IAEA-TECDOC-1449

Radiological aspects of non-fixed contamination of packages and conveyances

Final report of a coordinated research project 2001–2002

IAEA
International Atomic Energy Agency

June 2005

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IAEA-TECDOC-1449

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Final report of a coordinated research project
2001–2002

IAEA
International Atomic Energy Agency

June 2005

The originating Section of this publication in the IAEA was:

Radiation, Transport and Waste Safety Section
International Atomic Energy Agency
Wagramer Strasse 5
P.O. Box 100
A-1400 Vienna, Austria

RADIOLOGICAL ASPECTS OF
NON-FIXED CONTAMINATION OF PACKAGES AND CONVEYANCES

IAEA, VIENNA, 2005
IAEA-TECDOC-1449
ISBN 92–0–103105–X
ISSN 1011–4289

Printed by the IAEA in Austria
June 2005

FOREWORD

The production and use of radioactive material in activities relating to the generation of nuclear power involve the transport of radioactive material in the public domain. In accordance with its statutory authorization, the IAEA publishes safety standards, including its Regulations for the Safe Transport of Radioactive Material, which undergo continuous review and are revised as necessary. The IAEA transport regulations, which are prepared in consultation with experts from IAEA Member States, are adopted essentially in their entirety by various international transport organizations and by many IAEA Member States.

The regulations, inter alia, stipulate the limits for the level of radioactive contamination on the external surfaces of packages that are used for the transport of radioactive material. The majority of packages, conveyances and associated equipment typically are not contaminated. However, packages, conveyances and equipment in use or associated with parts of the nuclear fuel cycle are prone to become contaminated during transport.

The approach to non-fixed contamination control in the IAEA transport regulations consists in applying derived limits for non-fixed surface contamination together with the requirement to keep contamination 'as low as practicable'. This approach was used in the first edition of the IAEA's Transport Regulations and has remained virtually unchanged up to the present day. The original basis for the derived limits for non-fixed contamination was consistent with the models used for derived limits for contamination in laboratories, hospitals and industry. The methodology has proved to be sound.
It was reported in the International Journal of Transport of Radioactive Material (Volume 10, No.2, 1999) that contamination in excess of the regulatory limits was observed in some flasks and wagons transporting radioactive material. However, no significant radiation exposure to the workers or the public occurred. It was recognized that certain safety issues relating to package/conveyance needed to be re-evaluated. The Transport Safety Standards Committee (TRANSSC) of the IAEA (formerly the Transport Safety Standards Advisory Committee (TRANSSAC) recommended at its May 2000 meeting that the IAEA undertake a coordinated research project (CRP) on contamination. The scientific scope and the programme goals that were identified included collection of data on contamination; sources of contamination; steps for reducing contamination and an evaluation of the radiological safety impact of the current requirements for non-fixed contamination. This TECDOC represents the final results of the research undertaken during this CRP.

A CD-ROM including the working material and the reports used in the CRP is supplied separately with this TECDOC.

The IAEA Officers responsible for this publication were M.E.Wangler and A.N.Nandakumar of the Division of Radiation Transport and Waste Safety.

EDITORIAL NOTE

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1. INTRODUCTION

The external surfaces of packages containing radioactive material, specifically those containing spent fuel, may routinely or inadvertently become contaminated with radioactive substances in the form of fixed (non-removable) or non-fixed (removable) surface contamination during transport. The non-fixed radioactive contaminants may subsequently be transferred from external package surfaces to the transport equipment and transport personnel or to areas accessible to the public. Such transfer of contamination has the potential to give rise to radiation exposures of people and the environment. Surface contamination on radioactive material packages, conveyances and the transport equipment is thus an issue that is related to occupational health and radiological safety and therefore requires careful control.

The IAEA Regulations for the Safe Transport of Radioactive Material, TS-R-1 [1.1] prescribe limits and controls for fixed and non-fixed contamination on the surfaces of packages and conveyances. To ensure an adequate level of safety and protection of persons, property and the environment the radioactive surface contamination must be restricted so as not to exceed the prescribed limits.

The regulatory requirements and controls for limiting the level of radioactive surface contamination on packages and conveyances have been in existence for some four decades [1.2]. The basis of the derivation of the regulatory non-fixed surface contamination limits is a model originated by A. Fairbairn [1.3]. In recent years, however, it has been recognised that a review of these regulatory limits is required, taking into account the current philosophy in radiological protection and the current transport practices.

REFERENCES TO SECTION 1

[1.1] INTERNATIONAL ATOMIC ENERGY AGENCY, Regulations for the Safe Transport of Radioactive Material, 1996 Edition, Safety Standards Series No. TS-R-1 (As Amended 2003), IAEA, Vienna (2004).

[1.2] INTERNATIONAL ATOMIC ENERGY AGENCY, Regulations for the Safe Transport of Radioactive Materials, Safety Series No. 6, IAEA, Vienna (1961).

[1.3] INTERNATIONAL ATOMIC ENERGY AGENCY The derivation of maximum permissible levels of radioactive surface contamination of transport containers and vehicles, Notes on Certain Aspects of the Regulations for the Transport of radioactive materials, Safety Series No. 7, IAEA, Vienna (1961), p.79.

2. SCOPE AND OBJECTIVES

In recognition of this need to review and assess the current contamination limits, an IAEA Coordinated Research Project (CRP) on the Radiological Aspects of Package and Conveyance Non-Fixed Contamination was initiated. The objective was to review the scientific basis for the current regulatory limits for surface contamination. The CRP was also to develop guidance material for evaluating the radiological significance of surface contamination to workers and the public in the light of state of the art research and technical developments and current transport practices. The specific objectives of the work undertaken within this multi-national CRP were, in accordance with the terms of reference:

- to ensure that appropriate models exist for all package types including consideration of the aspects pertinent to assessing and revising a surface contamination model for transport,
- to collect — where possible — contamination, operational and dosimetric data to ensure modelling consistency,
- to develop models for assessing the radiation doses incurred in transport operations, and
- to consider preventive methods for package and conveyance contamination.

The CRP was to include the review and analysis of these critical issues and was to characterise the dose pathways for non-fixed surface contaminants on packages and conveyances under routine conditions of transport and to quantify the resulting internal and external radiation dose to transport workers and the public (per unit of surface contamination on packages). This assessment was to take into account the different categories of radioactive material packages used in practice and related transport conditions typically encountered in the transport of radioactive material.

This publication presents and summarises the principal findings and conclusions of the collaborative work undertaken by six of IAEA Member States and one international organization within this IAEA CRP.

3. BACKGROUND

The IAEA's Regulations for the Safe Transport of Radioactive Material [3.1] contain specific requirements for contamination levels on packages and conveyances used for the transport of radioactive material (paragraphs 508 – 514). It was reported in the International Journal of Transport of Radioactive Material [3.2] that contamination in excess of the regulatory limits was observed in some flasks and wagons transporting radioactive material. However, no significant radiation exposure to the workers or the public had occurred. It was recognized that issues related to package and conveyance needed to be re-evaluated. The Transport Safety Standards Committee (TRANSSC) recommended at its May 2000 meeting that the IAEA undertake a coordinated research project on non-fixed contamination.

3.1. Formation of the CRP

Following the announcement by the IAEA that the CRP was to be undertaken, six countries and one international organization agreed to undertake research to support the project. These were:

France
Germany
Japan
Sweden
United Kingdom
United States of America
World Nuclear Transport Institute (WNTI)

With regard to the Terms of Reference for the CRP, the participants agreed the research efforts should include

- ensuring appropriate models exist for all package types, including consideration of the following aspects
    - public exposures
    - protection of the environment
    - worker exposures
    - critical groups and their exposures
    - appropriate radiological recommendations and data
    - routine conditions of transport only
    - package types, conveyances, materials, modes of transport
    - environmental conditions
    - all relevant exposure pathways,
- collecting – where possible – contamination, operational and dosimetric data to validate modelling results,
- using the models for assessing contamination limits and optimisation of doses in operations, and
- considering preventive methods for package and conveyance contamination

Based on the high priority placed on the CRP by TRANSSC, an accelerated schedule was developed to reduce the normal 3-year CRP cycle. It was agreed that efforts would be directed at completing the CRP by early 2003. There were a pre-meeting and three Research Coordination Meetings (RCMs) held to coordinate the research efforts and share results as the work progressed. These were:

- 2–4 May 2001, Paris, (pre-meeting)
- 7–9 November 2001, London
- 18–21 June 2002, Williamsburg, VA, USA
- 19–22 November 2002, Berlin

A consultants services meeting was held during 17-21 March 2003 in Vienna to prepare the final report of the CRP.

It was agreed early in the project that a single model should be developed for most of the dose calculations. This model, called the Basic Model, is described in detail in Section 6. Participants from France, Germany, United Kingdom and WNTI met three times to coordinate the development of the Basic Model and input from other participants was considered during this time. The calculation techniques, assumptions and other parameters used in the Basic Model were considered in detail and final agreement on these was achieved at the third RCM in Berlin. Calculations were performed independently by the Basic Model group participants to verify the accuracy of the spreadsheet being used. Forty nine radionuclides were modelled during the development and testing of the Basic Model. Once completed, the Basic Model was used to calculate values for all radionuclides listed in the IAEA Transport Regulations.

A coordinated effort was made to validate the assumptions used for exposure distances and times. Questionnaires were developed and circulated to shippers to obtain estimates of these parameters for a variety of package types in different industries (radiopharmaceutical, nuclear fuel cycle, nuclear power reactors and industrial radiography). Previous studies were also identified that detailed spent fuel handling operations and these were used to ensure the appropriateness of assumptions on worker task assumptions.

3.2. Research undertaken by participants

This section contains a summary of work performed by each participating country and organisation. Details of the research results are provided in the annexes, which can be found in the CD supplied with this publication.

3.2.1. France

The contribution of France to the CRP consisted first of a detailed analysis of the Fairbairn model [3.3] by the Institut de Radioprotection et de Sûreté Nucléaire (IRSN) presented at the first RCM in London.

Within the framework of the CRP, EDF performed experimental research [3.4] on the resuspension of the non-fixed contamination present at the surface of the spent fuel casks.

A detailed questionnaire was developed by IRSN [3.5], and circulated to the following companies to collect detailed information on transport operations and package handling conditions:

COGEMA (Compagnie Générale des Matières nucléaires)– spent fuel, vitrified waste, uranyl nitrate, plutonium oxide powder, unirradiated fuel and uranium oxide

- CIS BIO – radiopharmaceuticals
- ALSTOM – industrial radiography
- EDF (Electricité de France) – unirradiated and spent fuel and waste
- FBFC (Franco Belge de Fabrication de Combustibles) – unirradiated fuel, uranium oxide and uranium hexafluoride
- CEA (Commissariat à l'Energie Atomique) – tritium
- EURODIF – uranium hexafluoride
- COGEMA LOGISTICS

The responses to the questionnaire were compiled and taken into account during the discussions on exposure parameters for the Basic Model.

IRSN contributed to the development and the implementation of the Basic Model [3.6, 3.7]. Calculations were performed for 350 radionuclides and the results were given at the third RCM in Berlin. A final version of the contribution of IRSN to the Basic Model with scenarios and parameters reviewed during the Berlin meeting is given in this report, in Section 6, with results obtained for 350 radionuclides and recommendations.

In parallel to the CRP, and on the behalf of the European Commission, COGEMA LOGISTICS led a study [3.8] performed with the participation of COGEMA (France), EDF (France), SKB (Sweden), GRS (Germany) and BNFL (United Kingdom). This study deals with decontamination and preventive methods of contamination.

3.2.2. Germany

The recognition that the safety of transport workers and public health are of paramount importance in the transport of radioactive material has resulted in research work in Germany aiming to collect, compile and analyse information pertinent to judging the achieved level of safety and protection in transport. The research work undertaken by Germany has inter alia resulted in an accumulation of expertise and radioactive material transport databases with relevance to this CRP. The available databases include records on the type, nature and frequency of surface contamination in nuclear fuel transport and time and motion studies for various transport disciplines. The databases and records available have been analysed to assist the model development process and to provide input data characterising the type, duration and source-receptor distances for various transport operations, i.e. fuel cycle and non-fuel cycle transport operations.

The radiation survey and assessment results available in Germany confirm the general understanding that the radiation exposures received by transport workers and members of the public under normal conditions of transport are generally low for all major categories of radioactive material and well below the applicable regulatory dose limits [3.9, 3.10].

3.2.3. Japan

Japan explored two models to calculate doses from contamination on spent fuel casks. These models use representative radionuclides and include a conservative model and a realistic model. The conservative model uses parameters chosen so that they are very unlikely to be exceeded. The realistic model uses parameters more closely aligned with actual shipping experience. Sensitivity analyses were performed and identified the most sensitive parameters in the models [3.11, 3.12]. The resuspension factors, which were experimentally determined by Japan were adopted in the Basic Model developed in the CRP.

3.2.4. Sweden

Sweden, SKB, has contributed to the CRP with a study concerning the ‘weeping’ phenomena (Weeping: A surface phenomenon characterized by egress of small quantities water which had been adsorbed on the external surface layer of a package – typically a spent fuel cask - following periods of immersion in fuel storage pools or decontamination) [3.13].

3.2.5. United Kingdom

NRPB in particular provided a major input into the development of the exposure model. Calculations giving exposures for unit contamination were developed using a spreadsheet, and the modelling group report was prepared by NRPB staff [3.14]. These contributions were made in cooperation with other participants in the modelling group. Working parameters of times, distances and other variables, were discussed with representatives of the UK nuclear industry, who provided helpful comments. Representatives of the UK nuclear industry also provided relevant material used as guidance within the UK nuclear industry. Staff from the UK competent authority for safe transport of radioactive material also contributed to the discussions during the RCM and also hosted the London RCM.

3.2.6. United States of America

Participants from the United States performed two studies to support the CRP. The Spent Fuel Project Office of the US Nuclear Regulatory Commission sponsored a study on optimising the doses from contamination on spent fuel casks. A study on the nature of dissolved caesium fixation to metal surfaces was sponsored by the National Transportation Program of the US Department of Energy.

The research undertaken by the Nuclear Regulatory Commission [3.15] focused on developing an approach to optimise doses resulting from removable contamination on spent fuel casks. This work built on the Basic Model to the greatest extent possible and expanded the doses taken into account to include collective internal and external doses to workers and members of the public.

Detailed evaluation of time and motion information on spent fuel handling operations in the United States was performed to characterize the doses to workers performing decontamination and monitoring tasks and other workers involved in cask handling operations. By calculating collective as well as individual doses, the overall effects of higher allowable contamination limits could be calculated.

These results confirmed the extremely low magnitude of doses to the public from contamination on spent fuel casks and examined the dose implications of higher allowable contamination limits.

The research undertaken by the Department of Energy [3.16] examined a potential cause of cask weeping and how it might be prevented or minimized. Experiments were performed that involved immersing stainless steel coupons in contaminated water (simulating spent fuel pool water), measuring surface contamination, and evaluating the efficacy of removing the contamination with an ion exchange mechanism. Some of the coupons were pre-treated to determine if pre-treatment is effective in minimizing contamination adherence. The work focused on caesium as dissolved contaminant since it is a major constituent of pool water contamination. The working hypothesis for the experiments was that there is a nanolayer of clay minerals on all the surfaces and the contamination bonds electrostatically between the clay layers (‘fixing’ the contamination in place). An ion exchange mechanism can then result in the contamination coming free from the clay, resulting in 'non-fixed' or removable contamination (the weeping phenomena). Some coupons were exposed to road grime and subsequently washed to remove large particles. Some coupons were pre-treated and all were immersed in the contamination solution, counted and washed (some with water and some with ion exchange solutions) to determine what can be removed. Results showed that pre-treatment with non-radioactive caesium is the most effective pre-treatment and coupons treated with road grime picked up more caesium.

3.2.7. World Nuclear Transport Institute (WNTI)

The WNTI contribution to the CRP consisted primarily of a detailed evaluation of the Fairbairn model, the initial model development (a first version of the current Basic Model was developed for the first RCM in London, November 2001), and a recommendation of the Verein der Groβ kraftwerksbetreiber (VGB), Germany Working Group on Practical Radiation Protection on prevention and control of contamination in the transport of radioactive materials [3.17]. In addition, the final version of the Basic Model has been used as the basis for preparing a WNTI report [3.18], which provides proposals for revised transport contamination levels.

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[3.2] COMMON REPORT OF THE COMPETENT AUTHORITIES OF FRANCE, Germany, Switzerland and the United Kingdom, Surface contamination of Nuclear Spent Fuel Shipments, International Journal of Radioactive Materials Transport, 10 No. 2, (1999).

[3.3] PERRIN, M.L., First conclusions on the analysis of the Fairbairn Model, Report No. SEGR/03-61, Institut de Radioprotection, et de Surete Nucleaire, France (2001) (included in the CD supplied with this report)
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[3.7] Lizot, M.T., et al Surface Contamination Model for Transport, Summary of IRSN Contribution to the Working Group on modelisation of the IAEA Coordinated Research programme, SSTR/03.1105 indice 1, (2003).
[3.8] TCHATALIAN, B., Harmonisation of Procedures for assuring non-contamination on containers for irradiated fuel transports, EEC Contract 4.1020/D/99-001. Reference 12510-B-2 Rev.0, NUSYS, France, (2002) (included in the CD supplied with this report separately).
[3.9] SCHWARZ, G., Characterisation of the type and duration of selected transport operations giving rise to occupational and public exposures during the transport of radioactive material in Germany. Working Paper prepared for the Modelling Group of the IAEA Coordinated Research programme (CRP) on contamination, Gesellschaft für Anlagen- und Reaktorsicherheit (GRS) mbH, Cologne, Germany and Bundesamt für Strahlenschutz, Salzgitter, Germany, 2002 (included in the CD supplied with this report separately).
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[3.13] CARLSSON, M., Physico-chemical investigation of surface contamination on nuclear fuel containers, Department of Analytical Chemistry and Nuclear Chemistry, Uppsala University, Sweden, 2002 (included in the CD supplied with this report separately).
[3.14] NATIONAL RADIOLOGICAL PROTECTION BOARD, Report of the Modelling Group, Final Report, Version 5, Chilton, UK, (2003).
[3.15] RAWL, R.R., et al., Establishing Risk-Informed Limits for Non-Fixed Surface Contamination on Spent Fuel Transportation Casks, Oak Ridge National Laboratory and Nuclear Regulatory Commission, (2003) (included in the CD supplied with this report separately)..
[3.16] McCONNELL, P., KRUMHANSL, P., KAPOOR, A., Ion Exchange Mechanism for Non-fixed Cesium Surface Contamination on Spent Fuel Packages: Hypothesis, Confirmation, and Mitigation - Final Report; (2002) (included in the CD supplied with this report separately).
[3.17] VGB WORKING GROUP ON PRACTICAL RADIATION PROTECTION: Recommendation “Prevention and Control of Contamination in the Transport of Radioactive Materials”, Verein der Großkraftwerksbetreiber (VGB), Germany (2002).
[3.18] WNTI, “Radiological Model for Deriving Transport Contamination Limits”, World Nuclear Transport Institute, (2003), (included in the CD supplied with this report separately).

4. DESCRIPTION OF THE CONTAMINATION ISSUE

4.1. Current regulatory requirements

Limits on removable contamination on the external surfaces of transport packages have been included in the IAEA Transport Regulations since their inception [4.1]. These requirements are based on a dosimetric model developed by Fairbairn [4.2] that has served as the technical basis for these limits. This section describes the original model, and some earlier studies that have been performed, and illustrates some of the issues that have arisen concerning removable surface contamination.

Currently, the regulatory requirements applicable to removable surface contamination are found in paragraphs 508 and 509 of the IAEA Regulations for the Safe Transport of Radioactive Material. These paragraphs state:

> 508. The *non-fixed contamination* on the external surfaces of any *package* shall be kept as low as practicable and, under routine conditions of transport, shall not exceed the following limits:
>
> (a) 4 Bq/cm² for beta and gamma emitters and *low toxicity alpha emitters*, and
>
> (b) 0.4 Bq/cm² for all other alpha emitters.
>
> These limits are applicable when averaged over any area of 300 cm² of any part of the surface.
>
> 509. Except as provided in paragraph 514, the level of *non-fixed contamination* on the external and internal surfaces of *overpacks*, *freight containers*, *tanks* and *intermediate bulk containers* shall not exceed the limits specified in paragraph 508.

There are a wide variety of conditions in which radioactive materials packages are prepared for transport. In some cases newly manufactured packaging components are used to assemble the package in very clean environments (e.g., radiopharmaceutical manufacturers) and it is highly unlikely that any contamination will be present on the outer surfaces of the package. Under other conditions, such as the reuse of packages that must be loaded in potentially contaminated areas (e.g., a spent fuel cask) the packages may require extensive efforts to decontaminate and monitor the exterior surfaces to ensure compliance with the regulations. Despite this wide variety of package use scenarios, a single set of requirements for removable contamination have been applied to all package types throughout the history of the IAEA Transport Regulations.

It has been recognized for a number of years that some packages, such as spent fuel casks, may require extensive decontamination in order to comply with the regulations, both on dispatch and throughout the journey. Some of this difficulty is due to a phenomenon referred to as ‘weeping’ or ‘sweating’ where a surface that has been decontaminated becomes recontaminated. This phenomenon is due to fixed contamination becoming removable due to changes in humidity, temperature, etc. The advisory material for the transport regulations [4.3] includes the following guidance:

> 508.3. In practice, contamination which appears fixed may become non-fixed as a result of the effects of weather, handling, etc. In most instances where small packages are slightly contaminated on the outer surfaces, the contamination is almost entirely removable or non-fixed, and the methods of measurement should reflect this. In some situations, however, such as in the case of fuel flasks which may have been immersed in contaminated cooling pond water whilst being loaded with irradiated fuel, this is not necessarily so. Contaminants such as ^{137}Cs may strongly adhere onto, or penetrate into, steel surfaces. Contamination may become ingrained in pores, fine cracks and crevices, particularly in the vicinity of lid seals. Subsequent weathering, exposure to rain or even exposure to moist air conditions may cause some fixed contamination to

be released or to become non-fixed. Care is necessary prior to dispatch to utilize appropriate decontamination methods to reduce the level of contamination such that the limits of non-fixed contamination would not be expected to be exceeded during the journey. It should be recognized that on some occasions the non-fixed contamination limits may be exceeded at the end of the journey. However, this situation generally presents no significant hazard because of the pessimistic and conservative assumptions used in calculating the derived limits for non-fixed contaminations. In such situations the consignee should inform the consignor so that the latter can determine the causes and minimize such occurrences in the future.

The weeping phenomenon has been recognized and addressed in the United States by a special domestic provision [4.4] that allows for a 10 times increase in removable contamination on spent fuel casks between dispatch and receipt, when the shipment is made under exclusive use. In some cases, consignors decontaminate the cask to a fraction, such as 1/10, of the allowable limit in order to accommodate 'weeping' and prevent a non-compliance situation.

4.2. Contamination events involving railway wagons in Europe

Following the discovery of non-fixed radioactive contamination on spent fuel flasks and associated transport equipment in France, a working group of the competent regulatory authorities was set up by agreement between the German Minister for the Environment and the French Minister for the Environment, on May 26, 1998. On June 2, 1998, a meeting of the German, French and Swiss competent regulatory authorities convened a working group on radioactive contamination in Cologne, Germany. The United Kingdom joined the working group at its second meeting in Paris on June 22, 1998. The working group had the task to ensure a coordinated action by experts in each country to establish the causes and effects of the contamination incidents and propose appropriate actions.

A report [4.5] was issued that presented a compilation of information collected from all four countries, setting out the history of the events, their causes and consequences. The report describes remedial actions that were put in place and made recommendations for future actions.

> "[A]n assessment of dose commitments was done. For COGEMA workers of Valognes rail terminal the analysis of the dose readings showed that the maximum dose detected was 3.2 mSv/year compared with the limit of 20 mSv/year for other workers in the nuclear industry. The Office for Protection against Ionising Radiation (OPRI) carried out inspections which focused on determining the level of radioactivity in the body of SNCF workers who are most involved in the transport of radioactive fuel. Four different sites were selected: the departure area at Paluel Nuclear power plant, the Rouen marshalling yard, the arrival area at Valognes station and the wagon maintenance centre at Ternier. The results of the 131 whole-body monitoring operations carried out on SNCF workers from all four sites are normal. They show that the level of contamination of artificial origin in their body was not above the detection threshold. The persons examined had been engaged in various work areas: track workers, drivers, operations and maintenance.
>
> For the public, the highly conservative estimates made by the Institute for Nuclear Safety and Protection show:
>
> - for the risk of irradiation: an effective dose rate of 0.003 mSv/h at two metres from the external surface of the vehicle, whereas the regulatory limit under conditions likely to be encountered in routine transport is 0.1 mSv/h.
>
> - for internal contamination: an effective dose rate equivalent of 0.0001 mSv/h.
>
> In conclusion, as far as health is concerned, the non-compliance with the 4 Bq/cm² standard did not have any radiological consequence. The health regulations were respected both in the case of COGEMA workers, who carry out unloading and monitoring operations of contamination at

Valognes, and the public. In Switzerland a whole-body monitoring was carried out on the rail workers involved in spent fuel shipments. The results of the monitoring of 124 persons have shown no intake of radioactivity which could originate from the spent fuel shipments. In Germany and the UK similar monitorings and/or assessments lead to the same conclusions." [4.5]

The report also identified the need for technical improvements by the consignors of the fuel flasks and made recommendations regarding reporting incidents and the establishment of a database on spent fuel shipments.

4.3. Measurements according to ISO 7503

The ISO standard 7503 [4.6] sets out procedures for the measurement of contamination. This procedure has been adopted in a wide variety of countries and provides for a standard methodology to be applied. There were no particular problems identified in the methodology for assessing the activity on a wipe as contained in the ISO standard as far as transport was concerned. However the techniques involved and difficulties of performing wipe tests are not covered in sufficient detail in this ISO standard.

Users should ensure that the lower limit of detection is lower than required taking all these efficiencies into account. On the other hand, attention is drawn to the detail contained in the ISO standard relating to the efficiency with which wipes can be counted using detector systems. For transport a particular circumstance that should be kept in mind is the presence of a high radiation field that prevents direct measurement in many cases.

While the ISO standard gives a good methodology for monitoring loose contamination there are differences between this and the methodologies set out in the transport regulations. For example, the ISO standard sets the contamination limits on the basis of 100 cm^2 as the area to be wiped, while the transport regulations use 300 cm^2.

4.4. Fairbairn model and its limitations

4.4.1. Derivation of contamination limits

The limits for non-fixed contamination on packages and conveyances are specified in the IAEA transport regulations as 4 Bq/cm^2 for beta, gamma and low toxicity alpha emitters and 0.4 Bq/cm^2 for all other alpha emitters. Before conversion to SI units, these limits were specified as 10^{-4} $\mu Ci/cm^2$ and 10^{-5} $\mu Ci/cm^2$ respectively. These limits were derived using a simple model of worker exposure. The derivation of the limits was published in advisory and explanatory material [4.7] to accompany the 1961 edition of the IAEA Regulations [4.8].

The Fairbairn model considers only inhalation of airborne contamination and transfer of contamination to the human hands under a specified set of exposure scenarios. The permissible levels of contamination are constrained so as not to result in a) an airborne concentration greater than the maximum permissible concentration in air (MPCa) specified in the 1959 recommendations of the International Commission on Radiological Protection (ICRP) [4.9] and b) contamination of the hands beyond what was considered to be good practice in the 1960s. These constraints were applied to those alpha and beta emitting radionuclides, then considered most hazardous, i.e., to ^{239}Pu (alpha emitter) and ^{90}Sr (beta emitter). Good practice regarding hand contamination was based on limiting the irradiation of the basal layer of the skin since the potential ingestion of contamination from the hands was found to be less restrictive. The model resulted in limits for non-fixed contamination of 0.4 and 4 Bq/cm^2 for alpha and beta emitters, respectively. These derived limits have been in use for many years.

Recently, there have been a number of changes in radiation protection philosophy and associated dosimetry, mainly as a result of the recommendations of the International Commission on

Radiological Protection. These include changes in the dose coefficients for inhalation of radionuclides and a change in the specification of the annual dose limit for workers. Also, during the period since the contamination limits were derived much experience has been gained in their use in many types and stages of the transport of radioactive materials. In particular, the experience of contamination of packages and conveyances in the transport of irradiated fuel, mainly by rail, has indicated the need for a review of these limits.

The participants in this CRP provided comments and criticisms of the original derivation of the contamination limits by the Fairbairn model [4.10, 4.11, 4.12]. The main issues raised concerning the Fairbairn model are summarised here.

4.4.2. Radionuclides

The radionuclides used in the Fairbairn model, ^{90}Sr and ^{239}Pu, were considered to be the most restricting. However, these are not the most relevant radionuclides in practical transport situations, nor do they represent the range of dose coefficient values. Any review of the Fairbairn model should include a range of radionuclides.

4.4.3. Resuspension mechanism

The inhalation of contamination resuspended from a surface is an important mode of exposure and the way this mechanism is modelled is critical in determining the contamination limits. The resuspension factor that was used in the Fairbairn model, 4×10^{-5} m^{-1}, was for dusty operations in confined working areas, and it was acknowledged that for outdoor situations this factor might be some 20 times less than the value used. Most transport operations are carried out in large working areas, or outdoors, for which a lower resuspension factor would be appropriate. Also, the resuspension factor used in the Fairbairn model was for dusty operations, whereas most packages, in particular INF flasks are handled in clean areas. The value of the resuspension factor used in the Fairbairn model may therefore be over restrictive by at least an order of magnitude.

New experimental work has been called for which would provide data that would support a more appropriate way to model the resuspension mechanism. The resuspension mechanism might be better represented by determining a resuspension rate from the surface, together with the dispersion of the contamination away from the surface using a suitable air exchange rate. This could enable other factors to be taken into account, such as the type of surface and the effect of weather.

4.4.4. Aerosol characteristics

In the Fairbairn model, no account was taken of the characteristics of the resuspended contamination, in particular the particle size. This factor has implications for the fraction of activity assumed to be inhaled, and therefore of the resulting dose. The particle size also determines the relevant dose coefficient to be used in the assessment of dose from inhalation. New experimental work could provide useful data on aerosol characteristics.

4.4.5. Occupancy times

In the Fairbairn model, the dose from inhalation of resuspended activity was taken into account using the maximum permissible concentration in air. This quantity is specified on the assumption of an annual exposure of 2000 hours to the resuspended activity. This level of occupancy time is achieved in only a very few specialised situations, and for most transport workers occupancy times are likely to be very much lower. Information on occupancy times in modern transport practices should be obtained so that realistic exposure scenarios can be specified, for both workers and members of the public.

4.4.6. Exposure pathways

Only two modes of exposure are used in the Fairbairn model: inhalation of resuspended contamination and dose to the skin from activity transferred from the surface. It has been observed that the Fairbairn

model implicitly assumed a factor of 10% for the transfer of non-fixed contamination from the surface to the hand, and that the suitability of this should be re-examined. A replacement model should take into account other relevant exposure pathways such as external irradiation and inadvertent ingestion of activity.

4.4.7. Dose coefficients

The Fairbairn model uses the maximum permissible air concentration for assessment of inhalation of resuspended contamination, and these were based on dose coefficients current at the time. The model is therefore based on the outdated critical organ approach to radiological protection that poorly, if at all, reflects actual health risk. The more recent ICRP recommendations are risk-based. The current dose coefficients are based on the latest lung model recommended by ICRP [4.13] and in general are significantly different from those used in the Fairbairn model. The indications are that if the calculations in the Fairbairn model were carried out with modern dose coefficients, the contamination limits so derived would be at least an order of magnitude less restrictive.

4.4.8. Dose criteria

The Fairbairn model uses the maximum permissible air concentration for assessment of inhalation of resuspended contamination, and a restriction on hand dose for activity transferred to the skin. Both of these made use of the limitation of dose to workers that were current at the time the Fairbairn model was formulated, which corresponded to 50 mSv/a. The recommendations of ICRP have been revised a number of times since then and recommendations on dose limitation have been revised. Also, the assumption that a worker exposed to specified contamination limits should imply that a dose limit is reached may not be in keeping with modern radiation protection objectives. It may therefore be more appropriate to specify dose criteria below the annual dose limits for workers. The potential exposure of members of the public also needs to be taken into account. These issues need to be discussed within the review of the contamination model.

4.4.9. Exposure of members of the public

The Fairbairn model is based on a transport worker that works in the vicinity of packages continuously for 2000 hours per year. It does not take into account exposures to the public during the course of transport.

4.4.10. Application to spent fuel flasks

The issue of contamination of packages has mainly concerned flasks used to transport spent nuclear fuel (SNF), most other types of packages tend to be free of such contamination. Hence the applicability of a model for such contamination has a particular significance for the transport of these flasks. The following issues are therefore relevant:

- The radionuclides used in the Fairbairn model, while limiting, are inappropriate for LWR SNF packaging applications. For example, the use of ^{134}Cs, ^{137}Cs, ^{58}Co and ^{60}Co are more appropriate for SNF applications as these are the radionuclides most frequently encountered on casks surfaces following immersion in spent fuel pools.

- The exposure scenarios (hand contact without subsequent near-term monitoring or decontamination, dusty conditions, etc.) considered in the model are not representative of the operational practices and environments associated with handling of spent fuel casks.

- The inhalation intakes are based on a scenario occurring “in a confined space with conditions arranged to simulate very dusty conditions” [4.7]. This approach is very conservative considering the conditions under which SNF packages are stored, prepared, and transported today.

The Fairbairn model is based on exposure scenarios that are not appropriate for spent fuel flasks. Limits for non-fixed surface contamination on spent fuel flasks should ideally be established using a model that considers and optimizes the appropriate exposure scenarios both in the workplace and in the environment. The existing conservative limits may lead to operational problems, non-compliance issues, and public perceptions of significant health risks when the actual risks are very low.

4.4.11. Conclusions on the limitations of the Fairbairn model

The Fairbairn model has served as the basis for the surface contamination limits for four decades. However, some of the underlying factors used in the derivation of contamination limits, such as the dose coefficients, have been significantly revised in recent years and therefore a review of those limits is necessary. Almost all of the observations on the Fairbairn model conclude that a number of the aspects of the calculations use very conservative assumptions. When these are combined the resulting limits may therefore carry an unwarranted level of conservatism, and therefore a replacement model should use the most appropriate values at each stage of the calculation of contamination limits. Additional calculations are needed to ensure that appropriate values are calculated for all relevant radionuclides.

In the case of spent fuel flasks, a risk-informed approach would ensure optimal use of personnel and material resources for SNF-based packaging operations.

The implicit use of an annual dose limit as a criterion for the contamination limits might now not be appropriate and a lower criterion should be used, so that this aspect may therefore be not conservative enough. The comments and observations on the original Fairbairn model therefore strongly suggest a review of the contamination limits is necessary, supported by information on modern transport practices and new experimental work. The use of the word "limits" may be inappropriate, while the terms "derived limits" or "derived levels" may be more accurate.

4.5. Previous studies

4.5.1. Results of previous studies

Over the years, studies have been made of the potential exposure pathways and doses that might result from removable surface contamination. The results of these studies generally confirmed that very low doses are likely to result from surface contamination on transport packages.

4.5.2. NRPB

The United Kingdom National Radiation Protection Board (NRPB) conducted a study [4.14] for the UK Department of Transport to examine a dose-based approach for setting limits for non-fixed contamination the surface of transport packages. This effort included formulation of an assessment model to translate the appropriate dose limits into derived limits on surface contamination.

The exposure pathways for workers considered in the NRPB study included:

- External irradiation of the skin
- Inhalation of resuspended surface contamination
- Ingestion of surface contamination transferred to the hands

The model only addressed exposures to workers and, while suggesting that the only significant pathway should be inhalation of dispersed contamination during transport, it is noted that such exposures are “exceedingly low.” Thus the model included no provisions for assessing the exposure to members of the public. The focus of the NRPB report was on deriving contamination limits.

Derived limits were reported for a number of radioisotopes under various parametric assumptions. Derived limit results of 600 and 700 Bq/cm^2 were reported for ^{137}Cs and ^{134}Cs, respectively, assuming average distance and time parameter values ('choice estimate' terminology used in the report).

4.5.3. Pacific Northwest Laboratory

The US Nuclear Regulatory Commission (NRC) sponsored a study at Pacific Northwest Laboratory (PNL) of removable surface contamination on radioactive material transportation containers [4.15]. The study addresses various categories of radioactive material and includes consideration of the dose to workers and the public associated with the transport of spent fuels. Estimates of the additional economic costs incurred by lowering the current contamination limits by factors of 10 and 100 were presented.

The modelling effort in the study included the development of a generic model that was applied to several categories of transport containers. The potentially significant exposure pathways considered were

- Ingestion (intakes of non-fixed contamination transferred to hands)
- Inhalation (intakes of airborne contamination resuspended from container surfaces, and worker's contaminated hands)
- Direct exposure (to container surface contamination, skin contamination, and during decontamination of container surfaces).

The dosimetric evaluations are based on the pre-ICRP Publication 30 [4.16] models and thus are more than twenty years out of date. An interesting aspect of the study is the significance attributed to the ingestion pathway. The exposure scenario involves a transfer of non-fixed contamination from the package surfaces to the hands, foodstuffs, and other items, which enter the mouth. The model did consider both workers and members of the public.

In applying the model to spent fuel casks it was noted that such containers are in exclusive use and hence not recycled for other uses which potential might lead to contamination entering other pathways that lead to members of the public. Transportation workers were assumed to ingest surface contamination from 5 10^{-5} m^2 for each container. No release of the surface contamination was considered during the actual transport and thus the model does not consider inhalation intake by members of the public. The assumed lack of release from the container also means that members of the public were not directly exposed to non-fixed contamination (only direct exposure to the contents of the container). However, despite stating the assumption of no exposure to the public, an ingestion intake by 10 individuals was assumed as the only pathway for public exposure to non-fixed surface contamination.

The impact on the collective radiation dose of decontamination activities was evaluated. The analysis considered the total time spent decontaminating at a fixed rate of decontamination, the exposure rate of the worker to the contents of the cask and to the non-fixed contamination, and the reduction in the public dose resulting from the reduced surface contamination. The relative impact of these factors is shown in adapted from the study. A surface contamination level of 4 Bq/cm^2 was assumed and a decontamination factor of 10 was assumed for each 30 minutes of decontamination effort. That figure indicates an optimum additional decontamination time of approximately 20 minutes for a direct exposure rate of 10 μSv/h and such dose savings from additional decontamination are only possible when there is a low direct exposure rate. At higher direct exposure rates (0.1 and 1 mSv/h), any additional decontamination efforts resulted in higher total doses as the increase in worker dose was much greater than the resulting reduction in public dose. This reflects the finding that it is the worker direct exposure considerations that were driving this model. Figure 1 shows the estimated collective dose plotted against the decontamination time adopted from [4.15].

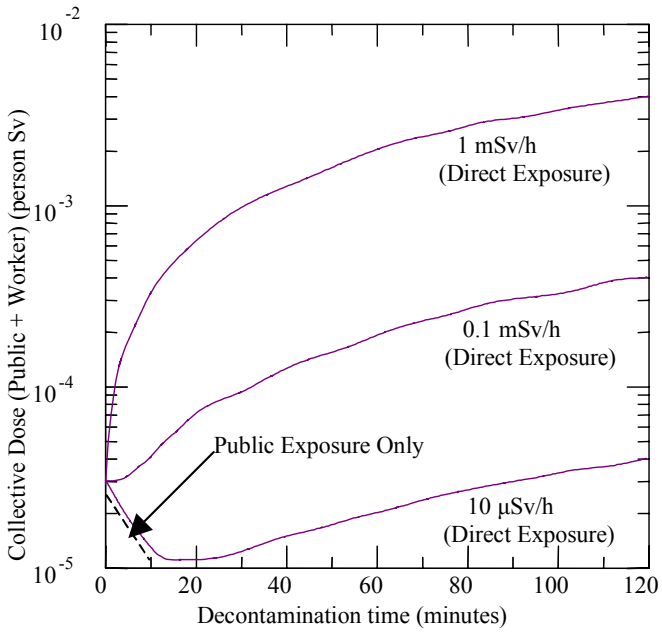

Figure 1. Collective dose vs. decontamination time for spent fuel container. Adapted from [4.15].

4.5.4. IAEA meetings

4.5.4.1. Meetings and dates

A number of meetings have been held specifically to examine the contamination issue, and other meetings have addressed the topic. The main meetings have been :

- Consultant Services Meeting on Transport Package Surface Contamination. CS-86, IAEA Vienna, 28–30 June 1999.
- Consultant Services Meeting on Contamination Issues Relating to the Safe Transport of Radioactive Material. CS-158, DETR London, 13–15 December 1999.
- Technical Committee Meeting to identify issues relating to the requirements of the current transport regulations and to develop approaches to guide the revision of these requirements. TCM 1156, IAEA Vienna, 6–10 March 2000.

4.5.4.2. Outcome of Meeting CS-86

The CSM held at IAEA, Vienna, in June 1999 agreed on the following overall conclusions.

(1) The values for non-fixed contamination in the first transport regulations had a sound basis.

(2) The realistic exposures of workers and members of the public are low at the current non-fixed contamination levels. Recent work shows that the original basis remains conservative.

(3) The 4 and the 0.4 Bq/cm² values are not primary limits but are derived limits. There are no health consequences of any significance associated with exposures at these values.

(4) Most packages are free of surface contamination. The few that are contaminated must be decontaminated to below the derived limits for non-fixed contamination prior to transport, taking into account compliance with these derived limits under routine conditions of transport.

(5) Further work is required on the controls for contamination for conveyances and equipment under exclusive use. The relevant text in the IAEA Regulations should be revised in the future.

(6) There are considerable margin of safety associated with the current requirements.

(7) The presentation of the basis, application and consequences of the non-fixed surface contamination values must be improved. Agency publications should be used in this regard.

4.5.4.3. Outcome of Meeting CS-158

The issues relating to contamination in transport were reviewed during this meeting, which was held in London in December 1999.

It was noted that while there was no reason to doubt the fact that the current limits are low in terms of radiological effect, the age of the model and the fact that it is not complete in any one document, would make it hard to defend.

It was concluded that a new model should be derived starting from first principles. The group suggested that the following aspects should be taken into account:

(1) The models should cover workers, the public and the environment (non-controlled area accumulation).

(2) They should be based on critical groups.

(3) Transport specific scenarios should be identified covering all types of shipments under routine conditions of transport.

(4) The models should be realistic, but with conservative unknowns.

(5) A sensitivity study should be included where appropriate.

(6) Significant uncertainties should be identified.

(7) The models should use appropriate radiological data.

(8) Further detailed considerations include:

   (a) Package types should be considered.

      (i) Excepted, Type A, Type B etc.

      (ii) Weight

      (iii) Surface temperature

      (iv) Surface dose rate

   (b) Conveyances should be considered.

   (c) Resuspension should be examined in detail.

   (d) Weather conditions should be covered.

   (e) Different contamination distributions should be examined (gross surface or spot/particle).

   (f) The interface with measurement procedures should be considered.

   (g) Modes of transport and intermodal transfer should be included.

   (h) The effects of segregation requirements should be considered.

4.5.4.4. Outcome of Meeting TCM 1156

A Technical Committee Meeting was held at Vienna in March 2000, entitled "TCM to Identify Issues Relating to the Requirements of the Current Transport Regulations and to Develop Approaches to Guide the Revision of these Requirements". The chairman's report included in the recommendations for guidance material the following: "4.3.1.5 - a new model that considers contamination should be developed."

A working group considered this topic and its report included the following:

"*New Model for Contamination"*

The Working Group recommends a new model that considers contamination. The purpose of the model is not to establish a limit but to evaluate the radiological consequences of a cleanliness goal. The reasons include the following: dose optimisation, consideration of all pathways, Special Arrangement justification, and demonstrating the suitability of cleanliness levels. The modellers should use practical data and Member States are encouraged to make this information available on request.

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5. EXAMINATION OF PHYSICAL PARAMETERS

5.1. Resuspension rates and factors

5.1.1. Relevance for the Basic Model

The resuspension rates or resuspension factors are a key element of the Basic Model. In fact, the conservative choice of values for the resuspension factor has already been a very important part of the Fairbairn model [5.1]. The assumptions on resuspension govern the entire inhalation pathway. As resuspension rates or resuspension factors may vary over several orders of magnitude, careful choice of realistic yet conservative enough values is necessary.

It is necessary to describe the difference between the resuspension rate and the resuspension factor.

- The resuspension rate is given in h^{-1} (derived from Bq/h per Bq). and describes the percentage of matter present on a surface (here: activity) that is suspended from the surface into the surrounding air. It mainly depends on the properties of the surface, the interaction between matter on the surface (here: the contaminants), and the air speed. It is independent of the size of the room or, more generally, the volume of air into which resuspension occurs.

- The resuspension factor is given in m^{-1} (derived from Bq/m³ per Bq/m²). This value is only meaningful for a given combination of package type, room size, air exchange rate etc. and incorporates the resuspension rate as well. While the resuspension factor is less transparent than the resuspension rate because it needs detailed descriptions of the conditions, it has the advantage that it can be measured directly in experiments.

Within the course of this CRP, a number of experimental approaches for evaluating resuspension factors and rates have been performed. Experiments in France and Japan are described in the following sections 5.1.2 and 5.1.3. A literature survey has been performed by the WNTI group before developing their model that is described in section 5.1.4.

5.1.2. Experimental work from France

Within the framework of the CRP, The Electricite de France (EDF) has evaluated in an experimental way the resuspension of the non-fixed contamination present on spent nuclear fuel transport cask [5.2]. For the purpose of these experiments, EDF has designed a wind tunnel, ERMITAGE, that made it possible to carry out 20 experiments of resuspension due to wind action with:

- representative particles of the contamination possibly present on a transport container surfaces,

- various surfaces: stainless steel of roughness 1.6 μm, stainless steel of roughness 0.02 μm, and elastomer; these surfaces have not been cleaned after deposition of the contamination in order to obtain pessimistic conditions of resuspension;

- a duration of 30 minutes of air flow;

- two different surface temperatures: ambient and 80°C;

- different speeds of wind of 3 m/s, 10 m/s and 30 m/s.

The resuspension factor thus obtained is valid within the framework of the resuspension conditions in ERMITAGE:

- very close to the inner surface, the experiment cell being 10 cm high,

- for the duration of the experiment (30 minutes);
- its extrapolation to other conditions requires correction factors (in particular a dilution factor to simulate the concentration in the air at a given distance from the surface).

Under these intentionally pessimistic conditions and after statistical analysis, the study, based on 20 tests, has shown that:

- the resuspension factor found varies overall from 1 10^{-6} to 9.6 10^{-6} m^{-1}
- its average is 2 10^{-6} m^{-1},
- resuspension of 70% to 90% of the total resuspended activity occurs during the first 2 minutes of air sweeping,
- neither wind speed nor the temperature have a decisive influence on the resuspension factor, unlike the nature of surface.

It is possible for EDF to eliminate the particles of important diameter that are not hazardous in terms of inhalation to carry out other experiments.

5.1.3. Experimental work in Japan

Experimental work was performed in Japan to determine experimental values for resuspension rates (dispersal rates) for the main radioisotopes used (Table 1). The experimental device was a simulated structure of a hood for handling of radioisotopes. The wind velocity around the specimen was 10–30 cm/s [5.3].

Table.1. Values for resuspension rates on the basis of Japanese experiments

Resuspension rate value	**Valid for:**
1 10^{-4} h^{-1}	^{3}H, ^{14}C, ^{35}S, ^{75}Se
1 10^{-5} h^{-1}	^{77}As, ^{103}Ru, ^{125}Sb, ^{125}I, ^{131}I, ^{137}Cs, ^{197}Hg
1 10^{-6} h^{-1}	For other metal radionuclides
1 10^{-7} h^{-1}	^{22}Na, ^{24}Na, ^{32}P, ^{45}Ca, ^{51}Cr, ^{59}Fe, ^{67}Ga, ^{99}Mo, ^{99m}Tc, ^{111}In, ^{147}Pm, ^{198}Au, ^{201}Tl

5.1.4. Literature study performed by the WNTI

The WNTI has performed a literature survey on resuspension rates. In the literature, only very few experimental data can be found that fit the contamination situation of transport package surfaces and that are good enough to be used as references. Many data on resuspension rates or resuspension factors are related to resuspension of soil or dust by wind over open country, on roads etc. Literature values should, therefore, only be used as reference after checking the scenario carefully.

A relevant publication in this respect is NUREG, 1981 [5.4]. For the Basic Model presented here, a resuspension rate of 10^{-4} h^{-1} is at the upper end of the data given in Table 3.1-3 in [5.4]. The experimental evidence provided in sections 5.1.2 and 5.1.3 above indicates that real resuspension rates will be much lower, suggesting that a value of 10^{-4} h^{-1} is a conservative approach.

5.2. Operational data

5.2.1. Data from participants

The operational data used in the Basic Model takes into account the type and duration of selected transport operations and the related worker-package distances have been derived from work performed and relevant national databases and records collected and compiled in France, Germany, UK, USA and by WNTI.

5.2.2. Data from France

The French operational data describe the type and nature of a range of transport operations typically encountered in the land-based transport of radioactive material. The operational transport data provided in [5.5] are based on a questionnaire survey performed in France in 2002 and relate specifically to the type and duration of transport operations and the related average worker-package distances associated with the shipments of commercial spent nuclear fuel, vitrified radioactive waste, pre-fuel material (e.g. uranium hexafluoride, uranyl nitrate, PuO_2-powder), radioactive waste and the supply and distribution of radioisotopes.

5.2.3. Data from Germany

Similarly, Germany provided operational transport data [5.6] in terms of working-exposure patterns relevant for the assessment of radiation doses of workers and the public from potential radioactive surface contamination on packages and conveyances and from workplaces where radioactive material packages are prepared and handled. The data have in part been derived from recent transport worker dose monitoring campaigns and cover shipments of commercial spent nuclear fuel, vitrified waste, LLW/ILW and operations involving radioisotopes for medical, scientific and industrial applications. In addition a dose assessment approach has been devised on the basis of empirically determined effective worker-package distances derived from transport worker monitoring data [5.6].

5.2.4. Other operational data sources

The US operational data reflect a comprehensive review and analysis of transport operations involving postulated commercial spent nuclear fuel shipments [5.7]. The UK and WNTI provided an extensive transport database on volumes of radioactive material shipments and operational procedures with emphasis on large remotely handled ISO containers, spent nuclear fuel and other shipments in the UK and continental Europe [5.8, 5.9, 5.10, 5.11].

REFERENCES TO SECTION 5

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[5.7] SCHNEIDER, K.J., et al., Analysis of Radiation Doses from Operation of Postulated Commercial Spent Fuel Transportation Systems, DOE-CH/TPO-001, (1987).
[5.8] NATIONAL RADIOLOGICAL PROTECTION BOARD RADIATION EXPOSURE FROM THE NORMAL TRANSPORT OF RADIOACTIVE MATERIALS WITHIN THE UNITED KINGDOM - 1991 Review, NRPB-R255, Chilton, UK, (1992).
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6. DEVELOPMENT OF THE BASIC MODEL

6.1. Modelling group

During the 1st RCM in London, a number of working groups proposed similar approaches to carrying out modelling and calculations for the radiological aspects of radioactive contamination on packages. However, it was noted that there w0ere some differences in methods and assumptions. It was, therefore, proposed that those groups and individuals, with a particular interest in developing these models should meet to discuss the development of a Basic Model.

The first meeting was held at NRPB, Chilton, on 14-15 February 2002. Exposure scenarios and parameters were agreed and the four main groups involved, IRSN, GRS, WNTI and NRPB carried out calculations based on those agreements. A second meeting was held at RWE, Essen, on 29-30 April 2002, to review the scenarios and parameters, and to resolve some difficulties that had been recognised. Also at that meeting, scenarios and parameters were developed to model the exposure of members of the public. A meeting was held at GRS, Cologne, on 21-22 August 2002, to review the scenarios and parameters in the light of the discussions held at the second RCM in Williamsburg held in June 2002. At this meeting in Cologne additional scenarios were considered and the existing scenarios were modified. During the 3rd RCM in Berlin all the scenarios and many of the parameters were reviewed, and those final values are those used in the Basic Model described in this report.

6.2. Modelling approach

It is recognised that the transport and handling conditions of radioactive material packages and shipments may vary between operators and for transport operations. However, there are sufficient common features and working procedures to enable the development of representative scenarios and exposure parameters. The transport and handling conditions of workers and the public that should be reflected in such an exposure model, i.e., the model structure and model parameter values, do not necessarily represent a particular transport situation or geographical environment. Rather, the essential elements of the transport phases are abstracted to form a general model for these operations. Such a model should provide a practical means for the assessment of radiation exposures of workers and the public that is unlikely to be exceeded under actual transport conditions.

In order to assess exposures of workers and members of the public to radioactive contamination on packages and conveyances, it is necessary to consider the various stages of transport and how individuals may become exposed. These include the preparation of the package, labelling, marking, handling, transport/ storage and final receipt and checking by the consignee. During the London meeting, the initial operations relevant to contamination control during transport were discussed. In order to establish a consistent starting point for all package types, it was decided that the Basic Model would begin with the final inspection stage immediately before shipment.

The Basic Model is intended to cover all the main types of transport operations. It was considered that focusing on different package types, each representing the main types of transport, was an efficient way of achieving this approach. The Basic Model was developed around the transport procedures in a way that follows the steps taken during a typical shipment. The exposure of workers and the public was then considered at each phase of the transport process, for each of the package types. The details of this approach are described in the next sections.

6.3. Exposure situations

6.3.1. Package types and load parameters

It was suggested at the London meeting, and subsequently affirmed by the modelling meetings, that four categories of packages would be considered for the modelling process that would cover the main types of radioactive material shipments.

These are:

- Small manually handled packages (SM), such as those used for industrial and medical materials;
- Small remotely handled packages (SR), such as waste or UO_2 drums moved using fork-lift trucks;
- Large remotely handled packages (LR), such as standard ISO freight containers or other large packages such as those used to transport uranium hexafluoride (this category may also include large overpacks); and
- Irradiated Nuclear Fuel flasks (FF).

The dimensions of the four package types are given in Table 2, and have been chosen to represent typical dimensions of the four types of packages. The total surface area and volume, of the packages are given in the table, with the number of packages assumed per load, and the cross sectional area of the load.

Table 2. Package types and load parameters

		Package Type			
Parameter	**Unit**	**SM**	**SR**	**LR**	**FF**
Height	M	0.3		2.4	
Width	M	0.3		2.4	
Length	M	0.3	0.9	5.9	6
Diameter	M		0.6		2.5
Surface area of package	m^2	0.54	2.3	68	130
Volume of package	m^3	0.03	0.25	34	29
Cross section of a package (external)	m^2	0.9	5.4	58	49
Cross section of a load (external)	m^2	22.5	32	58	49
No. per load, small van	[-]	25	n/a	n/a	n/a
Loads per day, small van	d^{-1}	2	n/a	n/a	n/a
No. per load, large van	[-]	100	42	1	1
Loads p.day., large van	d^{-1}	1	1	1	0.5

The dimensions of the FF package are representative of PWR/BWR flasks, that is, the larger examples of this type of package. Flasks used in the UK for Magnox/AGR fuel have smaller dimensions. A larger package will represent more conservative conditions.

For the SM packages two exposure conditions are modelled. The use of a small van, carrying 25 packages involves inhalation of resuspended material by the driver. For the larger van , it is assumed that 100 packages are carried in a separate compartment, so there is no inhalation dose to the driver during the movement phase.

For LR and FF packages the average number of packages despatched per day is 1 and 0.5 respectively. However, it is assumed that the same number of returning empty packages are also handled and therefore the total number handled is twice those given in Table 2.

6.3.2. Stages of transport operations assumed in the Basic Model

The Basic Model considers a series of steps representing the stages of the shipment and each stage consists of a number of tasks. The steps are shown below in Table 3.

6.3.3. Worker parameters

6.3.3.1. Worker tasks

The worker tasks are listed in Table 4 for each package type. Steps 1.1 to 1.4 are the final operations before despatching the package, after the cleaning and decontamination of the package. The tasks end at the consignees' premises with acceptance checks. For each type of package movement, groups of workers carrying out each task are identified by letters. Each of these groups may consist of one or more workers.

Table 3. Steps in the transport operations within the Basic Model

Main Step	Sub-Step
1. Final inspection of package	1.1 Visual inspection 1.2 Dose rate measurement 1.3 Contamination measurement 1.4 Labelling of package
2. Loading onto conveyance	2.1 Transfer from site to vehicle 2.2 Fastening, lifting, loading and fixing 2.3 Dose rate meas. at conveyance 2.4 Contamination measurement of conveyance 2.5 Placarding of conveyance
3. Movement phase	3.1a Movement during transport – road/rail 3.1b Movement during transport – air 3.1c Movement during transport – sea 3.2 Unforeseen interruptions 3.3 Regular stop
4. Transfers during transport	4.1 Unloading (including sub-steps) from first conveyance 4.1.1 Dose rate and contamination measurements 4.1.2 Unfixing, fastening, lifting 4.2 Loading (including sub-steps) onto second conveyance 4.2.1 Fixing 4.2.2 Dose rate measurement at conveyance 4.2.3 Contamination measurement of conveyance 4.2.4 Placarding of conveyance 4.3 Regular stops
5. Receiving inspection	5.1 Visual inspection (conveyance/package) 5.2 Dose rate measurement, conveyance 5.3 Unfixing, fastening, lifting, unloading 5.4 Transfer from conveyance to consignee 5.5 Dose rate measurement on package 5.6 Contamination measurement on package 5.7 Contamination measurement on conveyance

Table 4. Tasks and exposure scenarios for workers within the Basic Model

Sub-Step	Exposed Workers, Package Type SM	SR	LR	FF
1.1 Visual inspection	A	A	A	A
1.2 Dose rate measurement	A	A	A	A
1.3 Contamination measurement	A	A	A	A
1.4 Labelling of package	A	A	A	A
2.1 Transfer from site to vehicle	C	B,C	B,C	B
2.2 Fastening, lifting, loading and fixing	C	B,C	B,C	B
2.3 Dose rate measurement at conveyance	A,C	A,C	A	A
2.4 Contamination measurement of conveyance	A,C	A,C	A	A
2.5 Placarding of conveyance	A,C	A,C	A	A
3.1 Movement	C	C	C	C
3.2 Interruptions during transport	C	C	C	C
4.1 Unloading (including Sub-steps) from conveyance #1				
4.1.1 Dose rate & contamination measurement	-	F	H	H
4.1.2 Unfixing, fastening, lifting, unloading	F	F	F,G	F,G
4.2 Loading (including Sub-steps) on conveyance #2				
4.2.1 Fixing	F	F	F,G	F,G
4.2.2 Dose rate measurements at conveyance	F	F	H	H
4.2.3 Contamination measurements of conveyance	F	F	H	H
4.2.4 Placarding of conveyance	F	F	F	F
5.1 Visual inspection (conveyance/package)	T	T	T	T
5.2 Dose rate measurements of conveyance	-	-	-	T
5.3 Unfixing, fastening, lifting, unloading	C	U,C	U,C	U
5.4 Transfer from conveyance to consignee	C	U,C	U,C	U
5.5 Dose rate measurements of package	T	T	T	T
5.6 Contamination measurements of package	-	T	T	T
5.7 Contamination measurements of conveyance	-	-	T	T

- Operation not normally applicable.

The tasks carried out by each type of worker are described below.

Worker A - Package preparation and transfer worker

A worker in Worker Group A carries out the final labelling and preparation of the packages, before carrying out checks on the vehicle. Worker A is likely to be a health physics worker at the consignors site.

Worker B – Fork-lift truck / loading / transfer vehicle operator

A worker in Worker Group B is a fork-lift truck / loading / transfer vehicle operator who transfers the package to the conveyance, and carries out the loading and fixing operations. There is no worker B in the case of small manually handled packages as these are assumed to be taken by hand to the conveyance.

Worker C - Conveyance driver

A worker in Worker Group C is the driver of the conveyance, and may also carry out some loading and unloading operations, apart from fuel flasks, where Worker B would carry out those tasks. For SM and SR packages the driver may also carry out dose rate and contamination checks, and placard the conveyance. Worker C can be the driver for the first or second mode.

At the transfer site:

Worker F – General transfer site worker

Worker G – Loading operator (e.g. crane operator)

Worker H - Health physics worker

Step 4 of the transport operation covers loading and unloading operations en-route. This may involve transferring the package between two modes of transport, for example, a fuel flask being transferred from a road vehicle to a rail wagon or packages delivered by van to an air cargo warehouse for loading onto an aircraft. Step 4 also covers situations where packages are handled at a distribution centre. In this case packages are unloaded from one vehicle and put onto another. The storage in transit step has been incorporated into this step. A worker in Worker Group F represents a general worker who performs a number of tasks. In reality, this would normally involve different individuals, but for simplicity in the model it is assumed that this could be one worker. However, for LR and FF packages, a dedicated loading operator (e.g. a crane operator) may be involved (Worker G). Also, at sites where LR and FF packages are handled there are likely to be dedicated health physics operators who carry out monitoring (Worker H).

At the consignee's premises:

Worker T - Health physics worker

Worker U - Unloading worker

A worker in Worker Group T is an employee at the consignee's premises who, as part of his duties, carries out visual and health physics checks on the received package. These checks vary between the four package types and Table 4 shows which sub-steps are involved in these checks. It is assumed that there are dedicated unloading operators (Worker U) for FF packages. These workers may also be involved in unloading SR and LR packages otherwise it is assumed that the driver unloads the vehicle.

6.3.3.2. Working times and distances

The times allocated for each task are listed in Table 5, with the relevant distances. Other relevant parameters are also included in the table.

Times are given for each operation either per package (pa), conveyance (co) or task (ta). Within the Basic Model, annual times are calculated from the individual task time and number of operations/ packages handled per annum. Annual times taken for individual tasks were summed for each of the worker groups. In a few cases, the total working time is in excess of the maximum annual transport working time, which was assumed to be 1500 h. This total is for transport operations, the remainder of the annual working time is assumed to be taken up with non-transport activities. In these instances, the doses were scaled from the calculated total time to the 1500 h working time. The 'excess time' would be distributed among other workers in that group. It is assumed that the driver of the conveyance (Worker C), spends 600 h per annum driving the conveyance in the movement phase. This was chosen as a representative fraction, about one-third, of the annual working time.

The task times and distances shown in Table 5 were discussed during the third and final RCM in Berlin. Operational data such as those described in section 5.2 were considered in arriving at the values shown in this table. It was acknowledged that for most transport operations the handling/ measurement times assumed may be too long, but it was argued that for a few transport operations assumed times may be shorter than those encountered in practice. Overall, these parameters have taken into account a wide range of working conditions.

Table 5. Parameters for worker exposure within the Basic Model

	Parameter	Unit	Package Type SM	SR	LR	FF
Working cond.	Working days per annum	d/a	250	250	250	250
	Fraction of annual working time	[-]	75%	75%	75%	75%
	Working hours per day	h/d	8	8	8	8
	Working hours per annum	h/a	1500	1500	1500	1500
Exposure time	1.1 Visual inspection	min/pa; dist, m	0.5; 1.0	1.0; 1.0	5.0; 1.0	10.0; 1.0
	1.2 Dose rate measurement	min/pa; dist, m	3.0; 1.0	10.0; 1.0	30.0; 1.0	45.0; 1.0
	1.3 Contamination final measurement	min/pa; dist, m	1.0; 0.5	3.0; 0.5	30.0; 0.5	30.0; 0.5
	1.4 Labelling of package	min/pa; dist, m	0.3; 0.5	1.0; 0.5	2.0; 0.5	2.0; 0.5
	2.1 Transfer from site to conveyance	min/pa; dist, m	0.5; 0.3	3.0; 1.0	10.0; 1.0	30.0; 3.0
	2.2 Fastening, loading, lifting and fixing	min/pa; dist, m	0.5; 0.3	4.0; 1.0	5.0; 1.0	60.0; 1.0
	2.3 Dose rate measurement at conveyance	min/co; dist, m	5.0; 1.0	10.0; 1.0	5.0; 1.0	30.0; 1.0
	2.4 Contamination measurement of conv.	min/co; dist, m	0.1; 1.0	5.0; 1.0	5.0; 0.5	30.0; 1.0
	2.5 Placarding of conveyance	min/co; dist, m	2.0; 1.0	2.0; 1.0	2.0; 1.0	5.0; 1.0
	3.1 Movement (with packages)	h/a	600; 1.0	600; 1.0	600; 1.0	600; 5.0
	3.2 Unforeseen interruptions	h/a	1.0; 1.0	1.0; 1.0	1.0; 1.0	1.0; 1.0
	4.1 Unloading (incl. Sub-steps) from conv. #1					
	4.1.1 Dose rate & contam. Measurement	min/co; dist, m	n/a	5.0; 1.0	5.0; 1.0	20.0; 1.0
	4.1.2 Unfixing, fastening, lifting, unloading	min/pa; dist, m	1.0; 0.3	7.0; 1.0	15.0; 1.0	30.0; 1.0
	4.2 Loading (incl. Sub-steps) on conv. #2					
	4.2.1 Fixing	min/pa; dist, m	1.0; 0.3	7.0; 1.0	15.0; 1.0	30.0; 1.0
	4.2.2 Dose rate measurement at conveyance	min/co; dist, m	5.0; 1.0	10.0; 1.0	5.0; 1.0	10.0; 1.0
	4.2.3 Contamination meas. of conveyance	min/co; dist, m	0.1; 1.0	5.0; 1.0	5.0; 1.0	10.0; 1.0
	4.2.4 Placarding of conveyance	min/co; dist, m	2.0; 1.0	2.0; 1.0	2.0; 1.0	5.0; 1.0
	4.3 Regular stops	h/a	n/a	n/a	n/a	n/a
	5.1 Visual inspection (conveyance/package)	min/ta; dist, m	0.2; 1.0	1.0; 1.0	5.0; 1.0	3.3; 1.0
	5.2 Dose rate measurement, conveyance	min/ta; dist, m	n/a	n/a	n/a	10.0; 1.0
	5.3 Unfixing, fastening, lifting, unloading	min/ta; dist, m	0.5; 0.3	4.0; 1.0	5.0; 1.0	60.0; 1.0
	5.4 Transfer from conveyance to consignee	min/ta; dist, m	0.5; 0.3	3.0; 1.0	10.0; 1.0	30.0; 3.0
	5.5 Dose rate measurement, package	min/ta; dist, m	1.0; 1.0	3.3; 1.0	10.0; 1.0	30.0; 1.0
	5.6 Contamination measurement, package	min/ta; dist, m	n/a	3.0; 0.5	5.0; 0.5	20.0; 0.5
	5.7 Contamination measurement, conveyance	min/ta; dist, m	n/a	5.0; 1.0	5.0; 1.0	30.0; 1.0
Room/Outdoor	Area	m^2	100	100	200	200
Inhalation	Height	M	2.5	2.5	10	20
	Effective Height (contam. spread)	M	2.5	2.5	10	10
	Effective Volume	m^3	250	250	2000	2000
	Air exchange rate	1/h	2	2	2	2
	Wind speed (outdoors)	m/s	2	2	2	2
	volume of cabin of small van	m^3	10	10	10	10
	air exchange rate car	h^{-1}	10	10	10	10
Transfer factors	Transfer package->hand	[-]	0.2	0.2	0.2	0.2
	Transfer hand->face	[-]	0.2	0.2	0.2	0.2
	Transfer hand->mouth for specific area	[-]	1	1	1	1
Contam. Area	Area for ingestion from cont. hands	cm^2	4	4	4	4
	Contaminated area of 2 hands, palms	cm^2	400	400	400	400
	Contaminated area of face	cm^2	100	100	100	100
	UV exposed area of skin	cm^2	3000	3000	3000	3000
	Time contamin.-washing, hands	h/d	4	4	4	4
	Time contamin.-washing, face	h/d	8	8	8	8
Ingestion	Ingestion rate from hands	d^{-1}	1	1	1	1
Ground shine	Average percentage of activity on ground	[-]	0.1	0.1	0.1	0.1

Note: Pa: package co: consignment

6.3.4. Parameters for the exposure of the public

6.3.4.1. Public exposure situations

The steps, and sub-steps, where members of the public may be exposed are shown in Table 6. The scenarios considered apply for the most exposed members of the public, i.e. the critical group.

Table 6. Public exposure situations within the Basic Model

	Exposed Members of Public, Package Type			
Sub-Step	**SM**	**SR**	**LR**	**FF**
3.1a Movement/interruptions during transport –road/rail	√	√	√	√
3.1b Movement during transport – air	√	√	-	-
3.1c Movement during transport – sea	√	√	√	-
3.2 Interruptions during transport	√	√	√	√
3.3 Regular stop	√	√	√	√
4.1 Unloading (incl. Sub-steps) from conveyance #1				
4.1.1 Dose rate & contamination measurement	-	√	√	√
4.1.2 Unfixing, fastening, lifting, unloading	√	√	√	√
4.2 Loading (incl. Sub-steps) on conveyance #2				
4.2.1 Fixing	√	√	√	√
4.2.2 Dose rate meas. at conveyance	√	√	√	√
4.2.3 Contamination meas. Of conveyance	√	√	√	√
4.2.4 Placarding of conveyance	√	√	√	√
4.3 Regular stop	√	√	√	√
5.1 Visual inspection (conveyance/package)	√	-	-	-
5.3 Dose rate measurement, package	√	-	-	-
5.4 Transfer from conveyance	√	-	-	-

For exposure situations during the movement phase (Step 3) the exposed members of the public are different for each sub-step and package type. That is, the member of public exposed during air transport is unlikely to be exposed also during sea transport of radioactive material. In the case of Step 4, the members of the public are assumed to live in close proximity to the transfer facility site, and therefore the exposures are summed for each sub-step.

6.3.4.2. Public exposure times and distances

Occupancy times and distances for members of the public were allocated to the steps in the transport operations for each package type where the exposure of members of the public is likely/ possible. These are listed in Table 7. Except in the case of aircraft passengers, members of the public are assumed to be exposed outdoors.

The occupancy times will not be limited by working hours as in the case of workers; in one instance they are in excess of 2500 hours. Members of the public will always be at a greater distance from the transport operations than will transport workers for each step. Their exposures are, therefore, expected to be much lower than that of the transport workers. However, the dose criteria applicable to members of the public are generally lower and, therefore, their exposures should be assessed to determine if they are more limiting in some situations.

Table 7. Public exposure scenarios and parameters within the Basic Model

Main Step	Sub-Step		Package Type SM	SR	LR	FF
1. Final Inspection of Package	1.1 Visual inspection		n/a	n/a	n/a	n/a
	1.2 Dose rate meas.		n/a	n/a	n/a	n/a
	1.3 Contamination meas.		n/a	n/a	n/a	n/a
	1.4 Labelling of package		n/a	n/a	n/a	n/a
2. Loading onto conveyance	2.1 Transfer from site to vehicle		n/a	n/a	n/a	n/a
	2.2 Fastening, lifting, loading and fixing		n/a	n/a	n/a	n/a
	2.3 Dose rate meas. at conveyance		n/a	n/a	n/a	n/a
	2.4 Contamination meas. of conveyance		n/a	n/a	n/a	n/a
	2.5 Placarding of conveyance		n/a	n/a	n/a	n/a
3. Movement phase	3.1a Movement during transport –road/rail					
		Exposure time (h)	**5**	**5**	**5**	**5**
		Distance (m)	**5**	**5**	**5**	**5**
	3.1b Movement during transport – air					
		Exposure time (h)	**50**	**5**	**-**	**-**
		Distance (m)	**1.5**	**1.5**	**-**	**-**
	3.1c Movement during transport – sea					
		Exposure time (h)	**250**	**250**	**250**	**-**
		Distance (m)	**10**	**10**	**10**	**-**
	3.2 Unforeseen interruptions					
		Exposure time (h)	**10**	**10**	**10**	**10**
		Distance (m)	**10**	**10**	**10**	**10**
	3.3 Regular stop					
		Exposure time (h)	**10**	**10**	**250**	**250**
		Distance (m)	**5**	**5**	**50**	**50**
4. Transfers during transport	4.1 Unloading (incl. Sub-steps) from conv. #1					
	4.1.1 Dose rate & contam. Meas.					
		Exposure time (minutes per conveyance)	n/a	**5**	**5**	**20**
		Distance (m)	**50**	**50**	**50**	**50**
	4.1.2 unfixing, fastening, lifting					
		Exposure time (minutes per package)	**1**	**7**	**15**	**30**
		Distance (m)	**50**	**50**	**50**	**50**
	4.2 Loading (incl. Sub-steps) on conv. #2					
	4.2.1 fixing					
		Exposure time (minutes per package)	**1**	**7**	**15**	**30**
		Distance (m)	**50**	**50**	**50**	**50**
	4.2.2 Dose rate meas. at conveyance					
		Exposure time (minutes per conveyance)	**5**	**10**	**5**	**10**
		Distance (m)	**50**	**50**	**50**	**50**
	4.2.3 Contamination meas. Of conveyance					
		Exposure time (minutes per conveyance)	**0.1**	**5**	**5**	**10**
		Distance (m)	**50**	**50**	**50**	**50**
	4.2.4 Placarding of conveyance					
		Exposure time (minutes per conveyance)	**2**	**2**	**2**	**5**
		Distance (m)	**50**	**50**	**50**	**50**
	4.3 *Regular stops*					
		Exposure time (minutes per conveyance)	**100**	**150**	**150**	**20**
		Distance (m)	**50**	**50**	**50**	**30**
5. Receiving inspection	5.1 Visual inspection (conveyance/package)			n/a	n/a	n/a
		Exposure time (minutes per conveyance)	**0.2**	-	-	-
		Distance (m)	**10**	-	-	-
	5.2 Dose rate measurement, conveyance		n/a	n/a	n/a	n/a
	5.3 Unfixing, fastening, lifting, unloading			n/a	n/a	n/a
		Exposure time (minutes per package)	**0.5**			
		Distance (m)	**10**			
	5.4 Transfer from conveyance to consignee			n/a	n/a	n/a
		Exposure time (minutes per package)	**0.5**			
		Distance (m)	**5**			
	5.5 Dose rate measurement, package		n/a	n/a	n/a	n/a
	5.6 Contamination meas. Package		n/a	n/a	n/a	n/a
	5.7 Contamination meas. Conveyance		n/a	n/a	n/a	n/a

The exposure of members of the public from transport of packages in the different modes and stages of transport are described below.

6.3.4.3. Step 3.1a - Road and Rail

Members of the public may be exposed to surface contamination during the movement phase of transport and this is covered by Step 3.1a. During transport by road there are normally no passengers who could be exposed. However, a vehicle could be parked for a few hours in a public car park close to members of the public or due to traffic congestion, members of public in their own vehicle, could be in close proximity to a vehicle containing radioactive materials for several hours. These situations are not likely to affect a member of the public more than once a year for a few hours. An exposure time of 5 h at a distance of 5 m was assumed for this scenario. It should be noted that Steps 3.2 and 3.3 cover situations where an unforeseen, or a regular interruption is made in a journey, and regular public exposures at a transfer site is included in Step 4.

Packages containing radioactive materials are transported by rail and members of the public may be exposed at distances that are subject to the regulatory requirements for segregation of packages from occupied areas. It is assumed that this would be an infrequent scenario for an individual member of the public and the same parameters as for road have been chosen; that is, 5 h at 5 m. As in the case of road, no shielding was assumed in the calculations of direct exposure from the surface contamination. However, the inhalation pathway was not included as the packages are assumed to be within an enclosed conveyance.

6.3.4.4. Step 3.1b - Air

On cargo aircraft, the Unit Loaded Devices (ULDs) containing packages carrying radioactive materials are always located away from the flight deck so as to reduce doses to the flight crew. Therefore, flight crew doses are negligible. The majority of radioactive materials are transported via cargo airplanes, rather than passenger airplanes. Therefore, the likelihood of a member of the public flying in the same aircraft as the radioactive materials is relatively low. Also, on passenger aircraft it is assumed that exposure of air crew would be less than that of members of the public who could be closer to the packages in the hold for longer periods of time. Cabin crew members do not remain in the same seat for the duration of a journey, therefore the exposure distances are likely to be much longer for the majority of the flight time than for the most exposed members of the public. For the purposes of this model, the most exposed passenger is assumed to be a member of public on a long haul flight in a seat directly above the package. This scenario is covered in the Advisory Material for the Regulations TS-G-1.1 [6.1]; the person-package distances used in that scenario are reflected in this study.

The model used to establish segregation distances of packages from occupied areas assumes an exposure time of 500 h. However, this represented conditions of a few decades ago, and current Radioactive Traffic Factors are considerably lower than assumed in that model [6.2]. Consequently, an exposure time of 50 h was assumed for this scenario. The minimum distance of a passenger from the closest package in the hold is assumed to be 1.5 m. These parameters were applied to the air transport of SM packages. The transport of SR packages is much less frequent and the exposure time is assumed to be by a further factor ten lower, that is 5 h.

Information from a UK air company suggests that air does not tend to be circulated through the cargo hold, and any air that does is filtered prior to being recirculated in the passenger cabin. HEPA filters with at least 99.9% filtration capability are used on the majority of passenger aircraft. Therefore it is very unlikely that inhalation of resuspended material will occur. The Boeing website [6.3] states that the pressurised air in the cabin comes from the compressor stages in the aircraft's jet engines. About half of the air inside the cabin is exhausted from the plane on each cycle of the air conditioning units. The remaining air is drawn by fans through special filters under the cabin floor and is then mixed with fresh outside air coming in from the engine compressors. In addition, packages are usually transported in Unit Loaded Devices (ULDs) within the aircraft which will reduce the fraction of material that is

dispersed throughout the cargo hold. For these reasons, it was agreed that the inhalation pathway can be ignored for this step.

6.3.4.5. Step 3.1c – Sea

Passengers and crew are unlikely to come into direct contact with packages containing radioactive materials while on board a ship. However, they may be in public areas in the vicinity of vehicles carrying packages, and this scenario is covered by step 3.1c. Only external exposure is considered for this scenario, as the radioactive materials are likely to be stored within vehicles on the ship. As crew are not radiation workers and unlikely to know that there are radioactive materials carried onboard these workers have been grouped with members of the public. A maximum exposure time of 250 h/year at 10 m has been assumed for these people. Fuel flasks are transported in dedicated ships and workers on the ships carry out a number of checking and monitoring operations. No passengers are carried on these ships.

6.3.4.6. Step 3.2 – Unforeseen interruptions

This scenario is intended to include breakdowns of the conveyance or other infrequent events. It is assumed that a member of the public would not be exposed for more than 10 h a year at 10 m, which applies to all package types.

6.3.4.7. Step 3.3 – Regular stop

This scenario is intended for temporary stops such as for refuelling, traffic lights and meal breaks. In the case of LR and FF packages, it also includes marshalling and handling of the packages in areas where the public may have close access. These situations are similar to the type of public exposure included in Step 4, but that step is specifically for sites that are not in a public area.

6.3.4.8. Step 4 – Exposure of members of the public close to a transfer site

The sites referred to in these operations are not public areas. Members of the public are assumed to be exposed outside the perimeter of the site at a distance of at least 50 m. The exposure times per package or conveyance are the same as for workers. For regular stops involving FF packages, a site has been identified where the closest member of the public may be at 30 m and this distance was assumed in this case.

6.3.4.9. Step 5 – Package reception

In almost all circumstances, packages are received into the consignee's premises, where there is no public access. However, the delivery of SM packages to a hospital could result in the temporary exposure of members of the public, such as patients or non-radiation workers, as indicated in Steps 5.1, 5.3 and 5.4. A short exposure time is assumed, at a distance of 10 m for Steps 5.1 and 5.3 and 5 m for Step 5.4.

6.4. Exposure pathways

6.4.1. Pathways of importance

The following pathways are used to assess doses to transport workers and members of the public from operations involving the shipment of contaminated packages. The pathways of importance to workers are: skin contamination, inadvertent ingestion following skin contamination, inhalation of resuspended activity and external exposure to contamination on the package and on the ground.

Members of the public should not come into direct contact with contaminated packages, therefore, the only pathways that need to be considered here are the inhalation of resuspended activity and external exposure from the contamination on the surface of the package and ground contamination.

6.4.2. Skin contamination

6.4.2.1. Dose due to skin contamination

Skin contact with the contamination will lead to a dose to the basal layer of skin. As the person is assumed to be in contact with the contaminated packages, it is assumed that there will be a transfer of non-fixed contamination to the hands and wrists, and further transfer to the face. In the limiting case, it is assumed that the persons would not be wearing gloves, but would wash their hands prior to leaving work as a normal behaviour.

6.4.2.2. Equivalent skin dose from contamination

In the case of hand contact with contaminated packages, the equivalent dose to this exposed skin is calculated using:

$$H_{skin} = A_C \times f \times T \times (\beta_{skin} + \gamma_{skin})$$

where:

H_{skin} = annual equivalent dose to skin (Sv/a),

A_c = activity per unit area of package (Bq/cm²)

T = exposure time (h/a),

f = fraction of the activity on the package transferred to the hands, assumed to be 20%

β_{skin} = skin equivalent dose rate to the basal layer of skin epidermis for beta irradiation (Sv/h/ (Bq/cm²)), dependent on skin thickness

γ_{skin} = skin equivalent dose rate to the basal layer of skin epidermis for gamma irradiation (Sv/h/ (Bq/cm²)), dependent on skin thickness

For unit activity concentrations, the annual skin equivalent dose is presented in the units Sv/a per Bq/cm²

6.4.2.3. Calculation of the effective skin dose from contamination

Assuming that outstretched hands have an area of 200 cm^2 on each side, and that contamination would only occur to the palm side, following the touching or carrying a package, 400 cm^2 of 400 μm [6.4] thick skin (palm side) will be contaminated. In the calculation of effective dose, the hands are assumed to be contaminated for 4 hours per day, on each day that packages are handled. When assessing doses to the listed workers, an exposure time of 8 h (2 periods of 4 h) was used in cases where the worker exposure times are longer than about 1000 h/a. This assumes that a worker's skin is contaminated at the start of the working day, is washed off during a mid-day break, and re-contaminated during the afternoon.

The skin equivalent dose resulting from hand contamination is given by:

$$H_{skin\ hands} = A_c \times f \times T \times (\beta_{skin} + \gamma_{skin})$$

where:

A_c = level of contamination, assumed to be 1 Bq/cm^2

f = fraction transferred, assumed to be 20%

T = time exposed, assumed to be 4 h/day, for each day that packages are handled.

β_{skin} = Skin equivalent dose rate to the basal layer of skin epidermis for beta irradiation for 400 μm thick skin, Sv/h per Bq/cm^2

γ_{skin} = Skin equivalent dose rate to the basal layer of skin epidermis for gamma irradiation, Sv/h per Bq/cm^2

The annual effective dose from skin contamination due to handling a contaminated package is given by:

$$E_{skin} = H_{skin_{hands}} \times w_{skin} \times \frac{area\ exposed}{total\ area}$$

where:

w_{SKIN} = tissue weighting factor for skin, 0.01 [6.5]

area exposed = area of skin contaminated, here assumed to be 400 cm^2

total area = total area of exposed skin, 3000 cm^2 [6.5]

Other areas of skin, for example the face, could become contaminated indirectly from the contamination on the hands. Assuming 4% (20% of 20%) of the activity level on the surface of the package is transferred from the hands to the face, and the affected area of face is 100 cm^2, approximately the area of one cheek. The frequency and efficiency of face washing is much lower than that of hand washing, most people wash their face only once per day, usually in the morning, leading to an exposure time of 12 to 24 hours per contamination event. Here, an average exposure time of 8 hours has been assumed, taking into account more frequent washing due to possibly dirty conditions, worker awareness of a potential contamination hazard and appropriate cleanliness practices. This estimated time may not always be conservative, but, together with the assumption of daily occurrence, will not lead to an under-estimate of skin dose.

The average annual skin equivalent dose resulting from face contamination is given by:

$$H_{skin_{face}} = \chi \times f \times T \times (\beta_{skin} + \gamma_{skin})$$

where χ = activity of contamination available for transfer to the face:

$$\chi = A_c \times f_{face}$$

where:

A_c = level of contamination, assumed to be 1 Bq/cm^2

f = fraction transferred to the hands, assumed to be 20%

f_{face} = fraction transferred from the hands to the face, assumed to be 20%

T = time exposed, assumed to be 8 h/day, on each day that packages are handled.

β_{skin} = Skin equivalent dose rate to the basal layer of skin epidermis for beta irradiation for 400 μm thick skin, Sv/h per Bq/cm^2

γ_{skin} = Skin equivalent dose rate to the basal layer of skin epidermis for gamma irradiation, Sv/h per Bq/cm^2

The annual effective dose from contamination of the face due to the transfer of contamination from the hands to the face is given by:

$$E_{SKIN} = H_{SKIN_{FACE}} \times w_{SKIN} \times \frac{area\ \ exposed}{total\ \ area}$$

where:

w_{SKIN} = tissue weighting factor for skin, 0.01 [6.5]

area exposed = area of skin contaminated, here assumed to be 100 cm^2

total area = total area of exposed skin, 3000 cm^2 [6.5]

6.4.3. Ingestion

Normally, at a workplace where radioactive contamination is a possibility, measures are taken to prohibit eating, drinking and smoking (in the work areas). It will be assumed here that these measures have been taken.

It is assumed, therefore, that the only possible transfer of the contamination from the hands to the mouth is due to touching the lips and mouth, from where it is ingested. The fraction ingested is taken as 1% of the activity on the hands, which, assuming uniform contamination is equivalent to ingestion of 100% of the activity on two fingertips, assuming the surface area of the hands is 400 cm^2. The annual committed effective dose from ingestion of contamination following skin contamination from contact with non-fixed contamination on a package is given by:

$$E = A_c \times area \times f \times f_{ing} \times R_9 \times (events / year)$$

where:

A_c = level of contamination, assumed to be 1 Bq/cm^2

area = area of hands from which contamination is ingested, here assumed to be 4 cm^2

f = fraction of the activity on the package transferred to the hands, 20%

f_{ing} = fraction ingested of ingestion area, 100%

R_9 = ingestion dose coefficient, Sv/Bq [6.6]

events / year = number of occasions per year that the skin is contaminated[1]

Where decay products need to be considered, those with a half-life of less than 10 days were summed with that of the parent radionuclide. These are listed in ref. [6.7]. The dose coefficients for ingestion were obtained from the IAEA Basic Safety Standards [6.6], which are taken from the publications of ICRP. The dose coefficients corresponding to the default chemical species were chosen for both workers and members of the public.

[1] Assumed to occur once per day that packages are handled.

6.4.4. Inhalation

The conditions in the vicinity of a package cannot be specified with precision. Therefore, in this simple model only very basic assumptions are made. An indoor and an outdoor calculation of the air concentration has been produced. These models utilise the resuspension rate method of calculation.

The indoor air concentration model relies on the assumption that the resuspended activity will distribute evenly over the entire volume of a room, and that this occurs instantaneously. The chosen air concentration calculations are shown below. The airborne activity concentration (C_d) indoors is given by:

$$C_d = \frac{RR \times A \times A_c}{V \times f_{ex}} \times contain$$

where:

RR is the resuspension rate. The value chosen was 10^{-4} h^{-1}, based on the considerations of values from the literature given in Section 5.1. For tritium, the contamination is likely to be in the form of tritiated water which can resuspend (evaporate) from the surface at a much higher rate, and could disappear from the surface in a matter of hours. A resuspension rate of 10^{-1} h^{-1} was, therefore, chosen for tritium, while recognising that the uncertainty on this value is large.

A is the surface area of the package from which resuspension occurs, m^2

V is the room volume, m^3

f_{ex} is the air exchange rate of the room, 2/h.

contain is a containment factor, included for packages that are covered, this has a default value of 1% for covered packages, and 1 for uncovered packages.

In cases where the work takes place outside, additional assumptions are made concerning the mechanism of release. The activity concentration in the air outdoors is given by:

$$C_d = \frac{RR \times A \times A_c}{v \times Q} \times contain$$

where:

v is the average wind speed, 2 m/s

Q is the cross section of the volume into which the resuspended contamination will disperse, m^2.

This area, Q, is assumed to be 10 times the cross sectional area of the package. This method is only suitable for calculating doses to persons in close proximity to a package. It was assumed that the cross section around a package for the assessment of doses to members of the public would be 1.5, 2, 6 and 10 times greater than the cross section used above for distances of 5, 10, 30 and 50 metres respectively. These values are given in Table 8.

Table 8. Cross sections around package configurations

Person	**Distance** (m)	**Q load** (m^2)				
		Description	**SM**	**SR**	**LR**	**FF**
Worker	0-3	10 × package/load cross section	22.5	32	58	49
Public	5	15 × package/load cross section	33.75	48	87	73.5
Public	10	20 × package/load cross section	45	64	446	98
Public	30	60 × package/load cross section	135	192	348	294
Public	50	100 × package/load cross section	225	320	580	490

Inhalation will only occur following the resuspension of non-fixed contamination. It will be assumed that all of this resuspended activity will be available for inhalation, although there is a facility in the model to enter a scenario-specific respirable fraction. The exposure time for inhalation of resuspended activity from contaminated packages is up to 2000 hours for workers who always work in close proximity to the packages. In the case of remote handling, time spent in the vicinity of packages will be much reduced.

The annual committed effective dose from inhalation of resuspended activity from a contaminated package is given by:

$$E = C_d \times frac \times T \times INH \times R_{10}$$

where:

C_d = airborne concentration, Bq/m^3, as described above

frac = respirable fraction, scenario dependent, but is conservatively assumed to be 1 for all assessed scenarios

T = exposure time, h/a, scenario dependent

INH = breathing rate, m^3/h, 1.2 for workers and 0.9 for members of the public [6.8]

R_{10} = inhalation dose coefficient [6.6]

Where decay products needed to be considered, those with a half-life of less than 10 days were summed with that of the parent radionuclide. These are listed in ref. [6.9]. The dose coefficients for inhalation were obtained from the IAEA Basic Safety Standards [6.6], which are taken from the publications of ICRP. The dose coefficients corresponding to the default chemical type were chosen for both workers and members of the public. For workers the default values for 5 μm particles were used, and 1 μm in the case of the public.

6.4.5. External exposure

6.4.5.1. External exposure pathways

External exposure to the body due to contaminated packages can arise in a number of ways. Fixed and non-fixed contamination will cause external dose from the package surface. Non-fixed contamination could also fall or be washed off the package onto the ground, which would lead to ground-shine exposure, this is described in the following section. Resuspended non-fixed contamination could also lead to cloud-shine due to submersion in a cloud of resuspended activity, deposition of this

resuspended activity will lead to ground-shine. However, both of these pathways have been found to be insignificant exposure pathways.

The calculations of doses from external irradiation have been performed using MicroShield 5.05 [6.10] using the ICRP-38 nuclide data library [6.11]. This gives external dose rates from the gamma contribution of the contamination. Beta radiation is not taken into account since it is much less significant than the gamma contribution because of air absorption over the distances chosen. Doses from neutron emissions have not been included, but would only be relevant for the three neutron emitting nuclides ^{248}Cm, ^{252}Cf, and ^{254}Cf.

6.4.5.2. Ground-shine

Ground-shine will occur because of direct transfer of contamination from the package to the ground. Ground deposition of resuspended activity will also occur, causing contamination to be spread around the ground in the vicinity of the package. However, this mechanism results in much lower levels of contamination than from direct transfer, as shown in section 6.4.7.

Direct transfer of radioactive material from a contaminated package to the ground may occur, for example, by heavy particles falling under gravity to the ground. Other mechanisms may involve transfer by rainwater onto the ground. It is assumed that the ground around an area where contaminated packages are handled, for example at a loading/transfer point, may become contaminated to 10% of the activity level on the package.

The annual dose from ground-shine was calculated as follows:

$$E = T \times DE / 1000$$

where:

T = exposure time h/a

DE = Effective dose equivalent rate in rotational geometry, calculated using MicroShield 5.05 [6.10], mSv/h for an activity concentration of 0.1 Bq/cm^2.

6.4.5.3. External exposure from contamination on the package

Regulations [6.12] state that the maximum surface dose rate of a package is 2 mSv/h, except for packages under exclusive use where the maximum radiation level on any surface of a package shall not exceed 10 mSv/h. Where all of this radiation is due to fixed contamination on the package surface, the dose to workers and the public is restricted by segregating packages from occupied areas. Segregation distances are calculated using dose criteria of 5 mSv/a for workers and 1 mSv/a for members of the public.

Doses to individuals from non-fixed contamination are limited by non-fixed contamination activity limits. The external exposure from non-fixed contamination is evaluated as described below.

External exposure to uniform contamination of a package would result in a dose given by:

$$E = A_c \times T \times DE / 1000$$

where:

A_c = activity concentration on package, 1 Bq/cm^2, scaled to Bq/m^2

T = exposure time h/a

DE = Effective dose equivalent rate in rotational geometry, calculated using MicroShield 5.05 [6.10], mSv/h

1000 = conversion from mSv/h to Sv/h

6.4.6. Geometries for external exposure

6.4.6.1. Types of packages

This section compiles the exposure geometries for which dose coefficients for external irradiation are calculated. The exposure geometries depend on the type of package (small manual, small remote, large remote and fuel flask) as well as on the sub-step (working conditions). These geometries are presented in the following sections grouped according to package type.

6.4.6.2. Small manually handled packages

For the small packages that are manually handled three configurations are used:

- single package:
- 10 packages in 2 rows of 5 (used for small van with 25 packages):
- 25 packages in 5 rows of 5 (used for large van with 100 packages):

The exposure geometries and the relation between exposure geometries and sub-steps of the model are given in Fig. 2.

Table 9 and Table 10 provide the details of the exposure geometries of small manually handled packages for workers and the public respectively.

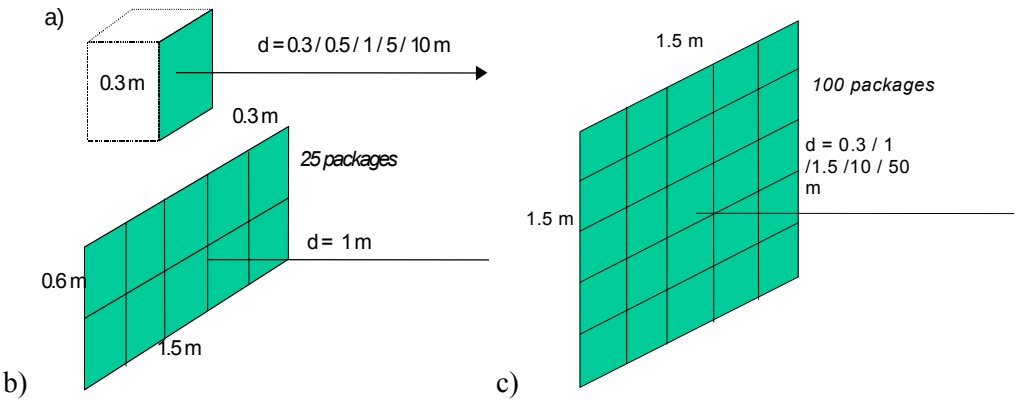

Figure 2. Exposure geometries for package type 'small manually handled'.

Table 9. Exposure geometries of small manually handled packages and sub-steps (worker)

No. of packages	Area of Exposure	Distance d	For sub-steps	Figure ref.
1	0.3 × 0.3 m²	0.3 m	5.4	2 a
1	0.3 × 0.3 m²	0.5 m	1.3, 1.4	2 a
1	0.3 × 0.3 m²	1 m	1.1, 1.2	2 a
25, visible 2 × 5	0.6 × 1.5 m²	1 m	3.1	2 b
100, visible 5 × 5	1.5 × 1.5 m²	0.3 m	2.1, 2.2, 4.1.2, 4.2.1, 5.3	2 c
100, visible 5 × 5	1.5 × 1.5 m²	1 m	2.3, 2.4, 2.5, 3.2, 4.2.2, 4.2.3, 4.2.4, 5.1, 5.5	2 c

Table 10. Exposure geometries of small manually handled packages and sub-steps (public)

No. of packages	Area of Exposure	Distance d	For sub-steps	Figure ref.
1	0.3 × 0.3 m²	5 m	Public 5.4,	2 a
1	0.3 × 0.3 m²	10 m	Public 5.1, 5.3	2 a
100, visible 5 × 5	1.5 × 1.5 m^2	1.5 m	Public 3.1b	2 c
100, visible 5 × 5	1.5 × 1.5 m^2	5 m	Public 3.1a, 3.3	2 c
100, visible 5 × 5	1.5 × 1.5 m^2	10 m	Public 3.1c, 3.2	2 c
100, visible 5 × 5	1.5 × 1.5 m^2	50 m	Public 4.1.1,4.1.2, 4.2.1 4.2.2, 4.2.3, 4.2.4, 4.3	2 c

6.4.6.3. Small remotely handled packages

For the small packages that are handled remotely (200 l drums) three configurations are used:

- single drum:
- 42 packages in 2 layers of 3 times 7 packages (on a van carrying the 42 packages without overpack):
- 15 packages in one layer of 5 times 3 for transport by air.

The exposure geometries and the relation between exposure geometries and sub-steps of the model are given in Figure 3.

Table 11 and Table 12 provide the details of the exposure geometries of small remotely handled packages for workers and the public respectively. Rectangular areas approximate the exposure geometry for the array of drums, the size of the (projected) surface. This is equivalent to the dose rate calculated from an array of cylindrical objects if the contribution of the back surface is reduced by shielding (wall material of the waste and material contained in the drum).

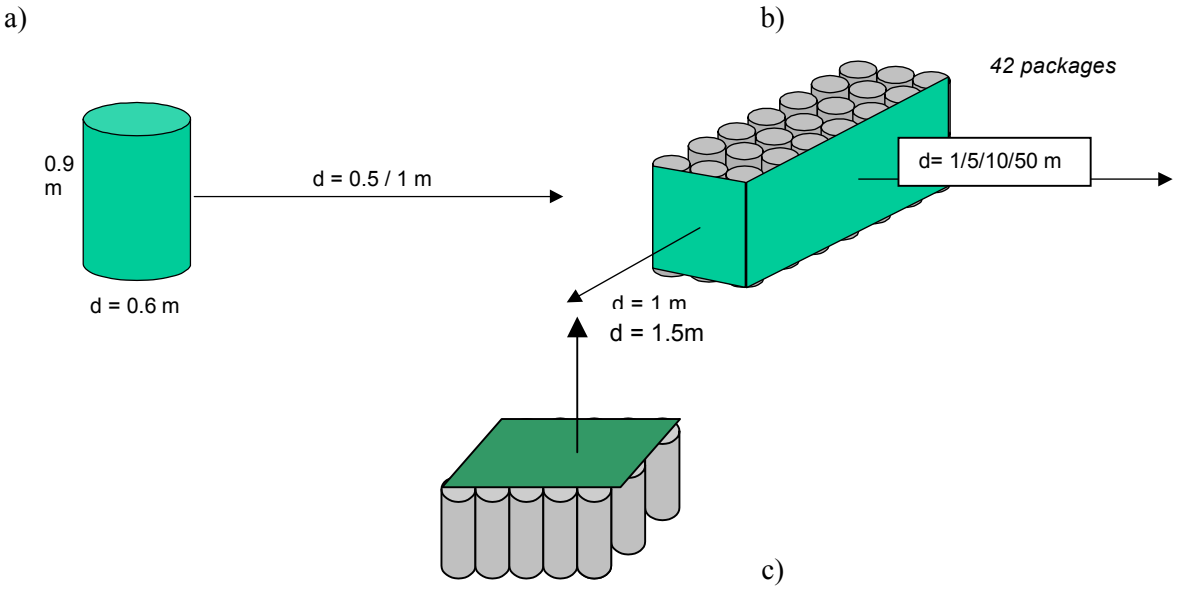

Figure 3. Exposure geometries for package type 'small remote'.

Table 11. Exposure geometries of small remotely handled packages and sub-steps (workers)

No. of packages	Area of Exposure	Distance d	For sub-steps	Figure ref.
1	Cyl.: ∅ = 0.6 m, h = 0.9 m	0.5 m	1.3, 1.4, 5.6	Figure 3 a
1	Cyl.: ∅ = 0.6 m, h = 0.9 m	1 m	1.1, 1.2, 2.1, 5.4, 5.5	Figure 3 a
42 (14 visible)	rectangle. area 1.8 × 4.2 m²	1 m	2.2, 2.3, 2.4, 2.5, 3.2, 4.1.1, 4.1.2, 4.2.1, 4.2.2, 4.2.3, 4.2.4, 5.1, 5.3, 5.7	Figure 3 b
42 (6 visible)	rectangle. area 1.8 × 1.8 m²	1 m	3.1	Figure 3 b

Table 12. Exposure geometries of small remotely handled packages and sub-steps (public)

No. of packages	Area of Exposure	Distance d	For sub-steps	Figure ref.
15	rectangle area 1.8 × 3 m²	1.5 m	Public 3.1b	3 c
42 (14 visible)	rectangle area 1.8 × 4.2 m²	5 m	Public 3.1a, 3.3	3 b
42 (14 visible)	rectangle area 1.8 × 4.2 m²	10 m	Public 3.1c, 3.2	3 b
42 (14 visible)	rectangle area 1.8 × 4.2 m²	50 m	Public 4.1.1,4.1.2, 4.2.1 4.2.2, 4.2.3, 4.2.4, 4.3	3 b

6.4.6.4. Large remotely handled packages

For the large packages that are handled remotely (20' container) exposure by only a single package is taken into account. Exposure may take place from the large side or the small side.

The exposure geometries and the relation between exposure geometries and sub-steps of the model are given in Fig. 13

Table 13 and Table 14 provide the details of the exposure geometries of large remotely handled packages for workers and the public respectively.

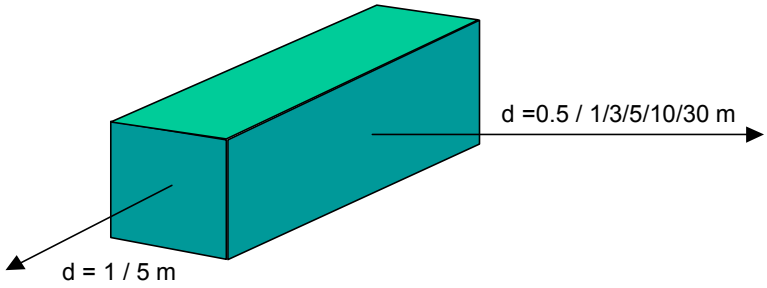

Figure 4. Exposure geometries for package type "large remote" and "fuel flask".

Table 13. Exposure geometries for package type “large remote” and sub-steps (worker)

No. of packages	Area of Exposure	Distance d	For sub-steps	Figure ref.
1 (large side)	2.4 × 5.9 m²	0.5 m	1.3, 1.4, 2.4, 5.6	4
1 (large side)	2.4 × 5.9 m²	1 m	1.1, 1.2, 2.1, 2.2, 2.3, 2.5, 3.2, 4.1.1, 4.1.2, 4.2.1, 4.2.2, 4.2.3, 4.2.4, 5.1, 5.3, 5.4, 5.5, 5.7	4
1 (small side)	2.4 × 2.4 m²	1 m	3.1	4

Table 14. Exposure geometries for package type “large remote” and sub-steps (public)

No. of packages	Area of Exposure	Distance d	For sub-steps	Figure ref.
1 (large side)	2.4 × 5.9 m²	5 m	Public 3.1a	4
1 (large side)	2.4 × 5.9 m²	10 m	Public 3.1c, 3.2	4
1 (large side)	2.4 × 5.9 m²	50 m	Public 3.3, 4.1.1, 4.1.2, 4.2.1, 4.2.2 4.2.3, 4.2.4, 4.3	4

6.4.6.5. Model for fuel flask

The fuel flask uses similar exposure geometries as for the large remotely handled package.

The exposure geometries and the relation between exposure geometries and sub-steps of the model are given in figure 4. *Table 15* and *Table 16* provide the details of the exposure geometries of large remotely handled packages for workers and the public respectively. The geometry considered here includes the floor of the wagon, 2.7 x 6.2 m^2 .

Table 15. Exposure geometries for a fuel flask and sub-steps (worker)

No. of packages	Area of Exposure	Distance	For sub-steps	Figure ref.
1 (large side)	2.4 × 5.9 m²	0.5 m	1.3, 1.4, 5.6	4
1 (large side)	2.4 × 5.9 m²	1 m	1.1, 1.2, 2.2, 2.3, 2.4, 2.5, 3.2, 4.1.1, 4.1.2, 4.2.1, 4.2.2, 4.2.3, 4.2.4, 5.1, 5.2, 5.3, 5.5, 5.7,	4
1 (large side)	2.4 × 5.9 m²	3 m	2.1, 5.4,	4
1 (small side)	2.4 × 2.4 m²	5 m	3.1	4
Floor of wagon	2.7 × 6.2 m2	1 m	5.7	4

Table 16. Exposure geometries for a fuel flask and sub-steps(public)

No. of packages	Area of Exposure	Distance	For sub-steps	Figure ref
1 (large side)	2.4 × 5.9 m²	5 m	Public 3.1a	4
1 (large side)	2.4 × 5.9 m²	10 m	Public 3.2	4
1 (large side)	2.4 × 5.9 m²	30 m	Public 4.3	4
1 (large side)	2.4 × 5.9 m²	50 m	3.3, 4.1.1, 4.1.2, 4.2.1, 4.2.2, 4.2.3, 4.2.4	4

6.4.6.6. Ground-shine

For ground contamination from direct deposition, ground-shine is modelled assigning a rectangular area of 10 × 10 m². The person is assumed to be situated in the middle of this area. Dose rate is calculated at a point 1 m above ground. This geometry is depicted in Figure 5.

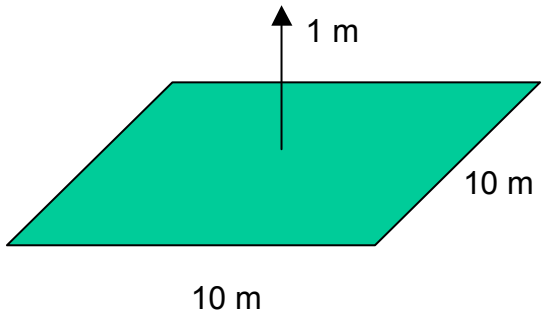

Figure 5. Exposure geometry for ground shine.

For the case of contamination from direct deposition on a flatrol (i.e. surface of a conveyance), the flatrol area is assumed as a rectangular area of 6.2 × 2.7 m². The person is again assumed to be situated in the middle of the area with the dose rate calculated at 1m above ground. This geometry is depicted in figure 6.

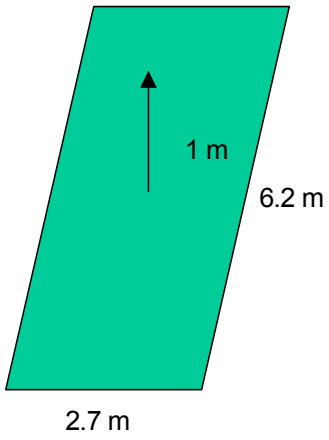

Figure 6. Exposure geometry for ground-shine – flatrol.

6.4.7. Minor pathways

There are other pathways that have been excluded from the study because of the very low doses that would be received following the exposure. They all result from resuspension of non-fixed contamination.

6.4.7.1. Cloud-shine

Resuspension of contamination will result in a 'cloud' of activity around the package as described earlier for the inhalation pathway. This resuspended activity would result in an annual dose to workers (or members of the public) given by:

$$E = \frac{C_d \times T \times [(R_1 \times CF_1) + (R_2 \times CF_2 \times w_{SKIN})]}{H}$$

where:

C_d = activity concentration in air (from section 6.4.4), Bq/m^3

T = exposure time, h/a

R_1 = the average photon energy per disintegration, MeV [6.4]

R_2 = the average beta energy per disintegration, MeV [6.4]

H = hours per year, 8760 h/a

CF_1 = dose rate from a semi-infinite cloud for 1 Bq/m^3 per MeV of gamma energy, 1.6 10^{-6} Sv/a per Bq/m^3 MeV [6.4]

CF_2 = skin equivalent dose rate from a semi-infinite cloud for 1 Bq/m^3 per MeV of beta energy, 2 10^{-6} Sv/a per Bq/m^3 MeV [6.4]

W_{SKIN} = tissue weighting factor for skin, 0.01 [6.13]

This pathway results in doses several orders of magnitude less than those from direct transfer (Section 6.4.5.2).

6.4.7.2. Deposition of resuspended activity onto the ground

Deposition can occur through two processes: wet deposition, due to precipitation falling through the cloud of resuspended activity (also known as washout) or dry deposition caused by cloud particles settling on to the underlying surface. Wet deposition will be ignored here as wet packages will have a much lower resuspension rate and so the deposition from that resuspended material will be negligible.

The rate at which material is dry-deposited from a cloud of resuspended activity is known as the deposition velocity, v_g, which is the ratio of the amount of material deposited on a surface per unit area per unit time to the air concentration per unit volume above the surface. It has a generic value of 10^{-3} m/s for all radionuclides except that the deposition velocity of organic iodine is higher, viz., 10^{-5} m/s and that of inorganic iodine is lower, viz., 10^{-2} m/s [6.14]. The dry deposition rate, D is given in Bq/s/m^2 by

$$D = V_g \times C$$

where C is the activity concentration in air at ground level.

The activity deposited onto the ground per unit area is therefore D×t, where t is the deposition time (s). As the activity concentration in air is dependent on the activity concentration on the package surface, and the resuspension factor, the activity deposited onto the ground (A_g) can be given in Bq/m² by:

$$A_g = D \times t = v_g \times t \times A_c \times f_{resus}$$

where A_c is the contamination activity concentration on the surface of the package scaled to Bq/m², and f_{resus} is the resuspension factor.

Therefore the effective dose to a person from deposited activity is given by:

$$E = A_g \times T \times (R_\beta + R_\gamma)$$

where:

T = exposure time h/a,

R_β = effective dose rate 1 metre above an infinite plane for beta radiation, Sv/h per Bq/m² [6.4]

R_γ = effective dose rate 1 metre above an infinite plane for gamma radiation, Sv/h per Bq/m². [6.4]

Ground-shine doses calculated from this pathway are several orders of magnitude lower than those resulting from the direct transfer of contamination onto the ground.

This pathway could be extended further to deposition onto crops or grazing land, which would result in a secondary ingestion pathway. Here the deposited quantity would be so low as to not pose a radiological hazard, which is discussed in Section 6.4.7.5 below.

6.4.7.3. Deposition of resuspended activity onto the skin

It is possible that if non-fixed contamination is resuspended, a fraction of the airborne activity could be deposited onto the skin.

The activity deposited on the skin is given by:

$$A_{skin} = D \cdot t \cdot area$$

where D is the deposition rate (Bq/s/m²), given by the deposition velocity (v) multiplied by the activity concentration in air. The deposition velocity is assumed be 10^{-3} m/s as in ground deposition, and the air concentration is calculated as described in section 6.4.4.

The deposition time, t (s), has been taken to be the length of time spent in the area, where deposition could occur. The area (m^2) is the area of skin contaminated, this has been assumed to be 1000 cm^2, or 0.1 m^2, the approximate area of the hands and face.

The effective dose from this pathway can be shown to be many orders of magnitude below those from the main exposure pathways.

6.4.7.4. Deposition of activity onto clothing

Activity could also be deposited onto the clothing, it is assumed that the clothing is worn for 5 hours following the deposition prior to taking the clothes off and washing them. Here a clothing shielding factor (CSF) for beta particles is required [6.13]. It has been assumed that the mass thickness of clothing (χ_m) is 0.1 g/cm², this thickness is between the values for winter and summer clothing, and is suitable for general work ware for time averaged calculations [6.13]. The clothing shielding factor is given by:

$$CSF = e^{-(17\,E^{-1.14})\chi m}$$

Where E is the maximum energy of the beta particle.

Therefore, for ^{90}Sr with a maximum energy of 0.196 MeV [6.11], the clothing shielding factor would be:

$$CSF = e^{-(17E^{-1.14})\chi_m}$$

$$CSF = e^{-(17(0.196^{-1.14})0.1}$$

$$CSF = 0.00002$$

Applying this value to the skin dose formulae used above, for a contaminated area of 3000 cm^2, the resulting equivalent and effective skin dose is 2.36×10^{-16} Sv, and 2.4×10^{-16} Sv respectively.

This pathway results in doses several orders of magnitude lower than the other pathways discussed in this methodology and can, therefore, be ignored.

6.4.7.5. Doses from deposition within the biosphere

Internal exposure doses to the general public were evaluated due to ingestion of agricultural and livestock products that were assumed to be contaminated by radionuclides resuspended from the package surface [6.15]. This model is also described in the Annex.

A model for dose evaluation was developed and calculations for a limited number of radionuclides were performed, based on a removable surface contamination level of 1 Bq/cm². Exposures of members of the public were evaluated for a number of age groups and these are shown in Table 17. The highest dose was 1.9×10^{-7} Sv/a per Bq/cm² to a 3 month old infant from ^{227}Ac. The maximum doses to adults were from ^{227}Ac and ^{210}Po, each at about 3.1×10^{-8} Sv/a per Bq/cm². These doses are given in Table 17 and are several orders of magnitude lower than those from the main exposure pathways considered.

Table 17. Doses from deposition within the biosphere for various age groups

Radionuclide	**Dose for each age (mSv/a)/(Bq/cm^2)**			
	Adult	**10 years old**	**1 year old**	**3 months old**
^{3}H	$1.8\text{x}10^{-09}$	$1.4\ \text{x}10^{-09}$	$1.5\ \text{x}10^{-09}$	$1.5\ \text{x}10^{-09}$
^{14}C	$1.8\text{x}10^{-08}$	$1.3\ \text{x}10^{-08}$	$1.2\ \text{x}10^{-08}$	$1.0\ \text{x}10^{-08}$
^{55}Fe	$9.4\ \text{x}10^{-09}$	$1.6\ \text{x}10^{-08}$	$1.4\ \text{x}10^{-08}$	$4.3\ \text{x}10^{-08}$
^{60}Co	$9.7\ \text{x}10^{-08}$	$1.6\ \text{x}10^{-07}$	$1.5\ \text{x}10^{-07}$	$3.1\ \text{x}10^{-07}$
^{90}Sr	$8.3\ \text{x}10^{-0}$	$9.1\ \text{x}10^{-07}$	$4.7\ \text{x}10^{-07}$	$1.5\ \text{x}10^{-06}$
^{99}Tc	$1.9\ \text{x}10^{-07}$	$2.0\ \text{x}10^{-07}$	$3.5\ \text{x}10^{-07}$	$7.3\ \text{x}10^{-07}$
^{131}I	$3.3\ \text{x}10^{-06}$	$3.0\ \text{x}10^{-06}$	$1.7\ \text{x}10^{-06}$	$1.4\ \text{x}10^{-06}$
^{137}Cs	$4.0\ \text{x}10^{-07}$	$1.6\ \text{x}10^{-07}$	$8.9\ \text{x}10^{-08}$	$1.6\ \text{x}10^{-07}$
^{210}Po	$3.1\ \text{x}10^{-05}$	$3.4\ \text{x}10^{-05}$	$4.6\ \text{x}10^{-05}$	$1.4\ \text{x}10^{-04}$
^{226}Ra	$8.1\ \text{x}10^{-06}$	$1.2\ \text{x}10^{-05}$	$5.7\ \text{x}10^{-06}$	$2.8\ \text{x}10^{-05}$
^{237}Np	$3.2\ \text{x}10^{-06}$	$1.6\ \text{x}10^{-06}$	$1.2\ \text{x}10^{-06}$	$1.1\ \text{x}10^{-05}$
^{238}U	$1.3\ \text{x}10^{-06}$	$9.9\ \text{x}10^{-07}$	$7.0\ \text{x}10^{-07}$	$2.0\ \text{x}10^{-06}$
^{239}Pu	$7.2\ \text{x}10^{-06}$	$3.9\ \text{x}10^{-06}$	$2.4\ \text{x}10^{-06}$	$2.4\ \text{x}10^{-05}$
^{241}Am	$5.7\ \text{x}10^{-06}$	$3.2\ \text{x}10^{-06}$	$2.1\ \text{x}10^{-06}$	$2.1\ \text{x}10^{-05}$
^{227}Ac	$3.1\ \text{x}10^{-05}$	$2.1\ \text{x}10^{-05}$	$1.8\ \text{x}10^{-05}$	$1.9\ \text{x}10^{-04}$

6.4.7.6. Public exposure from an extended area of contamination

It may be postulated that operations involving the shipment by rail of a large number of packages over an extended period of many years could lead to a build up of ground contamination beside the railway tracks. This is an extension of the scenario described in Section 6.4.5.2 (ground-shine pathway), that would arise from direct transfer of contamination to a long strip of ground. In that scenario it was assumed that 10% of the contamination on the package is transferred to the ground due to build up at a site where the package is stationary. Where the packages are in transit, there will be further dispersion of the material transferred to the ground and, therefore, the contamination will be less than at a stationary position, it is assumed that only 1% of the contamination on the package is transferred to the ground. It is assumed that over a period of time the contamination becomes fixed and therefore external radiation is the predominant pathway of exposure. It is postulated that a member of the public spends a large amount of time at a short distance from the contaminated strip. For example, a member of public living a short distance from the railway line.

Calculations were carried out, using Microshield 5.05 [6.10] for ^{137}Cs and ^{60}Co contamination at 0.01 Bq/cm² at two distances. The individual was assumed to be exposed at a central position from a 200 m length of track, of width 2 m. The following annual doses (Table 18) were calculated for an annual exposure time of 500 h.

Table 18. Annual doses for ^{60}Co and ^{137}Cs for the public from an extended area of contamination

Distance from track, (m)	**Annual dose,** (Sv)	
	^{60}Co	^{137}Cs
10	6.5×10^{-9}	1.4×10^{-9}
50	7.3×10^{-10}	1.3×10^{-10}

These annual doses are comparable to the public dose from road/rail movements of fuel flasks Table 22, step 3.1a, which are 6.2×10^{-9} Sv and 1.4×10^{-9} Sv from ^{60}Co and ^{137}Cs respectively. However, the assumptions are very conservative. It is assumed that the whole track is contaminated to a level of 0.01 Bq/cm², whereas, if there is any contamination, it is likely to be very localised. While this pathway is not included in the main pathways, the calculation indicates that railway tracks carrying packages that are potentially contaminated need to be subject to periodic monitoring.

6.5. Consideration of particulate contamination

The dose assessment methodology assumes that the contamination is spread approximately evenly over surface areas relevant to the chosen pathways. That is, it is assumed that over areas of at least a few hundred square centimetres the surface activity that gives rise to radiation exposure is proportional to the area under consideration. This would particularly apply if the contamination has arisen from water-soluble materials or consists of colloidal or very fine particulate material. However, radioactive contamination on the surface of packages and conveyances may also consist of discrete particles. The incidents involving the higher levels of non-fixed surface contamination on flasks and wagons that were reported in France and Germany in 1998 included occurrences of particles with activities up to a few thousand becquerel. Individual particle activities found on conveyances ranged up to a maximum of about 10^5 Bq, and some particles found in areas around reactor buildings ranged up to 10^6 Bq.

The occurrence of such particulate contamination is primarily associated with PWR and BWR reactors. The source of such particles is generally recognized to be the corrosion and wear of metal components in the primary cooling circuit of these reactors. These particles then become activated by neutron irradiation in the core. They can become attached to fuel elements and thus leave the reactor when the spent fuel is removed to the cooling ponds. These particles can subsequently attach to the surfaces of flasks immersed in the cooling ponds, and although most will be removed during the cleaning process, some may be lodged in crevices and surface imperfections, and so be missed during the preparation of the flask for transport. During shipment, the normal vibrations and movements of the flask could result in the particles becoming detached. It is understood that this was the main route by which such particles accumulated on the surfaces of the rail wagons used to transport these flasks.

The occurrence of high activity particles, or 'hot particles', at nuclear sites has come under greater scrutiny over the last two decades, partly because the introduction of more sensitive monitoring equipment has shown their potential for worker exposure [6.16]. High-activity particles, if in contact with the skin, or if taken up into the body, could give rise to high, localised, doses. If the activity is sufficiently high, there is a potential for deterministic radiological effects over small volumes of the body.

Studies on the radiological risks of hot particle exposure of the skin indicate that stochastic effects are much less important than deterministic effects such as ulceration [6.5]. For large field irradiation of the skin, the International Commission on Radiological Protection recommends a dose limit of

500 mSv, but this is not appropriate for very small volumes of tissue. Some studies have attempted to define an acceptable level of dose to such small volumes. A review of experimental work, on the induction of acute ulceration of the skin by hot particles, concluded that for point sources the threshold for deep ulceration of the skin is about 10^{10} beta particles [6.16], and this corresponds to a local dose of about 5 Sv [6.5]. If one beta particle is emitted from each radionuclide disintegration, this threshold may be expressed as 10 GBq.s [6.16]. The dose threshold for transient ulceration is about 1 Sv [6.5].

A 10^5 Bq particle on the skin could produce, from beta emission, transient ulceration in about 5 hours, and deep ulceration in about 25 hours. However, these are conservative exposure times as no account is taken of self shielding by the particle or shielding from clothing. If a worker picks up such a particle on the hands, it is likely to be removed by washing after a few hours, and therefore it is very unlikely that any deterministic effect will be noticed.

It would be possible for hot particles to be ingested and they, therefore, may give rise to elevated local doses to the gastrointestinal (GI) tract. However, a review of the exposure conditions concluded that hot particles do not represent an unusually high risk to the GI tract [6.16]. They are not expected to be absorbed due to their limited solubility, and would normally be eliminated from the body within 42 hours [6.17].

If a particle with an activity of 10^5 Bq of ^{60}Co in soluble form was ingested such that the activity was dissolved and absorbed into the body, an effective dose of 0.3 mSv may be received. Assuming that a worker would ingest such a particle as a very rare event, this dose would not be unacceptably high.

The Basic Model presented here assumes that the contamination is evenly spread and that the resuspended activity is respirable; that is, the particle size is small (10 μm or below). This takes full account of the inhalation dose from the surface activity and, therefore, is a cautious assumption. If the activity is distributed within larger particles (10 μm to 100 μm or more), the dose from inhalation will be lower. If the surface contamination consists mainly of large particles, including 'hot particles' the main hazard is from prolonged skin exposure, as discussed above. A full assessment of the hazard from such large particles would include probabilistic estimates of the pick-up of particles onto the skin, and the retention time on the skin. However, this is not developed here because, as noted above, the stochastic risk from such particles is not as important as deterministic effects and these effects are out of the scope of this study which was oriented on stochastic effects only.

6.6. Calculation methods

Four groups (NRPB, IRSN, WNTI and GRS) carried out calculations using the agreed upon Basic Model parameters, for 49 selected radionuclides. GRS carried out the calculations using a computer program in Pascal language, with data input and output files. The three other groups used spreadsheet methods, using Microsoft Excel. The designs of the calculation methods were different in each case and, therefore, provided some independence in each assessment.

Additional calculations have been carried out by IRSN to assess the dose impact from contamination by the 350 radionuclides that are included in the Regulations TS-R-1 (noble gases, which normally cannot be included in non-fixed contamination, have not been considered) [6.12].

6.7. Results and assessment

6.7.1. Numerical results

6.7.1.1. Results for selected radionuclides

Annual doses have been assessed for a surface activity of 1 Bq/cm² for each of the four package types for both workers and members of the public for 49 selected radionuclides. These results are presented in Table 19 to Table 22 for small manually handled packages, small remotely handled packages, large

remotely handled packages and fuel flasks respectively. Figures 7 to 10 show graphical representations of the data for each of the 49 radionuclides. A summary table has been produced showing the limiting dose, worker, pathway and package type for all 49 radionuclides. (Table 23). These data are also shown graphically in Figure 11. All the doses are expressed in (mSv y^{-1})/(Bq cm²)

Table 19. Small Manually Handled Packages

	Worker				Public				
Radionuclide	**Dose**	**Worker**	**Pathway**	**Pathway**	**Dose**	**Step**	**Pathway**	**Pathway**	**Limiting person**
^{227}Ac	$6.18x10^{+01}$	C	inh	100%	$1.59x10^{-02}$	4	inh	100%	Worker
^{241}Am	$2.68x10^{+00}$	C	inh	98%	$1.21x10^{-03}$	4	inh	100%	Worker
^{198}Au	$3.36x10^{-03}$	F	ext	74%	$1.85x10^{-05}$	3.1b	ext	100%	Worker
^{14}C	$1.97x10^{-04}$	C	ing	59%	$5.78x10^{-08}$	4	inh	100%	Worker
^{144}Ce	$4.50x10^{-03}$	C	inh	50%	$2.28x10^{-06}$	3.1b	ext	100%	Worker
^{252}Cf	$1.29x10^{+00}$	C	inh	99%	$5.78x10^{-04}$	4	inh	100%	Worker
^{36}Cl	$1.03x10^{-03}$	C	inh	48%	$2.11x10^{-07}$	4	inh	100%	Worker
^{242}Cm-	$3.64x10^{-01}$	C	inh	99%	$1.50x10^{-04}$	4	inh	100%	Worker
^{244}Cm	$1.69x10^{+00}$	C	inh	99%	$7.80x10^{-04}$	4	inh	100%	Worker
^{60}Co	$1.79x10^{-02}$	F	ext	82%	$1.10x10^{-04}$	3.1b	ext	100%	Worker
^{51}Cr	$2.24x10^{-04}$	F	ext	82%	$1.36x10^{-06}$	3.1b	ext	100%	Worker
^{134}Cs	$1.54x10^{-02}$	F	ext	64%	$7.35x10^{-05}$	3.1b	ext	100%	Worker
^{137}Cs	$6.58x10^{-03}$	F	ext	50%	$2.67x10^{-05}$	3.1b	ext	100%	Worker
^{154}Eu	$1.13x10^{-02}$	C	ext	44%	$5.87x10^{-05}$	3.1b	ext	100%	Worker
^{55}Fe	$1.07x10^{-04}$	C	ing	62%	$1.10x10^{-08}$	4	inh	100%	Worker
^{3}H	$1.76x10^{-03}$	C	inh	100%	$1.30x10^{-06}$	4	inh	100%	Worker
^{125}I-	$4.07x10^{-03}$	C	ing	74%	$2.85x10^{-06}$	3.1b	ext	100%	Worker
^{131}I	$8.08x10^{-03}$	C	ing	54%	$1.93x10^{-05}$	3.1b	ext	100%	Worker
^{192}Ir	$6.44x10^{-03}$	F	ext	80%	$3.82x10^{-05}$	3.1b	ext	100%	Worker
^{40}K	$2.94x10^{-03}$	C	ing	42%	$6.87x10^{-06}$	3.1b	ext	100%	Worker
^{90}Mo	$2.42x10^{-03}$	F	ext	70%	$1.26x10^{-05}$	3.1b	ext	100%	Worker
^{59}Ni	$3.42x10^{-05}$	C	inh	63%	$3.76x10^{-09}$	4	inh	100%	Worker
^{63}Ni	$8.13x10^{-05}$	C	inh	62%	$1.39x10^{-08}$	4	inh	100%	Worker
^{32}P	$1.30x10^{-03}$	C	HAND	49%	$2.22x10^{-08}$	4	inh	100%	Worker
^{210}Pb	$2.50x10^{-01}$	C	ing	54%	$3.44x10^{-05}$	4	inh	100%	Worker
^{210}Po	$1.17x10^{-01}$	C	inh	59%	$9.53x10^{-05}$	4	inh	100%	Worker
^{238}Pu	$2.98x10^{+00}$	C	inh	98%	$1.33x10^{-03}$	4	inh	100%	Worker
^{239}Pu	$3.18x10^{+00}$	C	inh	98%	$1.44x10^{-03}$	4	inh	100%	Worker
^{240}Pu	$3.18x10^{+00}$	C	inh	98%	$1.44x10^{-03}$	4	inh	100%	Worker
^{241}Pu	$5.76x10^{-02}$	C	inh	98%	$2.60x10^{-05}$	4	inh	100%	Worker
^{226}Ra	$2.82x10^{-01}$	C	inh	76%	$1.01x10^{-04}$	4	inh	100%	Worker
^{228}Ra	$3.03x10^{-01}$	C	inh	55%	$7.60x10^{-05}$	4	inh	100%	Worker
^{106}Ru	$4.43x10^{-03}$	C	ing	32%	$9.35x10^{-06}$	3.1b	ext	100%	Worker
^{35}S	$1.60x10^{-04}$	C	inh	67%	$4.04x10^{-08}$	4	inh	100%	Worker
^{124}Sb	$1.28x10^{-02}$	F	ext	81%	$7.75x10^{-05}$	3.1b	ext	100%	Worker
^{89}Sr	$1.32x10^{-03}$	C	HAND	45%	$1.76x10^{-07}$	4	inh	100%	Worker
^{90}Sr	$1.03x10^{-02}$	C	ing	59%	$1.08x10^{-06}$	4	inh	100%	Worker
^{99m}Tc	$9.27x10^{-04}$	F	ext	85%	$5.82x10^{-06}$	3.1b	ext	100%	Worker
^{129m}Te	$1.78x10^{-03}$	C	ing	34%	$4.18x10^{-06}$	3.1b	ext	100%	Worker
^{232}Th	$2.88x10^{+00}$	C	inh	98%	$7.22x10^{-04}$	4	inh	100%	Worker
^{232}Th (nat)	$5.69x10^{+00}$	C	inh	96%	$2.04x10^{-03}$	4	inh	100%	Worker
^{201}Tl	$6.64x10^{-04}$	F	ext	81%	$3.96x10^{-06}$	3.1b	ext	100%	Worker
^{232}U	$2.61x10^{+00}$	C	inh	97%	$1.07x10^{-03}$	4	inh	100%	Worker
^{233}U	$6.84x10^{-01}$	C	inh	99%	$2.77x10^{-04}$	4	inh	100%	Worker
^{234}U	$6.74x10^{-01}$	C	inh	99%	$2.72x10^{-04}$	4	inh	100%	Worker
^{235}U	$6.06x10^{-01}$	C	inh	98%	$2.46x10^{-04}$	4	inh	100%	Worker
^{236}U	$6.25x10^{-01}$	C	inh	99%	$2.51x10^{-04}$	4	inh	100%	Worker
^{238}U	$5.67x10^{-01}$	C	inh	98%	$2.31x10^{-04}$	4	inh	100%	Worker
^{95}Zr	$6.08x10^{-03}$	F	ext	82%	$3.73x10^{-05}$	3.1b	ext	100%	Worker

Table 20. Small Remotely Handled Packages

	Worker				Public				
Radionuclide	**Dose**	**Worker**	**Pathway**	**Pathway**	**Dose**	**Step**	**Pathway**	**Pathway**	**Limiting person**
^{227}Ac	$5.23 \times 10^{+00}$	A	inh	96%	5.43×10^{-02}	4	inh	100%	Worker
^{241}Am	2.55×10^{-01}	A	inh	84%	4.14×10^{-03}	4	inh	100%	Worker
^{198}Au	3.62×10^{-03}	F	ext	62%	7.21×10^{-06}	3.1c	ext	100%	Worker
^{14}C	1.45×10^{-04}	A	ing	80%	1.97×10^{-07}	4	inh	100%	Worker
^{144}Ce	2.35×10^{-03}	F	ing	45%	3.78×10^{-06}	4	inh	94%	Worker
^{252}Cf-	1.21×10^{-01}	A	inh	85%	1.97×10^{-03}	4	inh	100%	Worker
^{36}Cl-	5.77×10^{-04}	A	HAND	49%	7.21×10^{-07}	4	inh	100%	Worker
^{242}Cm	3.18×10^{-02}	A	inh	92%	5.13×10^{-04}	4	inh	100%	Worker
^{244}Cm	1.59×10^{-01}	A	inh	85%	2.66×10^{-03}	4	inh	100%	Worker
^{60}Co	1.84×10^{-02}	F	ext	73%	4.46×10^{-05}	3.1c	ext	100%	Worker
^{51}Cr	2.33×10^{-04}	F	ext	71%	5.23×10^{-07}	3.1c	ext	100%	Worker
^{134}Cs	1.58×10^{-02}	F	ext	57%	2.93×10^{-05}	3.1c	ext	100%	Worker
^{137}Cs	6.78×10^{-03}	F	ext	44%	9.63×10^{-06}	3.1c	ext	100%	Worker
^{154}Eu	1.01×10^{-02}	F	ext	71%	2.36×10^{-05}	3.1c	ext	100%	Worker
^{55}Fe	7.76×10^{-05}	A	ing	85%	3.75×10^{-08}	4	inh	100%	Worker
^{3}H	1.47×10^{-04}	A	inh	98%	4.44×10^{-06}	4	inh	100%	Worker
^{125}I	3.47×10^{-03}	F	ing	86%	8.80×10^{-07}	3.1c	ext	100%	Worker
^{131}I	7.75×10^{-03}	F	ing	57%	7.53×10^{-06}	3.1c	ext	100%	Worker
^{192}Ir	6.68×10^{-03}	F	ext	70%	1.49×10^{-05}	3.1c	ext	100%	Worker
^{40}K	2.91×10^{-03}	F	ing	43%	2.80×10^{-06}	3.1c	ext	100%	Worker
^{99}Mo	2.68×10^{-03}	F	ext	57%	2.96×10^{-06}	3.1c	ext	100%	Worker
^{59}Ni	1.44×10^{-05}	A	ing	87%	1.28×10^{-08}	4	inh	100%	Worker
^{63}Ni	3.46×10^{-05}	A	ing	87%	4.74×10^{-08}	4	inh	100%	Worker
^{32}P	1.20×10^{-03}	A	HAND	53%	7.60×10^{-08}	4	inh	100%	Worker
^{210}Pb	1.46×10^{-01}	A	ing	93%	1.17×10^{-04}	4	inh	100%	Worker
^{210}Po	5.36×10^{-02}	A	ing	89%	3.26×10^{-04}	4	inh	100%	Worker
^{238}Pu	2.84×10^{-01}	A	inh	84%	4.54×10^{-03}	4	inh	100%	Worker
^{239}Pu	3.04×10^{-01}	A	inh	84%	4.93×10^{-03}	4	inh	100%	Worker
^{240}Pu	3.04×10^{-01}	A	inh	84%	4.93×10^{-03}	4	inh	100%	Worker
^{241}Pu	5.55×10^{-03}	A	inh	83%	8.88×10^{-05}	4	inh	100%	Worker
^{226}Ra	8.07×10^{-02}	A	ing	69%	3.45×10^{-04}	4	inh	100%	Worker
^{228}Ra	1.50×10^{-01}	A	ing	90%	2.59×10^{-04}	4	inh	100%	Worker
^{106}Ru	3.83×10^{-03}	F	ing	37%	3.80×10^{-06}	4	inh	73%	Worker
^{35}S	6.08×10^{-05}	A	ing	46%	1.38×10^{-07}	4	inh	100%	Worker
^{124}Sb	1.33×10^{-02}	F	ext	71%	3.12×10^{-05}	3.1c	ext	100%	Worker
^{89}Sr	1.20×10^{-03}	A	HAND	50%	6.02×10^{-07}	4	inh	100%	Worker
^{90}Sr	7.47×10^{-03}	A	ing	82%	3.69×10^{-06}	4	inh	100%	Worker
^{99m}Tc	9.44×10^{-04}	F	ext	75%	2.16×10^{-06}	3.1c	ext	100%	Worker
^{129m}Te	1.67×10^{-03}	F	ing	36%	8.13×10^{-07}	4	inh	81%	Worker
^{232}Th	2.75×10^{-01}	A	inh	84%	2.47×10^{-03}	4	inh	100%	Worker
^{232}Th (nat)	6.61×10^{-01}	A	inh	67%	6.97×10^{-03}	4	inh	100%	Worker
^{201}Tl	6.78×10^{-04}	F	ext	71%	1.45×10^{-06}	3.1c	ext	100%	Worker
^{232}U	2.73×10^{-01}	A	inh	76%	3.65×10^{-03}	4	inh	100%	Worker
^{233}U	6.48×10^{-02}	A	inh	85%	9.47×10^{-04}	4	inh	100%	Worker
^{234}U	6.39×10^{-02}	A	Inh	85%	9.27×10^{-04}	4	inh	100%	Worker
^{235}U	5.84×10^{-02}	A	Inh	83%	8.39×10^{-04}	4	inh	100%	Worker
^{236}U	5.93×10^{-02}	A	Inh	84%	8.58×10^{-04}	4	inh	100%	Worker
^{238}U	5.49×10^{-02}	A	Inh	83%	7.89×10^{-04}	4	inh	100%	Worker
^{95}Zr	6.25×10^{-03}	F	ext	73%	1.49×10^{-05}	3.1c	ext	100%	Worker

Table 21. Large Remotely Handled Packages

		Worker				Public			
Radionuclide	**Dose**	**Worker**	**Pathway**	**Pathway**	**Dose**	**Step**	**Pathway**	**Pathway**	**Limiting person**
^{241}Am	3.49x10^{-01}	A	inh	88%	3.37x10^{-04}	4	inh	100%	Worker
^{198}Au	2.64x10^{-03}	A	ext	70%	1.33x10^{-05}	3.1c	ext	100%	Worker
^{14}C	1.47x10^{-04}	A	ing	79%	1.60x10^{-08}	4	inh	100%	Worker
^{144}Ce	2.10x10^{-03}	A	ing	50%	1.62x10^{-06}	3.1c	ext	100%	Worker
^{252}Cf-	1.67x10^{-01}	A	inh	89%	1.60x10^{-04}	4	inh	100%	Worker
^{36}Cl-	4.53x10^{-04}	A	ing	41%	5.88x10^{-08}	4	inh	99%	Worker
^{242}Cm	4.47x10^{-02}	A	inh	95%	4.17x10^{-05}	4	inh	100%	Worker
^{244}Cm	2.18x10^{-01}	A	inh	89%	2.16x10^{-04}	4	inh	100%	Worker
^{60}Co	1.37x10^{-02}	A	ext	81%	8.22x10^{-05}	3.1c	ext	100%	Worker
^{51}Cr	1.71x10^{-04}	A	ext	80%	9.64x10^{-07}	3.1c	ext	100%	Worker
^{134}Cs	1.26x10^{-02}	A	ext	58%	5.39x10^{-05}	3.1c	ext	100%	Worker
^{137}Cs	5.70x10^{-03}	A	ing	46%	1.77x10^{-05}	3.1c	ext	100%	Worker
^{154}Eu	7.86x10^{-03}	A	ext	75%	4.34x10^{-05}	3.1c	ext	100%	Worker
^{55}Fe	7.45x10^{-05}	A	ing	89%	3.04x10^{-09}	4	inh	100%	Worker
^{3}H	2.09x10^{-04}	A	inh	98%	3.61x10^{-07}	4	inh	100%	Worker
^{125}I	3.42x10^{-03}	A	ing	88%	1.62x10^{-06}	3.1c	ext	100%	Worker
^{131}I	6.94x10^{-03}	A	ing	63%	1.39x10^{-05}	3.1c	ext	100%	Worker
^{192}Ir	4.98x10^{-03}	A	ext	77%	2.74x10^{-05}	3.1c	ext	100%	Worker
^{40}K	2.40x10^{-03}	A	ing	52%	5.16x10^{-06}	3.1c	ext	100%	Worker
^{99}Mo	1.93x10^{-03}	A	ext	66%	5.44x10^{-06}	3.1c	ext	100%	Worker
^{59}Ni	1.52x10^{-05}	A	ing	83%	1.04x10^{-09}	4	inh	100%	Worker
^{63}Ni	3.64x10^{-05}	A	ing	82%	3.85x10^{-09}	4	inh	100%	Worker
^{32}P	8.86x10^{-04}	A	ing	54%	6.17x10^{-09}	4	inh	100%	Worker
^{210}Pb	1.50x10^{-01}	A	ing	91%	9.54x10^{-06}	4	inh	100%	Worker
^{210}Po	5.61x10^{-02}	A	ing	86%	2.64x10^{-05}	4	inh	100%	Worker
^{238}Pu	3.89x10^{-01}	A	inh	88%	3.69x10^{-04}	4	inh	100%	Worker
^{239}Pu	4.16x10^{-01}	A	inh	88%	4.01x10^{-04}	4	inh	100%	Worker
^{240}Pu	4.16x10^{-01}	A	inh	88%	4.01x10^{-04}	4	inh	100%	Worker
^{241}Pu	7.57x10^{-03}	A	inh	88%	7.21x10^{-06}	4	inh	100%	Worker
^{226}Ra	9.12x10^{-02}	A	ing	61%	5.75x10^{-05}	3.1c	ext	100%	Worker
^{228}Ra	1.54x10^{-01}	A	ing	87%	2.11x10^{-05}	4	inh	100%	Worker
^{106}Ru	3.11x10^{-03}	A	ing	45%	6.82x10^{-06}	3.1c	ext	100%	Worker
^{35}S	6.46x10^{-05}	A	ing	43%	1.12x10^{-08}	4	inh	100%	Worker
^{124}Sb	9.83x10^{-03}	A	ext	79%	5.75x10^{-05}	3.1c	ext	100%	Worker
^{89}Sr	9.05x10^{-04}	A	ing	57%	4.91x10^{-08}	4	inh	100%	Worker
^{90}Sr	7.11x10^{-03}	A	ing	86%	3.00x10^{-07}	4	inh	100%	Worker
^{99m}Tc	6.95x10^{-04}	A	ext	84%	3.98x10^{-06}	3.1c	ext	100%	Worker
^{129m}Te	1.28x10^{-03}	A	ing	48%	1.25x10^{-06}	3.1c	ext	100%	Worker
^{232}Th	3.76x10^{-01}	A	inh	88%	2.00x10^{-04}	4	inh	100%	Worker
^{232}Th (nat)	8.60x10^{-01}	A	inh	74%	5.66x10^{-04}	4	inh	100%	Worker
^{201}Tl	5.05x10^{-04}	A	ext	79%	2.66x10^{-06}	3.1c	ext	100%	Worker
^{232}U	3.63x10^{-01}	A	inh	82%	2.96x10^{-04}	4	inh	100%	Worker
^{233}U	8.89x10^{-02}	A	inh	89%	7.69x10^{-05}	4	inh	100%	Worker
^{234}U	8.76x10^{-02}	A	inh	89%	7.53x10^{-05}	4	inh	100%	Worker
^{235}U	7.99x10^{-02}	A	inh	87%	6.83x10^{-05}	4	inh	100%	Worker
^{236}U	8.13x10^{-02}	A	inh	89%	6.97x10^{-05}	4	inh	100%	Worker
^{238}U	7.44x10^{-02}	A	inh	88%	6.41x10^{-05}	4	inh	100%	Worker
^{95}Zr	4.65x10^{-03}	A	ext	80%	2.75x10^{-05}	3.1c	ext	100%	Worker

Table 22. Fuel Flasks

	Worker				Public				
Radionuclide	**Dose**	**Worker**	**Pathway**	**Pathway**	**Dose**	**Step**	**Pathway**	**Pathway**	**Limiting person**
^{227}Ac	$9.20x10^{+00}$	A	inh	98%	$8.57x10^{-03}$	4	inh	100%	Worker
^{241}Am	$4.25x10^{-01}$	A	inh	91%	$6.55x10^{-04}$	4	inh	100%	Worker
^{198}Au	$2.29x10^{-03}$	A	ext	66%	$1.03x10^{-06}$	3.1a	ext	100%	Worker
^{14}C	$1.48x10^{-04}$	A	ing	78%	$3.12x10^{-08}$	4	inh	100%	Worker
^{144}Ce	$2.12x10^{-03}$	A	ing	49%	$6.47x10^{-07}$	4	inh	87%	Worker
^{252}Cf-	$2.03x10^{-01}$	A	inh	91%	$3.12x10^{-04}$	4	inh	100%	Worker
^{36}Cl-	$4.68x10^{-04}$	A	ing	40%	$1.14x10^{-07}$	4	inh	100%	Worker
^{242}Cm	$5.52x10^{-02}$	A	inh	96%	$8.11x10^{-05}$	4	inh	100%	Worker
^{244}Cm	$2.66x10^{-01}$	A	inh	91%	$4.21x10^{-04}$	4	inh	100%	Worker
^{60}Co	$1.17x10^{-02}$	A	ext	78%	$6.25x10^{-06}$	3.1a	ext	100%	Worker
^{51}Cr	$1.46x10^{-04}$	A	ext	77%	$7.54x10^{-08}$	3.1a	ext	100%	Worker
^{134}Cs	$1.13x10^{-02}$	A	ext	54%	$4.14x10^{-06}$	3.1a	ext	100%	Worker
^{137}Cs	$5.27x10^{-03}$	A	ing	49%	$1.37x10^{-06}$	3.1a	ext	100%	Worker
^{154}Eu	$6.88x10^{-03}$	A	ext	70%	$3.46x10^{-06}$	4	ext	76%	Worker
^{55}Fe	$7.54x10^{-05}$	A	ing	88%	$5.92x10^{-09}$	4	inh	100%	Worker
^{3}H	$2.60x10^{-04}$	A	inh	99%	$7.01x10^{-07}$	4	inh	100%	Worker
^{125}I	$3.39x10^{-03}$	A	ing	89%	$1.49x10^{-07}$	3.1a	ext	96%	Worker
^{131}I	$6.62x10^{-03}$	A	ing	67%	$1.09x10^{-06}$	3.1a	ext	99%	Worker
^{192}Ir	$4.28x10^{-03}$	A	ext	74%	$2.13x10^{-06}$	3.1a	ext	100%	Worker
^{40}K	$2.28x10^{-03}$	A	ing	54%	$3.93x10^{-07}$	3.1a	ext	99%	Worker
^{99}Mo	$1.70x10^{-03}$	A	ext	61%	$7.02x10^{-07}$	3.1a	ext	100%	Worker
^{59}Ni	$1.58x10^{-05}$	A	ing	80%	$2.03x10^{-09}$	4	inh	100%	Worker
^{63}Ni	$3.79x10^{-05}$	A	ing	79%	$7.48x10^{-09}$	4	inh	100%	Worker
^{32}P	$8.89x10^{-04}$	A	ing	54%	$1.20x10^{-08}$	4	inh	100%	Worker
^{210}Pb	$1.53x10^{-01}$	A	ing	89%	$1.86x10^{-05}$	4	inh	100%	Worker
^{210}Po	$5.81x10^{-02}$	A	ing	83%	$5.14x10^{-05}$	4	inh	100%	Worker
^{238}Pu	$4.74x10^{-01}$	A	inh	90%	$7.17x10^{-04}$	4	inh	100%	Worker
^{239}Pu	$5.06x10^{-01}$	A	inh	90%	$7.79x10^{-04}$	4	inh	100%	Worker
^{240}Pu	$5.06x10^{-01}$	A	inh	90%	$7.79x10^{-04}$	4	inh	100%	Worker
^{241}Pu	$9.21x10^{-03}$	A	inh	90%	$1.40x10^{-05}$	4	inh	100%	Worker
^{226}Ra	$9.59x10^{-02}$	A	ing	58%	$5.50x10^{-05}$	4	inh	99%	Worker
^{228}Ra	$1.59x10^{-01}$	A	ing	84%	$4.10x10^{-05}$	4	inh	100%	Worker
^{106}Ru	$2.96x10^{-03}$	A	ing	47%	$8.20x10^{-07}$	4	inh	53%	Worker
^{35}S	$6.77x10^{-05}$	A	ing	41%	$2.18x10^{-08}$	4	inh	100%	Worker
^{124}Sb	$8.40x10^{-03}$	A	ext	76%	$4.39x10^{-06}$	3.1a	ext	100%	Worker
^{89}Sr	$9.09x10^{-04}$	A	ing	57%	$9.53x10^{-08}$	4	inh	100%	Worker
^{90}Sr	$7.20x10^{-03}$	A	ing	85%	$5.83x10^{-07}$	4	inh	100%	Worker
^{99m}Tc	$5.87x10^{-04}$	A	ext	82%	$3.18x10^{-07}$	3.1a	ext	100%	Worker
^{129m}Te	$1.25x10^{-03}$	A	ing	49%	$2.13x10^{-07}$	3.1a	ext	97%	Worker
^{232}Th	$4.57x10^{-01}$	A	inh	90%	$3.90x10^{-04}$	4	inh	100%	Worker
^{232}Th (nat)	$1.02x10^{+00}$	A	inh	79%	$1.10x10^{-03}$	4	inh	100%	Worker
^{201}Tl	$4.31x10^{-04}$	A	ext	76%	$2.15x10^{-07}$	3.1a	ext	100%	Worker
^{232}U	$4.37x10^{-01}$	A	inh	85%	$5.77x10^{-04}$	4	inh	100%	Worker
^{233}U	$1.08x10^{-01}$	A	inh	91%	$1.50x10^{-04}$	4	inh	100%	Worker
^{234}U	$1.07x10^{-01}$	A	inh	91%	$1.47x10^{-04}$	4	inh	100%	Worker
^{235}U	$9.70x10^{-02}$	A	inh	90%	$1.33x10^{-04}$	4	inh	100%	Worker
^{236}U	$9.90x10^{-02}$	A	inh	91%	$1.36x10^{-04}$	4	inh	100%	Worker
^{238}U	$9.05x10^{-02}$	A	inh	90%	$1.25x10^{-04}$	4	inh	100%	Worker
^{95}Zr	$3.97x10^{-03}$	A	ext	77%	$2.11x10^{-06}$	3.1a	ext	100%	Worker

Table 23. All package types

Nuclide	Worker					Public					Limiting Person
	Dose	Package	Worker	Pathway	Pathway	Dose	Package	Step	Pathway	Pathway	
^{227}Ac	6.18x10^{+01}	SM	C	inh	100%	5.43x10^{-02}	SR	4	inh	100%	Worker
^{241}Am	2.68x10^{+00}	SM	C	inh	98%	4.14x10^{-03}	SR	4	inh	100%	Worker
^{198}Au	3.62x10^{-03}	SR	F	ext	62%	1.85x10^{-05}	SM	3.1b	ext	100%	Worker
^{14}C	1.97x10^{-04}	SM	C	ing	59%	1.97x10^{-07}	SR	4	inh	100%	Worker
^{144}Ce	4.50x10^{-03}	SM	C	inh	50%	3.78x10^{-06}	SR	4	inh	94%	Worker
^{252}Cf-	1.29x10^{+00}	SM	C	inh	99%	1.97x10^{-03}	SR	4	inh	100%	Worker
^{36}Cl-	1.03x10^{-03}	SM	C	inh	48%	7.21x10^{-07}	SR	4	inh	100%	Worker
^{242}Cm	3.64x10^{-01}	SM	C	inh	99%	5.13x10^{-04}	SR	4	inh	100%	Worker
^{244}Cm	1.69x10^{+00}	SM	C	inh	99%	2.66x10^{-03}	SR	4	inh	100%	Worker
^{60}Co	1.84x10^{-02}	SR	F	ext	73%	1.10x10^{-04}	SM	3.1b	ext	100%	Worker
^{51}Cr	2.33x10^{-04}	SR	F	ext	71%	1.36x10^{-06}	SM	3.1b	ext	100%	Worker
^{134}Cs	1.58x10^{-02}	SR	F	ext	57%	7.35x10^{-05}	SM	3.1b	ext	100%	Worker
^{137}Cs	6.78x10^{-03}	SR	F	ext	44%	2.67x10^{-05}	SM	3.1b	ext	100%	Worker
^{154}Eu	1.13x10^{-02}	SM	C	ext	44%	5.87x10^{-05}	SM	3.1b	ext	100%	Worker
^{55}Fe	1.07x10^{-04}	SM	C	ing	62%	3.75x10^{-08}	SR	4	inh	100%	Worker
^{3}H	1.76x10^{-03}	SM	C	inh	100%	4.44x10^{-06}	SR	4	inh	100%	Worker
^{125}I	4.07x10^{-03}	SM	C	ing	74%	2.85x10^{-06}	SM	3.1b	ext	100%	Worker
^{131}I	8.08x10^{-03}	SM	C	ing	54%	1.93x10^{-05}	SM	3.1b	ext	100%	Worker
^{192}Ir	6.68x10^{-03}	SR	F	ext	70%	3.82x10^{-05}	SM	3.1b	ext	100%	Worker
^{40}K	2.94x10^{-03}	SM	C	ing	42%	6.87x10^{-06}	SM	3.1b	ext	100%	Worker
^{99}Mo	2.68x10^{-03}	SR	F	ext	57%	1.26x10^{-05}	SM	3.1b	ext	100%	Worker
^{59}Ni	3.42x10^{-05}	SM	C	inh	63%	1.28x10^{-08}	SR	4	inh	100%	Worker
^{63}Ni	8.13x10^{-05}	SM	C	inh	62%	4.74x10^{-08}	SR	4	inh	100%	Worker
^{32}P	1.30x10^{-03}	SM	C	HAND	49%	7.60x10^{-08}	SR	4	inh	100%	Worker
^{210}Pb	2.50x10^{-01}	SM	C	ing	54%	1.17x10^{-04}	SR	4	inh	100%	Worker
^{210}Po	1.17x10^{-01}	SM	C	inh	59%	3.26x10^{-04}	SR	4	inh	100%	Worker
^{238}Pu	2.98x10^{+00}	SM	C	inh	98%	4.54x10^{-03}	SR	4	inh	100%	Worker
^{239}Pu	3.18x10^{+00}	SM	C	inh	98%	4.93x10^{-03}	SR	4	inh	100%	Worker
^{240}Pu	3.18x10^{+00}	SM	C	inh	98%	4.93x10^{-03}	SR	4	inh	100%	Worker
^{241}Pu	5.76x10^{-02}	SM	C	inh	98%	8.88x10^{-05}	SR	4	inh	100%	Worker
^{226}Ra	2.82x10^{-01}	SM	C	inh	76%	3.45x10^{-04}	SR	4	inh	100%	Worker
^{228}Ra	3.03x10^{-01}	SM	C	inh	55%	2.59x10^{-04}	SR	4	inh	100%	Worker
^{106}Ru	4.43x10^{-03}	SM	C	ing	32%	9.35x10^{-06}	SM	3.1b	ext	100%	Worker
^{35}S	1.60x10^{-04}	SM	C	inh	67%	1.38x10^{-07}	SR	4	inh	100%	Worker
^{124}Sb	1.33x10^{-02}	SR	F	ext	71%	7.75x10^{-05}	SM	3.1b	ext	100%	Worker
^{89}Sr	1.32x10^{-03}	SM	C	HAND	45%	6.02x10^{-07}	SR	4	inh	100%	Worker
^{90}Sr	1.03x10^{-02}	SM	C	ing	59%	3.69x10^{-06}	SR	4	inh	100%	Worker
^{99m}Tc	9.44x10^{-04}	SR	F	ext	75%	5.82x10^{-06}	SM	3.1b	ext	100%	Worker
^{129m}Te	1.78x10^{-03}	SM	C	ing	34%	4.18x10^{-06}	SM	3.1b	ext	100%	Worker
^{232}Th	2.88x10^{+00}	SM	C	inh	98%	2.47x10^{-03}	SR	4	inh	100%	Worker
^{232}Th (nat)	5.69x10^{+00}	SM	C	inh	96%	6.97x10^{-03}	SR	4	inh	100%	Worker
^{201}Tl	6.78x10^{-04}	SR	F	ext	71%	3.96x10^{-06}	SM	3.1b	ext	100%	Worker
^{232}U	2.61x10^{+00}	SM	C	inh	97%	3.65x10^{-03}	SR	4	inh	100%	Worker
^{233}U	6.84x10^{-01}	SM	C	inh	99%	9.47x10^{-04}	SR	4	inh	100%	Worker
^{234}U	6.74x10^{-01}	SM	C	inh	99%	9.27x10^{-04}	SR	4	inh	100%	Worker
^{235}U	6.06x10^{-01}	SM	C	inh	98%	8.39x10^{-04}	SR	4	inh	100%	Worker
^{236}U	6.25x10^{-01}	SM	C	inh	99%	8.58x10^{-04}	SR	4	inh	100%	Worker
^{238}U	5.67x10^{-01}	SM	C	inh	98%	7.89x10^{-04}	SR	4	inh	100%	Worker
^{95}Zr	6.25x10^{-03}	SR	F	ext	73%	3.73x10^{-05}	SM	3.1b	ext	100%	Worker

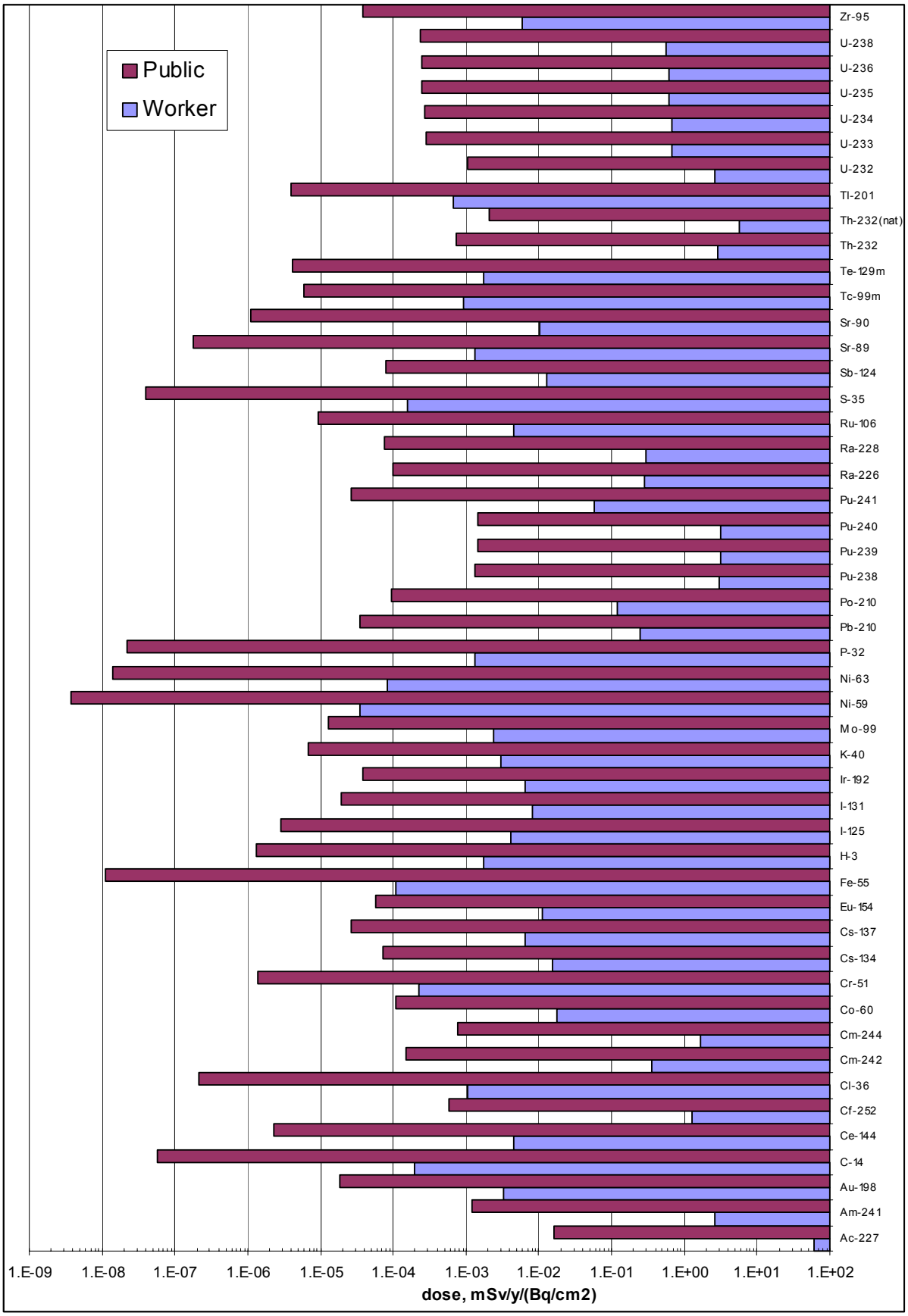

Figure 7. Worker and Public doses received from small manually handled packages.

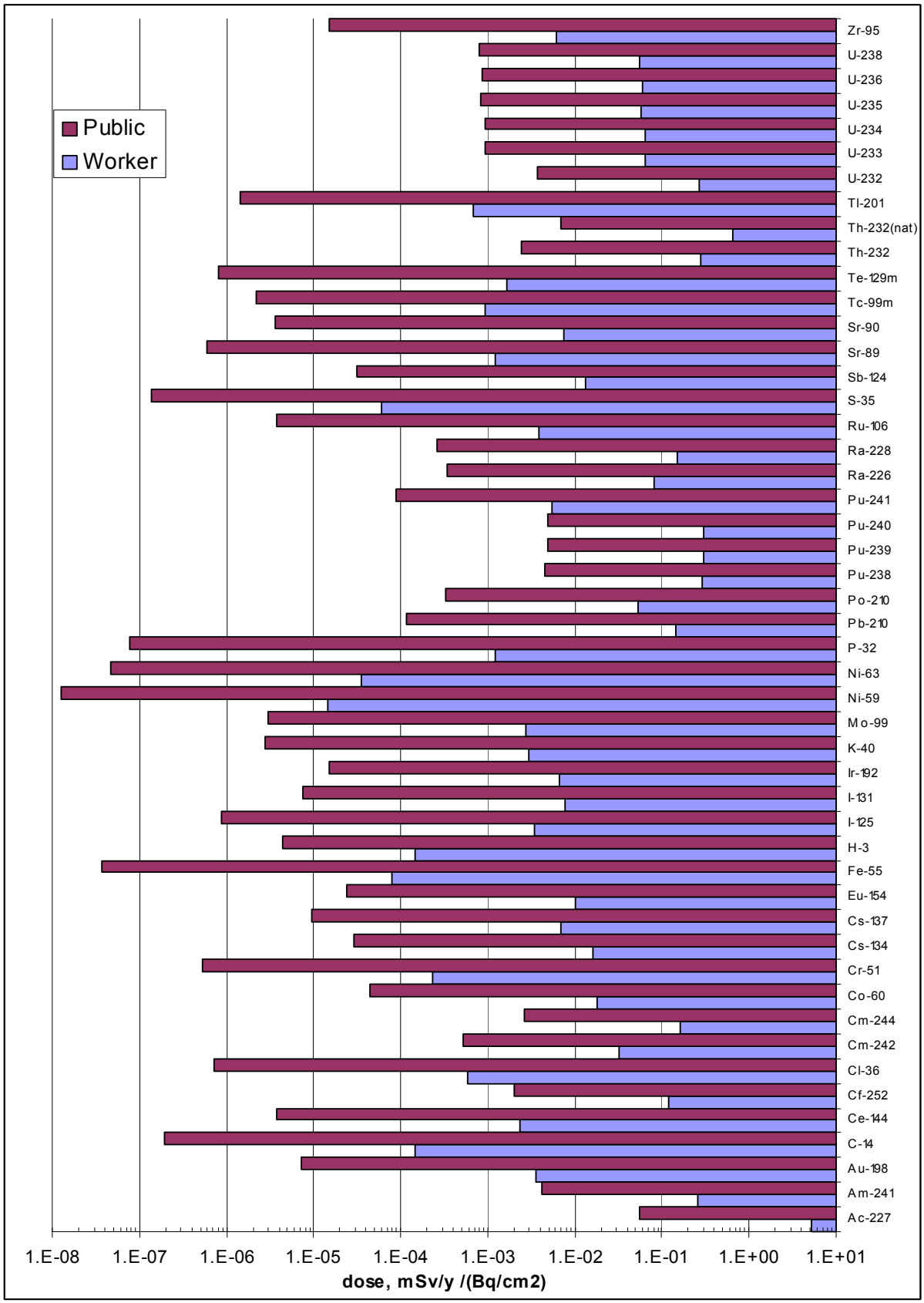

Figure 8. Worker and Public doses received from small remotely handled packages.

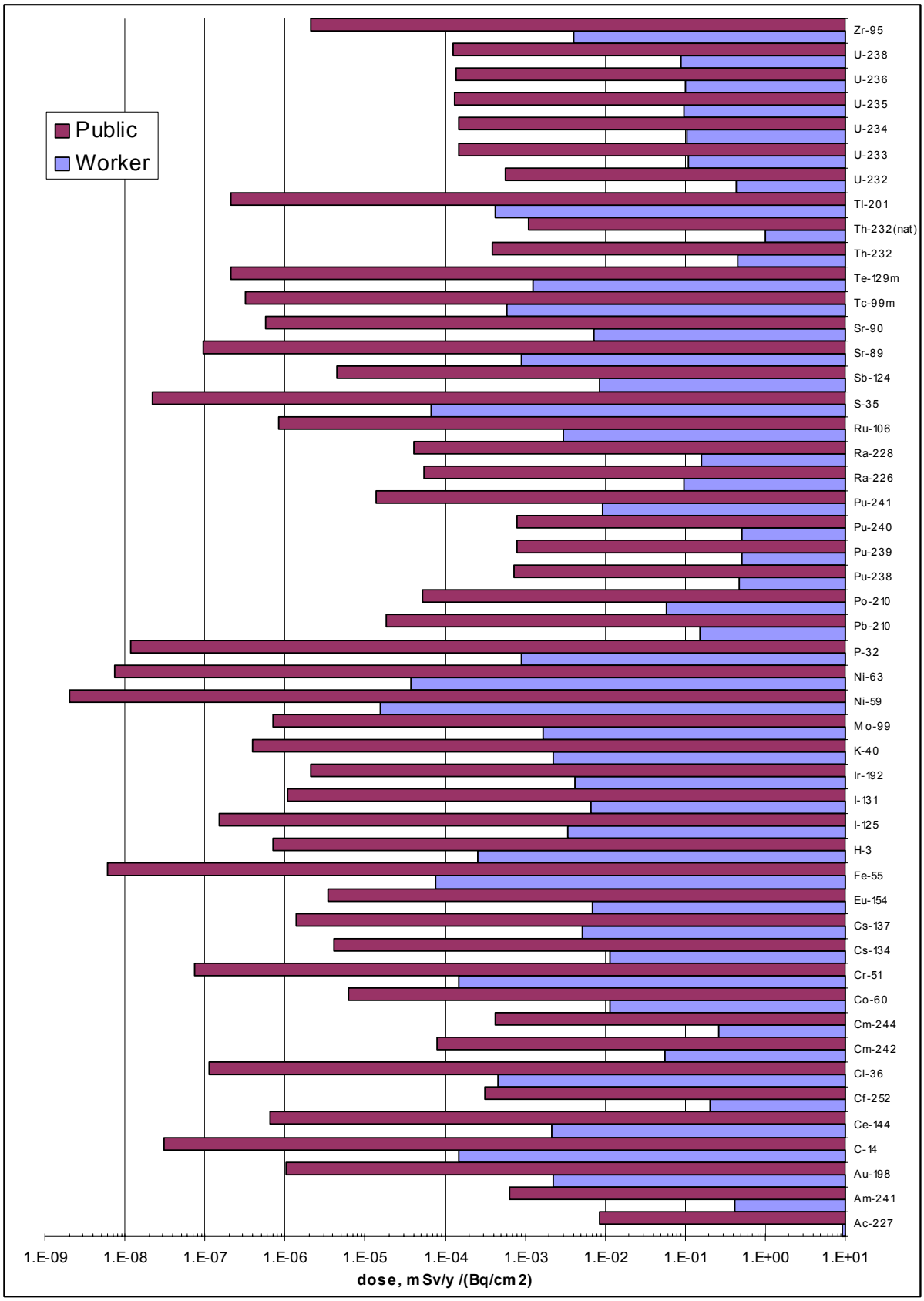

Figure 9. Worker and Public doses received from large remotely handled packages.

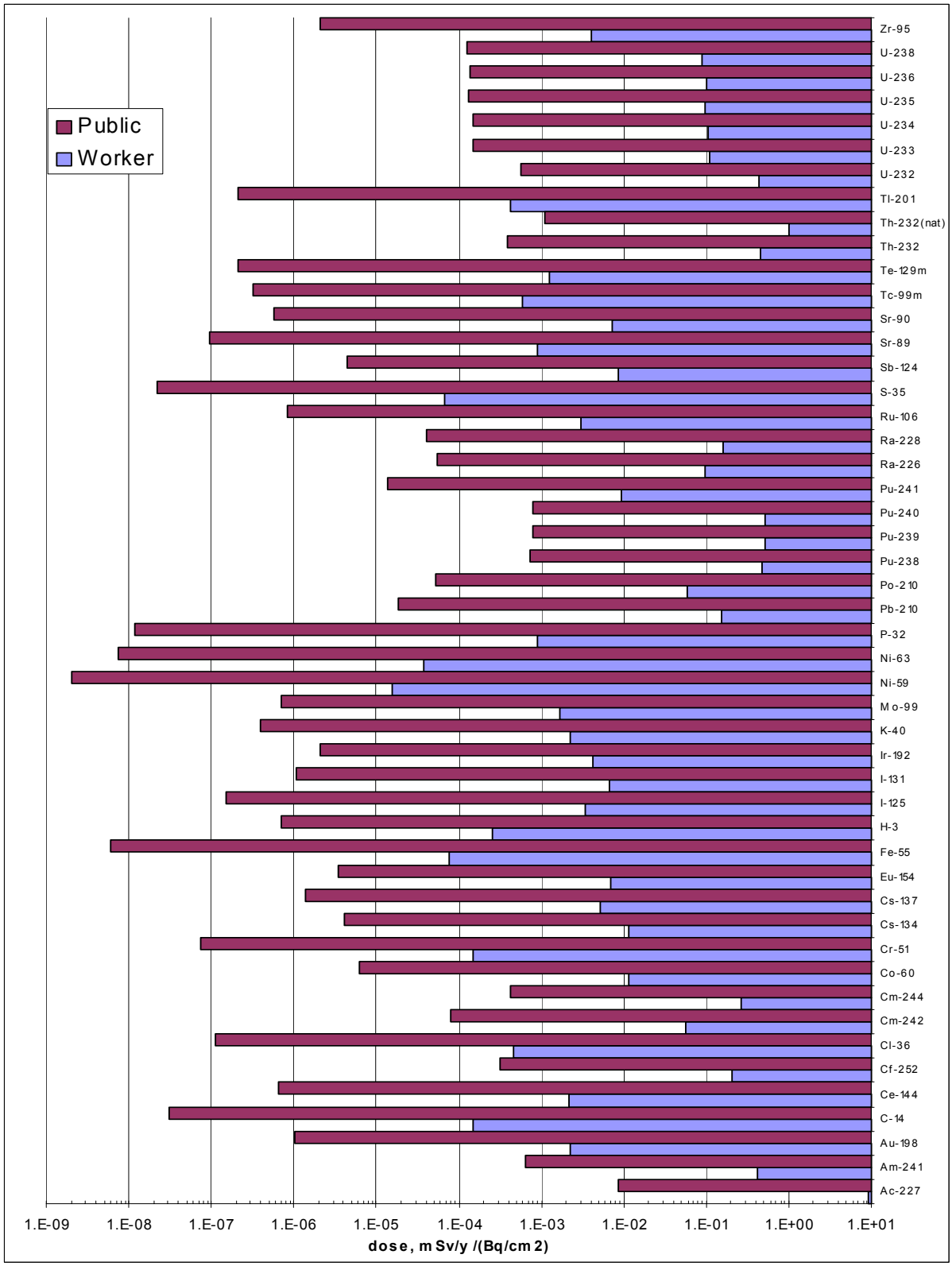

Figure 10. Worker and Public doses received from surface contaminated fuel flasks.

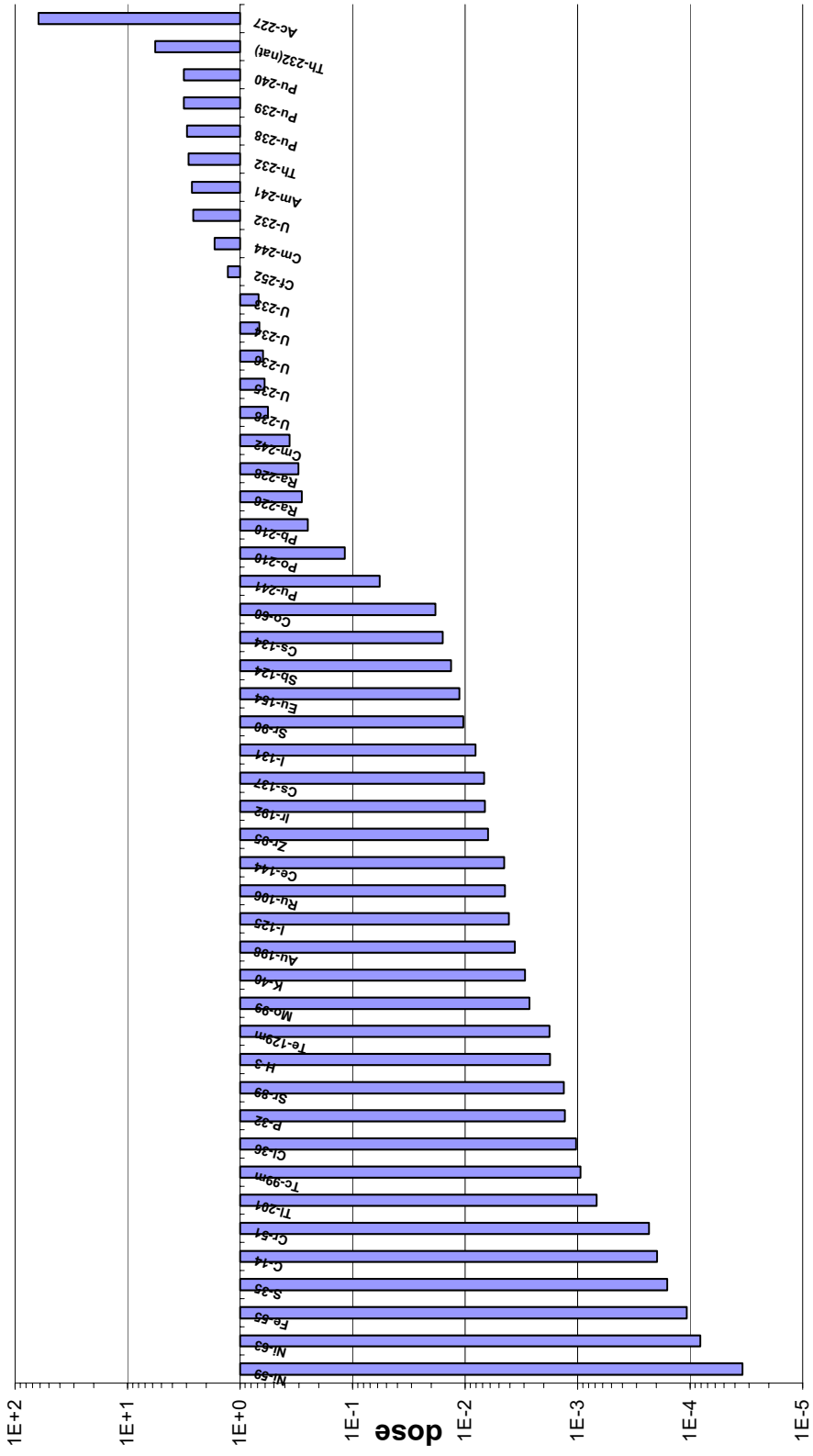

Figure 11. Annual dose received by the limiting worker for a surface activity concentration of 1 Bq/cm² in mSv/a.

The limiting workers (i.e., the persons who constitute the critical group from the point of view of dose received) for small manually handled packages are Workers C and F; the delivery driver who may also be involved in loading and unloading operations as may be the transfer site worker carrying out loading and unloading operations. The limiting pathway is radionuclide dependent for all package types. The limiting member of the public was either the flight passenger (Table 3, step 3.1b) for strong gamma emitters or the person living/working near a transfer point (Table 3, step 4) for all other radionuclides, here the limiting pathway was inhalation.

Workers A and F are the limiting workers for small remotely handled packages. They are the package preparation worker and the transfer site worker. For packages containing alpha emitters the limiting workers were Worker A, whereas for the other decay types it was either Worker A or F. The limiting member of the public was either a passenger on a ship carrying radioactive material (Table 3, step 3.1c) for strong gamma emitters, or the person living/working in the vicinity of a transfer point (Table 3, step 4) for all other radionuclides, here the limiting pathway was inhalation.

The limiting worker for large remotely handled packages and fuel flasks was Worker A for all radionuclides. The limiting member of the public for large remotely handled packages was either a passenger on a ship carrying radioactive material (Table 3, step 3.1c) for strong gamma emitters where external exposure is the dominating pathway. Or the person living/working in the vicinity of a transfer point (Table 3, step 4) for all other radionuclides, where the limiting pathway was inhalation.

In the case of fuel flasks the limiting member of the public was either a person present during a movement or interruption (Table 3, step 3.1a) for strong gamma emitters, or a person living/working near a transfer point (Table 3, step 4) for all other radionuclides where inhalation was the limiting pathway.

Small manually and small remotely handled packages are the dominating package types for all radionuclides for worker tasks; this is due to the high transport volumes considered together with the closer handling that occurs for these package types. Large remotely handled packages (here we used the example of ISO containers) give the lowest dose for the activity level. This package type has shorter handling times due to mechanical remote loading and unloading, which is the key factor in worker doses. Small manually handled packages and small remotely handled packages are the dominating package types for public exposures. Overall, workers rather than members of the public are the limiting persons for all radionuclides.

6.7.1.2. Results for a wider range of radionuclides

Following the Cologne meeting, IRSN carried out calculations for 350 nuclides. During the third and final RCM held in Berlin in November 2002 all the scenarios and many of the parameters were reviewed. The calculations were then adjusted to provide extrapolated values corresponding to the modified exposure parameters, as described below.

The calculations were performed for the 350 nuclides selected with the assumptions described previously with the scenarios and parameters adopted before the meeting in Berlin. The maximum total doses (for a surface contamination equal to 1 Bq/cm^2) have been calculated for each type of package and for each radionuclide. The results are given for SM, SR, LR packages and Fuel Flasks respectively in the report [6.18] where the dominating pathway has been indicated together with the percentage of the total dose that it represents and the category of workers or members of public who receive this dose. A very good agreement has been reached between the IRSN results and the Modelling Group results obtained on the 49 selected radionuclides.

Doses have then been assessed for each of the four package types and for the workers with the new parameters reviewed during the meeting of Berlin. The new parameters described in the report [6.7] do not appear more justified than the old ones except for the public exposure time even if they better correspond to the practices in some countries.

The doses to the public were shown to remain well below the doses for the workers, and, therefore, only the results for workers are presented here. The results using the revised parameters are presented in Table 24 for some typical radionuclides, such as some alpha emitters, a soft beta emitter, a strong beta emitter and a gamma emitter. They are compared to the previous results and the ratio of the pre- and post-Berlin meeting results are given in the last column.

Table 24. Comparison of results using revised parameters

Nuclide	Dose (mSv/a)/ (Bq/cm²)	Surface activity leading to 1 mSv/a (Bq/cm²)	Package	Worker	Pathway	Pathway	Ratio post/ pre Berlin meeting results
^{55}Fe	$1.07x10^{-04}$	$9.33x10^{+03}$	SM	C	Ing	62%	1.18
^{90}Sr0	$1.11x10^{-02}$	90	SM	C	Ing	55%	1.16
^{60}Co	$1.87x10^{-02}$	53.6	SR	F	Ext	62%	0.76
^{141}Ce	$9.63x10^{-04}$	$1.04x10^{+03}$	SM	C	Ext	31%	0.81
^{14}C	$1.97x10^{-04}$	$5.08x10^{+03}$	SM	C	Ing	59%	1.18
^{239}Pu	3.18	0.31	SM	C	Inh	98%	2.04
U depleted	$4.8x10^{-01}$	2.39	SM	C	Inh	99%	-
U natural	$6.2x10^{-01}$	1.61	SM	C	Inh	99%	-
U natural with daughters (for 1 Bq total)	$3.38x10^{-01}$	2.95	SM	C	Inh	93%	1.88
U4.5%	$6.61x10^{-01}$	1.51	SM	C	Inh	99%	-
U20%	$6.68x10^{-01}$	1.5	SM	C	Inh	99%	-
U50%	$6.70x10^{-01}$	1.49	SM	C	Inh	99%	-
U93%	$6.72x10^{-01}$	1.48	SM	C	Inh	99%	-
^{234}U	$6.74x10^{-01}$	1.48	SM	C	Inh	99%	2.04
^{235}U	$6.06x10^{-01}$	1.65	SM	C	Inh	98%	2.02
^{238}U	$5.67x10^{-01}$	1.60	SM	C	Inh	98%	2.03

Except for the alpha emitters, the results given in Table 24 show that the calculated values for the new scenarios characterized by the revised parameters are within $\pm$ 20% of the previous ones. For the alpha emitters the maximum dose is multiplied by a factor 2.

It was considered that due to the uncertainties of the resuspension rate, a difference of 20% is not very significant, so the previous results were kept for beta and gamma emitters and the maximum dose to the worker for the alpha emitters was recalculated multiplying the maximum dose by a factor 2.

The results so extrapolated are given in Table 25 for the 350 radionuclides, for the limiting scenarios, giving the doses corresponding to 1 Bq cm^{-2}.

Table 25. Maximal doses to the most exposed worker due to 1 Bq/cm²

Main Decay Mode	**Isotopes**	**Maximum Total Dose** (mSv/a)/(Bq/cm²)
α	^{225}Ac	$1.12\text{x}10^{-01}$
α	^{227}Ac	$6.00\text{x}10^{+01}$
β–, γ	^{228}Ac	$1.46\text{x}10^{-02}$
γ	^{105}Ag	$1.90\text{x}10^{-03}$
γ	^{108}Ag	$1.73\text{x}10^{-02}$
β–, γ	^{110m}Ag	$2.71\text{x}10^{-02}$
β–, γ	^{111}Ag	$9.88\text{x}10^{-04}$
β+, γ	^{26}Al	$2.58\text{x}10^{-02}$
α	^{241}Am	$2.63\text{x}10^{+00}$
α	^{242m}Am	$2.35\text{x}10^{+00}$
α	^{243}Am	$2.64\text{x}10^{+00}$
β+,γ	^{72}As	$1.76\text{x}10^{-02}$
γ	^{73}As	$2.37\text{x}10^{-04}$
β–, β+, γ	^{74}As	$8.45\text{x}10^{-03}$
β–, γ	^{76}As	$5.68\text{x}10^{-03}$
β–, γ	^{77}As	$4.89\text{x}10^{-04}$
α	^{211}At	$1.59\text{x}10^{-02}$
γ	^{193}Au	$2.94\text{x}10^{-05}$
β+, γ	^{194}Au	$9.92\text{x}10^{-03}$
γ	^{195}Au	$8.55\text{x}10^{-04}$
β–, γ	^{198}Au	$4.76\text{x}10^{-03}$
γ	^{198m}Au	$6.38\text{x}10^{-03}$
β–, γ	^{199}Au	$1.12\text{x}10^{-03}$
γ	^{131}Ba	$5.28\text{x}10^{-03}$
γ	^{133}Ba	$4.33\text{x}10^{-03}$
γ	^{133m}Ba	$8.56\text{x}10^{-04}$
β–, γ	^{140}Ba	$2.39\text{x}10^{-02}$
β–	^{10}Be	$5.37\text{x}10^{-04}$
γ	^{7}Be	$5.14\text{x}10^{-04}$
β+, γ	^{205}Bi	$2.27\text{x}10^{-04}$
γ	^{206}Bi	$3.09\text{x}10^{-02}$
γ	^{207}Bi	$1.56\text{x}10^{-02}$
β–, γ	^{210}Bi	$3.62\text{x}10^{-03}$
α	^{210m}Bi	$2.04\text{x}10^{-01}$
α	^{212}Bi	$3.09\text{x}10^{-02}$
α	^{247}Bk	$4.39\text{x}10^{+00}$
β–, γ	^{249}Bk	$4.92\text{x}10^{-03}$
β+, γ	^{76}Br	$1.19\text{x}10^{-04}$
β+,γ	^{77}Br	$3.11\text{x}10^{-03}$
β–, γ	^{82}Br	$2.63\text{x}10^{-02}$
β+, γ	^{11}C	$1.03\text{x}10^{-02}$
β–	^{14}C	$1.67\text{x}10^{-04}$
γ	^{41}Ca	$6.70\text{x}10^{-05}$
β–	^{45}Ca	$3.05\text{x}10^{-04}$
β–, γ	^{47}Ca	$1.37\text{x}10^{-02}$
γ	^{109}Cd	$2.02\text{x}10^{-03}$

Main Decay Mode	Isotopes	Maximum Total Dose (mSv/a)/(Bq/cm²)
β−, γ	^{113m}Cd	$1.07x10^{-02}$
β−, γ	^{115}Cd	$4.70x10^{-03}$
β−, γ	^{115m}Cd	$1.85x10^{-03}$
γ	^{139}Ce	$1.79x10^{-03}$
β−, γ	^{141}Ce	$1.18x10^{-03}$
β−, γ	^{143}Ce	$3.61x10^{-03}$
β−, γ	^{144}Ce	$3.54x10^{-03}$
α	^{248}Cf	$5.88x10^{-01}$
α	^{249}Cf	$4.40x10^{+00}$
α	^{250}Cf	$2.14x10^{+00}$
α	^{251}Cf	$4.49x10^{+00}$
α	^{252}Cf	$1.26x10^{+00}$
α	^{253}Cf	$9.53x10^{-02}$
α	^{254}Cf	$2.26x10^{+00}$
β−, γ	^{36}Cl	$7.77x10^{-04}$
β−, γ	^{38}Cl	$1.42x10^{-02}$
α	^{240}Cm	$2.20x10^{-01}$
α	^{241}Cm	$2.82x10^{-03}$
α	^{242}Cm	$3.55x10^{-01}$
α	^{243}Cm	$1.95x10^{+00}$
α	^{244}Cm	$1.65x10^{+00}$
α	^{245}Cm	$2.64x10^{+00}$
α	^{246}Cm	$2.64x10^{+00}$
α	^{247}Cm	$2.45x10^{+00}$
α	^{248}Cm	$9.29x10^{+00}$
β+, γ	^{55}Co	$8.22x10^{-04}$
β+, γ	^{56}Co	$3.21x10^{-02}$
γ	^{57}Co	$1.21x10^{-03}$
β+, γ	^{58}Co	$9.97x10^{-03}$
γ	^{58m}Co	$2.01x10^{-05}$
β−, γ	^{60}Co	$2.46x10^{-02}$
γ	^{51}Cr	$3.29x10^{-04}$
γ	^{129}Cs	$3.02x10^{-03}$
γ	^{131}Cs	$3.51x10^{-04}$
β+, β−, γ	^{132}Cs	$7.43x10^{-03}$
β−, γ	^{134}Cs	$2.00x10^{-02}$
γ, β−	^{134m}Cs	$4.54x10^{-04}$
β−	^{135}Cs	$4.76x10^{-04}$
β−, γ	^{136}Cs	$2.20x10^{-02}$
β−, γ	^{137}Cs	$8.90x10^{-03}$
β−, β+,γ	^{64}Cu	$2.11x10^{-03}$
β−, γ	^{67}Cu	$1.16x10^{-03}$
γ	^{159}Dy	$3.18x10^{-05}$
β−, γ	^{165}Dy	$8.26x10^{-04}$
β−, γ	^{166}Dy	$2.33x10^{-03}$
β−, γ	^{169}Er	$1.85x10^{-04}$
β−, γ	^{171}Er	$4.29x10^{-03}$
β−, β+, γ	^{150}Eu (12,6 h)	$8.92x10^{-05}$

Main Decay Mode	Isotopes	Maximum Total Dose (mSv/a)/(Bq/cm²)
γ	^{150}Eu (34,2 y)	$1.87x10^{-03}$
γ	^{147}Eu	$1.35x10^{-04}$
γ, β+	^{148}Eu	$3.69x10^{-04}$
γ	^{149}Eu	$3.09x10^{-05}$
β–, β+, γ	^{152}Eu	$1.26x10^{-02}$
β–, β+, γ	^{152m}Eu	$3.73x10^{-03}$
β–, γ	^{154}Eu	$1.49x10^{-02}$
β–, γ	^{155}Eu	$9.11x10^{-04}$
β–, γ	^{156}Eu	$1.23x10^{-02}$
β+, γ	^{18}F	$1.04x10^{-02}$
β+, γ	^{52}Fe	$3.27x10^{-02}$
γ	^{55}Fe	$9.06x10^{-05}$
β–, γ	^{59}Fe	$1.19x10^{-02}$
β–	^{60}Fe	$3.17x10^{-02}$
γ	^{67}Ga	$1.44x10^{-03}$
β+, γ	^{68}Ga	$9.63x10^{-03}$
β–, γ	^{72}Ga	$2.48x10^{-02}$
γ	^{146}Gd	$7.54x10^{-04}$
α	^{148}Gd	$2.86x10^{+00}$
γ	^{153}Gd	$1.39x10^{-03}$
β–, γ	^{159}Gd	$9.09x10^{-04}$
γ	^{68}Ge	$9.93x10^{-03}$
γ	^{71}Ge	$6.86x10^{-06}$
β–, γ	^{77}Ge	$1.11x10^{-02}$
β–	^{3}H	$8.54x10^{-04}$
γ	^{172}Hf	$2.29x10^{-03}$
γ	^{175}Hf	$1.23x10^{-04}$
β–, γ	^{181}Hf	$6.04x10^{-03}$
β–, γ	^{182}Hf	$1.76x10^{-02}$
γ	^{194}Hg	$1.05x10^{-02}$
γ	^{195m}Hg	$1.56x10^{-04}$
γ	^{197}Hg	$8.08x10^{-04}$
γ	^{197m}Hg	$2.02x10^{-03}$
β–, γ	^{203}Hg	$2.52x10^{-03}$
β–, γ	^{166}Ho	$1.28x10^{-03}$
β–, γ	^{166m}Ho	$2.05x10^{-02}$
γ	^{123}I	$1.82x10^{-03}$
β+, γ	^{124}I	$1.31x10^{-02}$
γ	^{125}I	$3.95x10^{-03}$
β–, β+, γ	^{126}I	$1.14x10^{-02}$
β–, γ	^{129}I	$2.48x10^{-02}$
β–, γ	^{131}I	$8.99x10^{-03}$
β–, γ	^{132}I	$2.28x10^{-02}$
β–, γ	^{133}I	$7.83x10^{-03}$
β–, γ	^{134}I	$2.50x10^{-02}$
β–, γ	^{135}I	$1.79x10^{-02}$
γ	^{111}In	$4.07x10^{-03}$
γ	^{113m}In	$2.89x10^{-03}$

Main Decay Mode	Isotopes	Maximum Total Dose (mSv/a)/(Bq/cm²)
γ	^{114m}In	2.84x10^{-03}
β–, γ	^{115m}In	1.93x10^{-03}
γ	^{189}Ir	6.77x10^{-05}
γ	^{190}Ir	1.48x10^{-02}
β–, γ	^{192}Ir	8.94x10^{-03}
β–, γ	^{194}Ir	1.98x10^{-03}
β–, β+, γ	^{40}K	3.37x10^{-03}
β–, γ	^{42}K	2.60x10^{-03}
β–, γ	^{43}K	1.02x10^{-02}
γ	^{137}La	4.89x10^{-04}
β–, γ	^{140}La	2.19x10^{-02}
γ	^{172}Lu	3.40x10^{-04}
γ	^{173}Lu	1.23x10^{-04}
γ	^{174}Lu	1.91x10^{-04}
γ	^{174m}Lu	2.19x10^{-04}
β–, γ	^{177}Lu	6.85x10^{-04}
β–, γ	^{28}Mg	2.90x10^{-02}
β+, γ	^{52}Mn	3.37x10^{-02}
γ	^{53}Mn	7.75x10^{-06}
γ	^{54}Mn	8.48x10^{-03}
β–, γ	^{56}Mn	1.65x10^{-02}
γ	^{93}Mo	7.17x10^{-04}
β–, γ	^{99}Mo	3.26x10^{-03}
β+, γ	^{13}N	1.03x10^{-02}
β+, γ	^{22}Na	2.23x10^{-02}
β–, γ	^{24}Na	3.49x10^{-02}
γ	^{93m}Nb	6.11x10^{-05}
β–, γ	^{94}Nb	1.65x10^{-02}
β–, γ	^{95}Nb	7.83x10^{-03}
β–, γ	^{97}Nb	7.30x10^{-03}
β–, γ	^{147}Nd	2.09x10^{-03}
β–, γ	^{149}Nd	5.09x10^{-03}
γ	^{59}Ni	2.31x10^{-05}
β–	^{63}Ni	5.51x10^{-05}
β–, γ	^{65}Ni	5.74x10^{-03}
β–, γ	^{236m}Np	7.19x10^{-04}
γ	^{235}Np	9.03x10^{-05}
α	^{236}Np	1.99x10^{-01}
α	^{237}Np	1.46x10^{+00}
β–, γ	^{239}Np	2.05x10^{-03}
γ	^{185}Os	7.27x10^{-03}
β–, γ	^{191}Os	8.46x10^{-04}
γ	^{191m}Os	1.54x10^{-04}
β–, γ	^{193}Os	1.30x10^{-03}
β–, γ	^{194}Os	3.08x10^{-03}
β–	^{32}P	1.25x10^{-03}
β–	^{33}P	9.88x10^{-05}
β–, γ	^{230}Pa	2.87x10^{-02}

Main Decay Mode	Isotopes	Maximum Total Dose (mSv/a)/(Bq/cm²)
α	^{231}Pa	$8.70x10^{+00}$
β–, γ	^{233}Pa	$2.58x10^{-03}$
γ, β+	^{201}Pb	$9.54x10^{-04}$
γ	^{202}Pb	$2.40x10^{-03}$
γ	^{203}Pb	$3.11x10^{-03}$
γ	^{205}Pb	$9.16x10^{-05}$
β–, γ	^{210}Pb	$1.92x10^{-01}$
β–, γ	^{212}Pb	$2.03x10^{-02}$
γ	^{103}Pd	$2.67x10^{-04}$
β–	^{107}Pd	$8.96x10^{-06}$
β–, γ	^{109}Pd	$6.55x10^{-04}$
γ	^{143}Pm	$3.31x10^{-03}$
γ	^{144}Pm	$1.62x10^{-02}$
γ	^{145}Pm	$5.46x10^{-04}$
β–	^{147}Pm	$2.51x10^{-04}$
β–, γ	^{148m}Pm	$2.13x10^{-02}$
β–, γ	^{149}Pm	$8.00x10^{-04}$
β–,γ	^{151}Pm	$3.40x10^{-03}$
α	^{210}Po	$1.63x10^{-01}$
β–, γ	^{142}Pr	$9.32x10^{-04}$
β–	^{143}Pr	$7.73x10^{-04}$
γ	^{191}Pt	$3.12x10^{-03}$
γ	^{188}Pt	$3.36x10^{-04}$
γ	^{193}Pt	$9.34x10^{-06}$
γ	^{193m}Pt	$3.87x10^{-04}$
γ	^{195m}Pt	$8.22x10^{-04}$
β–, γ	^{197}Pt	$6.40x10^{-04}$
γ, β–	^{197m}Pt	$1.84x10^{-03}$
α	^{236}Pu	$1.26x10^{+00}$
γ	^{237}Pu	$5.49x10^{-04}$
α	^{238}Pu	$2.93x10^{+00}$
α	^{239}Pu	$3.12x10^{+00}$
α	^{240}Pu	$3.12x10^{+00}$
β–, γ	^{241}Pu	$2.83x10^{-02}$
α	^{242}Pu	$3.03x10^{+00}$
α	^{244}Pu	$2.94x10^{+00}$
α	^{223}Ra	$5.87x10^{-01}$
α	^{224}Ra	$2.95x10^{-01}$
β–, γ	^{225}Ra	$3.02x10^{-01}$
α	^{226}Ra	$3.58x10^{-01}$
β–	^{228}Ra	$2.26x10^{-01}$
β+, γ	^{81}Rb	$6.12x10^{-03}$
γ	^{83}Rb	$5.56x10^{-03}$
β+, β–, γ	^{84}Rb	$9.54x10^{-03}$
β–, γ	^{86}Rb	$2.21x10^{-03}$
β–	^{87}Rb	$3.36x10^{-04}$
γ	^{184}Re	$9.02x10^{-03}$
γ	^{184m}Re	$4.24x10^{-03}$

Main Decay Mode	Isotopes	Maximum Total Dose (mSv/a)/(Bq/cm²)
β–, γ	^{186}Re	$9.58x10^{-04}$
β–	^{187}Re	$1.14x10^{-06}$
β–, γ	^{188}Re	$1.64x10^{-03}$
β–, γ	^{189}Re	$1.77x10^{-04}$
γ	^{101}Rh	$1.90x10^{-04}$
γ	^{102}Rh	$9.41x10^{-04}$
β–, β+, γ	^{102m}Rh	$3.30x10^{-04}$
γ	^{103m}Rh	$2.40x10^{-05}$
β–, γ	^{105}Rh	$1.03x10^{-03}$
β+, γ	^{99}Rh	$1.25x10^{-04}$
α	^{222}Rn	$3.73x10^{-02}$
β–, γ	^{103}Ru	$5.17x10^{-03}$
β–, γ	^{105}Ru	$9.71x10^{-03}$
β–	^{106}Ru	$4.87x10^{-03}$
γ	^{97}Ru	$2.41x10^{-03}$
β–	^{35}S	$1.04x10^{-04}$
β–, β+, γ	^{122}Sb	$5.42x10^{-03}$
β–, γ	^{124}Sb	$1.79x10^{-02}$
β–, γ	^{125}Sb	$4.81x10^{-03}$
β–, γ	^{126}Sb	$2.64x10^{-02}$
β+, γ	^{44}Sc	$2.10x10^{-02}$
β–, γ	^{46}Sc	$2.01x10^{-02}$
β–, γ	^{47}Sc	$1.37x10^{-03}$
β–, γ	^{48}Sc	$3.27x10^{-02}$
γ	^{75}Se	$4.39x10^{-03}$
β–	^{79}Se	$6.56x10^{-04}$
β–, γ	^{31}Si	$6.30x10^{-04}$
β–	^{32}Si	$2.87x10^{-04}$
γ	^{145}Sm	$9.40x10^{-05}$
α	^{147}Sm	$5.96x10^{-01}$
β–, γ	^{151}Sm	$1.43x10^{-04}$
β–, γ	^{153}Sm	$1.19x10^{-03}$
γ	^{113}Sn	$3.39x10^{-03}$
γ	^{117m}Sn	$1.71x10^{-03}$
γ	^{119m}Sn	$2.44x10^{-04}$
β–, γ	^{121m}Sn	$1.20x10^{-04}$
β–, γ	^{123}Sn	$5.62x10^{-04}$
β–, γ	^{125}Sn	$4.37x10^{-03}$
γ	^{126}Sn	$1.80x10^{-02}$
γ	^{82}Sr	$1.25x10^{-02}$
γ	^{85}Sr	$5.43x10^{-03}$
γ	^{85m}Sr	$2.11x10^{-03}$
γ	^{87m}Sr	$3.32x10^{-03}$
β–, γ	^{89}Sr	$1.26x10^{-03}$
β–	^{90}Sr	$9.51x10^{-03}$
β–, γ	^{91}Sr	$1.09x10^{-02}$
β–, γ	^{92}Sr	$1.67x10^{-02}$
γ, β+	^{178}Ta	$2.03x10^{-05}$

Main Decay Mode	Isotopes	Maximum Total Dose (mSv/a)/(Bq/cm²)
γ	^{179}Ta	1.91x10^{-05}
β–, γ	^{182}Ta	1.20x10^{-02}
γ	^{157}Tb	9.54x10^{-05}
β–, γ	^{158}Tb	1.64x10^{-03}
β–, γ	^{160}Tb	1.14x10^{-02}
β+, γ	^{95m}Tc	7.08x10^{-03}
γ	^{96}Tc	2.54x10^{-02}
γ	^{96m}Tc	2.48x10^{-02}
γ	^{97}Tc	1.71x10^{-04}
γ	^{97m}Tc	3.92x10^{-04}
β–, γ	^{98}Tc	1.45x10^{-02}
β–	^{99}Tc	1.83x10^{-04}
γ	^{99m}Tc	1.20x10^{-03}
γ	^{121}Te	6.05x10^{-03}
γ	^{121m}Te	2.72x10^{-03}
γ	^{123m}Te	1.85x10^{-03}
γ	^{125m}Te	7.91x10^{-04}
β–, γ	^{127}Te	3.94x10^{-04}
γ, β–	^{127m}Te	1.15x10^{-03}
β–, γ	^{129}Te	1.24x10^{-03}
β–, γ	^{129m}Te	2.20x10^{-03}
β–, γ	^{131m}Te	2.45x10^{-02}
β–, γ	^{132}Te	2.58x10^{-02}
	Th nat	5.79x10^{+00}
α	^{227}Th	5.92x10^{-01}
α	^{228}Th	2.50x10^{+00}
α	^{229}Th	6.72x10^{+00}
α	^{230}Th	2.73x10^{+00}
β–, γ	^{231}Th	3.95x10^{-04}
α	^{232}Th	2.83x10^{+00}
β–, γ	^{234}Th	1.90x10^{-03}
γ	^{44}Ti	2.69x10^{-02}
γ, β+	^{200}Tl	1.27x10^{-02}
γ	^{201}Tl	9.16x10^{-04}
γ	^{202}Tl	4.81x10^{-03}
β–, γ	^{204}Tl	6.68x10^{-04}
γ	^{167}Tm	1.59x10^{-04}
β–, γ	^{170}Tm	9.73x10^{-04}
β–, γ	^{171}Tm	7.91x10^{-05}
	U nat	4.82x10^{+00}
α	^{230}U	1.16x10^{+00}
α	^{232}U	2.59x10^{+00}
α	^{233}U	6.72x10^{-01}
α	^{234}U	6.62x10^{-01}
α	^{235}U	5.99x10^{-01}
α	^{236}U	6.14x10^{-01}
α	^{238}U	5.58x10^{-01}
β+, γ	^{48}V	2.92x10^{-02}

Main Decay Mode	Isotopes	Maximum Total Dose (mSv/a)/(Bq/cm²)
γ	^{49}V	$4.83x10^{-06}$
γ	^{178}W	$7.00x10^{-05}$
γ	^{181}W	$4.32x10^{-04}$
β–	^{185}W	$2.68x10^{-04}$
β–, γ	^{187}W	$5.32x10^{-03}$
β–, γ	^{188}W	$2.11x10^{-03}$
β+, γ	^{87}Y	$8.18x10^{-03}$
β+, γ	^{88}Y	$2.47x10^{-02}$
β–	^{90}Y	$1.42x10^{-03}$
β–, γ	^{91}Y	$1.43x10^{-03}$
γ	^{91m}Y	$5.40x10^{-03}$
β–, γ	^{92}Y	$4.79x10^{-03}$
β–, γ	^{93}Y	$2.24x10^{-03}$
γ	^{169}Yb	$3.37x10^{-03}$
β–, γ	^{175}Yb	$6.85x10^{-04}$
β+,γ	^{65}Zn	$6.49x10^{-03}$
β–	^{69}Zn	$4.12x10^{-04}$
γ	^{69m}Zn	$4.74x10^{-03}$
γ	^{88}Zr	$4.22x10^{-03}$
β–	^{93}Zr	$1.44x10^{-03}$
β–, γ	^{95}Zr	$7.77x10^{-03}$
β–, γ	^{97}Zr	$1.71x10^{-02}$

In addition, the maximum dose has been calculated for various forms of uranium. The results are summarised in Table 26 for unirradiated uranium including depleted uranium, natural uranium, natural uranium with daughters and enriched uranium (4.5%, 20%, 50%, 93%).

Table 26. Results for unirradiated uranium

	Surface activity resulting in 1 mSv/year for worker (Bq/cm²)
Natural uranium with daughters	2.9
Depleted, natural and enriched uranium (U-234, U-235 and U-238) without daughters	1.5 to 2.4

6.7.2. Estimation of conservatism

It is an important element of any model analysis to at least estimate (if not quantify) the level of conservatism the model presents. The Basic Model contains a number of assumptions that have been identified to be conservative, i.e. which lead to upper estimates of dose for a given surface contamination. The most important assumptions are:

- Homogeneous distribution of the contamination on the surface of all packages (consignments).

- A resuspension rate of 10^{-4} h^{-1} in all scenarios that seems to be at the upper end of the range of reported values (cf. section 5.1).

- All six surfaces of each package are assumed to contribute to the airborne activity, regardless whether these surfaces are covered or rest on the floor.

- The combination of the steps and sub-steps of the model has been effected in a way that leads to a maximum estimate of the annual working time. A conservative cut-off value for the annual working time has been used.

However, it should be noted that certain uncertainties inherent to this model, some of which are not conservative. These include the resuspension rate (cf 5.1), which varies from 1×10^{-3} per h [6.19] to 1×10^{-7} per h [6.15]. The external doses due to beta and neutron emitters, and the influence on the skin dose or on ingestion of 'hot spots' are not taken into account.

While it may be argued that more pessimistic values for crucial parameters can always be found, it must be emphasized that the aim is to ensure that the entire model is conservative enough, but not to choose all parameters at their upper end of their probability distributions, i.e. at a high level of conservatism. It is important to note that the multiplication of a number of parameter values that have all been chosen at a high percentile of their respective probability distributions (e.g. at the 95-percentile) will yield a result that is far beyond that overall percentile. In this way, the overall effect of such a combination of factors could lead to a result which is overly conservative.

In order to get the conservatism of the overall model right, the parameter values need to be chosen with care, to be realistic but still on the conservative side. An indication how to achieve this proceed can be found in the Advisory Material for the IAEA Regulations TS-G-1.1 [6.1], which contains guidance on how to perform calculations for the segregation distances. In paragraph 306.1 it is stated that "the dose values given in paragraph 306 (i.e. 5 mSv/a for occupationally exposed workers and of 1 mSv/a to the critical group) are ... required to be used together with *hypothetical but realistic parameters* in order to obtain appropriate segregation distances". Such an approach has also been followed in implementing the Basic Model.

It is also important to link the degree of conservatism of a model with the dose range for which it is intended. Obviously, a model that is devised to establish a link between activities and the dose limit (as was the Fairbairn model [6.20]) or dose criteria not far away from that limit will require a sufficiently high degree of conservatism. This is clearly different if a model is devised to operate in the range of trivial doses (i.e. the range 10 to 100 μSv/a individual effective dose), where explicitly realistic models are required according to Safety Series 89 [6.21]. The CRP and especially the Modelling Group considered that the Basic Model possesses a sufficiently high degree of conservatism to operate with dose criteria between around 1 mSv/a up to the dose limit for workers and a few tenths of mSv/a up to 1 mSv/a for members of the general public. France disagrees with paragraph 6.7.2. It should be noted that large uncertainties remain in the model, some of which are not conservative, such as:

- uncertainties on the resuspension rate (two orders of magnitude)
- the external doses due to the beta and neutron emitters are not taken into account
- the influence on the skin dose or on ingestion of the 'hot spots' is not taken into account.

6.7.3. Sensitivity analysis

The aim of a sensitivity analysis is to gain knowledge on the relative influence of each parameter on the result. This is equivalent to testing how sensitive the result will be to changes in each parameter. As the Basic Model is rather complex, a straightforward analysis of the importance of certain parameters is not easily possible for the following reasons:

- The nuclides for which the calculations are performed have very different radiological properties. This leads to different pathways (external exposure, inhalation, ingestion) becoming most relevant for different nuclides. Therefore, the importance of parameters depends on the radionuclides.

- The doses are calculated by summing contributions from various pathways. In addition, the Basic Model uses maximum and minimum functions (e.g. taking the maximum dose contribution from any worker and afterwards from any of the four package types), which does not permit any simple linear dependence between parameter values and the resulting dose.

- The Basic Model uses cut-off criteria for that part of the dose calculation that depends on the annual working time, which of course becomes effective only for those workers whose working time adds up to exceed this cut-off criteria.

This means that a straightforward sensitivity analysis by analytical methods is not feasible here. Instead, the sensitivity analysis was performed by using a numerical simulation tool [6.22], which can be used to test the influence of each parameter on the result (determining the contribution of each parameter to the variance of the result).

The relative contribution of the exposure pathways: external irradiation, inhalation, ingestion, and skin contamination differs for each radionuclide. Therefore, the following radionuclides were chosen, each of which represents distinctive groups of radionuclides:

- ^{60}Co for strong γ emitters with a relatively low ingestion or inhalation dose coefficient;

- ^{63}Ni for weak β - emitters that do not give rise to external irradiation and that have a low ingestion or inhalation dose;

- ^{90}Sr for stronger β - emitters with no external irradiation;

- ^{241}Am for α - emitters with a high ingestion and inhalation dose coefficient.

For the actual sensitivity analysis, all (relevant) parameter values were allowed to vary according to the following assumptions:

- For all parameters except the resuspension rate, a normal distribution was chosen with the mean value μ corresponding to the value used in the Basic Model calculations and a standard deviation σ which equals 0.1 μ.

- The resuspension rate was treated with a lognormal distribution with a mean value of 10^{-4} h^{-1} and a standard deviation of 5 10^{-4} h^{-1}.

- Dose coefficients were not varied (implying that exposure distances were also kept constant because the exposure geometry directly affects the dose coefficients of external irradiation).

The following results have been obtained for the four nuclides listed above:

- ^{60}Co : The results for ^{60}Co are most strongly influenced by all parameters relating to working time (number of working days per year, working hours per day, cut-off of time which is apportioned to the transport scenarios; annual duration of transport for the driver, person C), to configuration of the loads and the number of loads per day (e.g. for package type SR: number of packages in van and package related exposure times), and to the percentage of activity deposited on the ground (ground shine).

- ^{63}Ni : The results for ^{63}Ni are most strongly influenced by all parameters relating to resuspension rates (especially for package type SM, transport of parcels in a small van), to inadvertent ingestion (transfer factors, contaminated area of hands) and to inhalation (inhalation rates), to exposure time

(cut-off of annual working time, working hours per day), and to skin exposure (transfer factors, contaminated area of face).

- ^{90}Sr : The results for ^{90}Sr are most strongly influenced by all parameters relating to resuspension rates (especially for package type SM, transport of parcels in a small van), and to inadvertent ingestion (transfer factors surface→hands and hands→mouth).

- ^{241}Am : The results for ^{241}Am are most strongly influenced by all parameters relating to resuspension rates (for all package types), to inhalation (surface area of the packages, air exchange rate in the room, size of the room; inhalation rate of workers), to working time (exposure times in various sub-steps, cut-off of the annual working time which is apportioned to the transport scenarios), to load configuration (number of packages per load), to inadvertent ingestion (transfer factors package→hands, contaminated area of hand from which activity is ingested).

In summary, the analysis shows that

- assumptions on annual working time,

- the resuspension rate and

- assumptions on transfer of activity

are those parameters that have the highest influence on the overall result. Any attempts to improve the scenario parameters should, therefore, be focused on these parameters. This means that parameter values relating to working steps (e.g. the exposure times for individual sub-tasks), air exchange rates and wind speed, percentage of activity deposited on the ground etc. warrant less attention.

It is important to distinguish this sensitivity analysis from an uncertainty analysis that would go one step beyond the results presented here by determining the value range over which the result will vary (on a given confidence level). This would, however, require knowledge of the uncertainties in the parameter values and of the probability distributions of these variations for each parameter. Currently, there are no quantitative data from which this information could be derived.

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7. ADDITIONAL EVALUATIONS OF DOSES DUE TO NON-FIXED CONTAMINATION

7.1. Deriving optimized limits for non-fixed contamination on spent fuel casks

As described in Section 4.4, the Fairbairn model [7.1] is based on exposure scenarios and radionuclides that are not appropriate for spent fuel casks. The exposure scenarios considered in the model (hand contact without subsequent near-term monitoring or decontamination, dusty conditions, etc.) are not representative of the operational practices and environments associated with handling of spent fuel casks. The Fairbairn model also does not consider the potentially significant doses to workers resulting from actions necessary to achieve compliance with the surface contamination limits. Finally, the Fairbairn model does not consider doses to other groups such as members of the public. All of these dose pathways and the resulting collective doses must be considered if optimized contamination limits for spent fuel casks are to be derived.

The IAEA Basic Safety Standards [7.2] call for the optimization of protection and safety "in relation to exposures from any particular source within a practice..." and states that "...protection and safety shall be optimized in order that the magnitude of individual doses, the number of people exposed and the likelihood of incurring exposures all be kept as low as reasonable achievable...within the restriction that the doses to individuals delivered by the source be subject to dose constraints". In the case of non-fixed surface contamination on spent fuel casks there are exposures to both workers and the public that must be taken into account.

Optimization of protection resulting from removable surface contamination on spent fuel casks requires considering all doses that could result from the contamination itself and doses that vary as a function of the allowable contamination limit. When evaluating the doses due to surface contamination, the doses due to the contents of the cask can be disregarded except where these doses vary due to the contamination limits. That is, where the doses due to the contents are the same regardless of the contamination limits, they can be considered as a constant and need not be factored into the evaluation.

The total effective dose due to removable surface contamination on the spent fuel casks can be expressed as:

$$E_{tot} = E_{wkr} + E_{pub}$$

where:

E_{tot} = total collective dose

E_{wkr} = collective dose to workers (both internal and external)

E_{pub} = collective dose to the public (both internal and external)

The dose components of the factors in the above equation are derived in more detail in the Annexes for all significant exposure pathways.

Limits for non-fixed surface contamination on spent fuel casks should be based on a model that considers all appropriate exposure scenarios (both in the workplace and in public areas) and that optimizes the resulting doses. The existing limits (based on the Fairbairn model) may lead to operational problems, non-compliance issues, and public perceptions of significant health risks when the actual risks are very low. An approach based on optimized protection should be applied to ensure that spent nuclear fuel packaging operations do not result in unnecessary radiation exposure.

There are doses that, while not resulting directly from the non-fixed contamination on the surface of a spent fuel cask, are strongly influenced by and are received during decontamination to meet the

regulatory limits on contamination. For example, actions necessary to comply with the current derived limits for light water reactor spent nuclear fuel casks can result in significant external doses to workers. This is due to the relatively high radiation levels around the loaded casks where workers must function during the measurement of contamination levels and while decontaminating the cask. Optimization of protection, in compliance with the regulatory limits, warrants taking into account all the dose components, received in the various operations related to transport of a package and resulting from the cask contamination.

A detailed overall assessment of the doses from contamination on spent fuel casks was performed as part of the CRP. Where possible, this study built on the parameters and exposure scenarios developed for the Basic Model. The Basic Model pathways for inhalation, ingestion and direct radiation from contamination were adjusted with more detailed times and distances selected to be representative for cask handling operations at light water reactors (cf. Section 4.5.3 and [7.3]). This allowed a more detailed assessment of the effect of higher contamination limits on workers' doses. Doses to the public resulting from inhalation and direct radiation from the contamination were also calculated in order to support an overall assessment of doses as a function of allowable contamination limits.

The results show that higher allowable contamination limits for spent fuel casks can result in lower overall doses when there is a corresponding reduction in the effort required to decontaminate and monitor surface contamination. For a representative spent fuel cask shipment, using actual population and route information, the overall dose decreases when the workers are able to complete the decontamination and monitoring tasks in such as way as to save 2 to 20% of their dose. Doses to members of the public (both individual and collective) are orders of magnitude lower than the doses to workers. Consequently, fractional improvements in the time required to complete decontamination and monitoring activities can result in overall dose savings.

7.2. Development of the Conservative Model and the Realistic Model for spent fuel transportation

A study has been carried out [7.4] in order to evaluate the safety margin provided by the Regulations TS-R-1 [7.5] with respect to surface contamination of packages. The focus of the study was on spent fuel transports. Two approaches were taken - conservative and realistic - to evaluate doses from non-fixed surface contamination to workers and to the general public.

The conservative approach employs a model the parameters of which are conservatively chosen (referred to as "Conservative Model") to estimate doses that are very unlikely to be exceeded. Dose evaluations are made for vehicle transportations of spent fuel packages with an assumed uniform surface contamination of 1 Bq/cm^2 all over the surfaces. Basic parameters of the exposure scenario for the Conservative Model are summarized in Table 27.

Table 27. Basic parameters for dose estimation by the conservative mode

Package	Spent fuel cask, cylindrical shape (diameter: 2.3m, length: 5.9m)	
Surface contamination	Uniform contamination, 1 Bq/cm^2	
Annual shipments	50 times (1 cask per week)	
Time needed for shipments	4 h per 1 cask. (The time include monitoring, decontamination, handling, loading or unloading and other works)	
Distance from the package	RW	1 m from the package surface
	TR	3 m from the package surface with 3mm steel plate of vehicle's body
	GP	10 m from the package surface
Annual working time	RW	200 h/year (4 h × 50 times/yeara)
	TR	1000 h/year (half of annual working time for general worker)
	GP	200 h/year (4 h × 50 times/yeara)

RW = radiation worker; TR = transporter; GP = general public

Dose calculation methodology is similar to that of the Basic Model and considers a total of nine pathways, such as external, submersion, ground-shine, inhalation, three pathways for ingestion, including biosphere and two pathways for the skin. Details of the methodology are presented in [7.4].

Eighteen radio-nuclides are selected as representatives for calculation of doses to Radiation Worker (RW), Transporter (TR, vehicle driver) and to the General Public (GP). Direct radiation from radioactive content of the package is not considered in this model.

The realistic approach to scope practical doses considers marine transportation, which is a main mode for spent fuel transport in Japan. Surface contamination of 0.4 Bq/cm² for alpha emitters and 4.0 Bq/cm² for beta emitters is assumed on the 1 m² of the cask surface. The exposure scenario for the Realistic Model is summarized in Table 28.

Table 28. Exposure scenario for dose estimation by the realistic model

Worker	Annual working Hours (h/year)	Evaluation values
Onboard radiation control member (ORM)	10	0.5 (h/day), 2 (day/navigation), 10 (navigation/year)
Crew (CR)	16.7	10 (min/package), 10 (package/navigation) 10 (navigation/year)
Inspector at port (IP)	33.3	20 (min/package), 10 (package/navigation) 10 (navigation/year)
Handling worker including lashing work (HW)	33.3	(10 + 10) (min/ package), 10 (package/navigation), 10 (navigation/year)

External, internal exposure and skin contamination to workers involved in the transportation, such as the onboard radiation control member (ORM), crew (CR), inspector at port (IP) and handling worker, including lashing worker (HW) are evaluated under the condition that proper radiation protection measures are taken.

The calculation results from the dose and the safety margin for the Conservative Model can be summarized as follows.

- The maximum dose rate from external radiation due to ^{60}Co is 6.6 10^{-6} mSv/year to a radiation worker.
- The maximum dose rate from inhalation of ^{239}Pu is 1.5 x 10^{-1} mSv/year to a transporter.
- The maximum dose rate from ingestion of ^{239}Pu is 5.0 x 10^{-3} mSv/year to a radiation worker.
- The maximum dose rate to the skin due to ^{90}Sr is 6.3 x 10^{-2} mSv/year to a radiation worker.

The calculation results for the dose and the safety margin for the Realistic Model can be summarized as follows.

- The maximum dose rate for external radiation including doses from the radioactive content of the package is 0.83 mSv/year to handling worker.
- The maximum dose rate from inhalation of ^{239}Pu is 7.7 x 10^{-7} mSv/cask to an onboard radiation control member.
- The maximum dose rate for ingestion of ^{90}Sr is 5.0 x 10^{-7} mSv/cask to an onboard radiation control member.
- The maximum dose rate to the skin due to ^{90}Sr is 4.2 x 10^{-3} mSv/cask to an onboard radiation control member.

Sensitivity analyses were carried out and complementary cumulative distribution functions of dose rates for the Conservative Model and for the Realistic Model were obtained. A major uncertainty, which is sensitive to the results of the dose rate exists in the parameter for inhalation.

As a result of evaluations either by the Conservative Model or by the Realistic Model, the dose is confirmed to be lower than the dose limit set by the IAEA Transport Regulations [7.5].

REFERENCES TO SECTION 7

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8. CONTROL OF CONTAMINATION

8.1. Background

Following some investigations by the competent authority in France, DSIN, the existence of excess levels of contamination on flasks and wagons used to transport irradiated nuclear fuel was published in May 1998. The contaminated flasks and wagons had been used routinely to transport irradiated fuel between Germany, Switzerland and France, and some shipments to the UK may have been involved. This led to the cessation of these shipments for a period in several European Union (EU) member states including France, Germany and Switzerland. In the follow-up period, extensive surveys and investigations were carried out to establish the causes, nature and magnitude of the contamination, and to seek remedies [8.1]. These circumstances had an impact throughout the EU, and more thorough decontamination and monitoring procedures were introduced. In some cases, this led to increased exposures of the workers carrying out these duties. These events have raised the question of how to ensure the optimisation of radiation protection while maintaining the levels of non-fixed contamination below regulatory limits. Information, data and experience from four member states of the EU were reviewed in a study funded by the European Commission [8.2]. Investigations in France into flask monitoring and preparation for transport showed that considerable improvements could be made in preventing contamination and achieving dose savings for the workers involved. These findings, and other information published more recently, are summarised here.

8.2. Occurrence of contamination on nuclear fuel flasks

Nuclear fuel transport flasks may become contaminated with radioactive materials on both internal and external surfaces, from contact with water in the fuel storage ponds at nuclear installations when loading or unloading fuel. This effect has been well known and reported for several decades [8.3–8.7].

Transport flask surface contamination arises mainly from dissolved or particulate radioactive materials in the storage and cooling pond water. These materials are activated corrosion products from the primary reactor cooling loop and fission or activation products from the fuel elements.

Surface contamination of flasks may also occur by direct contact with the irradiated fuel elements. The extent and magnitude of surface contamination is also dependent on the concentrations and chemical forms of the radioactive constituents in the pond water, and the contact time in the pond. On UK flasks and wagons, the main potential contaminant is ^{137}Cs. This radionuclide is a soluble fission product, leached from spent fuel rods through stress-related pin holes in the fuel cladding when fuel elements are kept in the cooling ponds at each nuclear power plant (NPP), prior to despatch to the reprocessing factory or storage site. In most UK NPPs, fuel elements are loaded into a ‘skip’ - an open-top steel basket used to transfer elements into flasks. Generally, this takes place by immersing the flask into the pond water and this gives rise to flask surface contamination. Unlike flask surfaces, skips are not decontaminated, since they are held within the flask until delivered to the reprocessing site or returned to the NPP. Skips can be significant contributors of pond water activity and the more contaminated skips are frequently returned as rapidly as possible without delayed immersion in the ponds.

In pressurised and boiling water reactors (PWRs and BWRs), the main contaminants are the activated corrosion products ^{60}Co, ^{110}Ag, or ^{54}Mn and the fission product ^{137}Cs. In France the activity of the pond water has been reported to be up to 400 MBq/m^3. In order to limit occupational exposure during down-time, a special water chemistry treatment is undertaken just before a planned shutdown; in order to precipitate a large fraction of the corrosion and activation products. However, this is a major cause of deposition of particles on the fuel assemblies. In France, there are two main systems used for loading spent fuel into flasks, depending on the NPP type. The flask may be immersed in the pond or connected to the underside of the pond. The main surfaces of the flask are covered during these operations, but there are phases in both of these processes when pond or leakage water can contaminate parts of the flask surface.

8.3. Minimization of contamination

8.3.1. Techniques

The contamination management strategies generally available to plant operators may in broad terms be divided in to three categories: prevention, decontamination and minimisation [8.8]. In the present context, prevention and decontamination relate to operational practices and procedures, whereas minimisation of contamination pertains primarily to design-related issues at the planning stage of a facility, operation or flask. Information was collected for analysis, and comparison, from the four countries participating in the study [8.2]. The data collected included incidence of contamination, exposures of workers involved in the transport operations, and techniques adopted to reduce contamination incidence and exposures.

8.3.1.1. Removal of contamination from the storage or loading pond water

Contamination may be removed from the loading ponds by cleaning or filtration. However, both of these operations will result in worker exposure and will produce radioactive waste that will require either storage or disposal. In some NPPs, contamination on the skips used to hold the fuel elements has been found to be a significant factor in controlling pond water activity. The introduction of skip cleaning and maintenance facilities can yield reductions in pond water activity. In recent years in the UK increased pond concentrations of radionuclides were experienced, caused mainly by the inter-pond transfer of contaminated skips that are used to hold the fuel within the flasks. The use of ion exchange materials in cartridges has been effective in controlling this increase in concentration.

8.3.1.2. Protection of contaminated flasks during transport

The use of canopies or covers over irradiated fuel transport flasks during carriage is common in many countries. The covers protect the flasks from the effects of weather, particularly wind and rain, and restrict unauthorised access to the surfaces of the INF flasks during transport.

8.3.1.3. Design of containers to limit contamination

The effort of decontaminating surfaces should be considered during the design stage of INF flasks. However, there may be competing requirements, such as the number and design of cooling fins for heat removal.

8.3.1.4. Pre-wetting the flask

Many operators have found that wetting the surface of the flask with clean water before its immersion in the pond can reduce the uptake of contaminants on the flask surface.

8.3.1.5. Use of covers and skirts

Flasks become contaminated by pond water when immersed in the pond during loading or unloading of the fuel. A metal skirt fixed around the central area of the flask can be used to prevent contact with the pond water. In addition, a further vinyl overskirt is used over the body of the flask to cover the end parts of the flask beyond the area covered by the skirt.

8.3.1.6. Strippable sealant

Exposed areas at the ends of the flask can be covered with adhesive film or peelable paint which can be removed after the flask is taken out of the pond.

8.3.2. Good Practices Guides

Good practices guides of current working procedures [8.9], [8.10] have been issued for irradiated fuel containers and include techniques listed above.

8.4. Experiments into the mechanisms of 'weeping'

The phenomenon of 'weeping', or 'sweating', is characterized by the unexpected occurrence of non-fixed contamination on the surface of a spent fuel transport cask, at a level exceeding regulatory limits, and after prior demonstration that such contamination was within the allowable limits. Given that this phenomenon is not clearly understood, it is probable that, even if all currently suggested methods and procedures for mitigation of the contamination are consistently applied, the phenomenon will eventually, unexpectedly recur.

^{137}Cs and ^{60}Co are common radionuclides observed as non-fixed surface contamination when weeping incidents occur. A number of attempts have been made to provide a mechanistic explanation for the seemingly capricious occurrence of cask weeping incidents. Work has been performed to determine the mechanism(s) of non-fixed surface contamination on package surfaces. This work led to the conclusions that the contamination is: 1) an adsorption/desorption phenomenon; or, 2) a physical-chemical process in which radionuclides are incorporated (physically trapped and/or ionically bonded) into the oxide layer on the package surface. Re-evaluation of the literature suggests that while the phenomenological mechanisms previously conjectured were reasonable, a more likely mechanism for ^{137}Cs weeping is ion exchange involving clay particles affixed to the exterior surfaces of shipping casks.

Weeping incidents usually are associated with releases of ^{60}Co and ^{137}Cs, though on occasion other radionuclides may be involved. Generally, these two radioisotopes behave differently and may be influenced by different environmental factors. Studies to date have focused on the roles played by the surface coatings, paint and metal oxide passivating coatings, but have ignored the impact of road grime. This component could significantly influence the retention and later release of ^{137}Cs. It should be kept in mind that the various mechanisms discussed in the literature – and the mechanism examined in this study – are not mutually exclusive. Thus, depending on local history and environment all may operate in concert and it may not be possible to isolate a single cause for a particular weeping incident.

A coating of clay inherently exists on spent-fuel package surfaces. Caesium has a high affinity to clay; Cs^+ is ionically bonded between layers of clay. Fixed Cs^+ in the clay layers manifests as non-fixed contamination (weeps) via an ion exchange process when exposed to solutions containing preferential cations. Experiments were performed to: 1) confirm the physical / chemical process of caesium ion exchange as it occurs on spent fuel packages – this was confirmed; and 2) select an optimum 'wash' for mitigation of Cs weeping.

The ion-exchange mechanism was demonstrated for Cs adsorption/removal. There was no pH effect on Cs uptake, which supports the ion-exchange hypothesis. Soiling the surface resulted in a marked increase in ^{137}Cs retention (compared to a freshly washed – 'as received' surface), but produced only a small increment in the amount of ^{60}Co retained.

A factor of >100 reduction in absorbed Cs was achievable. Optimal Cs mitigation was achieved via a stable Cs solution pre-wash followed with multiple water or cation post-washes. Pre-treatment with non-radioactive Cs is the most effective process for minimizing the uptake of radioactive Cs on metal surfaces and markedly decreases retention of Cs. The uptake of Cs is correlated with the presence of road grime. None of the pre-treatments had an impact on Co uptake. Co uptake involves a different mechanism and is unrelated to the presence of road grime. Post-washing removes some Co, but the amount is not related to the chemistry of the wash fluids or the presence of road grime. Tap water, K^+, and NH_4^+ post-wash solutions work well for Cs removal for all pre-treatment conditions. In contrast to pre-treatment, post-washing with Cs is least effective apparently due to a reduced driving force for ion exchange. This trend holds for coupons pre-treated with tap water, K^+, NH_4^+ or with Cs^+.

Annex 6.2 contains a full report of this study and its results.

8.5. Harmonisation of procedures in the EU for assuring non contamination on containers for irradiated fuel transports

The European Commission, in accordance with Council Decisions 1999/21 and 1999/25 of December 14th, 1999, adopted multi-annum framework and programme for actions in the nuclear energy sector (1998 to 2002). These actions are related to the safe transport of radioactive materials and are divided into Fields. The study's aim is "Harmonisation of procedures for assuring non contamination on containers for irradiated fuel transports" [8.10].

The main objectives of the study are to establish guidelines for:

- prevention of contamination,
- practices of decontamination,
- procedures and equipment implemented for measuring non-fixed contamination.

These objectives are treated according to the following fields:

- inventory of the current practices for clean transportation of irradiated fuel flasks in several countries (France, Germany, UK, Sweden, Switzerland),
- analysis and comparison of the different methods,
- proposal of recommendations based on best practices.

The report contains:

- a description of the different types of fuel assemblies.
- a description of different types of fuel transportation flasks (divided into three categories: with cooling fins, with cooling ribs and with smooth body) used in the European countries,
- an overview of the spent fuel management and associated transportation in Europe (between nuclear power plants, interim storage plants, railway and marine terminals and reprocessing plant).

The recommendations are to keep the spent fuel pools as clean as possible and to use a plastic or vinyl overskirt around the whole flask when it must be immersed for loading or unloading operations. On the other hand, the under-pit loading is the best practice.

Decontamination processes are divided into five categories: mechanical, chemical, physico-chemical, electrochemical, and use of ultrasonic waves. After a comparison of the different techniques in nuclear power plants, it appears that automatic pressure washing seems to be the best method in continuation with the wetting of the flask before immersion. For reprocessing plants, automatic pressure washing, electrochemical decontamination and the ultrasonic wave method seem to be good practices.

The participants of the study have successfully exchanged data on implemented decontamination processes, protective means and equipment for measuring non-fixed contamination. Differences exist between countries, but all implemented techniques are compatible to achieve the intended result and are leading today to the clean transportation of irradiated fuel flasks.

However, the participants of this CRP recommend the following :

- to load spent fuel flasks under-pit, because it avoids the contact between the outer flask surfaces and the contaminated pool water,

- to provide a complete external protection of the flask (plastic overskirt or plastic skirt), when immersing the flask for loading/unloading,
- to wet the flask prior to its immersion into the spent fuel pool,
- to clean the flask thoroughly (manual or automatic) before transportation in case of wet loading or unloading,
- to use the same reference source for calibration,
- to use probes having similar performances, and sensibilities,
- to implement a quality management process for self-checking of the operator; and to decrease the number of routine checkpoints on flasks,
- to use the same check points for non-contamination on the flasks in all procedures.

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9. GENERAL DISCUSSION AND CONCLUSIONS

9.1. Overview

This document presents and summarises the results of a multi-national CRP undertaken with the objectives of reviewing and evaluating the scientific basis for the current regulatory limits for non-fixed surface contamination on packages and conveyances and of developing guidance material for evaluating the radiological significance of surface contamination for transport workers and the public.

The approach taken was to characterise and model the dose pathways for non-fixed contamination on packages and conveyances under normal transport conditions and to calculate potential radiation doses to workers and the public. The transport and exposure conditions reflected in the Basic Model do not reflect a particular transport situation or geographical transport region, but are believed to provide a practical and reliable means for the assessment of occupational and public exposures that are unlikely to be exceeded under actual transport conditions.

The following sections cover the observations on the results of the Basic Model and how these may be applied to derived limits for non-fixed surface contamination. However, since no choice is made on the appropriate dose criteria, no numerical limits are derived in this report.

9.2. Observations on the results of the Basic Model

The CRP members agreed at the beginning of the CRP work that the Fairbairn model [9.1] was limited and outdated and that a new comprehensive model (the Basic Model) should be produced to provide the basis for eventually deriving new contamination limits for transport situations. It was agreed that the Basic Model should provide individual doses for at least 49 selected radionuclides. Results for each of the 356 radionuclides that are listed in Table I of the Regulations TS-R-1 [9.2] were also calculated. The results for 49 radionuclides for four different package types have been presented in Section 6 of this TECDOC.

The following observations can be made with regard to the dose conversion coefficients (i.e. the annual dose values, which are calculated for unit contamination, 1 Bq/cm², on the package):

- With respect to the entire set of 49 nuclides, the values of the annual dose per unit surface contamination span nearly seven orders of magnitude for workers and nearly eight orders of magnitude for members of the public.

- The highest of all values of annual dose per unit surface contamination (both for workers and the public) is calculated for ^{227}Ac. ^{227}Ac decays by both alpha and beta emission, emitting a range of alpha and beta particles together with electrons and gamma ray photons. This radionuclide produced doses of at least an order of magnitude higher than all other radionuclides assessed in this study, and is several orders of magnitude higher than the majority of radionuclides. However, ^{227}Ac is not normally associated with practical contamination problems in transport.

- The lowest range of values for the annual dose per unit surface contamination is calculated for very weak beta/gamma emitters like ^{59}Ni which have an extremely low radiological relevance.

- Special consideration should be given to tritium, since this radionuclide raises important difficulties for prevention of contamination as tritium is present in water or steam form.

- The categorisation of radionuclides into ‘beta and gamma emitters’, ‘low toxicity alpha emitters’, and ‘all other alpha emitters’ (vide paragraph 214 of TS-R-1) with only two values (4 and 0.4 Bq/cm²) does not represent the span of seven orders of magnitude in the results of the Basic Model.

- Furthermore, the distinction between "low toxicity alpha emitters" from the other alpha emitters is not valid in the context of the results of the Basic Model because some of the low toxicity alpha emitters listed in TS-R-1 are responsible for some of the highest doses, these being ^{232}U, ^{232}Th and Th-nat.

- Worker doses were found to be about three orders of magnitude higher than doses for critical members of the public for all radionuclides. Therefore, in deriving revised limits for the non-fixed surface contamination, workers would be more limiting than members of the public.

- The dominating package type for the majority of radionuclides was 'small, manually handled packages', although there is little difference between the maximum doses obtained for all package types, assuming the same level of surface contamination. For each of the strong gamma emitters, the dominant package type was small, remotely handled packages, where worker F was the limiting worker (cf. Table 23).

The Basic Model assumes a uniform contamination of the package surface. In reality, small, manually handled packages are very rarely contaminated, and all other package types that are contaminated tend to have non-uniform contamination on the surface. It is possible that a scaling factor could be incorporated into the results to take this into account, but as this factor would vary from package to package it would create added uncertainty. Care should be taken if the results for small, manually handled packages are used for future regulations, and non-uniformity of contamination should also be considered. Small, remotely handled packages often contain waste, regularly consisting of surface contaminated materials, such as used protective clothing or cemented waste. As a result of the contents and environment in which these packages are prepared, the possibility of surface contamination on small remotely handled packages is much higher than that of new packages containing medical and industrial isotopes. Fuel flasks are the package type most likely to be contaminated and also, therefore, the package type most likely to be affected by any changes to the regulations.

9.3. Specifying surface contamination limits

The Basic Model can be used to establish a direct relation between the non-fixed contamination level per unit area on the package and the annual dose limit for workers or public for each radionuclide. However, no such relation has been worked out because the CRP did not address the issue of making decisions on dose limits. The Basic Model results are linear and can be simply scaled for any given dose criterion. That is, given a dose criterion, surface contamination levels can be derived with the Basic Model.

Regardless of the choice of the value of the dose limit or dose criterion as the basis for deriving surface contamination limits, consideration must be given to the way in which the final results are presented. There are at least three basic options:

1. Values rounded to 1 or 2 significant digits for each nuclide;

2. Values rounded to orders of magnitude for each nuclide;

3. Grouped values: The grouping process may be based either on properties of the nuclides (i.e. "alpha emitters with a high radio toxicity", "strong beta/gamma emitters", "weak beta emitters" etc.) or on numerical considerations.

At the time of the completion of the CRP work, no decision was made on this topic. It would be, however, an easy task to provide radionuclide specific rounded values (according to points 1 or 2 of the above list) for each radionuclide in a separate column of Table I in Section IV of the Regulations TS-R-1.

If radionuclide specific values were chosen, a summation rule such as the one described in paragraph 404 of the Regulations TS-R-1, would need to be used for mixtures of radionuclides.

9.4. Practical considerations when performing contamination measurements

9.4.1. Radionuclide specific approach

When taking a radionuclide specific approach, it may be argued that radionuclide specific surface contamination levels might require extra effort when performing the measurements, as it may appear to warrant spectroscopic measurements every time surface contamination measurements are performed. This, however, would be a misconception because of the following reasons:

- It is necessary to know the key radionuclides for performing surface contamination measurements. Otherwise, no interpretation of the readout of the instrument would be possible. If a high percentage of nuclides present are hard to measure, then one should establish scaling factors between those nuclides and easy-to-measure nuclides (key nuclides) and derive the activity of the hard-to-measure nuclides using the scaling factors.

- If spectroscopic measurements are used to establish the initial nuclide vector ("fingerprint"), these results could be normally used. The radionuclide mixture is not likely to change dramatically for a particular working situation.

- From the readout of a surface contamination monitor it is simple to decide whether the radionuclide specific surface contamination limits are complied with or exceeded. The appropriate readout of the instrument (e.g. given in counts per second or Bq/cm²), which corresponds to the limit value for the specific radionuclide mixture, could be determined before starting the contamination measurements and could then be used for all subsequent measurements on surfaces with similar radionuclide vectors (fingerprints).

9.4.2. Radionuclide group approach

If a radionuclide group approach is taken, it will be necessary to determine which radionuclides are present and into which grouping they are to be categorized. Depending on the limits applicable to each radionuclide group, the measurements would be done on a gross alpha or gross beta basis, as in the current system.

9.5 Conclusion

The CRP is a culmination of the efforts taken by many researchers all over the world to determine the dose that may result from the presence of non-fixed contamination on the external surfaces of packages and conveyances. Methods for estimating the dose values were evaluated during the CRP. The studies carried out under the CRP indicate that the present limits on non-fixed contamination on the surfaces of packages and conveyances are conservative.

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Pre-meeting

Paris, France: 2–4 May 2001

Research Coordination Meetings

London, United Kingdom: 7–9 November 2001
Williamsburg, VA, United States of America: 18–21 June 2002
Berlin, Germany: 19–22 November 2002

Modelling Group Meetings

Chilton, United Kingdom: 14–15 February 2002,
Essen, Germany: 29–30 April 2002
Cologne, Germany: 21–22 August 2002

Consultant Service Meeting

Vienna, Austria: 17–21 March 2003