Bq/g of activity from the U-238 series up to Pb-210, and approximately 0.15 Bq/g from the Th-232 series. Dust in the blast furnace gas contains 3-9 Bq/g of Po-210 and 6-22 Bq/g of Pb-210. This dust forms scales in which activity levels increase up to 200 Bq/g of Pb-210/Po-210. Hence, in the iron smelting industry, concentrates (blast furnace slag, blast furnace dust scales) are formed in which the specific activity is significantly increased, meriting special attention. The waste produced by this industry would therefore require registration or licensing under the draft Council Directive. However, with regard to waste disposal, companies active in this area are already subject to licensing obligations under the Nuclear Energy Act.

In the iron processing industries, few problems occur with the material itself because it is supplied in purified form by the smelting industry. There are, however, secondary problems with some products, as with welding rods, for instance (cf. § 4.3).

One raw material which does approach the new limits is bauxite, used in the aluminium industry (cf. § 4.7); bauxite could therefore only be used subject to monitoring.

The ZAC bricks used in the construction of the furnaces would present problems both in use and as waste. In steel and iron foundries, zircon sand is used because of its resistance to heat (cf. § 4.2). This sand could no longer be used freely under the new regulations.

5.6 Phosphor/Fertilizer (cf. § 4.6)

Phosphate ore is supplied in bulk in the Netherlands for extraction and production of pure phosphor, fertilizer and phosphoric acid. In the process of extracting the phosphor from the ore, Pb-210 and Po-210 are released. Depending on the production process applied, enrichment can also occur. Emission into the air or discharge into surface water takes place.

phosphor

In the extraction of phosphor, the ore is first ground and then granulated with a binding agent. The resulting grains are sintered at 800°C. The phosphate is then reduced to phosphor in a furnace at 2200°C. Slag is drained from the furnace with a concentration of U-238 and its daughter products of 1-1.5 Bq/g. This slag is currently used mainly in hydraulic engineering and road construction, but this would cause problems under the proposed Council Directive. The stack gases are cleaned. The residue of this clean-up process contains about 800 Bq/g of Po-210 and is produced at a rate of about 1000 tonnes/a. Using special techniques, the activity levels of this waste can be reduced to about 100 Bq/g. The latter level is sufficient to satisfy the present regulations, but would not be acceptable under the proposed Council Directive.

fertilizer/phosphoric acid

The phosphates are extracted from the crushed ore by treatment with acid. The addition of sulphuric acid leads to the release of gypsum and the creation of phosphoric acid.

Fertilizer is made by introducing additives. The composition of these additives depends on the desired quality of the final product. The product is granulated and dried.

During the extraction process, Pb-210 and Po-210 are released and largely emitted. The clean-up residues would require reporting or licensing if the draft Council Directive is implemented, since they generally possess levels of activity above the exemption level {1 Bq/g}. Enrichment can lead to fertilizers having an activity level 1.5 to 2 times as high as that of the ore, so that these too would exceed the limit (cf. Table 3).

5.7 Other ores/minerals (cf. § 4.7)

Other companies, processing ores or minerals such as in Table 4 with activity levels approaching the limits, would only be allowed to use them subject to monitoring and due attention to potential concentration effects. However, even where ores with lower specific activity levels (< 0.1 Bq/g) are processed, concentration can occur, with the result that the specific activity in residues and/or waste can be much higher. Concentration factors in excess of 100 have been reported.

5.8 Energy supply (cf. § 4.5)

extraction of fossil fuels

Natural gas and oil are extracted on land and on the continental shelf. Coal is no longer mined in the Netherlands. Contaminants, mainly radon, are extracted along with the gas and oil. The companies in question are thoroughly aware of the risks involved. The present Dutch regulations already provide for mandatory monitoring for the presence of natural radioactive nuclides. If the draft Council Directive were implemented, monitoring would have to be significantly extended to include far more sites and locations. Large amounts of concentrates would also exceed the proposed limits, with the result that they would perhaps have to be disposed of as radioactive waste.

underfiring

The fossil-fuel-firing electricity generating industry produces about 800,000 tonnes of fly ash per year. If the draft Council Directive were implemented, product certification would be necessary to ensure that the limits were not exceeded. This would make disposing of coal residues more difficult (cf. § 4.5), whereas recycling might be an effective way to reduce risks (cf. § 5.2).

6 WORKSHOP DISCUSSIONS

The respondents to the questionnaire were given the opportunity to add to their written responses during a workshop. A number of important comments were made:

- In reviewing the draft Council Directive, the difference between the levels at which reporting becomes compulsory and those at which licensing becomes compulsory could perhaps be widened. This would provide an opportunity for the establishment of a less administratively complex (registration) system.

- From the inventory it would appear that liberalizing the limits on naturally occurring radionuclides by a factor of ten would remove a great many problems.

- The exemption levels proposed in the draft Council Directive have been calculated using models and scenarios whose ability to provide a realistic picture of the industrial and non-industrial risk from radioactivity is highly dubious. In a number of cases, the model parameter values are improbably conservative and not based on verifiable scientific sources. In many cases, scenarios which reduce the risk to workers would lead to far reaching restrictions – such as for instance on the recycling or disposal of residues and waste – with repercussions for the general population. The likelihood of contact varies enormously, however. The draft includes no scenarios covering emission into the air or discharge into water. No account is taken of the specific chemical and physical forms in which natural radiation sources are found in non-nuclear industries. It was also observed that the calculation and estimation methods used are such that even fairly minor changes in the basic assumptions would result in the exemption levels being ten times as high.

- A possible solution to at least some of the problems might be to keep raw materials outside the scope of the Directive. This would, however, lead to inconsistencies:

    - The specific activity of raw materials can be higher than that of products or residues.

    - Situations could arise during industrial processing whereby a raw material becomes processed material, and thus subject to reporting or licensing regulations.

    - The use of newly extracted ores or materials would be made more attractive than the recycling of residues.

- If the Council Directive were implemented in its present form, large amounts of waste which is currently recycled or disposed of at dump sites would have to be treated as radioactive.

- Specific and absolute activity levels should not be the only criteria considered. Where radon-exhaling materials are concerned, risk is the sum of internal contamination and external exposure to radiation. The use of residues in building materials may increase external exposure to radiation, but reduces internal contamination as the exhalation is lower. Thus, the aggregate risk represented by the above sum depends on the degree to which the residues are processed. A distinction could also be made for instance between the use of fly ash in embankments and its use in house-building materials.

- In many cases radioactivity certificates would have to be obtained from the raw material suppliers, many of whom are from outside the EC. This would effectively shift the problems, which cannot always easily be solved, onto these companies.

- With a few exceptions, products would generally be below the absolute radioactivity limit. However, during transport and storage problems could arise due to several product units being considered collectively as a single source possessing more than the permitted total amount of radioactivity.

During the working group discussions the following specific observations were made:

Problems relating to environmental protection

- The radioactivity of imported phosphate ores is generally above the proposed limits (the few "clean" ores are very hard to obtain). Phosphor extraction produces about 700,000 tons of slag per annum, which would probably not be freely disposable. This could effectively mean the end of this industry. In the preparation of certain types of fertilizer which contain large amounts of phosphor or other phosphates, the uranium and thorium activity is concentrated by a factor 1.5 to 2. Such products might require reporting or certification under the new Council Directive, thus making them unsaleable.

- About 10,000 tons of blast-furnace flue dust per annum would be designated as waste requiring registration or licensing.

- The amount of waste generated during the gas/oil production that would be classed as radioactive would increase substantially.

- The use of fly ash (approx. 10^6 tonnes/a) in building materials is already under threat. If some of the fly ash required registration or licensing, this would make selling it even more difficult, while storage is not a realistic alternative. The electricity generating industry feels that this would make the firing of coal a less attractive option.

- The remaining industries have as yet little insight into the amounts of wastes which would be classified as requiring registration or licensing. They fear, however, that the volumes involved could be considerable. It should be noted in this context that the price for disposal of radioactive waste in the Netherlands is about NLG 1000 (i.e. about ECU 450) per 200 litre drum. The total cost involved could therefore run into tens of millions of guilders.

It was argued that increasing the volumes of waste treated as radioactive is at odds with the Dutch policy of encouraging the recycling of residues and the conservation of raw materials. It was questioned whether the proposed policy would indeed lead to a reduction of the collective dose and what the cost per manSievert saved would be.

It is most improbable that the costs associated with the implementation of the draft Council Directive could be justified from the viewpoint of optimization.

Problems relating to occupational health

The dose limits for workers proposed in the draft Council Directive are broadly similar to those laid down by the Dutch authorities. It is as yet not entirely clear what repercussions the new standards would have in the field of occupational health. In a number of instances, attention has to be paid to certain issues of radiological protection against naturally active sources. The following general observations were made regarding radiological protection protocols:

- Would the permissible levels of surface contamination decrease too? In practice, the present limits already present technical problems with respect to measurements.
- The size of the monitored areas and controlled zones would increase.
- Health and safety education programmes would have to be increased. Recruitment might become a problem.

Which nuclides are relevant?

All the nuclides in the uranium and thorium series are relevant. However, processing disturbs the secular equilibriums. In the phosphate industry the relevant nuclides are mainly Pb-210 and Po-210 which occur in the residues. Where natural gas is concerned, it is the daughter products of radon which are significant. In the other industries, because of its high volatility, Po-210 is the principal nuclide emitted (in waste gas).

All this means that a simple determination of the activity from a whole series would not suffice; the entire nuclide spectrum would have to be determined, and that would involve considerable cost.

Implementation of measures

On the subject of implementation, it was questioned whether the government was itself capable of building an effective supervising structure in such a short term. For companies which already have obligations under the Nuclear Energy Act, the extension will not lead to any significant problems. For companies which as yet have no such obligation, the process of gathering sufficient know-how would take three to four years. Serious capacity problems are anticipated in material certification. There is insufficient capacity as it is, even taking external bodies into consideration. The development of sufficient capacity would be a long and expensive process, partly because of the complicated measurements needed. Material certification would in effect call for a new administrative structure.

Costs associated with implementation

With regard to the costs associated with implementation, the following observations were made:

- Phosphate industry: if slag and fertilizer could no longer be freely sold, it would mean "the end of the industry", with all the consequences that would have for employment.
- Gas/oil extraction: significant rise in operating costs per contaminated installation and an increase in the volume of waste classed as radioactive, with associated processing costs. Considerable secondary costs since additional pretreatment would be required prior to subcontracted maintenance, and loss of production as a result of necessary operational and maintenance strategy changes.
- Energy supply: if fly ash could no longer be recycled, this would mean "the end" for coal-fired power generation.
- Other cost items, in such areas as administration, training, additional facilities, measurements, waste and reduced flexibility are difficult to estimate at the present time. However, it is expected that profitability could be badly affected.

Competitive position

It is feared that the competitive position of Dutch industry would be undermined, not only relative to countries outside Europe, but also relative to other EC countries, due to differences in the way the new regulations are perceived and implemented in the various member states, especially where interpretation of the registration and licensing obligations are concerned.

As the term radioactivity has strongly negative connotations in the minds of the general public, the fear is that the image of companies affected by the proposed Council Directive would be damaged, which would in turn adversely affect the saleability of their products. It is impossible to tell to what extent this would be the case, since it is not known what the level of social acceptance would be.

Support

The indications are that support from the business community for the new Council Directive would probably be limited in view of the limited benefits and high costs. Winning industry over to the new regulations would be considerably easier if:

- the regulations were clear; the excessive caution employed in setting the limits is regarded with suspicion. There is a general willingness to accept the ICRP standards, provided they are applied in a realistic manner.
- a clear section on the financial aspects were included.
- the Nuclear Energy Act (KEW) were adapted to take specific account of the unintentional use of natural radioactive materials in non-nuclear industries; or if special covenants were drawn up.
- it were clear that the restrictive policy pursued did not create additional problems, such as preventing the recycling of residues and necessitating their disposal as (radioactive) waste.

In view of the above, an integral approach to environmental policy is advocated, whereby consideration is given to the effect measures in one policy area will have on other areas. The proposed Council Directive is perceived as counterproductive, precisely because this type of global view is lacking.

7 CONCLUSIONS

The range of materials utilized and consumed by Dutch industry is enormous. A large proportion of these materials either possess higher levels of natural radioactivity in their natural state, or are associated with products or wastes which, as a result of concentration, have higher levels of specific activity. It is not known which materials, products and wastes exceed the limits proposed in the new Council Directive.

The survey reported here has not provided a comprehensive picture of the potential problems associated with the new limits. Nor was this expected, in view of its scope (in relation to the extent of the problem) and the time available for its completion (six months). Only a few companies in each industrial sector could be approached. These companies had however been selected on the basis of their representativeness of their particular branch. In general, the response level from the companies approached was good. Unfortunately, many companies had no information whatsoever about the radioactivity of their raw materials or products. This gap could be filled only in part by conducting measurements.

Nevertheless, a number of general conclusions can be drawn, based in part on the discussions during the workshop.

- In their natural state, a number of raw materials and products possess specific activity levels well below the present limits, but well in excess of the levels at which registration or licensing would be compulsory under the draft Council Directive.

- A large number of materials generally have specific radioactivity levels that are several times less than the levels at which registration or licensing would be compulsory under the draft Council Directive. However, the level of specific activity in raw materials of these types has been known to vary by a factor 10 or more. This means that companies not obliged to register or license would still have to set up inspection systems for incoming goods, to enable them to reject batches with high levels of radioactivity. Some companies reported that it is the batches with the highest radioactivity that possess the best product characteristics.

- The raw materials found to have activity levels above those at which registration or licensing would be compulsory appear in many cases to be difficult to substitute by alternative products with lower activity levels, for both technical and economic reasons.

- While some products contain excessive specific activity levels, individual product units do not also exceed the limit on absolute activity. However, the proposed obligation to report or license would be based on the total radioactivity present at a given location, not on the activity of individual product units. This would mean that, for instance, a shop's stock of incandescent light bulbs could be subject to reporting or licensing, while the product itself could be freely traded.

- If the new regulations were implemented, a considerable volume of waste and residues would be subject to reporting or licensing. If such materials had to be classified as (low) radioactive waste, this would have to be disposed of at high cost.

- The use of newly extracted ores and materials could be made more attractive than the recycling of residues. This would be contrary to the Dutch environmental policy. If raw materials were excluded from the scope of the new regulation, this might lead to the use of raw materials in preference to recycled residues, even where the former carry a greater risk than the latter.

- (Specific) activity should not be the only factor taken into account. The aggregate risk should be considered. Where radon-exhaling materials are concerned, risk is the sum of internal contamination and external exposure to radiation.

- A distinction should be made between the different applications of a given material. The risk associated with the use of fly ash in embankments clearly cannot be compared with the risk associated with its use in building materials.

- Companies would be faced with higher costs and damage to their corporate images, especially in relation to companies outside the EC.

- It would be desirable to have an integral approach to environmental policy, whereby consideration was given to the effect measures in one policy area will have on other areas. The proposed Council Directive is perceived by the affected industries as counterproductive, precisely because this type of global view is lacking.

- Generally speaking, there are doubts as to whether the implementation of the draft Council Directive would lead to an improvement in radiological protection in the non-nuclear industry such that the associated costs could be justified as optimal.

With regard to occupational health and safety, analyses suggest that the radiation hazards associated with working with materials possessing specific activity levels around the proposed exemption levels should not be ignored. However, companies almost always have the facility to take the measures necessary to manage the problem effectively. The reduced dose limits proposed in the draft Council Directive are broadly in line with the policy already pursued in the Netherlands. However, the size of the monitored areas and the number of people classed as radiological workers would increase.

If the proposed exemption levels for the uranium and thorium series are liberalized by a factor 10, many of the objections outlined above would be removed. Considerable doubt has been expressed as to whether the models, scenarios and parameter values employed are suitable for use in relation to the non-nuclear industries.

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THE REGULATORY AND POLICY FRAMEWORK FOR THE RECYCLE OF RADIOACTIVE SCRAP METALS IN THE UNITED STATES OF AMERICA

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Abstract

The issue of safely recycling materials contaminated with low levels of radioactivity is not a new one to the radiation industries and regulators. It has been discussed for decades. Major factors in the recycling issue are the high cost of disposing of these materials as waste and their potential economic value. Unfortunately, regulatory solutions to the problem have not been easily forthcoming. Although the U.S. is among the most significant users of radioactive materials in the world, passing regulations related to de minimis quantities of radiation in the environment is a very difficult and contentious endeavor. A recycling regulation will have to undergo a great deal of scientific and political scrutiny before it is passed and implemented in the U.S. In this paper, we will discuss recycling in the United States from a regulatory and policy standpoint.

INTRODUCTION

The U.S. Environmental Protection Agency (EPA) is a regulatory agency responsible for the administration of a number of environmental laws legislated by the U.S. Congress. The implementation of these laws is accomplished through the development of policies, guidance and regulations for the control of releases of pollutants and hazardous substances to the environment, including radionuclides. All rulemaking must be done with open participation of the American public and must be consistent with laws and policies of the U.S. Government.

The EPA is currently engaged in developing regulations for the cleanup of sites contaminated with radioactive materials. The first phase of this effort is to establish soil, groundwater, and building cleanup levels. Later, the EPA intends to address low level wastes generated through decommissioning and site cleanup. The EPA is evaluating the feasibility of establishing criteria for recycling radioactive scrap metal into general commerce, or for restricted uses. Many questions remain regarding recycling of radioactively contaminated scrap metal that will require resolution before recycling could be implemented in the U.S.

BACKGROUND

The U.S. Government agencies are pursuing the decommissioning and cleanup of many of nuclear research and production facilities. In so doing, the issue of the potential recycling of radioactive scrap metal (RSM) has gained increasing attention.

RSM is mostly relatively low activity surface-contaminated pipe, process components, and structural metal from facilities where radionuclides have been used or produced. These facilities, both Federal and commercial, include uranium fuel cycle facilities, weapons production facilities, nuclear power reactors, various research and development laboratories, and other industries. Large quantities of RSM have been accumulating at several Federal facilities since the days of the Manhattan Project. Over the years, on a site-by-site basis, some accumulated scrap metal was cleaned and/or surveyed and sold to scrap vendors, while other scrap was simply buried.

The RSM issue began to gain attention in the late 1970's with the U.S. Department of Energy's (DOE) Cascade Upgrade and Cascade Improvement Programs at the three U.S. uranium enrichment plants - Oak Ridge, TN, Paducah, KY, and Portsmouth, OH. Scrap metal generated from these programs consisted of approximately 35,000 tons of iron and steel; 9,200 tons of nickel; and 1,600 tons of copper[1]. Since this metal was contaminated with special nuclear and byproduct material (i.e., low-enriched uranium and technetium), scrap metal dealers would not purchase the metal, because specific licenses to handle it might have to be acquired from the U.S. Nuclear Regulatory Commission (NRC).

Therefore, the DOE petitioned the NRC to allow an exemption from licensing of these metals below the concentration limits of 5 ppm Tc-99, and 17.5 ppm low-enriched uranium[1]. The NRC denied the petition and recommended an integrated Federal policy with the U.S. Environmental Protection Agency (EPA) concerning residual levels of radioactivity[2]. Shortly thereafter, the DOE ceased releases of materials from contaminated zones pending a resolution to the problem.

Private oil and gas industries, and some mining operations, are also generators of radioactive scrap metal. Certain operations concentrate radium as a scale on process equipment, such as drilling pipe and casing. The radium-rich scale can reach concentrations as high as 3.7 kBq/g, as was found in some North Sea drilling operations[3]. This began to cause alarm in the U.S. commercial scrap metal recycling industry around 1985. It also became apparent that radiation sources, such as Co-60 and Cs-137 radiography and teletherapy devices, were inadvertently entering the commercial metal recycling stream. Metal melting firms and scrap metal vendors have since been forced to install state-of-the-art detection devices to screen loads of incoming scrap.

Some RSM generated by the commercial nuclear industry has been recycled based on NRC decommissioning criteria (i.e., NRC Regulatory Guide 1.86), or based on limits of detectability of hand-held survey meters. This practice varies from state to state. However, it has long been recognized that consistent national standards would be necessary if RSM recycling were to be implemented nationally.

It appears that the DOE currently has the largest quantity of RSM available for recycle. The major concern, however, is the projected generation of very large quantities of RSM as full-scale decommissioning of government and commercial facilities gets underway.

REGULATORY FRAMEWORK FOR RECYCLING

Any policy, guidance, or rule the U.S. Government may propose concerning RSM recycling must be compatible with other U.S. laws, policies, and guidance. These may be existing environmental laws, general policies, or previous precedents that may apply. Briefly described below are some of the major environmental laws that a RSM recycling rule would have to conform to in the U.S.:

The Atomic Energy Act[4] (AEA), 1954 - Radioactive materials were first regulated in the United States by the AEA, which gave the Atomic Energy Commission (later divided into the Nuclear Regulatory Commission and an agency that later became the Department of Energy) authority to regulate the management, processing and utilization of radionuclides in a manner that protects the public health and the environment. In 1970, Reorganization Plan No. 3 gave the newly established Environmental Protection Agency authority to issue Federal guidance and generally applicable regulations on radiation protection matters[5]. The EPA was also given authority to write regulation for high and low level waste disposal. All radioactive materials licensing matters for AEA-regulated materials have remained with the NRC. Regulations developed by the EPA are binding on all Federal agencies, NRC licensees, and states.

The AEA regulates radionuclides according to three groupings: source material (uranium and thorium), special nuclear material (primarily plutonium and enriched uranium), and byproduct material (radioactive byproducts of exposure to special nuclear materials, and source material mine tailings). Naturally Occurring and Accelerator-produced Radioactive Material (NORM and NARM) are not covered by the AEA. (The radionuclide of particular concern in the NORM category is radium, because it is ubiquitous and highly radiotoxic).

The Comprehensive Environmental Response, Compensation, and Liability Act[6] (CERCLA), 1980 - CERCLA (and subsequent amendments) establishes a program for identifying and cleaning up abandoned hazardous waste sites, and includes a fund which finances the costs of cleaning up sites when no responsible party can be identified. As such, it is commonly called the Superfund Program.

Sites contaminated with radioactivity are typically cleaned up on a site-by-site basis through the Superfund program under broad guidelines established by CERCLA. This requires consensus between EPA, the state, and the responsible party, often using past cleanup levels as precedents. Superfund has established a target risk range for site cleanup of 10E-4 to 10E-6 lifetime risk of fatal cancer.

Both the AEA and the CERCLA authorize the EPA to write generally applicable radiation cleanup standards.

The Resource Conservation and Recovery Act[7] (RCRA), 1976 - RCRA (and subsequent amendments) establishes detailed regulation of hazardous wastes from generation to final disposal. Owners and operators of treatment, storage, and disposal facilities must obtain RCRA permits, and comply with all provisions of the law. Materials covered by the AEA are specifically excluded from RCRA. Naturally occurring, and accelerator-produced radioactive materials, however, may be subject to RCRA regulations.

The Toxic Substances Control Act[8] (TSCA), 1976 - TSCA (and subsequent amendments) regulates the manufacture, distribution in commerce, processing, use, and disposal of chemical substances and mixtures. Materials covered by the AEA are specifically excluded from TSCA. Naturally occurring, and accelerator-produced radioactive materials, however, are covered by TSCA.

The Federal Water Pollution Control Act[9] (FWPCA), 1972 - The FWPCA (and subsequent amendments) protects the nation's water quality chiefly through the use of technology-based effluent limits, the national pollutant discharge elimination permitting system, pretreatment requirements for industrial discharges, and toxicity based water quality standards. A 1976 Supreme Court opinion held that source, special nuclear, and byproduct material are not subject to the Act. (Other radionuclides would seem to be included inasmuch as radionuclides can be defined as a pollutant).

The Clean Air Act[10] (CAA), 1970 - The CAA (and subsequent amendments) protects the nation's air quality through national ambient air quality standards, new source performance standards, and other provisions. Radionuclides are a hazardous air pollutant under Section 112 of the Act. The Act authorized the establishment of the National Emission Standards for Hazardous Air Pollutants (NESHAPs) which specify standards for hazardous emissions, including radionuclides.

The Safe Drinking Water Act[11] (SDWA) - The SDWA (and subsequent amendments) seeks to protect public water supply systems through protection of the groundwater. Any radioactive substances that may be found in water are regulated under the Act.

While provisions exist for the U.S. Government to issue a regulation controlling the recycling of RSM, it is worth noting that some potential complications could arise. For instance, scrap metal from the oil and gas industries, which may be contaminated with naturally occurring radioactive material (NORM), would not be regulated by the AEA, but might be regulated by RCRA and TSCA. This could mean that multiple legislative authorities would need to be relied upon to cover the range of radionuclides or that the standards would not apply to some radionuclides, such as NORM.

RSM decontamination and recycling processes would be subject to several environmental laws and regulations. For example, all radionuclide emissions to the air resulting from decontamination processes would have to meet the National Emissions Standards for Hazardous Air Pollutants[12] (NESHAPs), authorized under the CAA. The use of hazardous substances, such as acids used in decontamination processes, would be regulated under the RCRA and the TSCA. Also, the release of liquid effluent would be subject to the SDWA and the FWPCA.

One of the most important aspects of rulemaking in the U.S. is the nature of Federal and state regulatory authorities. In general, the Federal Government regulates matters that concern the entire country. For example, the Commerce Clause of the U.S. Constitution[13] states that all interstate commerce shall be governed by the Federal Government, while states are empowered to regulate within their own boundaries. Federal preemption of state authority has been a source of dispute in the past.

Where Federal regulations apply, states may still retain some authority. This is especially true in the environmental arena. For example, under the CERCLA, hazardous waste cleanup goals have been set. The state in which the cleanup action is taking place, however, maintains the right to mandate more stringent criteria. Also, the Energy Policy Act[14] of 1992 established the right of states to regulate the disposal of low level radioactive waste and incineration of low level radioactive waste on the basis of radiological hazard (if the NRC were to set a national policy exempting wastes below a specified level from regulation). Exactly what the states regulatory boundaries are with regard to the recycle of RSM is still unclear.

In addition to existing laws and regulations, RSM recycling would have to be compatible with established policy goals and statements of the U.S. Recycling itself is a significant national policy. Also, the policy of waste minimization strongly favors RSM recycling. However, the virtues of RSM recycling may be less obvious when considering, for example, the policy of environmental equity - national, international, and intergenerational.

THE REGULATORY CLIMATE IN THE U.S.

Whether warranted, or not, the American public has very strong feelings about radiation. According to Spencer Weart, Director of the Center for History of Physics, American Institute of Physics, nuclear energy, from bombs and reactors, excites "more emotion and public protest than any other technology." It has been "associated with potent images: not only weapons, but also uncanny scientists with mysterious rays and mutant monsters."[15]

Effectively communicating risk to the public is important for any environmental rulemaking. Risk represented in terms of cancer deaths per million persons may be useful to regulators, but does not mean very much to the public. Comparing risks to scenarios more easily comprehended by the public, such as automobile accident risk, has proven effective in certain cases. However, despite our best efforts to present comparative radiation risk analyses, public anxiety remains.

The reasons for this are very complex, and beyond the scope of this paper. But, one key aspect is how the public views risk in terms of whether it is voluntary or involuntary. For example, it may be claimed that an individual is exposed to far more risk from radon in the home, or from cosmic radiation during airplane flights (voluntary risks), than from that found in recycled metals (imposed risk). The public also tends to fear man-made radiation more than natural radiation. The result is that comparisons that seem logical to radiation professionals fail to be persuasive with the general public. Similar observations have been made with non-ionizing radiation and certain chemical risks.

Notwithstanding the perceptions of radiation danger, public involvement on environmental rule and policy development is very important. In fact, according to the Administrative Procedures Act of 1946[16], regulations cannot be passed in the U.S. without allowing for public participation. The Act requires rulemaking to be open for public notice and comment. Federal Agencies have learned, however, that it is important to go beyond the minimum when proposing a regulation that may be controversial.

The NRC has noteworthy first-hand experience with the importance of properly approaching the public when promulgating a rule. In 1990, the NRC published its intent to establish a policy[17] that was already being applied to site decommissioning. The policy would have set an individual effective dose equivalent level of 100 μSv/yr from residual radioactivity that would be below regulatory concern, or BRC. The BRC policy was opposed by nearly all sectors of the public, generating a large public response.

Two important outcomes are that: 1) we have learned a great deal about how to promulgate a regulation on a potentially controversial subject, 2) the public is now highly sensitized to the concept of BRC as applies to residual radiation levels.

At a recent meeting of the National Council on Radiation Protection and Measures in Washington, DC (April 7, 1993), former NRC Chairman Kenneth Carr recalled the BRC experience. Several key points may be drawn from the discussions:

* **Opposition crystallizes when you approach the final stages.** It is only at the last moment before establishing a new law, or policy, that the opposition unleashes its full battery of resistance. The potential opposition must be involved in the process from the start.

* **It is important to court other waste regulators.** In this case, the states and the EPA were not won over. The NRC found a powerful ad hoc alliance had built up against the policy.

* **A broad-based rule that would apply to many circumstances has less chance of succeeding than a selective rule.** A narrowly applicable rule can more easily be defended, because risk analyses are scenario specific and tend to be more complete. Broad-based rules leave too much to the imagination.

The primary questions raised by the EPA about the BRC policy were that there were not sufficient risk analyses for the wide range of exposure scenarios that could occur, and that questions remained about the magnitude of the dose level selected.

It is plain that RSM recycling is essentially a BRC issue, and it is therefore a difficult one for the EPA. Still, much of the radiation community agrees that the concept is very important, and perhaps unavoidable. Currently, the U.S. regulatory community is extremely cautious with the use of the term "BRC" in relation to any cleanup standards, or other residual radioactivity rulemaking. While generally viewed by the public with both hope and suspicion, the EPA may be in danger of the appearance of collaboration with industries, or not representing the public's best interest, if it does not handle the BRC issue with extreme caution. U.S. Government Agencies can also be sued in Federal court over the legality, or fairness, of regulations that they promulgate. Thus, even if recycling criteria were established, they could be overturned in Federal court.

The NRC is currently drafting new decommissioning criteria for all licensees. The approach they have taken is to hold a series of workshops around the country with all interested parties invited, to discuss the options available. These workshops took place before decisions about the rule were made sending a clear message that they wanted full public input and participation before any rule is written. It also afforded the opportunity to hear first hand what environmental groups, industry representatives, local community advocates, and average citizens thought and felt about the subject.

FREE RELEASE

Free release refers to the complete deregulation of a material for use for any purpose based on established criteria. Establishing free release criteria that adequately protect human health, or release criteria that are acceptable, is the first step in successfully promulgating RSM recycling on a national scale. That is where science ends and politics begins.

In 1989 the EPA drafted proposed regulations[18] for the disposal of low level radioactive waste (LLW). These have been delayed pending resolution of outstanding issues with other Federal agencies. In the draft proposed rule, BRC was intended to apply only to wastes classified as low level waste. The proposed BRC level would allow a less restrictive disposal method for wastes below the proposed level. Such a level could be used as a reference point by other practices, such as for RSM recycling.

Currently, what comes the closest to free release criteria for RSM recycling in the U.S. is Regulatory Guide 1.86 (RG 1.86), Termination of Operating Licenses for Nuclear Reactors[19]. The NRC, which oversees the licensing of nuclear facilities, produced RG 1.86 in 1974 (the NRC was called the U.S. Atomic Energy Commission at that time). Table I. of the Guide (see Fig. 1) gives levels of radioactivity per surface area that a licensee would have to achieve before the property could be released for unrestricted use. These levels have been used to decommission facilities from 1974 to the present.

Though intended only for license termination and the release of buildings for unrestricted use, RG 1.86 levels have been used as acceptable free release criteria for numerous other applications, including for recycling RSM. RG 1.86, however, was not developed to provide criteria for the recycle of metals with low levels of radioactivity into general commerce. One reason why, is that the levels established are surface limits only, not volumetric.

The usefulness of RG 1.86 as a general release criterion is limited by the fact that the levels prescribed were determined by the limits of dectability at that time, not by potential health effects from the residual radioactivity. Today, when such criteria are set, extensive modelling is done to demonstrate that the proposed criteria are protective of human health and the environment. The NRC is re-evaluating their decommissioning criteria, including the levels set in RG 1.86.

There are few other free release (or de minimis) precedents for residual radiation in the U.S. Perhaps the only true radiation free release levels are those set internally by the US Navy. The Navy has set limits for the free release of materials at 100 cpm above background, for beta-gamma emitters, and no detectable alpha decay. A nonradioactive de minimis precedent in the U.S. is the PCB rule[20] (polychlorinated biphenyl), which designated a lower limit below which the material in question would be considered free of PCB, and was therefore not regulatable. The approach is essentially BRC, in that there is a concentration level at which regulators determined that a given

Figure 1., NRC Regulatory Guide 1.86, Table I

TABLE I

ACCEPTABLE SURFACE CONTAMINATION LEVELS

NUCLIDE[a]	AVERAGE[bc]	MAXIMUM[bd]	REMOVABLE[be]
U-nat., U-235, U-238, and associated decay products	5000 dpmα/100 cm^2	15,000 dpmα/100 cm^2	1000 dpmα/100 cm^2
Transuranics, Ra-226, Ra-228, Th-230, Th-228, Pa-231, Ac-227, I-125, I-129	100 dpm/100 cm^2	300 dpm/100 cm^2	20 dpm/100 cm^2
Th-nat., Th-232, Sr-90, Ra-223, Ra-224, U-232, I-126, I-131, I-133	1000 dpm/100 cm^2	3000 dpm/100 cm^2	200 dpm/100 cm^2
Beta-Gamma emitters (nuclides with decay modes other than alpha emission or spontaneous fission) except Sr-90 and others noted above.	5000 dpmßγ/100 cm^2	15,000 dpmßγ/100 cm^2	1000 dpmßγ/100 cm^2

[a]Where surface contamination by both alpha- and beta-gamma-emitting nuclides exists, the limits established for alpha- and beta-gamma-emitting nuclides should apply independently.

[b]As used in this table, dpm (disintegrations per minute) means the rate of emission by radioactive material as determined by correcting the counts per minute observed by an appropriate detector for background, efficiency, and geometric factors associated with the instrumentation.

[c]Measurements of average contamination should not be averaged over more than 1 square meter. For objects of less surface area, the average should be derived for each such object.

[d]The maximum contamination level applies to an area of not more than 100 cm^2.

[e]The amount of removal radioactive material per 100 cm^2 of surface area should be determined by wiping that area with dry filter or soft absorbent paper, applying moderate pressure, and assessing the amount of radioactive material on the wipe with an appropriate instrument of known efficiency. When removable contamination on objects of less surface area is determined, the pertinent levels should be reduced proportionally and the entire surface should be wiped.

concentration constitutes an acceptable level of risk. But, the PCB rule also defines the material to be PCB free and is therefore not regulatable by definition.

Widely accepted free release criteria have been actively pursued both in the U.S. and abroad. International scientific and professional organizations concerned with standards for radiation protection, such as the International Atomic Energy Agency (IAEA), have provided recommendations and guidance for the protection of the public. The IAEA recently published Safety Series No. 111, "Application of Exemption Principles to the Recycle and Reuse of Materials from Nuclear Facilities[21]." This report applies earlier IAEA guidance (Safety Series No. 89[22], individual effective dose equivalent of 10 μSv/yr) to the development of radiological criteria for recycling contaminated materials. It is intended to be guidance for regulatory authorities in IAEA member states in the setting of standards.

The recommendations of independent organizations are very important to achieving consensus. Consensus is a process that must take place in the scientific community, the regulatory community, and the public community. All of the national and international work that has been done will play an important role in the U.S. response to the problem. But, it may be that in the U.S., the numbers are not the problem, the concept is.

RSM RECYCLING POLICY CONSIDERATIONS

For full scale RSM recycling to be established in the U.S. using free release criteria, a number of important policy questions must first be addressed. The questions revolve around two central themes: who, or what, could be hurt by RSM recycling? and, what is to be gained? The EPA is currently evaluating the various costs and benefits of recycling RSM belonging to the DOE.

Without question, there are quantifiable benefits from RSM recycling, and the benefits may be significant. The decommissioning of all of the DOE nuclear facilities could generate tens of millions of tons of steel and other metals with a potential market value in the billions of dollars. Currently, the cost of disposing of this metal as low level waste is about $200/cubic foot. Depending on decontamination and associated costs, RSM recycling could save substantial amounts of money, all of which would otherwise be paid for by the public, either through taxes or increased costs for utilities or consumer products.

Metals from decommissioning are valuable national resources that, if not recycled, most likely would be procured by one means or another. To avoid mining the equivalent amount of metal would be desirable, given the environmental damage that mining causes. Recycling metal also uses considerably less energy than producing metal from ore, especially in the case of aluminum, where recycling takes about one fourth the amount of energy.

However, the recycling of RSM is an irreversible action. Unlike low-level waste, for instance, which can be dug up if the regulations are later determined to be inadequate, metals that are released, with their radioactive components, are gone forever from any control. Once the metal pool has radioactive materials in it, they cannot reasonably be extracted.

Because of the irreversibility of recycling, particular care must be taken with regard to risk modelling and risk scenarios. Risk assessments have been done for several consumer use scenarios, but they are quite generic when one considers the extreme diversity of the uses of metals such as stainless and carbon steel, copper, and aluminum. It is necessary to also examine plausible, if conservative, scenarios as well. It could be assumed, for example, that a merchant marine could be exposed over a lifetime to a 20,000 ton ship made entirely of steel with the maximum allowable concentration. Such scenarios may be considered extreme, but they may also be very realistic. Unless the free release criteria are truly valid for any real scenario, deregulating RSM may be premature.

It is also recognized that risks must be viewed comprehensively. From the standpoint of total deaths averted, one has to consider, on the one hand, the radiological risks to the public from recycled metal and all of the associated process risks (including worker exposures, waste handling and disposal, transportation, etc.), and on the other hand, the risks associated with burying the same amount of contaminated metal with the associated risks (RSM handling and transportation, disposal, mining costs and risks, processing an equivalent amount of ore, etc.). Since ore mining is a particularly hazardous occupation, the result may be that more lives would be lost by not recycling, due to the additional mining necessary if that metal were to be replaced in the market.

The issues we should consider are not limited to individual and collective national health risks alone. We must also consider "environmental equity" as well. The nature of world trade leads one to believe that other countries would soon be receiving radionuclides from countries that recycle RSM in the form of manufactured products without their awareness. An extreme example occurred in 1984 when a Mexican steel company exported products to the U.S. contaminated with Co-60. In this case, a large cobalt source inadvertently entered the recycling stream at a Mexican mill and was incorporated into the finished products. If it hadn't been incidentally detected, the accident might never have been discovered[23].

By and large, the net flow of products is from developed countries to underdeveloped countries. This fact could leave the U.S. vulnerable to charges that it is engaging in an inequitable practice at the expense of disadvantaged nations. Equity is also an issue from an intergenerational standpoint. While the benefits of recycling are mostly received by the current generation, the price, in terms of health impacts, would be paid by subsequent generations.

Commercial considerations are also relevant. The U.S. should evaluate very closely the possible effects not only on human health and the environment, but also on radio-sensitive industries. Advancements in integrated circuit, photographic film, and microchip technology are increasing their susceptibility to the effects of radiation. The cost of obtaining truly "clean" raw materials could drive the cost of certain products disproportionately high. Some American and Japanese companies have already been struggling with the very low levels of residual radioactivity in commercially available steel. In one case in Japan, pieces of steel from an old sunken battleship were recovered to make shielding for a whole-body counter[24] because other available steel contained too much residual radioactivity. Early evidence from work the EPA is doing on recycling indicates that certain industries could suffer from additional radioactivity in metals. It is conceivable that an imprudent recycling policy could limit the growth of our own industries.

CONCLUSIONS

Any U.S. policy on RSM recycling must consider a variety of existing laws, regulations, and policy goals. Policies, in particular, must be closely examined for what they intend to accomplish. Recycling RSM appears to be consistent with several well established U.S. policies, such as waste minimization, but it is not clear that the policy of environmental equity will readily accommodate recycling.

The U.S. regulatory community received a clear message from the NRC's experience in attempting to establish a "below regulatory concern" policy. States are protective of their rights to govern: their message was for the U.S. government not to preempt state authority. Public feelings regarding radiation are very deeply set and are, for the most part, negative. The message was, that the public wants radioactive materials regulated. In the United States, the public is actively involved in Federal rulemaking. If states and the public are not amenable to the concept of recycled metal with low levels of radiation, no matter how low those levels may be, no standards are likely to be achieved. Thus, the political realities could derail what to many might seem to be very stringent and protective free release levels, simply because the "concept" was wrong.

The obvious benefits of RSM recycling do not necessarily overshadow the potential consequences. Though there may be significant economic advantages to recycling RSM, and there may be environmental and even risk reduction advantages, questions remain on appropriate modelling of recycled metal use scenarios.

To adequately support a RSM recycling policy in the U.S., the approach must include; 1) explicit risk goals that are substantially below standards currently set in specific radiation protection regulations, 2) detailed and comprehensive risk modelling of conservative, but real, exposure scenarios, and, 3) careful and early involvement of key constituents.

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DERIVED EXEMPTION AND CLEARANCE LEVELS

(Session III)

Chairman

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THE DERIVATION OF UNCONDITIONAL CLEARANCE LEVELS FOR RADIONUCLIDES IN SOLID MATERIALS

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Abstract

In December 1992 the IAEA convened an advisory group meeting with the aim of recommending a set of unconditional exemption levels relevant to solid materials contaminated with, or containing, radionuclides (now called clearance levels).

This paper will report on the results of that meeting. It will describe the methodology used to derive clearance levels from the results of radiological assessment studies related to landfill disposal and incineration of wastes and to the recycling and reuse of metals and concrete.
It will also recall the general exemption principles and explain the change of terminology in the light of the IAEA Basic Safety Standards, which are currently under revision.
Finally, the recommended clearance levels will be discussed in the framework of a regulatory system.

INTRODUCTION.

The environment contains naturally occurring radionuclides and exposures to these radionuclides cannot be avoided. In fact, almost everything must be considered as radioactive material from a strict physics point of view.
However, in most cases, the radioactive material content is very low and in practice the management and control of very low level radioactive materials is restricted to materials which become contaminated or activated as a result of industrial processing. Exposures from natural background radiation are mostly not amenable to control and are excluded from the regulatory control system which involves notification, registration and licensing of practices and sources.

Some practices involving sources of ionising radiation may present such a low risk that it would be a waste of resources to exercise control by regulatory processes. These sources are said to be exempted from regulatory control; the corresponding levels of radioactivity or radioactivity concentration are called exemption levels (also reporting levels).
It may also happen that radioactive materials which are subject to control have decayed or have been processed in such a way that their use no longer presents a radiological risk and therefore they may be removed from the radiological protection system, since it would be a waste of time and effort to continue supervision. This is called clearance and the corresponding levels of radioactivity or radioactivity concentration are called clearance levels.

The clearance may be restricted to certain conditions or specific uses or management routes e.g. to recycling or reuse, to landfill disposal or to incineration; this is called "conditional clearance". The term "unconditional clearance" is used when there are no restrictions on the use of the material after clearance.

The clearance of very low radioactive material is particularly important in the case of decommissioning and dismantling of nuclear installations, or after large repair operations, where large quantities of low active or slightly contaminated material are produced which are suitable to be treated so as to reach very low residual activity levels and to be cleared.

Materials cleared in one country are no longer distinguishable from non radioactive material and may be moved from one country to another. It is therefore necessary to develop an approach and to derive clearance/exemption levels which are agreed upon on an international level.

PRINCIPLES FOR EXEMPTION.

In 1988 an international (IAEA, NEA) consensus was reached on the general principles for exemption from radiological protection measures. (1)
The exemption of a practice or a source from regulatory control (notification, registration, licensing) must be seen in relation to the basic radiological protection principles : justification of a practice, optimisation of protection, individual risk and dose limits.
A "practice" is defined as "a set of co-ordinated and continuing activities involving radiation exposure which are aimed at a given purpose, or the combination of similar such sets".
The "source" is then defined as "the physical entity (e.g. radioactive material, nuclear installation) whose use, manipulation, operation, decommissioning and/or disposal are constituents of the coordinated set of activities defined as practice".

From a radiological protection standpoint, there are two basic criteria for determining whether or not a practice can be a candidate for an exemption :

- the individual risks must be trivial, i.e. sufficiently low as not to warrant regulatory concern;

- the radiological protection must be optimised, taking the cost of regulatory control into account.

An individual dose is likely to be regarded as trivial if it is of the order of some tens of microsieverts per year. Because an individual may be exposed to radiation from several exempt practices, it is reasonable to apportion a fraction to each exempt practice. This could lead to individual doses to the critical group of the order of 10 μSv in a year from a single practice.

In the optimisation assessment, the relevant quantity is the collective dose commitment per year of practice. A generic study of the available options should be made and the conclusion reached that exemption is the option that optimizes protection. If this generic study indicates that the collective dose commitment from one year of the unregulated practice will be less than about 1 man.Sv, it may be concluded that the total detriment is low enough to permit exemption without more detailed examination of other options.

In its 1990 recommendations, the International Commission on Radiological Protection (ICRP) recognizes "that the exemption of sources is an important component of the regulatory functions". (2) The Commission reiterates the two basic criteria for exempting a source or an environmental situation from regulatory control. One is that the source gives rise to small individual doses and small collective doses in both normal and accident conditions. The other is that no reasonable control procedures can achieve significant reductions in individual and collective doses.

Although the general principles were established for "exemption", it is clear that these same principles also apply to "clearance".

APPLICATION OF EXEMPTION PRINCIPLES.

Unregulated practices give rise to small individual doses (10 or some tens of micro Sv per year), which are not measurable in practice. Therefore, the exemption criteria in terms of dose must be converted to more practical and measurable quantities, such as radioactivity concentration levels in Bq/g or surface contamination levels in Bq/cm^2.

Several dose assessments of conditional clearances have been performed by national and international organisations. In these assessments, various scenarios and exposure pathways are considered.

In December 1992, an advisory group met at the IAEA in Vienna to derive and to recommend unconditional clearance levels for solid materials containing radionuclides, on the basis of studies which had been directed towards the low activity streams of material generally considered to be the most likely candidates for exemption/clearance. These are :

- low level solid wastes from the nuclear fuel cycle;
- slightly contaminated metals and concrete which may arise in the decommissioning of nuclear installations;
- low level wastes generated during the application of radioisotopes in industry, hospitals and research laboratories.

In these studies, potential individual radiation exposures were evaluated for a range of scenarios linked to each of the practices considered.

The main scenarios considered are the following :

Landfill disposal (3)-(9) (15)	- transport workers
	- landfill site workers
	- disturbance of the site after closure
	- radionuclide transfer via groundwater
	- fires in the landfill
Incineration (3)-(7) (15)	- operators
	- emissions
	- ash (to landfill)

Recycling (steel) - scrap transport workers
(7) (8) (10) (11) (15)

- scrap processing workers
- workers at smelter, and fabrication plant
- consumer use
- emissions
- use of slag

(similar groups of scenarios are considered for non-ferrous metals (7) (10) (12) and concrete (7) (8) (10) (13) (15))

Reuse - small tools and equipment
(10) (11) (13) (14) (15)

- large equipment
- buildings (use and renovation)

The evaluation of radiation exposure in each of the scenarios takes account, as necessary, of exposure due to external irradiation, and to inhalation and ingestion of radionuclides.

The individual dose criterion used as the basis for the unconditional clearance levels was 10 μSv per year, and nuclide specific values of clearance levels, in terms of Bq/g or Bq/cm^2 were obtained from each of the dose assessment scenarios considered in the reviewed studies ([1]).

By this means, it was possible to produce a range of clearance level values for each radionuclide considered. This range reflects the range of assumptions made by each of the assessment groups involved.

The general criteria for accepting the data from the assessment studies was that the data sets should be based on credible exposure scenarios which have a reasonable likelihood of occuring and with realistic assumptions concerning transfer parameters, exposure times etc.

Some assessment or components of them were judged to be :

a) incomplete, with important exposure scenarios missing

b) not credible, because of the nature of the postulated exposure scenarios

c) unrealistic, because of the assumptions made.

The results of these studies were, therefore, excluded from further consideration and were not included when constructing the range of values.

([1]) In some of the studies, other dose criteria were used [50 μSv for small groups of exposed individuals] to derive clearance levels. Where possible, the published data have been normalized to the 10 μSv criterion.

A summary of the considered data is given in table 1 and visualized in figure 1.

A procedure was devised for determining unconditional clearance levels from the ranges of values obtained from the published reports. The aim of the procedure was to select values which would provide a high degree of assurance that doses from likely scenarios would not exceed 10 μSv per year. At the same time it was considered desirable to avoid having the final values determined by unlikely scenarios. The procedure allows doses from unlikely scenarios of up to about 100 μSv per year.

Given the inherent inaccuracies of the results of these assessments, e.g. because of the use of parameters of which only the order of magnitude is more or less known, the radionuclides have been grouped according to the order of magnitude of the derived clearance level.

It was proposed to use the geometric mean of lower and upper bounds of the groups as recommended clearance level (for example : 3 for the group "1 to 10").
The result of the analyses and the deliberations is given in table 2.

It can be seen that the lowest clearance levels are found for high energy gamma-emitters and for alpha-emitters.

The recommended clearance levels are intended as average values for medium sized items, surfaces (less than a few m^2) or amounts of material (less than a few hundred kg).

For radionuclides that are not listed in table 2, it was proposed to use the following formula to find the group :

$$\text{Minimum} \left\{ \frac{\text{ALI (inhalation)}}{1000\ \text{Bq}} \ , \ \frac{\text{ALI (ingestion)}}{100\ 000\ \text{Bq}} \ , \ \frac{1\ \text{MeV}}{E_\gamma + 0{,}1\ E_\beta} \right\}$$

where ALI (inhalation/ingestion) = the annual limit on intake for inhalation/ingestion (ICRP-61) (16)

E_γ/E_β = effective gamma/beta energy per desintegration (ICRP-38) (17)

A comparison of the levels obtained by using the formula and the derived levels in table 2, is given in table 3.

The group also recommended to use the same numerical values for the activity concentration levels (in Bq/g) and for the surface contamination levels (in Bq/cm^2).

FINAL CONSIDERATIONS.

It is important to note that the values in Table 2 apply to unconditional clearance. The derivation of unconditional clearance levels must necessarily take into account radiation exposure during all of the reasonably possible uses and movements of the materials intended for clearance. For a given radionuclide, the derived clearance level will be determined by the scenario and exposure pathway which give rise to the highest radiation dose. When the practice which is a candidate for clearance

Table 1 - Derivation of unconditional clearance levels (activity concentration)

Radionuclide	DATA FROM STUDIES																					DERIVED LEVEL
	Landfill disposal					Incineration					Recycling								Reuse		Uncond Clear	
											Steel		Aluminium		Copper	Concrete		All				
	IAEA 87	Muller-Neumann 88	Poschner 91	Sumerling-Sweeney87	Guetat 92b	IAEA 87	Muller-Neumann 88	Poschner 91	Sumerling-Sweeney87	Guetat 92b	IAEA 92	CEC 88	IAEA 92	Garbay 91	Garbay 91	IAEA 92	Haristoy 92	Guetat 92b	IAEA 92	NUREG 90	Guetat 92a	
H 3		2000		4000	4000	700000	1000000		40000	30000							800000	10000		100000	10000	3000
C 14	100	600		100	800	900	3000		100	1000							7000	2000		4000	1000	300
Na- 22	0.6	2	0.2		0.6	0.6	10	2		1							1	0.7		0.5	3	0.3
Na 24					0.3					2000							INF	0.6		0.3	2	0.3
P- 32	1000	3000	10000	700	1000	8000	20000	80000		900							200000	200		200	10000	300
S- 35	5000	5000	2000		10000	30000	400000	200000		2000							30000	30000		20000	200000	3000
Cl- 36					6					400						20000	1000	100		1000	20000	30
Ca- 45	200000	8000			8000	20000			4000	1000							8000	3000		2000	70000	3000
Cr- 51		100		300	40					100							6000	70		30	200	30
Mn 54	2	4			1	2				4	0.4	10	1	20	60	1	4	0.2	4	1	7	0.3
Fe- 55					600000					100000	10000	200000	1000	200000	200000	200000	20000	400	900	2000	200000	300
Fe 59		3			1					3							30	2		1	5	3
Co- 57		30			10					30							60	3		10	50	30
Co- 58		40			1					3							20	1		1	6	3
Co- 60		1		0.1	0.5	0.5	90			1	0.1	2	0.3	5	20	0.3	0.7	0.06	1	0.4	2	0.3
Ni- 63		10000			50000		400000			90000	20000	200000	40000	1000000	50000	100000	20000	8000	20000	10000	100000	3000
Zn- 65		6			2					5	0.6		2	30	80	2	6	0.5	6	2	10	0.3
Sr- 89		4000			4000					700							3000	60		300	10000	300
Sr- 90 +		2		30	10	80	50			300	50	40	200	800	300	300	90	7	70	80	2000	3
Y- 90		20000			1000					4000							1.0E+14	300		100	7000	300
Nb- 94		2			0.7					2	0.2		0.5	7	30	0.5	1	0.07	2	0.7	4	0.3
Tc- 99 m				90	20				2.0E+10	1.0E+10							INF	20		10	80	30
Tc- 99					40		6			1000	7000		20000			50000	3000	2000	9000	2000	60000	30
Ru- 106 +	6	200			6	8				20				40	70		10	0.9		5	30	3
Ag- 110 m+		1			0.4					1				6	20		1	0.09		0.4	2	0.3
Cd 109		400			200		8000			500							200	10		200	900	300
In- 111					3					50							4.0E+12	6		4	20	3
Sb- 124		2			0.7					2							7	0.9		0.6	3	0.3
I 123					10					200000							INF	20			50	30
I- 125		700		300	100				1000	300							3000	30		80	50	30
I 129					50		4			70							200	10		30	800	30
I 131	4	9		30	3	8	100		2000	20							400000	6		3	20	3
Cs- 134		2			0.8					2		5		10	30		1	0.09		0.7	4	0.3
Cs- 137 +		6		20	2	3	40			5	0.5	8	1	20	90	1	3	0.2	4	2	10	0.3
Ce- 144 +	100	100			20	90				60							70	6		70	100	30
Pm 147		4000		20000000	10000					1000							2000	500		5000	10000	3000
Eu- 152		3			1					3	0.4		1	10	50	1	1	0.1	4	0.9	5	0.3
Ir- 192					2					4							20	1		1	8	3
Au- 198					3					50							7.0E+12	6			20	3
Tl- 201					20					200							6.0E+12	30			80	30
Pb 210 +			10		0.2			100		4							7	0.7		1	30	0.3
Po- 210			20		200			100		40							20	3		4	50	3
Ra- 226 +			0.3	0.1	0.7			200		2							0.6	0.07		7	6	0.3
Ra- 228 +			0.3		1			300		3							2	0.1		8	7	0.3
Th 228 +			0.3		0.8			4		1							0.2	0.06		2	1	0.3
Th- 230			0.5		8			4		2							0.3	0.1		2	1	0.3
Th- 232			0.1	10	2			0.8		0.4							0.08	0.02		0.5	0.3	0.3
U- 234			1		10			10		3				50	7		0.5	0.2		6	3	0.3
U- 235 +			1		8			10		3				30	7		0.5	0.2		4	3	0.3
U- 238 +			1	10	10			10		3	1		5	50	7	3	0.5	0.2	1	6	4	0.3
Np- 237 +		0.3			3					0.7				10	2		0.2	0.04		0.7	1	0.3
Pu- 239	0.6	0.3		400	6	0.2	3			1	0.3	0.1	1	10	2	90	0.3	0.08	0.3	2	1	0.3
Pu- 240		0.3			6					1							0.3	0.08		2	1	0.3
Pu- 241 +					100					30	10		70			50	20	2	20	100	40	30
Am 241	0.6	0.3		200	6	0.2				1	0.3	0.1	1	10	2	0.9	0.3	0.02	0.3	0.9	1	0.3
Cm 244		0.6			10					2		0.2		30	3		0.4	0.1		2	2	0.3

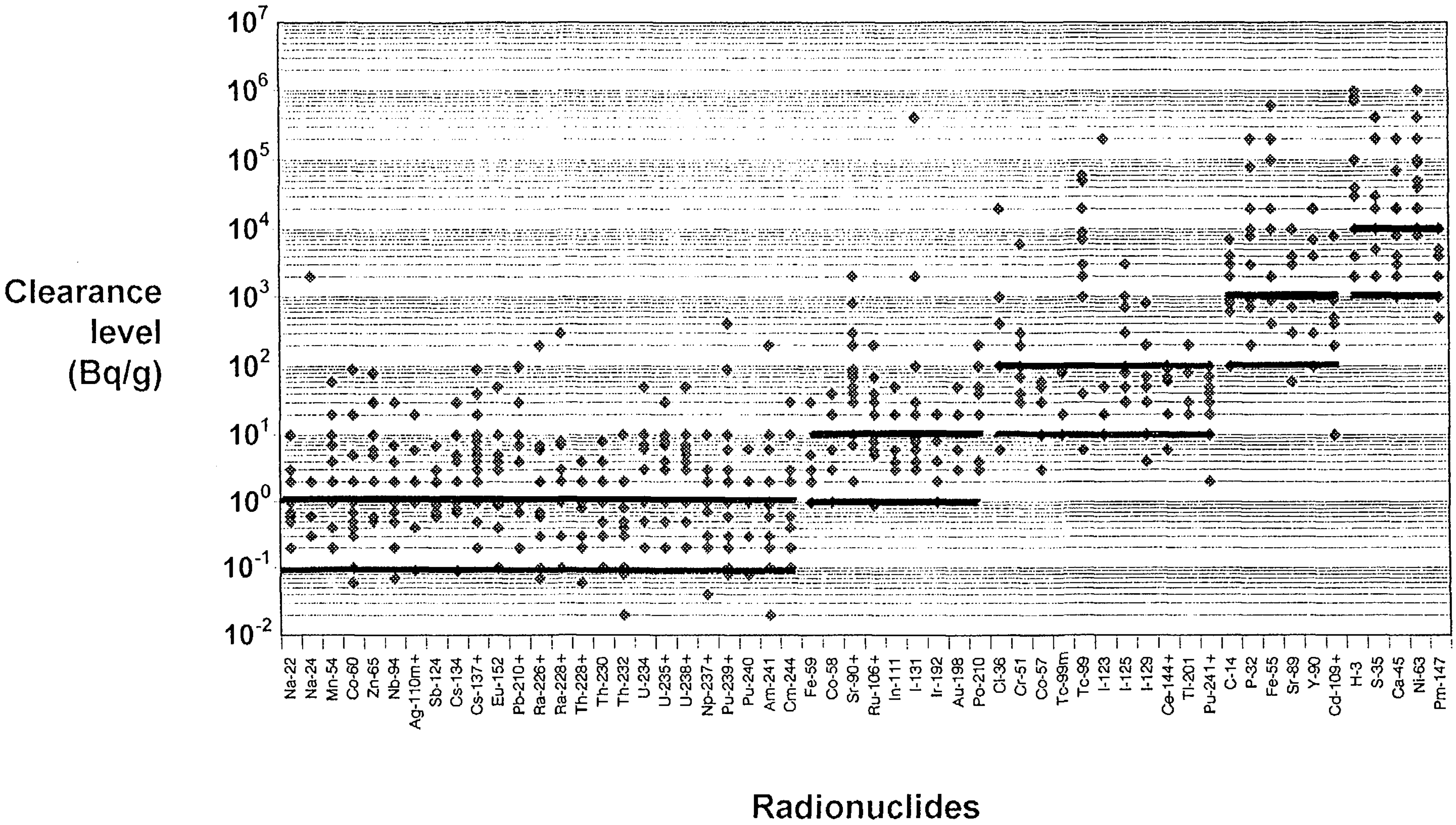

FIG. 1. Derivation of clearance level

TABLE 2 - RECOMMENDED UNCONDITIONAL CLEARANCE LEVELS

Ranges of Activity Concentration (Bq/g)	Radionuclides			Representative Single Values of Activity Concentration (Bq/g)
0.1				
	Na-22	Cs-137	U-234[1]	
	Na-24	Eu-152	U-235	
	Mn-54	Pb-210	U-238[1]	
	Co-60	Ra-226[1]	Np-237	0.3
	Zn-65	Ra-228	Pu-239	
	Nb-94	Th-228[1]	Pu-240	
	Ag-110m	Th-230[1]	Am-241	
	Sb-124	Th-232[1]	Cm-244	
	Cs-134			
1.0				
1.0				
	Fe-59	Ru-106	Ir-192	
	Co-58	In-111	Au-198	3
	Sr-90	I-131	Po-210	
10				
10				
	Cl-36	Tc-99	Ce-144	
	Cr-51	I-123	Tl-201	30
	Co-57	I-125	Pu-241	
	Tc-99m	I-129		
100				
100				
	C-14	Fe-55	Y-90	
	P-32	Sr-89	Cd-109	300
1000				
1000				
	H-3	Ca-45	Pm-147	
	S-35	Ni-63		3000
10000				

Note 1. Specific exposure to Rn-220 and Rn-222 was not considered in this classification.

is well defined, e.g. landfill disposal, it will usually be possible to take account of the known features of the practice. These considerations may be expected, in general, to lead to higher clearance levels as compared to the unconditional levels.

From a regulatory viewpoint, it is also necessary to be able to verify the applicable clearance levels. This can be done by direct measurement on the material to be cleared, by laboratory measurements on representative samples, by use of properly defined scaling factors or by other means which are

Table 3
Formula for unconditional Exemption Levels

Radionuclide	DERIVED LEVEL	FORMULA
Na- 22	0.3	0.3
Na- 24	0.3	0.3
Mn- 54	0.3	3
Co- 60	0.3	0.3
Zn- 65	0.3	3
Nb- 94	0.3	0.3
Ag- 110 m+	0.3	0.3
Sb- 124	0.3	0.3
Cs- 134	0.3	0.3
Cs- 137 +	0.3	3
Eu- 152	0.3	0.3
Pb- 210 +	0.3	0.3
Ra- 226 +	0.3	0.3
Ra- 228 +	0.3	0.3
Th- 228 +	0.3	0.3
Th- 230	0.3	0.3
Th- 232	0.3	0.03
U- 234	0.3	0.3
U- 235 +	0.3	0.3
U- 238 +	0.3	0.3
Np- 237 +	0.3	0.3
Pu- 239	0.3	0.3
Pu- 240	0.3	0.3
Am- 241	0.3	0.3
Cm- 244	0.3	0.3
Fe- 59	3	0.3
Co- 58	3	3
Sr- 90 +	3	3
Ru- 106 +	3	3
In- 111	3	3
I- 131	3	3
Ir- 192	3	3
Au- 198	3	3
Po- 210	3	0.3
Cl- 36	30	30
Cr- 51	30	30
Co- 57	30	3
Tc- 99 m	30	3
Tc- 99	30	30
I- 123	30	3
I- 125	30	30
I- 129	30	3
Ce- 144 +	30	3
Tl- 201	30	30
Pu- 241 +	30	30
C- 14	300	300
P- 32	300	30
Fe- 55	300	300
Sr- 89	300	30
Y- 90	300	30
Cd- 109	300	30
H- 3	3000	3000
S- 35	3000	300
Ca- 45	3000	300
Ni- 63	3000	300
Pm- 147	3000	300

accepted by the competent authorities. Of course, the choice of measurement strategy and appropriate instruments will depend upon the type of material and radionuclides present.

As a final remark, the clearance concept in the context of naturally occurring radionuclides will usually only be relevant in cases where some enhancement has taken place. In these cases, the problem is in establishing the "normal" level in the workplace or the environment and in applying the clearance criteria to levels in excess of the "normal" level.

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CONTROL CRITERIA FOR RESIDUAL CONTAMINATION IN MATERIALS CONSIDERED FOR RECYCLE AND REUSE

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Abstract

Pacific Northwest Laboratory (PNL) is collecting data and conducting technical analyses to support the U.S. Department of Energy (DOE), Office of Environmental Guidance, Air, Water, and Radiation Division (DOE/EH-232) in determining the feasibility of developing radiological control criteria for recycling or reuse of metals or equipment containing residual radioactive contamination from DOE operations. The criteria, framed as acceptable concentrations for release of materials for recycling or reuse, will be risk-based and will be developed through analysis of radiation exposure scenarios and pathways. The analysis will include evaluation of relevant radionuclides, potential mechanisms of exposure, and non-health-related impacts of residual radioactivity on electronics and film. The analyses will consider 42 key radionuclides that are generated during DOE operations and may be contained in recycled or reused metals or equipment.

1. INTRODUCTION

Pacific Northwest Laboratory (PNL) is collecting data and conducting technical analyses to support joint efforts by the DOE, the U.S. Environmental Protection Agency (EPA), and the U.S. Nuclear Regulatory Commission (NRC) to develop radiological control criteria for the recycling and reuse of scrap materials and equipment that contain residual radioactive contamination. The initial radiological control criteria are the concentrations in or on materials considered for recycling or reuse that meet the individual or industrial (electronics/film) dose criteria. The analysis includes determining relevant radionuclides, potential mechanisms of exposure, and methods to determine possible non-health-related impacts from residual radioactive contamination in materials considered for recycling or reuse.

The data and models described in this paper may be considered by DOE (in coordination with other U.S. Federal agencies) with other information to set radiological control criteria for recycling that are as low as reasonably achievable (ALARA) and to support environmental regulations. To determine if recycling is the "preferred" action or approach for management of material, DOE has identified two criteria. The action must be 1) environmentally acceptable and cost effective, or 2) environmentally preferred. Under this approach it is recognized that some situations exist under which the direct costs may be higher for recycling than for burial, but environmental costs avoided by recycling balance the short-term costs associated with the recycling activity (e.g., the recycling option reduces environmental insults associated with certain secondary impacts).

The preliminary results described in this paper are based on generic exposure scenarios and pathway analyses using 42 radionuclides determined to be

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potentially present as residual contamination in metals or equipment from DOE operations. The scenarios and information developed by the International Atomic Energy Agency (IAEA) in Safety Series No. 111-P-1.1 (1992), Application of Exemption Principles to the Recycle and Reuse of Materials from Nuclear Facilities [1], were considered in developing the initial radiation exposure pathway and scenario analysis. Additional analyses were conducted to determine the potential non-health-related impacts industry may experience from residual contamination in recycled metals, such as those used in industries producing and using electronics and X-ray film.

Although alternative public dose limits were considered, the initial control criteria in this report are based on 1) a dose of 10 μSv y^{-1} (1 mrem y^{-1}) to a worker in a smelter or to an individual who uses consumer products made from recycled materials, 2) a dose of 1 μSv y^{-1} (0.1 mrem y^{-1}) to an individual downwind from a smelter used to process recycled metals, and 3) minimizing non-health-related impacts associated with potential radiation effects on electronics or film.

2. DOSE ASSESSMENT METHODS

To determine if radioactively contaminated materials can be released from regulatory controls, it is necessary to first determine the potential future uses for the materials and then the potential radiation doses resulting from those future uses. Generic radiation exposure scenarios were used to conceptually model likely future uses for recycled materials. These scenarios are a combination of radiation exposure pathways that contain specific exposure conditions. This section contains a summary of the basic radiation exposure pathways, scenarios, and methods used to estimate the preliminary control criteria for recycling or reuse of materials.

2.1. General Assumptions

For the preliminary calculations that follow, it was assumed that 100 t of contaminated steel, aluminum, and concrete and 10 t of copper were recycled during a year. This assumption was intended to lead to the development of bulk contamination control criteria. For the development of surface contamination control criteria, individual tools or pieces of equipment for reuse were considered. A unit concentration of each radionuclide was assumed and control criteria were derived in terms of Bq g^{-1} for volume and Bq cm^{-2} for surface contamination. For the scenario calculations that follow, 40 reference radionuclides (plus two additional concrete activation products) were selected. The radionuclides considered and their physical half-lives are listed in Table I.

The choice of the 42 radionuclides also takes into account other considerations, among them the following:

- the origin of the radionuclides; whether natural uranium (^{238}U), uranium activation products (^{239}Pu, ^{241}Pu, and ^{241}Am), fission products (^{90}Sr, ^{99}Tc, and ^{137}Cs), or activation products (^{36}Cl, ^{41}Ca, ^{54}Mn, ^{55}Fe, ^{60}Co, ^{63}Ni, ^{65}Zn, ^{94}Nb, ^{99}Tc, and ^{152}Eu)
- the half-life of the radionuclides; whether relatively short (^{65}Zn) or very long (^{94}Nb or ^{239}Pu)
- the importance of the radionuclides in the context of bulk activation or surface contamination; that is, over the short term (^{55}Fe, ^{65}Zn, and ^{60}Co), long term (^{63}Ni, ^{137}Cs, and ^{152}Eu), or very long term (^{238}U, ^{239}Pu, ^{94}Nb, and ^{99}Tc)

TABLE I. RADIONUCLIDES CONSIDERED IN THE RECYCLING AND REUSE ANALYSIS

NUCLIDE	HALF-LIFE (y)	NUCLIDE	HALF-LIFE (y)
^{3}H	12.3	^{144}Ce	0.78
^{14}C	5.7×10^{3}	^{147}Pm	2.62
^{36}Cl	3×10^{5}	^{151}Sm	87
^{41}Ca	1.3×10^{5}	^{152}Eu	13.6
^{54}Mn	0.86	^{154}Eu	8.8
^{55}Fe	2.7	^{226}Ra	1.6×10^{3}
^{57}Co	0.74	^{228}Th	1.91
^{60}Co	5.3	^{229}Th	7.34×10^{3}
^{63}Ni	99.9	^{230}Th	7.7×10^{4}
^{65}Zn .	0.67	^{232}Th	1.4×10^{10}
^{79}Se	6.5×10^{4}	^{232}U	72
^{90}Sr + Y	28.5	^{233}U	1.5×10^{5}
^{93}Zr	1.5×10^{6}	^{234}U	2.47×10^{5}
^{94}Nb	2.0×10^{4}	^{235}U	7.1×10^{8}
^{99}Tc	2.13×10^{5}	^{238}U	4.51×10^{9}
^{106}Ru	1.01	^{237}Np	2.14×10^{6}
^{110m}Ag	0.68	^{238}Pu	87.6
^{125}Sb	2.8	^{239}Pu	2.4×10^{4}
^{129}I	1.6×10^{7}	^{240}Pu	6.57×10^{3}
^{134}Cs	2.06	^{241}Pu	14.4
^{137}Cs	30.1	^{241}Am	4.34×10^{2}

- the mode of decay and internal dose conversion factors (DCF), including alpha emitters with large DCFs (^{228}Th, ^{230}Th, ^{232}Th, ^{232}U, ^{233}U, ^{234}U, ^{235}U, ^{238}U, ^{226}Ra, ^{237}Np, ^{238}Pu, ^{240}Pu, ^{241}Pu, and ^{241}Am); beta/gamma emitters with large DCFs (^{60}Co, ^{65}Zn, ^{94}Nb, ^{90}Sr, ^{110m}Ag, ^{129}I, ^{134}Cs, ^{137}Cs, ^{144}Ce, ^{147}Pm, ^{151}Sm, ^{152}Eu, and ^{154}Eu); non-photon emitters with moderate DCFs (^{90}Sr, ^{106}Ru, and ^{241}Pu); beta/gamma emitters with low DCFs (^{36}Cl, ^{54}Mn, ^{55}Fe, ^{57}Co, ^{99}Tc, and ^{125}Sb); or non-photon emitters with low DCFs (^{3}H, ^{14}C, ^{41}Ca, ^{63}Ni, and ^{79}Se)

- the behavior of the radionuclides during recycling operations; that is, whether they are volatilized and escape, are concentrated, or are partitioned into slag or ingots (products).

Early daughter products in equilibrium with parent radionuclides have been assumed in all cases. For smelting, it is probable that the majority of some radionuclides, such as ^{60}Co, remain in the ingot. However, a fraction of material will remain in the slag, and another portion will likely volatilize and be released with fumes and gases. The behavior of a specific radionuclide will depend on the chemistry of the radionuclide in question and the type of smelting process considered. Because the partitioning is not known for most radionuclides during smelting, the dose calculations that follow are based on the conservative assumption that, for each radionuclide, all of the activity is retained in each of the three phases of smelting: as the metal (steel, aluminum, or copper), in the slag, and as released out of the stack. The slag is assumed to equal about 10% of initial mass of the steel, or about 10 t in the steel and aluminum analyses and 1 t in the copper analysis. This triple accounting approach will overestimate the true doses; however, it will maximize the potential importance of the scenarios and should serve as an adequate basis for the initial development of radiological control criteria for recycling and reuse.

2.2. Radiation Exposure Pathways

Humans may be exposed to radiation in three main ways:

- exposure to external radiation
- inhalation of radioactive gases or small particles
- ingestion of radioactive material.

The following paragraphs describe the specific ways in which these pathways have been used as part of the assessment methods in this study.

2.2.1. External Radiation Exposure

The radioactive sources considered in this study were generally represented by a self-absorbing, homogeneous, cylindrical volume or a surface-contaminated source with the dose point on the central, longitudinal axis of the cylinder [1, 2]. Except for exposure conditions that represent exposure to molten metals contained in a furnace, external absorbers and shields were ignored. This procedure tends to maximize the estimated dose equivalents. In some situations, a source can be represented better by a half cylinder than by a full cylinder. For these situations, a full cylinder was defined such that the area of its flat surface was twice that actually needed; the effective dose equivalent was then calculated by using the full-cylinder model and dividing the resulting dose by two. The external dose calculations were performed using the EXTDF module of the GENII Software System [3].

2.2.2. Inhalation Exposure

The committed effective dose equivalent factors used in this study were taken from International Commission on Radiological Protection Publication No. 30 (ICRP 1977-1982) and its supplements [4]. The concentration of respirable dust in the air will vary depending upon a variety of factors, including the physical condition of the material being handled, the quantity of the material present, and the building's ventilation. Thus, it is difficult to predict the concentrations that may be present during any recycling step. However, so that a complete analysis may be performed, air concentrations have been assumed based on the information in IAEA Safety Series No. 111-P-1.1 [1] for those recycling steps in which the potential for inhalation is most likely. In general, the air concentrations were assumed to vary between about 10^{-3} and 10^{-5} g m^{-3}.

2.2.3. Ingestion Exposure

For this study, ingestion is assumed to occur by one of three separate routes:

- ingestion of removable radioactive materials
- ingestion of corroded material from using frying pans or water pipes
- ingestion of food products contaminated by airborne plumes released from a smelter.

Ingestion of removable radioactive contamination found on recycled metals or reused equipment can occur when the contamination is transferred from a surface to hands, foodstuffs, cigarettes, or other items that enter the mouth. Since very little information exists on the estimated radiation doses associated with this pathway, the methods outlined by the IAEA for recycling and reuse [1] are used for this study. Thus, an assumed quantity of 10 mg of contamination per hour of direct contact exposure is assumed for ingestion by adults. The individuals in direct contact with the recycled metals are assumed to be a selected group of individuals associated with the various manufacturing steps.

Ingestion of contaminated metal corroded from frying pans during cooking and ingestion from copper water pipes are considered as separate ingestion pathways. In their study of the potential impact of recycling, O'Donnel et al. [5] considered cast iron pans and used an assumed corrosion rate of 0.127 cm y^{-1} [5]. Since the use of stainless steel and aluminum pans with a much lower corrosion rate is perhaps more consistent with current domestic practice, a lower value of 0.13 mm y^{-1} is used in this study for all types of pans.

2.2.4. Downwind Exposures

In the recycling of metals, materials that may volatilize and be released through the stack during smelting could result in public exposure. The potential radiation doses to the downwind public from the radioactive effluents assumed to leave the smelter stack were estimated using the CAP88-PC [6] computer code. This software was developed by the U.S. Environmental Protection Agency to perform dose and risk assessments for demonstrating compliance with the National Emission Standards for Hazardous Air Pollutants (NESHAPS) rules in 40 CFR 61.93a [7]. The exposure pathways considered in the analysis included inhalation of airborne material, external exposure to penetrating radiation, and ingestion of contaminated foods. The default meteorological data files for the Chicago area contained in the CAP88-PC code were used in this study. Meteorological data for Chicago were selected for this study because they were felt to be representative of many midwestern U.S. industrial settings with a large population in the vicinity.

3. RADIATION EXPOSURE SCENARIOS AND ASSUMPTIONS

For this analysis, six separate categories of contaminated (or activated) materials and future conditions were considered:

- recycling of steel
- recycling of aluminum
- recycling of copper
- recycling of concrete (as aggregate)
- reuse of a contaminated room within a facility
- reuse of tools or equipment (with surface contamination).

These six categories were further subdivided into various exposure scenarios describing the activities of specific individuals or groups of individuals. To adequately represent those scenarios likely to be of generic importance and relevance to all DOE nuclear facilities, previous dose estimates [1,5,8,9] were used in choosing which scenarios to evaluate. The scenarios presented here yield the highest potential doses for each category of recycled material and radionuclide grouping, as determined from the IAEA [1] study. Details of the scenarios considered, the relevant assumptions, and the values assigned to the important parameters are discussed in the following sections.

3.1. Scenarios for Steel Recycling

Recycled steel may contain both activation products and surface contamination from reactor coolant or other sources. The three most limiting scenarios identified by the IAEA [1] were for a slag worker at the smelter, for the occupant of an automobile containing recycled steel, and for a worker using a piece of large equipment made of recycled steel. These scenarios are the bounding scenarios selected for consideration in this study.

3.2. Scenarios for Aluminum Recycling

Although the long-lived activation of aluminum is negligible, surfaces may become contaminated through contact with reactor coolant or other sources. The scenarios described by the IAEA [1] for recycling of aluminum were evaluated, and the three most limiting ones were used in this study: 1) an operator at a furnace, 2) the occupant of an automobile, and 3) a person using a frying pan.

3.3. Scenarios for Copper Recycling

The IAEA [1] study did not consider copper recycling. However, within the DOE complex, copper recycling may be quite significant; thus, it is included in this analysis. By analogy with steel and aluminum, three scenarios for the recycling of copper are identified for this study. They are doses to a furnace operator and to individuals who use copper pans or live in houses with copper pipes made from recycled copper.

3.4. Scenarios for Concrete Recycling

Large quantities of activated or contaminated concrete will be encountered during decommissioning of DOE defense and research facilities. Because there is an economic incentive to avoid the costs of transporting and disposing of radioactive concrete as radioactive waste, recycling of concrete as feedstock for further concrete manufacture has been considered by European countries [1]. Before such reuse could be authorized, it is clear that any existing building that is a candidate for release would have to pass an extensive radiation survey to assure compliance with existing national regulations.

For the IAEA [1] study, the recycled concrete was assumed to be used to build a new structure in which individuals live or work for 6000 h y^{-1}. The initial concrete was assumed to be contaminated to an undiluted unit concentration. Although a very large dilution could occur during the manufacture of new concrete structures, for this analysis a 1:10 dilution was assumed. The limiting scenarios used were a concrete worker and a resident in a room made from recycled concrete. The concrete activation products ^{36}Cl and ^{41}Ca were included in the analysis to account for the potential activation of concrete. These scenarios account for bulk or volume sources in concrete.

3.5. Scenario for Reuse of Concrete Buildings

Concrete buildings may be decontaminated and reused for other purposes after decommissioning. The scenario considered for building reuse was intended to account for normal occupancy, as described in an evaluation of residual radioactive contamination conducted for the NRC by Kennedy and Strenge [10]. For the building occupancy scenario, individuals were assumed to work in a building after unrestricted release. Although the residence time could vary, a normal work year of 2000 h y^{-1} was assumed. Because decontamination efforts before release focus on the removal of surface sources, the removable level of surface contamination was assumed to be 10% of the levels assumed for other scenarios in this report. That is, for this scenario the air concentration was assumed to be 10^{-5} g/m^3, and the ingestion rate was assumed to be 1.0 mg h^{-1} of exposure.

3.6. Scenarios for Reuse of Equipment or Tools

During decommissioning, discrete pieces of contaminated equipment (including hand tools, pumps, small motors, furniture, and storage tanks) may be salvaged and released for unrestricted use if they can meet radiological control criteria. For the IAEA [1] study, it was assumed that the fixed contamination present on the surfaces of the tools or equipment was ten times higher than the removable fraction, as measured by swabbing. For this study, the radiation exposure scenarios that may be most limiting were used. Because of the potential presence of difficult-to-monitor contamination on the inner surfaces of the motor, these include the use of hand tools that incorporate a small motor (i.e., an electric hand drill or saw). A high exposure duration of 600 h y^{-1} is assumed because of the relatively close proximity of power tools to workers during construction conditions. The exposure pathways considered by the IAEA included exposure to external radiation, ingestion of contamination transferred from the surfaces of the tool to hands and then to the mouth, and inhalation of localized airborne material from the hand tool. In addition to the reuse of small items, the IAEA also considered larger items as possible candidates for reuse. These items are likely to contain surface contamination; therefore, the same exposure considerations apply as for hand tools, with modifications accounting for the size of the item.

4. POTENTIAL EFFECTS ON ELECTRONICS

Another concern centers on the potential effects of unrestricted use of radioactively contaminated recycled metals on electronic components. The threshold range for radiation-induced damage to electronic components varies with the type of component. In anticipation of their use in high-radiation fields (such as applications in space travel), selected electronic components can be "hardened" against radiation effects. In general, for non-hardened components, the damage thresholds for electronic components ranges from about 5 Gy (500 rad) to about 500 Gy (50,000 rad) [11]. Assuming a 10-y lifetime for electronic components, this translates to a dose-rate range of about 5×10^{-5} Gy h^{-1} (5×10^{-3} rad h^{-1}) to 5×10^{-3} Gy h^{-1} (0.5 rad h^{-1}) [11]. For comparison, natural background radiation is about 1×10^{-3} Gy y^{-1} (0.11 rad y^{-1}), or about 1×10^{-7} Gy h^{-1} (1×10^{-5} rad h^{-1}). Since the dose limits considered for the development of control criteria are a fraction of annual background, the development of special control criteria for electronic components is deemed unnecessary.

5. POTENTIAL EFFECTS ON FILM

One potential concern related to recycling of metals and concrete containing residual radioactive contamination is that these recycled materials may be used as material for making film-storage boxes. It is well known that film is sensitive to exposure to radiation and that two of the major uses of film are in the fields of medical and industrial radiography. To help prevent undesirable darkening or fogging of films prior to use, the National Council on Radiation Protection and Measurements (NCRP) has recommended that radiographic film stored in darkrooms or storage areas should not be exposed to more than 2 μGy (0.2 mrad) of radiation prior to being developed [12]. For design specifications for film-storage areas, the NCRP recommends assuming a one-month storage time as an average, if the exact time is not known. In this analysis, an estimate of the potential doses to film resulting from storage was made for storage in four different types of containers constructed from recycled materials.

6. RESULTS AND DISCUSSION

The results of this preliminary study are based on generic exposure scenarios and pathway analyses using 42 radionuclides determined to be potentially present as residual contamination in metals or equipment from DOE operations that may be considered from recycling or reuse. The scenarios and information developed by the IAEA and others were considered in developing this initial radiation exposure pathway and scenario analysis. Although alternative public dose limits were considered, the initial control criteria in this report are based on 1) a dose of 10 μSv y^{-1} (1 mrem y^{-1}) to a worker in a smelter or to an individual who uses consumer products made from recycled materials, 2) a dose of 1 μSv y^{-1} (0.1 mrem y^{-1}) to an individual downwind from a smelter used to process recycled metals, or 3) non-health-related impacts associated with potential radiation effects on electronics or film.

Table II summarizes the limiting concentrations based on individual radiation dose for residual contamination in (or on) recycled materials. For the radionuclides in Table II, doses to smelter workers or to users of consumer products provided the most restrictive (i.e., the smallest) derived residual concentrations. This table shows the initial radiological control criteria for bulk materials, in units of Bq g^{-1}, for steel, aluminum, copper, and concrete, and the initial control criteria for surface contamination in units of Bq cm^{-2}.

TABLE II. DRAFT RADIOLOGICAL CONTROL LEVELS BASED ON AN INDIVIDUAL DOSE OF 10 μSv y^{-1} FOR RECYCLING AND REUSE OF DOE METALS OR EQUIPMENT CONTAINING RESIDUAL RADIOACTIVE CONTAMINATION[a]

	Bulk Contamination Bq g^{-1}			Surface Contamination Bq cm^{-2}	
Radionuclide	Steel	Aluminum	Copper	Concrete	Tools & Equip.
^{3}H	2.1E+05	3.3E+05	3.3E+05	6.3E+06	9.6E+04
^{14}C	7.0E+03	9.6E+03	9.6E+03	2.1E+04	2.9E+03
^{36}Cl	NA[b]	NA	NA	6.3E+02	NA
^{41}Ca	NA	NA	NA	3.1E+02	NA
^{54}Mn	4.8E-01	1.3E+00	7.0E+00	2.2E-01	1.2E+02
^{55}Fe	5.6E+02	9.3E+02	7.8E+03	1.4E+02	5.6E+03
^{57}Co	3.3E+00	8.9E+00	5.2E+01	2.9E+00	6.7E+02
^{60}Co	1.6E-01	4.8E-01	2.5E+00	6.3E-02	4.1E+01
^{63}Ni	1.9E+04	3.7E+04	3.7E+04	2.0E+05	1.1E+04
^{65}Zn	6.3E-01	1.8E+00	9.6E+00	2.5E-01	1.2E+02
^{79}Se	1.6E+03	2.5E+03	2.5E+03	2.7E+04	7.4E+02
^{90}Sr	9.3E+01	1.6E+02	1.6E+02	1.2E+03	4.8E+01
^{93}Zr	1.4E+03	7.0E+03	7.0E+03	5.2E+03	2.5E+03
^{94}Nb	2.6E-01	7.4E-01	4.1E+00	1.3E-01	6.3E+01
^{99}Tc	4.8E+03	1.1E+04	1.6E+04	5.2E+03	4.4E+03

TABLE II. (Cont'd)

Radionuclide	Bulk Contamination Bq g^{-1} Steel	Aluminum	Copper	Surface Contamination Bq cm^{-2} Concrete	Tools & Equip.
^{106}Ru	1.6E+00	4.8E+00	2.5E+01	8.9E-01	1.6E+02
^{110m}Ag	1.4E-01	4.1E-01	2.1E+00	6.3E-02	3.6E+01
^{125}Sb	8.9E-01	2.4E+00	1.3E+01	4.8E-01	2.1E+02
^{129}I	4.8E+01	4.4E+01	7.4E+01	2.0E+01	2.2E+01
^{134}Cs	2.3E-01	6.7E-01	3.6E+00	1.1E-01	3.6E+01
^{137}Cs	7.0E-01	2.0E+00	1.0E+01	3.6E-01	7.4E+01
^{144}Ce	2.7E+01	5.9E+01	4.4E+02	2.2E+01	2.4E+02
^{147}Pm	2.9E+03	1.5E+04	1.5E+04	9.3E+03	4.4E+03
^{151}Sm	3.7E+03	1.9E+04	1.9E+04	1.3E+04	8.9E+03
^{152}Eu	3.4E-01	9.6E-01	5.2E+00	1.4E-01	8.1E+01
^{154}Eu	3.3E-01	9.2E-01	4.8E+00	1.4E-01	7.4E+01
^{226}Ra	7.4E+00	1.9E+01	1.9E+01	4.8E+01	5.2E+00
^{228}Th	4.1E-01	2.0E+00	2.0E+00	1.3E+00	1.5E+00
^{229}Th	7.0E-02	3.6E-01	3.6E-01	2.3E-01	2.6E-01
^{230}Th	4.8E-01	2.3E+00	2.3E+00	1.5E+00	1.7E+00
^{232}Th	1.1E-01	5.5E-01	5.5E-01	3.5E-01	3.7E-01
^{232}U	1.8E-01	9.2E-01	9.2E-01	5.9E-01	7.8E-01
^{233}U	9.6E-01	4.8E+00	4.8E+00	3.0E+00	4.1E+00
^{234}U	9.6E-01	4.8E+00	4.8E+00	3.0E+00	4.1E+00
^{235}U	1.0E+00	5.2E+00	5.2E+00	3.2E+00	4.1E+00
^{238}U	1.0E+00	5.2E+00	5.2E+00	3.2E+00	4.1E+00
^{237}Np	2.4E-01	1.2E+00	1.2E+00	7.8E-01	6.3E-01
^{238}Pu	4.1E-01	2.0E+00	2.0E+00	1.3E+00	1.7E+00
^{239}Pu	3.7E-01	1.8E+00	1.8E+00	1.2E+00	1.5E+00
^{240}Pu	3.7E-01	1.8E+00	1.8E+00	1.2E+00	1.5E+00
^{241}Pu	2.1E+01	1.1E+02	1.1E+02	6.7E+01	8.9E+01
^{241}Am	2.2E-01	1.1E+00	1.1E+00	7.4E-01	5.9E-01

(a) Calculations were made using the EXTDF module from the GENII Software System [3] and selected scenarios based on the methods in IAEA Safety Series No. III-P-1.1 [1].

(b) "NA" indicates that this concrete activation product was Not Applicable to this scenario and was considered only for concrete-recycling scenarios.

Doses to the public downwind of a smelter were estimated using the generic data on atmospheric dispersion and medium-high population density in the EPA's CAP88-PC software. Doses were calculated to the maximally exposed individual (MEI) downwind of a smelter, assuming a unit release. For all radionuclides considered (except ^{238}U), the individual doses were more restrictive than the collective doses to the downwind public.

Also evaluated were non-health-related impacts industry may experience from residual contamination in recycled metals, such as those used in the electronics and film industries. Upon investigation, it was found that most electronic components can withstand doses well in excess of the DOE individual dose limit.

Thus, recycling the materials considered in this report at or below the contamination levels indicated Table II would have little impact on the electronics industry. On the other hand, use of recycled metals was found to have potential impacts on the film industry. Table III summarizes the limiting concentrations in recycled materials based on a 2-μGy (0.2-mrad) exposure to film stored for one month (12) in a box constructed of either undiluted steel or concrete for each of the 42 radionuclides considered. This table shows the initial radiological control criteria (Bq g^{-1}) for bulk materials for steel and concrete both with and without a non-contaminated 0.5-cm lead lining. The initial control levels for film are more restrictive than those derived from doses to smelter workers or to consumers for the photon-emitting radionuclides. This result is considered to be preliminary because of the highly conservative assumptions used in the analysis and because it is unlikely that storage areas for film would be constructed exclusively of undiluted (i.e., 100%) recycled steel or concrete. Further evaluation of the assumptions and data associated with the film scenario is needed.

TABLE III. DRAFT RADIOLOGICAL CONTROL LEVELS BASED ON 2-μGy EXPOSURE TO FILM STORED FOR ONE MONTH[a]

Radionuclide	Initial Radiological Control Levels for Bulk Contamination (Bq g^{-1}) Steel	Lead-lined Steel	Concrete	Lead-lined Concrete
^{3}H	1.3E+08	---[b]	6.3E+07	---[b]
^{14}C	7.8E+03	8.5E+10	4.4E+03	3.0E+11
^{36}Cl	NA[c]	NA[c]	2.0E+02	1.7E+04
^{41}Ca	NA[c]	NA[c]	4.4E+01	---[b]
^{54}Mn	1.3E-01	2.5E-01	1.0E-01	1.9E-01
^{55}Fe	1.3E+02	---[b]	2.0E+01	---[b]
^{57}Co	9.6E-01	2.4E+02	1.1E+00	2.0E+02
^{60}Co	4.4E-02	7.4E-02	3.1E-02	4.8E-02
^{63}Ni	1.0E+05	SpA[d]	4.8E+04	SpA[d]
^{65}Zn	1.8E-01	2.8E-01	1.2E-01	1.8E-01
^{79}Se	1.0E+04	SpA[d]	5.6E+03	SpA[d]
^{90}Sr	2.8E+02	5.6E+04	3.6E+02	1.2E+05
^{93}Zr	1.3E+05	SpA[d]	5.6E+04	SpA[d]
^{94}Nb	7.4E-02	1.6E-01	5.9E-02	1.3E-01
^{99}Tc	1.5E+03	2.5E+07	1.2E+03	6.7E+07
^{106}Ru	4.4E-01	1.3E+00	4.1E-01	1.1E+00
^{110m}Ag	4.1E-02	7.8E-02	3.0E-02	5.5E-02
^{125}Sb	2.5E-01	7.8E-01	2.1E-01	6.3E-01
^{129}I	3.2E+01	SpA[d]	5.5E+00	SpA[d]
^{134}Cs	6.7E-02	1.4E-01	5.2E-02	1.1E-01
^{137}Cs	1.9E-01	4.8E-01	1.6E-01	4.1E-01
^{144}Ce	8.1E+00	8.9E+04	8.1E+00	1.2E+05
^{147}Pm	3.0E+03	2.0E+08	2.2E+03	3.7E+08
^{151}Sm	1.9E+04	SpA[d]	3.5E+03	SpA[d]
^{152}Eu	9.6E-02	1.8E-01	6.7E-02	1.2E-01
^{154}Eu	9.2E-02	1.7E-01	6.7E-02	1.2E-01
^{226}Ra	2.7E+01	3.7E+04	3.2E+01	3.6E+04

TABLE III. (Cont'd)

Initial Radiological Control Levels for Bulk Contamination (Bq g^{-1})				
Radionuclide	Steel	Lead-lined Steel	Concrete	Lead-lined Concrete
^{228}Th	6.7E+01	1.0E+04	3.6E+01	1.1E+04
^{229}Th	1.8E+00	7.0E+02	1.6E+00	7.8E+02
^{230}Th	2.6E+02	1.4E+07	5.9E+01	1.9E+07
^{232}Th	3.3E+02	SpA(d)	6.3E+01	SpA(d)
^{232}U	2.3E+02	1.3E+07	4.4E+01	1.8E+07
^{233}U	3.4E+02	5.2E+06	1.2E+02	7.0E+06
^{234}U	2.8E+02	2.4E+07	5.2E+01	3.2E+07
^{235}U	1.1E+00	4.1E+02	1.2E+00	4.4E+02
^{238}U	3.7E+02	SpA(d)	6.3E+01	SpA(d)
^{237}Np	7.4E+00	1.5E+04	4.1E+00	1.6E+04
^{238}Pu	3.0E+02	SpA(d)	4.8E+01	SpA(d)
^{239}Pu	5.9E+02	2.0E+07	1.2E+02	2.7E+07
^{240}Pu	3.2E+02	SpA(d)	5.2E+01	SpA(d)
^{241}Pu	4.1E+07	---(b)	1.9E+07	---(b)
^{241}Am	1.5E+01	SpA(d)	5.9E+00	SpA(d)

(a) Calculations were made assuming that the film was stored for one month in a rectangular container made from either steel or concrete, with or without lead shielding lining (0.5-cm thickness) the box. The radiological control levels were determined based on the 2-μGy (0.2-mrad) limit recommended by the NCRP (1989) for diagnostic x-ray film.

(b) For radionuclides having no gamma emissions, the lead lining reduced the dose to zero, resulting in initial control levels that approached infinity. This is represented by (---) in the table.

(c) "NA" indicates that this concrete activation product was Not Applicable to this scenario and was considered only for concrete-recycling scenarios.

(d) "SpA" indicates that the calculated control level exceeds the specific activity possible for the radionuclide shown.

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METHODOLOGY FOR CALCULATING ACTIVITY LEVELS FOR EXEMPTION FROM THE REQUIREMENTS OF REPORTING IN THE EURATOM BASIC SAFETY STANDARDS

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Abstract

Practices may be exempt from the reporting requirements in the EURATOM Basic Safety Standards if the relevant activity quantities or concentrations are below certain specified levels. This paper describes the methodology adopted for calculating those levels together with the results for selected radionuclides. The methodology was developed as a collaborative project co-ordinated by CEC. Briefly, a practice may be exempt from the requirement of reporting if the corresponding radiological risks are sufficiently low. Therefore, practices involving small amounts of radioactivity were identified and appropriate exposure scenarios developed representing the use, misuse and subsequent disposal of radionuclides. The calculated exposures were used to derived exempt quantities and concentrations by comparison with the appropriate dose criteria.

1 Introduction

This paper describes the methodology adopted for calculating activity levels for exemption from the requirement of reporting in the EURATOM Basic Safety Standards. This methodology was developed as a collaborative project between National Radiological Protection Board (NRPB), Centre d'Etudies sur L'Evaluation de la Protectin dans le Domaine Nucléaire (CEPN) and Institute de Protection et de Sûreté Nucléaire (IPSN) and co-ordinated by the Commission of European Communities, (CEC).

It is generally accepted that in order to avoid excessive regulatory procedures, certain categories of sources or practices may be exempted from some or all regulatory provisions if the corresponding radiological risks are sufficiently small. The paper by Janssens et al [1] describes the basis for the radiological dose criteria that were adopted in this study and these criteria are reproduced in Table 1. Two types of situation are considered: normal and accidental. Since accidental exposures are not certain to occur it is necessary to consider the probability of the exposure occurring as well as the dose delivered if the event occurs. ICRP [2] recommend that if the dose is below the dose limit then the probability weighted dose can be compared with the dose criteria adopted for normal situations. This approach has been taken in this study. The product of the annual probability and the dose if the event occurs is termed the annual average dose.

Although the use of the annual average dose ensures that accidental exposures are compared with the same criteria as normal exposures, it was felt that an additional constraint on accidental exposures was relevant for exemption level calculations since the aim is to ensure that any exposures received from exempt sources are acceptable. Therefore an additional constraint on the dose to an individual, should the accident occur, was used. In this study the accompanying exempt levels are set such that in accidental situations the dose limits for members of the public (1 mSv y^{-1} effective and 50 mSv y^{-1} to skin) would not be exceeded.

TABLE 1 RADIOLOGICAL PROTECTION CRITERIA FOR CHOOSING EXEMPT ACTIVITIES AND ACTIVITY CONCENTRATIONS

Annual dose criteria (mSv)		
	Effective	Skin
normal situations	0.01	50
accidental situations (assuming the exposure occurs)	1	50

Generally, exemption is expressed in terms of derived quantities such as activity concentrations or activities (quantities) of radionuclides which are related to the dose criteria by a set of defined models representing the practices being considered. This paper describes the choice of radionuclides, exposure scenarios and parameter values used to derive the exempt levels from the requirements of reporting and presents results for selected radionuclides.

2 Choice of radionuclides and physical form

The European Directive applies to any practices or intervention situations which involve a hazard from ionising radiation. However, Article 3 which refers to "practices for which no reporting is required" specifies that the term exemption covers "the use of radioactive substances or their subsequent disposal". Since the purpose of Article 3 is to avoid imposing inefficient requirements on users of small quantities of radioactive substances, it refers to practices involving small scale usage of radioactivity where the radiological risks incurred from the use, misuse and subsequent disposal are too small to warrant regulatory concern. Such practices may include the following uses of radioactivity.

- surface density gauges (β emitters)
- testing the integrity of semiconductors and leak testing generally (e.g. ^{85}Kr)
- in education (e.g. sealed sources for demonstrating properties of radiation)
- technological application (e.g. ^{63}Ni in gas chromography)
- smoke detectors (e.g. ^{241}Am)
- research laboratories (e.g. ^{14}C and ^{32}P as tracers in biochemical research)
- hospital laboratories (e.g. radio-immunoassay techniques)

A review of the radionuclides currently in use in these practices identified about 100 radionuclides with actual or potential uses and a further 200 radionuclides were identified as being of interest in consultation with European experts. Therefore a total of 299 radionuclides were considered in this study and they are listed in Table 2. These radionuclides are used in one or more different physical forms, and since the physical form influences the way in which the radionuclide is used, the way in which it may be misused, and its subsequent disposal method, it is important to identify the relevant physical form. The likely physical forms of those radionuclides for which no current use was identified were determined by consideration of the physical and chemical properties of the element in question. 33 of the radionuclides have decay products which are themselves radioactive and need to be taken into account when assessing exposure. These are identified in Table 2. Since the half lives of the daughters are short (< 1 y) and shorter than their parents, secular equilibrium would be likely to be established within the timescales of the exposure scenarios and so these radionuclides were considered to be in secular equilibrium in the study [3,4,5]. A total of 6 different physical forms were identified and these are described below.

2.1 Gas/vapour

Gaseous radionuclides such as ^{85}Kr are supplied in sealed glass or metal containers. These may be either used as beta sources (for example, in surface density gauges) or the gas may be used in the unsealed form, for example, in testing the integrity of some semiconductor devices. Both routine disposal and accidental releases will give rise to dispersion of the radionuclide in the atmosphere.

TABLE 2 RADIONUCLIDES CONSIDERED IN THIS STUDY

H-3	Co-56	Sr-87m	*Pd-103	Cs-137+	Eu-155	*Au-199	*Pa-231	Cm-244
Be-7	Co-57	Sr-89	*Pd-109	*Cs-138	*Gd-153	*Tl-200	*Pa-233	*Cm-245
C-14	Co-58	Sr-90+	Cd-109	*Te-123m	*Gd-159	Tl-201	*U-230+	*Cm-246
O-15	*Co-58m	*Sr-91	*Cd-115	*Te-125m	*Tb-160	*Tl-202	*U-231	*Cm-247
F-18	Co-60	*Sr-92	*Cd-115m	*Te-127	*Dy-165	Tl-204	*U-232+	*Cm-248
Na-22	*Co-60m	Y-90	*Ag-105	*Te-127m	*Dy-166	Bi-206	*U-233	Bk-249
Na-24	*Co-61	*Y-91	*Ag108m+	*Te-129	*Ho-166	*Bi-207	U-234	*Cf-246
*Si-31	*Co-62m	*Y-91m	Ag-110m	*Te-129m	Er-169	*Bi-210	*U-235+	*Cf-248
P-32	*Ni-59	*Y-92	Ag-111	*Te-131	*Er-171	*Bi-212+	*U-236	*Cf-249
*P33	Ni-63	*Y-93	In-111	*Te-131m	Tm-170	*Pb-203	*U-237	*Cf-250
S-35	*Ni-65	Rb-86	In-113m	Te-132	*Tm-171	Pb-210+	U-238+	*Cf-251
Cl-36	Cu-64	*Zr-93+	*In-114m	*Te-133	*Yb-175	*Pb-212+	U-238 N	Cf-252
*Cl-38	Zn-65	Zr-95	*In-115m	*Te-133m	*Lu-177	*Po-203	*U-239	*Cf-253
Ar-37	*Zn-69	*Zr-97+	*Sn-113	*Te-134	Ta-182	*Po-205	*U-240	*Cf-254
Ar-41	Zn-69m	*Nb-93m	*Sn-125	*Xe-131m	*Hf-181	*Po-207	*U-240+	*Es-253
*K-40	*Ge-71	*Nb-94	Sb-122	Xe-133	*W-181	Po-210	*Np-237+	Es-254
K-42	Ga-72	Nb-95	Sb-124	*Xe-135	W-185	*At-211	*Np-239	*Es-254m
*K-43	*As-73	*Nb-97	Sb-125	Ce-139	*W-187	*Rn-220+	*Np-240	*Fm-254
Ca-45	As-74	*Nb-98	I-123	Ce141	Re-186	Rn-222+	*Pu-234	*Fm-255
Ca-47	*As-76	*Tc-96	I-125	*Ce143	*Re-188	*Ra-223+	*Pu-235	
Sc-46	*As-77	*Tc-96m	*I-126	Ce-144+	*Os-185	*Ra-224+	*Pu-236	
*Sc-47	Se-75	*Tc-97	*I-129	*Ba-131	*Os-191	*Ra-225	*Pu-237	
*Sc-48	Br-82	*Tc-97m	*I-130	Ba-140+	*Os-191m	Ra-226+	Pu-238	
*V-48	*Kr-74	*Tc-99	I-131	La-140	*Os-193	*Ra-227	Pu-239	
Cr-51	*Kr-76	Tc-99m	I-132	*Pr-142	*Ir-190	*Ra-228+	*Pu-240	
Fe-52	*Kr-77	*Mo-90	*I-133	Pr-143	Ir-192	*Th-226+	*Pu-241	
Fe-55	*Kr-79	*Mo-93	*I-134	Pm-147	*Ir-194	*Th-227	*Pu-242	
Fe-59	*Kr-81	Mo-99	*I-135	*Pm-149	*Pt-191	Th-228+	*Pu-243	
*Mn-51	*Kr-83m	*Mo-101	*Cs-129	*Nd-147	*Pt-193m	*Th-229+	*Pu-244	
*Mn-52	Kr-85	*Ru-97	Cs-131	*Nd-149	*Pt-197	Th-230	Am-241	
*Mn-52m	*Kr-85m	Ru-103	*Cs-132	*Sm-151	*Pt-197m	*Th-231	*Am-242	
*Mn-53	*Kr-87	*Ru-105	*Cs-134m	*Sm-153	Hg-197	Th-232 N	*Am-242m+	
Mn-54	*Kr-88	Ru-106+	Cs-134	Eu-152	*Hg-197m	*Th-234+	*Am-243+	
Mn-56	Sr-85	*Rh-103m	*Cs-135	*Eu-152m	Hg-203	*Ac-228	*Cm-242	
*Co-55	Sr-85m	*Rh-105	*Cs-136	Eu-154	Au-198	*Pa-230	*Cm-243	

PREFIX (*) REFERS TO RADIONUCLIDE OF UNKNOWN USE AND WASTE FORM.

SUFFIX (+), (N) radionuclides with short-lived daughters assumed to be in secular equilibrium

2.2 Liquid/solution

Many radionuclides are used in the form of liquid solutions, in a wide range of applications. Examples include the use of ^{99m}Tc in diagnostic nuclear medicine and ^{32}P in biochemical research. Occupational exposure may occur as a result of handling containers of liquid (external exposure) and inadvertent intakes of split material (contamination). Liquid wastes arise inevitably from the use of these materials.

2.3 Dispersible solid

Radionuclides can exist in the form of dispersible solids (e.g. powders) in a variety of ways such as, for example, finely divided process materials containing isotopes of the natural radioactive elements uranium and thorium. Almost all the radionuclides considered in the study could potentially be present in the form of dispersible solid low level wastes.

2.4 Non-dispersible solid

Exposure pathways arising from non-dispersible solid forms were also considered for natural radionuclides.

2.5 Thin film/foil

One form of sealed radioactive source is a thin film, usually fixed on a carried substrate, examples include ^{241}Am sources used in ionisation chamber smoke detectors and ^{63}Ni sources in some gas chromatographs.

2.6 Sealed source/capsule

The other common form of sealed source is an encapsulated pellet or similar. This is commonly used for gamma emitting radionuclides such as ^{137}Cs and ^{60}Co.

3 Exposure scenarios and pathways

The aim of this study was to calculate the doses arising from the use, misuse and disposal of radioactive materials and then to compare the resulting dose with the appropriate dose criteria and derive exemption levels. The first step was, therefore, to establish a set of exposure scenarios and pathways that covered the range of possible exposures. In this study the term "scenario" is used to identify a generic situation or sequence of events which may lead to exposures via a range of external, inhalation and ingestion pathways. A review of the literature identified over 20 possible exposure scenarios with over 80 different exposure pathways [3,4,5]. However, only a few of these exposure scenarios and pathways give rise to doses that dominated the results and therefore, the most radiologically important scenarios and pathways were selected for this study, in consultation with European experts. Three exposure scenarios were chosen. These are normal use (workplace), accidental (workplace) and disposal to landfill (public). Each of these scenarios may give rise to doses via one or more pathways and when reviewing these pathways, it became obvious that the same pathways and parameter values could not be used to derive both exempt concentrations and quantities. This is because the concentration calculations are more pessimistic since it is assumed that users may hold as much activity as they wish, provided that the calculated activity concentration limits are not exceeded.

A total of 23 exposure pathways were identified, 8 for activity concentration calculations, and 15 for total activity calculations. These scenarios and pathways are listed in Table 3 and described below.

3.1 Normal Use (workplace) scenario

This scenario represents the use of small amounts of radionuclides in industry etc in the manner for which they are intended, and involves external exposures and inadvertent intakes of radioactive materials. Only doses to the person(s) using the source are assessed.

3.1.1 Pathways for activity concentration calculations

Five pathways were identified for activity concentration calculations. The first, external exposure from handling a source [A1.1], assumes that the individual picks up and handles a source for a limited proportion of

TABLE 3 LIST OF EXPOSURE SCENARIOS AND PATHWAYS *CONSIDERED IN CALCULATIONS OF DOSES*

1	**NORMAL USE (WORKPLACE) SCENARIO:**		
A1	**Activity concentration**		
	A1.1	External exposure from handling a source	
	A1.2	External exposure from a 1 m^3 source	
	A1.3	External exposure from a gas bottle	
	A1.4	Inhalation of dusts	
	A1.5	Ingestion from contaminated hands	
B1	**Activities/quantities**		
	B1.1	External exposure from a point source	
	B1.2	External exposure from handling a source	
2	**ACCIDENTAL (WORKPLACE) SCENARIO:**		
A2	**Activity concentration** **this is covered by Normal use (workplace) scenario**		
B2	**Activities/quantities**		
	B2.1	Spillage:	External exposure from contaminated hands
	B2.2	Spillage:	External exposure from contaminated face
	B2.3	Spillage:	External exposure from contaminated surface
	B2.4	Spillage:	Ingestion from hands
	B2.5	Spillage:	Inhalation of resuspended activity
	B2.6	Spillage:	External dose from aerosol or dust cloud
	B2.7	Fire:	Contamination of skin
	B2.8	Fire:	Inhalation of dust or volatiles
	B2.9	Fire:	External from combustion products
3	**DISPOSAL (PUBLIC) SCENARIO:**		
A3	**Activity concentration**		
	A3.1	External exposure from a landfill site	
	A3.2	Inhalation of dust from a landfill site	
	A3.3	Ingestion of an object from a landfill site	
B3	**Activities/quantities**		
	B3.1	External exposure from a landfill site	
	B3.2	Inhalation from a landfill site	
	B3.3	External exposure to skin from handling an object from a landfill site	
	B3.4	Ingestion of an object from a landfill site	

the working day (< 5%). Typical situations, involving handling sources, include fitting of sources into jigs for calibrating instruments, packaging and machining small radioactive components. The second exposure pathway, external exposure from a 1 m^3 source [A1.2], assumes that the operator is exposed to a source of 1 m^3 for 100 hours per year. Examples of this are exposure from small stock piles of ores containing natural radionuclides, process materials or a store (cabinet) of radioactive material. The third exposure pathway, external exposure from a 0.1 m^3 gas bottle assumes that the operator works at a distance of 1 m from a single gas bottle [A1.3], containing the radionuclide in question for 100 hours per year. Examples of this are hospitals or research laboratories, where the person may be unaware of the potential dose from the radiation emitted by the gas. It is unlikely that beta particles will penetrate the gas cylinder walls and therefore this exposure pathway considers gamma doses only. It is also assumed that the gas cylinder walls provide negligible shielding from gamma rays. The fourth exposure pathway, inhalation of dust [A1.4], assumes that the operator is exposed for a normal working year to an atmosphere with a dust concentration similar to the average air concentrations allowed for some elements for industrial processes. This dust is then assumed to be contaminated with radionuclides. Examples of this are exposure to natural radionuclides in the processing of mineral ores and during the manufacture of specialist refactories. Further exposures may occur when radioactively contaminated metals may be sawed or milled to produce a usable product. The fifth exposure pathway, ingestion from contaminated hands [A1.5], assumes that an individual working in the contaminated dusty atmosphere described above, inadvertently picks up and ingests a small fraction of the deposited dust each day. The resultant ingestion rate (32 mg per year) is less than 1% of the value typically used for inadvertent ingestion of soil while gardening.

The effective doses from each of the five pathways were summed to give a total dose from this scenario for comparison with the dose criteria. The skin dose from pathway A1.1 was also compared with the dose criteria.

3.1.2 Pathways for total activity calculations

Two pathways were identified for total activity calculations. The first, external exposure from a point source [B1.1], assumes that the operator is working near a small source, represented by a point source at 1 m. Typical situations involving this exposure pathway include repetitive use of a small sealed source to test equipment, fitting small sealed sources into devices (eg smoke detector), use of sources in tracer studies and the presence of small sources packed in containers. The second pathway, external exposure from handling a source [B1.2], assumes that the individual picks up and handles a source for a limited proportion of the working day. This pathway is the same as the handling pathway for activity concentrations except that the exposure time is less. The effective dose from these two pathways were summed to give a total dose from this scenario, for comparison with the dose criteria. The skin dose from pathway B1.2 was also compared with the dose criteria.

3.2 Accidental (Workplace) scenario

This scenario represents exposure arising from accidents and misuse in the workplace. Only doses to the person(s) using the source are assessed. Both the probability of the exposure occuring and the dose delivered if the event occurs are considered (see section 1).

3.2.1 Pathways for activity concentration calculations

The possible exposure pathways are external exposure, ingestion and inhalation. However, all these pathways have already been considered in the normal use (workplace) scenario and since the parameters chosen for that scenario were generally pessimistic, it was decided to review the combination of dose rate, exposure time and probability of exposure to see if the accidental pathways would give rise to lower or higher average annual doses. In fact, the accidental pathways give rise to lower average annual doses and therefore the Accidental (workplace) scenario is adequately covered by the Normal use (workplace) scenario and was not treated explicitly.

As discussed above, the doses from accidental scenarios, should they occur, are also compared with an effective dose limit of 1 mSv per year and a skin dose limit of 50 mSv. This was done for the four exposure pathways as follows. Firstly, for skin dose, the contact time assumed for normal use is 25 h per year. If an accident occurs, the contact time is likely to be much shorter (eg 10 mins) and therefore if normal use gives doses below the 50 mSv skin dose limit then so will an accident, if it occurs. Secondly, for external exposure the exposure time assumed for normal use is 100 hours and gives doses below 10 μSv per year. Therefore, even assuming continual occupancy, the maximum individual dose would be about 0.7 mSv, below 1 mSv. Thirdly, for inhalation, a dust loading of 40 $\mu g\ m^{-3}$ was assumed throughout the year. In order to incur a dose of 1 mSv, the dost loading would have to be 4 $mg\ m^{-3}$ throughout the year which is verging on an intolerably dusty atmosphere. Finally, for ingestion, an intake rate of 32 mg per year was assumed for the normal situation. In order to incur a dose of 1 mSv, over 3 g of material would have to be ingested in an accident.

3.2.2 Pathways for total activity calculations

The operator is assumed to be exposed to external and internal dose pathways from two basic situations: accidental spillage of radionuclides and contaminated smoke from a fire. These two situations are treated separately when determining exempt levels ie the spillage and fire doses were not added together.

Spillage was assumed to give rise to doses via 6 exposure pathways. The first, exposure from contaminated hands [B2.1], assumes that an individual accidentally spills a radioactive solution or powder over a working surface and 10% is assumed to contaminate the back of an individual's hands and part of their arms. The contamination is assumed to be washed off after 10 minutes. The second pathway, external exposure from contaminated face [B2.2], assumes that following a spillage as described above, 10% of the material on the hands is transferred to the face and it is washed off after 10 mins. The third pathway, external exposure from contaminated surface [B2.3], assumes that a spill results in a working surface of area 7 m^2 becoming contaminated and that an individual works at a distance of 1 m from the spilt source for 10 minutes. The fourth pathway, ingestion from hands [B2.4], assumes that the individual inadvertently ingests 1 mg of spilt material for his or her hands. The fifth pathway, inhalation of dust or volatiles, assumes that the individual inhales dust or volatiles [B2.5], from a spilt source for 10 minutes. Finally, the sixth pathway, external exposures from an aerosol or dust cloud [B2.6], assumes that an aerosol or dust cloud is formed from spilling a solution or powder and disperses in a room, remaining airborne for at least 10 mins. The individual is then assumed to remain in the room for 10 mins. The average annual effective doses from these six pathways were added together to give a total dose from spillage.

Fire was assumed to give rise to doses via 3 exposure pathways. The first, contamination of skin [B2.7], considers a Laboratory fire in which the radioactive source is ignited and the resulting ash is deposited over a large area. It is assumed that the skin (face or back of hands) is exposed to the deposit for 10 minutes. The second pathway, inhalation of dust or volatiles [B2.8], assumes that a person inhales the combustion products in a Laboratory fire for 10 minutes. The third pathway, external dose from combustion products [B2.9], assumes that the fire results in a contaminated cloud of smoke which persists for at least 10 minutes, during which time an individual is exposed to the dose from the gamma and beta emitters in the cloud. The average annual effective doses from these three pathways were added together to give a total dose from fire.

3.3 Disposal (Public) scenario

This scenario considers the normal and accidental exposure of a member of the public who is visiting a landfill site in which a radioactive source has been disposed. Most sites are accessible to the public, especially if they are those which allow individual members of the public to dump their own rubbish, such as Local Authority sites in the UK. The landfill site is assumed to be a generic small site with a capacity of domestic waste of 1.5 10^4 tonnes over an area of 10^{-2} km^2. A delay of 24 hours is assumed to occur between use of the source and its subsequent disposal at the landfill site, during which time the source is assumed to decay. Radioactive decay over the exposure time of the individual is not included to allow for the possibility of sequential disposal of sources.

3.3.1 Pathways for activity concentration calculations

Three pathways were identified for activity concentration calculations. The first, external exposure [A3.1], was considered as an accidental exposure pathway. This pathway assumes that a member of the public walks outside for 300 h y^{-1}, a time typical of outdoor recreational activities, and spends 1% of this time walking over the landfill site. The second pathway, inhalation of dust [A3.2], is also considered as an accidental exposure pathway. In this pathway, the individual walking over the site is assumed to inhale dust from the exposed contaminated soil from a single source for 1 hour per year. The third pathway, ingestion of activity from the source [A3.3], represents three inadvertent ingestion mechanisms. These are: a person finding a source or an object contaminated with radioactivity, a person ingesting contaminated soil from their hands, or a child inadvertently swallowing an object. An intake rate of 1 g per year was assumed, approximately half the quantity typically used for inadvertent ingestion of soil while gardening. The average annual effective doses from these three pathways were summed to give a total dose from this scenario, for comparison with the dose criteria.

3.3.2 Pathways for total activity calculations

Four pathways were identified for total activity calculations. The first, external exposure [B3.1)] is the same as for the activity concentration calculations. The second, inhalation of dust [B3.2], considers two routes by which dust may become inhaled by a member of the public from radioactive contaminated material on a landfill site. The first route is accidental exposure by a member of the public walking over the landfill site, as described in 3.3.1. The second route considers normal exposure of a member of the public residing close to a landfill site where they inhale dust for 5000 h/y. In both cases the doses are the same. The third pathway, external exposure to skin [B3.3], assumes that a person walking over a landfill site finds an object which is of interest, picks it up and holds or places it in their pocket for 8 hours. The fourth pathway, ingestion of an object from a landfill site [B3.4], assumes that a member of the public inadvertently ingests a small fraction (0.1%) of the source. The average annual effective doses from these four pathways were summed to give a total dose from this scenario, for comparison with the dose criteria. The skin dose from pathway was B3.3 was also compared with the dose criteria.

4 Calculation of exempt levels

Doses to individuals in the workplace and to members of the public are obtained for an activity concentration of 1 Bq g^{-1} and an activity of 1 Bq. It is assumed that the total inventory of radioactive substances in the considered entity at any time remains 1 Bq g^{-1} or 1 Bq. This is a conservative assumption since in the case of short-lived nuclides, the average inventory will be much smaller.

The generic formulae used to calculate doses are as follows:

$D = (A \text{ or } C)\, f\, T\, R\, s$	for effective dose from whole body external exposure
$D_{skin} = \dfrac{(A \text{ or } C)\, f\, T\, R\, s}{\text{CONTACT}}$	for skin dose external exposure
$D = D_{skin}\, W_{skin}\, \dfrac{\text{CONTACT}}{\text{BODY}}$	for effective dose from skin dose
$D = (A \text{ or } C)\, f\, T\, R\, s\ \text{INH DUST}$	for effective dose from inhalation
$D = (A \text{ or } C)\, f\, R\, s\ \text{ING}$	for effective dose from ingestion

D_{skin} is the dose equivalent for skin doses, D is the effective dose for whole body doses or the committed effective dose for intakes of radionuclides.

A and C are the activity (1 Bq) or activity concentration (1 Bq g^{-1}) respectively.

f is the fraction of A or C which contributes to the dose, D. This may be expressed, for example, as a fraction which contaminates the individual, eg, Accidental-spillage or Accidental-ingestion from contaminated hands.

T is the time for which an individual is exposed to the source, (h y^{-1}). The exposure time taken is generally realistic, for example, in the Accident (workplace) scenario, it is assumed that an individual is exposed for 10 minutes before decontaminated takes place. Values used in this study are given in Table 4.

R is the radionuclide dependent dose factor for a given pathway e.g. dose per unit ingestion, external dose rate from a point source. This factor may be modified by a geometry factor if the size of the source is smaller than the geometry assumed when deriving the dose factors; for example, when calculating the external dose from a 0.1 m^3 gas bottle the dose factor for an infinite slab is modified to account for the size of the gas bottle by multiplying by a geometric factor.

The parameters used for external dose calculations (both whole body and skin dose) are given in Table 5. Dose per unit intake values for ingestion and inhalation were taken from Reference 6.

The term s represents the probability of an exposure occurring in a year. This is used in situations where it is not certain that a dose will occur in a year, ie, Accident (workplace) scenario and some Disposal (public) pathways. The probability chosen for all these situations was 1 10^{-2} per year.

TABLE 4 EXPOSURE TIME USED IN THE CALCULATIONS

Value $h\ y^{-1}$	Scenario/pathway
0.16	All accidental (workplace) pathways
1	A3.2, B3.2
8	B3.3
10	B1.2
25	A1.1
100	A1.2, A1.3, B1.1 (liquid, powder)
200	B1.1 (others)
300	A3.1, B3.1
2000	A1.4

TABLE 5 EXTERNAL DOSE PARAMETERS USED IN THE CALCULATION

Value/formula	Reference	Scenario/pathway	Comment
β dose 1 m above infinite slab	3	A1.2	Shielding factor = 0.1
β dose 1 m above infinite plane	7	B2.3	Geometry factor = 0.1 [ref 13]
β dose from point source at 1 m	3	B1.1	-
β dose in semi-infinite cloud $2\ 10^{-6}$ (Sv y^{-1})/(Bq m^{-3} MeV)	8,9	B2.6, B2.9	-
β dose at 4 mg cm^{-2}	10, 14	B2.1, B2.2	area = 100 cm^2 liquids = 600 cm^2 solids
		B2.7	area = $2\ 10^4$ cm^2 liquids = $2\ 10^2$ cm^2 other
β dose at 40 m g cm^{-2}	10, 14	A1.1, B1.2,	area = 20 (powder) 0.5 (gas), 2 (other)
		B3.3	area = 200
γ dose 1 m above semi infinite slab $3\ 10^{-7}$ (Sv h^{-1})/Bq g^{-1} MeV)	8, 11	A1.2 A1.3 A3.1 B3.1	Geom = $2\ 10^{-2}$ [ref 8] Geom = $3\ 10^{-3}$ [ref 11] Geom = $6\ 10^{-9}$ [ref 3] Geom = $6\ 10^{-11}$ [ref 3]
γ dose 1 m above infinite plan	7	B2.3	Geom = 0.1 [ref 13]
γ dose from point source at 1 m	3	B1.1	-
γ dose in semi infinite cloud $1.6\ 10^{-6}$ (Sv y^{-1})/(Bq m^{-3} Mev)	8, 9	B2.6, B2.9	
γ dose at 7 mg cm^{-2}	12, 14	A1.1, B1.2, B2.1 B2.2, B2.7, B3.3	

W_{skin} is the tissue weighting factor for skin (10^{-2}), BODY is the total skin area (10^4 cm^2) and CONTACT is the contact area for the pathway under consideration (see Table 5).

The exemption levels were calculated using the dose criteria in Table 1 and dividing these by the maximum doses obtained for each scenario and radionuclide, as follows:

$$\text{Exempt level for each scenario} = \frac{\text{Annual individual dose criteria}}{\text{Dose per unit activity (Bq) or activity concentration (Bq g}^{-1})}$$

These were calculated for both skin doses and effective doses. For the Normal use (workplace) and Disposal (public) scenarios the effective dose for each scenario was the sum of the effective doses from all the pathways considered. For the Accidental (workplace) scenario, the two basic types of accident (spillage and fire) were treated separately.

The smallest (most restrictive) exempt level for each radionuclide and physical form was determined from the workplace and public scenarios.

These values were rounded up or down as follows: if the calculated value lies between 3 10^{X} and 3 10^{X+1}, then the rounded exemption level is 10^{X+1}. For example, 6 10^{7} would be rounded up to 10^{8} whereas 2 10^{5} would be rounded down to 10^{5}.

5 Results and discussion

Table 6 gives the rounded exempt levels for 25 of the most commonly used radionuclides.

5.1 Exemption activity concentrations

From Table 6 it can be seen that the rounded exemption activity concentration values range from 1 Bq g^{-1} the value obtained for most α-emitting actinides to 1 10^{6} Bq g^{-1} for ^{3}H.

For the majority of radionuclides the exemption concentrations are based on external exposures arising in the workplace from standing near a source such as a store cupboard. The exemption activity concentrations for radionuclides which are used in a gas form (including ^{37}Ar) are due to external doses from standing close to a gas bottle. For the remaining radionuclides (notably some of the actinides) the exemption activity concentrations are a result of doses from inhalation in the workplace or ingestion by a member of the public on a landfill site.

5.2 Exemption activities

From Table 6 it can be seen that rounded activity values range between 1 10^{4} and 1 10^{9} Bq. In fact, the range over all 299 radionuclides was from 1 10^{3} Bq for most α-emitting nuclides up to 1 10^{12} Bq for ^{83m}Kr which is a short lived noble gas.

The exemption activity for α-emitters is determined by the exposure from inhalation or ingestion by a member of the public on a landfill site or by inhalation following an accident at the workplace.

As a general rule the exemption values for gamma and beta emitters tend to be based on skin doses to a worker handling a source, or external effective doses. However, some of them have exemption activities based on internal doses to a worker accidentally inhaling smoke from a fire, or to a member of the public ingesting material from a landfill site.

5.3 Discussion and conclusion

In calculating exemption concentrations and quantities it has been assumed that there is no limit on the quantity of radioactivity that can be held provided it is below the exemption concentration; similarly, there is no limit to the activity concentration provided that the total activity limit is not exceeded. It should be remembered that the scenarios consider only moderate masses corresponding to small users.

Application of the rounded levels is expected to result in effective doses to the critical group of no more than around 10 μSv per year with doses of no more than 50 mSv per year to irradiated areas of the skin. Furthermore, these exposures are calculated for realistic exposure situations but even if unlikely, or pessimistic, exposure situations prevailed the dose limits for members of the public of 1 mSv per year (effective) and 50 mSv per year (skin) would not be exceeded.

In conclusion, exemption concentrations and quantities for around 300 radionuclides have been calculated using defined exposure scenarios and pathways. The calculated values (Table 4) apply to practices

TABLE 6 RESULTS FOR SELECTED RADIONUCLIDES

Nuclide	Use	Activity concentration $Bq\ g^{-1}$	Critical pathway	Activity Bq	Critical pathway
^{3}H	H,R,T	10^{6}	Ing acc (P)	10^{9}	Ing acc (P)
^{14}C	H,R,T	10^{4}	Ing acc (P)	10^{7}	Ing acc (P)
^{32}P	H,E,R,T	10^{3}	Ext (W)	10^{5}	Skin (W)
^{35}S	R,T	10^{5}	Ing acc (P)	10^{8}	Ing acc (P)
^{36}Cl	R,T	10^{4}	Ext (W)	10^{6}	Skin (W)
^{55}Fe	T	10^{4}	Ing acc (P)	10^{6}	Skin (W)
^{59}Fe	H,R,T	10^{1}	Ext (W)	10^{6}	Ext (W)
^{57}Co	H,R	10^{2}	Ext (W)	10^{6}	Skin (W)
^{58}Co	H,R,T	10^{1}	Ext (W)	10^{6}	Ext (W)
^{60}Co	H,R,T	10^{1}	Ext (W)	10^{5}	Skin (W)
^{63}Ni	G,T	10^{5}	Ing acc (P)	10^{8}	Ing acc (P)
^{85}Kr	R,L,G,T	10^{5}	Extg (W)	10^{4}	Skin (W)
^{89}Sr	H,T	10^{3}	Ext (W)	10^{6}	Skin (W)
$^{90}Sr+$	G,H,E,R,T	10^{2}	Ext (W)	10^{4}	Skin (W)
^{99m}Tc	H	10^{2}	Ext (W)	10^{7}	Ext (W)
^{111}In	H	10^{2}	Ext (W)	10^{6}	Ext (W)
^{123}I	H	10^{2}	Ext (W)	10^{7}	Ext (W)
^{125}I	H,R	10^{3}	Ext (W)	10^{6}	Ing acc (P)
$^{137}Cs+$	G,H,R,T	10^{1}	Ext (W)	10^{4}	Skin (W)
^{147}Pm	T	10^{4}	Inh (W)	10^{7}	Ing acc (P)
^{198}Au	H,T	10^{2}	Ext (W)	10^{6}	Ext (W)
^{201}Tl	H	10^{2}	Ext (W)	10^{6}	Skin (W)
$^{226}Ra+$	H,T	10^{1}	Ing acc (P)	10^{4}	Ing acc (P)
^{239}Pu	E,T	1	Inh (W)	10^{4}	Inh acc (P)
^{241}Am	E,S,T	1	Inh (W)	10^{4}	Inh acc (P)

Key

Ing acc (P) Accidental ingestion by public from landfill (A3.3, B.34).
Ext (W) External in workplace (effective skin + point source) (A1.1, A1.2, B1.2, B1.2).
Skin (W) Skin dose in workplace (A1.1, B1.2).
Extg (W) External in workplace from gas cylinder (A1.3).
Inh (W) Inhalation in workplace (A1.4).
Inh acc (P) Accidental inhalation by public from landfill (A3.2, B3.2).
\+ Short lived daughters included.
G Surface density gauges.
L Leak testing.
E Education.
T Technological.
S Smoke detectors.
R Research laboratories.
H Hospital laboratories.

involving small scale usage of activity where the quantities involves are at most of the order of a tonne. The values take into account use, misuse and subsequent disposal.

These values have been presented in report [16] and are the values recently proposed for a revision of the Basic Safety Standards Directive.

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CLEARANCE LEVEL RECOMMENDATIONS FROM THE GERMAN COMMISSION ON RADIOLOGICAL PROTECTION FOR METAL EQUIPMENT AND SCRAP:
Derivation of clearance levels

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Abstract

In Germany the peaceful use of nuclear power is regulated by the German Atomic Energy Act and its ordinances. The exemption of radioactive material from the system of regulatory control is governed by the Radiation Protection Ordinance, which also entails surface specific clearance levels for material leaving regulatory control. No mass specific clearance levels are prescribed, but recommendations from the German Commission on Radiation Protection have been published. The clearance of material from a nuclear site is made on a case by case basis within a licensing procedure. Release is only allowed if it can be shown to be "non-detrimental". The authorities in Germany have come to a common understanding of the non-detrimental clause in the Atomic Energy Act and interpret it to mean the "de-minimis", which recommends an individual effective dose of 10μSv/a for a single practice. A discussion of the clearance level recommendations, which are based on the "de-minimis" concept, are presented.

1. INTRODUCTION

In Germany the peaceful use of nuclear power is regulated by the German Atomic Energy Act, AtG [1]. The legal basis for recycling and reusing material from nuclear installations is section 9a which prescribes that radioactive material as well as dis-assembled or dismantled radioactive components

1. be ... recycled or reused in a non-detrimental manner, or
2. if this is not possible, using present scientific and technical means, or not economically feasible, ... be systematically stored as radioactive waste.

Section 9a of the AtG gives recycling and reuse priority over radioactive disposal when this is feasible and shown to be non-detrimental. An individual dose[1] of 10 μSv in one year, "de minimis", which has an associated risk according to the ICRP of 10^{-7}, has been recommended by the International Atomic Energy Agency, IAEA, as a

1 ICRP 31 effective dose equivalent or in ICRP 60 effective dose.

basis for rule making for a single practice [2]. In Germany "de minimis" forms the understanding upon which recommendations and licensing procedures for release of slightly radioactive material are based. In the recommendations of the German Commission on Radiological Protection the "de minimis" is interpreted as fulfilling the "non-detrimental" clause from section 9a of the AtG [3,4].

2. THE GERMAN COMMISSION ON RADIOLOGICAL PROTECTION'S RECOMMENDATIONS

2.1 Metal from Nuclear Power Plants

In order that the licenses from state to state are consistent recommendations have been published by the German Commission on Radiological Protection, SSK. The first recommendation "Radiological Protection Principles for the non-detrimental Recycling and Reuse of slightly radioactive Iron and Steel from Nuclear Power Plants" was published in 1987 [3]. This recommendation has since been used for defining clearance levels in licenses. In 1992 a second recommendation for non-ferrous metals was issued, which validated the practice of using the ferrous metal clearance levels for non-ferrous metals [4].

The recommendations are based on an analysis of the post release processes which lead to critical scenarios. In fig. 1 a flow diagram shows the expected paths the metal will follow after being released. The possibility of direct reuse is shown by the arrow connecting the release box with the product box. Direct reuse is an issue for tools, installation parts like electric motors or pumps or equipment like forklifts for which the economic value is higher than the scrap value. The majority of the metal released will be treated as normal scrap and enter into the metal production industry. Since after release no control can be exercised the possibility that the metal is reused instead of melted must be considered. This possibility is most likely to occur with scrap pieces like large plates, stairs or structural elements like I-beams. In the flow diagram this is indicated by the arrow connecting the scrapyard box to the product box. Scrap which is smelted into new products goes through a number of processes which lead to a reduction in the activity concentration in the final product compared to the released scrap. This includes the mixing with scrap from other origins and nuclide separation during melting. For example after smelting ^{137}Cs is found predominantly in the slag and dust while ^{60}Co remains in the metal [5].

Averaging conditions, which are part of the release criteria, are important factors influencing the average released activity. If the area and mass over which it is

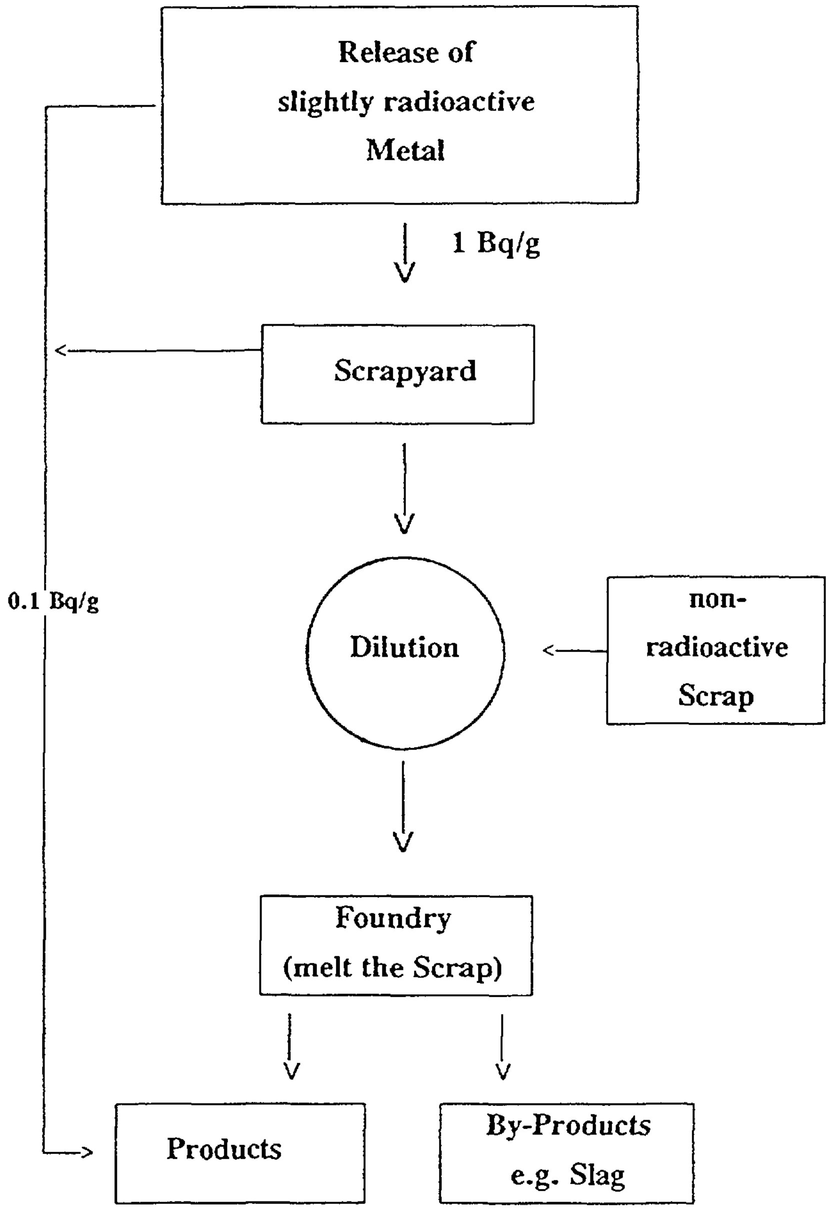

Fig. 1
Flow diagram representing the post release process for metal

allowed to average the activity is kept relatively small and no piece is allowed to be released with higher activity than the clearance level, then due to the inhomogeneity of the activity the released material will have activity levels well below the clearance level [6,7]. A simple example is given in table I to illustrate this point. The scrap in the example is released using a clearance level of 1 Bq/g but the mass over which averaging is allowed varies. For an averaging mass of 1000 kg the entire scrap can be released, including the 50 kg with 10 Bq/g. If the averaging mass is more restrictive, then the 50 kg with 10 Bq/g and the 50 kg with 5 Bq/g can no longer be released, see table I. In this example a reduction in the averaging mass from 1000 kg to 100 kg results in a factor of 4 reduction in the average activity concentration.

Table I: SIMPLE RELEASE EXAMPLE; EFFECT OF REDUCING THE MASS OVER WHICH AVERAGING IS ALLOWED, THE CLEARANCE LEVEL IS 1 Bq/g

Scrap to be released		**Release allowed ?**		
Mass kg	**Activity Bq/g**	**Averaging Mass kg**		
		1000	500	100
50	10	yes	no	no
50	5	yes	yes	no
100	1	yes	yes	yes
200	0.5	yes	yes	yes
600	0.05	yes	yes	yes
Average Activity Bq/g	0.98	0.98	0.51	0.26

Section 28 paragraph 1 of the German Radiation Protection Ordinance, RPO [1], requires that

1. every unnecessary radiation exposure or contamination of persons, property or the environment be avoided,

2. every radiation exposure ... be kept as low as possible.

In fulfilment of section 28 paragraph 1 the SSK requires that, if technically feasible and economically justifiable, the material not be released but rather reused or recycled within the nuclear industry. By keeping the material under nuclear license the exposure to the surroundings can be monitored and controlled and exposure to the general public can be avoided.

With these points in mind the SSK issued recommendations regulating the release of metal scrap and components leaving a nuclear power plant, which are valid for activity spectra from light water reactors, LWR. Such spectra are dominated by ^{60}Co and ^{137}Cs. Furthermore the recommendations are explicitly not to be used if α-contamination is suspected. The recommendations for the release of ferrous and non-ferrous metal give clearance levels for the total activity and are summarized below.

- The first priority is to reuse or recycle within the nuclear industry, if this is not feasible then several release options are available.

1. Unconditional release is possible if the mass specific activity is below 0.1 Bq/g for each piece and the surface activity is below the limits prescribed by the RPO in annex IX column 4.

2. Release of scrap on the condition that it is melted is allowed if the mass specific activity is below 1.0 Bq/g for each piece and the surface activity is below the limits prescribed by the RPO in annex IX column 4.

3. A controlled recycling can be considered when the mass specific activity is higher than 1 Bq/g and/or the surface activity higher than the limits prescribed by the RPO in annex IX column 4. Within a controlled recycling the scrap is not released but rather melted under license. After melting the product metal can be unconditionally released if it meets the requirements put forth in point 1.

   On a case by case basis the licensing authorities can allow product metal with a mass specific activity between 0.1 and 1 Bq/g to be used in applications where an increased radiation exposure is not expected. This option is not of great practical importance.

The surface contamination clearance levels from annex IX column 4 of the RPO require that it be shown, for the entire surface, that the activity is less than

0.05 Bq/cm^2 for α emitters,
0.5 Bq/cm^2 for ß/γ emitters and
5.0 Bq/cm^2 for weak ß/γ emitters,

which are averages over any 100 cm^2 for total surface activity (fixed plus non-fixed). The requirement that both mass and surface specific clearance levels be implemented and the restrictive contamination clearance levels guarantee that the mass specific clearance level cannot be fully used. This leads to a significant reduction in the amount of activity which can be released and therefore to a reduction of the exposure to the general public.

2.1 Metal from Uranium Mining and Milling Operations

After reunification Germany became responsible for remediation of the eastern German uranium mining and milling operation WISMUT SADG. Two recommendations regulating the release of scrap metal and equipment from the mining and milling operations have been issued by the SSK [8]. Neither recommendation is applicable to facilities outside the uranium mining and milling operations in eastern Germany. Furthermore scrap and equipment from processing and enrichment operations cannot be released using these recommendations.

Unconditional release is possible if the total surface activity is less than 0.05 Bq/cm^2 averaged over large surfaces. Since α-emitting radionuclides are not expected to be present in the metal, a surface contamination limit is sufficient. For the release of smooth surfaced objects it is sufficient when inspection shows that the surfaces are clean and that no enclosed volumes, where dust and dirt could go undetected, are pre sent. The release after inspection is based on experience and spot check measurements and is valid for mining equipment which has been in contact with only uranium ore. The inspection should guaranty that the contamination is predominantly fixed.

A surface contamination clearance level of 0.5 Bq/cm^2 for the total α-contamination is allowed on the condition that the scrap be melted. Release of the scrap is possible if after sample measurements statistically show that the contamination is below 0.5 Bq/cm^2 averaged over 100 cm^2. Once again bulk contamination is not expected so that the surface limit is sufficient. Furthermore the scrap must be cut to furnace dimensions before it is released. This is meant to ensure that reusable pieces are destroyed beyond use and that segmenting the scrap occurs under radiological control, since it has been shown that reuse and segmenting of α-contaminated scrap lead to the highest doses [9,10].

3. SCIENTIFIC BASIS FOR THE CLEARANCE LEVELS

In the last years a number of studies using deterministic methods have been published which calculate the expected doses from recycling metal [11,12,13]. In these studies the parameters used to calculate the doses are set numbers, which imply that the processes leading to the radiation exposure are predictable. In reality the process of recycling is guided by chance, so that it is not possible to predict how a piece of scrap will be recycled and into which product it will be formed. Take for example the dilution in fig. 1, which is typically set to 1/10 in deterministic scenarios. The value of 1/10 has no experimental basis and cannot be guarantied, although it is very probable.

In Germany stochastic models of the complete process of releasing scrap, smelting, manufacturing and product use [6,7,9] have been developed and taken into account by the SSK. Unlike deterministic models, which produce fixed values for the dose, the stochastic models produce a distribution of individual doses coming from the exposures to the scrap during the various stages in the recycling process. From the individual dose distributions the collective dose can be calculated by appropriate integration. The stochastic models account for the release procedures described in section 2 and for the variation in the parameters, such as dilution. The model randomly choses parameters from the possible values such that the most likely values are chosen most often. This means that unlikely cases, such as no dilution, are not excluded from consideration but allowed to occur with a low probability. The individual dose distributions from the stochastic models for recycling scrap having a nuclide spectra typical of a LWR are shown in fig. 2 [6,7]. The maximum dose is greater than 10 μSv/a but the number of exposures is less than 1. This is the result of averaging the dose distribution over many simulations and means that the doses where the number of exposures are less than 1 did not occur in every simulation.

The preliminary results of a stochastic model for calculating the individual dose distribution from recycling α-contaminated scrap is summarized in table II. By far the most critical sector is the small scrapyard which includes manual segmentation of the scrap. The next most critical sectors are melting the scrap in a foundry and slag use or disposal. The Commission of the European Communities, CEC, in their research and development program are sponsoring a contract to study the behaviour of activity during segmentation [10]. The study will enhance model reliability by producing data for critical exposure scenarios like manual segmenting of scrap. Within this study revised dose calculations are being carried out [14].

4. APPLICATION OF THE RECOMMENDATIONS IN DECOMMISSIONING

The two recommendations for LWR scrap have been used in decommissioning operations. The problems associated with showing that the activity is below the clearance levels have been over come in two separate approaches [15,16], which will not be discussed any further here. In general large pieces are decontaminated and the activity measured while small pieces and hard to measure geometries are collected in containers according to material type before being sent for melting and eventual release.

In the CEC pilot nuclear reactor dismantling project, Gundremmingen (KRB) approximately 3300 tons of metal have been removed during dismantling to date and non-detrimentally recycled or reused at a cost of about 10 DM/kg. Using clearance

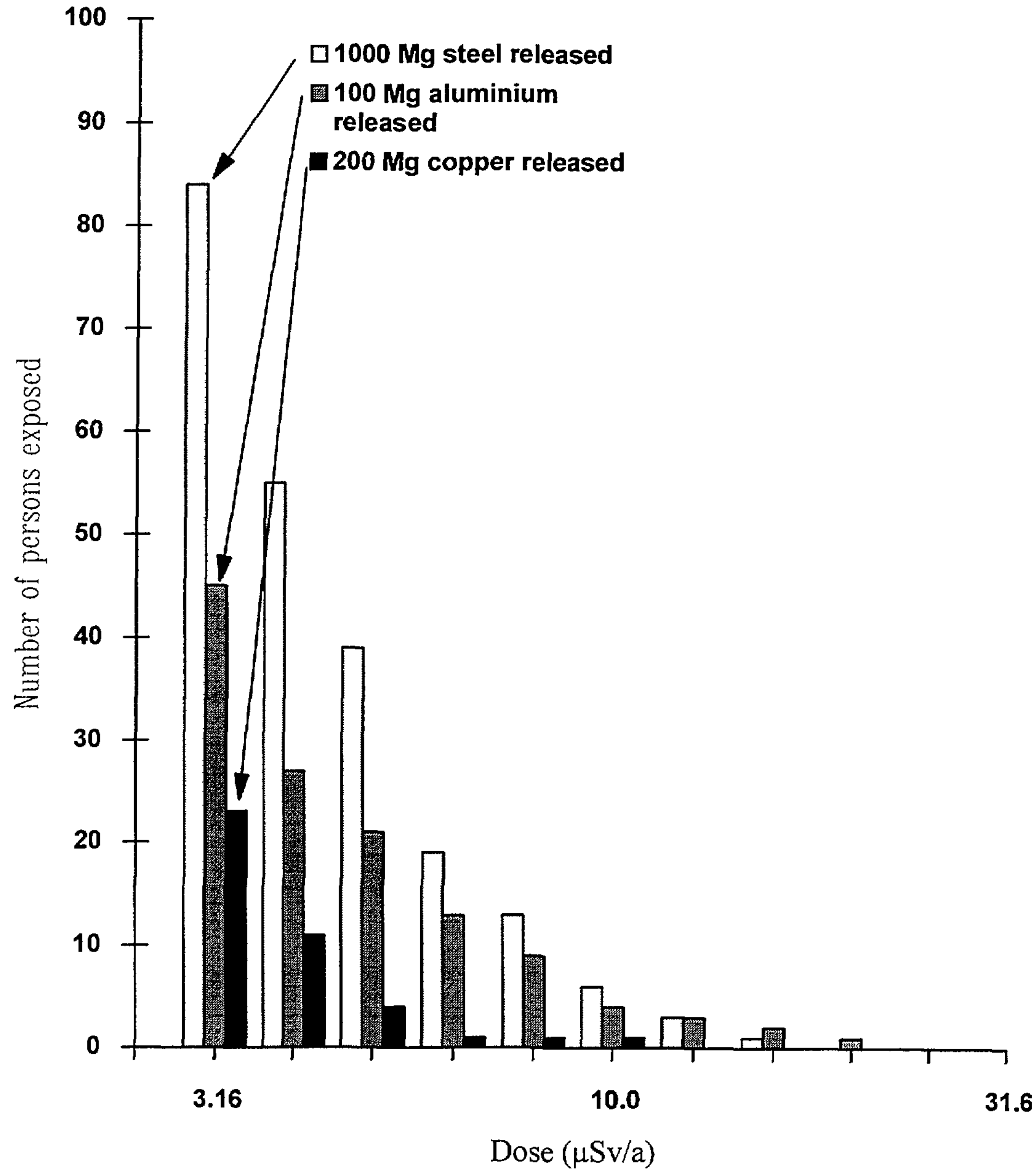

Fig. 2

Average Individual Dose Histogram
from 200 Simulations
Clearance Levels 0.37 Bq/cm² and 1 Bq/g

levels in accordance with the SSK recommendations KRB has unconditionally released about 1520 tons, the rest has, after melting, either been used within the nuclear industry or unconditionally released [17]. Of this metal about 118 tons are non-non-ferrous metals of which 112 tons were unconditionally released. A further 293 tons of electrical components, including electrical cables, have also been released. Reasonable estimates set the amount of ferrous metal released in Germany at around 1000 tons per year [6]. Of the non-ferrous metals aluminium, copper and lead are the most significant. The quantity of non-ferrous metal released per year depends strongly on the repair and decommissioning work in progress. Estimates for the

Table II: PRELIMINARY RESULTS OF THE STOCHASTIC MODEL FOR RECYCLING α-CONTAMINATED SCRAP [9]

	DOSE INTERVALS		
PATHWAY	**1 - 10 μSv**	**10 - 100 μSv**	**> 100 μSv**
small scrapyard	6	3	0.1
large scrapyard	0.01	---	---
foundry	0.4	0.01	---
slag use	0.3	0.01	---

10,000 simulations
200 tons of metal released
0.05 Bq/cm^2 as clearance level
50 % manually segmented (small scrapyard)

quantity released in Germany lie around 30 tons per year for aluminium, up to 200 tons of copper and copper alloys (a significant portion being electrical cables) and around 150 tons of lead. These quantities are expected to increase significantly during the next decade as more facilities are decommissioned [6,7].

The recommendations for α-contaminated scrap and equipment have not been extensively used due to the restrictive clearance levels and the large quantity of metal in the WISMUT operations, 20,000 tons is estimated for Crossen alone [8]. The average contamination on scrap and equipment from WISMUT mining operations is estimated to be between 0.5 and 2.4 Bq/cm^2 depending on the site, so that significant decontamination is required. The cost of unrestricted release is calculated to be much higher than the cost of the planned disposal options and therefore it is unlikely that unrestricted release will be pursued, although the final decision has not been made. The WISMUT is not required to recycling, as prescribed by the AtG, since the remediation is regulated by the German Mining Laws and not the AtG.

5. RELATED QUESTIONS OF REUSE AND RECYCLING

The SSK is working on a nuclide specific clearance level recommendation for disposal of waste and building rubble in conventional landfills. This recommendation is in an advanced stage and is expected to be published in 1994.

At present Brenk Systemplanung is working on studies for the German Ministry for Environment, Nature Protection and Reactor Safety, BMU, which can serve as the basis for clearance level decisions [9,18]. These studies include work on release criteria for α-contaminated metal, criteria for buildings and building material, criteria for the direct reuse of equipment and criteria for disposal to a conventional landfill. Both studies will be finished in 1993 and publicly available in 1994.

Standard procedures for measuring radioactivity, which are suitable for release criteria, are published by the Nuclear Technology Standards Committee from the German Standards Institute, DIN. Measurement standards and release procedures for metal containing beta and gamma activity are to be published in the near future [19]. At present standards for alpha activity are being drafted and standards for the release of buildings and building rubble are planned. Measurement standards are an integral part of the licensing procedure and the release criteria. They ensure that the activity is adequately characterized and quantified before a decision to release the material is made.

6. EXPECTED DEVELOPMENTS

The recommendations discussed above contain mass specific clearance levels which are not nuclide specific. From todays point of view, it appears desirable to reflect the relative contribution to the exposure of each radionuclide in the clearance levels. It is not adequate to give, for example ^{60}Co and ^{55}Fe the same ranking. Therefore it is greatly appreciated that the CEC has established a working party to achieve that goal [20].

Another important issue is the establishment of unconditional release criteria which are not restricted to a certain material or a special recycling pathway. Such criteria are required in cases where it is not certain that the material will be treated as expected or for the release of mixed material which can be separated after release. In practical cases the question wether certain items could be directly reused instead of being scrapped is frequently raised, the arrow leaving the scrapyard in fig. 1. As the criteria for unconditional release are seen as more fundamental from the point of view of radiological protection, the IAEA's efforts to establish such criteria are strongly supported [21].

7. CONCLUSIONS

Germany has established a concept of recycling which is designed to meet strict criteria concerning the exposure of the general public as well as the industrial need to recycle material. The essential elements of this concept are recycling within the nuclear industry, unconditional and conditional criteria for release, development of measurement devices and practices and regulatory control of release procedures. Probabilistic methods have been applied successfully to supplement the traditional deterministic scenario analysis in deriving clearance levels. Experience with this concept in recent decommissioning projects shows that industrial scale recycling can be conducted without jeopardizing the 10 μSv/a criterion. It is expected that international developments towards nuclide specific clearance levels and unconditional release criteria can be used to further improve this concept and bring the nuclear community closer to the required international consensus in this area of fundamental importance.

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ASSESSMENT AND PROPOSAL OF ACTIVITY LEVELS FOR RELEASE OF VERY LOW LEVEL RADIOACTIVE WASTE TO LANDFILLS

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Abstract

This paper reports on the results of a common NIRAS/ONDRAF (Belgium) - CEA/IPSN (France) project, within the framework of a study-contract with the Commission of the European Communities (No. CT-F12W/0060).
It describes the different landfill disposal techniques in France and in Belgium together with the exposure pathways and the parameters needed as input to the computer codes CERISE and GEOLE, which have been developed to assess the radiological impact of landfill disposal of very low level radioactive waste.
The results of the individual dose calculations for workers and the public and the derived activity concentration and contamination levels for acceptance to landfill sites will be presented.
Finally, a comparison will be made between the sets of derived levels as they are related to the different landfill disposal practices in the two countries.

1. INTRODUCTION.

Landfill disposal is a well-known, commonly used and rather cheap method for disposal of household and industrial waste. Waste management without landfill disposal is almost impossible.

In recent years a legal and regulatory framework has been developped in most European countries. In order to limit the environmental nuisances, waste acceptance criteria have been set up for the different categories of landfill disposal sites.
However, such quantitative acceptance criteria with respect to the radioactive content of the waste, do not exist in Belgium, and exist only since a short time in France.

In a study by NIRAS/ONDRAF (Belgium) and CEA/IPSN (France), acceptance criteria for landfill disposal have been derived in terms of radioactivity concentration (in Bq/g) and surface contamination (in Bq/cm^2). This study has been performed under contract with the Commission of the European Communities (No. CT-F12W/0060)[1].

In deriving these acceptance criteria (conditional clearance levels), and in order to make the assumptions as realistic as possible, the regulatory constraints for landfill disposal of non radioactive waste have been taken into account, as well as the daily operational practices on such sites.

2. METHODOLOGY.

For the purpose of exemption/clearance, an individual effective dose of some tens of microsieverts in a year can be reasonably regarded as trivial. Because an individual may be exposed to radiation doses from several practices that may have been judged exempt, it may be reasonable to apportion a fraction to each practice. Doses to individuals of the critical group of the order of 10 μSv in a year from each exempt practice have been suggested [(2)].

The following dose criteria have been adopted for the derivation of clearance levels :

- 10 μSv/year for exposure to the population, where the likelihood of being exposed to several exempt practices and sources is rather high;

- 50 μSv/year for exposure to smaller, specific groups, where the likelihood of being exposed to several exempt practices is rather small.

The types of landfill disposal sites have been selected on the basis of the national laws and regulations for landfill disposal of non radioactive waste. In most scenarios, it has been assumed that the very low level waste, which is a candidate for clearance, is mixed with ordinary industrial and/or household waste.

For each scenario, the doses to individuals have been calculated for a unit activity concentration of several nuclides (1 Bq/g) for different exposure pathways; a weighting factor of 0,01 has been used for skin dose. The total doses are then compared to the appropriate dose criterion to derive the corresponding activity concentration levels. Finally, the clearance level is the most limiting of these activity concentration levels.

A similar procedure has been followed to derive clearance levels in terms of surface contamination (in Bq/cm^2).

The following computer codes have been used (separately or coupled to each other) to perform the dose calculations :

- CERISE, to calculate doses to an individual due to external gamma- and beta-radiation, inhalation and ingestion of radionuclides; the dose-factors are derived from ICRP-publication no. 61 [(3)];

- GEOLE, to calculate the radioactivity released through the geosphere to an aquifer;

- ABRICOT, to calculate the radiological impact of the use of contaminated water.

3. DESCRIPTION OF THE SCENARIOS.

3.1. Types of landfill disposal sites.

In France, three types of landfill disposal sites have been considered :

- class 1 (industrial waste), with a capacity of 50 000 tonnes/year;

- class 2 (household waste), with a capacity of 150 000 tonnes/year;

- class 2, with a capacity of 15 000 tonnes/year

In Belgium, two types have been considered

- class 1 (industrial waste), with a capacity of 100 000 tonnes/year,
- class 3 (inert waste), with a capacity of 33 000 tonnes/year

3 2 Source terms

It must be realised that a landfill disposal site accepts wastes from different origins Following a conservative approach, the following masses of very low level radioactive waste have been assumed

- in France 1 000 tonnes per year and per site,
- in Belgium 100 000 tonnes over a period of 30 years sent to one class 1 landfill or to ten different class 3 landfills

3 3 Scenarios and Pathways

For the French landfill disposal sites, the following scenarios have been selected (see fig 1)

- engine driver external irradiation, external contamination (dust deposit on hands and face), inhalation of dust, ingestion via external contamination of hands, external irradiation and inhalation in case of trench fire,
- chemist external contamination of hands and face, ingestion via external contamination of hands,

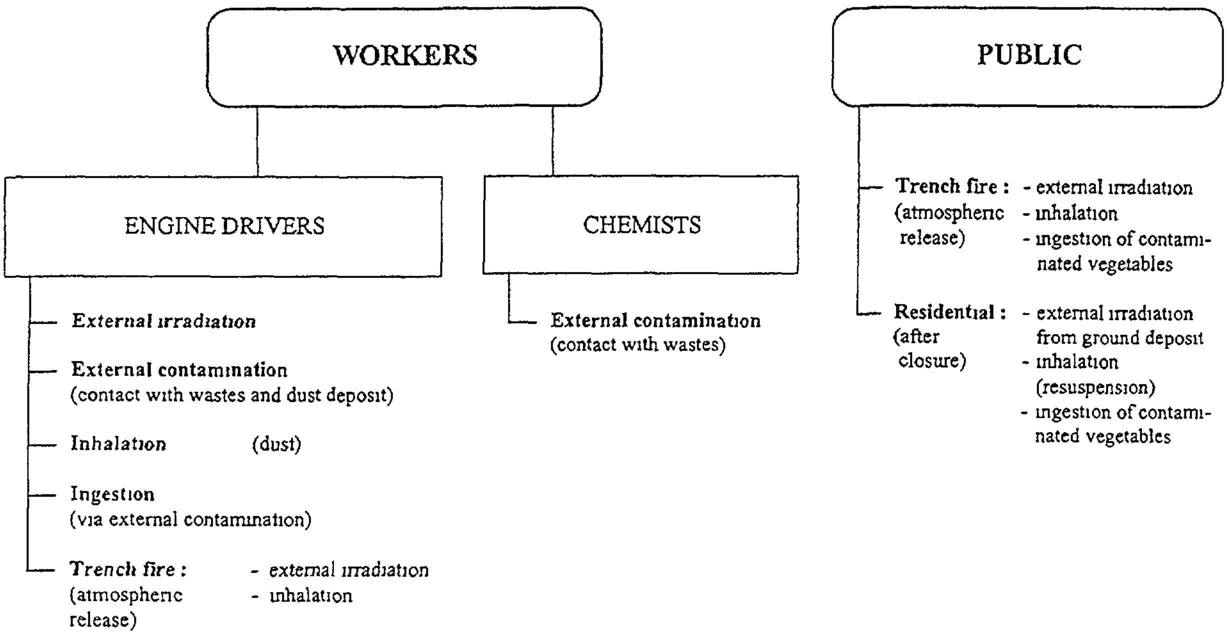

FIGURE 1 - SCENARIOS FOR FRENCH LANDFILL DISPOSAL SITES

- general public : external irradiation, inhalation and ingestion of contaminated vegetables after a trench fire; external irradiation from ground deposit, inhalation (due to resuspension) and ingestion of contaminated vegetables in case of intrusion after closure of the site (adult, 10 year old child, 1 year old baby).

For the Belgian landfill disposal sites, the following scenarios have been selected (see fig. 2) :

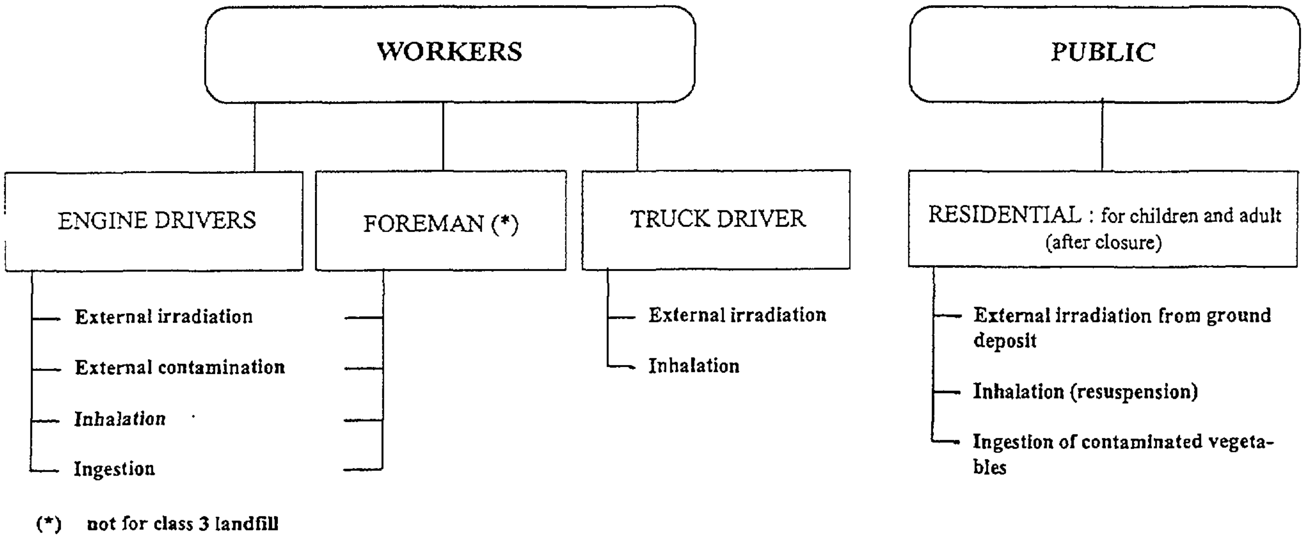

FIGURE 2 - SCENARIOS FOR BELGIAN LANDFILL DISPOSAL SITES

- engine driver : external irradiation and contamination, inhalation of dust, ingestion via external contamination of hands;

- foreman (only for class 1 landfill) : external irradiation and contamination, inhalation of dust, ingestion via external contamination of hands;

- truck driver (transport of waste to the landfill) : external irradiation, inhalation of dust;

- general public : external irradiation from ground deposit, inhalation (due to resuspension) and ingestion of contaminated vegetables in case of intrusion after closure of the site (adult, 1 year old baby).

In addition, the water pathway has been studied, based on data for a specific large class 2 landfill in France and a specific class 1 landfill in Belgium.

In all cases the exposure parameters have been determined on the basis of practical operating procedures and on the basis of measurements of dust concentration at different workplaces.

4. RESULTS AND DISCUSSION.

The clearance levels derived for landfill disposal are given in table 1 for the activity concentration and in table 2 for surface contamination (for a selected number of radionuclides only).

TABLE 1

Specific activity level, in Bq/g.

	Belgian Results		French Results		
Radionuclides	Class 1	Class 3	Class 1	large Class 2	small Class 2
H3	8,2E+05	2,5E+06	1,3E+05	1,3E+05	1,3E+04
C14	7,1E+03	2,1E+04	3,6E+03	1,1E+04	1,1E+03
Na22	1,9E+00	2,6E+00	3,0E+00	8,9E+00	3,6E+00
P32	1,9E+03	2,5E+04	1,1E+04	2,6E+04	7,8E+03
S35	1,4E+04	1,4E+05	1,9E+05	7,8E+04	2,4E+04
Cl36	3,1E+03	3,1E+04	1,1E+04	7,8E+03	2,5E+03
Mn54	5,0E+00	6,7E+00	7,7E+00	2,3E+01	9,2E+00
Fe55	1,8E+05	1,4E+06	1,1E+06	3,3E+06	4,2E+05
Co60	1,6E+00	2,2E+00	2,5E+00	7,5E+00	3,0E+00
Ni59	6,4E+04	6,4E+05	3,8E+05	1,1E+06	1,1E+05
Ni63	3,0E+05	1,2E+06	1,7E+05	4,4E+05	4,4E+04
Sr90+	2,3E+02	6,8E+02	7,0E+01	1,3E+02	1,3E+01
Nb94	2,6E+00	3,5E+00	4,1E+00	1,2E+01	3,2E+00
Tc99m	1,7E+03	2,3E+03	4,2E+04	2,0E+03	3,1E+03
Tc99	2,4E+02	7,1E+02	1,2E+02	3,6E+02	3,6E+01
Ru106+	2,1E+01	2,9E+01	3,3E+01	9,6E+01	3,8E+01
Ag108m+	2,5E+00	3,3E+00	3,8E+00	1,1E+01	3,1E+00
Sb125+	1,1E+01	1,4E+01	1,7E+01	5,0E+01	2,0E+01
I125	2,6E+03	5,6E+03	3,6E+03	3,6E+03	1,2E+03
I129	8,3E+02	2,5E+03	4,3E+02	7,0E+02	1,3E+02
I131	1,4E+01	1,8E+01	2,3E+01	5,9E+01	2,5E+01
Cs134	2,7E+00	3,6E+00	4,2E+00	1,3E+01	5,0E+00
Cs137+	7,6E+00	1,0E+01	1,2E+01	3,6E+01	1,2E+01
Pm147	9,4E+03	9,4E+04	6,8E+04	2,0E+05	2,8E+04
Sm151	1,5E+05	1,9E+05	2,3E+05	6,6E+05	7,9E+04
Eu152	3,0E+00	4,0E+00	4,6E+00	1,4E+01	5,6E+00
Eu154	3,4E+00	4,5E+00	5,3E+00	1,6E+01	6,3E+00
Ra226+	1,6E+00	2,1E+00	2,4E+00	7,3E+00	1,9E+00
U235+	1,7E+01	2,0E+01	2,4E+01	7,0E+01	1,1E+01
U238+	2,4E+01	2,6E+01	3,3E+01	9,5E+01	1,2E+01
Np237+	8,9E+00	1,0E+01	1,3E+01	3,6E+01	5,4E+00
Pu238	1,4E+01	1,4E+01	1,9E+01	5,2E+01	6,1E+00
Pu239+	1,4E+01	1,4E+01	1,9E+01	5,2E+01	6,1E+00
Pu240	1,4E+01	1,4E+01	1,9E+01	5,2E+01	6,1E+00
Pu241+	9,1E+02	9,6E+02	1,2E+03	3,5E+03	4,1E+02
Am241	1,4E+01	1,4E+01	1,8E+01	5,2E+01	6,1E+00
Cm244	2,3E+01	2,4E+01	3,1E+01	8,7E+01	1,0E+01

The results are also represented in fig. 3-4 for the activity concentration levels and in fig. 5-6 for the surface contamination levels.

TABLE 2

Surface activity level, in Bq/cm²

	Belgian Results		French Results		
Radionuclides	Class 1	Class 3	Class 1	large Class 2	small Class 2
H3	9,1E+05	9,1E+06	5,56E+05	4,2E+06	8,3E+05
C14	1,1E+03	2,2E+03	2,92E+03	8,8E+03	1,8E+03
Na22	1,4E+01	1,9E+01	4,01E+01	1,2E+02	4,6E+01
P32	5,9E+01	6,9E+01	3,64E+02	1,0E+03	2,5E+02
S35	1,1E+03	2,3E+03	3,03E+03	9,0E+03	1,8E+03
Cl36	1,1E+02	1,4E+02	7,06E+02	2,1E+03	4,4E+02
Mn54	3,5E+01	4,7E+01	1,05E+02	3,1E+02	1,2E+02
Fe55	4,5E+03	5,3E+03	3,68E+04	1,1E+05	2,3E+04
Co60	1,3E+01	1,7E+01	3,75E+01	1,1E+02	4,3E+01
Ni59	1,2E+03	1,3E+03	1,31E+04	3,9E+04	8,1E+03
Ni63	3,4E+04	9,8E+04	4,99E+04	2,2E+05	4,4E+04
Sr90+	3,5E+01	4,5E+01	2,08E+02	6,2E+02	1,5E+02
Nb94	1,9E+01	2,5E+01	5,26E+01	1,6E+02	6,0E+01
Tc99m	5,2E+03	6,9E+03	2,19E+05	1,0E+04	1,5E+04
Tc99	5,6E+02	1,1E+03	1,64E+03	4,9E+03	9,8E+02
Ru106+	3,4E+01	5,0E+01	1,50E+02	4,5E+02	1,3E+02
Ag108m+	1,6E+01	2,1E+01	5,21E+01	1,6E+02	6,1E+01
Sb125+	6,5E+01	8,7E+01	1,73E+02	5,2E+02	1,8E+02
I125	5,6E+02	9,8E+02	5,67E+02	3,0E+03	6,9E+02
I129	1,5E+02	9,8E+02	1,10E+02	7,3E+02	1,5E+02
I131	7,3E+01	1,1E+02	1,82E+02	4,8E+02	1,5E+02
Cs134	1,9E+01	2,5E+01	5,14E+01	1,5E+02	5,7E+01
Cs137+	5,1E+01	6,8E+01	1,17E+02	3,5E+02	1,1E+02
Pm147	7,9E+02	1,6E+03	2,14E+03	6,4E+03	1,3E+03
Sm151	2,5E+04	6,5E+04	4,72E+04	1,7E+05	3,4E+04
Eu152	2,1E+01	2,8E+01	7,41E+01	2,2E+02	8,3E+01
Eu154	2,5E+01	3,3E+01	6,48E+01	1,9E+02	7,0E+01
Ra226+	1,2E+01	1,6E+01	3,19E+01	9,6E+01	2,8E+01
U235+	1,2E+02	2,1E+02	2,41E+02	7,2E+02	1,9E+02
U238+	4,3E+01	6,1E+01	2,04E+02	6,1E+02	1,5E+02
Np237+	2,2E+01	1,4E+02	1,65E+01	1,1E+02	2,2E+01
Pu238	3,6E+01	3,6E+02	2,22E+01	1,7E+02	3,3E+01
Pu239+	3,6E+01	3,6E+02	2,22E+01	1,7E+02	3,3E+01
Pu240	3,6E+01	3,6E+02	2,22E+01	1,7E+02	3,3E+01
Pu241+	1,8E+03	1,8E+04	1,11E+03	8,3E+03	1,7E+03
Am241	2,7E+01	2,5E+02	1,67E+01	1,2E+02	2,5E+01
Cm244	5,4E+01	5,4E+02	3,33E+01	2,5E+02	5,0E+01

It can be seen that the derived levels vary between several orders of magnitude :

- from 1 to 1 000 000 Bq/g for the French class 1 landfill;
- from 0.4 to 3 000 000 Bq/g for the French class 2 landfills;
- from 1 to 800 000 Bq/g for the Belgian class 1 landfill;
- from 1 to 2 000 000 Bq/g for the Belgian class 3 landfill.

They reflect the very different radiological characteristics of the radionuclides.

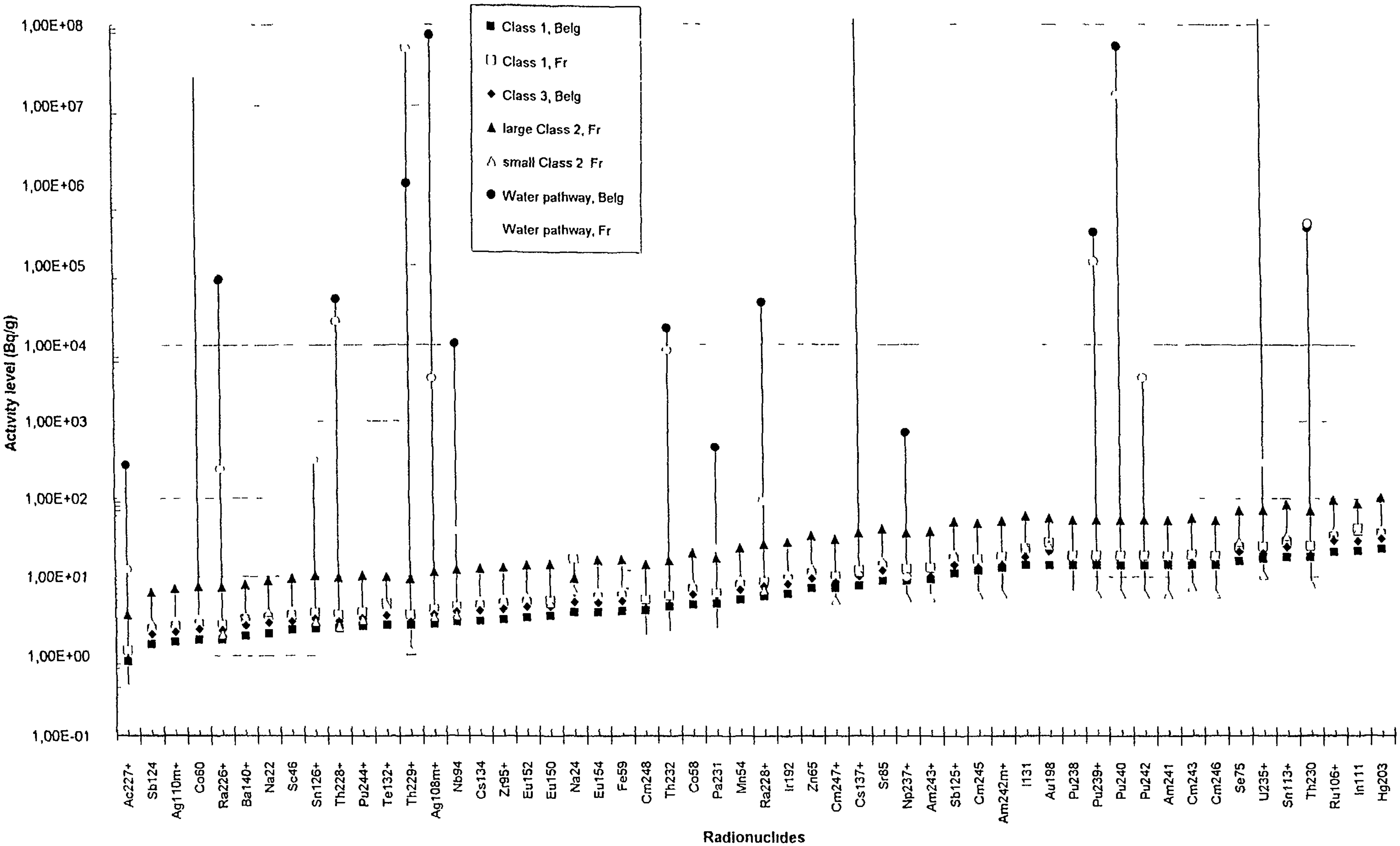

FIG. 3 **Specific activity level, in Bq/g, for France and Belgium (1)**

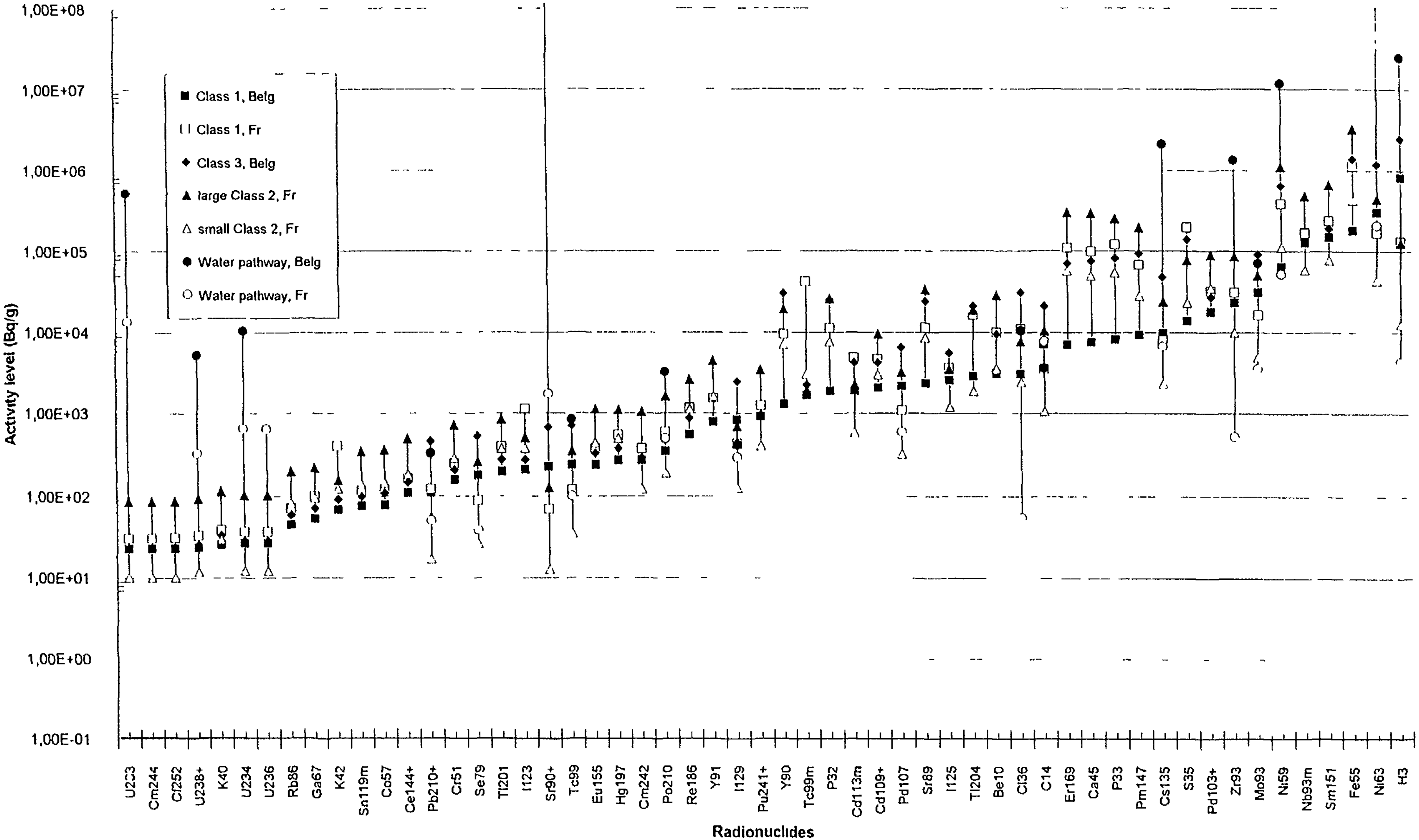

FIG. 4 Specific activity level, in Bq/g, for France and Belgium (2)

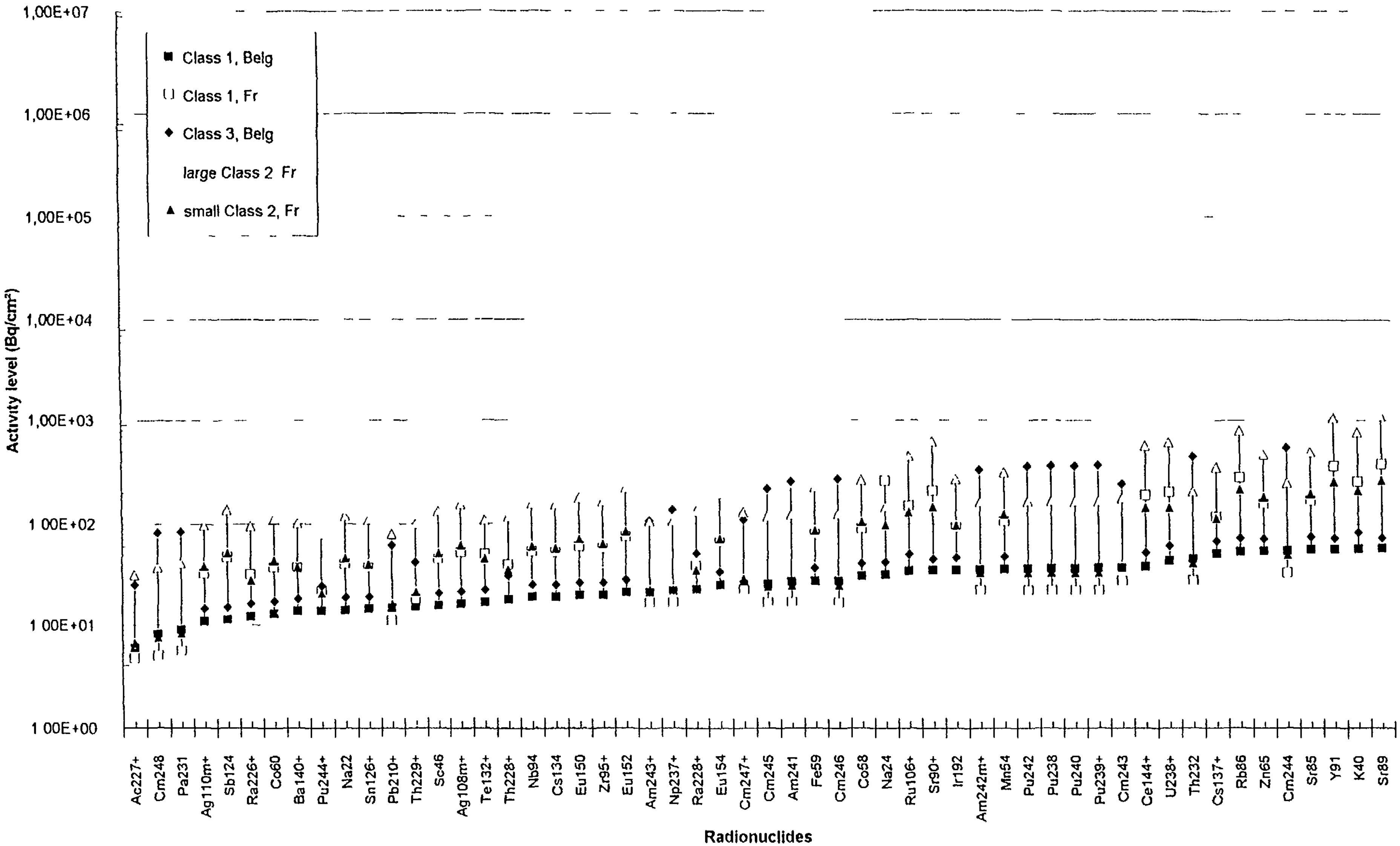

FIG. 5 **Surface activity levels, in Bq/cm², for France and Belgium.**

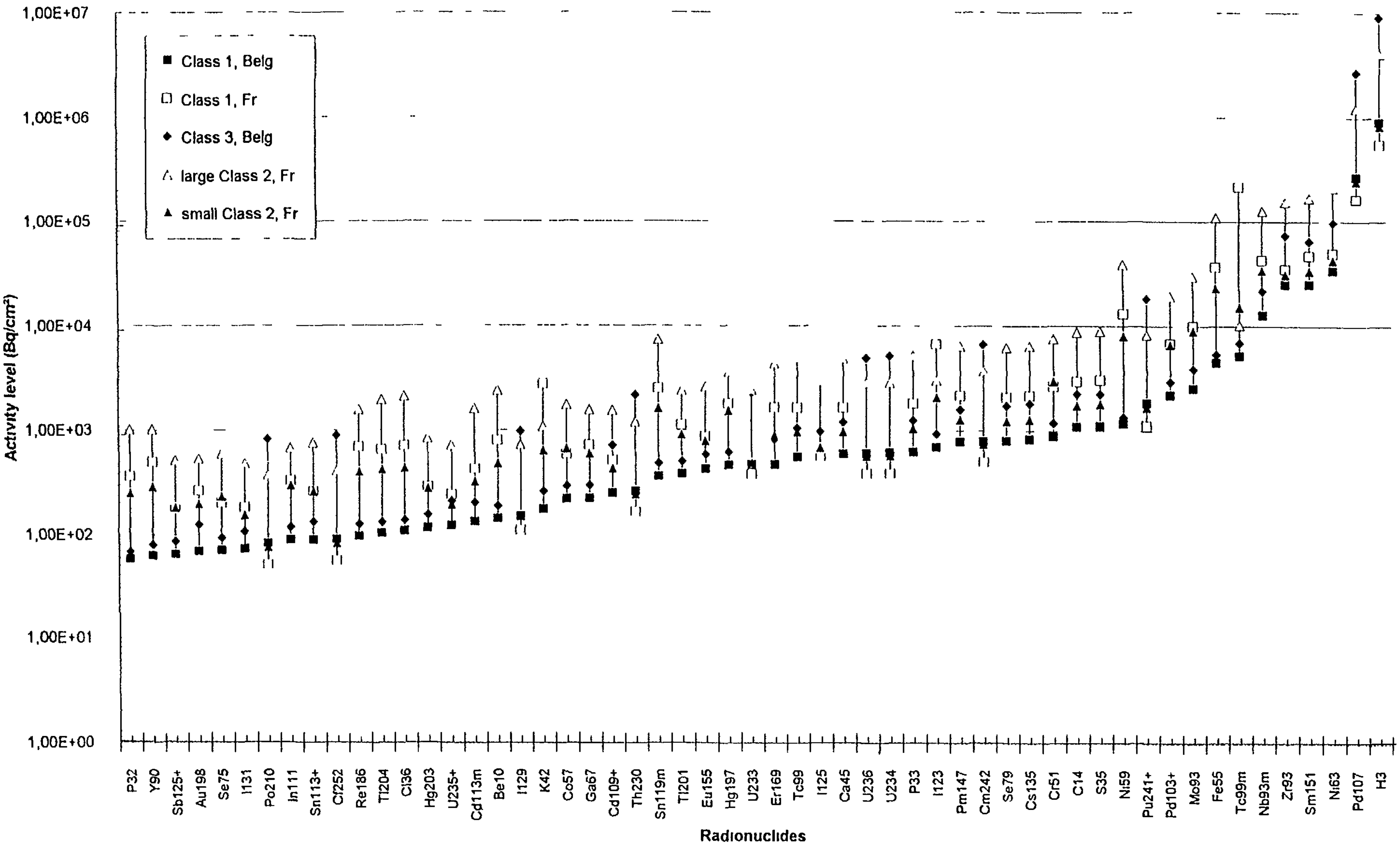

FIG. 6 **Surface activity levels, in Bq/cm², for France and Belgium.**

In all cases it is easy to see two groups of radionuclides for which the levels are lower :

- radionuclides with high gamma energy, such as Na-22, Co-60, Ag-110m; the limiting scenarios are those involving external irradiation;

- alpha emitters, such as Pu-239 and Am-241; the limiting scenarios are those where inhalation is involved.

For the other radionuclides, the derived levels are very variable, because not a single pathway is dominant.

The derived levels in terms of surface contamination are always higher than those in terms of activity concentration (sometimes an order of magnitude).

It can also be seen that the derived levels for Belgium and France (water pathway not taken into account) are comparable and in most cases differ not more than a factor of 10.
The clearance levels for landfill disposal are in general larger than those for recycling.

The water pathway is not limiting for the Belgian landfill; for the French landfill, it seems to be significant only for tritium.

Finally, the dose to the truck driver appears to be the limiting factor for many radionuclides. It is therefore important to define clearly the "practice" to be studied and the associated scenarios and exposure pathways.

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THE EXEMPTION AS A METHOD FOR THE MANAGEMENT OF BY-PRODUCTS CONTAINING RADIOACTIVITY. THE SPANISH SITUATION

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Abstract

At the present there are no specific criteria in Spain governing exemption from regulatory control for radioactive sources and practices involving radioactive material, as regards final management. Nevertheless, the regulations currently in force, which are in line with those established in the CE, indicate those cases in which certain such sources and practices are outside their field of application. Spain also participates actively in the most up to date developments relating to these issues at international level, fundamentally within the framework of the IAEA and CE. Some activities of interest undertaken in Spain over the last few years are described in this paper.

1. INTRODUCTION

The basic standards and recommendations regarding Radiological Protection published by different International Organizations provide a basis for Regulatory Control of activities involving ionizing radiations. This Regulatory Control is based on a system of Notification, Registration and Authorization, allowing the designated authority to establish the requirements necessary for protection. In certain cases, however, this level of control is either impossible or unnecessary, as a result of which, exclusions and exemptions are established.

Exemptions are established in order to avoid the application of excessive regulatory procedures to clearly justified practices, whose low level of significance removes the need for regulatory requirements. They may be applied to various types of practices ranging from the use of a given product to its final disposal at the end of its useful lifetime.

On the other hand, the use of radioactive materials and sources in various fields of science and technology, such as the generation of electricity by nuclear means, generates considerable volumes of by-products containing (or contaminated with) radioactive isotopes, both during the operating lifetime of the installations involved and during subsequent dismantling.

In many cases, these by-products have a very low level of activity, as a result of which they might be disposed of as normal wastes, using conventional methods, or be reused. In order for this to be possible, it is necessary to establish conditions for these by-products - subject during generation to the control system - to be manage as exempted from regulatory controls during their final disposal.

There is, therefore, a conceptual distinction between the exemption applied to new practices involving radioactive materials, and regarding their being subject or not to regulatory controls, and the exemption applied to by-products or wastes generated by practices already subject to regulation. This conceptual and theoretical distinction may in practice lead to a series of different and rather serious disadvantages, since it is quite feasible that controls will be demanded in the final management of materials not subject to any such control during use. This question is undoubtedly important as regards public understanding and acceptance, and it may well be advisable to consider the applicability of the principle that exemption of a given practice should take into account the entire process, including final management of the wastes or by-products generated by it, as recommended by the ICRP.

As regards the coherent setting up of an internationally well founded policy of exemption, mention should be made of the developments under way at the IAEA and the EC, with respect both to principles and criteria for practical application. Also to be underlined is the revision of the Basic Safety and Radiological Protection Standards performed by these two Organizations, a process in which the area of exemption from regulatory controls is expressly included and covered to some degree of detail, in keeping with ICRP recommendations.

The advantages to be gained through development in Spain of a widespread policy of exemption are clearly the same as would be achieved elsewhere. In this respect the following, among others, may be singled out for special mention:

- Reduction of the necessary capacity of radioactive waste disposal and storage installations, licensed as such.

- Optimization of the use of the capacities of Organizations involved in Radiological Protection.

- Reduction of the overall cost of radioactive waste management.

- Possibilities for advantage to be taken of certain materials following recycling.

- Improvements to the operating conditions and performance of licensed installations, due to reductions in the volume of radioactive wastes disposed of and, consequently, simplification of management constraints.

Additionally, and in a less tangible manner, a contribution might be made to improving understanding by society of the reality and perspectives of radiological risk.

2. THE SPANISH SITUATION

The existing regulations, which are in line with those established in the CE, indicate those cases in which certain sources and practices are exempted. Additionally and within the framework of its habitual licensing and operational control activities in relation to certain installations, the Nuclear Safety Council (CSN) has paid specific attention to the free release of certain quantities of radioactive materials, usually very low level liquid wastes. This is the case in general for hospital facilities, especially those in which small quantities of short-lived radionuclides are handled.

The basic criterion applied by the CSN is the obligation of the licensee to provide documented evidence, in the evaluation of radioactive liquid waste management, that the so-called Regulatory "Annual Incorporation Limits" have not been exceeded.

It should also be pointed out that, in very specific cases, the CSN has favourably viewed conventional disposal of the low level solid wastes generated by these facilities, when such wastes have had a specific activity of less than 74 Bq/g (2nCi/g), this being the value for exemption included in the Community and Spanish standards. In such cases, the licensee of the facility must provide documentary proof of the availability of suitable procedures and present his estimates of activity on the basis of the most restrictive radionuclides.

Generally speaking, the conventional disposal practices favourably looked on by the CSN have been very few, and have been practised by facilities equipped with Radiological Protection Technical Units (or Services).

In relation to this question, attention should be brought to the invitation for comments in relation to the draft version of CSN Guideline 7.8, "Criteria and technical basis for the exemption of radioactive materials from Regulatory Control", distributed in June 1993. This guideline refers fundamentally to methodology, and does not include the derived values which are essential for producers to be able to manage their wastes directly by means of conventional techniques.

3. EVOLUTION OF THE TREATMENT OF EXEMPTION BY ORGANIZATIONS OF RELEVANCE TO SPAIN

In recent years, the importance of this issue has led different international organizations to define principles for exemption, with a view to determining the quantities of activity in materials and by-products required for them to be considered exempt and allow the management of such materials and by-products to be accomplished by means of less controlled methods.

3.1 IAEA

In this respect, and in view of the obvious need to reach agreements allowing basic principles and criteria to be established at international level, for subsequent development by each nation, the IAEA has promoted different tasks in cooperation with other significant Organizations.

These tasks finally resulted, in September 1988, in publication of safety guide number 89, "Principles for the Exemption of Radiation Sources and Practices from Regulatory Control". This document may be said to have clearly established the basic principles for exemption, and as from its publication these principles began to be applied to specific cases, and levels are being derived for practical application.

These well-known criteria are graphically summarized in Figure 1. Using them as input, the IAEA has undertaken an ambitious programme of development of methods and practical guidelines for exemption, for possible direct (or simple) application by different countries. Spain participates actively in several of these developments.

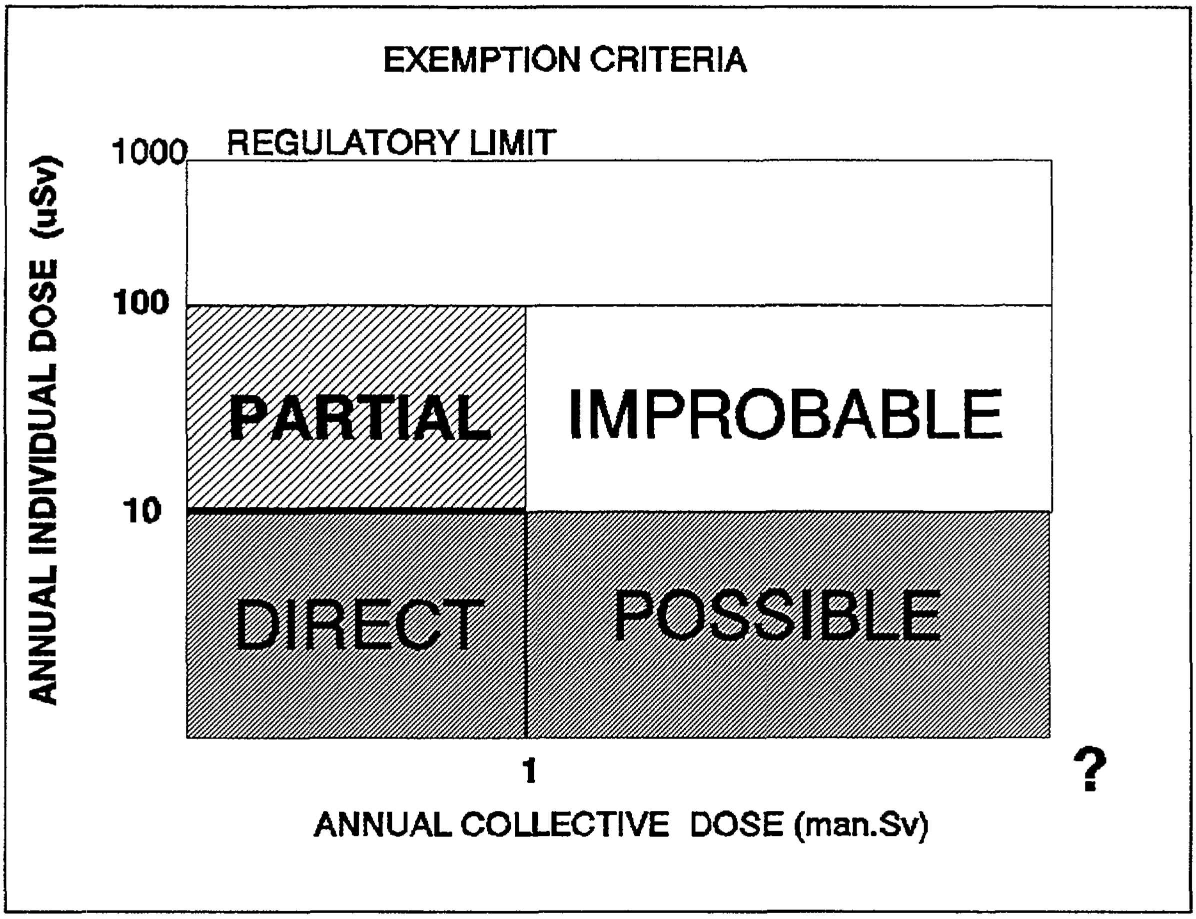

FIGURE 1

The IAEA's Basic Safety and Radiological Protection Standards, currently under revision, are considered to be especially relevant in this respect. These Standards expressly include exemption from regulatory control, although when it comes to establishing numerical derived levels problems occur, as a result, may be, of attempts to achieve a situation of coherence with the currently on-going revision of the CEC's Directive.

The basic difficulty arises when attempting to achieve a degree of coherence between the levels derived for exemption of regulatory requirements (notification, registration and authorization) and those for the less controlled disposal of materials and sources previously subjected to Regulatory Control. This has given rise to the appearance of a new concept: the "clearance" levels, which would be applied in order to free from Regulatory Control requirements those materials and by-products arising from practices and sources subject to such controls during their useful lifetime.

Unfortunately, in order to achieve this objective of coherence in terms of derived values, agreement is being reached as to these clearance values not being higher than the values used for exemption. While this statement may be acceptable in term of dose or risk to the population, it is not, in our view, a solidly based approach, because if different scenarios are assumed for use and disposal, the resulting derived levels may be different. Furthermore, the draft text available, issued in July 1993, indicates that clearance values should not be higher than those established in terms of total activity and concentration of activity, isotope by isotope, in a table showing the values derived for "direct exemption".

Fortunately, the IAEA leaves open a door for decisions by each individual Regulatory Body, which may approve other values as long as these meet the basic criteria in terms of dose limits, which are perfectly expressed.

It should also be pointed out that, although the IAEA considers exemption for the final disposal of radioactive wastes to be feasible in the terms indicated, application thereof is not clear with respect to those products whose use is exempt, the decision being left open to the Authorities of each country. This leaves unresolved the long-standing contradiction that something exempt in use may not be exempt as regards disposal. In the same way, it is indicated that conditional exemption is possible in those situations determined by the Authorities, based on physical or chemical form or for the use or disposal of radioactive materials.

3.2 ICRP

Parallel to the above, the new recommendations of the International Commission on Radiological Protection (ICRP), issued in November 1990, explicitly mention exemption from regulatory control as a necessity, and include a reference to the aforementioned IAEA guide (SS-89).

These recommendations indicate that sources and practices may be exempt either because they produce low individual and collective doses or because the control procedures which would have to be established would not be warranted in view of the dose reduction achieved.

In any case, exemption is based on insignificant doses, event though these are very difficult to establish and there is no clear definition as to when an individual or collective dose is sufficiently low for exemption to be possible. In this respect, optimization of Radiological Protection continues to play a fundamental role.

3.3 CEC

As regards the European Community, it should first be pointed out that the basic Regulations governing Radiological Protection currently in force (80/836 and 84/467) state that the use of certain radioactive materials is exempt from all requirements regarding declaration or authorization when the levels of activity involved are below certain values indicated in tables. Nothing is said of the final disposal of such materials nor of the use of the aforementioned values for the general disposal of radioactive materials.

In view of their mandatory nature, these Regulations have been reflected in the regulatory documents of the Member States; thus, for example, they appear in the latest revision (January 1992) of the Spanish Regulations governing Protection against Ionizing Radiations. These values have been used, and continue to be used at random, as values for exemption of disposal (clearance), without any consistency as regards Radiological Protection.

The aforementioned basic Regulations are now being revised in view of the latest ICRP recommendations and the progress made in establishing criteria for exemption. In this respect, the available draft of the new Regulations (August '93) includes exhaustive tables of values, in terms both of total activity and of concentrations thereof, below which a source or practice is exempt from application of the Standard, including final disposal. These values could be interpreted also as upper limits which may be used for the disposal of other materials and by-products arising as a result of regulated activities.

These new values differ appreciably from those previously existing, and have been calculated taking into account all scenarios of use and misuse of the different radionuclides in all their different forms (solids, liquids, gases, flat sources, encapsulated sources, etc.) and using a value of 10 μSv/year (a dose value to the skin of 50 mSv has also been used for specific practices and radionuclides) as the upper dose value for individuals and a reference collective dose of 1 Sv x man, values corresponding to automatic or direct exemption in accordance with the criteria and principles of SS-89.

Unfortunately, as it stands, the draft of the Regulation includes certain potential problems, as indicated thereafter: 1) Nothing is indicated with respect to the disposal of products having levels of activity higher than those reflected in the appendix thereto, and use of which is exempt due to their being homologated; 2) Another statement which would be difficult to explain is included in the preamble to the aforementioned appendix, and states that the values in terms of concentration of activity correspond to the use of moderate quantities, and 3) the Regulations indicate that the disposal of radioactive substances or the recycling of materials from licensed installations are subject to authorization, and the possibility for any national authority to supress this requirement may be interpreted to be constrained by the values in the appendix.

Be that as it may, it should be recognized that work on exemption began within the framework of the EC before 1989, and that since that date it has received considerable impulse, among other reasons because the establishment of an harmonious and homogeneous policy of exemption has been recognized as an important factor as regards various aspects relating to the single market and the social acceptance of radioisotope use.

Among the works performed in this respect, special mention should be made of the publishing of document No 43 of the "Radiation Protection" series, in 1989, which refers specifically to the exemption of recycled steels and defines a methodology and a series of practical, easily applied derived limits. This document is currently being revised.

An important document has also been published in relation to the exemption of waste streams generated in hospitals, research centres and industry (minor producers) (EUR 14520-1992). This document is not, however, a regulatory instrument and simply describes the current situation; on the other hand it is of significant importance, since it might serve as a basis for development of standards at community level.

Finally, it should be pointed out that basic activities have been initiated in the EC with a view to exempting contaminated concrete materials and metallic materials, such as copper, aluminium, etc.

4. ACTIVITIES CARRIED OUT BY ENRESA IN SPAIN

Since 1988, ENRESA has been carrying out a programme of activities ultimately aimed at facilitating the application of exemption for the Spanish regulatory authorities, with particular attention paid to its use for the final management of radioactive wastes and by-products by conventional routes.

This programme has been performed with participation by two Spanish engineering companies (Empresarios Agrupados and INITEC) and the collaboration of the Centre for Energy-related, Environmental and Technical Research (CIEMAT), and has included adaptation to the Spanish situation of assessment methodologies developed by organizations from other countries, for specific application. The programme has included different waste streams, both currently existing and/or which might appear with time in the country and might be disposed of via conventional routes under different conditions.

The following are examples of such waste streams:

- Biological and organic wastes not included in the fuel cycle.
- Non-homogeneous solid wastes.
- Sources, encapsulated or otherwise.
- Metallic scrap.
- Rubble, conditioned or otherwise.

Different specific applications have also been undertaken within this project, such as the one relating to ionic smoke detectors and already submitted to the authorities (April, 1993), proposing exemption for the disposal of disused ion smoke detectors with Am-241 activity levels below 37 KBq.

The values obtained for the different streams are in keeping with those included in the different documents of the IAEA and the EC for free disposal, and even with those indicated for the exemption of practices and sources.

The documents drawn up during the course of these projects were delivered to the CSN in July 93, and a request was filed for acceptance of the methodology applied in the studies submitted and, where applicable, for approval of the "derived clearance values" proposed for the streams analyzed, for use both by ENRESA and, potentially, by those national waste producers having agreements with ENRESA.

The scope of these projects and the results obtained are described below.

4.1 Exemption of biological solids and organic liquid wastes

The waste streams and annual quantities were selected on the basis of a survey performed among those minor producers (hospitals and research centres) who are clients of ENRESA. From this study it was deduced that most solid and aqueous liquid wastes are managed by the producers themselves, a type of exemption in fact being applied. Wastes containing isotopes with longer half-lives and those posing difficulties as regards disposal are delivered to ENRESA for final management; these are biological solid wastes, mainly animal cadavers, and organic liquid wastes (scintillation liquids).

Generically speaking, disposal of the wastes involved in the selected streams would consist of incineration and disposal of the resulting ashes in a municipal landfill, although more specifically two different disposal practices are contemplated: incineration of the annual national production of the two streams in a municipal incinerator and disposal of the resulting ashes at a municipal landfill, and incineration of the wastes from both streams produced at a single centre of maximum production in an incinerator located at this centre itself, with subsequent disposal of the ashes at a municipal landfill.

The aforementioned disposal practices are performed in three clearly defined stages: transport, both of the active wastes to a municipal incinerator and of the resulting ashes to the landfill; incineration of these wastes, along with other non-active wastes; and disposal of the ashes at a municipal landfill.

Following definition of the practices and of the stages included in each, scenarios were determined in order to calculate their potential environmental impact. Table I summarizes these scenarios and exposure pathways.

Evaluation of environmental impact included estimation of the dose received by the most exposed worker and by the overall group of workers for each operating stage, and of the dose received by an individual belonging to the most exposed population and the overall population in a radius of 80 km. As regards transport, the population selected was that living on the transport route.

The results obtained show that the least favourable case is incineration without filters. It may also be appreciated that the highest doses, both individual and collective, correspond to incineration. The doses for transport are significant, in comparative terms, in the case of biological solids, and insignificant in the case of organic liquids, as a result of the isotopes considered. In any case, the resulting doses are much lower than the values recommended by the IAEA (9,36E-2 μSv/year for the most exposed individual).

TABLE I EXEMPTION OF WASTES ARISING FROM HOSPITALS AND RESEARCH LABORATORIES (BIOLOGICAL SOLID WASTES AND CONTAMINATED LIQUID SCINTILLANT)

INCINERATION AND DISPOSAL OF ASHES IN A SANITARY LANDFILL

SCENARIOS	EXPOSURE PATHWAYS
1) Waste transport . Transport workers . People on route	 - External - External
2) Waste incineration . Plant workers . People off-site incinerator	 - External - Inhalation - Inhalation stack emissions
3) Waste disposal . Landfill workers . People off-site landfill	 - External - Inhalation - Inhalation of resuspended particles and erosioned particles windspread
4) Postclosure scenarios . Intruder construction and residential . Surface and ground water migration . Erosion scenarios	 - Inhalation and submersion in cloud - External - Water ingestion

4.2 Exemption of non-homogeneous solids waste

In both this study and in those described in the following sections, the main objective was to obtain derived levels for generic exemption which might be applied directly by the producer.

Two sub-streams were considered in this case: non-homogeneous solid wastes generated by minor producers and those produced by nuclear power plants. Annual quantities were defined.

The first type consists of solid wastes of different types, with the exception of biological solids, such as the rags, cotton waste, paper, gloves, plastics, glass, etc. generated at hospitals and medical and research centres.

Those generated at nuclear power plants correspond to two types of contaminated solids: compressible solids such as clothing, rags, plastics, filters, etc., and non-compressible solids such as wooden boxes and planks, fundamentally. Evidently, these wastes originate in all the nuclear power plants in the country.

The components of these wastes are similar to those found, in large quantities, in the urban solid wastes sent to incinerators or tips; consequently, incineration and disposal of the resulting ashes at the tip or direct disposal along with conventional refuse are the most suitable conventional management methods for these wastes. These scenarios are the ones considered in the study, and coincide generically with those considered for the exemption of organic liquids and biological solids (Table I), since the management method proposed is the same.

In this case the individual and collective radiological impact associated with disposal of the aforementioned wastes was calculated, with a concentration of activity of 1 Bq/g, the derived levels having been determined from the result obtained and through comparison with the dose levels for exemption. The results obtained are shown in Table II.

It should be pointed out that when the levels derived for exemption are expressed in terms of activity concentration (Bq/g), reference is to the wastes themselves, without any dilution at the point of origin. Likewise, when calculating these levels, consideration is given to the total quantities in terms of mass, such that the total activity released to the environment is limited.

4.3 Exemption of encapsulated or non-encapsulated sources

Sources, encapsulated or otherwise, are one of the most hotly discussed streams as regards analysis of the feasibility of their exemption. The problem is that in the case of sources, the activity is highly concentrated and although activity might be irrelevant the values involved may, in terms of concentration, be very high, this causing considerable confusion, especially among the general public.

In any case, the exemption of very low level sources disposal is a feasible proposition, and a study centering on the four following streams, typical of very low activity sources, has been performed: ionic smoke detector sources, calibration sources,

TABLE II REFERENCE EXEMPT QUANTITIES FOR SOLID DISPOSAL TO AN INCINERATOR OR TO A LANDFILL

RADIONUCLIDE	ACTIVITY CONCENTRATION (Bq/g)
H-3	1,98E2
C-14	5,29
P-32	4,01E2
S-35	3,34E3
Ca-45	4,63E3
Cr-51	1,51E2
Mn-54	5,17
Fe-59	3,69
Co-57	7,63E1
Co-58	4,37
Co-60	1,74
Ga-67	4,52E1
Rb-88	9,16
Sr-90	9,13E1
Y-90	1,33E3
Zr-95	5,62
Tc-99m	6,94E1
In-111	1,76E1
Sb-124	2,22
I-125	1,21E1
I-131	8,02
I-132	2,19
I-133	1,10E1
I-134	2,69
Cs-134	2,65
Cs-137	6,67
Tl-201	8,64E1

Reference quantity 148 t/year (all production)

sources for luminous devices and sources for chromatography equipment. Information on these sources has been obtained through interviews with different agents, covering practically the entire scope of supply. Also used have been data on wastes compiled by ENRESA between 1987 and 1991.

As regards the activity of each of the sources analyzed, it may be pointed out that the value of the smoke detectors varies from 18 KBq (0.5 μCi) to 37 KBq (1 μCi); that the calibration sources vary from 370 KBq (10 μCi) to 3700 KBq (100 μCi); and that the chromatograph sources are in the range of 370 MBq (10 mCi) to 555 MBq (15 mCi). In the case of the lightning devices, and since only tritium with its very low radiotoxicity is used, the activity values may reach 2,220 GBq (60 Ci), with a source range of between 22,2 GBq (0.6 Ci) and 74 GBq (2 Ci).

The methodology used to obtain exemption values does not vary with respect to that used for other streams, although in this case certain peculiarities are included, such as rupture of gaseous sources or the ingestion of solid sources.

In any case, the management mode is, once again, disposal to a tip with domestic refuse and possible incineration, the method normally used for processing of this type of refuse. For this reason, the scenarios and exposure paths are those indicated in Table I, with addition of the accidental inhalation and ingestion scenarios.

4.4 Exemption of metallic scrap

In this study consideration has been given to the steel, copper and aluminium scrap produced during the operating lifetime of Spanish nuclear power plants as a result of maintenance tasks, enhancements and design changes.

It is difficult to estimate the volume of production of this type of scrap, since it is not produced regularly; nevertheless, and on the basis of the experience gleaned to date (Almaraz, Trillo and Santa María de Garoña plants), an average volume of 21 tons/year may be considered for all the Spanish nuclear plants.

This waste stream is mainly made up of small metallic components such as pipes, supports, valves and filters, the base material of which is stainless and carbon steel. No segregation of any kind is carried out at the point of origin. The copper and aluminium components are not discarded, but decontaminated and reused at the plant itself.

From the radiological point of view, the origin of the activity may be of two types: contamination and activation.

In addition, and with a view to completing this waste stream, the metallic materials which might be generated during dismantling of the Vandellós I nuclear power plant have been included, along with an estimate of those which might arise during dismantling of a 1,000 MWe water-cooled reactor in a year, obtained from bibliographical studies (Reference 4).

In this case, the conventional management modes analyzed are recycling of scrap and the reuse of components. The associated scenarios and exposure pathways are shown in Table III. In these scenarios the main exposure path is external radiation, and the difference between them is the size of the source, its geometrical shape and the distance to the affected individual, this leading to identification of up to 21 different source models. Consideration is given also in the calculations to the fact that a part of the activity present in the scrap is transferred during the different processes to the slag or by-products of recycling. The results obtained in the case of steel are shown in Table IV.

4.5 Exemption of dismantling rubble

As in the previous cases, this project has been aimed at obtaining derived values for exemption in terms of activity concentration, although the methodology used has been somewhat different.

The volumes of concrete which may be expected during the dismantling of a nuclear power plant will either not be contaminated or activated, or will be so only to very low levels. Consideration has been given to some 5,000 to 8,000 tons, distributed in different intervals of activity and with a given radionuclide content. Likewise, consideration has been given to the time elapsing following reactor shutdown, since this parameter has an important influence on the spectrum of radionuclides present in the concrete.

The streams having been defined, two types of management modes have been considered: disposal to a tip and recycling or reuse of the material.

TABLE III EXEMPTION OF RECYCLE AND REUSE OF METALLIC SCRAP

STEEL	
SCENARIOS	**EXPOSURE PATHWAYS**
- Workers . Scrap delivery . Scrap storage . Smelting . Slag storage . Sheet . Fabrication plant . Distribution	 - External - External, inhalation and ingestion - External, inhalation and ingestion - External, inhalatión and ingestion - External, inhalatión and ingestion - External - External
- Consumer user	- External and ingestion
- Reuse of equipment and tools	- External, inhalation e ingestion
ALUMINIUM AND COPPER	
SCENARIOS	**EXPORURE PATHWAYS**
- Workers . Scrap delivery . Refinery furnace . Storage . Copper tail refinery . Slag refinery . Copper electrolysis . Fabrication plant . Zn recovery . Unload slag	 - External - Inhalation - External - Inhalation - External - Inhalation (Al) - External - Inhalation - Inhalation - External, inhalation e ingestion
- Consumer user	- External and ingestión

TABLE IV DERIVED EXEMPTION LEVELS FOR THE RECYCLE OF CONTAMINATED STEEL

RADIONUCLIDE	DECOMMISSIONING (Bq/g)	REUSE (Bq/cm^2)
Mn-54	3,13E1	5,18E1
Fe-55	2,15E5	2,90E3
Co-58	6,49E1	1,06E2
Co-60	3,01	1,19E1
Ni-63	4,24E5	7,94E3
Zn-65	5,46E1	8,33E1
Sr-90	2,38E2	5,18E1
Nb-94	1,61	1,70E1
Tc-99	1,74E4	2,39E3
Cs-134	1,10E2	2,01E1
Cs-137	7,52E1	4,10E1
Sb-124	4,12E1	6,54E1
Eu-152	5,81	4,18E1
U-238	4,42	4,08
Pu-239	1,91	1,21
Pu-241	9,98E1	6,25E1
Am-241	1,86	1,18
Cm-244	3,25	2,08

Disposal includes two stages: transport of the rubble from the centre to a controlled tip, and disposal at this tip, where it is mixed with inactive material.

Recycling, on the other hand, includes processing of the material for use in new construction projects and the stage of use itself.

The different scenarios analyzed and the exposure pathways considered in the calculations are similar to those indicated for the tip in Table I. For recycling and reuse, the scenarios included in Table V have been used. Individual and collective doses have been calculated for both workers and the public, the values derived for exemption having been determined on the basis of the results.

The values obtained are included in Table VI, and are the highest values meeting the defined criteria with a reference quantity 10.000 t/year.

TABLE V EXEMPTION OF RECYCLING AND REUSE OF CONCRETE

SCENARIOS	EXPOSURE PATHWAYS
1) Concrete delivery Transport, load/unload material	- External - Inhalation
2) Concrete processing . Recycle . Distribution	- External, inhalation and ingestion - External (storage) - External and inhalation (load trucks) - External (transport)
3) Product use . Construction . Parking lot worker . New building user	- External and inhalation - External - External
4) Building reuse . Building renovation . Building occupancy	- External, inhalation and ingestion - External, inhalation and ingestion

5. CONCLUSIONS

- There is an internationally felt need for the establishment of national exemption policies, based on common, well defined principles and serving to identify practical criteria in the form of derived values.

- Concrete, active programmes exist in the framework of both the IAEA and the EC, aimed at providing a response to the aforementioned need. The results obtained are essentially reasonably consistent and meet the objective of facilitating direct use. Nevertheless, the two most relevant developments currently under way contain certain statements which are debatable as regards conceptual coherence and which should be reconsidered.

- Concrete approaches currently exist in Spain and have been submitted to the Administration for the exemption of certain streams, these being based on background work carried out by ENRESA over the last few years.

TABLE VI DERIVED EXEMPTION LEVELS FOR CONTAMINATED CONCRETE

RADIONUCLIDE	ACTIVITY CONCENTRATION (Bq/g)
Cl-36	3
Ca-41	3E3
Mn-54	3
Fe-55	3E5
Co-60	3
Ni-63	3E4
Zn-65	3
Sr-90	3E2
Nb-94	3
Tc-99	3E1
Cs-137	3
Eu-152	3
U-238	3E2
Pu-239	3
Pu-241	3E2
Am-241	3

- Development of clearance criteria applicable to certain practices requires first in-depth knowledge of these practices, based on control at the point of origin of the materials to be disposed of and adequate documentation, and second high degrees of technical and administrative rigour throughout the process.

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PRACTICAL APPLICATION OF EXEMPTION AND CLEARANCE CONCEPTS

(Session IV)

Chairman

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DECONTAMINATION OF METAL COMPONENTS TO CLEARANCE LEVELS BY MEANS OF ABRASIVE BLASTING

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Abstract

At the decommissioning of nuclear installations, large quantities of metals are produced together with other important amounts of low radioactive wastes. These materials are mostly surface contaminated. Having been used or even only having stayed for some time in a controlled area, marks them as 'suspected material'.

Costs for treatment and disposal as radioactive wastes of these materials are very high. These very high costs make people look for alternatives in dealing with this kind of wastes. Recycling through melting or through thorough decontamination of metals to generally accepted unconditional clearance levels could be such an alternative.

To a larger extent recycling of materials can be considered as a first order ecological action to limit the amounts of wastes to be disposed of, and at the same time reduce the technical and economic problems involved with the management of radioactive wastes and to make economic use of primary material and in this way conserve our natural resources of basic material for future generations.

Other evaluations as the environmental impact of recycling compared to non recycling (mining or production of new material) and waste treatment, with the associated risks involved, can also be considered, as well as social and political impacts of recycling.

The paper deals with practical examples where in Belgium recycling of material will be carried out using decontamination techniques, and based on the results of different technical and economic evalutions.

1. INTRODUCTION

At the decommissioning of nuclear installations, large quantities of metals are produced together with other important amounts of low radioactive wastes. These materials are mostly surface contaminated. A small part of them can be activated. In any case, having been used or even only having stayed for some time in a controlled area, marks them as 'suspected material'. They can only be withdrawn from the radioactive waste circuit by thorough and intensive proving that a possible residual contamination in the remaining structures is well below still not yet generally accepted unconditional clearance levels.

In Belgium, N.I.R.A.S., the National Institute for Radioactive Wastes and Nuclear Fuel, has estimated waste quantities resulting from the Belgian nuclear programme up to the year 2050 (Ref [1]). In the category A, low and middle active wastes (containing isotopes with a half life time of less than 30 years or a non-significant amount of isotopes with a half life time of more than 30 years), 70,000 m^3 of on the whole 120,000 m^3 is believed to result from decommissioning activities. At the actual rates for waste treatment and shallow land burial in Belgium, the costs for treatment and disposal as radioactive wastes of these amounts of material will be more than 30 milliard Belgian francs (1 billion US dollar).

This large amount of money makes people look for alternatives in dealing with this kind of wastes. Thorough decontamination of metals to generally accepted unconditional clearance levels could be such an alternative. In this way these materials could be either :

- reused without any further transformation as for tanks, pumps, motors, valves, ..., that could be applied for direct reuse in industry;
- reused after further transformation as for metal, after melting and bringing the ingots back in the industrial circuit;
- disposed of as conventional waste on an industrial dumping ground, mainly for materials with limited economical value, or for materials actually requiring very expensive recuperation techniques.

In a broader context recycling of materials can be considered as a first order ecological action. Indeed, from the ecological point of view actual tendencies with no doubt point to a maximum reuse of materials :

- to limit the amounts of wastes to be disposed of, and at the same time to reduce the technical and economic problems involved with the management of radioactive wastes;
- to make economic use of primary material and in this way conserve our natural resources of basic material for future generations.

Other evaluations as the environmental impact of recycling compared to non recycling (mining or production of new material) and waste treatment, with the associated risks involved, can also be considered, as well as social and political impacts of recycling.

It has to be pointed out, however, that general applicability of possible recycling techniques is largely influenced by the existence or the lack of internationally accepted unconditional clearance levels for different kind of accepted practices.

In this paper practical examples are described where in Belgium recycling of material has or will be carried out using decontamination techniques and based on the results of different technical and economic evaluations.

2. EXAMPLES OF RECYCLING IN BELGIUM

2.1. Decontamination of 18 tons of carbon steel from a drip tray

After successful completion of the decommissioning of two smaller storage buildings for end products of reprocessing as a pilot project, in 1991 Belgoprocess started the dismantling of the Eurochemic reprocessing facilities on an industrial scale. To keep control on the costs for waste treatment and intermediate and final disposal, the pilot project has shown that it is worth making great effort in the decontamination of radioactive materials to unconditional clearance levels. Based on this conclusion a test programme was set up to demonstrate the feasibility of different decontamination techniques. As an example the decontamination of about 18 ton of contaminated rusty carbon steel by means of abrasive blasting systems was evaluated.

Different tests with chemical techniques did not give acceptable results in view of a possible release of the material. By means of a small abrasive blasting system, however, on a laboratory scale samples, such as :

- a very rusty plate with contamination levels up to 4 Bq/cm^2 in alpha and 25 Bq/cm^2 in beta,
- a very rusty plate but not contaminated,
- a plate covered with a layer of bitumen and a contamination level of 4 Bq/cm^2 in alpha and 10 Bq/cm^2 in beta,
- a plate covered with two layers of paint, but not contaminated,

could all be cleaned and/or decontaminated to background levels

Based on the results of these laboratory tests, Belgoprocess started a demonstration programme to decontaminate 18 tons of carbon steel from a drip tray by means of two semi-industrial and mobile systems for dry and wet blasting. The demonstration programme mainly was set up :

- to evaluate advantages and disadvantages of dry and wet blasting techniques, to detect specific problems and to look for solutions,
- to evaluate decontamination factors and other characteristics, and to gather information on secondary waste production,
- to test methods for secondary waste treatment and evaluate effects of wet blasting on existing ventilation and filtering systems,
- to evaluate measurements for unconditional clearance and possibilities to increase measurement speed,
- to further define the installation of a fully automised industrial blasting unit.

On the other hand in this demonstration programme, costs for decontamination by abrasive blasting systems were in detail compared to normal waste treatment and disposal costs for the same material. In this cost comparison in a conservative way total installation costs for both the dry and wet abrasive systems were included.

Each time for 9 ton of material three alternative treatment methods were considered : cutting and decontaminating to unconditional clearance levels by means of a wet and a dry mobile abrasive blasting system and waste treatment and disposal of secondary wastes or cutting, supercompaction, cementation and shallow land burial as low level waste.

More in detail the first two alternatives include :

- adapting existing building infrastructure to allow the installation of the abrasive blasting equipment,
- the total investment costs for both the dry or the wet blasting installation,
- cutting and transport of material to the decontamination facility,
- decontamination of the material in the dry or the wet abrasive blasting installation,
- measurement of the decontaminated material,
- treatment, conditioning and disposal of secondary waste as low level waste.

The third alternative considers :

- cutting and transport of material to the waste treatment facility,
- cutting, packaging in 220 litre drums, supercompaction and packaging in 400 litre drums,
- cementation and disposal of conditioned waste.

An overview of the costs (in Belgian francs 1992) related to the three alternatives is given in Tables 1 and 2 below. Both tables clearly point to the cost reductions achievable in using adapted decontamination techniques as an alternative for waste treatment, conditioning and disposal. Typical values for this case resulted in a cost of 467 Bfr/kg, respectively 716 Bfr/kg for decontamination to unconditional clearance levels by the dry, respectively the wet abrasive blasting system. These cost figures have to be considered to a comparable unit cost of 825 Bfr/kg for a conventional radioactive waste treatment scenario for the same material in this case.

Table 1 : Comparison of costs for the decontamination of 9 ton of carbon steel with a dry abrasive blasting system versus waste treatment costs (in 1,000 Bfr 1992).

Task	Decontamination by dry abrasive blasting	Supercompaction and disposal
Capital costs	1,279,-	
Installation costs	358,-	
Cutting and transport	619,-	619,-
Reducing and packaging		1,669,-
Decontamination	841,-	
Measurements	54,-	
Waste treatment	817,-	3,963,-
Intermediate disposal	35,-	175,-
Final disposal	200,-	1,000,-
TOTAL COSTS	4,203,-	7,426,-

Table 2 : Comparison of costs for the decontamination of 9 ton of carbon steel with a wet abrasive blasting system versus waste treatment costs (in 1,000 Bfr 1992).

Task	Decontamination by wet abrasive blasting	Supercompaction and disposal
Capital costs	3,795,-	
Installation costs	307,-	
Cutting and transport	619,-	619,-
Reducing and packaging		1,669,-
Decontamination	840,-	
Measurements	54,-	
Waste treatment	640,-	3,963,-
Intermediate disposal	28,-	175,-
Final disposal	160,-	1,000,-
TOTAL COSTS	6,443,-	7,426,-

Based on this cost evaluation the decision has been taken to start the demonstration programme. Both the dry and the wet systems have been delivered.

The dry system has been installed, and preliminary tests have been carried out. The whole test programme is scheduled to be finished and evaluated mid 1994. Based on the results of this demonstration programme a decision will be taken to set up an adapted industrial abrasive blasting infrastructure.

2.2. Decontamination of 309 tons of steel from the decommissioning of the main process building of the Eurochemic reprocessing plant

Following the demonstration programme on decontamination by abrasive blasting systems, Belgoprocess has evaluated the possible installation of an automised industrial abrasive blasting infrastructure, considering that large amounts of comparable metal components, decontaminated in a same installation, can only result in standard operations and cost reductions.

Estimates indicate that during the next years, and limited to the proper decommissioning programme on the Eurochemic reprocessing plant, Belgoprocess will produce more than 309 ton of low contaminated metal, including flat sheets of metal, casings of cutted tanks and neutron shields, UPN and IPN profiles, ..., which could be decontaminated to unconditional clearance levels. This is only a first evaluation. In real terms other metal structures will be removed that could also be qualified to be decontaminated by abrasive systems.

Therefore, as in the demonstration programme, costs for decontamination by abrasive blasting systems of the 309 ton of metal components from the decommissioning project were in detail compared to normal waste treatment and disposal costs for the same material. In this study the installation of a fully automised industrial system was considered which might be either a dry or a wet unit, depending on the results of the demonstration programme.

Table 3 : Comparison of costs for the decontamination of 309 ton of metal with a dry abrasive blasting system versus waste treatment costs (in 1,000 Bfr 1992).

Task	Decontamination by dry abrasive blasting	Supercompaction and disposal
Capital costs	2,867,-	
Installation costs	10,251,-	
Reducing and packaging	32,106,-	64,212,-
Decontamination	10,629,-	
Measurements	6,464,-	
Waste treatment	13,855,-	135,850,-
Intermediate disposal	616,-	5,936,-
Final disposal	3,520,-	33,920,-
TOTAL COSTS	80,308,-	239,918,-

Table 4 : Comparison of costs for the decontamination of 309 ton of metal with a wet abrasive blasting system versus waste treatment costs (in 1,000 Bfr 1992).

Task	Decontamination by wet abrasive blasting	Supercompaction and disposal
Capital costs	40,000,-	
Installation costs	10,658,-	
Reducing and packaging	32,106,-	64,212,-
Decontamination	22,169,-	
Measurements	6,464,-	
Waste treatment	10,467,-	135,850,-
Intermediate disposal	448,-	5,936,-
Final disposal	2,560,-	33,920,-
TOTAL COSTS	124,872,-	239,918,-

Also for the 309 ton of material, the three alternative treatment methods as mentioned above were considered : cutting and decontaminating to unconditional clearance levels by means of an industrial wet or dry abrasive blasting system and waste treatment and disposal of secondary wastes or cutting, supercompaction, cementation and shallow land burial as low level waste.

An overview of the costs (in Belgian francs 1992) related to the three alternatives is given in Tables 3 and 4 above.

Typical values for this case study resulted in a cost of 260 Bfr/kg, respectively 404 Bfr/kg for decontamination to unconditional clearance levels by the dry, respectively the wet abrasive blasting system. These cost figures again have to be considered to a comparable unit cost of 776 Bfr/kg for a normal radioactive waste treatment scenario for the same material.

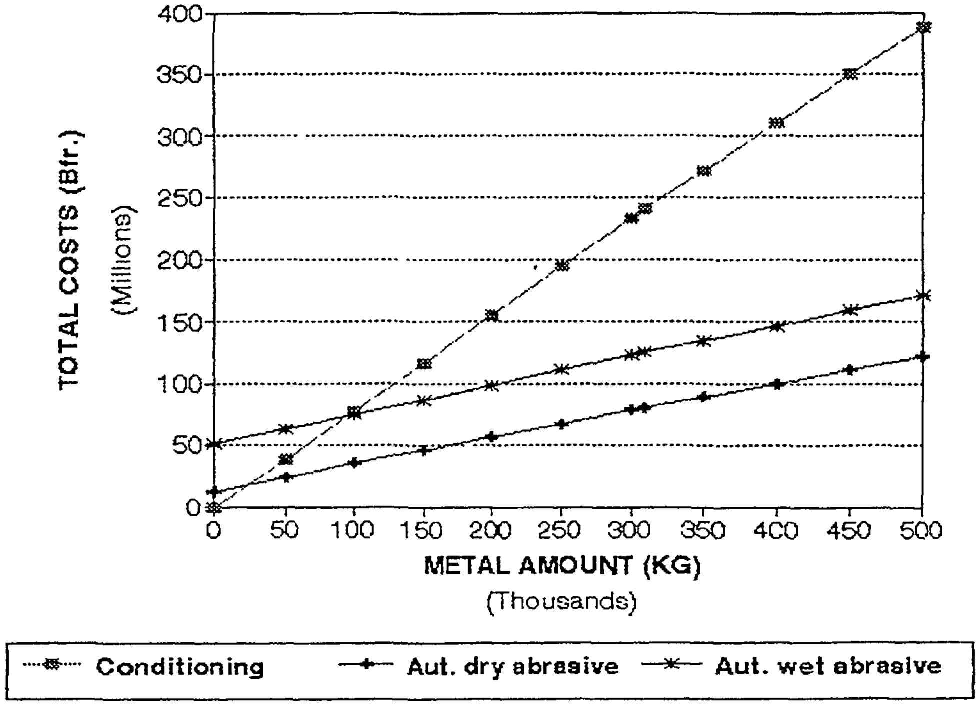

Figure 1 above gives a view of the total costs for installation, waste treatment, and intermediate and final disposal for the alternatives considered, and this depending on the amount of material available for decontamination.

3. Verification of the suitability of the abrasive installation

Release of material is based on actual procedures : all equipment, material and areas with contamination levels above background levels are considered radioactive. Materials monitored and found to be under the background levels of the used PCM counters can be disposed of without any restrictions. Surface area have to be monitored 100 %, and surfaces/areas that can not be monitored are considered radioactive.

In addition, to verify the suitability of the abrasive decontamination system more specifically, impact of abrasive material into the surface of a component, and able to introduce at the same time contamination into the surface layer, is checked by means of two independent control actions on samples taken from the components to be decontaminated. To avoid introduction of contamination at melting during plasma cutting of the material, these samples are taken using mechanical cutting methods.

First of all, together with a weight check, contamination levels on components to be decontaminated are monitored by non-destructive gamma measurements on the samples before and after decontamination operations.

Additionally, electrochemical control monitoring is carried out on some samples after decontamination. By electrolysis during a well known period and at a specific current in an HCl solution with a well defined concentration, some surface material of the sample is removed and dissolved in the electrolyte. A sample of this electrolyte is then analysed and radiologically characterised.

4. CONCLUDING OBSERVATIONS

It was surely not the aim of this paper to go very deeply into the technical details of some recycling projects. More emphasis was put on cost aspects and on the comparison of actual radioactive waste treatment and waste disposal costs versus costs for alternative techniques as shown in some projects or case studies on decontamination to unrestricted clearance levels.

The examples mentioned indicate that thorough decontamination of metals to generally accepted unconditional clearance levels could be a cost-effective alternative for actual waste treatment, waste conditioning and waste disposal of some metal components.

Moreover recycling of materials can be considered as a first order ecological action in limiting the amounts of wastes to be disposed of, and in making economic use of primary material.

Other evaluations as the environmental impact of recycling compared to non recycling (mining or production of new material) and waste treatment, with the associated risks involved, can also be considered, as well as social and political impacts of recycling.

It has to be pointed out, however, that general applicability of possible recycling techniques is largely influenced by the existence or the lack of internationally accepted unconditional clearance levels for different kind of accepted practices.

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RADIOLOGICAL CHARACTERIZATION OF METAL COMPONENTS IN VIEW OF MELTING

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Abstract

Within the framework of one of their main activities to process and store radioactive wastes produced in Belgium, Belgoprocess compared different scenarios for treatment, conditioning and disposal or recycling of 9 heat exchangers and fuel racks.

The 9 heat exchangers were mainly horizontal shell-in-tube aluminum heat exchangers with a total weight of about 42 ton and additionally about 2 ton of aluminum piping. The radioactive contaminated primary water was on the tube side and the cooling water on the shell side. So activity was limited to the inside of the tubes and the primary piping.

The 7 fuel racks were stainless steel constructions with a total weight of 32 ton. Data available on the material mentioned only limited use due to inadequate construction.

For both kinds of material two alternative treatment methods were considered : cutting, melting, storing of ingots for decay and waste treatment and disposal of secondary wastes or cutting, supercompaction, cementation and shallow land burial as low level waste.

Based on a cost evaluation the melting option has been chosen. Meanwhile transport of all the material to Sweden has started and is expected to be finished by the end of the year 1993.

The paper deals with the practical problems encountered with the characterisation of the material at the decision making process and at the preparation of the transport.

1. INTRODUCTION

Within the framework of one of their main activities to process and store radioactive wastes produced in Belgium, Belgoprocess compared different scenarios for treatment, conditioning and disposal or recycling of 9 heat exchangers and fuel racks.

The 9 heat exchangers were mainly horizontal shell-in-tube aluminum heat exchangers with a total weight of about 42 ton and additionally about 2 ton of aluminum piping. The radioactive contaminated primary water was on the tube side and the cooling water on the shell side. So activity was limited to the inside of the tubes and the primary piping.

The 7 fuel racks were stainless steel constructions with a total weight of 32 ton. Data available on the material mentioned only limited use due to inadequate construction.

For both kinds of material two alternative treatment methods were considered : cutting, melting, storing of ingots for decay and waste treatment and disposal of secondary wastes or cutting, supercompaction, cementation and shallow land burial as low level waste.

More in detail the first alternative included :
- cleaning and packaging of the material for transport to the melting facility (Studsvik in Sweden),
- transport to the melting facility,
- melting of the material at Studsvik,
- leaving the melted material as ingots in Sweden for decay and later reuse, and transporting back the secondary wastes to Belgium,
- treatment, conditioning and disposal of secondary waste as low level waste.

The second alternative considered :
- preparing materials for cutting,
- transport, cutting, packaging in 220 litre drums, supercompaction and packaging in 400 litre drums,
- cementation and disposal of conditioned waste.

Based on technical and economic evaluations the melting option has been chosen. Meanwhile transport of all the material by road and by boat to Sweden has started and is expected to be finished by the end of the year 1993.

During the decision making process and at preparation of the transport, a lot of difficulties have been encountered at the characterisation of the material. These difficulties were mainly based on a lack of information about the material, but also on problems encountered with representativity of measurements or calculations on the activity content of the material.

2. EVALUATION OF RADIOLOGICAL CHARACTERISTICS FOR BOTH WASTE COMPONENTS

The information based on which the final characteristics of the fuel racks and the heat exchangers were adopted has been obtained in different stages.

2.1. Available basic information

After taking over the activities of the former Waste Treatment Department of the Belgian Nuclear Research Centre in March 1989, as a main priority

Belgoprocess made an inventory of all wastes and other materials physically still present on site. It was also tried to identify these materials to information available in the waste bookkeeping system.

At a first evaluation neither for the fuel racks, nor for the heat exchangers any link could be demonstrated with any descriptive waste form found in the bookkeeping system. As a result, for both type of components waste forms were created and introduced in the actual bookkeeping system.

Without any reference or any report on measurements carried out, at the same time a total activity content of $3.0.10^9$ Bq was assigned to the fuel racks, corresponding to a specific activity of about 94 Bq/gram. No specific radiologic data were at that moment assigned to the heat exchangers.

2.2. Additional basic characterisation work

To improve the organisation of the waste storage facilities both fuel racks and heat exchangers were later on transported to adapted storage buildings. In view of this transport, additional investigation work was carried out in the record offices of the Waste Treatment Department. Forms were found showing some limited information about the kind of material and the volume and weight of the components mentioned, as they were transported in 1985 from the initial waste production centres to the site of the Waste Treatment Department. Only poor and incomplete data were given on the radiological characteristics of the material.

Additional verbal information stated that especially the fuel racks had been put for some time in a fuel pool, but never had been used for actually storing fuel elements due to their inadequate construction.

Indications on the forms did not give reasons to suppose that the racks ever had been activated. Indeed, it was mentioned that the maximum surface contamination based on analyses of wipe tests was 60 Bq/cm^2. The free surface of each rack being 158 m^2, this resulted in a maximum contamination level of 94.8 MBq or $2.6.10^{-3}$ Ci.

So the additional verbal information that the racks had never been used, made everyone understand that the activity content indicated on the form, $1.5.10^{-3}$ Ci with isotopes Co^{60} and Cs^{137}, was only due to surface contamination.

Taking into account the kind of isotopes and their decay and considering the storage conditions for the material during many years (in open air, fully exposed to local atmospheric conditions), it was decided to take new wipe tests from the different components in view of what was thought of a more reliable characterisation of the material for melting. These wipe tests were taken and analysed before transporting both sets of components to the more adapted intermediate storage facilities on site. Again it had to be pointed out that based on the available information, there was in no way any assumption of the presence of activation in the material.

The analyses of the wipe tests from the fuel racks showed (removable) surface contamination levels from 0.81 to 6.27 Bq per 900 cm^2 in alpha and 130 to 743 Bq/cm^2 per 900 cm^2 in beta-gamma.

When taking wipe tests from the heat exchangers, and also during their transport, some hot spots with higher dose rates were detected, which however were moved depending on the position of the component. At opening one of the heat exchangers, an accumulation of non fixed dust particles was found in the inlet chamber. This dust could easily be removed by means of a vacuum cleaner, and as a result no significant differences were measured at the surfaces of the heat exchanger any longer. Based on this experience it was however decided to open and clean all the other heat exchangers as well.

Radiological analyses of the wipe tests from the heat exchangers showed (removable) surface contamination levels from 1.93 to 4.43 Bq/cm^2 in alpha, 3.15 to 12.59 Bq/cm^2 in beta and 2.93 to 3.70 Bq/cm^2 in gamma, resulting in a maximum activity content of about 12 Bq/gram per heat exchanger.

Later also some 2 ton of aluminum piping has been characterised based on direct measurements of samples with a portable contamination monitor. Resulting contamination levels proved to be between 32.2 and 53.7 Bq/cm^2 in alpha and 43 to 118.2 Bq/cm^2 in beta-gamma, resulting in a total activity content for the pipes between 49 and 98 Bq/gram. These contamination levels could be reduced to some 20 Bq/cm^2 in alpha and 10 Bq/cm^2 in beta-gamma by high pressure water spraying.

In addition dose level measurements on contact of the fuel racks as well as on the heat exchangers showed no values above the background levels in the storage facility (5 to 10 μSv/hour). So they were not taken into account for characterisation of the components.

2.3. Characterisation problems at transport preparation of the fuel racks

After regrouping of the material in a more suited storage facility, the transport of the material to the melting facility in Sweden was prepared. A first transport could be carried out without any major problem, neither on handling nor on dose limitations. Doses on contact of the transport containers were lower than 5 μSv/hour, i.e. at background levels in the loading area. Preparing the second transport however, dose levels of 5 μSv/hour at contact of the transport container were detected, with some spots up to 8 μSv/hour.

Based on these differing observations, new dose rate measurements in different cells of the fuel rack with the highest detected dose levels were made, showing some spots up to 90 μSv/hour in some well defined cells. With these dose rate measurements a computer simulation was carried out, calculating the most probable value for the total specific activity of this fuel rack to be 4 Bq/gram. In these approximating calculations it was adopted that the activity distribution over the total material of the fuel rack was homogeneous. Taking into account the concentration of radioisotopes in nearby cells, it was also calculated that the isotopes in one cell only contributed for 40 % of the total dose level in that specific cell.

In addition, also samples have been taken from the material of the same fuel rack. Direct measurements with a portable contamination monitor on these samples gave values between 3.5 and 7 Bq/gram total activity. Analyses carried out showed specific activity values of the samples up to 1.8 Bq/gram.

Attempts have been made to decontaminate the samples by means of agressive methods as abrasive blasting and electrochemical techniques. It was shown that only part of the activity could be removed. This was an indication that the total activity was partly due to removable contamination, partly to non removable contamination, but also due to activation of the material. At the same time this was a possible explanation for the low initial results of contamination measurements by wipe tests carried out as indicated in paragraph 2.2., where surely only the removable contamination was detected.

The presence of Niobium as an activation product also gave a view of the characteristics of the fuel racks different from the information available untill that moment. It had indeed to be accepted that the fuel racks really had been used under nuclear circumstances and that certainly, at least temporarily, fuel elements had been stored in these racks.

Based on the above considerations, it seemed reasonable to accept that the total activity content of the fuel rack with the highest dose levels would be about 4 Bq/gram. Evaluating the dose levels detected for the other fuel racks, it was estimated that this value could be considered as an upper limit.

2.4. Final evaluation of the radiological characteristics of the fuel racks at melting

As indicated above, different attempts have been carried out to increase the reliability of the radiological characterisation of the material. Based on the results given, the fuel racks have been transported to Sweden and were melted in the melting facility at Studsvik. The ingots, the slag and the dust resulting from the specific melting process have been thoroughly analysed by gammaspectrometry. Based on the results of these measurements, apparently in all the evaluations and estimates carried out as indicated before, the activity of the material of the fuel racks still had been underestimated. Indeed, the total activity measured at melting at last proved to be equivalent to about 10 Bq/gram Co^{60}.

Based on the known details about the history of the fuel racks and on the number of differing activity measurements carried out to qualify the initial and revised total activity estimates for the fuel racks, and considering the results of the analyses after melting, there were no arguments to assume for each of the fuel racks other activity contents than those values recorded when analysing the samples of the melting process.

2.5. Evaluation of the radiological characteristics of the heat exchangers

Keeping in mind the experiences in characterising the material of the fuel racks, samples have been taken from the piping and tubing of the heat exchangers and were sent to the melting facility for prior characterisation work. The complete results of the melting tests are not available yet. Preliminary results give indications that also for the heat exchangers differing values will be obtained as was the case with the fuel racks.

3. CONCLUDING OBSERVATIONS

During the decision making process and at preparation of the transport in view of melting of material from fuel racks and heat exchangers, a lot of difficulties have been encountered to determine the radiologic characteristics of the material. The difficulties were mainly due to a lack of information about the material, but also to problems encountered with the representativity of measurements or calculations on activity content of the material.

The whole experience has proved that though radiological characterisation of material could be considered as a qualified and known, obvious process, much care has to be taken, especially when handling waste material with an unknown history or which is only poorly documented.

This consideration easily can be extended to decommissioning tasks or remedial action programmes where the actual work is carried out many years after shutdown of the installation, when experienced personnel that operated the plant have disappeared and only limited operational data are left.

Key elements in this kind of problems are the quality and the amount of the available information and the fact how at this very low levels activity estimates are made including considerations about removable or non removable contamination, or activation.

Much concern has to be attended to the interpretation of characterisation measurements, and to what extent :
- present contamination can be considered as removable contamination,
- the efficiency and representativity of wipe tests can be considered as sufficient,
- the representativity of direct measurements with a portable contamination monitor in view of estimates for non-removable contamination is acceptable,

- the representativity of direct measurements with a portable contamination monitor in view of estimates for activity measurements of activated material is acceptable,
- in alpha characterisations, absorption by the material itself is taken into account in a sufficient way.

Based on the results of the examples mentioned above, apparently there are arguments to assume that the analyses thoroughly carried out on the ingots, the slag and the dust resulting from the melting process have to be considered as basic elements from perhaps one of the most valuable methods in characterisation work, certainly in view of releasing material at clearance levels.

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RECYCLING OF RADIOACTIVE CONTAMINATED METALS FROM NUCLEAR INSTALLATIONS

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Abstract

This paper gives a historical survey of the SIEMPELKAMP process, which produces components for nuclear industry from radioactively contaminated scrap from nuclear power plants. It describes the bases for this safe method of material re-use and evaluates the experiences gained in the Krefeld plant within the past twelve years. In total, approx. 7,000 tons have been recycled.
The licensing procedure in Germany is described, and the activity limits for treating the metal are given.
Great importance was attached to the information of the public to obtain acceptance by the workers, the population, and the political parties.

1. The Development Task for SIEMPELKAMP Foundry

During the past twelve years, SIEMPELKAMP foundry developed the melting process for recycling radioactive contaminated scrap from nuclear installations. Regarding the problems with the licensing authorities, the political parties, and the general public, the basic decision was taken to recycle the material for reuse in nuclear industry by producing castings needed in this field.

The first test melts were made in 1984 by melting the superheaters from Obrigheim nuclear power plant. It took five years until the development was completed and the process was finally accepted as optimal way for radioactive metal scrap treatment.

One of the major concerns was to limit radiation exposure of the personnel in the environment of the furnace during their working period to an absolute minimum.

The development philosophy was to apply remote control in this process to the maximum possible extent. When the new plant was taken into operation in 1989, all processes were operated by remote control, except for the slag removal.

In 1992, the development of a robotic slag removal system began, together with Ansaldo/Italy. Presently, this robotic system is being put into operation and, immediately on successful completion of the cold tests, the melting process shall start - from charging of the furnace to the casting of the liquid steel into the ingot - completely remotely controlled.

Thus, optimum protection of the staff in the furnace surrounding is given.

Continuous supervision of the personnel by dosimeters and bodycounter checks show that there is no measurable dose above natural background.

From the beginning of the development until today, approx. 4,000 tons have been successfully melted in the CARLA plant.

2. Licensing

German regulations and laws demand licensing on two subjects:

- licensing for protection of environment against emissions
- radiation protection ordinance.

The first law regulates the plant being run as a commercial melting shop; especially the quantity of dust in the off-gas of the filter system is limited to be below 20 mg/m^3. Furthermore, noise level on the border to the neighbourhood is limited. In this special case, radioactivity is not considered.

For the first plant, the licence was given for a specific activity of 74 Bq/g. In those days, this was the limit for free handling of radioactive material. The reason why a license has to be granted was that the activity being delivered to the plant was determined by a rough measurement; the exact content of radioactivity could only be determined after the melting process and the homogenisation of the activity in the melt. On the other hand, it was known by pre-tests that some radioactive nuclides were set free during the melting process.

Due to the small amount of filter dust, the specific activity of this dust was comparatively high so that the limit of 74 Bq/g was exceeded very easily.

Several licenses have been granted regarding storage conditions, melting procedures and quantities, melting of different materials etc. For the newly built CARLA Plant for beta-gamma materials, the specific activity limit of the material to be melted was increased to 200 Bq/g for the incoming material.

Considerations showed that the specific activity of the filter dust may rise, under extreme conditions, to up to 25,000 Bq/g.

3. Public Acceptance

From the early beginning it was obvious that this project could only be successful if public acceptance was obtained. To this end, in a first step, the local press was invited to report about the project to keep the population informed.

It was clear that this step was only the beginning of a walk through the media. Until today, a lot of newspapers reported about this melting process; five times different German, the French and the Japanese TV stations came to look at and report about the facility, as well as several local institutions. Special attention had to be paid to the workers in the foundry and its neighbourhood.

Consequently, all visitors had the chance to see everything that was done in the plant and no secrets were made about the procedure, even if there was a high risk of misunderstandings.

4. Results of Beta-Gamma Melting

During the development of the melting process, the definition "mass specific activity" grew to major importance. While in the early days, for the measurement of the activity, only the gamma nuclides have been considered, a definition of the radiation protection committee showed that "Bq/g" makes no difference between beta and gamma radiators. Alpha-contamination did not have to be considered as the amount of alpha-contamination of scrap coming from nuclear power plants always was three to four orders of magnitude below the relevant radiators.

In the first test campaign, the dose rate of the material being delivered to the SIEMPELKAMP plant was measured. Low energy beta radiators do not have any effect on the dose rate measurement. Thus, the specific activity was underestimated. Melting this material, a second problem arose as there is an activity level of material being delivered to be melted, and there is a second activity distribution in the melt, the slag, and the filter dust. As known by pretests, only some nuclides stay in the melt while others will be found in the slag and the dust.

Discussions with the authorities led to the decision that a correction of the nuclide specific activity can be made by taking samples from the material which are suitable to detect all beta and gamma radiating nuclides. This vector is calibrated to the activity of $Cobalt_{60}$ and $Caesium_{137}$; this scale-up factor being calculated from the ratio of total activity to the activity of Co_{60} and Cs_{137} is considered to come close to the real amount of "Bq/g".

After melting, it is difficult to calculate residual activity with this procedure because - during the melting process itself - a split-up of the activity happens so that only the experience of several melting tests can help to find nuclides in the melt, the slag, and the filter dust.

During the entire development of the melting process, it was a question of economy to which extent secondary waste arises from this melting process. Not only the quantity of slag and filter dust was to be considered; but also the contaminated cladding of the furnace which was to be added.

Melting of zinc-plated steel sheets brought an other new experience. In a first step, all zinc leaves the melt and leads to a large amount of secondary waste; dust quantities in the range of 5 % of the initial quantity of material were usual.

The second effect - caused by the high velocity of the off-gas - was the burning of the adapter on top of the furnace; zinc in this high volume produces high temperatures which can destroy all steel structures of the off-gas ducts. Therefore, zinc-plated steel had to be melted at a very reduced melting rate which influenced the economy of the process very strongly.

5. Components to be Produced from this Material

There is a broad range of applications for the re-use of the material. The least strict requirements are those for shielding beams, plates and stones. These products allow for a comparatively high contamination as they do not have to fulfill high-quality demands. Chrome and molybdenum contents may be as high as 5 %.

The second type of applications are shielding doors and partitions. These components have to meet certain static requirements and pass their own weight into the building structure, to which in certain cases loads from earthquakes may even have to be added. Furthermore, other external forces from accidents like gas cloud explosion, air crash etc. are to be considered.

The third grade of demand are IP-II and type-A-packages resp. according to the regulations of the International Atomic Energy Agency, Vienna. Containers in this category have to endure a drop from a height of 1 m and they have to fulfill the regulations for the final depository.

Category four are type-B-containers (IAEA). The activity content in these containers is significantly higher. Leakage from these containers must be avoided; helium tightness must be guaranteed.

Fifth category, with the highest demands, are the "CASTOR"-casks which are produced to take up spent fuels from nuclear power plants for the purpose of transport and intermediate storage. The weight of these containers ranges from 50 to 150 t per piece. The complete spectrum of nuclear safety demands has to be fulfilled.

6. Conclusion

From the beginning up to 1993, SIEMPELKAMP foundry has melted more than 7,000 t of radioactive metallic waste. During the development phase, not only steel had to be melted, it also included melting of copper, brass, aluminium, and lead.

Melting of radioactive contaminated metal scrap to be recycled for nuclear applications is a common job for SIEMPELKAMP foundry, Krefeld. This method is established not only in Germany; developments in other countries like France, the United States, Sweden and Japan came to the same results.

Scale-Up Factor

$$SC = \frac{A_{total}}{A_{Co60} + A_{Cs137}}$$

After measurement of the total activity in representative samples, the specific activity is

determined by measuring the activity of Co60 and Cs137.

A_{total}	$SC \times (A_{Co60} + A_{Cs137})$
A_{total}	total specific activity
A_{Co60}, A_{Cs137}	activity of Co60/Cs137 as measured

Aerosol Release in the Melting Area

Work phase	Nuclide	max. Values / Bq/m³
sampling / casting	Cs-137	0.14
during various		
melting procedures	Cs-137	0.17
furnace deslagging	Cs-137	0.42

Aerosol Release in Storage Area / Mold Storage

ambient air	-	below detection limit
samples taken at various places		

Contamination in Various Zones of CARLA Plant
(wipe test, high-purity Germanium detector; 4 to 8/1990)

Area	Min.-value Bq/cm²	Max.-value Bq/cm²	Average Bq/cm²	Distribution in the Area
drum storage	< 0.5	< 0.5	< 0.5	homogeneous
filter area	< 0.5	< 0.5	< 0.5	partly slight concentrations in area of dust drums
housing	< 0.5	< 0.5	< 0.5	partly slight concentrations in furnace area
mold storage	< detection limit			homogeneous

Contamination Measurements of Plant Components
("wipe test", evaluation in a high-purity Germanium detector)
(11/1993)

Sample taken from	max. values found Bq/cm²
filter plant piping	
- inside	7.9
- outside	< 0.5
charging device	
- inside	1.8
- outside	< 0.5
filter box behind HEPA filters	< below detection limit

Measurement of Activity Concentration in the Off-gas

(11/1993)

Nuclide	Activity Concentration (Bq/m^3)	Measuring fault (%)
total Alpha	5.2 E-6	15
total Beta	1.4 E-4	3
Co-60	6.7 E-5	16
Cs-137	2.7 E-5	30
Am-241	< 2.1 E-5	-
Pu-238	< 0.01 mBq/m^3	
Pu-239/240	< 0.01 mBq/m^3	

Dose Rate Measurements at Various Points of Plant (Measuring Period April to August 1990)

11/1993

Area	Min.-value µSv/h	Max.-value µSv/h	Average µSv/h
drum storage	0.35	3.70	1.20
filter room	0.18	1.50	0.61
housing	0.13	0.80	0.36
furnace control	0.08	0.20	0.13
mold storage	0.80	2.40	1.55
outside:			
sliding doors	0.18	0.80	0.38
doors	0.08	0.25	0.14
windows	0.06	0.20	0.12

Additional Demands for Melting α-Contaminated Scrap	
Fissionable nuclides	U_{233}, U_{235}, Pu_{239} (Pu_{241})
Radiation protection law	
Limit for free handling	100 Bq/g
Limit for fissionable product	1 g / 100 kg
Limit for waste incl. fissionable nuclides	3 g / 100 kg
Maximum allowable specific activities for U_{235}	
Limit	1 g U_{235} per 100 kg scrap
Conversion factor	1 g U_{235} equals 84,500 Bq
Limit	84,500 Bq / 100 kg
Limit (specific)	0.845 Bq U_{235}/g
a-melting campaign	
Nuclides	U_{238}, U_{235}
Enrichment	4.3 %
Total activity of the scrap	
U_{233}	0.15 Bq/g
U_{235}	0.28 Bq/g
U_{234}	8.25 Bq/g
U_{238}	1.00 Bq/g

Activity in primary material is found in	ingot	slag	dust	exhaust air
Alpha emitters:				
- Uran 235, 238	1	98	1	
- Plutonium 241	1	98	1	
- Americium 244	1	98	1	
Beta emitters:				
- Tritium	-	-	-	x
- Iron	100	<1		
- Nickel 63	90	10	-	-
- Strontium 90	3	95	2	
Gamma emitters:				
- Cobalt 60	90	10	<1	
- Cesium 134, 137	<1	45	55	
- Silver 110M	95	5	<1	
- Europium 154	5	95	<1	
- Cer 144	50	50	<1	
- Mangan 54	95	5	<1	
- Zinc 65	<1	10	90	

DECOMMISSIONING OF FINAL PRODUCT STORAGE BUILDINGS AT THE FORMER EUROCHEMIC REPROCESSING PLANT

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Abstract

The paper discusses the final results of a pilot decommissioning project, carried out at the former Eurochemic reprocessing plant in Dessel, Belgium. The pilot project consisted of the dismantling of two, rather small, storage buildings for final products from the reprocessing process.

The aims of the project were : the verification of the assumptions made in a previous paper study on decommissioning, the demonstration and development of dismantling techniques and the training of personnel.

Both buildings have been emptied and decontaminated to background levels. Monitoring for de-restriction has been carried out by the Belgoprocess Health Physics Department and has been confirmed by an independent control organisation for radiation protection. Consequently, the buildings have been withdrawn from the controlled area and were demolished. Concrete debris from demolition has been removed to an industrial dumping ground for inert wastes and green field conditions restored.

The report deals with techniques used in the decommissioning operations, with radioactive decommissioning waste and secondary waste generation, required manpower resources and total costs of the dismantling project.

1. INTRODUCTION

The Eurochemic reprocessing facility, at the Mol-Dessel site in Belgium, was constructed in the early sixties (1960-1966). A consortium of 13 OECD countries operated this demonstration plant from 1966 to 1974 and reprocessed 180 ton of low-enriched and 30.5 ton of high-enriched uranium fuels.

After shutdown, the plant was decontaminated during the years 1975 to 1979 to keep it in a safe standby condition at a reasonable cost. Radiation levels in the cells were decreased to an average working level of 0.2 mSv/h.

Belgoprocess was established in 1984 to take over the activities of the former Eurochemic reprocessing company.

In 1986 it was decided not to resume reprocessing activities in Belgium and to dismantle facilities built for that purpose.

At the end of 1986, Belgoprocess became a subsidiary to NIRAS/ONDRAF. The main activities of Belgoprocess then changed to processing and storage of radioactive wastes produced in Belgium.

In 1991, Belgoprocess started the dismantling of the reprocessing facilities on an industrial scale. It was estimated that the programme will require 835 man-years and cost 5,750 MBEF (1987 value). In both costs and manpower estimates, the benefit of the decontamination works, carried out by Eurochemic between 1975 and 1979, has been taken into account.

Decontamination and demolition of two storage buildings for uranyl nitrate, plutonium dioxide and spent solvents were carried out as a pilot project to demonstrate the feasibility of decommissioning up to restoring green field conditions, to check techniques and costs and to train personnel.

All process equipment has been removed and floors, walls and ceilings have been decontaminated to background levels.

Monitoring for de-restriction has been carried out twice by the Belgoprocess Health Physics Department and has been confirmed by an independent control organisation for radiation protection. Consequently, both buildings have been withdrawn from the controlled area and were demolished. Concrete debris from demolition has been removed to an industrial dumping ground for inert wastes and green field conditions restored.

All equipment, material and areas with contamination levels above background levels were considered radioactive. Surface area had to be monitored 100 %. Surfaces/areas that could not be monitored were considered radioactive. Materials monitored and found to be under the background levels could be disposed of without any restrictions.

The total amount of wastes were within $\pm$ 20 % of earlier estimates. Some 60 % of the metallic and concrete waste has been disposed of as non-radioactive. In the case of the metallic waste, chemical decontamination methods contributed significantly to the releasable fraction.

2. DESCRIPTION OF THE STORAGE BUILDINGS 6A/6B

Both buildings 6A and 6B were relatively small buildings, 20 x 15 x 6 m and 20 x 18 x 8 m. During the reprocessing activities, they were used as storage buildings for uranyl nitrate, plutonium dioxide and spent solvents.

The total volume of both buildings was 3,290 m^3, with 4,860 m^2 of concrete surfaces. They contained 946 m^3 of concrete, 23,485 kg of process components and 15,825 kg of other material.

The average contamination levels were below 5 Bq/cm^2 with spots up to 100 Bq/cm^2 for both alpha and beta emitters. Dose rates varied between 300 and 500 $\mu Sv/h$.

3. GENERAL DISMANTLING ACTIVITIES

Within the variety of dismantling activities, five phases can be distinguished :

- the preparation of the actual dismantling works,
- the dismantling of the installations inside and outside of the buildings,
- the decontamination of the buildings,
- the contamination measurements in the decontaminated buildings,
- the withdrawal of the buildings from the controlled area, their demolition, removal of the demolition waste to an industrial dumping ground for inert wastes and restoration of the green field conditions.

Before the actual dismantling works were started, first existing utility supplies were replaced by mobile units and access locks for personnel and materials were constructed.

Access openings to cells were made or enlarged using a diamond cable saw. This technique allows high cutting capacities quite independent of wall thicknesses and the removal of entire concrete blocks to be decontaminated. It also gives resulting surfaces which are very neat as compared to those left by pneumatic hammering.

4. REMOVAL AND DECONTAMINATION OF METALLIC COMPONENTS

Actual dismantling of the installations started with the removal of piping, followed by the evacuation of the larger components. To prevent contamination spreading, piping was mainly removed with pipe cutters, saws and hydraulic shears. Pipes with larger diameters were cut with grinding disks. Some storage tanks were cut up on the spot using plasma torches. However, most of the large vessels were removed as a whole and chemically decontaminated, some for reuse, others for free release. Only metallic components, which could be measured completely on residual activity were processed for free release.

Decontamination was performed chemically, both by rinsing with nitric acid and sodium hydroxide solutions and by treatment with HF-containing pastes. Decontamination proved to be specially feasible for large vessels, but the feasibility also largely depended on the history of the vessel. No residual contamination above the detection limits of 0.04 Bq/cm^2 for alpha and 0.4 Bq/cm^2 for beta emitters was found.

5. DECONTAMINATION OF CONCRETE STRUCTURES

After the removal of all the installations, the emptied concrete structure was decontaminated, the activity levels required to be achieved being 0.04 Bq/cm^2 for alpha and 0.4 Bq/cm^2 for beta-gamma emitters.

Concrete decontamination was carried out by pneumatic hammering or by scabbling small layers of contaminated surfaces from floors, walls and ceilings, limiting the amount of waste production. Different types of hand- and floorscabblers were used, while also an automated scabbling machine was developed to increase scabbling speeds. During operation itself, waste production (dust and concrete particles) could easily and efficiently be collected in 220 litre drums by use of adapted systems with vacuum pump and self cleaning filter.

6. REMOVAL OF DECONTAMINATED CONCRETE WASTE TO AN INDUSTRIAL DUMPING GROUND

After decontamination of the concrete structures, both buildings have been entirely monitored twice by the inhouse Health Physics Department. The results, showing no residual contamination above detection limits, were confirmed by an at random control of an independent control organisation for radiation protection.

Additionally, core samples have been taken on the previously most contaminated spots. The specific activities of these samples proved to be well below 1 Bq/g. Measurements and analyses only confirmed the presence of natural radioisotopes.

Consequently, the buildings have been withdrawn from the controlled area.

The final steps in the pilot decommissioning project were the demolition of the two buildings, the removal of the demolition waste to an industrial dumping ground for inert wastes, and the restoration of green field.

To enable this last step in the objectives of the pilot project, it was necessary to supply sufficient evidence to justify such action to the public opinion. In doing so, the actual absence of any Belgian regulation for unlimited reuse or uncontrolled dumping of suspected and/or decontaminated materials, other than the qualitative definition of radioactive waste, had to be taken into account.

The evidences to be supplied had to be evaluated in the context of a number of exceptive regulations, which were applied in other decommissioning projects, and in the context of recommendations for recycling of materials resulting from the decommissioning of nuclear installations (Ref [1]), by radiation protection experts of the European Community :

- removable surface contamination in alpha <= 0.04 Bq/cm^2,
- removable surface contamination in beta-gamma <=0.4 Bq/cm^2,
- total specific beta-gamma activity <= 1 Bq/g, mean value over an arbitrary mass of 1000 kg with an individual maximum of 10 Bq/g.

The results of the multiple 100 % surface measurements in both buildings and the additional controls on selective core samples (gammaspectrometry and total alpha and beta measurements) showed that the requirements of the first two criteria were met, and that the third criterion, limited to the core samples taken, was also complied with.

The only thing that remained to be demonstrated was, that also a possible alpha contamination in the remaining structure was characterised by sufficiently low specific activities.

6.1. Evaluation of gamma measurements on the core samples

Based on experiences during operation of the plant (localised contamination) and on located contamination during decommissioning operations, it was decided not to take concrete core samples at random, but at the previously most contaminated spots. This should increase the probability on detection of possible remaining contamination.

By doing so, it was accepted in a conservative way that the obtained core samples were representative for all concrete volumes, as if all of them had originally known the same history, and as if all of them were also brought to their actual characteristics in a similar manner.

This means that it was accepted that every part of both buildings was originally contaminated to the same degree as those parts where core samples were taken, and where before decontamination the highest degree of contamination was found, which, in reality was certainly not so.

This hypothesis, however, gave the possibility of extrapolating in a conservative way, the results of the analyses on the core samples, expressed in the mean value (X) for the measurements and the standard deviation (S), to the whole of the building material. And for this hypothesis the core sampling carried out could be considered as random.

The total analysis of the core samples via gammaspectrometry, by means of a 305 x 102 mm NaI(Tl)-detector in a screened bunker with low background radiation, revealed no indication of artificial contamination via low level activity measurements. The obtained spectrum gives a qualitative view with high sensitivity. In the core samples only natural nuclides were found.

By means of gammaspectrometry (Ge-detector) of the individual core samples a quantitative measurement of the gamma emitters was carried out and compared to the results of similar measurements on a reference, non contaminated, core sample. This method has but little sensitivity. In fact a surface measurement is carried out which can only detect radioisotopes in the specimen at low depths.

Detected radionuclides were :

- K^{40}, Ra^{226} and daughters as natural radionuclides, which normally can be found in elements of concrete and other construction material. The mean

values of the detected activities for both nuclides were comparable to the values found in a non contaminated reference core sample (X_K = 0.4 mBq/g, S_K = 0.3 mBq/g and X_{Ra} = 0.27 Bq/g, S_{Ra} = 0.34 Bq/g respectively).
Operational history of the installations, and the fact that in the spectrum no peaks for U^{238} or U^{235} were found, do not give reasons to suppose that the installations have ever artificially been contaminated with K^{40} or Ra^{226}. Moreover, this argument would only be important if the detected values for K^{40} and Ra^{226} were inexplicably high.

- Cs^{137} as nuclide due to artificial radioactivity.
  Compared to the detected activity in the non contaminated reference core sample, the resulting total contamination of 2 mBq/g in the core samples, for a homogeneous distribution of the specific activity, was not higher than 1 Bq/g, mean value over an arbitrary mass of 1000 kg, taking into account the gamma penetration in concrete.
  By means of the adopted gammaspectrometry, no Am241 was found. Based on the operational history of the installations, this was an indication for the absence of alpha contamination by plutonium.

6.2. Evaluation of alpha and beta measurements on the core samples

If there was still some remaining contamination in the decontaminated floors or in other surfaces of both buildings, then, if the remaining material of the construction was still the original one, parts close to the surface would normally show the highest contamination concentration. Therefore some cross-sections of the upper part of the core samples were analysed for alpha and beta activity control.

For the beta activity, analyses carried out on six cross-sections of the core samples taken, resulted in a mean value X_B = 16.7 mBq/cm^2 and a standard deviation S_B = 8.2 mBq/cm^2.

With the hypotheses of paragraph 6.1., probabiliorism told us, that for a random sampling of N values from a normal population with calculated mean value X_B and standard deviation S_B, the most probable value for the mean would be somewhere between $X_B \pm t \times S_B / N^{1/2}$, where t represents the Student-factor, depending on one single parameter, the degree of freedom of the sampling problem.

For a degree of freedom N - 1 = 5, and for a confidence interval of 99 %, it could be stated with 99 % certainty that the most probable mean value for the measurements carried out, would not be higher than 30.2 mBq/cm^2. In practice it would even be lower, taking into account the hypotheses from paragraph 6.1..

It was furthermore accepted that the fission products Sr^{90}, Y^{90} and Cs^{137} each were detected showing equal activity (resulting mass-absorption coefficient for the medium 0.026 cm^2/mg), and an homogeneous distribution in a concrete layer of 1 mm thickness. So it could be calculated that the transmission due to selfabsorption would be limited to 17.4 % (Ref [2]). In this way the mean specific beta activity of the analysed concrete layers proved to be 0.44 Bq/g, and the maximum for the mean value 0.79 Bq/g.

Further on, an evaluation of alpha measurements would only make sense in case also other than natural alpha activity had to be considered. Indeed, it could not be the aim to prove that natural radioactivity in the remaining concrete was by coincidence lower than somewhere else in other constructions or concrete elements. In the evaluation carried out, it had therefore to be taken into account that an important part of the considered alpha contamination was coming from the radionuclide Ra^{226}, due to natural radioactivity, as stated before.

So, to detect possible remaining alpha contamination in the core samples, the same six cross-sections as mentioned before were also at both sides analysed on alpha radiation, resulting in a mean value X_A = 0.96 mBq/cm^2 and a standard deviation S_A = 0.54 Bq/cm^2.

With the hypotheses of paragraph 6.1., in the same way probabiliorism told us, that for a random sampling of N values from a normal population with calculated mean value X_A and standard deviation S_A, the most probable value for the mean would be somewhere between $X_A \pm t \times S_A / N^{1/2}$, where t again represents the Student-factor, again depending on only one single parameter, the degree of freedom of the sampling problem.

For a degree of freedom N - 1 = 11, and for a confidence interval of 99 %, it could again be stated with 99 % certainty that the most probable mean value for the measurements carried out, would not be higher than 1.44 mBq/cm^2. In case of a homogeneous distribution of the activity in a concrete layer with thickness equal to the range R, it could be calculated that the transmission due to selfabsorption, for 2 π - measurements, would be limited to 50 % (Ref [3]). With a value for R = 6.5 mg/cm^2 (Ref [4]), the mean specific alpha activity in the analysed concrete layers proved to be 0.30 Bq/g, and the maximum for the mean value 0.44 Bq/g.

6.3. Destructive analyses

To check the indicated evaluations with reality, destructive analyses were carried out on the core samples. By means of alpha-gamma spectrometry the total alpha-beta activity in a cross-section of the core sample with the highest detected activity was determined. The results of these direct measurements revealed respectively a beta activity of 0.78 Bq/g and an alpha activity of 0.64 Bq/g.

In this way the practical analyses confirmed the evaluations carried out in calculating the specific values for possible alpha or beta activity in the core samples taken.

6.4. Concluding remarks

By means of practical measurements, and by means of evaluations, confirmed by the results of destructive alpha and beta analyses, it was demonstrated in a conservative way that, for a possible alpha or beta-gamma contamination in the remaining structure of both buildings, sufficiently low surface activities and specific activities were obtained.

Taking into account the hypotheses, where it was accepted in a conservative way, that all volumes originally showed the highest degree of contamination, despite the fact that an important part of the surfaces of the buildings originally showed no detectable contamination to the used detectors, the resulting specific activities were well below the proposed levels.

In this way sufficient evidence was delivered to carry out the last part of the decommissioning project for both buildings, i.e. the final demolition of the remaining structure, the removal of the demolition debris to an industrial dumping ground for inert wastes and restoration of green field conditions.

7. RADIOACTIVE WASTE MANAGEMENT

Decommissioning waste consisted of :
- metallic waste from the dismantling of the installations,
- concrete waste from the demolition of the structures,
- secondary waste generated during the above operations.

Total waste production was within 20 % of the estimates.

The total amount of metallic waste generated during the dismantling of the installations in the two storage buildings amounted to 47 ton. A large part of these metallic waste was later decontaminated. Finally therefore, only 16 tons or about 35 % of the metal materials routed to be radioactive waste. It should be mentioned that a considerable fraction of this waste only comprised suspect components, which could not be measured due to their shapes.

During the decontamination of both buildings about 76 ton of concrete waste was produced. About 53 % of the generated concrete waste was removed as inactive waste. This was partially due to decontamination of massive blocks, obtained from openings in the cell walls.

Secondary waste mainly consisted of protective clothing (17 m^3) and ventilation filters (1.5 m^3).

At last about 2,326 ton of concrete debris from demolition was removed to an industrial dumping ground for inert wastes.

8. HEALTH AND SAFETY

As a result of the intensive cleaning and preliminary decontamination operations, that were carried out between 1974 and 1978, after the shutdown of the reprocessing plant, the remaining radioactivity inventory in both buildings was extremely low as well as the remaining radiation levels.

Therefore, the total dose commitment during the decommissioning and dismantling operations, carried out by Belgoprocess, was limited to 7 mSv for 22,482 manhours or about 0.1 μSv/manhour.

9. MAN-POWER RESOURCES

From 01.01.1988 until 30.09.1989, 22,482 manhours have been performed within Belgoprocess. This value represents 13.8 manyears while 13.4 manyears were estimated.

The work done by the dismantling team itself represented 78 % of the total manhours.

The required assistance of the Health Physics Department was a factor that had been seriously underestimated. It took 16 % of the total manhours versus 10 % expected. An obvious explanation can be given in terms of the very low final contamination levels to be obtained in the remaining constructions, which necessitated intensive monitoring.

10. COST ESTIMATES

Costs were well within the estimates. Labour costs represented more than 50 % of the total costs. To significantly reduce future dismantling costs, Belgoprocess focuses on automation, especially in concrete decontamination.

A summary of the overall cost overview versus different cost factors is indicated in Table 1 below. It regards all the tasks carried out since the shutdown of the facility in 1975. It includes the costs for procurement of dismantling equipment, for radioactive waste conditioning, for intermediate storage and final disposal, and for the demolition of the concrete structures and the removal of demolition waste to an industrial dumping ground for inert wastes.

11. CONCLUSIONS

The pilot dismantling project provided a number of valuable results in terms of adopted techniques, waste production, manpower requirements and cost factors.

Table 1 : Summary of Cost Estimation versus different Cost Factors

Predecommissioning Operations	2.2 %
Facility Shutdown Activities (estimate)	4.0 %
Procurement of Equipment and Material	13.4 %
Dismantling Activities	15.2 %
Waste Management and Disposal	16.2 %
Security, Surveillance and Maintenance	0.6 %
Site Clean-up and Landscaping	2.7 %
Project Management, Engineering and Site Support	12.4 %
Research and Development	7.6 %
Other Costs	25.7 %
TOTAL	100.0 %

Practically, the main lines of the estimates made initially were more or less confirmed. The experience gained has been very useful for the study and the planning of the actual dismantling works in the main process building of the reprocessing facility, which started in October 1989.

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US NUCLEAR REGULATION WITH ENHANCED PUBLIC PARTICIPATION

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Abstract

Judging from recent years, the public in the United States can be expected to respond to and actively participate in the regulatory proposals concerning radioactivity--especially those regulations that may affect the environment. The US Nuclear Regulatory Commission (NRC) has responded to regulatory activism from a spectrum of interests, including citizens' groups and environmentalists, as well as industry and professional societies by enhancing participation from interested persons and organizations. Public, political, and legal opposition mounted against an NRC policy that stated that a radiation dose less than 100 μSv/a, and hence, corresponding levels of radioactivity were so low that they were "below regulatory concern," (BRC). The NRC did not implement the policy and eventually withdrew it. However, this experience stimulated the NRC to actively seek a broad spectrum of viewpoints and comments for the early and continued development of a controversial regulation on decommissioning criteria for lands and structures. This solicitation has led to a process that has increased the ability for the public, interested organizations, and local and state governments to provide early comment through public workshop meetings and easier access to information on the development of the regulation. As a result of these interactions, new concepts have been considered in the development of the regulation. Examples of concepts from the workshops are: the possibility of restricted release as well as unrestricted release; and the role of a site-specific advisory board in decommissioning.

1. Introduction

There is a strong commitment by the US Nuclear Regulatory Commission (NRC) to engage the public and interested organizations such as industry, professional societies, and local and state governments in the early development of a controversial regulation on decommissioning criteria. Interestingly, this commitment by the NRC came shortly after the intense public, political, and legal opposition mounted against an NRC policy that stated that 100 μSv/a was so low that it could be considered "below regulatory concern" (BRC). The evolution of the BRC policy, from its beginnings as a legal requirement of the Low-level Waste Policy Amendments Act of 1985 to the requirement of the Energy Policy Act of 1992 that caused the NRC revoke the BRC policy, illustrates lessons for regulators in a climate of active and concerned public interests. Similarly, NRC sponsorship of enhanced public participation illustrates a positive approach to account for a large variation of viewpoints early in the development of a controversial regulation.

2. *Background*

2.1 BRC Policy Statements--History

In 1985, a federal law on low-level waste policy directed the NRC to develop criteria and procedures to act upon petitions "to exempt specific radioactive waste streams from regulation...due to the presence of radionuclides...in sufficiently low concentrations or quantities as to be below regulatory concern." In 1986, the NRC issued such a policy statement and implemented the directive in regulations. Later, in 1990, the NRC issued a second, general, BRC Policy to provide guidance for making decisions on whether to grant specific exemptions in categories such as: (1) The release of sites containing residual radioactivity; (2) The distribution of consumer products containing small amounts of radioactivity; (3) The disposal of certain wastes containing very low levels of radioactivity; or (4) The recycling or reuse of radioactive materials that have very low levels of radioactivity.

After the issuance of the BRC Policy, the NRC staff conducted a number of public meetings to explain the rationale of the policy and receive comments. The staff were met by a largely angry public and demonstrations against the policy. The NRC was taken to court in a law suit against the policy. Numerous letters were written by concerned citizens to their congressmen and senators, the President, and directly to the NRC. The public reaction resulted in the introduction of legislation on the national, as well as legislation by a number of state and local governments, that would prevent the BRC Policy from taking effect. The NRC placed a moratorium on the implementation of the BRC Policy and then attempted to communicate with interested and concerned organizations in a mediated, consensus-building process that failed by December 1991. Eventually, in late 1992, the US Congress enacted a law, later signed by the President, that required the NRC to revoke both the BRC Policy Statements. The NRC formally withdrew both the BRC Policy Statements in August 1993.

2.2 Lessons Learned from Responses to BRC Policy Statements

Former NRC Chairman Kenneth Carr reflected on the experience with the BRC Policy Statements in the April 1993 meeting of the US National Council on Radiation Protection and Measurements. He suggested that the full force of opposition is felt at the time of establishing explicit criteria. Hence, potential opposition should be anticipated and included in the developmental process from the early stages to facilitate understanding and appropriate compromises and to better anticipate opposition. Nearly all groups contacted during the consensus-building attempts were in support of the consensus-building process--but not all. It seems clear that both sound policy and pragmatic analysis require that all affected and influential points of view be represented for consensus building. However, consensus building after the fact, across all interested parties was not possible at that time. Short of such all inclusive breadth, only a coalition, rather than a consensus, may be developed.

Public interest groups must be invited to participate in a manner that will not divorce them from their existing legislative and litigation alternatives. Opposition should not be expected to relinquish power for the opportunity to communicate with the regulatory agency.

The former Chairman also stated that a broad-based regulation that would apply to many circumstances has less chance of succeeding than a selective regulation. The analysis of anticipated effects of a narrowly focused regulation can be more complete and leave less to uncertainty and opposing speculation on application.

Coordination with other regulators was incomplete prior to the issuance of the BRC Policy Statements. Many of the states and the US Environmental Protection Agency (EPA) were not in agreement with the policy statements. A number of local governments passed laws that forbid disposal of any radioactivity in landfills under their jurisdiction. The states established the right to require more restrictive standards relating to the BRC Policy Statements than those that the NRC imposed through the 1992 Energy Policy Act. The EPA was not convinced that the NRC analyses of risk were sufficient for the wide range of exposure scenarios that could occur. Further, risks acceptable to the EPA implied dose levels less than 40 μSv/a, rather than the 100 μSv/a established in the 1990 BRC Policy Statement.

The above lessons were clear before the end of the consensus-building process in December 1991, and the NRC, building on its experiences, took another course as it began the development of a much needed, but controversial regulation on decommissioning criteria for lands and structures.

3. Enhanced Participation for Development of Decommissioning Criteria

3.1 The Process

In their June 28, 1991, directive to the staff, the Commissioners provided instruction that should the consensus-building process on the broad policy statement fail, "the NRC staff should select a single issue, such as the establishment of residual contamination criteria, and attempt to build consensus on that matter using a process that would go beyond the normal process of notice and comment rulemaking. The staff should consider a process that includes the conduct of early scoping meetings with all interested parties who are willing to participate, including public interest groups, Agreement and non-Agreement States, industry groups, professional associations, and societies, NRC licensees, and other Federal agencies, in order to ascertain the depth and breadth of the issues to be considered. Consideration should also be given to conducting workshops and holding public meetings to solicit views of the interest[ed] parties on these issues."

As noted above, the consensus-building process on the broad policy did fail, and the subsequent, final plan for the enhanced participatory rulemaking process on the radiological criteria for decommissioning was approved on October 28, 1992. The process enhanced the participation of a wide variety of interests thorough a series of seven workshops held regionally between January and May 1993. The workshops were facilitated by a professional organization and were not designed to seek consensus. Rather, the workshops were conducted at a very early stage to ensure that relevant issues had been identified, to exchange information on these issues, identify underlying concerns and areas of disagreement, and where possible, approaches for resolution.

EPA was represented at each of the workshops as well as representatives from state and local governments, and other interested parties as directed by the

Commission. Invited participants who would have been unlikely to attend the workshops because of a lack of funds were reimbursed for their travel and limited expenses by the NRC. In addition to the workshops, eight scoping meetings were held in four cities to solicit comment on the scope of the Generic Environmental Impact Statement (GEIS). The comments from all these workshops and meetings and submitted written comments are being used to prepare the GEIS and the draft proposed regulation. Comments from invited participants as well as from the observing public are included in the considerations. The EPA is a cooperating agency in the development of the GEIS and as such lends technical support. A summary of the comments from these workshops and meetings will be published as a NUREG report in November 1993.

An electronic bulletin board was established and dedicated specifically to this regulatory effort. This bulletin board provides, through a toll-free number, an additional communication link between the NRC and those outside the agency. The bulletin board is user-friendly, provides useful information concerning present decommissioning criteria and efforts, and enables the user to down-load files, to leave messages, and to up-load comments that will be a part of the official record. The bulletin board is attended daily and thus, is an easy way for users to follow the progress of the regulatory process. By October 1993, there were nearly 300 users and almost 200 calls per month.

A further, unprecedented, enhancement for participation is planned. Agreement States, workshop participants, and other interested persons, will be provided a staff draft of the GEIS, the draft regulation text, and the rationale in January 1994. That is, the materials will be released outside the agency before they will be sent to the Commissioners for approval as a proposed regulation package. Comments from this second tier of enhancement will be considered before forwarding the proposed regulation and its supporting documents to the Commission for consideration and approval.

3.2 Format for the Workshops

Comments from the workshops were stimulated and organized through a professionally facilitated discussion of an issues paper. The issues paper was divided into two parts. First were two primary issues: (1) The fundamental approach to the establishment of criteria; and (2) application of practicality considerations. Four distinct approaches for the regulation were presented: (a) risk limits, where a limiting value is selected and criteria are established below the limit using practicality considerations; (b) risk goals, where a goal is selected and practicality considerations are used to establish criteria as close to the goal as practical; (c) best effort, where the technology for decontamination considered to be the best available is applied; and (d) return to background, where the decontamination would continue until the radiological conditions were the same as existed prior to the licensed activities.

Secondary issues included discussions such as the time frame for dose calculations, the individuals or groups to be protected, the use of separate criteria for specific exposure pathways such as groundwater, the treatment of radon, and the treatment of previously buried materials.

Workshop participants were provided with the issues paper in advance, and the discussion actually was guided by cross-cutting issues, that is, central themes common with all of the issues raised in the issues paper. The cross-cutting issues were: advantages and disadvantages of generic standards; protection of human health, safety, and the environment; waste management implications; relationship to existing regulatory frameworks; technical capabilities and implementation considerations.

3.3 Comments from the Workshops

The following are selected viewpoints, expressed by some, but not all participants. These viewpoints are presented to represent some of the range of discussion on various issues. Clearly, some of the comments disagree with others.

> Unrestricted use of decommissioned facilities as an ultimate goal should be questioned. Restricted use should also be considered.
>
> The public, including local government, citizen groups, and others should participate at each phase of both rulemaking and implementation.
>
> The ultimate goal of decommissioning should be the return to background at the site.
>
> Background, as an objective, may be difficult to achieve and measure and may not be justifiable from a risk or cost standpoint.
>
> Limits are important to ensure compliance.
>
> Limits could allow too much health risk.
>
> Best effort types of approaches were not desirable from a risk standpoint and from a finality standpoint.
>
> Flexibility should be provided in the regulations to adjust for site-specific conditions.
>
> Optimization approaches are supported by industry and professional societies, but are distrusted by citizens and local government representatives for a variety of reasons.
>
> Costs should play a practical role in the selection of criteria.
>
> Costs have no role in the selection of criteria.
>
> The ability to demonstrate compliance is important.
>
> All types of risks should be included in the optimization assessments, such as occupational risk, public risk, non-radiological risks, and transportation risks.
>
> Waste disposal issues are closely tied to decommissioning.

4. Conclusions

US public interest and activism on the regulation of radioactivity and the environment have grown significantly over the last decade. Local and state governments are increasingly insisting on a role of involvement in the regulatory process. Thus far in the development of criteria for decommissioning, the NRC staff have found that early involvement of the public and others outside the agency is valuable in the formulation of the regulation. These interactions can increase the range and types of options considered, serve to provide better understanding of specific situations, positions and feasibility of implementation, and provide a forum for dialogue on an equal basis. The staff expects the continued interaction with those outside the agency through the normal solicitation of comments and the electronic bulletin board.

In the future, the NRC may consider using an enhanced participatory process for the establishment of other controversial regulations.

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