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Methodology for Safety Assessment Applied to Predisposal Waste Management

Report of the Results of the International Project on Safety Assessment Driving Radioactive Waste Management Solutions (SADRWMS) (2004–2010)

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METHODOLOGY FOR
SAFETY ASSESSMENT APPLIED TO
PREDISPOSAL WASTE MANAGEMENT

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IAEA-TECDOC-1777

METHODOLOGY FOR SAFETY ASSESSMENT APPLIED TO PREDISPOSAL WASTE MANAGEMENT

REPORT OF THE RESULTS OF THE INTERNATIONAL PROJECT ON SAFETY ASSESSMENT DRIVING RADIOACTIVE WASTE MANAGEMENT SOLUTIONS (SADRWMS) 2004–2010

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FOREWORD

The International Project on Safety Assessment Driving Radioactive Waste Management Solutions (SADRWMS) was launched in November 2004 and ran until April 2010. The SADRWMS Project focused on approaches and mechanisms for the application of safety assessment methodologies for the predisposal management of radioactive waste. It also gave consideration to how assessment results could be interpreted and used in decision making during design development and modification, safety upgrades, periodic safety assessment and licensing activities.

The initial outcome of the SADRWMS Project was achieved through the development of flowcharts. These flowcharts have since been incorporated into IAEA Safety Standards Series No. GSG-3, The Safety Case and Safety Assessment for the Predisposal Management of Radioactive Waste, issued in 2013. In 2005, an initial specification was developed for the Safety Assessment Framework (SAFRAN) software tool to apply the SADRWMS flowcharts. In 2008, an in-depth application of the SAFRAN tool and the SADRWMS methodology was carried out on the predisposal management facilities of the Thailand Institute of Nuclear Technology (TINT) Radioactive Waste Management Centre (known as the TINT Facility).

In 2010, the IAEA presented the draft of GSG-3 to the 29th meeting of the Waste Safety Standards Committee for approval. During discussions, it was proposed that the use of the graded approach be illustrated through the development of lower tier TECDOCs documenting application of the methodology on typical predisposal radioactive waste management facilities and activities. As the output of the SADRWMS Project provided input to the development of GSG-3 and was applied to the TINT Facility, the SADRWMS methodology is considered the first such TECDOC. The TECDOC will enable users to develop an initial framework for safety assessment for radioactive waste management through the application of the GSG-3 methodology and the SAFRAN tool. Subsequent TECDOCs will illustrate the use of the graded approach on representative predisposal management of radioactive waste facilities and activities.

This publication summarizes the content and outcomes of the SADRWMS Project. The Chairman's Report of the SADRWMS Project and the Report of the TINT test case are provided on the CD-ROM which accompanies this publication.

The IAEA wishes to express its gratitude to all those who participated in the work of the SADRWMS Project and gratefully acknowledges those who contributed to the preparation of the resulting reports. The IAEA officers responsible for the overall running of the SADRWMS Project was initially L. Jova Sed and subsequently J. Raicevic and M. Kinker of the Division of Radiation, Transport and Waste Safety.

CONTENTS

1. INTRODUCTION

1.1. BACKGROUND

In 2004, the IAEA organized the International Project on Safety Assessment Driving Radioactive Waste Management Solutions (SADRWMS) to examine and harmonize the approach to safety assessment for predisposal management of radioactive waste. The SADRWMS Project also gave consideration to how the assessment results could be interpreted and used in decision making during design development and modification, safety upgrades, periodic safety assessment and licensing activities.

The SADRWMS Project was developed to complement the experience gained with the IAEA's projects "Improvement of Long term Safety Assessment Methodologies for Near Surface Disposal Facilities" (ISAM) [1] completed in 2000. The SADRWMS Project ran over a six year period, between November 2004 and April 2010. The initial outcome of the SADRWMS Project was achieved through the development of flowcharts to improve the mechanisms for applying safety assessment methodologies to predisposal management of radioactive waste. These flowcharts have since been incorporated into the IAEA General Safety Guide on the Safety Case and Safety Assessment for the Predisposal Management of Radioactive Waste (GSG-3) [2].

GSG-3 provides guidance for safety case and safety assessment for facilities and activities associated with the predisposal management of radioactive waste; and emphasizes the importance of ensuring that the extent and complexity of the assessment is commensurate with the nature of the activity or facility and its attendant risk. When the Secretariat presented GSG-3 to the 29th meeting of the Waste Safety Standards Committee in 2010 for approval, during discussions it was proposed that the use of the graded approach be illustrated through the development of implementing publications documenting application of the methodology on typical waste management facilities and activities.

In 2005 an initial specification was developed for the Safety Assessment Framework (SAFRAN) [3] software tool to apply the SADRWMS flowcharts. Between 2005 and 2008, a number of test cases were used on a limited basis in order to refine the SAFRAN Tool and develop the supporting documentation. In 2009, an in-depth application of the SAFRAN tool and the SADRWMS methodology was carried out on the predisposal management facilities of the Thailand Institute of Nuclear Technology Radioactive Waste Management Centre (TINT Facility).

As the output of the SADRWMS Project provided input to the development of GSG-3 and was applied to the TINT Facility, it is expected that this TECDOC will enable users to develop an initial framework for safety assessment for radioactive waste management through the application of the GSG-3 methodology and the SAFRAN tool. Subsequent publications will illustrate use of the graded approach on representative predisposal management of radioactive waste facilities/activities.

1.2. OBJECTIVE

The objectives of the SADRWMS Project are as follows:

- To improve mechanisms for application of safety assessment methodologies for predisposal management of radioactive waste;

- To provide illustration and practical advice on application of safety assessment methodologies;
- To develop an integrated and graded approach to addressing a large variety of radioactive materials management challenges;
- To enhance confidence and public acceptability of predisposal radioactive waste management practices by scientific safety assessment approach;
- To coordinate approaches to regulatory and peer review and justification of safety assessment and development of associated procedures, reflecting state of the art international practice for predisposal management of various types of radioactive waste.

The objective of this publication is to enable users to develop an initial framework for application of safety assessment to predisposal management of radioactive waste using an internationally agreed methodology. It outlines the key components of the safety assessment within the context of predisposal waste management, describes the SADRWMS methodology (what is needed in the way of safety justification for establishing the context of the assessment, screening of hazards, assessment of doses during normal operations and accident scenarios), and explains the implementation of the SADRWMS methodology in the SAFRAN Tool.

The methodology described in the document is applicable to all predisposal waste management facilities and activities (e.g., waste identification, remedial action, clearance or discharge, processing, storage), and to all types of radioactive waste (e.g., disused sources, limited volumes, operational waste, spent fuel declared as waste, legacy and decommissioning waste, large volume NORM residues). The target audience includes experts and organizations with responsibilities for safety assessment of predisposal radioactive waste management facilities and activities, managers and facility operators and their technical experts who are involved in such safety assessment; as well as persons from regulatory authorities evaluating such safety assessments for the purposes of licensing and regulatory control.

1.3. SCOPE

The scope of the SADRWMS project is on approaches and mechanisms for application of safety assessment methodologies for the predisposal management of radioactive waste; encompassing all types of radioactive waste including disused sources, limited volumes, operational waste and spent fuel, legacy and decommissioning waste, and large volume NORM residues. The scope of the project includes the overall processes of predisposal waste management as well as the following process steps:

- Remedial action;
- Clearance;
- Treatment;
- Storage; and
- Disposal (operational issues).

1.4. STRUCTURE

The framework developed within the SADRWMS project and used as the basis to develop guidelines for the application of safety assessment methodologies is discussed in Section 2 of this report. Section 3 gives an overview of the key components in the approach to safety assessment, and Section 4 describes the assessment context for specific predisposal waste management processes. Section 5 describes the implementation of the safety assessment methodology into the SAFRAN Tool. Sections 6 and 7 describe the methodologies for screening of hazards and dose assessments for normal operation scenarios and accident scenarios, respectively, as they are implemented in the SAFRAN tool.

The report includes 3 Annexes. Annex I describes the SAFRAN models for evaluation of worker exposure resulting from accidental release of airborne radioactive materials and user's guide. Annex II describes the SAFRAN models for evaluation of public exposure resulting from accidental release of airborne radioactive materials and user's guide. Annex III provides a description of the SAFRAN models for evaluation of external exposure.

The Chairman's Report of the SADRWMS Project and the Report of the TINT Facility test case are given in the CD-ROM which accompanies this report.

2. WASTE IDENTIFICATION AND PREDISPOSAL WASTE MANAGEMENT PROCESSES

In the SADRWMS project a framework for the overall process of predisposal waste management was developed. This can serve as basis to develop guidelines for the application of existing safety assessment methodologies and the identification of what is needed in the way of safety justification.

In support of this activity, flowcharts have been developed covering the main steps in predisposal waste management. The emphasis lies on waste orientated activities, and other aspects such as political considerations and engineering aspects are not considered. Figures 1 to 6 provide an overview of predisposal waste management activities. Figure 1 describes the general process. Figures 2 to 6 provide details for the individual process steps defined in Figure 1. Figures 2 to 6 indicate activities requiring safety assessment by boxes with a shaded background. An acronym identifying the type of safety assessment required is indicated at the top of each of these boxes. In the following, a description of the activities indicated in the flowcharts is provided. The purpose and scope of the required safety assessments is described in Section 4.

2.1. OVERALL PROCESS

The first activity in the overall flowchart shown in Figure 1 is identification of the type of waste. This has to address all parameters for the particular type of waste required to decide about its classification in terms of the flowchart. An important distinction arises between waste types that already exist and which are kept in a storage facility, as opposed to waste that is newly generated. In cases of existing waste that has been put into a storage facility in the past, the safety and security of these storage arrangements may not be adequate based on current standards. This may require remedial action to upgrade safety and security measures by changes in the condition of the waste, improvements of the storage facility and/or retrieval of the waste and storage in another facility.[1]

For new waste as well as for waste retrieved from an old storage facility the next step consists in determining whether processing is required and, if so, which type of processing is necessary to allow for safe and secure storage of the waste. Ideally, the processing of the waste will also be planned and conducted such that the waste is suitable for later transport and disposal.

After processing to the extent required, the waste will be put into a storage facility unless direct disposal is possible. This storage facility serves as a hold point for the time required to establish a suitable disposal facility.

[1] The decision to consider remedial action, i.e. an intervention, for waste already in storage in TAble 1 does not apply to waste that is in interim storage pending processing within a practice. Rather, it applies to waste for which the decision to store it in their current form already has been made, so that any changes would be considered as an intervention. Waste in interim storage would be treated like newly arising waste within a practice and it would be decided within the processing box whether processing is required.

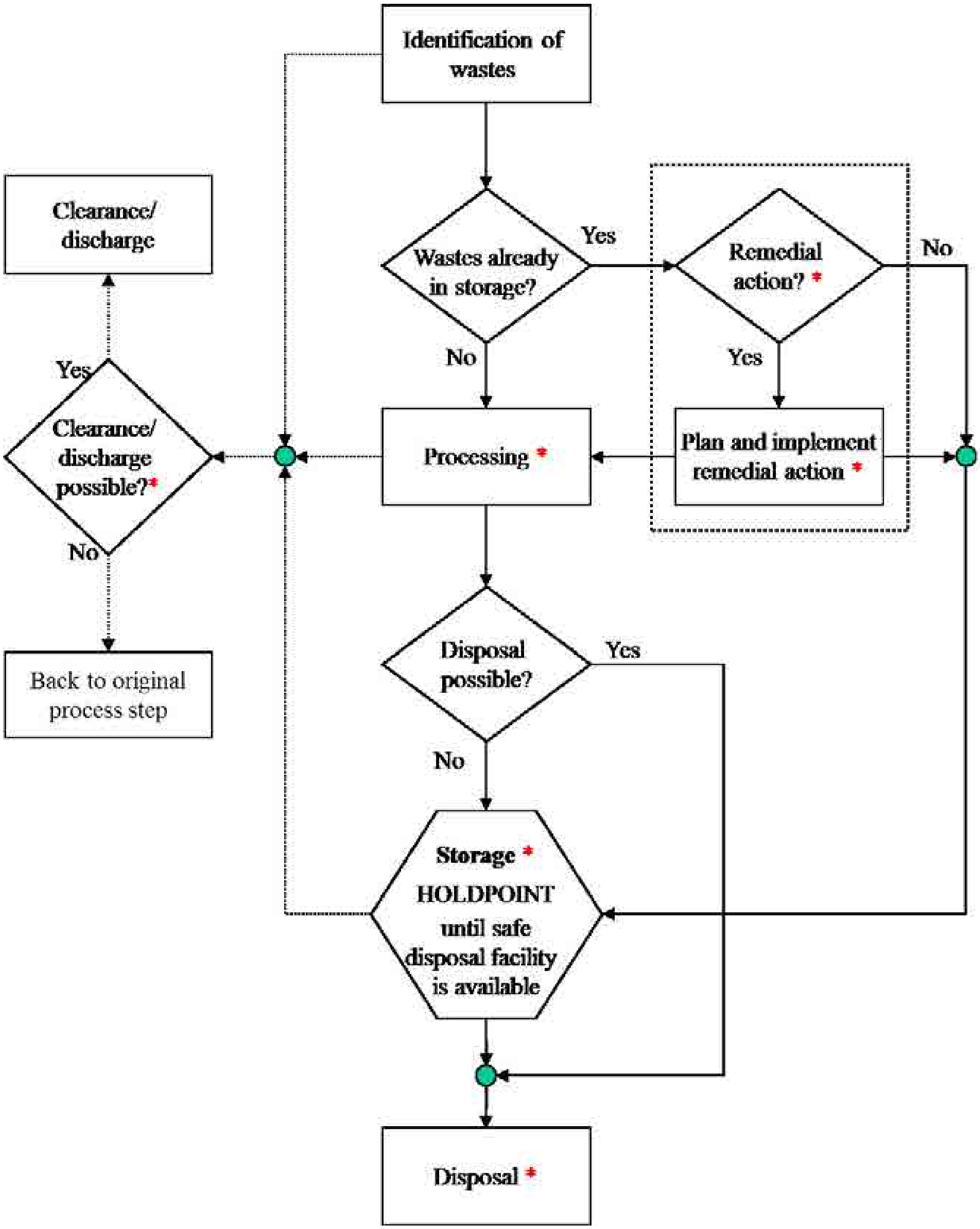

FIG. 1. Overall process.

Note: asterisks indicate activities that necessitate further steps which have decisions and safety assessments associated with them, as shown in Figures 2 to 6 (see also footnote 1).

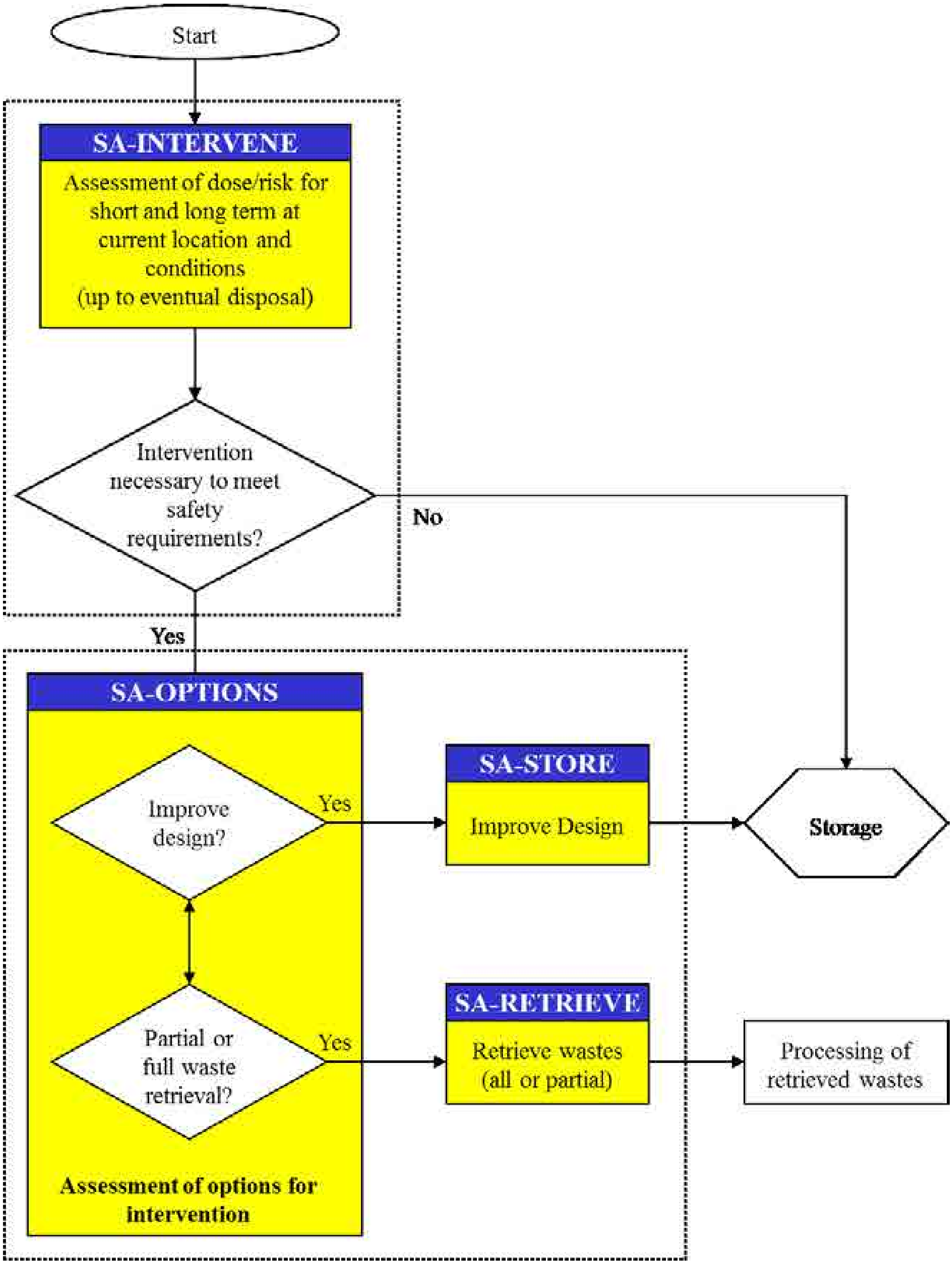

FIG .2. Remedial action?

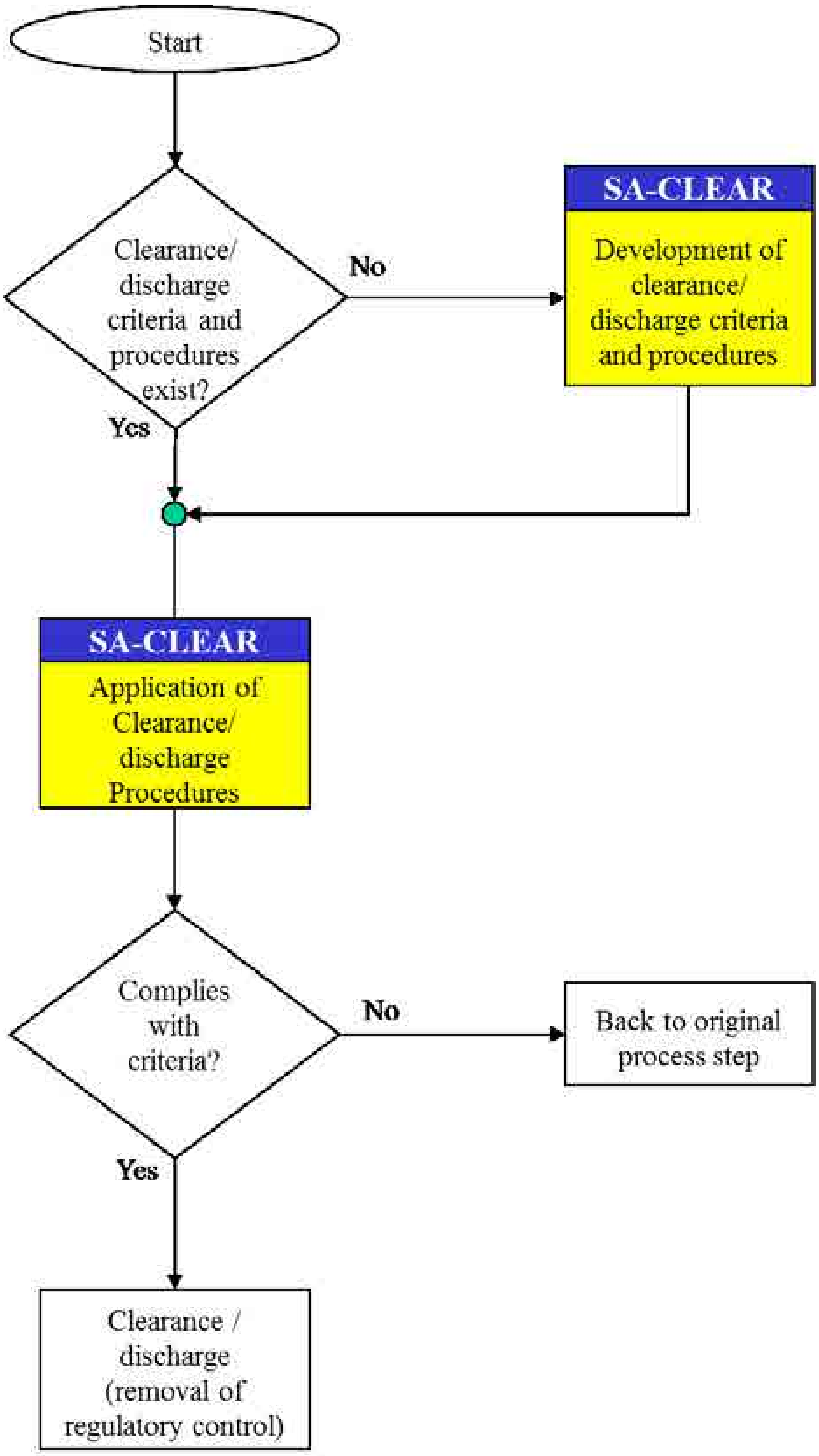

FIG. 3. Clearance/discharge possible?

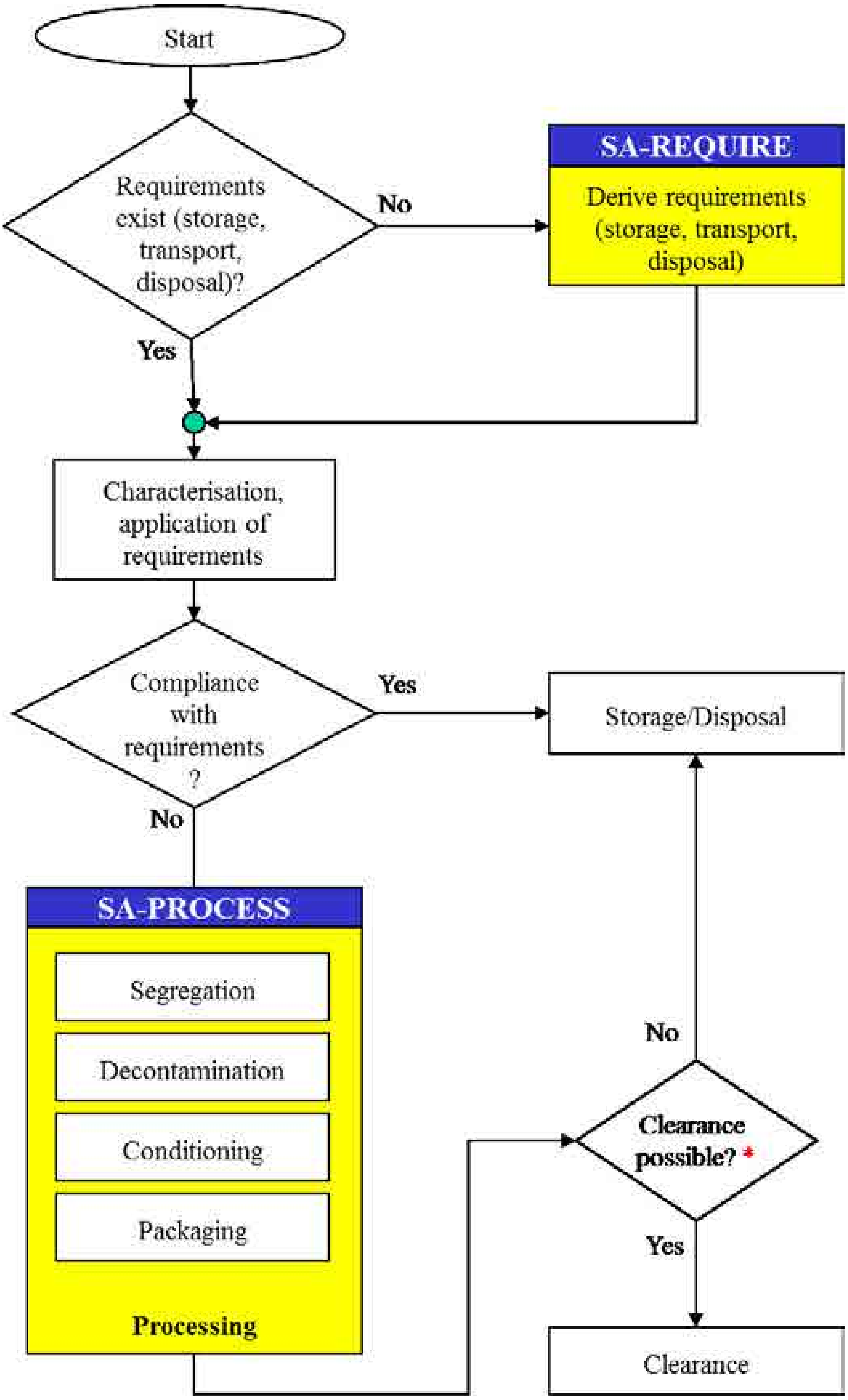

FIG. 4. Processing

Note: the asterisk indicates activities that necessitate further steps which have decisions and safety assessments associated with them (see also footnote 2).

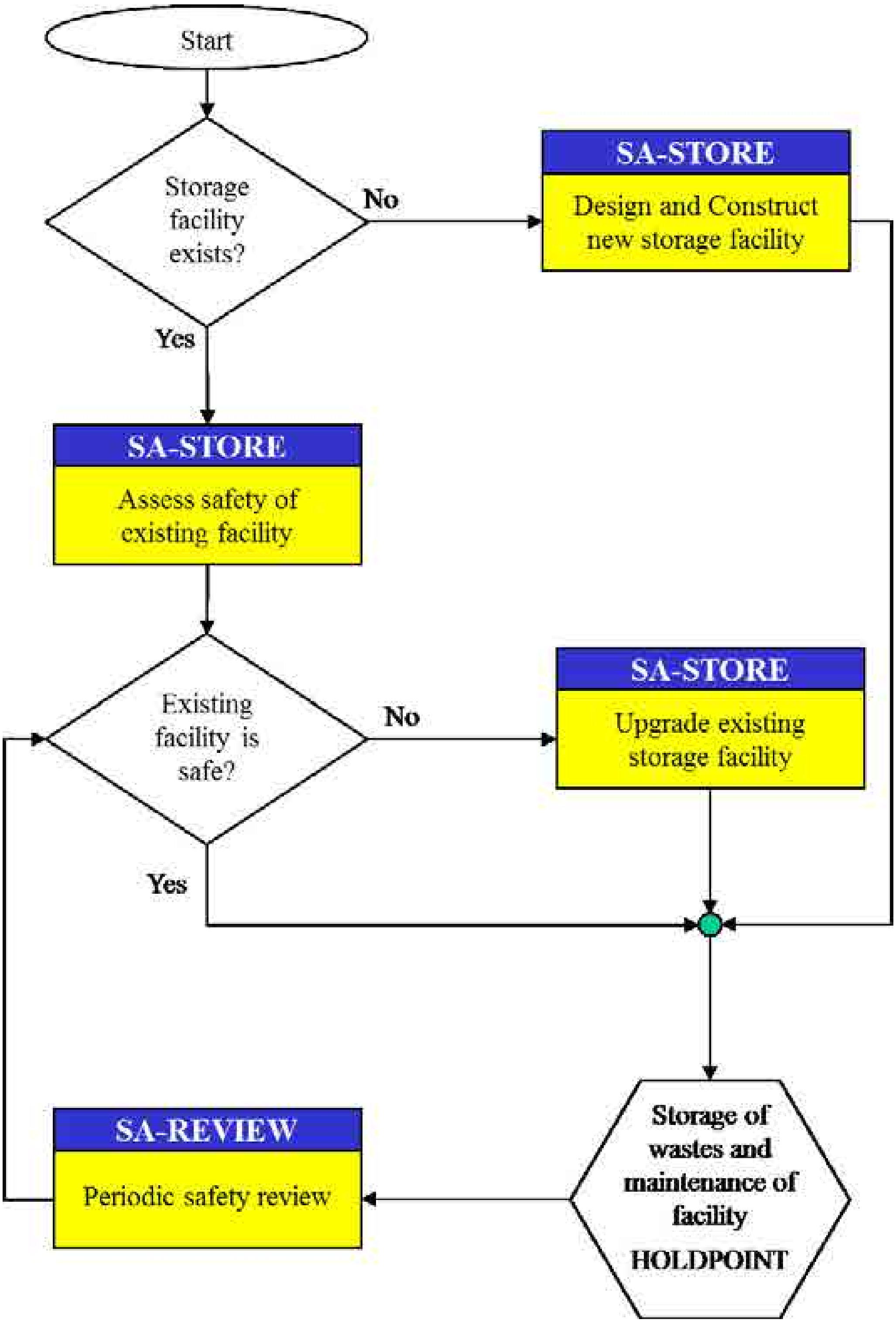

FIG. 5. Storage.

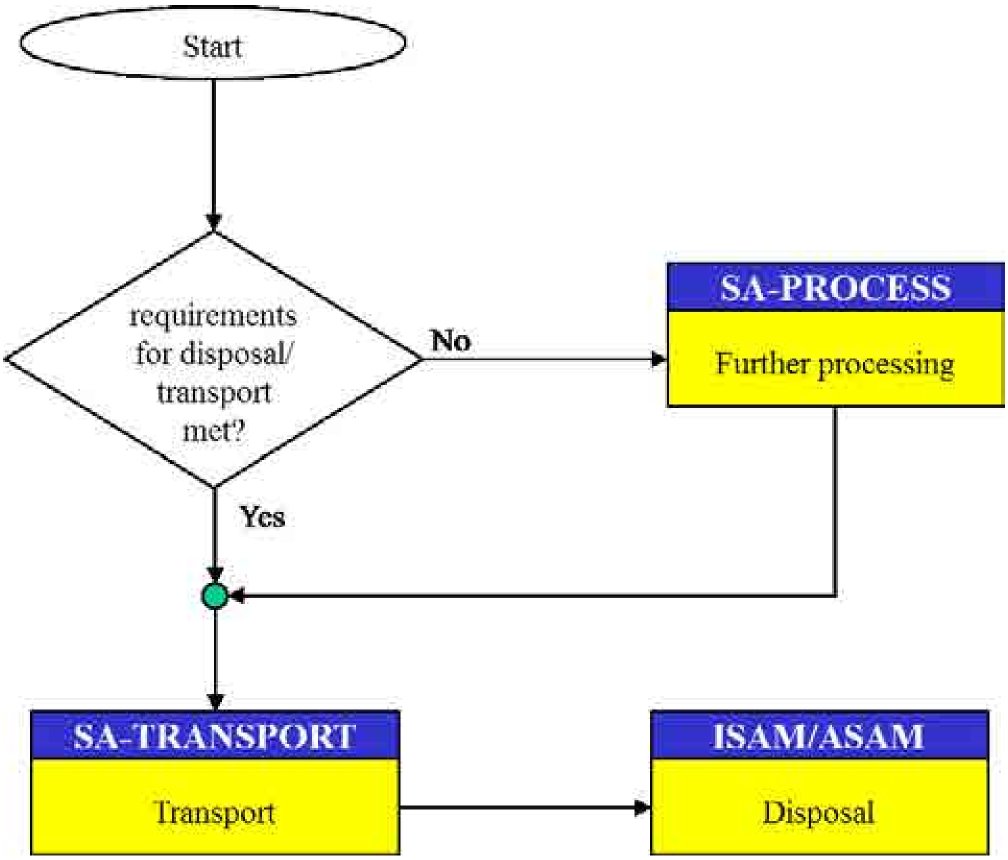

FIG. 6. Disposal.

During all stages of this process it may be possible to clear the waste, i.e. to remove it from regulatory control and dispose of it as non-radioactive waste or to recycle the waste material (e.g. in the case of metals). Clearance from regulatory control is a waste management option that may be available already at the very beginning of the process, i.e. following identification of the waste. Alternatively, clearance may be considered at later stages of the process because the option to clear waste may only be available after processing of the waste (segregation, decontamination) or after storage for radioactive decay.

For liquid or gaseous waste, an analogous waste management option consists in their discharge. As described for clearance above, discharge may be an option at any stage of the overall process. Examples of discharge of waste at later stages are discharge of liquid or gaseous waste arising during waste management activities (in particular, processing) and discharge of liquids after storage for radioactive decay.

2.2. WASTE IDENTIFICATION

In order to determine adequate management options for the waste in question, several of its key characteristics need to be known, such as:

- Liquid, solid or liquid/solid mixture;
- High or low dose rate;
- Dominant radionuclides: long lived or short lived;
- Flammable or non-flammable;
- Explosive or non-explosive;

- Containing alpha particles or not;
- Corrosive or non-corrosive;
- Gas emitter or non-gas emitter;
- Fissile or non-fissile;
- Contained or not contained;
- Well-contained or poorly contained;
- Records available;
- Waste properly labeled.

Waste characterization at this stage, however, is only of general nature and performed only to the extent necessary to decide about the further course of action and about immediate measures that might be necessary (e.g. to improve security or emergency response provisions). The collection of detailed data is performed as part of the preparation of safety assessments at later stages of the process to avoid sampling and measurements that are unnecessary (e.g. detailed chemical and physical characterization of waste that is later on identified as a candidate for clearance or discharge).

2.3. REMEDIAL ACTION

In case of waste in an old storage facility (not in interim storage as part of a current practice as explained in footnote of page 4), remedial action may be necessary to upgrade safety and security (Figure 2).

The first question to be addressed is whether the existing situation is acceptable from a safety and security point of view or whether corrective actions to upgrade safety and security are necessary. This means that only the question of the necessity to consider corrective action is addressed, not the question of which corrective action would be taken (in the event that this is considered necessary). The safety assessment required at this stage (SA-INTERVENE) considers in particular doses and risks arising from the current location and condition of the waste. The time span to be considered reaches up to the time at which it is anticipated that a disposal facility for the waste will become available.

If this safety assessment indicates the need for an intervention, it is necessary to identify and evaluate options to improve the situation (SA-OPTIONS). This may necessitate improvements in the design of the storage facility and/or the full or partial retrieval of the waste.

In the event that an intervention is found necessary within SA-INTERVENE, this safety assessment will in practice probably be combined with the safety assessment SA-OPTIONS to determine the type and extent of intervention. Nevertheless, these two safety assessments have different scopes and will be carried out consecutively. Therefore, they are treated separately from a methodological point of view.

In the event that the waste is being retrieved from an existing storage facility, the retrieved waste will be treated analogously to newly arising waste, i.e. options for its processing and safe storage and, when available, disposal will be determined. Special safety considerations are, however, necessary for the retrieval of waste. This is the case particularly when waste was stored originally without or with limited processing and in an unsuitable form (e.g. no packaging). The planning and execution of such retrieval activities will be based upon the safety assessment SA-RETRIEVE.

For the storage of wastes after retrieval and processing, the existing facility may be used, normally after the implementation of measures to upgrade its safety and security. Alternatively, such waste may be stored in another existing or in a new facility. The safety assessment SA-STORE required at this stage is in principle identical to the safety assessment required for storage in the case of newly arising waste which is discussed in Section 2.6

2.4. CLEARANCE OR DISCHARGE

Clearance (mainly for solid waste) and discharge (for liquid and gaseous waste) are important options for reducing the volume of waste to be stored and eventually to be disposed of. In some cases (e.g. stainless steel) the economic value may also provide an incentive to clear the waste.

The first question shown in Figure 3 is whether criteria and procedures for clearance or discharge, as appropriate, exist. If this is not the case these need to be developed (SA-CLEAR[2]).

With regard to clearance levels, the generic approaches recommended in Ref. [4] can be applied. Alternatively, specific clearance criteria and procedures can be developed for certain waste types or for certain disposal or recycling options. In the latter case, criteria for conditional clearance may be derived, i.e. regulatory control will only be removed if the waste producer can assure the regulatory body that certain restrictions on the disposal or recycling of waste are being complied with.

Guidance on the development of criteria and procedures for discharges is given in Ref. [5].

After the development of clearance and discharge criteria and procedures, the waste in question will be subject to these and it will be determined whether clearance or discharge is possible. The aim of the safety assessment SA-CLEAR is to provide, as part of the developed procedures for sampling and measurements, requirements for this decision.

If the waste complies with these criteria, it can be cleared or discharged. Otherwise, the waste remains with the overall scheme of radioactive waste management and will be subject to the appropriate processing step according to Figure 1.

In the case of unconditional clearance, the waste will be removed from regulatory control. For conditional clearance and discharge in general some regulatory requirements will remain, such as ensuring that clearance and discharge are performed according to the specified restrictions and prescribing, in particular in the case of discharges, requirements for monitoring.

2.5. PROCESSING

Processing of waste consists of any operation that changes the characteristics of waste, including pretreatment, treatment and conditioning. The goal of processing is to modify the

[2] For the sake of brevity, the acronym for this safety assessment refers to clearance only, but criteria and procedures for discharges are also addressed as appropriate.

waste form, as necessary, to comply with the requirements for its storage, transport and disposal (Figure 4).

If such requirements do not exist, they will need to be developed before any decision about waste processing can be made (SA-REQUIRE). As already stated ideally at this stage requirements for all further waste management steps, including transport and disposal will be derived. This avoids the necessity of further processing of the waste at a later stage, which would be economically unfavorable and which would also, if avoidable, conflict with the overall requirement to optimize the process. In practice, however, this will not be possible in all situations such as in the frequently occurring case in which a disposal facility or planning for such do not exist.

After development of requirements, or if these already exist, the waste in question will be characterized to the extent necessary in order to determine whether it complies with these requirements or not. The aim of the safety assessment SA-REQUIRE is to provide the necessary specifications for the required characterization.

If the waste in its current form does not comply with the requirements, processing is necessary. This may involve the following main steps:

- Segregation of waste types that are subject to different types of treatment, clearance and/or discharge;
- Storage of the waste to allow radioactive decay to facilitate its treatment or allow for clearance or discharge;
- Conditioning and packaging of the waste.

After processing, the waste will be sent for storage or disposal. Segregated or decontaminated portions of the waste that could potentially meet clearance or discharge levels will be subject to the application of clearance or discharge procedures (see Section 2.4).

The detailed activities associated with waste processing can be quite complex. Depending on the nature of the waste and the required changes of its chemical and physical form, risks for workers as well as for the public and the environment will have to be considered. These are addressed in the safety assessment SA-PROCESS carried out for the facility in which the waste processing is being performed and for all relevant activities therein.

2.6. STORAGE

As already discussed in Section 2.1, storage of waste is considered only as a hold point until a disposal facility becomes available. However, since in many States disposal facilities are not available and will not be available in the short term, safe and secure storage arrangements play an important role in the overall management of radioactive waste (Figure 5).

The first question arising is whether a storage facility already exists. In the case of an existing facility it is necessary to assess whether this facility allows for safe and secure storage of waste. If this is not the case, upgrading of the facility will be necessary. In this case, the situation is comparable to what is set out in Figure 2, for assessing the adequacy of storage arrangements for existing waste.

If no facility exists so far, it will be necessary to design and construct a new facility, with account taken of the safety and security requirements for the particular types of waste that have to be stored.

The safety assessment SA-STORE for addressing the adequacy of a storage facility will be in principle identical in both cases. The main difference arises from the fact that assessments will be based on the current situation and on options for its improvement in the case of an existing facility, while for a new facility the intended design will form the basis for the assessment.

After a storage facility has been commissioned, periodic safety review will be necessary in particular in the case of extended storage periods. Parameters that need to be addressed include changes in waste forms or containment structures as well as the appropriate functioning of all safety and security related systems. Details of the required review procedures will be determined by the safety assessment SA-REVIEW, which in most practical cases will be developed in conjunction with or may even form a part of SA-STORE.

2.7. DISPOSAL

The eventual target for radioactive waste is its safe disposal. When an adequate disposal facility exists, waste will be transported to this facility directly after processing or following a storage period.

Further processing may be necessary in order to meet transport and disposal criteria, although this necessity should be avoided to the extent possible (see Section 2.5). If, however, additional processing is necessary, the type of activities and the safety assessment SA-PROCESS required are identical to those described in Section 2.5.

For the transport of waste a safety assessment SA-TRANSPORT will be necessary. This may be very simple for unproblematic waste, and will involve only demonstration that criteria on activity contents, dose rates etc. stipulated in the Transport Regulations [6] are complied with. For more problematic waste (in particular for high level waste) more detailed assessments of the transport risks may be necessary.

The eventual disposal of the waste will require a thorough safety assessment covering the operational phase of the repository as well as its long term safety. A methodology for this purpose has been developed in the ISAM coordinated research project [1]. Consideration of this stage of radioactive waste management, therefore, was outside the scope of the SADRWMS project.

3. KEY COMPONENTS OF THE SAFETY ASSESSMENT

The recommended approach to safety assessment includes the following key components:

i. Specification of the context for the assessment (the purpose of the assessment, the philosophy underlying the assessment, the regulatory framework, the assessment end points, and the time frame for the assessment);

ii. Description of the predisposal waste management facility or activity and the waste;

iii. Development and justification of scenarios;

iv. Formulation of models and identification of data needs;

v. Performance of calculations and evaluation of results;

vi. Analysis of safety measures and engineering aspects, and comparison with safety criteria;

vii. Independent verification of the results;

viii. Review and modification of the assessment, if necessary (i.e. iteration).

These components are outlined in Figure 7. Solid black lines indicate the typical sequence of activities but this sequence is not mandatory. The components are interdependent and may be performed in an iterative manner.

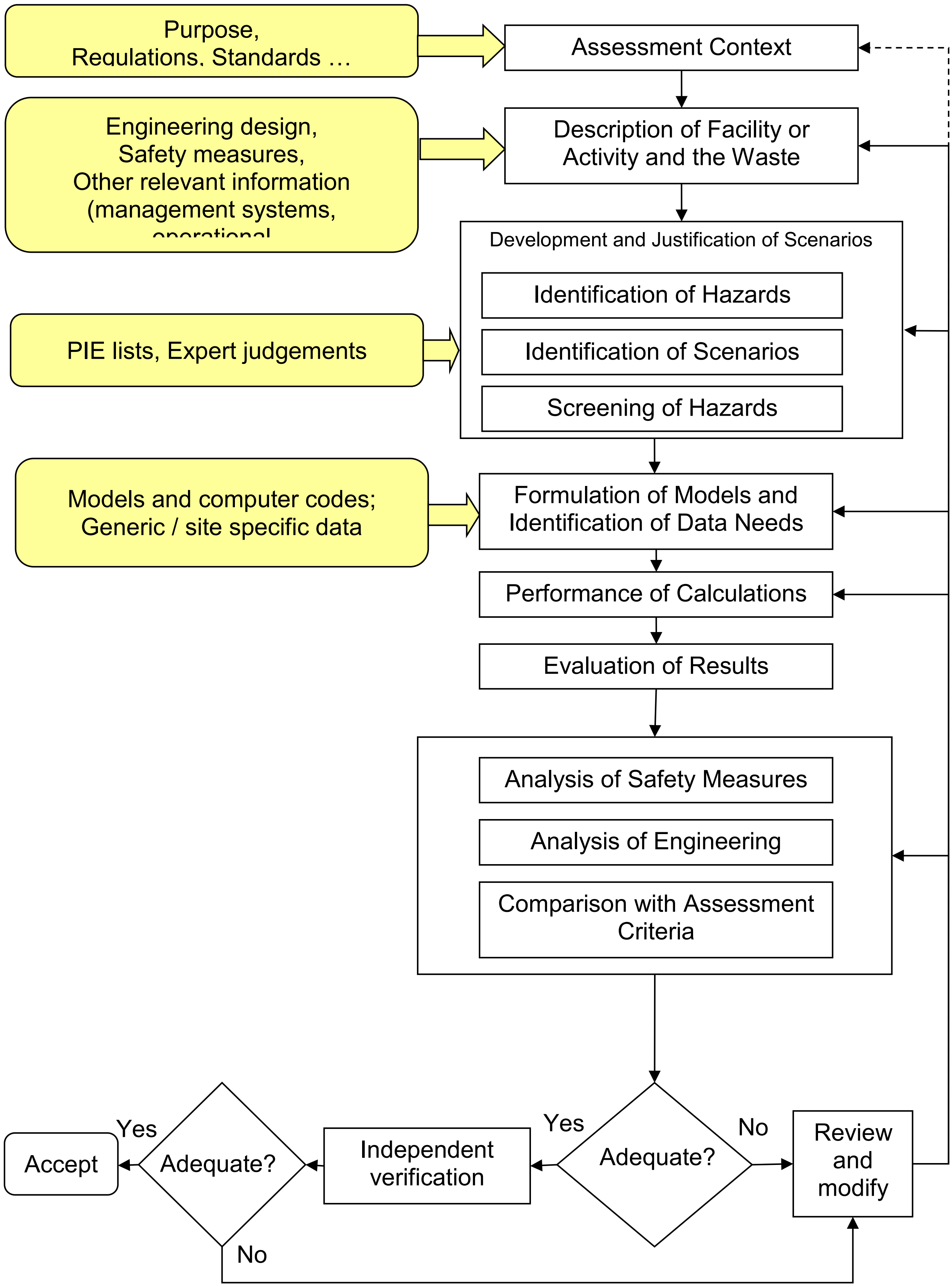

FIG. 7. *Safety assessment process for predisposal waste management.*

4. ASSESSMENT CONTEXT OF THE PREDISPOSAL WASTE MANAGEMENT PROCESSES

Based on the description of relevant waste management steps, the safety assessments identified as necessary for the individual process steps can be characterized with regard to the following aspects:

- Purpose of the safety assessment, i.e. questions that need to be addressed and answered;
- General aspects of the assessment context for each assessment.

The different safety assessments are identified using the acronyms already defined. For each safety assessment the following tables indicate key elements of their:

- Purpose;
- End points;
- Philosophy;
- Time frames.

In addition, remarks concerning the contents of each safety assessment and the relationship to other safety assessments are provided.

Some general aspects of the safety assessments are not mentioned in the tables, such as the regulatory framework as part of assessment context and the use of the safety assessment to contribute to enhancing public confidence. These will strongly depend on the specific conditions under which the assessments are undertaken.

TABLE 1. ASSESSMENT CONTEXT FOR SA-INTERVENE

Assessment of Safety of Waste Stored in an Existing Facility	**SA-INTERVENE**
Purpose of assessment	To determine whether the existing situation is acceptable from a safety and security point of view or whether corrective action to upgrade safety and/or security measures is necessary. Note: The identification of the required corrective action is not part of this assessment (see SA-OPTIONS).
Assessment end points	Assessment of impacts from the facility in current conditions and from possible changes (e.g. degradation of barriers, external or internal events); possible end points include: a. Radionuclide releases from the storage facility; b. Radionuclide concentrations in the surrounding environment; c. Doses and risks to workers for activities such as maintenance and surveillance; d. Doses to the public (potential exposure or actual exposure of a member of a certain group); e. Doses to non-human biota; f. Level of security at the facility.
Assessment philosophy	1. Use of cautious assumptions, but in view of the intervention situation these need to be as realistic as possible; i.e. the existing situation has to be addressed realistically and cautious assumptions only used to the extent that impacts from events and processes potentially affecting the assessment end points need to be assessed. 2. Use of actual data to the extent possible and warranted; i.e. the use of generic data is restricted to cases in which site specific data are not available (e.g. data concerning the impact of potential events and processes or data such as the contents of waste packages that cannot be measured at this stage) or to cases in which site specific sampling and measurements are not warranted by the importance of the data for the assessment results.
Assessment time frames	Anticipated time frame for establishing a disposal facility and for commencing the retrieval of the waste. Note: Frequently existing uncertainties in this respect are accounted for by using a contingency allowance.
Remarks	The aim of this assessment is only to determine whether there is a need for intervention. If this need is shown, SA-OPTIONS will be used to compare available intervention options and to lead to the identification of the upgrading option to be implemented.

TABLE 2. ASSESSMENT CONTEXT FOR SA-OPTIONS

Assessment of Options to Upgrade Safety		**SA-OPTIONS**
Purpose of assessment	1. To identify options to improve the existing situation of waste stored in the facility and/or the condition of the facility itself by a. Improving the design of the facility; and/or b. Retrieving part or all of the waste from the facility. 2. To compare identified options and to determine the optimal option with regard to all attributes relevant for the specific situation (doses, risks, costs etc.).	
Assessment end points	1. Assessment of the retrieval of waste and/or upgrading of the facility (to the extent that these are within the scope of options considered); possible end points include: a. Radionuclide releases caused by the retrieval and upgrading operations; b. Radionuclide concentrations in the surrounding environment; c. Doses and risks to workers during waste retrieval and upgrading of the facility; d. Doses to the public (potential exposure of a group member); e. Doses to non-human biota. 2. Assessment of impacts from the upgraded facility (i.e. improved design and/or partially retrieved waste); possible end points include: a. Radionuclide releases from the storage facility; b. Radionuclide concentrations in the surrounding environment; c. Doses and risks to workers for activities such as maintenance and surveillance; d. Doses to the public (potential exposure of a group member); e. Doses to non-human biota; f. Level of security at the facility. 3. Assessment of processing, storage or disposal of the retrieved waste (to the extent that waste retrieval is within the scope of options considered). Note: The necessity and scope of this part of the assessment will be very case specific and will depend on whether capacities for waste processing, storage or disposal already exist. In any case, it is important to include the fate of the retrieved waste (in particular, doses, risks and costs incurred by their management) into the comparison of options for intervention.	

TABLE 2. (Continued)

Assessment of Options to Upgrade Safety	**SA-OPTIONS**
Assessment philosophy	1. Use of cautious assumptions, but in view of the intervention situation these need to be as realistic as possible (see Table 1): a. The comparison of options has to be based in general on realistic assumptions; b. The assessment of compliance with regulatory standards within each of the options considered will require sufficiently cautious assumptions. 2. Use of actual data to the extent possible and warranted; i.e. the use of generic data is restricted to cases in which site specific data are not available (see Table 1).
Assessment time frames	1. Assessment of the retrieval of waste and/or upgrading of the facility: duration of these activities. 2. Assessment of impacts from the upgraded facility: anticipated time frame for establishing a disposal facility and for starting the retrieval of waste (including contingency allowance, see Table 1). 3. Assessment of processing storage or disposal of retrieved waste: case specific (see above).
Remarks	1. This assessment will only be required if SA-INTERVENE results in the need for intervention. 2. The actual planning of the measures to upgrade the facility and/or to retrieve the waste is not part of this safety assessment (see SA-STORE, SA-RETRIEVE). Therefore, assessments of these activities are required only to the extent and depth of allowing for a comparison of options. Detailed planning will only be necessary for the option identified as optimal (i.e. the option that is going to be implemented).

TABLE 3. ASSESSMENT CONTEXT FOR SA-RETRIEVE

Assessment of Waste Retrieval	SA-RETRIEVE
Purpose of assessment	1. Assessment of the safety of retrieval operations, to allow for their detailed planning. 2. Establishment of: a. Limits (qualitative or quantitative restrictions to any part of the activity, which are applied to ensure compliance with safety principles and requirements); b. Controls (processes, procedures or other instruments that are put in place to ensure compliance with safety principles and requirements); and c. Conditions (prerequisites, requirements for functions, facilities or organizations that must exist to ensure safety) for the retrieval operations.
Assessment end points	Assessment of the retrieval operations, possible end points include: a. Radionuclide releases caused by the retrieval and upgrading operations; b. Radionuclide concentrations in the surrounding environment; c. Doses and risks to workers during waste retrieval and upgrading of the facility; d. Doses to the public (potential exposure of a group member); e. Doses to non-human biota.
Assessment philosophy	1. Use of cautious assumptions, but, in view of the intervention situation, these should be as realistic as possible (see Table 1). 2. Use of actual data to the extent possible and warranted; i.e. the use of generic data is restricted to cases in which site specific data are not available (see Table 1).
Assessment time frames	Duration of retrieval activities.
Remarks	The assessment of the fate of retrieved waste is not part of this safety assessment. This will be covered by other relevant safety assessments addressing the management steps for such waste, i.e. its clearance, discharge, processing, storage, transport and disposal.

TABLE 4. ASSESSMENT CONTEXT FOR SA-CLEAR

Derivation of Clearance and Discharge Levels and Procedures		**SA-CLEAR**
Purpose of assessment	For clearance: 1. To establish generic clearance levels for waste in general or for certain waste types, possibly also including certain restrictions on clearance (e.g. clearance levels for metal scrap subject to smelting); or 2. To determine whether unconditional or conditional clearance of certain types of waste is possible (i.e. whether this particular waste types complies with criteria for clearance). For discharges: 3. To establish general or facility specific discharge limits. For clearance and discharges: 4. To develop clearance and discharge procedures (in particular the type and extent of required measurements and monitoring).	
Assessment end points	Assessment of exposure from waste after clearance or discharge; possible end points include: a. Doses to the public (potential exposure of a group member). Note: For clearance, scenarios to be determined on the basis of the type of material and possible (for unconditional clearance) or restricted (for conditional clearance) options for disposal and recycling of the materials.	
Assessment philosophy	1. In general, cautious assumptions are being used. However, in particular when applying the low dose levels that meet the criteria for clearance, overly conservative assumptions should be avoided (see Ref. [4]). 2. For generic clearance levels and discharge limits as well as for addressing the unconditional clearance of certain waste types, necessarily generic data have to be used. The use of site specific data will only possible for certain cases of conditional clearance (i.e. when the recycling or disposal routes are known and will be ensured by regulatory provisions) and for facility specific discharge limits.	

TABLE 4. (Continued)

Derivation of Clearance and Discharge Levels and Procedures		**SA-CLEAR**
Assessment time frames	1. Dose assessments for clearance, in principle, have to be carried out for unlimited time frames. However, in practice limitations of time frames to be considered arise from the half-lives of the radionuclides involved and from the fact that within the scenarios usually considered the highest exposures arise immediately or shortly after clearance (exception: water pathways). 2. For discharges, exposures usually occur within short time frames, with the exception of exposures resulting from the accumulation of radionuclides in the environment (e.g. through adsorption by river sediments or ground deposition of aerosols). The latter case has to be treated in analogy to clearance	
Remarks	1. As shown in Figure 1, clearance and discharges can be a waste management option at all stages of the overall process. Therefore, the derivation of general clearance levels and discharge limits is often easier and more effective than addressing clearance at each individual process stage. 2. Since scenarios and dose assessments used for the derivation of clearance levels are usually very general, it appears to be adequate for most cases to use generic clearance levels derived on an international basis (e.g. Ref. [4]). Specific assessments can then be limited to particular waste types or to establishing levels for conditional clearance. 3. The development of clearance procedures in general will have to consider waste types and radionuclides of interest in order to determine adequate sampling and measurement procedures.	

TABLE 5. ASSESSMENT CONTEXT FOR SA-REQUIRE

Derive Requirements (for Storage, Transport and Disposal)		**SA-REQUIRE**
Purpose of assessment	Derivation of requirements for different waste management steps: a. Storage; b. Transport; and c. Disposal. in order to define waste processing requirements.	
Assessment end points	End points depend on specific activity considered (see remarks).	
Assessment philosophy	1. In general, cautious assumptions are used. 2. Data are either generic (for waste management activities not addressing a specific facility) or site specific (when deriving requirements for a particular facility).	
Assessment time frames	Time frames depend on the specific activity considered (see remarks).	
Remarks	1. The derivation of requirements will part of the safety assessments conducted for the different waste management activities (see SA-STORE, SA-TRANSPORT and ISAM [1]). End points and time frames considered will be determined as part of these assessments. 2. The derived requirements are either of generic nature (such as in the case of transport) or they are based on safety assessments for specific storage or disposal facilities and, therefore, are valid only for certain waste management routes. 3. The derived requirements have to be sufficiently specific to determine the type and extent of waste processing required.	

TABLE 6. ASSESSMENT CONTEXT FOR SA-PROCESS

<table>
<tr><th colspan="2">Assessment of Processing of Waste</th><th>SA-PROCESS</th></tr>
<tr><td>Purpose of assessment</td><td colspan="2">1. Siting guidelines and/or site selection for the waste processing facility;
2. Assessment of the safety of the waste processing operations, to allow for their detailed planning;
3. Establishment of:
a. Limits;
b. Controls; and
c. Conditions
for the waste processing operation.</td></tr>
<tr><td>Assessment end points</td><td colspan="2">Assessment of the waste processing operations; possible end points include:
a. Radionuclide releases caused by the waste processing operations;
b. Radionuclide concentrations in the surrounding environment;
c. Doses and risks to workers during waste processing;
d. Doses to the public (potential exposure of a group member);
e. Doses to non-human biota.</td></tr>
<tr><td>Assessment philosophy</td><td colspan="2">1. In general, cautious assumptions are used.
2. Use of actual data to the extent possible and warranted; i.e. the use of generic data is restricted to cases in which site specific data are not available (e.g. data concerning the impact of potential events and processes) or to cases in which the collection of data concerning the waste to be processed are not warranted by the importance of the data for the assessment results.</td></tr>
<tr><td>Assessment time frames</td><td colspan="2">Duration of the waste processing activities.</td></tr>
<tr><td>Remarks</td><td colspan="2">The necessary type and extent of waste processing depend on the requirements derived for subsequent waste management steps (see Table 5).</td></tr>
</table>

TABLE 7. ASSESSMENT CONTEXT FOR SA-STORE

Assessment of Storage of Waste		SA-STORE
Purpose of assessment	1. Siting guidelines and/or site selection for the storage facility; 2. Assessment of the safety of the waste storage, allowing for detailed planning; 3. Establishment of: a. Limits; b. Controls; and c. Conditions for the waste storage.	
Assessment end points	Assessment of the storage facility, possible end points include: a. Radionuclide releases caused by the storage operation and the stored waste; b. Radionuclide concentrations in the surrounding environment; c. Doses and risks to workers during activities involved in storage of the waste and for activities such as maintenance and surveillance; d. Doses to the public (potential exposure of a group member) during the storage operation and during the storage period; e. Doses to non-human biota; f. Level of security at the facility.	
Assessment philosophy	1. In general, cautious assumptions are used. 2. Use of actual data to the extent possible and warranted; i.e. the use of generic data is restricted to cases in which site specific data are not available (e.g. data concerning the impact of potential events and processes) or to cases in which the collection of data concerning the waste to be stored are not warranted by the importance of the data for the assessment results.	
Assessment time frames	Anticipated time frame for establishing a disposal facility (including contingency allowance, see Table 1).	
Remarks	Controls and conditions for the safety of waste storage will require regular review. These are addressed in SA-REVIEW (Table 8).	

TABLE 8. ASSESSMENT CONTEXT FOR SA-REVIEW

Assessment of Regular Safety Reviews of a Storage Facility	**SA-REVIEW**
Purpose of assessment	To determine the frequency and scope of required regular reviews of the safety of a waste storage facility.
Assessment end points	Assessment end points are identical to those addressed in SA-STORE (Table 7) concerning the waste storage period.
Assessment philosophy	Identical to SA-STORE.
Assessment time frames	Identical to SA-STORE.
Remarks	1. This safety assessment addresses the same events and processes as already considered in SA-STORE Therefore, it will usually will be conducted in combination with or even as part of SA-STORE. 2. During the regular reviews, assumptions made in the underlying safety assessments (SA-STORE, SA-REVIEW) may turn out as inadequate (e.g. neglecting of certain events or processes, overly conservative assumptions). This may require updates of these safety assessments and additional measures to maintain safety.

5. IMPLEMENTATION OF THE SAFETY ASSESSMENT METHODOLOGY IN THE SAFRAN TOOL

The above described predisposal waste management safety assessment methodology is practically implemented in the SAFRAN tool. The SAFRAN tool incorporates a graded approach to the assessments, where a screening of hazards is carried out, followed by dose assessments for specific end points if required (Figure 8). The identification and screening of hazards, as well as the dose assessments, are carried out with the help of scenarios. Scenarios represent sequences of events and processes by which hazards can be realized, resulting in radiological impacts. Differentiation is made between assessments for normal operation scenarios (Section 6) and accidental scenarios (Section 7).

The assessment methodology is implemented in three main steps:

1. **Scenario definition.** Scenarios are postulated or assumed sets of conditions and/or events [7] that can lead to human exposure or environmental contamination during normal operation or accidental situations. The scenario can be seen as a representation of how identified hazards could be realized. Defining a scenario involves specifying the probability of occurrence (for accidental scenarios), the potential radiological impacts (consequences) and how relevant safety elements will perform. As shown in Figure 8, several scenarios can have the same potential impact. For assessments of accidents it is also possible to specify postulated initiating events (PIEs). Several initiating events can lead to the same sequence of events and the same impact.

2. **Hazard screening.** Hazard in this document is defined as the potential for a waste, activity or facility for causing radiological harm to workers, members of the public and the environment. The screening of hazards can be performed by performing quantification of the impacts corresponding to the scenarios and comparing the results with screening values. Impacts are quantified as releases (Bq) or doses (Sv) for accidental scenarios or release rates (Bq/y) or dose rates (Sv/y) for normal operation scenarios, obtained for standardized conservative conditions. For each impact a Hazard Quotient (HQ), defined as the ratio between the calculated value and a screening value, is calculated. The screening values are selected conservatively and therefore the calculated HQs can be used for screening purposes – if the HQ are below 1, then it can be considered that the hazards are not significant and there is no need to perform more realistic dose assessments. More generally, the HQ can be used for ranking the hazards, which could be useful for bounding of impacts and scenarios. The term radiological hazard is used here as a measure of the potential to cause radiological harm. This is different from radiological risk, which is defined as the radiological harm resulting from a specified exposure situation (see below).

3. **Dose assessments.** This step consists of calculating doses for specified assessment end points, such as specific workers and members of the public. For accidental scenarios effective doses (Sv) integrated over the whole exposure time are calculated. For normal operation scenarios annual effective doses (Sv/y) are calculated. Doses from internal exposure are in all cases defined as committed effective doses integrated over a period of 50 years for workers and over a period of 70 years for members of the public. The exposure assessment differs from the hazard characterization in that the assessments are not done for standardized conservative conditions, but for the realistic conditions that are expected will prevail for a given scenario.

Detailed descriptions of the methodologies used in assessments for normal operation and accidents are provided in Sections 6 and 7, respectively. These sections also include overviews of methods for analysis of results.

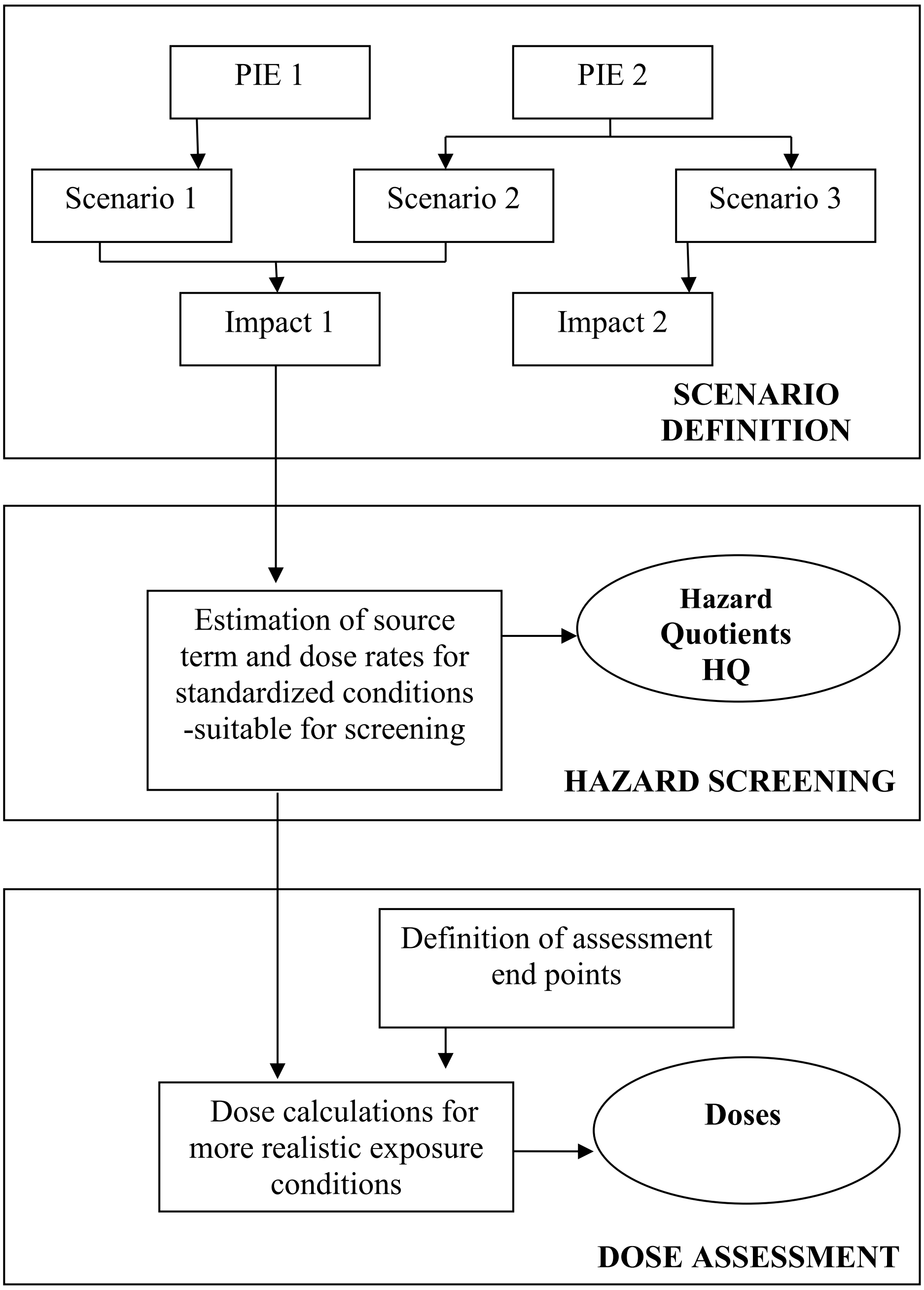

FIG. 8. Main steps of the graded approach to the hazards screening and dose assessments.

6. HAZARDS SCREENING AND DOSE ASSESSMENTS FOR NORMAL OPERATION SCENARIOS

This section describes the methodologies for screening of hazards and dose assessments for normal operation scenarios implemented in the SAFRAN tool.

6.1. TYPES OF IMPACTS

Impacts of normal operation scenarios can be of different types. Several classification schemes of impacts can be valid. In the SAFRAN tool impacts are classified by the radiological impact types and by the location where the exposure can take place: indoors (inside the rooms where waste management activities take place) and outdoors (outside the facilities). This distinction is required because the methods for quantification of impacts of different categories are different. Basing on this classification, the SAFRAN tool selects appropriate procedure and the model for calculation of impacts.

Impacts of normal operation scenarios in the SAFRAN tool are classified into the following categories:

- Impacts indoors:
  - Direct external exposure and exposure via inhalation;
  - Other.
- Impacts outdoors:
  - Releases to air;
  - Liquid discharges;
  - Other.

If the impact is classified as other, the SAFRAN tool does not use any particular procedure for quantification of the impact. The appropriate procedure for quantification of the impact shall be defined by user.

6.2. SCREENING OF HAZARDS

The impacts can be assessed qualitatively or quantitatively. Qualitative methods usually are used when the magnitude of the impact is either very high or very low. For situations in between usually there is a need for quantification of the impacts. It is possible to establish a scale of hazards relating qualitative categories with quantitative values (see Section 6.2.4). Quantification of impacts can be performed using the methods described in Sections 6.2.1, 6.2.2 and 6.2.3, which vary depending on the type of impact (see Section 6.1).

6.2.1. External irradiation and inhalation

For quantification of impacts consisting of external irradiation and exposure by inhalation indoors (i.e. inside the facilities) the following procedure is used by SAFRAN tool.

Dose rates from exposures via inhalation are calculated using the radionuclide concentrations in air inside the room where the exposure can take place for a given scenario:

$$DoseRate_{inh}^{i} = Conc_{air}^{i} * InhRate * DCC_{inh}^{i} \quad (1)$$

where,

$DoseRate_{inh}^{i}$ is the inhalation dose rate for the *i-th* radionuclide, Sv/h.

$Conc_{air}^{i}$ is the concentration in air of the *i-th* radionuclide, Bq/m^3. For operating facilities the radionuclide concentrations in air in the different rooms can be obtained from monitoring programs.

InhRate is the average inhalation rate for the given working conditions, m^3/h. The SAFRAN tool database includes inhalation rates for the normal operation conditions that are based on recommendations of the ICRP Publication 89 [8]. The following average breathing rates for adult male depending on level of physical activity can be selected from the SAFRAN tool database:

- 1.69 m^3/h – for the heavy worker (which 8 h working day consists of 7/8 light exercise and 1/8 heavy exercise resulting in the breathed air rate of 13.5 m^3/day);
- 1.20 m^3/h – for sedentary worker (which 8 h working day consists of 1/3 sitting and 2/3 light exercise resulting to the breathed air rate of 9.6 m^3/day).

The more conservative value (1.69 m^3/h) is set as default selection.

DCC_{inh}^{i} is the inhalation dose coefficient of the *i-th* radionuclide [Sv/Bq]. As default, the SAFRAN tool database incorporates values of committed effective dose per unit intake via inhalation for workers from General Safety Requirements Part 3 on Radiation Protection and Safety of Radiation Sources: International Basic Safety Standards – Interim Edition [9]. Dose coefficient values are taken from Schedule III of [9] following the lung absorption type as for compounds identified as “all compounds” or “all unspecified compounds” and more conservative value for 1 µm or 5 µm AMAD. Otherwise (i.e. when the lung absorption type is specified differently) the most conservative value of dose coefficient is used.

The total dose rate from inhalation of all radionuclides ($DoseRate_{inh}$) is calculated as follows:

$$DoseRate_{inh} = \sum_{i} DoseRate_{inh}^{i} \quad (2)$$

The total dose rate ($DoseRate_{total}$) is the sum of the total dose rate from inhalation (Eq. ((2)) and the external dose rate ($DoseRate_{ext}$) in the area or room where the exposure takes place for a given scenario:

$$DoseRate_{total} = DoseRate_{inh} + DoseRate_{ext} \quad (3)$$

For operating facilities the external dose rate in the different rooms can be obtained from monitoring programs.

The hazard quotient (*HQ*) for impact from external exposure and inhalation indoors for normal operation scenarios is calculated by dividing the total dose rate ($DoseRate_{total}$) by the screening dose rate (*ScreeningDoseRate*):

$$HQ = \frac{DoseRate_{total}}{ScreeningDoseRate} \quad (4)$$

The screening dose rate represents exposure conditions, under which the annual worker exposure is considered to be low:

$$ScreeningDoseRate = \frac{ScreeningDoseWorker}{ExposureTime} \quad (5)$$

As default, the screening dose rate for the SAFRAN tool database is calculated assuming that the screening dose to worker (*ScreeningDoseWorker*) is 0.1 mSv/y and the annual worker exposure duration (*ExposureTime*) is 2000 h/y (i.e. working time is 8 h per day, 5 days per week and 50 weeks per year):

$$ScreeningDoseRate = \frac{10^{-4}}{2000} = 5.0 \times 10^{-8}\ Sv/h \quad (5)$$

Default screening dose rate values in the SAFRAN tool database can be changed by user according to the assessment context.

6.2.2. Releases to air

The particular radionuclide related hazard quotient (HQ^i) for impacts of the type releases to air outdoors (i.e. outside the facility) for normal operation are obtained by dividing the release rate of each radionuclide ($ReleaseRate^i$) by a screening release rate ($ScreeningReleaseRate^i$):

$$HQ^i = \frac{ReleaseRate^i}{ScreeningReleaseRate^i} \quad (6)$$

Default values of the screening release rate for routine releases are available in the SAFRAN tool database.

The database default values of the screening release rates to the atmosphere for normal operation conditions are calculated by dividing the screening dose for members of the public by the appropriate dose calculation factor for releases to the atmosphere under normal operation conditions:

$$ScreeningReleaseRate^i = \frac{ScreeningDosePublic}{DoseCalculationFactor^i_{release}} \quad (7)$$

where,

ScreeningDosePublic is the screening dose for members of the public (Sv/y). For quantification of low impact it is assumed to be 0.01 mSv/y.

$DoseCalculationFactor^i_{release}$ is the dose calculation factor for release (Sv/Bq). For calculation of screening release rates the SAFRAN tool uses the IAEA Safety Report Series No. 19 [10], Annex I, Table I-III provided dose calculation factors, derived basing on the Generic Environmental Model.

The Generic Environmental Model [10] for releases into atmosphere is based on assumptions that are expecting being sufficiently conservative for most of usually encountered practices. The annual average doses to members of a hypothetical critical group are evaluated for a continuous release over a 30 year period. An atmospheric release is assumed to occur from a pipe with diameter 0.5 m, located on the side of a building with a cross-sectional area of 500 m^2. A hypothetical critical group member is assumed to live at a distance of 20 m from the release source. In terms of atmospheric dispersion, this receptor point is within the cavity zone of the building. It is assumed that terrestrial foods are produced at a greater distance from the source; crops from a distance of 100 m, and milk and meat from 800 m. The exposure pathways considered in these calculations are as follows: inhalation and external exposure from immersion in the cloud (at 20 m), external exposure from material deposited on the ground (at 20 m) and ingestion of crops (from 100 m) and milk and meat (from 800 m). Effective doses were calculated for both adult (age > 17 y) and child (age 1-2 y) members. The dose to the age group receiving the higher dose is included in the summary of results.

Further details on the Generic Environmental Model assumptions on dispersion conditions, habit parameters, effective dose coefficients for external and internal doses, can be found in the IAEA Safety Report Series No. 19 [10].

Default values of the screening release rate can be changed by user according to the assessment context.

The total hazard quotient for the scenario is obtained by summing the hazards from particular radionuclides:

$$HQ = \sum_i HQ^i \tag{8}$$

6.2.3. Liquid discharges

The particular radionuclide related hazard quotient (HQ^i) for impacts of the liquid discharges outdoors (i.e. from the facility) for normal operation are obtained by dividing the discharge rate of each radionuclide ($DischargeRate^i$) by the screening discharge rate ($ScreeningDischargeRate^i$).

$$HQ^i = \frac{DischargeRate^i}{ScreeningDischargeRate^i} \tag{9}$$

Default values of the screening discharge rate for routine releases are available in the SAFRAN tool database.

The database default values of the screening discharge rate for liquid discharges for normal operation conditions are calculated by dividing the screening dose for members of the public by the appropriate dose calculation factor for liquid discharges under normal operation conditions:

$$ScreeningDischargeRate^i = \frac{ScreeningDosePublic}{DoseCalculationFactor^i_{discharge}} \tag{10}$$

where,

ScreeningDosePublic is the screening dose for members of the public (Sv/y). For quantification of low impact it is assumed to be 0.01 mSv/y.

$DoseCalculationFactor^{i}_{discharge}$ is the dose calculation factor for discharge (Sv/Bq). For calculation of screening discharge rates the SAFRAN tool uses the IAEA Safety Report Series No. 19 [10], Annex I, Table I-IV provided dose calculation factors for discharges into sewer and Table I-V provided dose calculation factors for discharges into surface water. The higher value of dose conversion factor is selected for the SAFRAN tool database to envelope specific of the both considered environments.

The dose calculation factors for discharges are derived basing on the Generic Environmental Models.

The Generic Environmental Model [10] for discharges into sewer assumes that radionuclides are being discharged into the sewerage system and retained in the sewage sludge. The maximum annual average activity concentration in sludge was calculated assuming annual sewage production of 400 t/y (dry weight) for 2.5×10^{-6} Bq/kg per Bq/y discharged.

Two exposure pathways were considered, both of which arise within the sewage plant itself — external irradiation from radionuclides in the sludge and inhalation of radionuclides resuspended into the air from the sludge. The hypothetical maximum annual external dose from radionuclides at the surface of a container full of sewage sludge was calculated. The maximum annual surface activity concentration was estimated from the total sludge concentration by assuming a density of 1×10^{3} kg/m^{3} and a sludge container depth of 1 m. Workers within a sewage plant were assumed to be exposed for a working year of 2000 h/y.

The Generic Environmental Model [10] for discharges into surface water assumes that radionuclides are being discharged into a small river with a flow of 0.1 m^{3}/s. The hypothetical critical group is assumed to live at a distance of 500 m downstream from the discharge point and on the same side of the river. The assumed river dimensions are compatible with a river flow velocity of approximately 0.5 m/s and a partial mixing coefficient of 1.6.

The following exposure pathways were considered: drinking water, ingestion of freshwater fish and external irradiation from radionuclides in shore/beach sediment. The dose from drinking water was calculated assuming that the water was filtered before consumption. The total dose from the three pathways was estimated for adult (age > 17 y) and child (age 1-2 y) members. The dose to the age group receiving the higher dose is included in the summary of results.

Further details on the Generic Environmental Models assumptions on discharge conditions, habit parameters, effective dose coefficients for external and internal doses, can be found in the IAEA Safety Report Series No. 19 [10].

Default values of the screening discharge rate can be changed by user according to the assessment context.

The total hazard quotient for the scenario is obtained by summing hazards from particular radionuclides:

$$HQ = \sum_i HQ^i \tag{11}$$

6.2.4. Comparison of hazards

For normal operation scenarios probabilities are irrelevant and the hazard is characterized by the impact, measured by the HQ. It is possible to define the impacts qualitatively or quantitatively. To compare quantitative and qualitative hazards it is possible to create a scale that gives the correspondence between these. For example, SAFRAN tool comes with a default scale of hazards for normal operation scenarios with five categories: very low, low, medium, high and very high. The assumed correspondence between these categories and the hazard quotients (HQ) are shown in Table 9.

TABLE 9. IMPACT SCALE FOR NORMAL OPERATION SCENARIOS INCLUDED IN THE SAFRAN TOOL (DEFAULT VALUES ARE SHOWN)

Impact Category	**HQ values**
Very high	> 100
High	10-100
Medium	1-10
Low	0.1-1
Very low	< 0.1

6.3. DOSE ASSESSMENTS

The exposure assessment consists of calculating doses for specified assessment end points, such as specific workers and members of the public. For normal operation scenarios effective annual doses (Sv/y) are calculated. The assessment is performed for exposure conditions that are expected will prevail for a given scenario.

6.3.1. Assessment end points

The first step of the exposure assessment is to define the assessment end points, i.e. to define the specific results of the assessments: for example annual doses to a worker or a member of the public. The assessment end points can be different for different scenarios or the same assessment end point can be applicable to several scenarios.

6.3.2. Assessment of exposures indoors

In assessments of the exposure for scenarios with impacts indoors via external exposure and inhalation, the annual doses are calculated by multiplying the total dose rate quantified in the screening of hazards ($DoseRate_{total}$, see Eq. (3)) by the actual worker exposure time ($time_{exposure}$).

$$Dose = DoseRate_{total} \times time_{exposure} \tag{12}$$

6.3.3. Assessment of exposures outdoors

The method for assessment of doses from exposure outdoors will depend on the type of impact of the scenario of interest. The different situations that may arise are considered below.

6.3.3.1. Releases to air

The doses outdoors (i.e. outside the facilities) from particular scenario routine releases to air are calculated as follows:

$$Dose_j = \sum_i ReleaseRate_j^i \times DCF_{air,nor}^i \tag{13}$$

where,

$ReleaseRate^i_j$ is the particular scenario j release rate outdoors for particular radionuclide i. The release rates are quantified in the screening of hazards, see Section 6.2.2 and Eq. (7).

$DCF^i_{air,nor}$ is the dose conversion factor (DCF) for releases to air for normal operation. DCF is defined as the effective annual dose to a member of the public resulting from an unit constant release rate of a radionuclide during the whole period when the releases take place. The SAFRAN tool database includes values of the DCF. As default, the database includes the IAEA Safety Report Series No. 19 [10] Annex I, Table I-III provided values, that are derived basing on the Generic Environmental Model, see explanations in Section 6.2.2. If the specific of the site and the environment has to be addressed additionally, new DCF values can be calculated with the SAFRAN tool module SAFCALC model for routine atmospheric releases, which implements the screening models described in the IAEA Safety Report Series No. 19 [10]. The user may also input and use own DCF values, i.e. calculated with other relevant models and computer codes.

The SAFRAN tool calculates doses from releases to air of each particular radionuclide, the dose of each scenario (Eq. ((13)) and the total dose of all scenarios:

$$Dose = \sum_j Dose_j \tag{14}$$

6.3.3.2. Liquid discharges

The doses outdoors (i.e. outside the facilities) from particular scenario routine liquid discharges are calculated as follows:

$$Dose_j = \sum_i DischargeRate_j^i \times DCF_{liquid,nor}^i \tag{15}$$

where,

$DischargeRate^i_j$ is the particular scenario j discharge rate outdoors of particular radionuclide i. The discharge rates are quantified in the screening of hazards, see Section 6.2.3 and Eq. (10);

$DCF^{i}_{liquid,nor}$ is the dose conversion factors (DCF) for liquid discharges during normal operation. DCF is defined as the effective annual dose to a member of the public resulting from an unit constant release rate of a radionuclide during the whole period when the releases take place. The SAFRAN tool database includes values of the dose conversion factors. As default, database incorporates the IAEA Safety Report Series No. 19 [10] Annex I, Table I-IV (for discharges into sewer) and Table I-V (for discharges into surface water) provided values, that are derived basing on the Generic Environmental Models, see explanations in Section 6.2.3. The higher value of dose conversion factor is selected for the SAFRAN tool database to envelope specific of both considered environments. If the specific of the site and the environment has to be addressed additionally, new DCF values can be calculated with the SAFRAN tool module SAFCALC model for routine liquid discharges, which implements the screening models described in the IAEA Safety Report Series No. 19 [10]. The user may also input and use own DCF values, i.e. calculated with other relevant models and computer codes.

The SAFRAN tool calculates doses from discharges of each particular radionuclide, the dose of each scenario (Eq. (16)) and the total dose of all scenarios:

$$Dose = \sum_{j} Dose_j \tag{16}$$

7. HAZARDS SCREENING AND DOSE ASSESSMENTS FOR ACCIDENT SCENARIOS

This section describes the methodologies for screening of hazards and assessment of doses for accidental scenarios implemented in the SAFRAN tool. The first step is to identify the scenarios and impacts that will be quantified. Several scenarios can be linked to the same impact. For example, the radiological impact due to fire in a storage room might be the same for two scenarios with different initiating events leading to the same type of fire. Although these scenarios would have the same impact, they may have different probabilities.

7.1. TYPES OF IMPACTS

Impacts of accidental scenarios can be of different types. In the SAFRAN tool impacts are classified by the radiological impact types and by the location where the exposure can take place: indoors (inside the rooms where waste management activities take place) and outdoors. A classification of the impacts in the SAFRAN tool is needed in order to select an appropriate quantification method.

The impacts of accidental scenarios in the SAFRAN tool are classified into the following categories:

- Impacts indoors:
  - Increased direct external exposure. This could be the case in loose of shielding scenarios;
  - Release of radionuclides to air. This could be the case for scenarios when waste components are damaged resulting in releases to air inside and outside the facility;
  - Other.
- Impacts outdoors:
  - Release of radionuclides to air. This could be the case for scenarios when waste components are damaged resulting in releases to air outside the facility;
  - Liquid discharges of radionuclides. For scenarios involving accidental releases of liquid waste from the facility;
  - Other.

For accidents involving releases to air it is also necessary to define the type of effect which causes the releases, for example a thermal effect in case of fire accidents. This classification is needed the SAFRAN tool to be able choose values of the fraction of the affected radionuclide inventory that is released during the accident.

If the impact is classified as other, the SAFRAN tool does not use any particular procedure for quantification of the impact. The appropriate procedure for quantification of the impact shall be defined by user.

7.2. SCREENING OF HAZARDS

Quantification of impacts for screening of hazards is performed using the methods described in Sections 7.2.1, 7.2.2 and 7.2.3, which vary depending on the type of impact (see Section 7.1).

7.2.1. Potentially affected inventory (PAI)

The radionuclide inventory that can be potentially affected by the accident is denoted here as the potentially affected inventory (PAI). The PAI will depend on the radionuclide inventory present in the affected area at the moment of the accident and on the type of accident. The number of waste components that can be potentially affected by an accidental scenario will vary from case to case. Table 10 and Table 11 provide examples of how a PAI can be estimated in the SAFRAN tool for storage and processing facilities, respectively.

TABLE 10. CALCULATION OF POTENTIALLY AFFECTED INVENTORIES FOR STORAGE AREAS

Area	WS	WC	WF Type	N	RN	PAI Bq
Storage 1	WS1	WC1	Type 1	N_1	RN1	For one WC
					RN2	For one WC
	WS2	WC2	Type 2	N_2	RN 2	For one WC
					RN3	For one WC

Definition of columns in Table 10:

- Area is abbreviated name of the storage area;
- WS is abbreviated name of the waste stream (WS);
- WC is abbreviated name of the waste component (WC);
- WF type is type of the waste form of the WC;
- N is number of waste components for each WC (N);
- RN is abbreviated name of the radionuclide (RN);
- PAI is potentially affected inventory - radionuclide inventory in one WC.

Table 10 provides a summary of the information for all waste components (WC) in each storage area. The PAI refers to the inventory in one waste component. In cases when the waste streams are defined in a per year basis (waste volumes and activities per year) the number of waste components (N) can be calculated by multiplying the number of waste components per year by the duration in years of the waste stream.

TABLE 11. CALCULATION OF POTENTIALLY AFFECTED INVENTORIES FOR PROCESSING AREAS

Area	**WS**	**WF Type**	**WC**	**RN**	**PAI Bq**
Processing area 1	WS1	Type 1	WC1	RN1	conc*volume
				RN2	conc*volume
			WC2	RN1	conc*volume
				RN2	conc*volume
			WC3	RN1	conc*volume
				RN2	conc*volume
	WS2	Type 2	WC4	RN3	conc*volume
				RN4	conc*volume
			WC5	RN3	conc*volume
				RN4	conc*volume
			WC6	RN3	conc*volume
				RN4	conc*volume

Definition of columns in Table 11:

- Area is abbreviated name of the processing area;
- WS is abbreviated name of the waste stream (WS);
- WF type is type of the waste form involved in the waste management activity;
- WC is abbreviated name of the waste component (WC);
- RN is abbreviated name of the radionuclides (RN);
- PAI is PAI for the waste management activity.

Table 11 provides information on PAI associated with the waste management (WM) activity that takes place in a processing area. The PAI in this case is defined as the inventory involved in the process that can be affected during an accident. For each waste stream several values of PAI are calculated by multiplying the concentration in the waste components involved in the WM activity by the capacity (volume of waste involved each time the activity is carried out) of the WM activity. The capacity (m^3) is a property of the WM activity.

7.2.2. Direct external exposure

For quantification of impacts of the type direct external exposure the following procedure is used.

For each radionuclide i of a particular waste stream k the dose rate (Sv/h) resulting from exposure at a distance of one meter is calculated assuming point source geometry. This is done by multiplying the PAI value for one waste component ($PAI^i{}_k$) by the specific gamma ray dose constant ($SGRDC^i$) and the number of affected waste components (N_k) of a particular waste stream:

$$DoseRate_k^i = N_k \times PAI_k^i \times SGRDC^i \qquad (18)$$

The SGRDC (Sv/h per Bq) is a value for correlating the dose equivalent rate (per unit of activity) for a radionuclide at a specified distance of 1 meter. Values of the SGRDC are included in the SAFRAN tool database.

The total dose rate from affected waste components of a particular waste stream is obtained by summing the doses from all radionuclides:

$$DoseRate_k = \sum_i DoseRate_k^i \tag{19}$$

The total dose rate from all affected waste components of all waste streams is obtained by summing the dose rates from the affected waste streams:

$$DoseRate = \sum_k DoseRate_k \tag{20}$$

The hazard quotient (*HQ*) is calculated by dividing the dose rate by the screening dose rate for accidents ($ScreeningDoseRate_{acc}$):

$$HQ = \frac{DoseRate}{ScreeningDoseRate_{acc}} \tag{21}$$

The screening dose rate for accidents (Sv/h) is available in the SAFRAN tool database. This value has been obtained by dividing the assumed screening dose for accidents (0.1 mSv) by an exposure time of 1 hour:

$$ScreeningDoseRate_{acc} = \frac{10^{-4}}{1} = 1.0 \times 10^{-4}\ Sv/h \tag{22}$$

7.2.3. Releases to air

Impacts of the type releases to air can have radiological consequences inside and outside the facility. Below the procedure for quantification of the impacts for each of these situations is described.

7.2.3.1. Impacts indoors

For quantification of impacts inside a particular room of a facility of the type direct releases to air the following procedure is used.

For each waste component it is necessary to define values of the airborne release fractions (ARF). The ARFs are the fraction of the PAI (radionuclide specific) that is released to air during the accident and depend on the type of effect of the accident (for example thermal effect in case of fires) and the waste form type of the affected waste component. The SAFRAN tool database contains values of the ARF for a number of elements, types of effects and types of waste forms. As default, the SAFRAN tool database includes bounding values for the airborne release fractions (ARF) and airborne release rates (ARR) provided by the US DOE Handbook DOE-HDBK-3010-94 [11].The user can add own values of ARF to the database, as well as new types or effects and of waste forms.

For each radionuclide i of affected waste stream k the releases indoors (i.e. inside the room) are calculated by multiplying the number of affected waste components by the PAI and the ARF:

$$ReleaseInside_k^i = N_k * PAI_k^i * ARF_k^i \tag{23}$$

For storage areas the number of affected waste components could be equal to the number of waste components of all waste streams present in the area, but it could be only a fraction of this or even a larger number corresponding to future states of the facility. For processing areas the default value of the number of affected waste components is 1, since the PAI refers to the inventory involved in the process.

The HQs for each radionuclide i of affected waste stream k are obtained by dividing the calculated releases inside (Eq. (23)) by the screening release inside ($ScreeningReleaseInside^i$).

$$HQ_k^i = \frac{ReleaseInside_k^i}{ScreeningReleaseInside^i} \tag{24}$$

Values of the screening release inside for each supported radionuclide are available in the SAFRAN tool database. By default, the SAFRAN tool screening release inside considers the release that would give a dose equal to the screening dose for accidents (assumed equal to 0.1 mSv) to a person that is exposed at a distance of one meter from the release point during one hour. Default values of the screening release inside have been derived using the SAFRAN tool module SAFCALC model for assessment of indoor doses from accidental releases to air (see Annex I) under following conservative assumptions:

- Accident occurs in a relatively small room of 50 m^3 volume;
- There is no extraction of contaminated air from the room. i.e. the room air exchange rate is 0;
- All accident released activity is respirable and worker does not wear any protective equipment.

Further details on calculation of screening release inside default values can be found in Annex I, Section 7.1. The database default values can be changed by user according to the assessment context.

The total HQ for the waste stream is obtained by summing the HQs of all radionuclides released:

$$HQ_k = \sum_i HQ_k^i \tag{25}$$

The overall HQ from all affected waste streams is then:

$$HQ = \sum_k HQ_k \tag{26}$$

The total release inside is calculated as follows:

$$ReleaseInside^{i} = \sum_{k} ReleaseInside_{k}^{i} \tag{27}$$

7.2.3.2. Impacts outdoors

The total release inside (calculated with Eq. (27)) is used for calculating the releases outside ($ReleaseOutside^{i}$):

$$ReleaseOutside^{i} = ReleaseInside^{i} \times (1 - FiltrationEfficiency) \tag{28}$$

The filtration efficiency (*FiltrationEfficiency*) is the fraction of the released radionuclide that is retained in the filtration system and can take values between zero (assumes there is no filtration) and one (all airborne radionuclides are captured by filter).

The HQs for each radionuclide are calculated by dividing the calculated values of the release outside (Eq. (28)) by the corresponding value of the screening release outside ($ScreeningReleaseOutside^{i}$):

$$HQ^{i} = \frac{ReleaseOutside^{i}}{ScreeningReleaseOutside^{i}} \tag{29}$$

Values of the screening release outside for each supported radionuclide are included in the SAFRAN tool database. By default, the SAFRAN tool screening release outside considers the release that would lead to an annual exposure equal to the screening dose for accidents (assumed equal to 0.1 mSv). Default values of the screening release outside have been derived using the SAFRAN tool module SAFCALC model for assessment of outdoor doses from accidental releases to air (see Annex II) under following conservative assumptions:

- Release occurs from the building at the effective height of 10 m;
- Critical group members live at a distance of 30 m from the source in the downwind direction. In terms of atmospheric dispersion, this exposure point is inside the wake zone of the building from which the radioactive emission occurs. The terrestrial foods are produced at the same location also.

Calculated screening dose includes:

- External exposure by radiation from the radioactive air plume (air submersion);
- External exposure by radiation from contaminated ground (ground radiation);
- Internal exposure by radionuclides which are inhaled with the air (inhalation);
- Internal exposure as result of consumption of radionuclide contaminated foodstuffs (ingestion).

Further details on calculation of screening release outside default values can be found in Annex II, Section 7.1. The database default values can be changed by user according to the assessment context.

The total HQ for release outside is obtained by summing the HQ of all released radionuclides:

$$HQ = \sum_i HQ^i \tag{30}$$

The total HQ is used in the SAFRAN tool analysis module for comparison of hazards arising from different scenarios.

7.2.4. Comparison of hazards

Each hazard is characterized by two quantities - the impact (measured by the HQ) and the probability of the scenario. It is possible to define the probabilities and the impacts qualitatively or quantitatively.

Similarly to the scale of impacts (see Section 6.2.4), the SAFRAN tool includes a probability scale with five probability categories: very low, low, medium, high and very high. The assumed, as default, correspondence between these categories and annual probabilities (expressed in 1/y) and probabilities during the life time of the facility are shown in Table 12. This table also shows the expected number of occurrences of the accident during the life time of the facility. The values in Table 12 have been derived assuming that the occurrence of the accidents follows a Poisson distribution. This means that the annual probability of the accidents is constant during the life time of the facility and does not depend on previous occurrences. To derive the scale, at first the ranges of probability (*P*) that accident will occur during life time of the facility have been assumed. Then, considering life time of the facility 100 years, the annual probabilities (*p*) are calculated from the exponential distribution:

$$p = -\frac{\ln(1 - P)}{T_{LifeTime}} \tag{31}$$

TABLE 12. PROBABILITY SCALE FOR ACCIDENT SCENARIOS INCLUDED IN THE SAFRAN TOOL

Qualitative Category	**Probability during life time**	**Annual probability 1/y**
Very high	> 95 %	> 3,0E-02
High	75-95 %	1,4E-02 - 3,0E-02
Medium	5-75 %	5,0E-04 - 1,4E-02
Low	0,1-5 %	1,0E-05 - 5,0E-04
Very low	< 0,1 %	< 1,0E-05

Definition of columns in Table 12:

- Qualitative category is qualitative probability categories;
- Probability during life time is probabilities during lifetime of the facility or activity;
- Annual probability is annual probability of the accident.

7.3. DOSE ASSESSMENT

The exposure assessment consists of calculating doses for specified assessment end points, such as specific workers and members of the public. For accidental scenarios effective doses (Sv) integrated over the whole exposure time are calculated. Doses from internal exposure are in all cases defined as committed effective doses integrated over a period of 50 years for workers and over a period of 70 years for members of the public. The assessments are done for exposure conditions that are expected will prevail for a given scenario.

7.3.1. Assessment end points

The first step of the exposure assessment is to define the assessment end points, i.e. to define the specific results of the assessments: for example doses to a worker or a member of the public. Each end point can be linked to only one scenario, but one scenario can be linked to several end points. The assessment end points can be linked to regulatory criteria for analysis of compliance with the criteria.

7.3.2. Assessment of exposures indoors

Accidental exposures indoors can occur by different pathways. The different situations that may arise are considered below.

7.3.2.1. Direct external exposure

For assessment of doses from direct external exposure, the following calculation procedure is applied.

For each radionuclide i of a particular waste stream k doses are calculated considering the dose rate from a single component ($DoseRate^i_k$) and exposure duration ($time_k$):

$$Dose_k^i = DoseRate_k^i \times time_k \qquad (32)$$

Calculation of external exposure dose rates from a single waste component can be carried out using the SAFRAN tool module SAFCALC model for external exposure calculations (see Annex III). This model consists of several sub-models for different geometries: cube, cylinder, sphere, point source and disc. The models allow making calculations with or without consideration of shielding.

Once doses resulting from each radionuclide have been evaluated (with the SAFRAN tool models or external models), total dose from all radionuclides in the affected waste component can be calculated as follows:

$$Dose_k = \sum_i Dose_k^i \qquad (33)$$

The doses from several affected components of the same waste stream k can be calculated by multiplying the dose from one waste component (Eq. (33)) by the number of affected waste components (N_k):

$$Dose_{Nk} = N_k \times Dose_k \qquad (34)$$

The total dose for the scenario is obtained by summing the total doses of all considered waste streams:

$$Dose = \sum_{k} Dose_{Nk} \tag{35}$$

The total scenario dose is used in the SAFRAN tool analysis module for comparison of hazards.

7.3.2.2. Releases to air indoors

The dose from accidental release indoors (i.e. inside the facility) of radionuclide i is calculated taking into account exposures via inhalation and external irradiation from the cloud:

$$Dose^{i} = Dose^{i}_{inh} + Dose^{i}_{extCloud} = \tag{36}$$

$$ReleaseInside^{i} \times DispFact \times (InhRate \times ProtFact^{i} \times DCC^{i}_{inh} + DCC^{i}_{extCloud})$$

where,

ReleaseInside^{i} is the release to air inside the facility of the *i-th* radionuclide (Bq). The releases indoors are quantified in the screening of hazards, see Section 7.2.3.1 and Eq. (27).

DispFactor is the dispersion factor (h/m^3). The dispersion factor is defined as the concentrations in air, per unit release, at a given distance integrated over the exposure period. Values of dispersion factors depending on exposure distance, exposure time and room volume are available in the SAFRAN tool database. As default, the values in the database have been derived by use of the SAFRAN tool module SAFCALC model for assessment of indoor doses from accidental releases to air (for details refer to Annex I, Section 7.2). The user may also input and use its own dispersion factor values, i.e. calculated with other relevant models and computer codes.

InhRate is the inhalation rate of the exposed individual (m^3/h). The SAFRAN tool database included inhalation rates for the accidental conditions are based on recommendations of the ICRP Publication 89 [8]. The following momentary breathing rates for adult male depending on the level of physical activity can be selected:

- 3.0 m^3/h – for heavy exercise;
- 1.5 m^3/h – for light exercise;
- 0.54 m^3/h – sitting awake.

The most conservative value (3 m^3/h) is set as default selection.

DCC^{i}_{inh} is the dose coefficient for internal exposure to the *i-th* radionuclide via inhalation [Sv/Bq]. As default, the SAFRAN tool database incorporates GSR Part 3 [9] values of committed effective dose per unit intake via inhalation for workers. Dose coefficient values are taken from Schedule III of [9] following the lung absorption type for compounds

identified as “all compounds” or “all unspecified compounds” and more conservative value for 1 µm or 5 µm AMAD. Otherwise (i.e. when the lung absorption type is specified differently) the most conservative dose coefficient value is used.

$DCC^{i}_{extCloud}$ is the dose coefficient for external exposure from the cloud to the *i-th* radionuclide (Sv/h per Bq/m^3). As default, the SAFRAN tool database incorporates the U. S. Environmental Protection Agency Federal Guidance Report No. 12 [12] provided values of effective dose per time integrated activity concentration in the air. The publication [12] provides values for effective equivalent dose and equivalent skin dose separately. The effective dose factor is calculated as follows:

$$DCC^{i}_{extCloud} = h_E + 0.01 * h_T \tag{37}$$

where: h_E is the effective equivalent dose factor for air submersion (dose to skin is not included into summation) and h_T is the skin equivalent dose factor for air submersion.

The publication [12] provided dose factors do not include consideration of radiations emitted by radioactive decay products. Following the approach used in the IAEA Safety Report Series No. 19 [10], $DCC^{i}_{extCloud}$ values for some of radionuclides are additionally upgraded to account for impact arising from their progenies. These radionuclides and considered progenies are listed below:

TABLE 13. THE RADIONUCLIDES AND CONSIDERED PROGENIES

Radionuclide	Considered Progenies
Bi-212	Tl-208 (0.3593), Po-212 (0.6407)
Ce-144	Pr-144 (0.9857)
Cs-137	Ba-137m (0.9457)
Mo-99	Tc-99m (0.8860), Tc-99
Ru-106	Rh-106
Te-131m	Te-131 (0.2220), I-131 (0.7780)

*Where relevant, progeny branching ratios are shown in brackets.

$ProtectionFactor^{i}$ is the protection factor for human respiratory tract. The protection factor is the fractional reduction in incorporation of radionuclides via inhalation provided by protective equipment, such as respiration masks and may have a value between zero (there is no protection) and one (all radionuclides are captured by protective device).

The total dose for the scenario is calculated by summing doses from all radionuclides:

$$Dose = \sum_{i} Dose_i \tag{38}$$

The total scenario dose is used in the SAFRAN tool analysis module for comparison of hazards.

7.3.3. Assessment of exposures outdoors from releases to air

The dose from accidental release outdoors (i.e. outside the facility) of radionuclide *i* is calculated as follows:

$$Dose^i = ReleaseOutside^i \times DCF^i \tag{39}$$

where,

$ReleaseOutside^i$ is the release to air outside the facility of *i-th* radionuclide (Bq). The releases outdoors are quantified in the screening of hazards (see Section 7.2.3.2 and Eq. (28)).

DCF^i is the dose conversion factor (Sv/Bq). DCF for accidental releases to air are defined as the effective dose to a member of the public resulting from a unit instantaneous radionuclide release. The SAFRAN tool database includes values of the DCFs. As default, DCF values are calculated with the SAFRAN tool module SAFCALC model for assessment of outdoor doses from accidental releases to air (see Annex II), assuming exposure conditions similar to those that are used for calculation of the dose conversion factors for routine radioactive releases (see Sections 6.3.3.1 and 6.2.2):

- Release occurs from the building at the effective height of 10 m;
- Critical group members live at a distance of 30 m from the source in the downwind direction. The terrestrial foods are produced at a greater distance from the source: crops from a distance of 100 m and milk and meat from 800 m.

Calculated DCF includes:

- External exposure by radiation from the radioactive air plume (air submersion);
- External exposure by radiation from contaminated ground (ground radiation);
- Internal exposure by radionuclides which are inhaled with the air (inhalation);
- Internal exposure as result of consumption of radionuclide contaminated foodstuffs (ingestion).

Further details on calculation of default values of DCF can be found in the Annex II, Section 7.2. The user may also input and use own DCF values, i.e. calculated with other relevant models and computer codes.

The total dose for the scenario is calculated by summing doses from all radionuclides:

$$Dose = \sum_i Dose_i \tag{40}$$

The total scenario dose is used in the SAFRAN tool analysis module for comparison of hazards.

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ANNEX I. DESCRIPTION OF THE SAFRAN MODELS FOR EVALUATION OF WORKER EXPOSURE RESULTING FROM ACCIDENTAL RELEASE OF AIRBORNE RADIOACTIVE MATERIALS AND USER'S GUIDE

I-1. INTRODUCTION

This document describes the method used in the SAFRAN tool [I-1] for the calculation of exposure arising from accidental release of airborne radioactive materials. Models can be used for evaluation of occupational exposure to allow comparison with the relevant dose limiting criteria.

Presently, four models are available to address different exposure conditions. The first three are 'no dilution' model, 'gradual mixing' model and 'complete mixing' model. The fourth combined 'gradual mixing / complete mixing' model is a combination of the two last models and allows the user to run subsequently the 'gradual mixing and 'complete mixing' models.

Each of the models are created using Ecolego [I-2].

Each of the models is described and their practical use is explained in the sections below.

I-2. 'NO DILUTION' MODEL

I-2.1. Model description

The simplest and most conservative screening technique for assessment of potential inhalation dose is to assume that the all accident released respirable fraction of activity is inhaled by individual(s). Thus, the total effective dose incurred via inhalation E_{inh} (Sv) is calculated as follows:

$$E_{inh} = \sum_j Q_j \; RF_j \; e(g)_{inhj} \tag{I-1}$$

Where:

Q_j is the released airborne activity of radionuclide j (Bq).

RF_j is the respirable fraction of released airborne activity of radionuclide j (-). The RF defines amount of released activity which can be inhaled into human respiratory system. It is commonly assumed to include particles of 10 µm Aerodynamic Equivalent Diameter and less. Value of RF can range from 1 (most conservative value which means that the whole released activity consist of respirable size particles) to 0 (which means that none of airborne radioactive particles can cause human exposure via inhalation).

$e(g)_{inhj}$ is the committed effective dose per unit intake by inhalation for radionuclide j (Sv Bq^{-1}). As default, the model incorporates values from GSR Part 3 [I-3]; details are provided in Section I-6. Default values can be changed.

I-2.2. Parameters and values

The user defined parameters which are used in models of this toolbox can be separated into two groups - the general parameters, which have the same meaning, notification and value(s) in all models and the model specific parameters, which depends on the selected model.

The general parameters are specified and explained in the Table I-1 below. The ‘no dilution’ model has no additional specific parameters. To compile output from the ‘no dilution’ model, the user shall change default model output definition parameter to value 1.

TABLE I-1. USER DEFINED PARAMETERS THAT ARE GENERAL FOR ALL MODELS

Parameter	Notification	Default value	Comments
Model definition	Model	2	The parameter defines model, which is used for dose calculation and output of results. To use the ‘no dilution’ model this parameter shall be set to 1. To use the ‘gradual mixing’ model this parameter shall be set to 2. To use the ‘complete mixing’ model this parameter shall be set to 3. To use the ‘gradual mixing / complete’ this parameter shall be set to 4.
Released airborne activity, Q_j	Q	1 (Bq)	The user shall specify radionuclides under consideration and the released activity. The model default value is set to Q_j = 1 Bq for each of radionuclide.
Respirable fraction	RF	1.0 (-)	The model default value is set to RF=1.0 for each of radionuclide assuming that all released activity consists of respirable size particles.
Worker exposure time	Tex	60 (s)	This is time duration from accident occurrence till worker evacuates the room.

I-2.3. Limitations to use of the model

The model can be used for the assessment of bounding individual and collective doses resulting from the inhalation of radionuclides.

The user should keep in mind that model considers only exposure via inhalation and therefore may not be sufficient for assessment of total exposure from certain radionuclides which impact is governed by other exposure types. Examples of such radionuclides may be noble gases which generally do not participate in any biological processes and the exposure pathway of most concern is external exposure in a cloud of gas.

I-3. 'GRADUAL MIXING' MODEL

I-3.1. Model description

The 'gradual mixing' model (or 'cloud expansion' model), considers the radioactive material release to be instantaneous and the material to disperse at a constant velocity in all directions of a hemisphere whose plane coincides with the floor of the room. This assumption matches the physically intuitive picture in which the source term is diluted as the surrounding clean air mixes with it through the turbulent convection flow that is always present. Intuitively, the volume of clean air that enters the source cloud is proportional to the area of the interface between source cloud and clean air, or the hemisphere of radius r.

Assuming constant speed of the cloud expansion, the radius of the cloud r (m) is proportional to the time since the release occurred:

$$r = \alpha\, t \tag{I-2}$$

Where:

r is the radius of the cloud (m);

α is the cloud expansion speed (m s^{-1});

t is the time since the release occur (s).

The volume of the cloud V (m^3) at any time is that of a hemisphere of radius r:

$$V = \frac{2}{3}\,\pi\, r^3 \tag{I-3}$$

The concentration of the radioactive material in the cloud is continuously decreasing as the cloud disperses, see Fig. I-1. The exposure model considers worker that is initially located at some specified distance r_1 from the source. The worker is assumed to leave the room after a time that is longer than it takes the cloud to disperse. The worker is thus immersed in the cloud from the moment t_1 when the cloud reaches him to the time t_2 when he leaves the room.

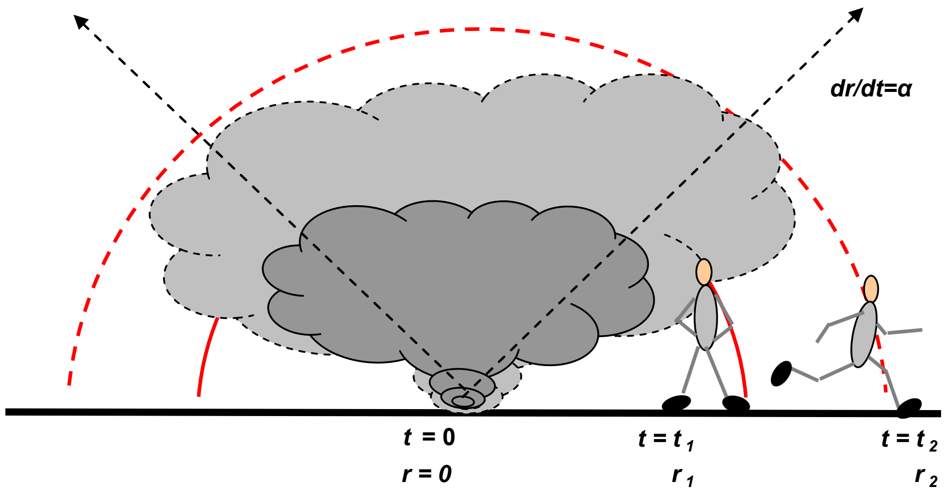

FIG. I-1. 'Gradual Mixing' dispersion model.

The total effective dose incurred via inhalation E_{inh} (Sv) is given by:

$$E_{inh} = \sum_j C_j \, RF_j \, B \, e(g)_{inhj} \tag{I-4}$$

Where:

C_j is the time integrated activity concentration in the air of the radionuclide j (Bq s m^{-3}).

RF_j is the respirable fraction of released airborne activity of radionuclide j, see explanations in Sections I-2.1 and I-2.2.

B is the worker breathing rate (m^3 s^{-1}). The model default value B = 8.33×10^{-4} (m^3 s^{-1}) is based on recommendations of the ICRP publication 89 [I-4] for the male conducting heavy work like fireman, construction worker, etc. Default value can be changed.

$e(g)_{inhj}$ is the committed effective dose per unit intake by inhalation for radionuclide j (Sv Bq $^{-1}$), see explanations in Sections I-2.1 and I-6.

The time integrated activity concentration in the air C_j for the exposure duration from the time t_1 till time t_2 is calculated as:

$$C_j = \int_{t_1}^{t_2} \frac{Q_j}{V} dt \tag{I-5}$$

Where:

Q_j is the released airborne activity of radionuclide j (Bq);

V is volume of the cloud (m^3).

Considering also Eq. (I-2) and Eq. (I-3), and after integration:

$$C_j = \frac{3\,Q_j}{4\,\pi\,\alpha^3}\left(\frac{1}{t_1^2} - \frac{1}{t_2^2}\right)$$

(I-6)

From the equation above it can be seen that the cloud diffusion factor χ (i.e. time integrated activity concentration in the air per unit of released activity (s m^{-3})) is given by:

$$\chi = \frac{C_j}{Q_j} = \frac{3}{4\,\pi\,\alpha^3}\left(\frac{1}{t_1^2} - \frac{1}{t_2^2}\right)$$

(I-7)

Total effective dose incurred via inhalation E_{inh} (Sv) then can be expressed, cf. Eq. (I-4):

$$E_{inh} = \sum_j \chi\, Q_j\, RF_j\, B\, e(g)_{inhj}$$

(I-8)

External exposure from submersion into the radioactive cloud (assuming that exposure begins with the worker submersion into the cloud and terminates when worker escapes the cloud) can be evaluated as:

$$E_{sub} = \sum_j \chi\, Q_j\, e_{subj}$$

(I-9)

Where:

e_{subj} is the effective dose per time integrated activity concentration in the air of the radionuclide j (Sv per Bq s m^{-3}) arising from submersion into radioactive cloud. In general, value of this dose conversion factor depends on number of parameters including shape and volume of the cloud. For preliminary evaluation the e_{airj} values derived for the semi-infinitive cloud [I-5] are used as default, details are provided in Section I-6. Default values can be changed.

The total accident effective dose is expressed as sum of the inhalation and the external exposure doses:

$$E = E_{inh} + E_{sub}$$

(I-10)

I-3.2. Parameters and values

The user defined general parameters for the model are the released airborne activity and the respirable fraction. These parameters are explained in the Table I-1.

The 'Gradual Mixing' model specific input parameters are discussed in table below.

TABLE I-2. 'GRADUAL MIXING' MODEL SPECIFIC PARAMETERS AND THEIR DEFAULT VALUES

Parameter	Notification	Default value	Comments
Distance from the centre of the release source to the worker location, r_1	r1	1.5 (m)	The model default value is r_1=1.5 m. This value can be imagined as a distance from the floor to the worker head height assuming the cloud is expanding form the floor level.
Cloud expansion speed, α	Alfa	0.15 (m s^{-1})	Average value, see discussion below the table.

The value for cloud expansion speed α requires understanding of the air currents likely to be present at the time of the release. The available research indicates that the mean air velocities in a room are directly proportional to the air change (AC) rate for the room [I-6]. That is, turbulent mixing in the room is dominated by ventilation flow, not natural convection. It also indicates that mean air speed depends on the size of the room, but less so on the furniture inside the room [I-7].

Experiments have shown that the cloud expansion speed α, is also directly proportional to the air change rate due to ventilation for a given room volume and increases with increasing room volume and that it will normally lie between 0.05 and 0.25 m s^{-1} for air change rates between 5 and 12 air changes per hour and room volumes from 70 m^3 up to 1350 m^3 [I-6], [I-7], [I-8], [I-9], [I-10], [I-11] and [I-12]. However, they also show that the cloud expansion speed is a factor of 2-3 lower than the average air velocity depending on room size and air change rate. From the available data, the factor of 3 is most applicable to high air-change rates (~10 ACh^{-1}) and large (~1000 m^3) room volumes. Note that, from Eq. (I-7), Eq. (I-8) and Eq. (I-9), the assumption of low cloud expansion speed is conservative in terms of dispersion coefficient and dose delivered.

I-3.3. Limitations to use of the model

It is considered that the model is only valid for situations wherein the time to evacuation is not very much greater than the time for the hemisphere to extend to the roof or surrounding walls of the enclosure, i.e. physical barriers which limit expansion of the cloud. For situations with longer exposure times use of the 'complete mixing' model (c.f. Section I-4) or combination of the 'gradual mixing / complete mixing' models (c.f. Section I-5) might be more appropriate.

It is also considered that the model is only valid for situations where the average directional air velocity is not excessively high.

Use of small values for parameter r_1 (distance from the center of the release source to the worker exposure location) should be cautious as it may not be supported by the physical nature of the model and the situation considered. For the small volume and concentrated sources of inhaled activity the use of 'no dilution' model (see Section I-2) might be more appropriate.

External exposure model assumes that worker external exposure begins with the worker submersion into the radioactive cloud. In reality, depending on released activity content, the worker expose can start with beginning of development of radioactive cloud. However the proposed exposure evaluation concept together with use of concentration to dose conversion factors derived for the semi-infinitive cloud is considered as still conservative approximation for the most of practically encountered situations where the distance from the accident point to the worker exposure location r_1 is not considerably long and the duration of worker exposure within the cloud t_{ex} prevails the time necessary the cloud to develop and submerge the worker t_1.

In the model, the release is assumed to be instantaneous. This is approximately true for spills or impacts, but not for releases that may last longer than it takes the worker to leave the room. In the latter case, it is appropriate to allow for this discrepancy by reducing the total source term Q by the fraction of the release time the worker is present in the room: t_2/t_s. Here t_s is the duration of the activity release.

I-4. 'COMPLETE MIXING' MODEL

I-4.1. Model description

The 'complete mixing' model, assumes that the released material Q_j spreads instantaneously and homogeneously throughout the room of volume V. In this case, the worker breathes and is submerged into the room averaged concentration of the released hazardous material for the time the worker stays in the room t_{ex}. Activity in the air of room is decreased due to radioactive decay and material removal by the room ventilation system. The process is described by differential equation as follows:

$$\frac{dq_{airj}}{dt} = -\lambda_j \, q_{airj} - \frac{G}{V} \, q_{airj} \tag{I-11}$$

Where:

q_{airj} is activity in the air of room of the radionuclide j (Bq);

λ_j is the radioactive decay constant of radionuclide j (s^{-1});

V is the room volume (m^3);

G is the room air change rate ($m^3\ s^{-1}$)

with initial condition:

$$q_{airj} = Q_j$$

(I-12)

The time integrated activity in the air Q_{airj} of the radionuclide j (Bq s) for the exposure duration t_{ex} is calculated as:

$$Q_{airj} = \int_0^{t_{ex}} q_{airj}\, dt$$

(I-13)

Neglecting radioactive decay and after integration:

$$Q_{airj} = Q_j\, V\, \frac{1 - \exp(-\frac{G}{V}\, t_{ex})}{G}$$

(I-14)

and the time integrated concentration in the air C_j of the radionuclide j (Bq s m^{-3}) is expressed as:

$$C_j = \frac{Q_{airj}}{V}$$

(I-15)

From the equations above the cloud diffusion factor χ (i.e. time integrated activity concentration in the air per unit of released activity (s m^{-3})) is given by:

$$\chi = \frac{C_j}{Q_j} = \frac{1 - \exp(-\frac{G}{V}\, t_{ex})}{G}$$

(I-16)

The total effective dose incurred via inhalation E_{inh} (Sv) is evaluated as:

$$E_{inh} = \sum_j \chi\, Q_j\, RF_j\, B\, e(g)_{inhj}$$

(I-17)

Where:

RF_j is respirable fraction of released airborne activity of radionuclide j (-), see explanations in Sections I-2.1 and I-2.2;

B is the worker breathing rate ($m^3\ s^{-1}$), see explanations in Section I-3.1;

$e(g)_{inhj}$ is the committed effective dose per unit intake by inhalation for radionuclide j ($Sv\ Bq^{-1}$), see explanations in Sections I-2.1 and I-6.

External exposure from submersion into the radioactive cloud can be evaluated as:

$$E_{sub} = \sum_j \chi\, Q_j\, e_{subj} \tag{I-18}$$

Where:

e_{subj} is the effective dose per time integrated activity concentration in the air for radionuclide j (Sv per Bq s m^{-3}), see explanations in Sections I-3.1 and I-6.

The total effective dose is expressed as sum of the inhalation and the external exposure doses:

$$E = E_{inh} + E_{sub} \tag{I-19}$$

I-4.2. Parameters and values

The user defined general parameters for the model are the released airborne activity and the respirable fraction. These parameters are explained in Table I-1.

The ‘complete mixing’ model specific input parameters are discussed in the table below.

TABLE I-3. ‘COMPLETE MIXING’ MODEL SPECIFIC PARAMETERS AND THEIR DEFAULT VALUES

Parameter	Notification	Default value	Comments
Volume of the room	V	600 (m^3)	Default value can be imagined as a 10×10 m wide and 6 m high room.
Air change rate	G	1.5 ($m^3\ s^{-1}$)	Correspond to the air change rate of 9 (AC h^{-1}) for the default room volume

I-4.3. Limitations to use of the model

The model assumes complete mixing of the released airborne activity within a certain volume and therefore shall be used for situations where actual conditions allow for such consideration. Complete mixing or close to complete mixing conditions may establish after some time following accident, especially in the smaller enclosures and where air is being mixed and exchanged by means of forced ventilation. Also, in comparison to ‘gradual mixing’ model, the use of ‘complete mixing’ model might be more appropriate for the exposure conditions where worker location is distant from the release source and / or close to the side wall of the room etc.

The model currently does not consider radioactive decay. Exposure from very short lived radionuclides and long worker exposure times may be overestimated.

The model should not be used for evaluation of exposure under the radioactive cloud developing conditions especially for exposure locations close to the accident point. In these cases use of the ‘gradual mixing’ or ‘no dilution’ models are more appropriate.

I-5. ‘GRADUAL MIXING / COMPLETE MIXING’ MODEL

I-5.1. Model description

The ‘gradual mixing / complete mixing’ model is a combination of the above described ‘gradual mixing’, see Section I-3, and complete mixing, see Section I-4, models.

The model considers the radioactive material release to be instantaneous and the material starts to disperse at a constant velocity in all directions of a hemisphere whose plane coincides with the floor of the room. When the cloud expands and fills up the room, the homogeneous mixing conditions are reached. The concentration in the room is then decreased due to radioactive decay and material removal by the room ventilation system. The switch from one model to another takes place when volume of the expanding hemisphere, cf. Eq. (I-2) and Eq. (I-3) in Section I-3.1, becomes equal to the volume of the room, cf. Section I-4.1:

$$\frac{2}{3}\,\pi\,\alpha^3\,t_2^3 = V \tag{I-20}$$

and the time when homogeneous mixing conditions are reached is calculated

$$t_2 = \sqrt[3]{\frac{3\,V}{2\,\pi\,\alpha^3}} \tag{I-21}$$

The worker exposure initial time t_1 is defined from data on the worker location r_1 and the cloud expansion speed α, see Table I-2. Then, basing on Eq. (I-7), Eq. (I-8), Eq. (I-9) and Eq. (I-10) the worker exposure within expanding cloud is evaluated.

When complete mixing conditions are reached, activity concentration in the room is described by Eq. (I-11) and Eq. (I-12), see Section I-4.1.

The time integrated activity in the air Q_{airj} of the radionuclide j (Bq s m^{-3}) for the exposure duration from t_2 till t_{exp} is calculated as:

$$Q_{airj} = \int_{t_2}^{t_{exp}} q_{airj}\, dt$$

(I-22)

and the cloud diffusion factor χ (i.e. time integrated activity concentration in the air per unit of released activity (s m^{-3})) is given by:

$$\chi = \frac{C_j}{Q_j} = \frac{1 - \exp(-\frac{G}{V}(t_{ex} - t_2))}{G}$$

(I-23)

The worker exposure (effective doses due to inhalation, submersion into the cloud and the total dose) is calculated as described by Eq. (I-17), Eq. (I-18) and Eq. (I-19), see Section I-4.1.

The doses calculated by both models are summed thus providing total worker exposure for period starting from the radioactive cloud development as hemisphere till worker evacuation when the complete mixing conditions in room are established.

I-5.2. Parameters and values

No additional parameters are necessary to those that are specified for the 'gradual mixing' and 'complete mixing' models, see Table I-2 in Sections I-3.2 and Table I-3 in I-4.2.

I-5.3. Limitations to use of the model

In general, model assumes that room dimensions (height, length, width) are of similar size. Therefore model should not be used for specific geometry enclosures like high shafts, long corridors etc.

'Complete mixing' model currently does not consider radioactive decay. Exposure from very short lived radionuclides and long worker exposure times may be overestimated.

I-6. DOSE CONVERSION FACTORS

As default, dose calculation models incorporate GSR Part 3 [I-3] provided values of committed effective dose per unit intake $e(g)_{inh}$ via inhalation (Sv Bq^{-1}) for workers. Values are selected from Table II-III following the lung absorption type as per Table II-V for compounds identified as "all compounds" or "all unspecified compounds" and more conservative value for 1 µm or 5 µm AMAD. Otherwise (i.e. when the lung absorption type is specified differently) the most conservative $e(g)_{inh}$ value is selected. For H-3 and C-14 more conservative values from Table II-IX are used.

Following the approach used in the IAEA Safety Report Series No. 19 [I-13], as default, the dose calculation models incorporate the U.S. Environmental Protection Agency Federal Guidance Report No. 12 [I-5] provided values of effective dose per time integrated activity concentration in the air e_{sub} (Sv per Bq s m^{-3}). The publication provides values for effective equivalent dose and equivalent skin dose separately. The effective dose factor is calculated as follows:

$$e_{sub} = h_E + 0.01\, h_T \tag{I-24}$$

Where:

h_E is the effective equivalent dose factor for air submersion (dose to skin is not included into summation);

h_T is the skin equivalent dose factor for air submersion.

The publication [I-5] provided dose factors do not include consideration of radiations emitted by radioactive decay products. Following the approach used in the IAEA Safety Report Series No. 19 [I-13], e_{sub} values for some of radionuclides are additionally upgraded to account for impact arising from their progenies. These radionuclides and considered progenies are listed in Table I-4 below:

TABLE I-4. RADIONUCLIDES AND THEIR PROGENIES WHICH EFFECTS TO EFFECTIVE DOSE ARE INCLUDED INTO DEFAULT DOSE FACTORS FOR AIR SUBMERSION

Radionuclide	Progeny *)
Bi-212	Tl-208 (0.3593), Po-212 (0.6407)
Ce-144	Pr-144 (0.9857)
Cs-137	Ba-137m (0.9457)
Mo-99	Tc-99m (0.8860), Tc-99
Ru-106	Rh-106
Te-131m	Te-131 (0.2220), I-131 (0.7780)

* Where relevant, progeny branching ratios are shown in brackets

The Ecolego tool [I-2] incorporated committed effective dose values per unit intake $e(g)_{inh}$ via inhalation and the effective dose values per time integrated activity concentration in the air e_{air} are listed in Table I-5 below:

TABLE I-5. DEFAULT VALUES OF EFFECTIVE DOSE PER UNIT INTAKE VIA INHALATION $e(g)_{inh}$ FOR WORKERS AND EFFECTIVE DOSE PER TIME INTEGRATED ACTIVITY CONCENTRATION IN THE AIR e_{sub}

No	Radionuclide	$e(g)_{inh}$ Sv Bq^{-1}	e_{sub} Sv per Bq s m^{-3}
1	Ac-228	2.9E-08	4.9E-14
2	Ag-110m	6.7E-09	1.4E-13
3	Am-241	3.9E-05	8.3E-16
4	As-76	9.2E-10	2.2E-14
5	At-211	9.8E-08	1.6E-15
6	Au-198	3.9E-10	2.0E-14
7	Bi-206	2.1E-09	1.6E-13
8	Bi-210	8.4E-08	2.6E-16

TABLE I-5. (Continued)

No	Radionuclide	$e(g)_{inh}$ Sv Bq^{-1}	e_{sub} Sv per Bq s m^{-3}
9	Bi-212	3.9E-08	7.4E-14
10	Br-82	8.8E-10	1.3E-13
11	C-14	5.8E-10	2.7E-18
12	Cd-109	9.6E-09	3.0E-16
13	Ce-141	3.1E-09	3.5E-15
14	Ce-144	3.4E-08	3.6E-15
15	Cm-242	4.8E-06	6.1E-18
16	Cm-244	2.5E-05	5.3E-18
17	Co-58	1.5E-09	4.8E-14
18	Co-60	9.6E-09	1.3E-13
19	Cr-51	3.0E-11	1.5E-15
20	Cs-134	9.6E-09	7.7E-14
21	Cs-135	9.9E-10	9.6E-18
22	Cs-136	1.9E-09	1.1E-13
23	Cs-137	6.7E-09	2.8E-14
24	Cu-64	6.8E-11	9.3E-15
25	Eu-154	5.0E-08	6.2E-14
26	Eu-155	6.5E-09	2.5E-15
27	Fe-55	9.2E-10	0.0E+00
28	Fe-59	3.0E-09	6.0E-14
29	Ga-67	1.1E-10	7.3E-15
30	H-3	1.8E-11	3.3E-19
31	Hg-197	2.8E-10	2.7E-15
32	Hg-197m	6.6E-10	4.2E-15
33	Hg-203	2.3E-09	1.1E-14
34	I-123	1.1E-10	7.4E-15
35	I-125	7.3E-09	5.4E-16
36	I-129	5.1E-08	3.9E-16
37	I-131	1.1E-08	1.8E-14
38	I-132	2.0E-10	1.1E-13
39	I-133	2.1E-09	3.0E-14
40	I-134	7.9E-11	1.3E-13
41	I-135	4.6E-10	8.1E-14
42	In-111	2.2E-10	1.9E-14
43	In-113m	1.9E-11	1.2E-14
44	Mn-54	1.1E-09	4.1E-14
45	Mo-99	3.6E-10	1.3E-14
46	Na-22	2.0E-09	1.1E-13
47	Na-24	5.3E-10	2.2E-13
48	Nb-95	1.4E-09	3.8E-14
49	Ni-59	2.2E-10	0.0E+00
50	Ni-63	5.2E-10	0.0E+00

TABLE I-5. (Continued)

No	Radionuclide	e(g)inh Sv Bq-1	esub Sv per Bq s m-3
51	Np-237	2.1E-05	1.0E-15
52	Np-239	1.1E-09	7.9E-15
53	P-32	1.1E-09	5.5E-16
54	Pa-231	1.3E-04	1.7E-15
55	Pa-233	3.1E-09	9.5E-15
56	Pb-210	1.1E-06	5.8E-17
57	Pd-103	1.2E-10	8.1E-17
58	Pd-107	3.3E-11	0.0E+00
59	Pd-109	2.1E-10	4.7E-16
60	Pm-147	4.7E-09	8.8E-18
61	Po-210	7.1E-07	4.2E-19
62	Pu-238	4.3E-05	5.3E-18
63	Pu-239	4.7E-05	4.4E-18
64	Pu-240	4.7E-05	5.1E-18
65	Pu-241	8.5E-07	7.4E-20
66	Pu-242	4.4E-05	4.3E-18
67	Ra-224	2.9E-06	4.8E-16
68	Ra-225	5.8E-06	3.1E-16
69	Ra-226	3.2E-06	3.2E-16
70	Rb-86	1.3E-09	5.3E-15
71	Rh-105	1.5E-10	3.8E-15
72	Rh-107	1.6E-11	1.5E-14
73	Ru-103	6.8E-10	2.3E-14
74	Ru-106	9.8E-09	1.1E-14
75	S-35	1.3E-09	3.2E-18
76	Sb-124	1.9E-09	9.3E-14
77	Sb-125	1.7E-09	2.0E-14
78	Se-75	1.7E-09	1.9E-14
79	Sn-113	7.9E-10	3.9E-16
80	Sr-85	5.6E-10	2.4E-14
81	Sr-87m	2.2E-11	1.5E-14
82	Sr-89	1.4E-09	4.5E-16
83	Sr-90	3.0E-08	1.0E-16
84	Tc-99	4.0E-10	2.9E-17
85	Tc-99m	2.0E-11	6.0E-15
86	Te-125m	6.7E-10	4.7E-16
87	Te-127m	2.0E-09	1.6E-16
88	Te-129m	1.8E-09	1.7E-15
89	Te-131m	1.2E-09	7.9E-14
90	Te-132	2.4E-09	1.0E-14
91	Th-228	3.1E-05	9.4E-17
92	Th-230	4.0E-05	1.8E-17

TABLE I-5. (Continued)

No	Radionuclide	e(g)inh Sv Bq-1	esub Sv per Bq s m-3
93	Th-232	4.2E-05	9.1E-18
94	Tl-201	7.6E-11	3.8E-15
95	Tl-202	3.1E-10	2.2E-14
96	U-232	3.5E-05	1.5E-17
97	U-234	8.5E-06	8.1E-18
98	U-235	7.7E-06	7.3E-15
99	U-238	7.3E-06	3.7E-18
100	Y-87	5.2E-10	2.2E-14
101	Y-90	1.6E-09	8.1E-16
102	Y-91	6.7E-09	6.5E-16
103	Zn-65	2.9E-09	2.9E-14
104	Zr-95	3.0E-09	3.6E-14

I-7. EMPLOYMENT OF THE MODELS

I-7.1. Calculation of release screening values

For quantification of impacts inside facility the SAFRAN tool [I-1] uses release screening values. Release screening values are calculated for the following exposure conditions:

- The effective dose to worker from release of any considered radionuclide is 0.1 mSv;
- The time from release till worker leaves the room is 1 hour;
- The distance from the release point to the worker exposure location is 1 m.

The considered worker exposure duration is rather long. It can be expected that following event of localized release the complete mixing conditions will establish within the most of reasonable size rooms. Therefore the ‘gradual mixing / complete mixing’ model is used for calculation of the release screening values.

Employment of the ‘gradual mixing / complete mixing’ model requires for specification of additional parameters:

- Volume of the room. Calculations are performed for the room of 50 m^3 volume.
- Air exchange rate for the room. It is conservatively assumed that there is no extraction of contaminated air from the room, i.e. air exchange rate is 0.
- It is conservatively assumed that all released activity is respirable.

The calculation parameters are summarized in Table I-6.

TABLE I-6. MODEL PARAMETERS USED FOR CALCULATION OF SCREENING RELEASE VALUES

Parameter	Notification	Value	Comments
Model definition for output of calculation results	Model	4	The output from the 'gradual mixing / complete mixing' model is used
Released airborne activity	Q	1 (Bq)	Unit release
Respirable fraction	RF	1 (-)	All released activity is respirable
Worker breathing rate	B	8.33E-04 (m^3 s^{-1})	Male conducting heavy work
Worker evacuation time	texp	3600 (s)	Exposure end time for screening release
Distance from the centre of the release source to the worker location, r_1	r1	1.0 (m)	Exposure distance for screening release
Cloud expansion speed, α	alfa	0.05 (m s^{-1})	Low speed for small room, no forced ventilation
Volume of the room	V	50 (m^3)	The cloud, expanding with default speed (α= 0.05 m/s), fills up this volume in approximately 58 s.
Air change rate	G	0 (m s^{-1})	No air change is assumed

Calculation of the screening release values have been conducted in two steps. At first the effective dose to worker is calculated for activity release of 1 Bq. In the second step the screening release value is obtained by scaling calculated dose from unit release to screening dose as follows:

$$Q_{S,j} = \frac{E_S}{E_j} Q_{unit} \tag{I-25}$$

Where:

$Q_{S,j}$ is the screening release value for particular radionuclide j (Bq);

Q_{unit} is 1 Bq is the unit release;

E_S is 1E-04 Sv is the considered screening dose for accidents;

E_j is the calculated worker exposure from unit release of particular radionuclide j (Sv).

The screening release calculation results and screening release values are presented in Table I-7 below.

TABLE I-7. CALCULATED UNIT RELEASE DOSE AND SCREENING RELEASE VALUES

No	Radionuclide	Unit release dose E, Sv	Screening release Q_S, Bq
1	Ac-228	1.82E-09	5.51E+04
2	Ag-110m	4.29E-10	2.33E+05
3	Am-241	2.44E-06	4.10E+01
4	As-76	5.92E-11	1.69E+06
5	At-211	6.13E-09	1.63E+04
6	Au-198	2.59E-11	3.86E+06
7	Bi-206	1.43E-10	6.98E+05
8	Bi-210	5.25E-09	1.90E+04
9	Bi-212	2.44E-09	4.09E+04
10	Br-82	6.48E-11	1.54E+06
11	C-14	3.63E-11	2.76E+06
12	Cd-109	6.00E-10	1.67E+05
13	Ce-141	1.94E-10	5.15E+05
14	Ce-144	2.13E-09	4.70E+04
15	Cm-242	3.00E-07	3.33E+02
16	Cm-244	1.56E-06	6.40E+01
17	Co-58	9.74E-11	1.03E+06
18	Co-60	6.10E-10	1.64E+05
19	Cr-51	1.99E-12	5.03E+07
20	Cs-134	6.06E-10	1.65E+05
21	Cs-135	6.19E-11	1.62E+06
22	Cs-136	1.27E-10	7.87E+05
23	Cs-137	4.21E-10	2.38E+05
24	Cu-64	4.95E-12	2.02E+07
25	Eu-154	3.13E-09	3.19E+04
26	Eu-155	4.07E-10	2.46E+05
27	Fe-55	5.75E-11	1.74E+06
28	Fe-59	1.92E-10	5.21E+05
29	Ga-67	7.42E-12	1.35E+07
30	H-3	1.13E-12	8.89E+07
31	Hg-197	1.77E-11	5.65E+06
32	Hg-197m	4.16E-11	2.41E+06
33	Hg-203	1.45E-10	6.92E+05
34	I-123	7.43E-12	1.35E+07
35	I-125	4.56E-10	2.19E+05
36	I-129	3.19E-09	3.14E+04
37	I-131	6.89E-10	1.45E+05
38	I-132	2.08E-11	4.82E+06
39	I-133	1.34E-10	7.49E+05

TABLE I-7. (Continued)

No	Radionuclide	Unit release dose E, Sv	Screening release Q_S, Bq
40	I-134	1.47E-11	6.81E+06
41	I-135	3.48E-11	2.87E+06
42	In-111	1.52E-11	6.59E+06
43	In-113m	2.09E-12	4.79E+07
44	Mn-54	7.18E-11	1.39E+06
45	Mo-99	2.35E-11	4.26E+06
46	Na-22	1.33E-10	7.50E+05
47	Na-24	4.96E-11	2.01E+06
48	Nb-95	9.04E-11	1.11E+06
49	Ni-59	1.38E-11	7.27E+06
50	Ni-63	3.25E-11	3.08E+06
51	Np-237	1.31E-06	7.62E+01
52	Np-239	6.94E-11	1.44E+06
53	P-32	6.88E-11	1.45E+06
54	Pa-231	8.13E-06	1.23E+01
55	Pa-233	1.95E-10	5.14E+05
56	Pb-210	6.88E-08	1.45E+03
57	Pd-103	7.51E-12	1.33E+07
58	Pd-107	2.06E-12	4.85E+07
59	Pd-109	1.32E-11	7.60E+06
60	Pm-147	2.94E-10	3.40E+05
61	Po-210	4.44E-08	2.25E+03
62	Pu-238	2.69E-06	3.72E+01
63	Pu-239	2.94E-06	3.40E+01
64	Pu-240	2.94E-06	3.40E+01
65	Pu-241	5.31E-08	1.88E+03
66	Pu-242	2.75E-06	3.64E+01
67	Ra-224	1.81E-07	5.52E+02
68	Ra-225	3.63E-07	2.76E+02
69	Ra-226	2.00E-07	5.00E+02
70	Rb-86	8.17E-11	1.22E+06
71	Rh-105	9.66E-12	1.03E+07
72	Rh-107	2.13E-12	4.70E+07
73	Ru-103	4.42E-11	2.26E+06
74	Ru-106	6.13E-10	1.63E+05
75	S-35	8.13E-11	1.23E+06
76	Sb-124	1.26E-10	7.95E+05
77	Sb-125	1.08E-10	9.28E+05
78	Se-75	1.08E-10	9.29E+05
79	Sn-113	4.94E-11	2.02E+06
80	Sr-85	3.68E-11	2.72E+06
81	Sr-87m	2.50E-12	4.00E+07

TABLE I-7. (Continued)

No	Radionuclide	Unit release dose E, Sv	Screening release Q_S, Bq
82	Sr-89	8.76E-11	1.14E+06
83	Sr-90	1.88E-09	5.33E+04
84	Tc-99	2.50E-11	4.00E+06
85	Tc-99m	1.70E-12	5.88E+07
86	Te-125m	4.19E-11	2.39E+06
87	Te-127m	1.25E-10	8.00E+05
88	Te-129m	1.13E-10	8.88E+05
89	Te-131m	8.09E-11	1.24E+06
90	Te-132	1.51E-10	6.63E+05
91	Th-228	1.94E-06	5.16E+01
92	Th-230	2.50E-06	4.00E+01
93	Th-232	2.63E-06	3.81E+01
94	Tl-201	5.04E-12	1.99E+07
95	Tl-202	2.10E-11	4.76E+06
96	U-232	2.19E-06	4.57E+01
97	U-234	5.31E-07	1.88E+02
98	U-235	4.81E-07	2.08E+02
99	U-238	4.56E-07	2.19E+02
100	Y-87	3.42E-11	2.93E+06
101	Y-90	1.00E-10	9.99E+05
102	Y-91	4.19E-10	2.39E+05
103	Zn-65	1.83E-10	5.45E+05
104	Zr-95	1.90E-10	5.26E+05

I-7.2. Calculation of diffusion factors for SAFRAN database

The SAFRAN tool [I-1] includes procedure for evaluation of worker exposure inside the facility due to accidental radionuclide releases to the air. The dose to worker is calculated as follows:

$$E = E_{inh} + E_{sub} = \sum_j Q_j \, \chi \, (B \; PF \; e(g)_{inhj} + e_{subj}) \tag{I-26}$$

Where:

Q_j is the release to air inside the facility of the j-th radionuclide (Bq);

χ is the cloud diffusion factor (h m^{-3});

B is the inhalation rate of the exposed individual (m^3 h^{-1});

$e(g)_{inhj}$ is the committed effective dose per unit intake by inhalation for radionuclide j (Sv Bq^{-1});

e_{subj} is the effective dose per time integrated activity concentration in the air for radionuclide j (Sv per Bq h m^{-3});

PF is the protection factor (due to use of respiratory protective equipment).

Values of the cloud diffusion factor for different exposure distances and exposure times are picked from the SAFRAN internal database. The user can also add own diffusion factor values.

Values of the cloud diffusion factor for SAFRAN internal database are calculated basing on the 'gradual mixing / complete mixing' model as follows:

$$\chi = \frac{1}{3600} (\chi_{GM} + \chi_{CM}) \tag{I-27}$$

Where:

χ_{GM} is the cloud diffusion factor calculated according to Eq. (I-7) when running the 'gradual mixing / complete mixing' model (s m^3);

χ_{CM} the cloud diffusion factor calculated according to Eq. (I-16) when running the 'gradual mixing / complete mixing' model (s m^3).

It can be seen from Eq. (I-7) and Eq. (I-16) that the cloud diffusion factor depends on exposure distance, cloud expansion speed, exposure time, room volume and the air exchange rate. Values for the SAFRAN internal data base are calculated conservatively assuming that there is no air extraction from the room. The cloud expansion speed therefore was considered being low, see discussion in Section I-3.2. Assumed cloud expansion speed is 0.05 (m s^{-1}).

Calculations are performed for the exposure distances 1m, 2m and the distance, above which diffusion factor for the same room volume becomes constant. Physically it means that complete mixing conditions within the room are established and the dose to worker depends on the exposure time. Calculations are performed for the exposure durations 1 min (1.67E-02 h), 5 min (8.33E-02 h), 10 min (1.67E-01 h), 0.5 h and 1 h.

Room volumes considered vary from relatively small rooms (50 m^3) to relatively large rooms (3000 m). With increasing of the room volume, variation of diffusion factor for the same exposure distance and exposure duration decreases. Therefore for large volume rooms, the diffusion factors calculated for 3000 m^3 room can be used as conservative extrapolation.

The diffusion factor calculation parameters and calculation results are summarized in Table I-8 below.

TABLE I-8. SAFRAN INTERNAL DATABASE OF CLOUD DIFFUSION FACTORS FOR AIRBORNE RELEASES INSIDE FACILITY

Exposure distance (m)	Exposure duration (h)	Room volume (m^3)	Diffusion factor (h/m^3)
1.0	1.67E-02	50	1.18E-03
1.0	8.33E-02	50	2.51E-03
1.0	1.67E-01	50	4.18E-03
1.0	5.00E-01	50	1.08E-02
1.0	1.00E+00	50	2.08E-02
2.0	1.67E-02	50	1.85E-04
2.0	8.33E-02	50	1.52E-03
2.0	1.67E-01	50	3.18E-03
2.0	5.00E-01	50	9.85E-03
2.0	1.00E+00	50	1.99E-02
3.0	1.67E-02	50	1.34E-05
3.0	8.33E-02	50	1.35E-03
3.0	1.67E-01	50	3.01E-03
3.0	5.00E-01	50	9.68E-03
3.0	1.00E+00	50	1.97E-02
1.0	1.67E-02	200	1.18E-03
1.0	8.33E-02	200	1.55E-03
1.0	1.67E-01	200	1.97E-03
1.0	5.00E-01	200	3.64E-03
1.0	1.00E+00	200	6.14E-03
2.0	1.67E-02	200	1.84E-04
2.0	8.33E-02	200	5.58E-04
2.0	1.67E-01	200	9.74E-04
2.0	5.00E-01	200	2.64E-03
2.0	1.00E+00	200	5.14E-03
5.0	1.67E-02	200	0.00E+00
5.0	8.33E-02	200	2.90E-04
5.0	1.67E-01	200	7.06E-04
5.0	5.00E-01	200	2.37E-03
5.0	1.00E+00	200	4.87E-03
1.0	1.67E-02	600	1.18E-03
1.0	8.33E-02	600	1.37E-03
1.0	1.67E-01	600	1.51E-03
1.0	5.00E-01	600	2.07E-03
1.0	1.00E+00	600	2.90E-03
2.0	1.67E-02	600	1.84E-04
2.0	8.33E-02	600	3.79E-04
2.0	1.67E-01	600	5.18E-04
2.0	5.00E-01	600	1.07E-03
2.0	1.00E+00	600	1.91E-03
7.0	1.67E-02	600	0.00E+00

TABLE I-8. (Continued)

Exposure distance (m)	Exposure duration (h)	Room volume (m^3)	Diffusion factor (h/m^3)
7.0	8.33E-02	600	7.78E-05
7.0	1.67E-01	600	2.17E-04
7.0	5.00E-01	600	7.72E-04
7.0	1.00E+00	600	1.61E-03
1.0	1.67E-02	1200	1.18E-03
1.0	8.33E-02	1200	1.34E-03
1.0	1.67E-01	1200	1.41E-03
1.0	5.00E-01	1200	1.69E-03
1.0	1.00E+00	1200	2.10E-03
2.0	1.67E-02	1200	1.84E-04
2.0	8.33E-02	1200	3.43E-04
2.0	1.67E-01	1200	4.13E-04
2.0	5.00E-01	1200	6.91E-04
2.0	1.00E+00	1200	1.11E-03
9.0	1.67E-02	1200	0.00E+00
9.0	8.33E-02	1200	3.10E-05
9.0	1.67E-01	1200	1.00E-04
9.0	5.00E-01	1200	3.78E-04
9.0	1.00E+00	1200	7.95E-04
1.0	1.67E-02	3000	1.18E-03
1.0	8.33E-02	3000	1.32E-03
1.0	1.67E-01	3000	1.35E-03
1.0	5.00E-01	3000	1.46E-03
1.0	1.00E+00	3000	1.63E-03
2.0	1.67E-02	3000	1.84E-04
2.0	8.33E-02	3000	3.28E-04
2.0	1.67E-01	3000	3.56E-04
2.0	5.00E-01	3000	4.67E-04
2.0	1.00E+00	3000	6.34E-04
12.0	1.67E-02	3000	0.00E+00
12.0	8.33E-02	3000	6.90E-06
12.0	1.67E-01	3000	3.47E-05
12.0	5.00E-01	3000	1.46E-04
12.0	1.00E+00	3000	3.12E-04

REFERENCES TO ANNEX I

[I-1] SAFRAN Tool and SAFRAN User's Guide, http://goto.iaea.org/safram

[I-2] ECOLEGO and ECOLEGO Toolbox, http://ecolego.facilia.se/ecolego/show/Ecoleg

[I-3] EUROPEAN COMMISSION, FOOD AND AGRICULTURE ORGANIZATION OF THE UNITED NATIONS, INTERNATIONAL ATOMIC ENERGY AGENCY, INTERNATIONAL LABOUR ORGANIZATION, OECD NUCLEAR ENERGY AGENCY, PAN AMERICAN HEALTH ORGANIZATION, UNITED NATIONS ENVIRONMENT PROGRAMME, WORLD HEALTH ORGANIZATION, Radiation Protection and Safety of Radiation Sources: International Basic Safety Standards, IAEA Safety Standards Series No. GSR Part 3, IAEA, Vienna (2014).

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ANNEX II. DESCRIPTION OF THE SAFRAN MODEL FOR EVALUATION OF PUBLIC EXPOSURE RESULTING FROM ACCIDENTAL RELEASE OF AIRBORNE RADIOACTIVE MATERIALS INTO THE ATMOSPHERE AND USER'S GUIDE

II-1 INTRODUCTION

This document describes the method used in the SAFRAN tool [II-1] for calculation of exposure arising from accidental release of airborne radioactive materials into the atmosphere. Model can be used for evaluation of public exposure to allow comparison with the relevant dose limiting criteria.

The model is based on the public exposure evaluation concept as described in the IAEA reports [II-2] and [II-3]. While both these reports in primary addresses impacts arising from routine (e.g. long time permanent) releases, the concept employed can be adapted for assessment of impacts arising from accidental (e.g. short time) releases. Another source which has also been extensively used is the German Incident calculation bases [II-4].

The material within this document is arranged as follows:

Section II-2 describes model for calculation of radionuclide dispersion in the atmosphere. Calculation of dispersion is based on the Gaussian plume model. Plume depletion considers radioactive decay, wet deposition (washout) and dry deposition (fallout). Building wake effects, if relevant, are accounted by modified-sigma method. Model evaluates radionuclides concentration in the air and contamination of the ground at a certain distances from the emission source.

Section II-3 describes model for evaluation of public exposure. Dose calculation includes evaluation of external exposure due to submersion into the plume, external exposure from contaminated ground, internal exposure due to inhalation, internal exposure due to consumption of contaminated food products. Following approach of Reference [II-3] doses are calculated for two members of the population: 1-2 years old child and adult.

Description of the models as provided in Sections II-2 and II-3 follows the same logic as these models are implemented in the Ecolego tool.

Section II-4 provides dose conversion and biosphere transfer factors used in the model

Section II-5 provides guidance on practical employment of the model in Ecolego tool.

Section II-6 includes some practical examples demonstrating how accident scenarios can be assessed by use of Ecolego tool.

Section II-7 describes calculation of SAFRAN databases of screening release values and released activity to dose conversion factors.

II-2. RADIONUCLIDE DISPERSION MODEL

II-2.1. Gaussian plume model

The concentration of airborne materials released into the atmosphere in the event of an accident is reduced in the downwind direction due to turbulent diffusion. The Gaussian plume model is one of the most widely used methods for estimating of downwind concentrations [II - 2], [II-3]. This is an empirical formula that is based on an analytical solution to the diffusion equation under the assumption of constant wind velocity and atmospheric conditions. For a continuous release from an elevated point source the Gaussian plume equation is:

$$c_{\mathrm{j}}\,(x, y, z) = \frac{q_j}{2\pi\sigma_y\sigma_z u}\exp\left(-\frac{y^2}{2\sigma_y^2}\right)\left[\exp\left(-\frac{(z-H_e)^2}{2\sigma_z^2}\right) + \exp\left(-\frac{(z+H_e)^2}{2\sigma_z^2}\right)\right] \tag{II-1}$$

Where:

$c_j(x,y,z)$ is the airborne material j concentration in the air (Bq/m^3) at a point (x, y, z) downwind to release;

x is the downwind distance (m).

y is the crosswind distance (m).

z is the height above ground (m).

q_j is the airborne material j release rate (Bq/s).

σ_y and σ_z are the diffusion parameters (m), which usually are functions of downwind distance x and atmospheric stability conditions. If dispersion conditions are influenced by certain obstacles (e.g. buildings) located nearby release location, diffusion parameters may depend on additional parameters, see Section II-2.1.3.

u is the mean wind speed (m/s).

H_e is effective height of release (m). H_e is the physical release height (e.g. stack height) point plus an allowance for any additional plume rise owing to momentum, for example fan driven exhaust, or buoyancy for significantly heated exhausts. Value of H_e may also be needed modified if dispersion conditions are influenced by certain obstacles (e.g. buildings) located nearby release location, see Section II-2.1.3.

II-2.1.1. Wind speed

The mean wind speed in Eq. (II-1) is the speed at the effective height of release. For the effective heights of release below 10 m, the wind speed at the height of 10 m is used. Wind speed for the effective height of release above 10 m is determined on the basis of wind speed u_1 at reference height z_1 = 10 m in accordance with the following equation:

$$u = u_1 \left(\frac{H_e}{z_1}\right)^m$$

(II-2)

Where:

u_1 is the mean wind speed (m/s) at the reference height z_1 = 10 m;

m is wind profile coefficient and depends on the atmospheric stability conditions (i.e. atmospheric stability category according to Pasquill) and the nature of the underlying terrain, see Table II-1 below.

TABLE II-1. WIND PROFILE COEFFICIENTS M FOR VARIOUS ATMOSPHERIC STABILITY CONDITIONS AND UNDERLYING TERRAINS

Underlying terrain	Diffusion category					
	A	B	C	D	E	F
Seas or lakes	0.03	0.05	0.06	0.08	0.10	0.12
Agricultural	0.10	0.15	0.20	0.25	0.35	0.40
Cities or woodlands	0.16	0.24	0.32	0.40	0.56	0.64

Definition of Pasquill atmospheric stability categories is provided in Table II-2. Stability category characteristics are as follows: A – extremely unstable, B – moderately unstable, C – slightly unstable, D – neutral (category is applicable to heavily overcast day or night conditions), E – slightly stable, F – moderately stable.

TABLE II-2. DEFINITION OF PASQUILL ATMOSPHERIC STABILITY CATEGORIES

Wind speed at height of 10 m (m s^{-1})	Daytime insolation			Night time conditions	
	Strong	Moderate	Slight	>3/8 cloudiness	≤3/8 cloudiness
<2	A	A-B	B		
2-3	A-B	B	C	E	F
3-5	B	B-C	C	D	E
5-6	C	C-D	D	D	D
>6	C	D	D	D	D

Note: the degree of cloudiness is defined as that fraction of the sky above the local apparent horizon that is covered by clouds.

The model reference values for wind profile coefficients and wind speed at the reference height are provided in Table II-3 and Table II-4 below.

TABLE II-3. REFERENCE VALUES OF WIND PROFILE COEFFICIENTS M FOR VARIOUS ATMOSPHERIC STABILITY CONDITIONS

Diffusion category					
A	B	C	D	E	F
0.09	0.20	0.22	0.28	0.37	0.42

TABLE II-4. REFERENCE VALUES OF WIND SPEED (m s^{-1}) AT THE REFERENCE HEIGHT z_1 = 10 m

Diffusion category					
A	B	C	D	E	F
1.0	2.0	4.0	5.0	3.0	2.0

II-2.1.2. Diffusion parameters

Diffusion parameters in Eq. (II-1)for non-disturbed dispersion conditions are calculated as follows:

$$\sigma_y = p_y\, x^{q_y}$$

$$\sigma_z = p_z\, x^{q_z}$$

(II-3)

Where:

σ_y is the diffusion parameter in crosswind direction (m);

σ_z is the diffusion parameter in vertical direction (m);

x is the downwind distance (m);

p_y, q_y, p_z, q_z – are diffusion coefficients. They are functions of effective height of release and atmospheric stability conditions (i.e. atmospheric stability category according to Pasquill) see Table II-5 below.

TABLE II-5. DIFFUSION COEFFICIENTS

Effective height of release, m	Atmospheric stability category	Diffusion coefficient			
		p_y	q_y	p_z	q_z
50	A	1.503	0.833	0.151	1.219;
	B	0.876	0.823	0.127	1.108
	C	0.659	0.807	0.165	0.996
	D	0.640	0.784	0.215	0.885
	E	0.801	0.754	0.264	0.774
	F	1.294	0.718	0.241	0.662
100	A	0.170	1.296	0.051	1.317
	B	0.324	1.025	0.070	1.151
	C	0.466	0.866	0.137	0.985
	D	0.504	0.818	0.265	0.818
	E	0.411	0.882	0.487	0.652
	F	0.253	1.057	0.717	0.486

TABLE II-5. (Continued)

Effective height of release, m	Atmospheric stability category	Diffusion coefficient			
		p_y	q_y	p_z	q_z
180	A	0.671	0.903	0.0245	1.500
	B	0.415	0.903	0.0330	1.320
	C	0.232	0.903	0.104	0.997
	D	0.208	0.903	0.307	0.734
	E	0.345	0.903	0.546	0.557
	F	0.671	0.903	0.484	0.500

For the effective height of release below 50 m, the data set for the 50 m is used.

For the effective height of release above 180 m, the data set for the 180 m is used.

For the effective height of release between 50 m and 100 m and between 100 m and 180 m, a logarithmic interpolation is carried out between tabulated values of p_y, or p_z and a linear interpolation between tabulated values of q_y or q_z as follows:

$$p_H = p_{UP}^{\left(\frac{H-H_{LO}}{H_{UP}-H_{LO}}\right)} p_{LO}^{\left(\frac{H_{UP}-H}{H_{UP}-H_{LO}}\right)}$$

$$q_H = \frac{(H-H_{LO})q_{UP} + (H_{UP}-H)q_{LO}}{H_{UP}-H_{LO}}$$

(II-4)

Where:

p_H are parameters p_y or p_z for specific height H;

p_{UP} and p_{LO} are parameters p_y or p_z for reference heights $H_{LO} < H < H_{UP}$, Table II-5;

q_H are parameters q_y or q_z for specific height H;

q_{UP} and q_{LO} are parameters q_y or q_z for reference heights $H_{LO} < H < H_{UP}$, Table II-5.

II-2.1.3. Consideration of the building wake effects

If the release is low and near buildings, the air flow rounding the building may cause the released material to get closer to the ground than in the case of an undisturbed flow. This is in the case if

- the effective height of release $H_e < (H_G + l_G)$ where l_G is defined as the smaller of two values, either building width b_G or height H_G; and

- the emission source is located either on/above the roof of the building or anywhere at a distance of $l_G/4$ from the building;
- or less than $3l_G$ away from the building in the downwind direction.

If the dispersion conditions as above are relevant, the effective height of release H_e has to be modified to the new value h_e as follows:

(a) $H_e > H_G \quad h_e = 0.5[3H_e - (H_G + I_G)]$
(b) $H_e \leq H_G \quad h_e = H_e - 0.5I_G$

(II-5)

Also:

- If h_e is smaller than $l_G/2$, then $h_e = l_G/2$ is used.
- If h_e is smaller than H_G, the diffusion parameters σ_y and σ_z are replaced by Σ_y and Σ_z:

$$\Sigma_y(x) = \left(\sigma_y^2(x) + \frac{I_G^2}{\pi}\right)^{0.5}$$

$$\Sigma_z(x) = \left(\sigma_z^2(x) + \frac{I_G^2}{\pi}\right)^{0.5}$$

(II-6)

II-2.1.4. Limitation of vertical dispersion with height

Limitation of vertical dispersion may be accounted by limiting values for diffusion parameter in vertical direction σ_z as indicated in TABLE II-6 below.

TABLE II-6. MAXIMAL VALUES OF DIFFUSION PARAMETER IN VERTICAL DIRECTION σ_z (m) FOR VARIOUS ATMOSPHERIC STABILITY CONDITIONS

Diffusion category					
A	B	C	D	E	F
1100	1100	800	800	-	-

II-2.2. Diffusion factor

The highest concentration leading to the most un-favorite exposure conditions are on the centre line of the plume. For evaluation of human exposure and activity deposition on a ground, a near ground plume concentration has to be considered. In case of the plume centre line (y=0) and near ground (z=0) conditions, the Eq. (II-1) simplifies to:

$$c_j\ (x, y = 0, z = 0) = \frac{q_j}{\pi \sigma_y \sigma_z u} \exp\left(-\frac{H_e^2}{2\sigma_z^2}\right)$$

(II-7)

The diffusion factor is defined as the time integrated activity concentration at ground level and on the diffusion axis (y=0, z=0) per unit of released activity.

The release period time integrated activity concentration in the air (Bq s m^{-3}) is:

$$C_j = \int_0^t c_j\ (x, y = 0, z = 0) dt = \frac{Q_j}{\pi \sigma_y \sigma_z u} \exp\left(-\frac{H_e^2}{2\sigma_z^2}\right)$$

(II-8)

Where:

Q_j is the total release of airborne material j (Bq) over time period t.

Thus, the diffusion factor χ (s m^{-3}) is calculated as follows:

$$\chi = \frac{C_j}{Q_j} = \frac{1}{\pi \sigma_y \sigma_z u} \exp\left(-\frac{H_e^2}{2\sigma_z^2}\right)$$

(II-9)

The diffusion factor is atmospheric dispersion conditions dependent parameter.

II-2.3. Consideration of fallout, washout, radioactive decay and release duration

The radionuclide dry fallout, washout and radioactive decay are additional factors, which decrease radionuclide concentration in the air in the downwind direction. These phenomena can be accounted by additional multiplies to the diffusion factor. The radionuclide fallout, washout and radioactive decay corrected diffusion factor χ_A (s m^{-3}) is expressed as:

$$\chi_{Aj} = \chi\ f_{Aj}\ f_{Rj}\ f_{Zj}\ f_\chi$$

(II-10)

Where:

f_{Aj} is the airborne concentration depletion factor due to dry deposition (fallout factor), see Section II-2.3.1 below;

f_{Rj} is the airborne concentration depletion factor due to interaction with precipitation like rain or snow (washout factor), see Section II-2.3.2 below;

f_{Zj} is airborne concentration depletion factor due to radioactive decay (radioactive decay factor), see Section II-2.3.3 below;

f_X is factor which accounts for duration of release, see Section II-2.3.4 below.

The f_{Aj}, f_{Rj} f_{Zj} are radionuclide dependent parameters, thus the corrected diffusion factor becomes atmospheric dispersion conditions and radionuclide dependent parameter.

The radionuclide fallout, washout and radioactive decay corrected time integrated ground level activity concentration in the air C_{Aj} (Bq s m^{-3}) is calculated:

$$C_{Aj} = Q_j \, \chi_{Aj} \tag{II-11}$$

II-2.3.1. Fallout factor

The airborne concentration depletion factor for dry deposition in downwind direction from the release source till distance x is calculated:

$$f_{Aj} = exp\left(-\frac{v_{gj}}{u}\sqrt{\frac{2}{\pi}}\int_0^x \frac{1}{\sigma_z} exp\left(-\frac{H_e^2}{2\sigma_z^2}\right) dx\right) \tag{II-12}$$

Where:

v_{gj} is the radionuclide specific dry deposition velocity (m s^{-1}). Reference values are provided in Table II-7 below.

TABLE II-7. REFERENCE VALUES FOR THE DRY DEPOSITION VELOCITY

Airborne material	Deposition velocity (m s^{-1})
Particulates (aerosols)	0.0015
Elemental iodine	0.01
Organic iodine compounds	0.0001

II-2.3.2. Washout factor

The airborne concentration depletion factor for washout in downwind direction from the release source till distance x is calculated:

$$f_{Rj} = \exp\left(-\frac{\Lambda_j \, x}{u}\right) \tag{II-13}$$

Where:

Λ is the radionuclide specific washout coefficient (s^{-1}). Washout coefficient is calculated as follows:

$$\Lambda_{\mathrm{j}} = \Lambda_{0j}\left(\frac{I}{I_0}\right)^{\mathrm{K}}$$

(II-14)

Where:

Λ_{0j} is the washout coefficient (s^{-1}) for precipitation intensity I_0 = 1 mm h^{-1}. Reference values are provided in Table II-8 below.

I is the precipitation intensity (mm h^{-1}). Reference values are provided in Table II-9 below.

K is the exponent. For particulate material and iodine K = 0.8.

TABLE II-8. REFERENCE VALUES FOR THE WASHOUT COEFFICIENT Λ_{0j}

Airborne material	Washout coefficient (s^{-1}) for the rain intensity I_0
Particulates (aerosols)	7E-05
Elemental iodine	7E-05
Organic iodine compounds	7E-07

TABLE II-9. REFERENCE VALUES FOR THE PRECIPITATION INTENSITY I

Release duration	Precipitation intensity (mm h^{-1})	Release relevant atmospheric stability conditions (diffusion categories)
up to 8 hours	5	C, D, E
from 8 to 24 hours	2	D
from 24 to 72 hours	1	D
from 3 to 7 days	0.5	D

II-2.3.3. Radioactive decay factor

The airborne concentration depletion factor for radioactive decay in downwind direction from the release source till distance x is calculated:

$$f_{Zj} = \exp\left(-\frac{\lambda_j\, x}{u}\right)$$

(II-15)

Where:

λ_j is the radionuclide specific radioactive decay constant (s^{-1}).

II-2.3.4. Consideration of release duration

Concentration of material in the air as calculated by Gaussian plume model is based on the assumption of constant wind velocity and atmospheric conditions. This is correct for the short time release. In case of prolonged release, the radionuclide concentration in the air may be reduced to account for the variability of the wind direction and weather conditions over this period. Concentration reduction factor f_T could be as follows, see Table II-10 below.

TABLE II-10. MULTIPLIES FOR REDUCTION OF THE AIRBORNE MATERIAL CONCENTRATION IN THE AIR DEPENDING ON RELEASE DURATION

Release duration	Concentration reduction factor	Release relevant atmospheric stability conditions (diffusion categories)
up to 8 hours	1	A, B, C, D, E, F
from 8 to 24 hours	½	C, D, E, F
from 24 to 72 hours	¼	C, D, E, F
from 3 to 7 days	1/8	C, D, E

II-2.4. Activity deposition and concentration on the ground

As it is described in the previous section, dry fallout and washout are processes which result in removal of airborne material from the plume with subsequent its deposition onto the ground surface. The amount of radionuclides depositing onto the ground depends on activity concentration in the air and activity fallout and washout intensity. Time integration is used for consideration of deposition over the whole release period. The concentration of radionuclide j deposited onto the ground C_{G0j} (Bq m^{-2}) can be expressed in the form:

$$C_{G0j} = Q_j(F_j + W_j) \tag{II-16}$$

Where:

Q_j is the total release of radionuclide j (Bq) over considered time period;

F_j is the radionuclide j specific release time integrated dry deposition (fallout) factor for ground contamination (m^{-2}), see Section II-2.4.1 below;

W_j is the radionuclide j specific release time integrated wet deposition (washout) factor for ground contamination (m^{-2}), see Section II-2.4.2 below;

f_T is factor which considers duration of release, see Section II-2.3.4.

Considering deposition of radionuclides onto aerial (i.e. above ground) surfaces of the plants the amount of the wet deposition is reduced assuming that only a part of activity adheres to the plant. The concentration of radionuclide j deposited onto aerial surfaces of the plants C_{GA0j} (Bq m^{-2}) is calculated:

$$C_{GA0j} = Q_j(F_j + f_w\, W_j) \tag{II- 17}$$

Where:

f_w is the fraction of activity remaining on the plant surface in case of wet deposition. The reference value is $f_w = 0.3$.

Deposited radionuclides result contamination of the ground and ground plants, migrate deeper into the soil and from there radionuclides can be further absorbed by ground plants through the root uptake. The total concentration of activity deposited on to the ground C_{G0} (Bq m^{-2}) is calculated as sum of the radionuclide specific ground concentrations:

$$C_{G0} = \sum_j C_{G0j}$$

(II-18)

II-2.4.1. Dry deposition factor for the ground contamination

The time integrated dry deposition factor for ground contamination on the diffusion axis F_j (m^{-2}) is calculated:

$$F_j = v_{gj}\ \chi_{Aj}$$

(II-19)

Where:

v_{gj} is the radionuclide j specific dry deposition velocity (m s^{-1}). Reference values are provided in the Table II-7.

χ_{Aj} is the radionuclide j specific atmospheric diffusion factor (s m^{-3}), corrected for airborne concentration depletion due to fallout, washout and radioactive decay till the location concerned, see Eq. (II-10).

II-2.4.2. Wet deposition factor for the ground contamination

The time integrated wet deposition factor for ground contamination on the diffusion axis W_j (m^{-2}) is calculated:

$$W_j = f_\chi \frac{\Lambda_j}{\sqrt{2\pi}\sigma_y u}$$

(II-20)

Where:

Λ_j is the radionuclide j specific washout coefficient (s^{-1}), see Eq. (II-14) and Section II-2.3.2;

f_X is factor which accounts for duration of release. In case of prolonged release, see Section II-2.3.4, the amount of wet deposition may be reduced to account for the variability of the wind direction and weather conditions over this period.

II-3. PUBLIC EXPOSURE MODEL

The radioactive materials released from a nuclear facility in the event of accident can cause human exposure via number of different exposure pathways. In general, the following exposure pathways have to be taken into consideration with regard to the release of radioactive materials into the atmosphere:

- External exposure by radiation from the radioactive air plume (air submersion);
- External exposure by radiation from contaminated ground (ground radiation);
- Internal exposure by radionuclides which are inhaled with the air (inhalation);
- Internal exposure as result of consumption of radionuclide contaminated foodstuffs (ingestion).

In order to determine the total dose received by a human, the dose contributions of the relevant radionuclides via various exposure pathways have to be summed.

II-3.1. Air submersion dose

Exposure due to submersion into radioactive air plume (i.e. air submersion dose) is most simple estimated by use of a semi infinitive cloud model. Implicit in this approach is the assumption that the cloud activity concentration is constant over the volume of the plume from which radiation are reaching the point at which the dose is to be evaluated.

Assuming radionuclide concentration in the air at the location concerned represents concentration of semi infinitive cloud the air submersion dose is calculated:

$$E_{subj} = e_{subj}\, C_{Aj} \tag{II-21}$$

Where:

E_{subj} is the radionuclide j specific effective dose from air submersion (Sv).

e_{subj} is the radionuclide j specific effective dose per time integrated activity concentration in the air (Sv per Bq s m^{-3}). Following approach given in Reference [II-3], dose coefficient values can be taken from [II-5]. Reference values are explained and provided in Section II-4.

C_{Aj} is the radionuclide j specific time integrated ground level activity concentration in the air (Bq s m^{-3}), see Eq. (II-11).

If the concentration distribution in the plume is sufficiently non uniform to invalidate semi infinitive cloud approach, a finite cloud model must be used (the finite cloud model involves representing the plume by a series of volume sources and integration over these sources). Because of the long mean free path of gamma rays in air, use of the semi infinitive cloud model can lead to considerable error close to the discharge point and under poor dispersion conditions, even at a distance, particularly for elevated releases. At large distances or closer, under good dispersion conditions, the predictions of both models converge. Beta rays have a range of only a few metres in air. Thus the beta radiation dose is proportional to the radionuclide concentration in the air at the exposure location concerned.

The total effective dose from air submersion of all radionuclides concerned is sum of the radionuclide specific effective doses:

$$E_{sub} = \sum_{j} E_{sub\,j}$$

(II-22)

II-3.2. Ground radiation dose

Similar to exposure from the radioactive air plume, radiation emitted from contaminated ground within a radius of some hundred meters around the point of concern can contribute to exposure. To simplify matters, it is sufficiently accurate to assume that the deposited activity within the whole radius is the same as that at the point of concern. The human dose is proportional to the exposure duration. At the same time, the radionuclide concentration is decreasing due to radioactive decay. The time period t_G averaged ground radiation dose E_G (Sv) therefore is calculated:

$$E_{Gj} = e_{gj}\, C_{G0j} \left(\frac{1 - \exp(-\lambda_{eff}\, t_G)}{\lambda_{eff}} \right) f_G\, b_G$$

(II-23)

Where:

E_{Gj} is the radionuclide j specific effective dose from the contaminated ground (Sv).

e_{gj} is the radionuclide j specific effective dose per time integrated activity concentration on the ground (Sv per Bq s m^{-2}). Following approach in Reference [II-3], dose coefficient values can be taken from the reference [II-8]. Reference values are explained and provided in Section II-4.

C_{G0j} is the radionuclide j activity concentration on the ground surface at the beginning of exposure (Bq m^{-2}), see Eq. (II-16).

t_G is the exposure duration for ground radiation dose (s). Examples of some frequently used or radiological safety requirements related exposure durations are shown in the Table II-11 below.

f_G is the fraction of the exposure duration t_G for which the human is exposed by ground radiation. The value of f_G will vary depending on particular circumstances of the exposure. For example, if annual dose is calculated (t_G = 3.16E+07 s) and human during the year spends on contaminated ground 1600 h (or 5.76E+06 s) then value of f_G is 5.76E+06 / 3.16E+07 = 0.18. Reference value for f_G is conservatively set to 1 which means that human is exposed during the whole considered exposure duration.

b_G is correction factor for the consideration of the soil roughness and activity migration into deeper soil layers. Defines fraction of activity which remains onto the ground surface starting from the second year after activity deposition and contributes to ground shine dose, i.e. value 1 means that none of activity migrates from the ground surface and value 0 means that all activity migrates out from the ground surface. The default value is set to 0.5.

λ_{eff} is the effective decay constant in the ground for the radionuclide j (s^{-1}):

$$\lambda_{eff} = \lambda_j + \lambda_g \tag{II-24}$$

Where:

λ_j is the radionuclide j specific radioactive decay constant (s^{-1}).

λ_g is the constant for contaminant removal from the ground by environmental processes other than radioactive decay. Parameter value is dependent on characteristics of considered exposure location such as climate, ground type, vegetative cover, chemical form of the radionuclide etc. Possible values for pasture or crop fields are discussed under Section II-3.4.2. Reference value is conservatively set to be 0 (s^{-1}). Thus no other radionuclide removal from the soil mechanisms excepting for radioactive decay are taken into consideration. Such approach can be considered as correct for relatively short exposure durations (e.g. up to 1 year). For longer exposure durations ground radiation dose is expected to be overestimated.

TABLE II-11.EXAMPLES OF SOME FREQUENTLY USED OR RADIOLOGICAL SAFETY REQUIREMENTS RELATED EXPOSURE DURATIONS

Parameter	Duration (s)
1 hour exposure	3600 = 3.60E+03
8 hours exposure	3600*8 = 2.88E+04
1 day (24 h) exposure	3600*24 = 8.64E+04
1 year exposure	3600*24*365.25 = 3.16E+07
5 years exposure	3600*24*365.25*5 = 1.58E+08
Whole life (53 years) exposure for adult	3600*24*365.25*53 = 1.67E+09

The total effective dose from ground radiation of all radionuclides concerned is sum of the radionuclide specific effective doses:

$$E_G = \sum_j E_{Gj} \tag{II-25}$$

II-3.3. Inhalation dose

The inhalation dose is proportional to the radionuclide concentration in the air at the location concerned, breathing rate and the exposure time. Maximal dose is calculated considering exposure during the whole duration of release:

$$E_{inhj} = e(g)_{inhj}\, B\, C_{Aj}\, RF \tag{II-26}$$

Where:

$e(g)_{inhj}$ is the committed effective dose per unit intake by inhalation for radionuclide j by the age group g (Sv Bq^{-1}). Reference values are explained and provided in Section II-4.

B is the average human respiratory rate ($m^3\ s^{-1}$) over considered exposure period. Respiratory rate over certain period depends on human age and levels / duration of physical activity [II - 6]. Reference values for the human breathing rate are provided in the Table II-12.

C_{Aj} is the radionuclide j specific time integrated ground level activity concentration in the air (Bq s m^{-3}), see Eq. (II-11).

RF is the respirable fraction, i.e. is a fraction of the total release of airborne material that is effectively inhalable. It is commonly assumed that respirable airborne activity includes particles of 10 µm Aerodynamic Equivalent Diameter (AED) and less. Reference value for RF is conservatively set to 1.

TABLE II-12. REFERENCE VALUES FOR THE HUMAN RESPIRATORY RATE

Exposure duration	Respiratory rate ($m^3\ s^{-1}$)	
	Child, from 1 up to 2 years	Adult, above 17 years
Up to 8 hours	8.7E-05	3.8E-04
More than 8 hours	6.0E-05	2.6E-04

The total effective dose from inhalation of all radionuclides concerned is sum of the radionuclide specific effective doses:

$$E_{inh} = \sum_j E_{inhj} \tag{II-27}$$

II-3.4. Ingestion dose

The ingestion dose assessment, similar to References [II-2] and [II-3], considers two main paths by which radionuclides may enter human body, i.e. the crop field path and the pasture field path. The crop field path accounts for exposure due to consumption of products grown up at the contaminated crop field or garden. The pasture path accounts for exposure due to consumption of milk and meat from animals which have been feed by grass from contaminated pasture field.

The foodstuff is subdivided into following main four groups:

- Green vegetables;
- Other plant products;
- Milk;
- Meat.

Contamination of the garden foodstuff and pasture grass results from direct deposition of radioactive materials on the vegetation and from uptake of radioactive materials from contaminated soil through the roots. The intake of radionuclides by human is both direct (green vegetables, other plant products) and indirect (milk, meat). Location of the garden and location of the pasture may also differ. Ingestion dose calculation method for the four groups of foodstuff is detailed in the subsections below.

The human nutrition, behaviour and biosphere parameters used in the dose calculations and their reference values are provided in Table II-13 and Table II-14 below. The element specific transfer factors for terrestrial foods are provided in Section II-4.

TABLE II-13. HUMAN NUTRITION PARAMETERS AND REFERENCE VALUES

Parameter	Child, from 1 up to 2 years	Adult, above 17 years
Annual consumption of leafy (green) vegetables (kg)	18	39
Annual consumption of other plant products (grains, potatoes, roots etc.) (kg)	366	610
Annual consumption of milk and milk products (kg)	480	390
Annual consumption of meat and meat products (kg)	26	180

TABLE II-14. HUMAN BEHAVIOR AND BIOSPHERE PARAMETERS AND REFERENCE VALUES

Parameter	Value	Comment
Environmental removal rate for plant surfaces (s^{-1})	5.7E-07	Corresponds to decay half-life of 14 days
Area-related mass of rooting zone for garden soil (kg m^{-2})	280	Based on rooting depth 0-20 cm with average soil density 1400 kg m^{-3}
Area-related mass of rooting zone for pasture soil (kg m^{-2})	120	Based on rooting depth 0-10 cm with average soil density 1200 kg m^{-3}
Annual yield (fresh weight) of green vegetables (kg m^{-2})	1.6	
Annual yield (fresh weight) of other plants (kg m^{-2})	2.4	
Annual yield (fresh weight) of pasture grass (kg m^{-2})	0.85	
Fodder daily consumption for milk producing animal (kg d^{-1})	65	Value for large cattle
Fodder daily consumption for meat producing animal (kg d^{-1})	65	Value for large cattle
Time between harvest and consumption of other plant products (s)	5.2E+06	Corresponds to 60 days
Time between slaughtering and meat consumption (s)	1.7E+06	Corresponds to 20 days

II-3.4.1. Time integrated activity concentration on aerial parts of the plants

For the time interval starting from initial activity deposition till certain time t_K, the time integrated activity concentration of radionuclide j on aerial parts of the plant C_{GPj} (Bq m^{-2}) is:

$$C_{GAj} = \frac{1 - \exp(-\lambda_{effA}\, t_K)}{\lambda_{effA}\, t_K} C_{GA0j} \tag{II-28}$$

Where:

C_{GA0j} is the initial concentration of radionuclide j deposited onto aerial surfaces of the plants (Bq m^{-2}), see Eq. (II-17).

t_K is the considered expose period for ingestion dose (s). Examples are given in Table II-11, above.

λ_{effA} is the effective decay constant on aerial parts of the plant for the radionuclide j (s^{-1}):

$$\lambda_{effA} = \lambda_j + \lambda_{ap}$$

(II-29)

Where:

λ_j is the radionuclide j specific radioactive decay constant (s^{-1}).

λ_{ap} is the constant for contaminant removal from aerial parts of the plant by other environmental processes than radioactive decay. Reference value is λ_{ap} = 5.7E-07 (s^{-1}) which correspond to decay half-life of 14 days.

II-3.4.2. Time integrated activity concentration in the soil

For the time interval starting from initial activity deposition till certain time t_B, the time integrated activity concentration of radionuclide j in the soil C_{Gj} (Bq m^{-2}) is:

$$C_{Gj} = \frac{1 - \exp(-\lambda_{effG}\, t_B)}{\lambda_{effG}\, t_B} C_{G0j}$$

(II-30)

Where:

C_{G0j} is the concentration of radionuclide j deposited onto the ground (Bq m^{-2}), see Eq. (II-16);

t_B is the considered expose period (s);

λ_{effG} is the effective decay constant in the soil root uptake zone for the radionuclide j (s^{-1}):

$$\lambda_{effG} = \lambda_j + \lambda_{rp}$$

(II-31)

Where:

λ_j is the radionuclide j specific radioactive decay constant (s^{-1}).

λ_{rp} is the constant for contaminant removal from the plant root uptake zone by other environmental processes than radioactive decay. Reference values are provided in the table below.

TABLE II-15. REFERENCE VALUES FOR CONSTANT FOR CONTAMINANT REMOVAL FROM THE PLANT ROOT UPTAKE ZONE BY OTHER ENVIRONMENTAL PROCESSES (s^{-1})

Element *	Garden	Pasture
Tc	1E-08	2E-08
Sr, Ru, I	1E-09	2E-09
Cs	1E-10	2E-10
Actinides	1E-11	2E-11

* Reference values for elements not indicated in the table can be conservatively assigned with values as for Actinides. Value 1E-11 (s^{-1}) corresponds to element decay half-life of about 2200 years.

For the garden exposure pathway it is considered that the all on the garden soil surface deposited activity accumulates in the soil volume up to certain depth, relevant for root uptake. The concentration of the radionuclide j in the (dry) garden soil C_{GGj} (Bq kg^{-3}) is:

$$C_{GGj} = \frac{C_{Gj}}{P_G} \tag{II-32}$$

Where:

P_G is the area-related mass of garden soil to the certain depth (kg m^{-2}). As reference value P_G = 280 (kg m^{-2}) is used, see Table II-14. This corresponds to ploughshare of the 20 cm depth with average soil density of 1400 (kg m^{-3}).

For the pasture exposure pathway it is considered that all on the pasture surface deposited activity accumulates in the pasture soil volume up to certain depth, relevant for root uptake. The mass concentration of the radionuclide j in the (dry) pasture soil C_{GPj} (Bq kg^{-3}) is:

$$C_{GPj} = \frac{C_{Gj}}{P_P} \tag{II-33}$$

Where:

P_P is the area-related mass of the pasture soil to certain depth (kg m^{-2}). As reference value P_P = 120 (kg m^{-2}) is used, see Table II-14. This corresponds to 10 cm depth pasture soil layer with average soil density of 1200 (kg m^{-3}).

II-3.4.3. Exposure due to consumption of green vegetables

It is considered that green vegetables are grown in a garden and the whole annual harvest is available for contamination during the accident. Green vegetables are consumed continuously without particular delay (e.g. for storage etc.) between collection of green vegetables from garden and ingestion of them by human. Possible (partial) removal of activity due to wash up of vegetables before consumption is not considered.

II-3.4.3.1 Activity admission through aerial parts of green vegetables

Considered exposure period t_K time integrated concentration of radionuclide j in the plants due to activity admission through the aerial parts of the plant C_{Gv1j} (Bq kg^{-1}) is:

$$C_{Gv1j} = \frac{C_{GAj}}{Y_{Gv}}$$

(II-34)

Where:

C_{GAj} is the considered exposure period t_K time integrated activity concentration of radionuclide j on aerial parts of the plant (Bq m^{-2}), see Eq. (II-28);

Y_{Gv} is the annual yield (fresh weight) of green vegetables. The reference value is $Y_{Gv} = 1.6$ (kg m^{-2}), see Table II-14.

The amount of green vegetables M_{GV1} (kg) consumed during considered exposure period t_K (s) is evaluated basing on continuous consumption rate:

$$M_{Gv1} = U_{Gv}\frac{t_K}{t_a}$$

(II-35)

Where:

U_{Gv} is the annual consumption of the green vegetables (kg). Reference human nutrition parameters are given in Table II-13.

t_K is the considered exposure period (s).

$t_a = 3.15E+07$ is the conversion factor, amount of seconds per year (s).

Consumption of annual harvest of green vegetables is calculated when assuming $t_K = t_a$, and this will lead to evaluation of the maximal annual exposure arising from activity admission through areal parts of plants. Exposure times $t_K > t_a$ are not used thus assuming that the accident contaminates only current season aerial parts of plants.

Effective dose E_{Gv1} (Sv) to human due to admission of radionuclide j through the aerial parts of green vegetables is:

$$E_{Gv1j} = e(g)_{ingj}\, C_{Gv1j}\, M_{Gv1}$$

(II-36)

Where:

$e(g)_{ingj}$ is the committed effective dose per unit intake by ingestion for radionuclide j by the group of age g (Sv Bq^{-1}) [II-7]. Reference values are explained and provided in Section II-4. Other parameters are defined above.

II-3.4.3.2 Activity admission through root uptake of green vegetables

Considered exposure period t_B time integrated concentration of the radionuclide j in the (fresh matter of) plant due to uptake from the soil C_{Gv2j} (Bq kg^{-1}) is:

$$C_{Gv2j} = C_{GGj}\, T_{Plj}$$

(II-37)

Where:

C_{GGj} is the considered exposure period t_B time integrated concentration of radionuclide j in the soil (Bq kg^{-1}), see Eq. (II-32).

T_{Plj} is the element j related garden soil to plant activity transfer factor (represents concentrations ratio in Bq kg^{-1} of fresh matter of the plant to Bq kg^{-1} of dry matter of the soil). Reference values for garden soil to plant transfer factors are provided in Section II-4.

The amount of green vegetables (kg) consumed during considered exposure period t_B (s) is calculated similarly to Eq. (II-35) and is:

$$M_{Gv2} = U_{Gv} \frac{t_B}{t_a}$$

(II-38)

The human exposure due to plants root uptake may be relevant for period t_B exceeding 1 year. Thus the long term effect due to residual contamination of the soil can be taken into account.

Effective dose E_{Gv2} (Sv) to human due to admission of radionuclide j through root uptake of green vegetables is:

$$E_{Gv2j} = e(g)_{ingj}\, C_{Gv2j}\, M_{Gv2}$$

(II-39)

II-3.4.3.3 Total dose due to consumption of green vegetables

Total radionuclide j specific effective dose due to consumption of green vegetables is:

$$E_{Gvj} = E_{Gv1j} + E_{Gv2j}$$

(II-40)

The total effective dose due to consumption of green vegetables of all radionuclides concerned is sum of the radionuclide specific effective doses:

$$E_{Gv} = \sum_{j} E_{Gvj}$$

(II-41)

II-3.4.4. Exposure due to consumption of other plant products

Differently to consumption of green vegetables, it is considered that other plant products (grains, potatoes, roots etc.) are consumed after some delay following to the harvest, see Table II-14.

II-3.4.4.1 Activity admission through areal parts of other plants

Considered exposure period t_K time integrated concentration of the radionuclide j in the plant at the consumption time due to activity admission through aerial parts of the plant C_{Op1j} (Bq kg^{-1}) is:

$$C_{Op1j} = \frac{C_{GAj} \exp(-\lambda_j t_{Op})}{Y_{Op}}$$

(II-42)

Where:

C_{GAj} is the considered exposure period t_K time integrated activity concentration of radionuclide j on aerial parts of the plant (Bq m^{-2}), see Eq. (II-28).

Y_{Op} is the annual yield (fresh weight) of other plants. The reference value is used $Y_{Op} = 2.4$ (kg m^{-2}), see Table II-14.

λ_j is the radionuclide j specific radioactive decay constant (s^{-1}).

t_{Op} is the time between harvesting and consumption of other plant products (s). The reference value is used $t_{Op} = 5.2E+06$ s, which correspond to 60 days duration, see Table II-14.

The amount of other plant products (kg) consumed during considered time period t_K is:

$$M_{Op1} = U_{Op} \frac{t_K}{t_a}$$

(II-43)

Where:

U_{Op} is the annual consumption of other plant products (kg);

t_K is the considered exposure period (s);

t_a = 3.15E+07 is the conversion factor, amount of seconds per year (s).

Effective dose E_{Op1j} (Sv) to human due to admission of radionuclide j through the aerial parts of other plant products is:

$$E_{Op1j} = e(g)_{ingj}\, C_{Op1j}\, M_{Op1} \tag{II-44}$$

Where:

$e(g)_{ingj}$ is the committed effective dose per unit intake by ingestion for radionuclide j by the group of age g (Sv Bq^{-1}) [II-7]. Reference values are explained and provided in Section II-4. Other parameters are defined above.

II-3.4.4.2 Activity admission through root uptake of other plant products

Considered exposure period t_B time integrated concentration of the radionuclide j in the plant at the consumption time due to activity admission through root uptake of the plant C_{Op2j} (Bq kg^{-1}) is:

$$C_{Op2j} = C_{GGj}\, T_{Plj}\, \exp(-\lambda_j\, t_{Op}) \tag{II-45}$$

Where:

C_{GGj} is the considered exposure period t_B time integrated concentration of radionuclide j in the soil (Bq kg^{-1}), see Eq. (II-32).

T_{Plj} is the element j related garden soil to plant activity transfer factor (represents concentrations ratio in Bq kg^{-1} of fresh matter of the plant to Bq kg^{-1} of dry matter of the soil). Reference values for garden soil to plant transfer factors for some elements are provided in Section II-4.

λ_j is the radionuclide j specific radioactive decay constant (s^{-1}).

t_{Op} is the time between harvesting and consumption of other plant products (s), see Table II-14.

The amount of other plant products (kg) consumed during considered exposure period t_B is:

$$M_{Op2} = U_{Op}\frac{t_B}{t_a} \tag{II-46}$$

The effective dose to the human due to ingestion of the root uptake contaminated other plant products E_{Op2} (Sv) is:

$$E_{Op2j} = e(g)_{ingj}\, C_{Op2j}\, M_{Op2}$$

(II-47)

II-3.4.4.3 Total dose due to consumption of other plant products

Total dose due to consumption of other plant products is:

$$E_{Opj} = E_{Op1j} + E_{Op2j}$$

(II-48)

The total effective dose due to consumption of other plant products of all radionuclides concerned is sum of the radionuclide specific effective doses:

$$E_{Op} = \sum_{j} E_{Opj}$$

(II-49)

II-3.4.5. Exposure due to consumption of milk

Similarly to the green vegetable exposure path, it is considered that contaminated forage from the pasture is consumed by milk producing animals directly in the pasture. Forage preparation, storage and contamination reduction due to later on forage consumption (e.g. in winter time) is not considered. This gives conservative dose evaluation for exposure periods exceeding duration of harvest growing season.

The long-term exposure effects are considered as follows. The grass direct contamination and activity admission through aerial parts of the plants is accounted for up to one harvest mass (e.g. maximal exposure time is used t_K = 1 year). However contamination of the pasture soil remains. No other radionuclide removal from the soil mechanisms excepting radioactive decay is considered. Thus activity admission through the grass roots remains and contributes the long-term exposure.

II-3.4.5.1 Activity admission through aerial parts of pasture grass

Considered exposure period t_K integrated activity concentration of radionuclide j in the fodder (i.e. pasture grass) due to activity admission through areal parts of plant C_{Pa1j} (Bq kg^{-1}) is calculated:

$$C_{Pa1j} = \frac{C_{GAj}}{Y_{Pa}}$$

(II-50)

Where:

C_{GAj} is the considered exposure period t_K time integrated activity concentration of radionuclide j on aerial parts of the plant (Bq m^{-2}), see II-3.4-1.

Y_{Pa} is the yield (fresh weight) of the pasture grass (kg m^{-2}). The reference value is used Y_{Pa} = 0.85 (kg m^{-2}), see Table II-14.

Corresponding concentration of the radionuclide j in the milk C_{Mi1} (Bq kg^{-1}) is:

$$C_{Mi1j} = C_{Pa1j}\, I_{Mi}\, T_{Mij}$$

(II-51)

Where:

I_{Mi} is the daily fodder (fresh weight) consumption by milk producing animal (kg d^{-1}). Reference value for large cattle is used I_{Mi} = 65 (kg d^{-1}).

T_{Mj} is the element specific fresh grass to milk transfer factor (d kg^{-1}). Reference values for some elements are provided in Section II-4. Element specific transfer factors for terrestrial foods are also discussed and values can be find in [II-2], [II-3].

The amount of milk M_{Mi1} (kg) consumed during considered exposure period t_K (s) is:

$$M_{Mi1} = U_{Mi}\frac{t_K}{t_a}$$

(II-52)

Where:

U_{Mi} is the annual consumption of milk (kg), see Table II-13. Other parameters are defined above.

Effective dose E_{Mi1} (Sv) to human due to consumption of milk which is contaminated by admission of radionuclide j through the aerial parts of pasture grass is:

$$E_{Mi1j} = e(g)_{ingj}\, C_{Mi1j}\, M_{Mi1}$$

(II-53)

Where:

$e(g)_{ingj}$ is the committed effective dose per unit intake by ingestion for radionuclide j by the group of age g (Sv Bq^{-1}) [II-7]. Reference values are explained and provided in Section II-4. Other parameters are defined above.

II-3.4.5.2 Activity admission through root uptake of pasture grass

Considered exposure period t_B time integrated concentration of the radionuclide j in the fresh matter of fodder (i.e. pasture grass) due to uptake from the soil C_{Pa2j} (Bq kg^{-1}) is:

$$C_{Pa2j} = C_{GPj}\, T_{Foj}$$

(II-54)

Where:

C_{GPj} is the mass concentration of the radionuclide j in the pasture soil (Bq kg^{-1}), see Eq. (II-33).

T_{Foj} is the element j pasture soil to forage plant transfer factor (represents concentrations ratio Bq kg^{-1} in wet matter of the plant per Bq kg^{-1} of dry matter of the soil). Reference values for pasture soil to forage plant transfer factors for some elements are provided in Section II-4. Element specific transfer factors for terrestrial foods are also discussed and values can be find in References [II-2] and [II-3].

Corresponding concentration of the radionuclide j in the milk C_{Mi2} (Bq kg^{-1}) is:

$$C_{Mi2j} = C_{Pa2j}\, I_{Mi}\, T_{Mij} \tag{II-55}$$

Amount of milk M_{Mi2} (kg) consumed during considered exposure period t_B (s) is:

$$M_{Mi2} = U_{Mi} \frac{t_B}{t_a} \tag{II-56}$$

Effective dose E_{Mi2} (Sv) to human due to consumption of milk which is contaminated by admission of radionuclide j through the root uptake of pasture grass is:

$$E_{Mi2j} = e(g)_{ingj}\, C_{Mi2j}\, M_{Mi2} \tag{II-57}$$

II-3.4.5.3 Total dose due to consumption of milk

Total dose due to consumption of milk is:

$$E_{Mij} = E_{Mi1j} + E_{Mi2j} \tag{II-58}$$

The total effective dose due to consumption of milk of all radionuclides concerned is sum of the radionuclide specific effective doses:

$$E_{Mi} = \sum_j E_{Mij} \tag{II-59}$$

II-3.4.6. Exposure due to consumption of meat

The exposure evaluation method is similar to the milk path exposure evaluation method with difference in consideration that the meat is consumed after certain delay following slaughtering of meat producing animal.

II-3.4.6.1 Activity admission through areal parts of pasture grass

Concentration of radionuclide j in the fodder due to activity admission through areal parts of pasture grass is calculated by Eq. (II-50). Concentration of the radionuclide j in the meat (Bq kg^{-1}) at the time of its consumption is calculated with accounting for radioactive decay due to time span from slaughtering of animal till meat consumption:

$$C_{Fl1j} = C_{Pa1j}\, I_{Fl}\, T_{Flj}\, \exp(-\lambda_j\, t_{Fl})$$

(II-60)

Where:

C_{Pa1j} is the considered exposure period t_K integrated activity concentration of radionuclide j in the fodder due to activity admission through areal parts of pasture grass (Bq kg^{-1}), see II-3.4-23.

I_{Fl} is the daily fodder (fresh weight) consumption by meat producing animal (kg d^{-1}). Reference value for large cattle is used I_F = 65 (kg d^{-1}).

T_{Flj} is the element specific fresh grass to meat transfer factor (d kg^{-1}). Reference values are provided in Section II-4. Element specific transfer factors for terrestrial foods are also discussed and values can be find in References [II-2] and [II-3].

t_{Fl} is the time between slaughtering and the consumption of the meat. Reference value is 1.7E+06 (s) which corresponds to 20 days delay time.

The amount of meat M_{Fl1} (kg) consumed during considered exposure period t_K (s) is:

$$M_{Fl1} = U_{Fl} \frac{t_K}{t_a}$$

(II-61)

Where:

U_{Fl} is the annual consumption of meat (kg), see Table II-13. Other parameters are defined above.

Effective dose E_{Mi1} (Sv) to human due to consumption of meat which is contaminated by admission of radionuclide j through the aerial parts of pasture grass is:

$$E_{Fl1j} = e(g)_{ingj} C_{Fl1j} M_{Fl1}$$

(II-62)

II-3.4.6.2 Activity admission through root uptake of pasture grass

Concentration of radionuclide j in the fodder due to activity admission through root uptake of pasture grass is calculated by Eq. (II-54). Concentration of the radionuclide j in the meat (Bq kg^{-1}) at the time of its consumption is calculated with accounting for radioactive decay due to time span from slaughtering of animal till meat consumption:

$$C_{Fl2j} = C_{Pa2j}\, I_{Fl}\, T_{Flj}\, \exp(-\lambda_j\, t_{Fl})$$

(II-63)

Where:

C_{Pa2j} is considered exposure period t_B time integrated concentration of the radionuclide j in the (fresh matter of) pasture grass due to uptake from the soil (Bq kg^{-1}), Eq. (II-54). Other parameters are defined above.

Amount of meat M_{Fl2} (kg) consumed during considered exposure period t_B (s) is:

$$M_{Fl2} = U_{Fl} \frac{t_B}{t_a}$$

(II-64)

Effective dose E_{Fl2} (Sv) to human due to consumption of meat which is contaminated by admission of radionuclide j through the root uptake of pasture grass is:

$$E_{Fl2j} = e(g)_{ingj}\, C_{Fl2j}\, M_{Fl2}$$

(II-65)

II-3.4.6.3 Total dose due to consumption of meat

Total dose due to consumption of meat is:

$$E_{Flj} = E_{Fl1j} + E_{Fl2j}$$

(II-66)

The total effective dose due to consumption of meat of all radionuclides concerned is sum of the radionuclide specific effective doses:

$$E_{Fl} = \sum_j E_{Flj}$$

(II-67)

II-3.4.7. Total ingestion dose

Total dose due to ingestion of radionuclide j is calculated as sum of individual doses arising from ingestion of foodstuff of all groups, i.e. green vegetables, other plant products, milk and meat:

$$E_{ingj} = E_{Gvj} + E_{Opj} + E_{Mij} + E_{Flj}$$

(II-68)

The total effective dose from ingestion of all radionuclides concerned is sum of the radionuclide specific effective doses:

$$E_{ing} = \sum_{j} E_{ingj}$$

(II-69)

II-3.5. Total Public Dose

In order to determine the total dose received by a human, the dose contributions of the relevant radionuclides via various exposure pathways have to be summed. Depending on assessment context and endpoints, particular exposure pathway related doses can be grouped in different way.

The particular radionuclide j stipulated effective dose E_j (Sv) from all external and internal exposure pathways is calculated:

$$E_j = E_{Subj} + E_{Gj} + E_{inhj} + E_{ingj}$$

(II-70)

Where:

E_{subj} is the radionuclide j specific effective dose from air submersion (Sv), see Section II-3.1;

E_{Gj} is the radionuclide j specific effective dose from the contaminated ground (Sv), see Section II-3.2;

E_{inhj} is the radionuclide j specific effective dose from inhalation (Sv), see Section II-3.3;

E_{ingj} is the radionuclide j specific effective dose from ingestion (Sv), see Section II-3.4.

The total effective dose E (Sv) of all radionuclides concerned is sum of the radionuclide specific effective doses:

$$E = \sum_{j} E_j$$

(II-71)

Particular radionuclide j stipulated effective dose due to submersion into passing through radioactive cloud E_{Cj} (Sv) is calculated as sum of doses from air submersion and inhalation:

$$E_{Cj} = E_{subj} + E_{inhj} \tag{II-72}$$

The total effective dose E_C (Sv) of all radionuclides concerned is sum of the radionuclide specific effective doses:

$$E_C = \sum_j E_{Cj} \tag{II-73}$$

Particular radionuclide j stipulated effective dose due to submersion into passing through radioactive cloud and from irradiation from radionuclides deposited onto the ground E_{CGj} (Sv) is calculated:

$$E_{CGj} = E_{subj} + E_{Gj} + E_{inhj} \tag{II-74}$$

The total effective dose E_C (Sv) of all radionuclides concerned is sum of the radionuclide specific effective doses:

$$E_{CG} = \sum_j E_{CGj} \tag{II-75}$$

II-4. DOSE CONVERSION AND BIOSPHERE TRANSFER FACTORS

II-4.1. Dose factors for inhalation and ingestion

As default, dose calculation model incorporate GSR Part 3 [II-7] provided values of committed effective dose per unit intake via inhalation $e(g)_{inh}$ (Sv Bq^{-1}) and via ingestion $e(g)_{ing}$ (Sv Bq^{-1}) for members of the public.

Dose values for intake via inhalation are selected from Table II-VII following the lung absorption type as per Table II-VIII for particulate aerosol when no specific information is available. Otherwise (i.e. when the lung absorption type is specified differently) the most conservative $e(g)_{inh}$ value is selected.

Dose values for intake via ingestion are selected from Table II-VI. Where several values available (depending on physical form or gut transfer factor), the most conservative $e(g)_{ing}$ value is selected.

The model included reference values for committed effective dose per unit intake via inhalation and via ingestion are provided in Table II-16.

TABLE II-16. REFERENCE VALUES OF COMMITTED EFFECTIVE DOSE PER UNIT INTAKE VIA INHALATION $e(g)_{inh}$ AND VIA INGESTION $e(g)_{ing}$ FOR MEMBERS OF THE PUBLIC

No	Radionuclide	Inhalation $e(g)_{inh}$, Sv Bq^{-1}		Ingestion $e(g)_{ing}$, Sv Bq^{-1}	
		Age 1-2 a	Age > 17 a	Age 1-2 a	Age > 17 a
1	Ac-228	1.6E-07	2.5E-08	2.8E-09	4.3E-10
2	Ag-110m	2.8E-08	7.6E-09	1.4E-08	2.8E-09
3	Am-241	6.9E-05	4.2E-05	3.7E-07	2.0E-07
4	As-76	4.6E-09	7.4E-10	1.1E-08	1.6E-09
5	At-211	3.7E-07	1.1E-07	7.8E-08	1.1E-08
6	Au-198	4.4E-09	8.6E-10	7.2E-09	1.0E-09
7	Bi-206	8.0E-09	1.7E-09	1.0E-08	1.9E-09
8	Bi-210	3.0E-07	9.3E-08	9.7E-09	1.3E-09
9	Bi-212	1.1E-07	3.1E-08	1.8E-09	2.6E-10
10	Br-82	3.0E-09	6.3E-10	2.6E-09	5.4E-10
11	C-14	6.6E-09	2.0E-09	1.6E-09	5.8E-10
12	Cd-109	3.7E-08	8.1E-09	9.5E-09	2.0E-09
13	Ce-141	1.1E-08	3.2E-09	5.1E-09	7.1E-10
14	Ce-144	1.6E-07	3.6E-08	3.9E-08	5.2E-09
15	Cm-242	1.8E-05	5.2E-06	7.6E-08	1.2E-08
16	Cm-244	5.7E-05	2.7E-05	2.9E-07	1.2E-07
17	Co-58	6.5E-09	1.6E-09	4.4E-09	7.4E-10
18	Co-60	3.4E-08	1.0E-08	2.7E-08	3.4E-09
19	Cr-51	2.1E-10	3.7E-11	2.3E-10	3.8E-11
20	Cs-134	7.3E-09	6.6E-09	1.6E-08	1.9E-08
21	Cs-135	9.9E-10	6.9E-10	2.3E-09	2.0E-09
22	Cs-136	5.2E-09	1.2E-09	9.5E-09	3.0E-09
23	Cs-137	5.4E-09	4.6E-09	1.2E-08	1.3E-08
24	Cu-64	5.7E-10	1.2E-10	8.3E-10	1.2E-10
25	Eu-154	1.5E-07	5.3E-08	1.2E-08	2.0E-09
26	Eu-155	2.3E-08	6.9E-09	2.2E-09	3.2E-10
27	Fe-55	1.4E-09	3.8E-10	2.4E-09	3.3E-10
28	Fe-59	1.3E-08	3.7E-09	1.3E-08	1.8E-09
29	Ga-67	1.0E-09	2.4E-10	1.2E-09	1.9E-10
30	H-3	2.7E-10	4.5E-11	1.2E-10	4.2E-11
31	Hg-197	1.2E-09	3.0E-10	1.6E-09	2.3E-10
32	Hg-197m	2.5E-09	5.3E-10	3.4E-09	4.7E-10
33	Hg-203	7.9E-09	2.4E-09	1.1E-08	1.9E-09
34	I-123	7.9E-10	7.4E-11	1.9E-09	2.1E-10
35	I-125	2.3E-08	5.1E-09	5.7E-08	1.5E-08
36	I-129	8.6E-08	3.6E-08	2.2E-07	1.1E-07

TABLE II-16. (Continued)

No	Radionuclide	Inhalation $e(g)_{inh}$, Sv Bq^{-1}		Ingestion $e(g)_{ing}$, Sv Bq^{-1}	
		Age 1-2 a	Age > 17 a	Age 1-2 a	Age > 17 a
37	I-131	7.2E-08	7.4E-09	1.8E-07	2.2E-08
38	I-132	9.6E-10	9.4E-11	2.4E-09	2.9E-10
39	I-133	1.8E-08	1.5E-09	4.4E-08	4.3E-09
40	I-134	3.7E-10	4.5E-11	7.5E-10	1.1E-10
41	I-135	3.7E-09	3.2E-10	8.9E-09	9.3E-10
42	In-111	1.2E-09	2.3E-10	1.7E-09	2.9E-10
43	In-113m	1.1E-10	2.0E-11	1.8E-10	2.8E-11
44	Mn-54	6.2E-09	1.5E-09	3.1E-09	7.1E-10
45	Mo-99	4.4E-09	8.9E-10	3.5E-09	6.0E-10
46	Na-22	7.3E-09	1.3E-09	1.5E-08	3.2E-09
47	Na-24	1.8E-09	2.7E-10	2.3E-09	4.3E-10
48	Nb-95	5.2E-09	1.5E-09	3.2E-09	5.8E-10
49	Ni-59	6.2E-10	1.3E-10	3.4E-10	6.3E-11
50	Ni-63	1.9E-09	4.8E-10	8.4E-10	1.5E-10
51	Np-237	4.0E-05	2.3E-05	2.1E-07	1.1E-07
52	Np-239	4.2E-09	9.3E-10	5.7E-09	8.0E-10
53	P-32	1.5E-08	3.4E-09	1.9E-08	2.4E-09
54	Pa-231	2.3E-04	1.4E-04	1.3E-06	7.1E-07
55	Pa-233	1.3E-08	3.9E-09	6.2E-09	8.7E-10
56	Pb-210	3.7E-06	1.1E-06	3.6E-06	6.9E-07
57	Pd-103	1.8E-09	4.5E-10	1.4E-09	1.9E-10
58	Pd-107	2.0E-09	5.9E-10	2.8E-10	3.7E-11
59	Pd-109	1.9E-09	3.7E-10	4.1E-09	5.5E-10
60	Pm-147	1.8E-08	5.0E-09	1.9E-09	2.6E-10
61	Po-210	1.1E-05	3.3E-06	8.8E-06	1.2E-06
62	Pu-238	7.4E-05	4.6E-05	4.0E-07	2.3E-07
63	Pu-239	7.7E-05	5.0E-05	4.2E-07	2.5E-07
64	Pu-240	7.7E-05	5.0E-05	4.2E-07	2.5E-07
65	Pu-241	9.7E-07	9.0E-07	5.7E-09	4.8E-09
66	Pu-242	7.3E-05	4.8E-05	4.0E-07	2.4E-07
67	Ra-224	8.2E-06	3.0E-06	6.6E-07	6.5E-08
68	Ra-225	1.8E-05	6.3E-06	1.2E-07	9.9E-08
69	Ra-226	1.1E-05	3.5E-06	9.6E-07	2.8E-07
70	Rb-86	7.7E-09	9.3E-10	2.0E-08	2.8E-09
71	Rh-105	1.7E-09	3.5E-10	2.7E-09	3.7E-10
72	Rh-107	9.7E-11	1.7E-11	1.6E-10	2.4E-11
73	Ru-103	8.4E-09	2.4E-09	4.6E-09	7.3E-10
74	Ru-106	1.1E-07	2.8E-08	4.9E-08	7.0E-09
75	S-35	4.5E-09	1.4E-09	5.4E-09	7.7E-10
76	Sb-124	2.4E-08	6.4E-09	1.6E-08	2.5E-09
77	Sb-125	1.6E-08	4.8E-09	6.1E-09	1.1E-09

TABLE II-16. (Continued)

No	Radionuclide	Inhalation $e(g)_{inh}$, Sv Bq^{-1}		Ingestion $e(g)_{ing}$, Sv Bq^{-1}	
		Age 1-2 a	Age > 17 a	Age 1-2 a	Age > 17 a
78	Se-75	6.0E-09	1.0E-09	1.3E-08	2.6E-09
79	Sn-113	1.0E-08	2.7E-09	5.0E-09	7.3E-10
80	Sr-85	3.1E-09	6.4E-10	3.1E-09	5.6E-10
81	Sr-87m	1.2E-10	2.0E-11	1.7E-10	3.0E-11
82	Sr-89	2.4E-08	6.1E-09	1.8E-08	2.6E-09
83	Sr-90	1.1E-07	3.6E-08	7.3E-08	2.8E-08
84	Tc-99	1.3E-08	4.0E-09	4.8E-09	6.4E-10
85	Tc-99m	9.9E-11	1.9E-11	1.3E-10	2.2E-11
86	Te-125m	1.1E-08	3.4E-09	6.3E-09	8.7E-10
87	Te-127m	2.6E-08	7.4E-09	1.8E-08	2.3E-09
88	Te-129m	2.6E-08	6.6E-09	2.4E-08	3.0E-09
89	Te-131m	5.8E-09	9.4E-10	1.4E-08	1.9E-09
90	Te-132	1.3E-08	2.0E-09	3.0E-08	3.8E-09
91	Th-228	1.3E-04	4.0E-05	3.7E-07	7.2E-08
92	Th-230	3.5E-05	1.4E-05	4.1E-07	2.1E-07
93	Th-232	5.0E-05	2.5E-05	4.5E-07	2.3E-07
94	Tl-201	3.3E-10	4.4E-11	5.5E-10	9.5E-11
95	Tl-202	1.2E-09	1.9E-10	2.1E-09	4.5E-10
96	U-232	2.4E-05	7.8E-06	8.2E-07	3.3E-07
97	U-234	1.1E-05	3.5E-06	1.3E-07	4.9E-08
98	U-235	1.0E-05	3.1E-06	1.3E-07	4.7E-08
99	U-238	9.4E-06	2.9E-06	1.2E-07	4.5E-08
100	Y-87	2.2E-09	3.9E-10	3.2E-09	5.5E-10
101	Y-90	8.8E-09	1.5E-09	2.0E-08	2.7E-09
102	Y-91	3.4E-08	8.9E-09	1.8E-08	2.4E-09
103	Zn-65	6.5E-09	1.6E-09	1.6E-18	3.9E-09
104	Zr-95	1.6E-08	4.8E-09	5.6E-09	9.5E-10

II-4.2. Dose factors for external exposure

Following the approach used in the IAEA Safety Report Series No. 19 [II-3], as default, the dose calculation model incorporate the US Environmental Protection Agency Federal Guidance Report No. 12 [II-5] provided values of effective dose per time integrated activity concentration in the air e_{sub} (Sv per Bq s m^{-3}) and of effective dose per time integrated activity concentration on the ground surface e_G (Sv per Bq s m^{-2}).

The publication provides values for effective equivalent dose and equivalent skin dose separately. The values of effective dose are calculated as follows:

$$e = h_E + 0.01\, h_T \qquad \text{(II-76)}$$

Where:

h_E is the effective equivalent dose factor for air submersion or ground contamination (dose to skin is not included into summation);

h_T is the skin equivalent dose factor for air submersion or ground contamination.

The publication [II-5] provided dose factors do not include consideration of radiations emitted by radioactive decay products. Following the approach used in the IAEA Safety Report Series No. 19 [II-3], e_{sub} and e_G values for some of radionuclides are additionally upgraded to account for impact arising from their progenies. These radionuclides and considered progenies are listed in Table 17. The model included reference values of effective dose per time integrated activity concentration in the air and of effective dose per time integrated activity concentration on the ground surface are provided in Table II-18.

TABLE II-17. RADIONUCLIDES AND THEIR PROGENIES WHICH EFFECTS TO EFFECTIVE DOSE ARE INCLUDED INTO REFERENCE DOSE FACTORS FOR AIR SUBMERSION e_{sub} AND GROUND CONTAMINATION e_g

Radionuclide	Progenies considered for e_{sub} *)	Progenies considered for e_G *)
At-211		Po-211 (0.5830)
Bi-210		Po-210
Bi-212	Tl-208 (0.3593), Po-212 (0.6407)	Tl-208 (0.3593), Po-212 (0.6407)
Ce-144	Pr-144 (0.9857)	Pr-144 (0.9857)
Cs-137	Ba-137m (0.9457)	Ba-137m (0.9457)
Mo-99	Tc-99m (0.8860), Tc-99	Tc-99m (0.8860), Tc-99
Np-237		Pa-233
Pb-210		Bi-210
Pd-103		Rh-103m
Ra-224		Rn-220, Po-216, Pb-212, Bi-212, Po-212 (0.6407), Tl-208 (0.3593)
Ra-225		Ac-225, Fr-221, At-217, Bi-213, Po-213 (0.978), Tl-209 (0.0216)
Ra-226		Rn-222, Po-218, Pb-214, Bi-214, Po-214, Pb-210, Bi-210, Po-210
Ru-106	Rh-106	Rh-106
Sb-125		Te-125m (0.231)
Sn-113		In-113m
Sr-90		Y-90
Te-127m		Te-127
Te-131m	Te-131 (0.2220), I-131 (0.7780)	Te-131 (0.222), I-131 (0.778)
Te-132		I-132
Th-228		Ra-224, Rn-220, Po-216, Pb-212, Bi-212, Po-212 (0.6407), Tl-208 (0.3593)
Th-230		Ra-226, Rn-222, Po-218, Pb-214, Bi-214, Po-214, Pb-210, Bi-210, Po-210

TABLE II-17. (Continued)

Radionuclide	Progenies considered for e_{sub} *)	Progenies considered for e_G *)
Th-232		Ra-228, Ac-228, Th-228, Ra-224, Rn-220, Po-216, Pb-212, Bi-212, Po-212 (0.6407), Tl-208 (0.3593)
U-232		Th-228, Ra-224, Rn-220, Po-216, Pb-212, Bi-212, Po-212 (0.6407), Tl-208 (0.3593)
U-234		Th-230, Ra-226, Rn-222, Po-218, Pb-214, Bi-214, Po-214, Pb-210, Bi-210, Po-210
U-235		Th-231
U-238		Th-234, Pa-234m, U-234, Th-230, Ra-226, Rn-222, Po-218, Pb-214, Bi-214, Po-214, Pb-210, Bi-210, Po-210
Y-87		Sr-87m
Zr-95		Nb-95

* Where relevant, progeny branching ratios are shown in brackets

TABLE II-18. REFERENCE VALUES OF EFFECTIVE DOSE PER TIME INTEGRATED ACTIVITY CONCENTRATION IN THE AIR e_{sub} AND OF EFFECTIVE DOSE PER TIME INTEGRATED ACTIVITY CONCENTRATION ON THE GROUND SURFACE e_g

No	Radionuclide	e_{air}, Sv per Bq s m^{-3}	e_G, Sv per Bq s m^{-2}
1	Ac-228	4.9E-14	9.8E-16
2	Ag-110m	1.4E-13	2.7E-15
3	Am-241	8.3E-16	2.8E-17
4	As-76	2.2E-14	5.4E-16
5	At-211	1.6E-15	4.1E-17
6	Au-198	2.0E-14	4.3E-16
7	Bi-206	1.6E-13	3.2E-15
8	Bi-210	2.6E-16	3.5E-17
9	Bi-212	7.4E-14	1.3E-15
10	Br-82	1.3E-13	2.6E-15
11	C-14	2.7E-18	1.7E-20
12	Cd-109	3.0E-16	2.3E-17
13	Ce-141	3.5E-15	7.5E-17
14	Ce-144	3.6E-15	1.8E-16
15	Cm-242	6.1E-18	1.0E-18
16	Cm-244	5.3E-18	9.7E-19
17	Co-58	4.8E-14	9.6E-16
18	Co-60	1.3E-13	2.4E-15
19	Cr-51	1.5E-15	3.1E-17
20	Cs-134	7.7E-14	1.6E-15

TABLE II-18. (Continued)

No	Radionuclide	e_{air}, Sv per Bq s m^{-3}	e_G, Sv per Bq s m^{-2}
21	Cs-135	9.6E-18	3.5E-20
22	Cs-136	1.1E-13	2.1E-15
23	Cs-137	2.8E-14	5.7E-16
24	Cu-64	9.3E-15	1.9E-16
25	Eu-154	6.2E-14	1.2E-15
26	Eu-155	2.5E-15	6.0E-17
27	Fe-55	0	0
28	Fe-59	6.0E-14	1.1E-15
29	Ga-67	7.3E-15	1.5E-16
30	H-3	3.3E-19	0
31	Hg-197	2.7E-15	6.5E-17
32	Hg-197m	4.2E-15	8.8E-17
33	Hg-203	1.1E-14	2.3E-16
34	I-123	7.4E-15	1.7E-16
35	I-125	5.4E-16	4.4E-17
36	I-129	3.9E-16	2.6E-17
37	I-131	1.8E-14	3.8E-16
38	I-132	1.1E-13	2.3E-15
39	I-133	3.0E-14	6.4E-16
40	I-134	1.3E-13	2.6E-15
41	I-135	8.1E-14	1.5E-15
42	In-111	1.9E-14	4.0E-16
43	In-113m	1.2E-14	2.6E-16
44	Mn-54	4.1E-14	8.2E-16
45	Mo-99	1.3E-14	2.9E-16
46	Na-22	1.1E-13	2.1E-15
47	Na-24	2.2E-13	3.7E-15
48	Nb-95	3.8E-14	7.6E-16
49	Ni-59	0	0
50	Ni-63	0	0
51	Np-237	1.0E-15	2.3E-16
52	Np-239	7.9E-15	1.7E-16
53	P-32	5.5E-16	8.6E-17
54	Pa-231	1.7E-15	4.2E-17
55	Pa-233	9.5E-15	2.0E-16
56	Pb-210	5.8E-17	3.8E-17
57	Pd-103	8.1E-17	1.3E-17
58	Pd-107	0	0
59	Pd-109	4.7E-16	4.0E-17
60	Pm-147	8.8E-18	3.5E-20
61	Po-210	4.2E-19	8.4E-21

TABLE II-18. (Continued)

No	Radionuclide	e_{air}, Sv per Bq s m^{-3}	e_G, Sv per Bq s m^{-2}
62	Pu-238	5.3E-18	9.3E-19
63	Pu-239	4.4E-18	4.0E-19
64	Pu-240	5.1E-18	8.9E-19
65	Pu-241	7.4E-20	2.0E-21
66	Pu-242	4.3E-18	7.4E-19
67	Ra-224	4.8E-16	1.5E-15
68	Ra-225	3.1E-16	2.8E-16
69	Ra-226	3.2E-16	1.8E-15
70	Rb-86	5.3E-15	1.7E-16
71	Rh-105	3.8E-15	7.8E-17
72	Rh-107	1.5E-14	3.5E-16
73	Ru-103	2.3E-14	4.7E-16
74	Ru-106	1.1E-14	3.5E-16
75	S-35	3.2E-18	1.8E-20
76	Sb-124	9.3E-14	1.8E-15
77	Sb-125	2.0E-14	4.4E-16
78	Se-75	1.9E-14	3.8E-16
79	Sn-113	3.9E-16	2.8E-16
80	Sr-85	2.4E-14	5.1E-16
81	Sr-87m	1.5E-14	3.2E-16
82	Sr-89	4.5E-16	6.9E-17
83	Sr-90	1.0E-16	1.1E-16
84	Tc-99	2.9E-17	8.0E-20
85	Tc-99m	6.0E-15	1.2E-16
86	Te-125m	4.7E-16	3.7E-17
87	Te-127m	1.6E-16	2.2E-17
88	Te-129m	1.7E-15	6.1E-17
89	Te-131m	7.9E-14	1.6E-15
90	Te-132	1.0E-14	2.5E-15
91	Th-228	9.4E-17	1.5E-15
92	Th-230	1.8E-17	1.8E-15
93	Th-232	9.1E-18	2.5E-15
94	Tl-201	3.8E-15	8.8E-17
95	Tl-202	2.2E-14	4.6E-16
96	U-232	1.5E-17	1.5E-15
97	U-234	8.1E-18	1.8E-15
98	U-235	7.3E-15	7.7E-16
99	U-238	3.7E-18	1.9E-15
100	Y-87	2.2E-14	7.7E-16
101	Y-90	8.1E-16	1.1E-16
102	Y-91	6.5E-16	7.5E-17

TABLE II-18. (Continued)

No	Radionuclide	e_{air}, Sv per Bq s m^{-3}	e_G, Sv per Bq s m^{-2}
103	Zn-65	2.9E-14	5.6E-16
104	Zr-95	3.6E-14	1.5E-15

II-4.3. Transfer factors for terrestrial foods

Reference values of the element specific transfer factors for terrestrial foods are taken from [II-4].Values are provided in Table II-19. Transfer factors for terrestrial foods are also discussed and values can be find in [II-2] and [II-3].

TABLE II-19. ELEMENT SPECIFIC TRANSFER FACTORS FOR TERRESTRIAL FOODS

Element	T_{Fo} Forage *)	T_{Pl} Plant **)	T_{Mi} Milk d Kg^{-1}	T_{Fl} Meat d Kg^{-1}
Ac	3.0E-03	3.0E-03	2.0E-05	3.0E-03
Ag	2.0E-01	2.0E-01	5.0E-02	2.0E-03
Am	3.0E-04	3.0E-04	2.0E-05	5.0E-04
As	6.0E-03	2.0E-03	7.0E-05	2.0E-03
At	3.0E-01	3.0E-01	5.0E-02	5.0E-01
Au	3.0E-03	3.0E-03	6.0E-06	3.0E-03
Bi	2.0E-01	2.0E-01	5.0E-04	2.0E-02
Br	1.0E-01	3.0E-01	5.0E-02	3.0E-02
C ***)	0	0	0	0
Cd	4.0E-01	4.0E-01	1.0E-03	4.0E-04
Ce	9.0E-03	9.0E-03	2.0E-05	2.0E-03
Cm	3.0E-04	3.0E-04	2.0E-05	2.0E-04
Co	2.0E-02	2.0E-02	2.0E-04	1.0E-02
Cr	1.0E-02	4.0E-03	3.0E-03	1.0E-02
Cs	5.0E-02	5.0E-02	5.0E-03	3.0E-02
Cu	2.0E-01	2.0E-01	2.0E-03	1.0E-02
Eu	3.0E-03	3.0E-03	2.0E-05	5.0E-03
Fe	5.0E-03	5.0E-03	3.0E-04	2.0E-02
Ga	3.0E-04	3.0E-04	5.0E-05	5.0E-01
H ***)	0	0	0	0
Hg	7.0E-02	2.0E-01	1.0E-05	3.0E-01
I	1.0E-01	2.0E-02	3.0E-03	1.0E-02
In	3.0E-01	3.0E-01	1.0E-04	8.0E-03
Mn	2.0E-01	2.0E-01	3.0E-04	5.0E-04
Mo	2.0E-01	5.0E-02	2.0E-03	7.0E-03
Na	4.0E-01	4.0E-01	4.0E-02	8.0E-02

TABLE II-19. (Continued)

Element	T_{Fo} Forage *)	T_{Pl} Plant **)	T_{Mi} Milk d Kg^{-1}	T_{Fl} Meat d Kg^{-1}
Nb	1.0E-02	1.0E-02	3.0E-03	3.0E-01
Ni	2.0E-02	2.0E-02	1.0E-02	2.0E-03
Np	2.0E-02	2.0E-02	5.0E-06	2.0E-04
P	5.0E-01	3.0E+00	3.0E-02	6.0E-02
Pa	3.0E-03	3.0E-03	5.0E-06	5.0E-03
Pb	1.0E-02	7.0E-03	3.0E-04	4.0E-04
Pd	2.0E-02	2.0E-02	1.0E-02	4.0E-03
Pm	3.0E-03	3.0E-03	2.0E-05	5.0E-03
Po	1.0E-02	5.0E-03	3.0E-04	5.0E-03
Pu	8.0E-05	4.0E-04	1.0E-07	3.0E-04
Ra	1.0E-02	5.0E-03	3.0E-03	9.0E-04
Rb	9.0E-01	9.0E-02	6.0E-03	1.0E-02
Rh	2.0E-02	2.0E-02	1.0E-02	2.0E-03
Ru	1.0E-02	1.0E-02	1.0E-06	2.0E-03
S	9.0E-01	9.0E-01	2.0E-02	1.0E-01
Sb	1.0E-01	2.0E-02	2.0E-03	1.0E-03
Se	5.0E-01	5.0E-01	5.0E-02	2.0E-02
Sn	2.0E-01	2.0E-01	3.0E-03	8.0E-02
Sr	4.0E-01	4.0E-01	2.0E-03	6.0E-04
Tc	3.0E+00	3.0E+00	1.0E-05	4.0E-02
Te	2.0E+00	2.0E+00	2.0E-04	8.0E-02
Th	2.0E-03	5.0E-04	5.0E-06	2.0E-04
Tl	3.0E-01	3.0E-01	2.0E-03	4.0E-02
U	3.0E-03	3.0E-03	5.0E-04	4.0E-04
Y	3.0E-03	3.0E-03	1.0E-05	1.0E-03
Zn	3.0E-01	3.0E-01	1.0E-02	1.0E-01
Zr	1.0E-03	3.0E-03	5.0E-06	2.0E-02

*) Represents concentrations ratio in Bq kg^{-1} of dry matter of the forage to Bq kg^{-1} of dry matter of the soil.
**) Represents concentrations ratio in Bq kg^{-1} of fresh matter of the plant to Bq kg^{-1} of dry matter of the soil.
***) No default values are specified in the model. To calculate intake of C and H through terrestrial food chain and corresponding ingestion doses, user shall specify relevant values for the transfer factors.

II-5. EMPLOYMENT OF THE MODEL (USER'S GUIDE)

II-5.1. Input parameters and values

The model includes about 70 parameters with preset (i.e. reference) values. Values of parameter can be changed seeking to adapt model to the situation being modelled. However not all of parameters have to be changed frequently.

To ease practical use of the model, parameters are grouped into several categories, which are available within Ecolego tool. Grouping is detailed in tables below. Additional scope of these tables is to give understanding how parameters in Ecolego tool correspond to the same parameters of description of mathematical model.

Most frequently changes are expected in parameters grouped into category 'scenario', Table II-20. These parameters define principal set of accident conditions like content and amount of released radionuclides, emission height, human exposure duration and basic meteorological parameters.

If the influence of buildings to dispersion has to be considered, the user has to change parameters of category 'building', Table II-21. They are 4 in total. Reference values for building height and width are set to be close to 0, thus no influence from the buildings is assumed as default.

It could be said that categories 'scenario'and 'building' include parameters which are dependent on the design of facility, safety elements and accident parameters.

Parameters of category 'human', Table II-22 and of category 'biosphere', Table II-23 describes environment around the facility. It could be said that these parameters are facility site related and accounts for local specific.

Parameters of category 'model', Table II-24, includes either general model parameters or biosphere parameters that are expected to be common on country or region level. Once preset, they are not expecting to be changed frequently. Reference values could be suitable for most of central and northern European countries.

Not categorized parameters, Table II-25, are not to be changed. There are two of them – amount of seconds per year and radionuclides decay half-lives.

TABLE II-20. PARAMETERS OF CATEGORY 'SCENARIO'

No	Sub-system	Parameter	Type	Dimension	Unit	Equation	Symbol	Reference value(s)
1		InActivityRelease	Vector	Materials	Bq	II-8	Q	1.0
2		InDistance	Vector	Locations	m	II-1	x	300
3	Diffusion	InHeightRelease	Scalar	-	m	II-1	H_e	30
4	Air	InPrecipitation	Vector	Stability class	mm/h	II-14	I	Table II-9
5		InReductionConcAir	Vector	Stability class	unitless	II-10	f_X	1.0
6	Inhalation	InRespirableFraction	Scalar	-	unitless	II-26	RF	1.0
7	GroundShine	InTimeExpoGround	Scalar	-	s	II-23	t_G	3.16E+07
8		InTimeExpoIngest	Scalar	-	s	II-30	t_B	3.16E+07
9	GroundShine	InTimeFraExpoGround	Scalar	-	unitless	II-23	f_G	1.0
10	Diffusion	InWindSpeed10	Vector	Stability class	m/s	II-2	u_1	Table II-4

TABLE II-21. PARAMETERS OF CATEGORY 'BUILDING'

No	Sub-system	Parameter	Type	Dimension	Unit	Equation	Symbol	Reference value(s)
1	Diffusion	InBuildingCaseDownwind	Scalar	-	unitless	-	-	1.0
2	Diffusion	InBuildingHeight	Scalar	-	m	II-5	H_G	0.1
3	Diffusion	InBuildingSourceDistance	Scalar	-	m	-	-	0
4	Diffusion	InBuildingWidth	Scalar	-	m	-	b_G	0.1

TABLE II-22. PARAMETERS OF CATEGORY ‘HUMAN’

No	Sub-system	Parameter	Type	Dimension	Unit	Equation	Symbol	Reference value(s)
1	Inhalation	BreathAdult	Scalar	-	m^3/s	II-26	B	3.8E-04
2	Inhalation	BreathChild1	Scalar	-	m^3/s	II-26	B	8.7E-05
3	Vegetables	InConsumpGreenVegAdult	Scalar	-	kg/yr	II-35	U_{Gv}	39
4	Vegetables	InConsumpGreenVegChild1	Scalar	-	kg/yr	II-35	U_{Gv}	18
5	Meat	InConsumpMeatAdult	Scalar	-	kg/yr	II-61	U_{Fl}	180
6	Meat	InConsumpMeatChild1	Scalar	-	kg/yr	II-61	U_{Fl}	26
7	Milk	InConsumpMilkAdult	Scalar	-	kg/yr	II-52	U_{Mi}	390
8	Milk	InConsumpMilkChild1	Scalar	-	kg/yr	II-52	U_{Mi}	480
9	Plants	InConsumpOthPlantAdult	Scalar	-	kg/yr	II-43	U_{Op}	610
10	Plants	InConsumpOthPlantChild1	Scalar	-	kg/yr	II-43	U_{Op}	366
11	Meat	InTimeConsumpMeat	Scalar	-	s	II-60	t_{Fl}	1.7E+06
12	Plants	InTimeConsumpOthPlant	Scalar	-	s	II-42	t_{Op}	5.2E+06

TABLE II-23. PARAMETERS OF CATEGORY ‘BIOSPHERE’

No	Sub-system	Parameter	Type	Dimension	Unit	Equation	Symbol	Reference value(s)
1	Meat	InConsumpFodderMeatAnimal	Scalar	-	kg/d	II-60	I_{Fl}	65
2	Milk	InConsumpFodderMilkAnimal	Scalar	-	kg/d	II-51	I_{Mi}	65
3	Soil	InMassSoilGarden	Scalar	-	kg/m^2	II-32	P_G	280
4	Soil	InMassSoilPasture	Scalar	-	kg/m^2	II-33	P_P	120
5	Vegetables	InYieldGreenVeg	Scalar	-	kg/m^2	II-34	Y_{Gv}	1.6
6	Plants	InYieldOthPlant	Scalar	-	kg/m^2	II-42	Y_{Op}	2.4
7	Milk	InYieldPastGrass	Scalar	-	kg/m^2	II-50	Y_{Pa}	0.85

TABLE II-24. PARAMETERS OF CATEGORY ‘MODEL’

No	Sub-system	Parameter	Type	Dimension	Unit	Equation	Symbol	Reference value(s)
1	Diffusion	InWindm	Vector	Stability class	unitless	II-2	m	Table II-3
2	Diffusion	py100	Vector	Stability class	unitless	II-3	p_y	Table II-5
3	Diffusion	py180	Vector	Stability class	unitless	II-3	p_y	Table II-5
4	Diffusion	py50	Vector	Stability class	unitless	II-3	p_y	Table II-5
5	Diffusion	pz100	Vector	Stability class	unitless	II-3	p_z	Table II-5
6	Diffusion	pz180	Vector	Stability class	unitless	II-3	p_z	Table II-5
7	Diffusion	pz50	Vector	Stability class	unitless	II-3	p_z	Table II-5
8	Diffusion	qy100	Vector	Stability class	unitless	II-3	q_y	Table II-5
9	Diffusion	qy180	Vector	Stability class	unitless	II-3	q_y	Table II-5
10	Diffusion	qy50	Vector	Stability class	unitless	II-3	q_y	Table II-5
11	Diffusion	qz100	Vector	Stability class	unitless	II-3	q_z	Table II-5
12	Diffusion	qz180	Vector	Stability class	unitless	II-3	q_z	Table II-5
13	Diffusion	qz50	Vector	Stability class	unitless	II-3	q_z	Table II-5
14	Diffusion	z1	Scalar	-	m	II-2	z_1	10
15	Air	InDryDepositionVelocity	Vector	Materials	m/s	II-12	v_g	0.0015
16	Air	InWashoutCoeff0	Vector	Materials	s^{-1}	II-14	$_0$	7.0E-05
17	Air	InWashoutK	Scalar	-	unitless	II-14	K	0.8
18	Ground	WetFraPlant	Scalar	-	unitless	II-17	f_W	0.3
19	AirSubmersion	DCFAirSubm	Vector	Materials	(Sv/s)/ (Bq/m^3)	II-21	e_{sub}	Table II-18
20	GroundShine	DCFGroundShine	Vector	Materials	(Sv/s)/ (Bq/m^2)	II-23	e_G	Table II-18
21	GroundShine	InDecayConstGround	Vector	Elements	s^{-1}	II-24	λ_g	0
22	GroundShine	InSoilMigrFactor	Scalar	-	unitless	II-23	b_G	0.5
23	Inhalation	DCFInhalAdult	Vector	Materials	Sv/Bq	II-26	$e(g)_{inh}$	Table II-16
24	Inhalation	DCFInhalChild1	Vector	Materials	Sv/Bq	II-26	$e(g)_{inh}$	Table II-16

TABLE II-24. (Continued)

No	Sub-system	Parameter	Type	Dimension	Unit	Equation	Symbol	Reference value(s)
25	Soil	InDecayConstPlantA	Scalar	-	s^{-1}	II-29	λ_{ap}	5.7E-07
26	Soil	InDecayConstPlantRGarden	Vector	Elements	s^{-1}	II-31	λ_{rp}	TABLE II-15
27	Soil	InDecayConstPlantRPasture	Vector	Elements	s^{-1}	II-31	λ_{rp}	TABLE II-15
28		DCFIngestAdult	Vector	Materials	Sv/Bq	II-36	$e(g)_{ing}$	Table II-16
29		DCFIngestChild1	Vector	Materials	Sv/Bq	II-36	$e(g)_{ing}$	Table II-16
30		TransferFactorForrageMeat	Vector	Elements	d/kg	II-60	T_{Fl}	Table II-19
31		TransferFactorForrageMilk	Vector	Elements	d/kg	II-51	T_{Mi}	Table II-19
32		TransferFactorSoilForrage	Vector	Elements	unitless	II-54	T_{Fo}	Table II-19
33		TransferFactorSoilPlant	Vector	Elements	unitless	II-37	T_{Pl}	Table II-19

TABLE II-25. NOT CATEGORIZED PARAMETERS

No	Sub-system	Parameter	Type	Dimension	Unit	Equation	Symbol	Reference value(s)
1		SecondsPerYear	Scalar	-	s		t_a	3.16E+07
2		RadDecayHalfLife	Vector	Materials	year		$T_{0.5}$	

II-5.2. Model output

Model calculates all equations as described in Sections I-2 and I-3, and user can check any results as needed. The Ecolego tool main outputs which usually could be used to summarize radiological impact assessment results and their correspondence to parameters of mathematical models are provided in Table II-26.

II-5.3. Limitations to use of the model

In theory Gaussian plume model is applicable for short range transport starting from about 100 m and do not exceeding about 20 km distance in downwind direction. Additional explanations on Gaussian plume model and its applicability, including references to the more detailed sources, can be found, for example in Annex V of [II-3].

TABLE II-26. MAIN ECOLEGO OUTPUTS USED TO SUMMARIZE RADIOLOGICAL IMPACT ASSESSMENT RESULTS

No	Sub-system	Block	Description	Equation
1	Air	ConcAirTimeCloud	Time integrated ground level radionuclide concentrations in the air at the ‘cloud’ exposure location (Bq s m^{-3}) for A, B, C, D, E, F atmospheric stability categories.	II-11
2	Ground	ConcGround0Cloud	Concentrations of radionuclides deposited onto the ground at the ‘cloud’ exposure location (Bq m^{-2}) for A, B, C, D, E, F atmospheric stability categories.	II-16
3	Ground	ConcGround0CloudSumM	Total (sum from all radionuclides) concentrations of radionuclides deposited onto the ground at the ‘cloud’ exposure location (Bq m^{-2}) for A, B, C, D, E, F atmospheric stability categories.	II-18
4	Dose	DoseAllPathChild1 DoseAllPathAdult	Radionuclide specific effective doses from all external and internal exposure paths for child and for adult respectively (Sv), for A, B, C, D, E, F atmospheric stability categories.	II-70
5	Dose	DoseAllPathChild1MaxS DoseAllPathAdultMaxS	Radionuclide specific effective doses from all external and internal exposure paths for child and for adult respectively (Sv), for atmospheric stability category which results in maximal total dose (i.e. dose for the most unfavourable atmospheric stability category).	II-70
6	Dose	DoseAllPathChild1SumM DoseAllPathAdultSumM	Total (sum from all radionuclides) effective doses from all external and internal exposure paths for child and for adult respectively (Sv), for A, B, C, D, E, F atmospheric stability categories.	II-71
7	Dose	DoseCloudChild1SumM DoseCloudAdultSumM	Total (sum from all radionuclides) effective doses from submersion into passing through radioactive cloud for child and for adult respectively (Sv), for A, B, C, D, E, F atmospheric stability categories.	II-73
8	Dose	DoseCloudGroundChild1SumM DoseCloudGroundAdultSumM	Total (sum from all radionuclides) effective doses from submersion into passing through radioactive cloud and from ground irradiation for child and for adult respectively (Sv), for A, B, C, D, E, F atmospheric stability categories.	II-75

II-6. EXAMPLE CALCULATIONS

This section provides some practical examples demonstrating how accident situations can be assessed using Ecolego tool. To make shorter output tables of results, example calculations are performed with 4 radionuclides.

II-6.1. Consideration of short-term impact

Short-term impact typically is associated with human exposure from submersion into passing through radioactive cloud. Radiological consequences of such impact usually (especially in case of short release) cannot be mitigated due to quick dispersion of activity in the atmosphere. The direct irradiation from onto the ground settled radionuclides may also be accounted for certain duration.

Example considered accident conditions assume release of Mn-54, Co-60, Sr-90 and Cs-137 radionuclides with 1E+10 Bq of activity each. Emission to atmosphere occurs from a stack of 30 m height. The release duration is short (i.e. does not exceed 8h), thus meteorological conditions assumed to be steady. The buildings nearby are low and their influence on atmospheric dispersion may not be accounted. Human happens to meet radioactive cloud staying at the distance of 300 m from the stack. The 24 h period is assumed to be sufficient for evaluation of on-site radiological situation and implementation of immediate response actions, so human exposure from contaminated ground can be limited to 24 h.

Ecolego tool [II-8] input parameters to be modified are given in the table below. For other parameters reference values are kept, see Section II-5.1.

TABLE II-27. PARAMETERS TO BE MODIFIED OR CHECKED FOR 'Consideration of short-term impact' EXAMPLE

No	Category	Parameter	Values and comment
1	Scenario	InActivityRelease	Assumed release values of 1E+10 Bq are set for radionuclides Mn-54, Co-60, Sr-90 and Cs-137 (other radionuclides are ticked out).
2	Scenario	InDistance	Set to 300 m value for 'cloud' exposure location. As the ingestion dose is not considered, values for 'garden' and 'pasture' locations can be left as it is.
3	Scenario	InHeightRelease	Set to 30 m
4	Scenario	InTimeExpoGround	Set to 8.64E+04 s which corresponds to 24 h exposure
5	Scenario	InTimeExpoIngest	Set to 0 - ingestion dose is not calculated and not added to total dose. As alternative, a very small value, which actually do not influence calculation results (e.g. 0.1 s), can be set.
6	Scenario	InTimeFraExpoGround	Reference value of 1.0 is kept

The Ecolego tool outputs to be considered are given in table below.

TABLE II-28. IMPACT SUMMARIZING OUTPUTS FOR 'CConsideration of short-term impact' EXAMPLE

No	Sub-system	Block	Description
1	DoseTotal	DoseCloudChild1SumM DoseCloudAdultSumM	Total effective dose from submersion into passing through radioactive cloud for child and for adult respectively (Sv)
2	DoseTotal	DoseCloudGroundChild1SumM DoseCloudGroundAdultSumM	Total effective dose from submersion into passing through radioactive cloud and from ground irradiation for child and for adult respectively (Sv)

II-6.2. Consideration of long-term impact

Long term impact is usually associated with accident stipulated total exposure from all relevant external and internal exposure pathways. This includes human exposure during the accident (i.e. includes short-term dose) and additional long-term exposure from onto the ground settled radionuclides. Fallout and washout deposited radionuclides result in external irradiation from contaminated ground and in internal exposure from consumption of contaminated food products. Exposure duration to be considered may depend on assessment context however typically these are times allowing evaluation of maximal annual dose, dose of several consecutive years following the accident or human whole life dose.

Similarly to previous case (see Section II-6.1), the example considered accident conditions assume the atmospheric emission to occur from a stack of 30 m height. The release duration is short (i.e. does not exceed 8h), thus meteorological conditions assumed to be steady. The buildings nearby are low and their influence on atmospheric dispersion may not be accounted. Human exposure location is at the distance of 300 m from the stack; annually human spends in this location up to 2000 h. Terrestrial foods are produced at a greater distance from the source; crops from a distance of 800 m, and milk and meat from 1200 m. As a bounding case the annual dose is calculated assuming that no special mitigation measures are taken to reduce radiological impact.

Ecolego tool input parameters to be modified are given in the table below. For other parameters reference values are kept, see Section II-5.1.

TABLE II-29. PARAMETERS TO BE MODIFIED OR CHECKED FOR 'Consideration of long-term impact' EXAMPLE

No	Category	Parameter	Values and comment
1	Scenario	InActivityRelease	Assumed release values of 1E+10 Bq are set for radionuclides Mn-54, Co-60, Sr-90 and Cs-137 (other radionuclides are ticked out).
2	Scenario	InDistance	Set to 300 m value for 'cloud' exposure location, 800 m values for 'garden' exposure location and 1200 m for 'pasture' exposure location.
6	Scenario	InTimeFraExpoGround	Set to 0.228 (2000 h equals to 7.2E+06 s, so 7.2E+06 / 3.16E+07 = 0.228)

The Ecolego tool outputs to be considered are given in table below.

TABLE II-30. IMPACT SUMMARIZING OUTPUTS FOR ‘Consideration of long-term impact’ EXAMPLE

No	Sub-system	Block	Description
1	DoseTotal	DoseAllPathChild1SumM DoseAllpathAdultSumM	Total effective dose from all external and internal exposure paths for child and for adult respectively(Sv)

II-6.3. Consideration of buildings influence

Example follows previous case (Section II-6.2) additionally assuming that stack of 30 m height belongs (or is close) to the building which is of 20 m height and 40 m width.

Ecolego tool input parameters to be modified are given in the table below. For other parameters reference values are kept, see Section II-5.1.

TABLE II-31. PARAMETERS TO BE MODIFIED OR CHECKED FOR ‘Consideration of buildings influence’ EXAMPLE

No	Category	Parameter	Values and comment
1	Scenario	InActivityRelease	Assumed release values of 1E+10 Bq are set for radionuclides Mn-54, Co-60, Sr-90 and Cs-137 (other radionuclides are ticked out).
2	Scenario	InDistance	Set to 300 m value for ‘cloud’ exposure location, 800 m values for ‘garden’ exposure location and 1200 m for ‘pasture’ exposure location.
3	Scenario	InTimeFraExpoGround	set to 0.228 (2000 h equals to 7.2E+06 s, so 7.2E+06 / 3.16E+07 = 0.228)
4	Building	InBuildingHeight	Set to 20 m
5	Building	InBuildingWidth	Set to 40 m

The Ecolego tool [II-8] outputs to be considered are annual effective doses to members of the public see Table II-30.

II-7. CALCULATION OF SAFRAN DATABASES

II-7.1. Calculation of release screening values

For quantification of impacts due to accidental release of radionuclides from facility to the atmosphere, the SAFRAN tool [II-1] uses release screening values. Release screening values are defined as the release that would lead to 0.1 mSv annual effective dose to members of the population under conservative exposure conditions.

Ecolego tool input parameters to be modified are given in TABLE II-32 below. For other parameters reference values are kept, see Section II-5.1.

TABLE II-32. PARAMETERS TO BE MODIFIED FOR CALCULATION OF SCREENING RELEASE VALUES

No	Category	Parameter	Values and comment
1	Scenario	InDistance	It is conservatively assumed that critical group member live in average of 30 m from the source. This value is set for the 'cloud', 'garden' and 'pasture' exposure locations. In terms of atmospheric dispersion, this exposure point is inside the wake zone of the building from which the radioactive emission occurs.
2	Scenario	InHeightRelease	Set to 10 m
3	Building	InBuildingHeight	Set to 10 m
4	Building	InBuildingWidth	Set to 10 m

Calculation of the screening release values have been conducted in two steps. At first the effective dose to child and adult members are calculated for activity release of 1 Bq. In the second step the screening release value is obtained by scaling calculated dose (higher value from child and adult doses) from unit release to screening dose as follows:

$$Q_{S,j} = \frac{E_S}{E_{max,j}} Q_{unit} \qquad \text{(II-77)}$$

Where:

$Q_{S,j}$ is the screening release value for particular radionuclide j (Bq);

Q_{unit} = 1 Bq is the unit release;

E_S = 1E-04 Sv is the considered screening dose for accidents;

$E_{max,j}$ is the calculated exposure (higher value from child and adult doses) from unit release of particular radionuclide j (Sv).

The screening release calculation results and screening release values are presented in TABLE II-33 below.

TABLE II-33. CALCULATED UNIT RELEASE DOSES AND SCREENING RELEASE VALUES

No	Radionuclide	Unit release dose E, Sv		Screening release Q_s, Bq
		Age 1-2 a	Age > 17 a	
1	Ac-228	1.43E-14	9.77E-15	7.01E+09
2	Ag-110m	3.79E-12	8.68E-13	2.64E+07
3	Am-241 *)	1.41E-11	2.36E-11	4.24E+06
4	As-76	2.16E-15	1.02E-15	4.62E+10
5	At-211	4.30E-13	8.81E-14	2.33E+08
6	Au-198	2.79E-15	1.50E-15	3.59E+10
7	Bi-206	2.56E-14	1.71E-14	3.91E+09
8	Bi-210 *)	3.72E-14	3.83E-14	2.61E+09
9	Bi-212 *)	9.74E-15	1.19E-14	8.37E+09
10	Br-82	6.23E-14	1.26E-14	1.61E+09
11	C-14 **)	3.50E-14	2.20E-14	2.86E+09
12	Cd-109	2.53E-13	8.64E-14	3.95E+08
13	Ce-141	2.89E-14	9.94E-15	3.46E+09
14	Ce-144	7.52E-13	2.07E-13	1.33E+08
15	Cm-242	2.78E-12	2.31E-12	3.60E+07
16	Cm-244 *)	1.12E-11	1.47E-11	6.79E+06
17	Co-58	9.44E-14	5.97E-14	1.06E+09
18	Co-60	9.77E-13	4.99E-13	1.02E+08
19	Cr-51	3.84E-15	1.32E-15	2.60E+10
20	Cs-134 *)	1.02E-12	2.22E-12	4.50E+07
21	Cs-135 *)	1.25E-13	2.24E-13	4.46E+08
22	Cs-136	1.50E-13	7.87E-14	6.67E+08
23	Cs-137 *)	7.26E-13	1.53E-12	6.54E+07
24	Cu-64	4.49E-16	1.58E-16	2.23E+11
25	Eu-154	4.43E-13	2.69E-13	2.26E+08
26	Eu-155	5.87E-14	2.46E-14	1.70E+09
27	Fe-55	6.48E-14	2.36E-14	1.54E+09
28	Fe-59	1.72E-13	8.83E-14	5.81E+08
29	Ga-67	1.20E-15	9.80E-16	8.34E+10
30	H-3 **)	2.58E-15	1.54E-15	3.88E+10
31	Hg-197	7.15E-16	4.52E-16	1.40E+11
32	Hg-197m	6.13E-16	3.55E-16	1.63E+11
33	Hg-203 *)	5.86E-13	6.25E-13	1.60E+08
34	I-123	1.27E-15	2.06E-16	7.84E+10
35	I-125	1.30E-12	5.62E-13	7.70E+07

TABLE II-33. (Continued)

No	Radionuclide		Unit release dose E, Sv		Screening release Qs, Bq
			Age 1-2 a	Age > 17 a	
36	I-129		8.62E-12	7.43E-12	1.16E+07
37	I-131		1.09E-12	1.52E-13	9.17E+07
38	I-132		5.73E-16	2.95E-16	1.75E+11
39	I-133		4.27E-14	4.64E-15	2.34E+09
40	I-134		2.45E-16	2.03E-16	4.08E+11
41	I-135		3.34E-15	6.93E-16	2.99E+10
42	In-111		1.29E-15	8.96E-16	7.73E+10
43	In-113m		3.32E-17	3.00E-17	3.01E+12
44	Mn-54		1.41E-13	1.02E-13	7.08E+08
45	Mo-99		7.22E-15	1.88E-15	1.39E+10
46	Na-22		4.01E-12	1.40E-12	2.49E+07
47	Na-24		2.07E-14	4.51E-15	4.83E+09
48	Nb-95		1.86E-13	1.78E-13	5.36E+08
49	Ni-59		2.41E-14	5.08E-15	4.15E+09
50	Ni-63		5.95E-14	1.21E-14	1.68E+09
51	Np-237	*)	8.10E-12	1.29E-11	7.74E+06
52	Np-239		1.82E-15	9.51E-16	5.49E+10
53	P-32		1.49E-12	2.03E-13	6.72E+07
54	Pa-231	*)	4.99E-11	8.60E-11	1.16E+06
55	Pa-233		3.08E-14	1.35E-14	3.25E+09
56	Pb-210		8.31E-11	2.69E-11	1.20E+06
57	Pd-103		4.01E-14	5.34E-15	2.50E+09
58	Pd-107		2.01E-14	3.31E-15	4.97E+09
59	Pd-109		8.13E-15	1.06E-15	1.23E+10
60	Pm-147		4.29E-14	1.32E-14	2.33E+09
61	Po-210		1.52E-10	4.19E-11	6.56E+05
62	Pu-238	*)	1.51E-11	2.61E-11	3.83E+06
63	Pu-239	*)	1.58E-11	2.84E-11	3.52E+06
64	Pu-240	*)	1.58E-11	2.84E-11	3.52E+06
65	Pu-241	*)	2.07E-13	5.21E-13	1.92E+08
66	Pu-242	*)	1.50E-11	2.73E-11	3.67E+06
67	Ra-224		2.89E-12	1.35E-12	3.45E+07
68	Ra-225	*)	2.64E-12	3.31E-12	3.02E+07
69	Ra-226		3.61E-11	1.56E-11	2.77E+06
70	Rb-86		3.97E-13	6.57E-14	2.52E+08
71	Rh-105		1.29E-14	1.69E-15	7.73E+09
72	Rh-107		3.48E-17	2.54E-17	2.87E+12
73	Ru-103		4.05E-14	2.03E-14	2.47E+09

TABLE II-33. (Continued)

No	Radionuclide		Unit release dose E, Sv		Screening release Qs, Bq
			Age 1-2 a	Age > 17 a	
74	Ru-106		9.87E-13	2.93E-13	1.01E+08
75	S-35		6.68E-13	1.85E-13	1.50E+08
76	Sb-124		3.38E-13	1.20E-13	2.96E+08
77	Sb-125		2.41E-13	1.04E-13	4.15E+08
78	Se-75		3.23E-12	6.49E-13	3.09E+07
79	Sn-113		2.37E-13	1.27E-13	4.21E+08
80	Sr-85		7.51E-14	3.19E-14	1.33E+09
81	Sr-87m		6.15E-17	4.58E-17	1.63E+12
82	Sr-89		2.93E-13	5.95E-14	3.41E+08
83	Sr-90		2.54E-12	1.43E-12	3.94E+07
84	Tc-99		2.55E-13	1.12E-13	3.92E+08
85	Tc-99m		3.46E-17	3.09E-17	2.89E+12
86	Te-125m		1.67E-13	1.09E-13	5.99E+08
87	Te-127m		6.76E-13	3.85E-13	1.48E+08
88	Te-129m		4.13E-13	2.57E-13	2.42E+08
89	Te-131m		4.54E-15	2.16E-15	2.21E+10
90	Te-132		2.12E-14	9.50E-15	4.72E+09
91	Th-228		1.90E-11	1.79E-11	5.27E+06
92	Th-230	*)	1.22E-11	1.34E-11	7.48E+06
93	Th-232	*)	1.44E-11	1.84E-11	5.43E+06
94	Tl-201		1.28E-15	3.67E-16	7.81E+10
95	Tl-202		1.82E-14	1.07E-14	5.51E+09
96	U-232		2.20E-11	1.61E-11	4.55E+06
97	U-234		4.34E-12	3.51E-12	2.30E+07
98	U-235		4.11E-12	3.14E-12	2.43E+07
99	U-238		3.97E-12	3.14E-12	2.52E+07
100	Y-87		2.54E-15	1.91E-15	3.93E+10
101	Y-90		5.80E-15	2.13E-15	1.72E+10
102	Y-91		1.73E-13	4.65E-14	5.78E+08
103	Zn-65	*)	4.86E-14	9.82E-13	1.02E+08
104	Zr-95		1.27E-13	9.19E-14	7.88E+08

*) Adults are limiting exposure group.

**) Intake of C and H through terrestrial food chains and corresponding ingestion doses are not considered, see note below Table II-19.

II-7.2. Calculation of released activity to dose conversion factors

The SAFRAN tool [II-1] includes procedure for evaluation of population exposure due to accidental radionuclide releases to the atmosphere. The dose to member of population is calculated as follows:

$$E = \sum_j Q_j \, DCF_{airj} \tag{II-78}$$

Where:

Q_j is the release to atmosphere of the j-th radionuclide (Bq).

DCF_{airj} is the unit released activity to effective dose conversion factor (i.e. dose conversion factor) for accidental release to the atmosphere of j-th radionuclide. Values of the DCF are picked from the SAFRAN internal database.

Values of the DCF_{airj} for SAFRAN database are calculated assuming exposure conditions similar to those that were used for calculation of the generic dose conversion factors for routine radioactive releases [II-2].

Ecolego tool [II-8] input parameters to be modified are given in TABLE II-34 below. For other parameters reference values are kept, see Section II-5.1.

TABLE II-34. PARAMETERS TO BE MODIFIED OR CHECKED FOR CALCULATION OF SCREENING RELEASE VALUES

No	Category	Parameter	Values and comment
1	Scenario	InDistance	It is assumed that critical group member live at a distance of 30 m from the source. This value is set for the 'cloud' exposure location. It is assumed that terrestrial foods are produced at a greater distance from the source: crops from a distance of 100 m and milk and meat from 800 m. These values are set for the 'garden' and 'pasture' exposure locations respectively.
2	Scenario	InHeightRelease	Set to 10 m
3	Building	InBuildingHeight	Set to 10 m
4	Building	InBuildingWidth	Set to 10 m

The dose conversion factors calculation results and SAFRAN tool [II-1] database selected values are presented in TABLE II-35. Similarly to approach used in [II-3], the higher value from calculated child and adult DCF_{airj} are selected for the database. This approach would lead to slight overestimation of the total dose, however DCF_{airj} in the database are addressed independently from age group of considered members of the population.

TABLE II-35. CALCULATED AND THE SAFRAN INTERNAL DATABASE SELECTED RELEASED ACTIVITY TO DOSE CONVERSION FACTORS

No	Radionuclide	DCF_{airj}, Sv/Bq		SAFRAN database DCF_{airj}, Sv/Bq
		Age 1-2 a	Age > 17 a	
1	Ac-228	1.42E-14	1.12E-14	1.42E-14
2	Ag-110m	4.82E-13	1.38E-13	4.82E-13
3	Am-241 *)	8.99E-12	2.09E-11	2.09E-11
4	As-76	1.18E-15	5.91E-16	1.18E-15
5	At-211	5.11E-14	5.04E-14	5.11E-14
6	Au-198	1.67E-15	8.13E-16	1.67E-15
7	Bi-206	1.39E-14	5.98E-15	1.39E-14
8	Bi-210 *)	2.81E-14	4.18E-14	4.18E-14
9	Bi-212 *)	9.72E-15	1.38E-14	1.38E-14
10	Br-82	5.29E-15	1.61E-15	5.29E-15
11	C-14 **)	1.33E-14	7.20E-15	1.33E-14
12	Cd-109	8.20E-14	2.52E-14	8.20E-14
13	Ce-141	1.16E-14	3.83E-15	1.16E-14
14	Ce-144	2.90E-13	7.04E-14	2.90E-13
15	Cm-242 *)	2.02E-12	2.41E-12	2.41E-12
16	Cm-244 *)	7.28E-12	1.33E-11	1.33E-11
17	Co-58	5.42E-14	1.95E-14	5.42E-14
18	Co-60	5.27E-13	1.67E-13	5.27E-13
19	Cr-51	9.31E-16	3.02E-16	9.31E-16
20	Cs-134	3.31E-13	3.12E-13	3.31E-13
21	Cs-135 *)	2.21E-14	2.65E-14	2.65E-14
22	Cs-136	2.51E-14	1.03E-14	2.51E-14
23	Cs-137 *)	1.92E-13	2.03E-13	2.03E-13
24	Cu-64	1.44E-16	9.13E-17	1.44E-16
25	Eu-154	2.67E-13	1.09E-13	2.67E-13
26	Eu-155	2.68E-14	9.63E-15	2.68E-14
27	Fe-55	1.90E-14	3.93E-15	1.90E-14
28	Fe-59	6.42E-14	2.01E-14	6.42E-14
29	Ga-67	5.07E-16	2.71E-16	5.07E-16
30	H-3 **)	9.68E-16	4.72E-16	9.68E-16
31	Hg-197	3.48E-16	2.13E-16	3.48E-16
32	Hg-197m	3.94E-16	2.89E-16	3.94E-16
33	Hg-203	5.96E-14	2.79E-14	5.96E-14
34	I-123	2.14E-16	7.23E-17	2.14E-16
35	I-125	2.33E-13	8.33E-14	2.33E-13
36	I-129	1.95E-12	1.31E-12	1.95E-12
37	I-131	9.07E-14	1.55E-14	9.07E-14

TABLE II-35. (Continued)

No	Radionuclide		DCF$_{airj}$, Sv/Bq		SAFRAN database DCF$_{airj}$, Sv/Bq
			Age 1-2 a	Age > 17 a	
38	I-132		3.33E-16	2.21E-16	3.33E-16
39	I-133		4.97E-15	1.15E-15	4.97E-15
40	I-134		2.17E-16	1.93E-16	2.17E-16
41	I-135		8.32E-16	3.50E-16	8.32E-16
42	In-111		8.95E-16	4.17E-16	8.95E-16
43	In-113m		3.19E-17	2.70E-17	3.19E-17
44	Mn-54		9.99E-14	3.86E-14	9.99E-14
45	Mo-99		1.39E-15	7.01E-16	1.39E-15
46	Na-22		5.40E-13	1.66E-13	5.40E-13
47	Na-24		2.54E-15	9.81E-16	2.54E-15
48	Nb-95		2.83E-14	1.27E-14	2.83E-14
49	Ni-59		3.53E-15	8.24E-16	3.53E-15
50	Ni-63		8.75E-15	2.03E-15	8.75E-15
51	Np-237	*)	5.22E-12	1.14E-11	1.14E-11
52	Np-239		1.04E-15	6.20E-16	1.04E-15
53	P-32		7.82E-14	9.23E-15	7.82E-14
54	Pa-231	*)	3.06E-11	7.02E-11	7.02E-11
55	Pa-233		1.29E-14	4.77E-15	1.29E-14
56	Pb-210		2.91E-11	8.00E-12	2.91E-11
57	Pd-103		2.87E-15	5.63E-16	2.87E-15
58	Pd-107		3.05E-15	7.16E-16	3.05E-15
59	Pd-109		6.15E-16	2.15E-16	6.15E-16
60	Pm-147		1.60E-14	4.97E-15	1.60E-14
61	Po-210		5.01E-11	1.08E-11	5.01E-11
62	Pu-238	*)	9.66E-12	2.30E-11	2.30E-11
63	Pu-239	*)	1.01E-11	2.50E-11	2.50E-11
64	Pu-240	*)	1.01E-11	2.50E-11	2.50E-11
65	Pu-241	*)	1.30E-13	4.52E-13	4.52E-13
66	Pu-242	*)	9.58E-12	2.40E-11	2.40E-11
67	Ra-224	*)	8.85E-13	1.35E-12	1.35E-12
68	Ra-225	*)	1.68E-12	2.90E-12	2.90E-12
69	Ra-226		9.51E-12	4.81E-12	9.51E-12
70	Rb-86		3.39E-14	5.83E-15	3.39E-14
71	Rh-105		9.12E-16	2.58E-16	9.12E-16
72	Rh-107		2.68E-17	2.61E-17	2.68E-17
73	Ru-103		2.14E-14	7.52E-15	2.14E-14
74	Ru-106		3.85E-13	9.37E-14	3.85E-13
75	S-35		5.47E-14	1.02E-14	5.47E-14

TABLE II-35. (Continued)

No	Radionuclide	DCF_{airj}, Sv/Bq		SAFRAN database DCFairj, Sv/Bq
		Age 1-2 a	Age > 17 a	
76	Sb-124	1.21E-13	3.89E-14	1.21E-13
77	Sb-125	1.04E-13	3.54E-14	1.04E-13
78	Se-75	2.34E-13	4.42E-14	2.34E-13
79	Sn-113	4.94E-14	1.53E-14	4.94E-14
80	Sr-85	3.08E-14	1.04E-14	3.08E-14
81	Sr-87m	4.73E-17	3.49E-17	4.73E-17
82	Sr-89	6.54E-14	1.49E-14	6.54E-14
83	Sr-90	7.08E-13	3.51E-13	7.08E-13
84	Tc-99	7.34E-14	1.31E-14	7.34E-14
85	Tc-99m	3.23E-17	2.24E-17	3.23E-17
86	Te-125m	3.13E-14	8.80E-15	3.13E-14
87	Te-127m	1.35E-13	2.99E-14	1.35E-13
88	Te-129m	6.68E-14	1.81E-14	6.68E-14
89	Te-131m	2.37E-15	1.10E-15	2.37E-15
90	Te-132	8.96E-15	3.39E-15	8.96E-15
91	Th-228 *)	1.43E-11	1.86E-11	1.86E-11
92	Th-230 *)	6.58E-12	8.61E-12	8.61E-12
93	Th-232 *)	8.31E-12	1.38E-11	1.38E-11
94	Tl-201	2.75E-16	1.03E-16	2.75E-16
95	Tl-202	4.55E-15	1.80E-15	4.55E-15
96	U-232	8.92E-12	7.16E-12	8.92E-12
97	U-234	2.26E-12	2.19E-12	2.26E-12
98	U-235	2.03E-12	1.94E-12	2.03E-12
99	U-238	2.05E-12	1.89E-12	2.05E-12
100	Y-87	1.94E-15	8.60E-16	1.94E-15
101	Y-90	2.68E-15	1.14E-15	2.68E-15
102	Y-91	6.62E-14	1.63E-14	6.62E-14
103	Zn-65 *)	4.86E-14	7.84E-14	7.84E-14
104	Zr-95	7.38E-14	2.82E-14	7.38E-14

*) Adults are limiting exposure group.

**) Intake of C and H through terrestrial food chains and corresponding ingestion doses are not considered, see note below Table II-19.

REFERENCES TO ANNEX II

[II-1] SAFRAN Tool and SAFRAN User's Guide, http://goto.iaea.org/safran

[II-2] INTERNATIONAL ATOMIC ENERGY AGENCY, Generic Models and Parameters for Assessing the Environmental Transfer of Radionuclides from Routine Releases: Exposures of Critical Groups, Safety Series No. 57, IAEA, Vienna (1982).

[II-3] INTERNATIONAL ATOMIC ENERGY AGENCY, Generic Models for Use in Assessing the Impact of Discharges of Radioactive Substances into the Environment, Safety Report Series No. 19, IAEA, Vienna (2001).

[II-4] DIE STRAHLENSCHUTZKOMMISSION, Störfallberechnungsgrundlagen für die Leitlinien des BMI zur Beurteilung der Auslegung von Kernkraftwerken mit DWR gemäß § 28 Abs. 3 StrlSchV Strahlenschutzkommission 1983, Neufassung des Kapitels 4: Berechnung der Strahlenexposition, Bonn, Germany (2003).

[II-5] ECKERMAN, K.F., RYMAN, J.C., External Exposure to Radionuclides in Air, Water and Soil, Federal Guidance Report No. 12, US Environmental Protection Agency, Washington, DC (1993).

[II-6] INTERNATIONAL COMMISSION ON RADIOLOGICAL PROTECTION, Basic Anatomical and Physiological Data for use in Radiological Protection: Reference Values, Publication 89, Pergamon Press, Oxford and New York (2003).

[II-7] EUROPEAN COMMISSION, FOOD AND AGRICULTURE ORGANIZATION OF THE UNITED NATIONS, INTERNATIONAL ATOMIC ENERGY AGENCY, INTERNATIONAL LABOUR ORGANIZATION, OECD NUCLEAR ENERGY AGENCY, PAN AMERICAN HEALTH ORGANIZATION, UNITED NATIONS ENVIRONMENT PROGRAMME, WORLD HEALTH ORGANIZATION, Radiation Protection and Safety of Radiation Sources: International Basic Safety Standards, IAEA Safety Standards Series No. GSR Part 3, IAEA, Vienna (2014).

[II-8] Ecolego and Ecolego Toolbox, http://ecolego.Facilia.se/ecolego/show/ecolego

ANNEX III. ADESCRIPTION OF THE SAFRAN MODELS FOR EVALUATION OF EXTERNAL EXPOSURE

III-1. INTRODUCTION

In this document an approximate external dose calculation methodology is presented for some source geometries: point, disc, drum (cylinder) and cube. Some general assumptions are made:

- All sources are assumed isotropic and homogenous.
- The shield is extended in x.y-plane with a thickness of t in the z-direction. The attenuation from the slab shield is energy and material dependent.

Primary photons are considered in the dose rate expressions. The contribution from scattered and secondary photons within the shield is accounted for by including buildup factors (e.g. Berger, Taylor and geometric progression (GP)). With the assumption of an isotropic point source and homogeneous infinite shield the total flux, ϕ [photons/cm^2 s]:

$$\phi(\varepsilon) = B_N \phi^0(\varepsilon) \qquad \text{(III-1)}$$

where:

B_N is number buildup factor [-];

$\phi^0(\varepsilon)$ is primary photon flux [photons cm^{-2} s^{-1}].

The number buildup factor is the ratio between the actual photon flux (the scattered photons included) and the primary photon flux. In case of a monoenergetic source, secondary photons may have energies up to the energy of the primary photons. The buildup factor depends on the primary photon energy, shield attenuation and the shield material. In order to calculate the dose, the dose buildup factor B_D should be used instead. The dose rate, $\dot{D}$ [mSv h^{-1}], received from the exposed photon flux per nuclide or element can be estimated by:

$$\dot{D} = 5{,}77x10^{-4} \sum_{\gamma} \mu_a(\varepsilon_\gamma) p_\gamma \varepsilon_\gamma B_D \phi^0\left(E_\gamma\right) \qquad \text{(III-2)}$$

where:

μ_a is mass absorption coefficient [cm^2 g^{-1}];

For example, for water the value is 0,031 cm^2 g^{-1} at 1 MeV [III-1];

ε_γ is photon energy [MeV];

p_γ is photon yield [-].

For nuclide data on its photon energy and yield see nuclear textbooks.

The total dose D_{tot} [mSv] is a sum of each individual radionuclide contribution to the dose:

$$D_{tot} = \sum_{nuclides} \dot{D} \cdot t, \tag{III-3}$$

where:

t is duration of exposure [h].

For a point-isotropic source the Geometric Progression (GP) buildup is used:

$$B(\varepsilon, \mu t) = 1 + (b-1)\cdot\left(K^{\mu t} - 1\right)/(K-1), \quad \textbf{if} \quad K \neq 1 \tag{III-4}$$

$$B(\varepsilon, \mu t) = 1 + (b-1)\cdot \mu t, \quad \text{if} \quad K = 1 \tag{III-5}$$

where:

μ is linear attenuation coefficient of shield [cm^{-1}];

t is thickness of shield [cm];

$$K(\varepsilon, \mu t) = c\cdot(\mu t)^a + d\cdot\frac{\tanh\left(\frac{\mu t}{X_k} - 2\right) - \tanh(-2)}{1 - \tanh(-2)};$$

$$\tanh(x) = \frac{\sinh(x)}{\cosh(x)} = \frac{e^x - e^{-x}}{e^x + e^{-x}};$$

a, b, c, d, X_k are parameters depending on the photon energy and the attenuating medium, and the nature of response (*e.g.* kerma* in air) [-].

Geometric Progression parameters for iron, lead, water and concrete can be found in *DataForExternalDoseCalc.xls*.

In order to incorporate the buildup factor into the dose expression for the other geometries the most straight-forward approximation is to use the Taylor form (less accurate compared to GP but conservative at low energy and low atomic number Z):

$$B(\varepsilon, \mu t) = \sum_{i=1}^{2} A_i \cdot e^{-\alpha_i \cdot \mu t}, \tag{III-6}$$

where:

μ is linear attenuation coefficient of shield [cm^{-1}];

t is thickness of shield [cm];

* Kerma= Kinetic energy released per unit mass.

$A_1, A_2, \alpha_1, \alpha_2$, where $A_2 = 1 - A_1$ are parameters depending on the photon energy and the attenuating medium [-].

Taylor parameters for iron, lead, water and concrete can be found in *DataForExternalDoseCalc.xls*.

The above buildup factors are valid for a point-isotropic monoenergetic source in an infinite media and can be used in the point-kernel method when integrating areal or volume sources. The use of buildup factors for an infinite shielding media instead of those for a finite shield is a conservative approach when assuming normally incidence on shield [III-2].

In Section III-2 dose rates for different geometries is presented with or without a slab shield. Due to the scope of the project, the expressions for the dose rate when the buildup from the shield is included have been derived in a 'back-of'-the-envelope' manner. However, some expressions (point, disc and cylinder 'on end' source) have been verified in the literature. In general, the expressions should be used with care before they have been properly validated with some other method.

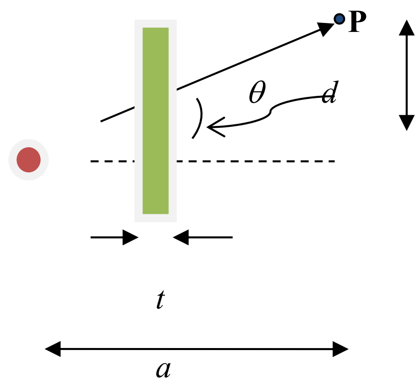
How to deal with multiple shields is discussed in Section III-3.

As mentioned earlier, included in the dose rate for different geometries presented below is the buildup from shield but not the buildup from the volume source itself. A less supported solution to this problem is discussed in Section III-4.

How to go from a measured dose rate to a source term is investigated in Section III-5. Finally, some dose rate calculations are performed in Section III-6. Since the calculations have not been validated with other methods, no guarantees can be made for its accuracy.

III-2. DOSE RATES FOR DIFFERENT GEOMETRIES

III-2.1. Point source



Given a point source and a slab shield in between the source and the point P of interest, the dose rate, $\dot{D}$ [mSv h^{-1}]:

$$\dot{D} = 5,77x10^{-4} \sum_{\gamma} \mu_a(\varepsilon_\gamma) p_\gamma \varepsilon_\gamma B_D S \frac{e^{-b_1 \sec\theta}}{4\pi(a\sec\theta)^2}, \qquad \text{(III-7)}$$

where:

μ_a is mass absorption coefficient [cm^2 g^{-1}];

ε_γ is photon energy [MeV];

p_γ is photon yield [-];

For nuclide data on its photon energy and yield see nuclear textbooks.

B_D is dose buildup factor (GP) [-];

Geometric Progression parameters for iron, lead, water and concrete can be found in *DataForExternalDoseCalc.xls*.

S is source strength. In this case the radionuclide activity [Bq];

$a\sec\theta$ is distance from source to point P [cm]:

$$\sec\theta = \frac{\sqrt{d^2 + a^2}}{a}$$

b_1 is shield attenuation [-].

$b_1 = \mu t$,

where:

μ is linear attenuation coefficient of shield [cm^{-1}];

t is thickness of shield [cm].

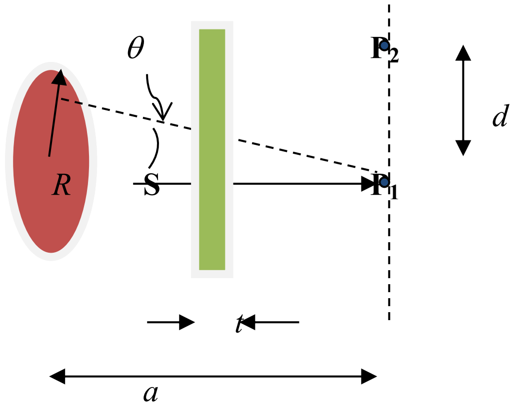
To get the linear attenuation coefficient μ (in cm^{-1}) just multiply the mass attenuation coefficients (in cm^2 g^{-1}) with density (g cm^{-3}). Values for iron, lead, water and concrete can be found in *DataForExternalDoseCalc.xls*.

Without shield $b_1 = 0$ and $B_D = 1$.

III-2.2. Disc source

III-2.2.1. With shield

III-2.2.2.



Given a circular disc source and a slab shield in between the source and the point of interest P (on- or off axis), situated at distance a away from the disc centre, the dose rate $\dot{D}$ [mSv h^{-1}] becomes:

$$\dot{D} = 5{,}77x10^{-4} \sum_{\gamma} \mu_a(\varepsilon_\gamma) p_\gamma \varepsilon_\gamma \frac{S}{2} \cdot$$
$$\sum_i A_i \left[E_1([1+\alpha_i]b_1) - E_1([1+\alpha_i]b_1 \sqrt{1+R^2/a^2}) \right] \quad \text{(III-8)}$$

where:

μ_a is mass absorption coefficient [cm^2 g^{-1}];

ε_γ is photon energy [MeV];

p_γ is photon yield [-];

For nuclide data on its photon energy and yield see nuclear textbooks.

S is source strength. In this case the radionuclide activity per area (πR^2) [Bq cm^{-2}];

A_i, α_i is Taylor buildup parameters [-];

Taylor parameters for iron, lead, water and concrete can be found in *DataForExternalDoseCalc.xls*.

R is radius of the disc source [cm];

a is distance from source to point P_1 [cm];

$E_1(x)$ is Exponential integral:

$$E_n(x) = x^{n.-1}\int_x^{\infty}\frac{e^{-y}}{y^n}dy,$$

Evaluated by the function in Java: double Expint(double x);

b_1 is shield attenuation [-];

$b_1 = \mu t,$

where:

μ is linear attenuation coefficient of shield [cm^{-1}];

t is thickness of shield [cm].

To get the linear attenuation coefficient μ (in cm^{-1}) just multiply the mass attenuation coefficients (in $cm^2\ g^{-1}$) with density ($g\ cm^{-3}$). Values for iron, lead, water and concrete can be found in *DataForExternalDoseCalc.xls*.

The dose rate $\dot{D}$ [$mSv\ h^{-1}$] at P_2 situated at a distance d away from P_1 (off axis point):

$$\dot{D} = 5{,}77x10^{-4}\sum_{\gamma}\mu_a(\varepsilon_\gamma)p_\gamma\varepsilon_\gamma\frac{S}{2}\cdot K\cdot$$
$$\sum_i A_i\left[E_1([1+\alpha_i]b_1) - E_1([1+\alpha_i]b_1\sqrt{1+R^2/a^2})\right] \qquad \text{(III-9)}$$

where:

K is correction factor depending on $d, a, \overline{\mu}$ (see Figure 6.3-3 on page 368 in III-3);

By letting $K = 1$, the dose estimation is conservative.

III-2.2.3. Without shield

Given a circular disc source without any shielding, the dose rate $\dot{D}$ [$mSv\ h^{-1}$] at the point P (P_1 or P_2) of interest is [III-4]:

$$\dot{D} = 5{,}77x10^{-4}\sum_{\gamma}\mu_a(\varepsilon_\gamma)p_\gamma\varepsilon_\gamma\frac{S}{4}\cdot$$
$$\ln\left(\frac{1}{2a^2}\left[a^2 + R^2 - d^2 + \sqrt{R^4 + 2R^2(a^2-d^2) + (a^2+d^2)^2}\right]\right), \qquad \text{(III-10)}$$

where:

μ_a is mass absorption coefficient [cm^2 g^{-1}];

ε_γ is photon energy [MeV];

p_γ is photon yield [-];

For nuclide data on its photon energy and yield see nuclear textbooks.

S is source strength. In this case the radionuclide activity per area (πR^2) [Bq cm^{-2}];

R is radius of the disc source [cm];

a is distance from source to point P in axial direction [cm];

d is distance from center point P_1 to P_2 (radial direction) [cm].

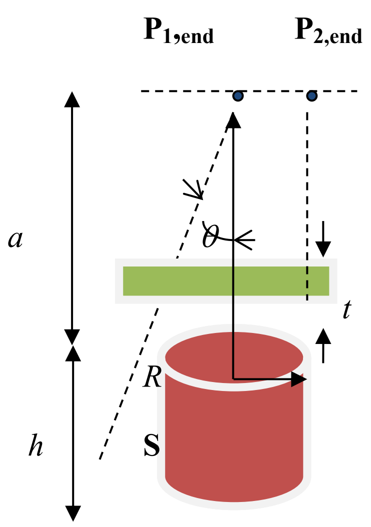
At P_1 $d = 0$ and the above expression becomes:

$$\dot{D} = 5{,}77x10^{-4} \sum_{\gamma} \mu_a(\varepsilon_\gamma) p_\gamma \varepsilon_\gamma \frac{S}{4} \cdot \ln\left(1 + R^2 / a^2\right)$$

III-2.3. Drum

A drum is defined as a cylindrical source with or without a slab shield outside the source. The point of interest is in the cylindrical axis direction outside the slab. Also, the case with a point in the radial direction outside a slab shield positioned at the end of the source is considered.

III-2.3.1. Exterior on end, with shield



For a uniformly distributed drum source (self-absorbing, $\mu_s \neq 0$) a slab shield is positioned outside the end of source and in between the source and the point of interest $P_{1,end}$. By replacing the cylinder with a large truncated cone outlined by the angle θ (see figure) the following 2 approximations (upper limits) of the dose rate, $\dot{D}$ [mSv h^{-1}], at $P_{1,end}$ are valid [III-5 and III-3].

For the case $h < 3/\mu_s$:

$$\dot{D} = 5{,}77x10^{-4} \sum_\gamma \mu_a(\varepsilon_\gamma) p_\gamma \varepsilon_\gamma \frac{S}{2\mu_s} \cdot$$

$$\sum_i A_i \cdot \left\{ \begin{array}{l} E_2([1+\alpha_i]b_1) - E_2([1+\alpha_i]b_1 + \mu_s h) + \\ \frac{1}{\sqrt{1+R^2/a^2}} \left[E_2(([1+\alpha_i]b_1 + \mu_s h)\sqrt{1+R^2/a^2}) - E_2([1+\alpha_i]b_1 \sqrt{1+R^2/a^2}) \right] \end{array} \right\}, \qquad \text{(III-11)}$$

For the case $h \geq 3/\mu_s$**:**

$$\dot{D} = 5{,}77x10^{-4} \sum_\gamma \mu_a(\varepsilon_\gamma) p_\gamma \varepsilon_\gamma \frac{S}{2\mu_s} \cdot$$

$$\sum_i A_i \cdot \left[E_2([1+\alpha_i]b_1) - \frac{1}{\sqrt{1+R^2/a^2}} E_2([1+\alpha_i]b_1 \sqrt{1+R^2/a^2}) \right], \qquad \text{(III-12)}$$

where:

μ_a is mass absorption coefficient [cm^2 g^{-1}];

ε_γ is photon energy [MeV];

p_γ is photon yield [-];

For nuclide data on its photon energy and yield see nuclear textbooks.

S is source strength. In this case the radionuclide activity per volume ($\pi R^2 h$) [Bq cm^{-3}];

A_i, α_i are Taylor buildup parameters [-];

Taylor parameters for iron, lead, water and concrete can be found in *DataForExternalDoseCalc.xls*.

R is radius of the cylinder source [cm];

h is height of the cylinder source [cm];

a is distance from end of source to point P_1 [cm];

μ_s is linear attenuation coefficient of the source [cm^{-1}];

To get the linear attenuation coefficient μ (in cm^{-1}) just multiply the mass attenuation coefficients (in cm^2 g^{-1}) with density (g cm^{-3}). Values for iron, lead, water and concrete can be found in *DataForExternalDoseCalc.xls*.

$E_2(x)$ is exponential integral:

$$E_n(x) = x^{n.-1}\int_x^{\infty}\frac{e^{-y}}{y^n}dy,$$

Evaluated by the function in Java: double Expint2(double x);

b_1 is shield attenuation [-];

$$b_1 = \mu t,$$

where:

μ is linear attenuation coefficient of shield [cm^{-1}];

t is thickness of shield [cm].

To get the linear attenuation coefficient μ (in cm^{-1}) just multiply the mass attenuation coefficients (in cm^2 g^{-1}) with density (g cm^{-3}). Values for iron, lead, water and concrete can be found in *DataForExternalDoseCalc.xls*.

III-2.3.2. Exterior on end, without shield

Using the approximation of the cylinder with a truncated cone as before, the dose rate $\dot{D}$ [mSv h^{-1}] at a distance a from the end of the cylinder becomes [III-5] [III-3]:

$$\dot{D} = 5{,}77x10^{-4}\sum_{\gamma}\mu_a(\varepsilon_\gamma)p_\gamma\varepsilon_\gamma\frac{S}{2\mu_s}\cdot \\ \left\{1 - E_2(\mu_s h) + \frac{1}{\sqrt{1+R^2/a^2}}\left[E_2(\mu_s h\sqrt{1+R^2/a^2}) - 1\right]\right\}, \qquad \text{(III-13)}$$

where:

μ_a is mass absorption coefficient [cm^2 g^{-1}];

ε_γ is photon energy [MeV];

p_γ is photon yield [-];

For nuclide data on its photon energy and yield see nuclear textbooks.

S is source strength. In this case the radionuclide activity per volume ($\pi R^2 h$) [Bq cm^{-3}];

R is radius of the cylinder source [cm];

h is height of the cylinder source [cm];

a is distance from end of source to point P_1 [cm];

μ_s is linear attenuation coefficient of the source [cm^{-1}];

To get the linear attenuation coefficient μ (in cm^{-1}) just multiply the mass attenuation coefficients (in cm^2 g^{-1}) with density (g cm^{-3}). Values for iron, lead, water and concrete can be found in *DataForExternalDoseCalc.xls*.

$E_2(x)$ is exponential integral:

$$E_n(x) = x^{n.-1}\int_x^\infty \frac{e^{-y}}{y^n}\,dy,$$

Evaluated by the function in Java: double Expint2(double x).

Given a uniformly distributed drum source with non-absorbing material (*i.e.* μ_s=0) the dose rate $\dot{D}$ [mSv h^{-1}] at the point $P_{1,end}$ of interest is:

$$\dot{D} = 5{,}77x10^{-4}\sum_\gamma \mu_a(\varepsilon_\gamma)p_\gamma\varepsilon_\gamma\frac{S}{4}\cdot$$
$$\left\{(h+a)\left[\ln\left(1+\frac{R^2}{(h+a)^2}\right)+\frac{2R}{(h+a)}\arctan\left(\frac{h+a}{R}\right)\right]-a\left[\ln\left(1+\frac{R^2}{a^2}\right)+\frac{2R}{a}\arctan\left(\frac{a}{R}\right)\right]\right\},\tag{III-14}$$

where:

μ_a is mass absorption coefficient [cm^2 g^{-1}];

ε_γ is photon energy [MeV];

p_γ is photon yield [-];

For nuclide data on its photon energy and yield see nuclear textbooks.

S is source strength. In this case the radionuclide activity per volume ($\pi R^2 h$) [Bq cm^{-3}];

R is radius of the cylinder source [cm];

h is height of the cylinder source [cm];

a is distance from end of source to point P_1 [cm].

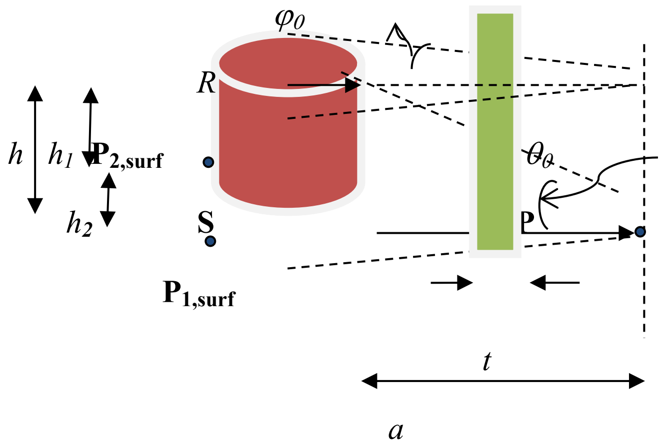
At $P_{2,end}$ situated at a distance R away from the cylinder axis:

$$\dot{D} = 5{,}77x10^{-4}\sum_{\gamma}\mu_a(\varepsilon_\gamma)p_\gamma\varepsilon_\gamma\frac{S}{2}\cdot\left[(h+a)\varphi_A\left(\frac{h+a}{R}\right)-a\varphi_A\left(\frac{a}{R}\right)\right], \tag{III-15}$$

where:

$$\varphi_A\left(\frac{h}{R}\right)=\frac{1}{2}\left[1+\ln\left(\frac{1}{2}+\frac{\sqrt{1+4R^2/h^2}}{2}\right)+\frac{2R}{h}-\sqrt{1+4R^2/h^2}\right].$$

III-2.3.3. Exterior on side, with shield



An approximation for an absorbing drum (cylindrical) source is to replace it with columnar fragment of cylindrical shell having a base area equal to that of the original cylinder. Introducing a slab shield outside the (self-absorbing) source, the dose rate $\dot{D}$ [mSv h^{-1}] at a point P in the same plane as the cylindrical bottom is approximated to [III-6]:

$$\begin{aligned}\dot{D} &= 5{,}77x10^{-4}\sum_{\gamma}\mu_a(\varepsilon_\gamma)p_\gamma\varepsilon_\gamma\frac{S}{2\pi\mu_s}\cdot\sum_i A_i\cdot\int_0^{\varphi_0}\begin{bmatrix}G(\theta_0,[1+\alpha_i]b_1\sec\varphi)-\\ G(\theta_0,[1+\alpha_i]b_1\sec\varphi+m_c\mu_s R)\end{bmatrix}d\varphi\\ &= 5{,}77x10^{-4}\sum_{\gamma}\mu_a(\varepsilon_\gamma)p_\gamma\varepsilon_\gamma\frac{S}{2\pi\mu_s}\cdot\sum_i A_i\cdot\left\{\int_0^{\varphi_0}\int_0^{\theta_0}\cos\theta\cdot e^{-[1+\alpha_i]b_1\sec\varphi\sec\theta}\left[1-e^{-m_c\mu_s R\sec\theta}\right]d\theta d\varphi\right\}\\ &= 5{,}77x10^{-4}\sum_{\gamma}\mu_a(\varepsilon_\gamma)p_\gamma\varepsilon_\gamma\frac{S}{2\pi\mu_s}\cdot\sum_i A_i\cdot F([1+\alpha_i]b_1,\mu_s,R,h,a),\end{aligned}$$

(III-16)

where:

μ_a is mass absorption coefficient [cm^2 g^{-1}];

ε_γ is photon energy [MeV];

p_γ is photon yield [-];

For nuclide data on its photon energy and yield see nuclear textbooks.

$F(b_1, \mu_s, R, h, a)$: Double integral evaluated by the function in Java:

double F(double b1, double ms, double r, double h, double a);

S is source strength. In this case the radionuclide activity per volume ($\pi R^2 h$) [Bq cm^{-3}];

A_i, α_i are Taylor buildup parameters [-];

Taylor parameters for iron, lead, water and concrete can be found in *DataForExternalDoseCalc.xls*.

φ_0 is angle (in radians, see figure).

$$\varphi_0 = \arcsin\left(\frac{R}{(a+R)}\right);$$

where:

R is radius of the cylinder source [cm];

a is distance from end of source to point P [cm];

μ_s is linear attenuation coefficient of the source [cm^{-1}];

To get the linear attenuation coefficient μ (in cm^{-1}) just multiply the mass attenuation coefficients (in cm^2 g^{-1}) with density (g cm^{-3}). Values for iron, lead, water and concrete can be found in *DataForExternalDoseCalc.xls*.

b_1 is shield attenuation [-]:

$$b_1 = \mu t,$$

where:

μ is linear attenuation coefficient of shield [cm^{-1}];

t is thickness of shield [cm].

To get the linear attenuation coefficient μ (in cm^{-1}) just multiply the mass attenuation coefficients (in cm^2 g^{-1}) with density (g cm^{-3}). Values for iron, lead, water and concrete can be found in *DataForExternalDoseCalc.xls*.

θ_0 is angle (in radians, see figure):

$$\theta_0 = \arctan\left(\frac{h}{a}\right);$$

where:

h is height of the cylinder source [cm].

Finally,

m_c is equivalent volume coefficient [-]:

$$m_c = \sqrt{(a/R)^2 + \pi/\varphi_0} - a/R.$$

III-2.3.4. Exterior on side, without shield

Without shield b_1=0, and the shield buildup parameters α_i=0 (dose buildup factor =1), then Eq. (III-16) becomes:

$$\dot{D} = 5{,}77x10^{-4}\sum_{\gamma}\mu_a(\varepsilon_\gamma)p_\gamma\varepsilon_\gamma\frac{S}{2\pi\mu_s}\cdot F(0,\mu_s,R,h,a), \tag{III-17}$$

The dose rate $\dot{D}$ [mSv h^{-1}] just outside an unshielded cylinder containing non-absorbing material (*i.e.* μ_s=0):

$$\dot{D} = 5{,}77x10^{-4}\sum_{\gamma}\mu_a(\varepsilon_\gamma)p_\gamma\varepsilon_\gamma\frac{Sh}{2}\varphi_A\left(\frac{h}{R}\right), \tag{III-18}$$

where:

$$\varphi_A(x) = \frac{1}{2}\left[1 + \ln\left(\frac{1}{2} + \frac{\sqrt{1+4/x^2}}{2}\right) + \frac{2}{x} - \sqrt{1+4/x^2}\right];$$

S is source strength. In this case the radionuclide activity per volume ($\pi R^2 h$) [Bq cm^{-3}];

R is radius of the cylinder source [cm];

h is height of the cylinder source [cm].

At the surface point $P_{2,surf}$, the dose rate becomes:

$$\dot{D} = 5{,}77x10^{-4}\sum_{\gamma}\mu_a(\varepsilon_\gamma)p_\gamma\varepsilon_\gamma\frac{S}{2}\left[h_1\varphi_A\left(\frac{h_1}{R}\right) + h_2\varphi_A\left(\frac{h_2}{R}\right)\right], \tag{III-19}$$

where:

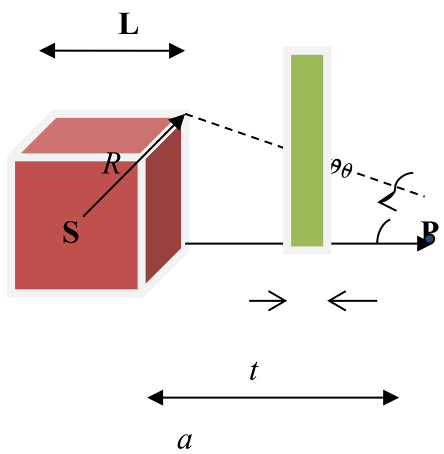
h_1 is height from point $P_{2,surf}$ to end of cylinder top [cm];

$h_2 = h - h_1$ is height to point $P_{2,surf}$ from cylinder bottom [cm].

III-2.4. Cube

III-2.4.1. With shield

In the case of a cubic volume source an approach is to look at a sphere encompassing the cube. This will overestimate the flux.



An approximation for an absorbing spherical source is to replace it by conical fragment of spherical shell having a volume equal to that of the original sphere. Given a uniformly distributed cube source with a slab shield outside the (self-absorbing) source, the dose rate $\dot{D}$ [mSv h^{-1}] at the point P of interest is approximated to [III-6]:

$$\dot{D} = 5{,}77x10^{-4} \sum_{\gamma} \mu_a(\varepsilon_\gamma) p_\gamma \varepsilon_\gamma \frac{S}{2\mu_s} \cdot \sum_i A_i \cdot \left(1 - e^{-m_s \mu_s R}\right) \cdot \left[E_2([1+\alpha_i]b_1) - \cos\varphi_0 \cdot E_2([1+\alpha_i]b_1 \sec\varphi_0)\right], \quad \text{(III-20)}$$

where:

μ_a is mass absorption coefficient [cm^2 g^{-1}];

ε_γ is photon energy [MeV];

p_γ is photon yield [-];

For nuclide data on its photon energy and yield see nuclear textbooks.

S is source strength. In this case the radionuclide activity per volume (L^3) [Bq cm^{-3}];

A_i, α_i are Taylor buildup parameters [-];

Taylor parameters for iron, lead, water and concrete can be found in *DataForExternalDoseCalc.xls*.

μ_s is linear attenuation coefficient of the source [cm^{-1}];

m_s is equivalent volume coefficient [-];

$$m_s = \left[2/(1-\cos\varphi_0) + (a/R)^3\right]^{1/3} - (a/R).$$

R is radius of the sphere $R = \dfrac{\sqrt{3}\cdot L}{2}$, where L is the length of the cube side [cm];

$E_2(x)$ is exponential integral;

$$E_n(x) = x^{n.-1}\int_x^{\infty}\frac{e^{-y}}{y^n}dy,$$

Evaluated by the function in Java: double Expint2(double x);

b_1 is shield attenuation [-];

$$b_1 = \mu t,$$

where:

μ is linear attenuation coefficient of shield [cm^{-1}];

t is thickness of shield [cm];

To get the linear attenuation coefficient μ (in cm^{-1}) just multiply the mass attenuation coefficients (in cm^2 g^{-1}) with density (g cm^{-3}). Values for iron, lead, water and concrete can be found in *DataForExternalDoseCalc.xls*.

φ_0 is angle (in radians, see figure);

$$\varphi_0 = \arcsin\left(\frac{R}{(a+R)}\right);$$

III-2.4.2. Without shield

Without shield b_1=0 and the shield buildup parameters α_i=0 (dose buildup factor =1). Then Eq. (III-20) becomes:

$$\dot{D} = 5{,}77x10^{-4} \sum_{\gamma} \mu_a(\varepsilon_\gamma) p_\gamma \varepsilon_\gamma \frac{S}{2\mu_s} \cdot \left(1 - e^{-m_s \mu_s R}\right) \cdot \left[1 - \cos\varphi_0\right] \qquad \text{(III-21)}$$

Given a uniformly distributed cube source with a non-absorbing material (*i.e.* μ_s=0) and unshielded, the dose rate $\dot{D}$ [mSv h^{-1}], at the point P_1 of interest is:

$$\dot{D} = 5{,}77x10^{-4} \sum_{\gamma} \mu_a(\varepsilon_\gamma) p_\gamma \varepsilon_\gamma \frac{S}{4a'} \left[2Ra' - (a'^2 - R^2) \ln\left(\frac{R + a'}{R - a'} \right) \right], \qquad \text{(III-22)}$$

where:

μ_a is mass absorption coefficient [cm^2 g^{-1}];

ε_γ is photon energy [MeV];

p_γ is photon yield [-];

For nuclide data on its photon energy and yield see nuclear textbooks.

S is source strength. In this case the radionuclide activity per volume (L^3) [Bq cm^{-3}];

R is radius of the sphere radius $R = \frac{\sqrt{3} \cdot L}{2}$, where L is the length of the cube side [cm];

a' is distance from center of the sphere/cube source [cm];

$a' = a + R,$

where:

a is distance from cube end [cm];

III-3. MULTIPLE SHIELDS

In the case of multiple shields the shield attenuation b_1 [-] is a sum of the optical path for the individual slabs:

$$b_1 = \sum_k \mu_k t_k,$$

The use of multiple shields (multilayers) will also affect the dose buildup. For instance, low Z material (water) produces higher fraction of scattered photons compared to high Z material (Pb) which absorbs more scattered photons.

The most simplified and conservative way of treating the buildup in multilayers is to use the product of the individual layers' buildup [III-5]:

$$B_{tot} = \prod_k B_k$$

The above formula is conservative since it does not take into account the saturation of the buildup in different layers.

A commonly applied rule for two layer shields is if low Z is closest to source ($Z_1 < Z_2$) then one can use $B_{tot} = B_2(\mu_1 t_1 + \mu_2 t_2)$, whereas if high Z is closest to source ($Z_1 > Z_2$) then $B_{tot} = B_1(\mu_1 t_1) \cdot B_2(\mu_2 t_2)$ [III-2].

Using Taylor's formula for two slabs: $B_{tot} = \sum_i \sum_j A_{1i} A_{2j} e^{-(\alpha_{1i}+\alpha_{2j})\mu T \sec\theta}$

The dose rate $\dot{D}$ [mSv h^{-1}] for the two layer shielded circular disc source (Eq. (III-8)) is then:

$$\dot{D} = 5{,}77x10^{-4} \sum_\gamma \mu_a(\varepsilon_\gamma) p_\gamma \varepsilon_\gamma \frac{S}{2} \cdot$$
$$\sum_i \sum_j A_{1i} A_{2j} \left[E_1(b_{ij}) - E_1(b_{ij}\sqrt{1 + R^2 / a^2}) \right] \qquad \text{(III-23)}$$

where:

$$b_{ij} = (1+\alpha_{1i})\mu_1 t_1 + (1+\alpha_{2j})\mu_2 t_2.$$

Adding an extra shield introduces simply an extra summation index (A_{3m}, b_{ijm}) in the above Eq. [III-5].

III-4. SCATTERING FROM VOLUME SOURCE

Multiple scattering in the volume source itself will also contribute to the total dose. In the case of a shielded volume source with the same material in source and shield Taylor coefficients could be included for the source as well. However, assuming different materials the total buildup factor should be the product of the shield buildup and source buildup (conservative and simplest approach as for multiple shields):

$$B_{tot} = B_{shield} \cdot B_{source}$$

How to solve the problem analytically could not been found in the literature. The saturation of the buildup in large volume source should decrease the dose contribution, especially in the case of a shielded source. Using Taylor form for the shield and source (a concept valid for points sufficiently deep inside source [III-5] but may be used for calculating the flux from volume sources behind a shield [III-3]):

$$B_{shield} = \sum_i A_{1i} e^{-\alpha_{1i}\mu T \sec\theta}$$

$$B_{source} = \sum_j A_{2j} e^{-\alpha_{2j}\mu_s(r-a\sec\theta)}$$

Thus, the dose rate $\dot{D}$[mSv h^{-1}] for a shielded cylindrical source at a point on the end when $h < 3/\mu_s$ is valid (Eq. (III-11)):

$$\dot{D} = 5{,}77x10^{-4}\sum_\gamma \mu_a(\varepsilon_\gamma)p_\gamma\varepsilon_\gamma \frac{S}{2}\cdot\sum_i\sum_j A_{1i}\frac{A_{2j}}{[1+\alpha_{2j}]\mu_s}\cdot$$
$$\left\{\begin{array}{l} E_2([1+\alpha_{1i}]b_1) - E_2([1+\alpha_{1i}]b_1 + [1+\alpha_{2j}]\mu_s h) + \\ \frac{1}{\sqrt{1+R^2/a^2}}\left[E_2(([1+\alpha_{1i}]b_1 + [1+\alpha_{2j}]\mu_s h)\sqrt{1+R^2/a^2}) - E_2([1+\alpha_{1i}]b_1\sqrt{1+R^2/a^2})\right]\end{array}\right\}. \tag{III-24}$$

The corresponding dose rate $\dot{D}$[mSv h^{-1}] for a shielded cylindrical source in the axial direction when $h \geq 3/\mu_s$ is valid (Eq. (III-12)):

$$\dot{D} = 5{,}77x10^{-4}\sum_\gamma \mu_a(\varepsilon_\gamma)p_\gamma\varepsilon_\gamma \frac{S}{2}\cdot$$
$$\sum_i\sum_j A_{1i}\frac{A_{2j}}{[1+\alpha_{2j}]\mu_s}\cdot\left[E_2([1+\alpha_{1i}]b_1) - \frac{1}{\sqrt{1+R^2/a^2}}E_2([1+\alpha_{1i}]b_1\sqrt{1+R^2/a^2})\right]. \tag{III-25}$$

The dose rate $\dot{D}$[mSv h^{-1}] for a cylindrical source (index *s* stands for source in the equation below) without a shield at a point in the axial direction (Eq. (III-13)):

$$\dot{D} = 5{,}77x10^{-4}\sum_\gamma \mu_a(\varepsilon_\gamma)p_\gamma\varepsilon_\gamma \frac{S}{2}\sum_s\frac{A_s}{[1+\alpha_s]\mu_s}\cdot$$
$$\left\{1 - E_2([1+\alpha_s]\mu_s h) + \frac{1}{\sqrt{1+R^2/a^2}}\left[E_2([1+\alpha_s]\mu_s h\sqrt{1+R^2/a^2}) - 1\right]\right\}. \tag{III-26}$$

Finally, including the source buildup in the case with a shielded cubic source the dose rate $\dot{D}$[mSv h^{-1}] would change into (Eq. (III-20)):

$$\dot{D} = 5{,}77x10^{-4}\sum_\gamma \mu_a(\varepsilon_\gamma)p_\gamma\varepsilon_\gamma \frac{S}{2\pi}\cdot$$
$$\sum_i\sum_j A_{1i}\frac{A_{2j}}{[1+\alpha_{2j}]\mu_s}\cdot F\left([1+\alpha_{1i}]b_1, [1+\alpha_{2j}]\mu_s, R, h, a\right). \tag{III-27}$$

For the corresponding unshielded cubic source the dose rate $\dot{D}$[mSv h^{-1}] becomes (Eq. (III-21)):

$$\dot{D} = 5{,}77x10^{-4} \sum_{\gamma} \mu_a(\varepsilon_\gamma) p_\gamma \varepsilon_\gamma \frac{S}{2} \sum_{s} \frac{A_s}{[1+\alpha_s]\mu_s} \cdot \left(1 - e^{-m_s[1+\alpha_s]\mu_s R}\right) \cdot \left[1 - \cos\varphi_0\right]$$

(III-28)

III-5. MEASURED DOSE RATE

It is also possible to make use of a measured dose rate at a distance from the source to find out the dose rate at other points. Assuming the radionuclide of the source is specified, Eq. (III-2) will give the source term S [Bq cm^{-n}, n=0,1,2,3]. Then this source term can be used in the calculation of flux, and later the dose rate, at a different distance from source.

For example, in the case of a circular disc source (neglecting buildup, and assuming no shield in between source and measuring point with $d = 0$):

$$S = \frac{5{,}77x10^{-4} \sum_{\gamma} \mu_a(\varepsilon_\gamma) p_\gamma \varepsilon_\gamma}{\dot{D}} \frac{4}{\ln\left(1 + R^2 / a^2\right)} \quad \text{(III-29)}$$

where:

$\dot{D}$ is measured dose rate [mSv h^{-1}];

μ_a is mass absorption coefficient [cm^2 g^{-1}];

For example, for water the value is 0,031 cm^2 g^{-1} at 1 MeV [III-1];

ε is photon energy [MeV];

p_γ is photon yield [-];

For nuclide data on its photon energy and yield, see nuclear textbooks.

R is radius of the disc source [cm];

a is distance from source to point P_1 [cm];

Finally when the source term is known, Eq. (III-8) can be use to find out the dose rate at a new distance a'. In the presence of a shield between source and the new point, Eq. (III-10) can be used instead.

III-6. EXAMPLES

III-6.1. Point Source

Calculate the dose rate for 10 MBq of Co-60 in a point source. With and without a slab concrete shield of thickness 20 cm ($t = 20$ cm). Measuring point P_1 200 cm (on axis) and P_2 200 cm (100 cm off axis) ($a = 200$ cm, $d = 100$ cm).

Solution:

From [III-3]:

Co-60: $E_{\gamma 1}$ 1,17 MeV (99,9%), $E_{\gamma 2}$ 1,33 MeV (100%).

From [III-1]:

$\frac{\mu}{\rho} = 5{,}496x10^{-2}\, cm \cdot g^{-1}$ Concrete (Al) material at 1 MeV, $\rho = 2{,}35\, g \cdot cm^{-3}$.

III-6.1.1. With shield

Buildup (Eq. (III-4):

$B_D \approx 4{,}167$

GP parameters for concrete material at 1 MeV: b=2,005 , c=1,350,a=-0,06906, X_k=15.,69, d=0,02349 (data taken from Radiological Toolbox).

At P_1

Use Eq. (III-7).

$$\phi^0 = \frac{10^7 \cdot e^{-\left(5{,}496x10^{-2} \cdot 2{,}35 \cdot 20\right)}}{4\pi \cdot 200^2} \approx 1{,}5 \text{ photons cm}^{-2}\text{ s}^{-1}.$$

$$\dot{D} = 5{,}77x10^{-4} \cdot 0{,}031 \cdot (1{,}17 \cdot 0{,}999 + 1{,}33) \cdot 4{,}167 \cdot 1{,}5 \approx 0{,}28\ \mu\text{Sv h}^{-1}.$$

At P_2

$$\sec\theta = \frac{\sqrt{d^2 + a^2}}{a} = \frac{\sqrt{5}}{2}$$

$$\phi^0 = \frac{10^7 \cdot e^{-\left(5{,}496x10^{-2} \cdot 2{,}35 \cdot 20 \cdot \sqrt{5}/2\right)}}{4\pi \cdot 200^2 \cdot 5/4} \approx 0{,}89 \text{ photons cm}^{-2}\text{ s}^{-1}.$$

$$\dot{D} = 5{,}77x10^{-4} \cdot 0{,}031 \cdot (1{,}17 \cdot 0{,}999 + 1{,}33) \cdot 4{,}167 \cdot 0{,}89 \approx 0{,}17\ \mu\text{Sv h}^{-1}.$$

III-6.1.2. Without shield

At P_1

Use Eq. (III-7).

$$\phi^0 = \frac{10^7}{4\pi \cdot 200^2} \approx 19{,}9 \text{ photons cm}^{-2}\text{ s}^{-1}.$$

$$\dot{D} = 5{,}77x10^{-4} \cdot 0{,}031 \cdot (1{,}17 \cdot 0{,}999 + 1{,}33) \cdot 19{,}9 \approx 89\ \mu\text{Sv h}^{-1}.$$

At P_2

$$\sec\theta = \frac{\sqrt{d^2 + a^2}}{a} = \frac{\sqrt{5}}{2}$$

$$\phi^0 = \frac{10^7}{4\pi \cdot 200^2 \cdot 5/4} \approx 15{,}9 \text{ photons cm}^{-2}\text{ s}^{-1}.$$

$$\dot{D} = 5{,}77x10^{-4} \cdot 0{,}031 \cdot (1{,}17 \cdot 0{,}999 + 1{,}33) \cdot 15{,}9 \approx 71\ \mu\text{Sv h}^{-1}.$$

III-6.2. Disc Source

Calculate the dose rate for 10 MBq of Co-60 in a disc source. With and without a slab concrete shield of thickness 20 cm ($t = 20$ cm), disc source with radius 50 cm ($R = 50$ cm). Measuring point P_1 200 cm (on axis) and P_2 200 cm (100 cm off axis) ($a = 200$ cm, $d = 100$ cm).

Solution:

From [III-3]:

Co-60: $E_{\gamma 1}$ 1,17 MeV (99,9%), $E_{\gamma 2}$ 1,33 MeV (100%).

From [III-1]:

$\frac{\mu}{\rho} = 5{,}496x10^{-2}\, cm \cdot g^{-1}$ Concrete (Al) material at 1 MeV, $\rho = 2{,}35\, g \cdot cm^{-3}$.

III-6.2.1. With shield

Taylor dose buildup parameters for concrete material at 1 MeV: A_1=53,82 , A_1=1- A_2, α_1= -0,0439, α_2 =-0,0174 (data taken from [III-2]).

At P_1

Use Eq. (III-8).

$$E_1([1-\alpha_1]\cdot b_1) = E_1([1+0{,}0439]\cdot 5{,}496x10^{-2}\cdot 2{,}35\cdot 20) = E_1(2{,}47) \approx 0{,}026$$

$$E_1\left([1-\alpha_1]\cdot b_1\cdot\sqrt{1+\frac{R^2}{a^2}}\right) = E_1\left(1{,}04\cdot 2{,}58\cdot\sqrt{1+\frac{50^2}{200^2}}\right) = E_1(2{,}54) \approx 0{,}024$$

$$E_1([1-\alpha_2]\cdot b_1) = E_1([1+0{,}0174]\cdot 5{,}496x10^{-2}\cdot 2{,}35\cdot 20) = E_1(2{,}54) \approx 0{,}024$$

$$E_1\left([1-\alpha_2]\cdot b_1\cdot\sqrt{1+\frac{R^2}{a^2}}\right) = E_1\left(1{,}02\cdot 2{,}58\cdot\sqrt{1+\frac{50^2}{200^2}}\right) = E_1(2{,}62) \approx 0{,}021$$

$$\hat{\phi} = \frac{10^7}{2\cdot\pi\cdot 50^2}\left[53{,}82\cdot(0{,}026-0{,}024) - 52{,}82\cdot(0{,}024-0{,}021)\right] \approx 4{,}9 \text{ photons cm}^{-2}\text{ s}^{-1}.$$

Note: The above expression is not a 'real' flux since the buildup factor is only valid for dose!

$$\dot{D} = 5{,}77x10^{-4}\cdot 0{,}031\cdot(1{,}17\cdot 0{,}999+1{,}33)\cdot 4{,}9 \approx 0{,}22\ \mu\text{Sv h}^{-1}.$$

At P_2

If $K \approx 1$ in Eq. (III-6), the dose rate is the same as in P_1. Using the table in [III-3], K can be estimated to 0,1, thus lowering the dose rate by a factor of 10.

III-6.2.2. Without shield

At P_1

$d = 0$:

$$\phi^0 = \frac{10^7}{4\cdot\pi\cdot 50^2}\ln\left(1+\frac{50^2}{200^2}\right) \approx 19{,}3 \text{ photons cm}^{-2}\text{ s}^{-1}.$$

From Eq.(III-10):

$$\dot{D} = 5{,}77x10^{-4}\cdot 0{,}031\cdot(1{,}17\cdot 0{,}999+1{,}33)\cdot 19{,}3 \approx 0{,}86\ \mu\text{Sv h}^{-1}.$$

At P_2

$$\phi^0 = \frac{10^7}{4 \cdot \pi \cdot 50^2} \cdot$$
$$\ln\left(\frac{1}{2 \cdot 200^2} \left[200^2 + 50^2 - 100^2 + \sqrt{50^4 + 2 \cdot 50^2 \cdot \left(200^2 - 100^2\right) + \left(200^2 + 100^2\right)^2} \right] \right)$$

$\approx 15{,}7$ photons cm^{-2} s^{-1}.

From Eq. (III-10):

$$\dot{D} = 5{,}77x10^{-4} \cdot 0{,}031 \cdot (1{,}17 \cdot 0{,}999 + 1{,}33) \cdot 15{,}7 \approx 0{,}70\ \mu \text{ Sv h}^{-1}.$$

III-6.3. Drum

Calculate the dose rate for 100 MBq of Sr-85 in a drum filled with water. With and without a slab concrete shield of thickness 10 cm (t). Drum radius 40 cm ($R = 40$ cm), height 100 cm ($h = 100$ cm),. Measuring point on end of drum axis P_1 200 cm away from drum end ($a = 200$ cm).

On side of drum axis P_2 400 cm away from drum side (100 cm from drum bottom) ($a = 400$ cm, $h_2 = 100$ cm).

Solution:

From [III-3]:

Sr-85: $E_{\gamma 1}$ 0,513 MeV (100%)

From [III-1]:

$\frac{\mu_s}{\rho} = 9{,}687x10^{-2}\, cm \cdot g^{-1}$ Water at 0,5 MeV, $\rho = 1\, g \cdot cm^{-3}$, $\frac{\mu_a}{\rho} = 3{,}299x10^{-2}\, cm \cdot g^{-1}$.

$\frac{\mu}{\rho} = 5{,}496x10^{-2}\, cm \cdot g^{-1}$ Concrete (Al) material at 1 MeV, $\rho = 2{,}35\, g \cdot cm^{-3}$.

III-6.3.1. With shield

Taylor dose buildup parameters for concrete material at 0,5 MeV: A_1=68,606 , A_1=1- A_2, α_1= -0,0685, α_2 =-0,0378 (data taken from [III-2]).

At P_1, on end

Since $\mu_s h \approx 10 \geq 3$ use Eq. (III-12) (neglecting source buildup).

$$E_2([1+\alpha_1]\cdot b_1)=E_2(0{,}9315\cdot 5{,}496x10^{-2}\cdot 2{,}35\cdot 10)=E_2(1{,}203)\approx 0{,}1106.$$

$$E_2\left([1+\alpha_1]\cdot b_1\sqrt{1+\frac{R^2}{a^2}}\right)=E_2\left(0{,}9315\cdot 1{,}2916\cdot\sqrt{1+\frac{40^2}{200^2}}\right)=E_2(1{,}296)\approx 0{,}0970.$$

$$E_2([1+\alpha_2]\cdot b_1)=E_2(0{,}9622\cdot 5{,}496x10^{-2}\cdot 2{,}35\cdot 10)=E_2(1{,}243)\approx 0{,}1046.$$

$$E_2\left([1+\alpha_2]\cdot b_1\sqrt{1+\frac{R^2}{a^2}}\right)=E_2\left(0{,}9622\cdot 1{,}2916\cdot\sqrt{1+\frac{40^2}{200^2}}\right)=E_2(1{,}338)\approx 0{,}0914.$$

$$\hat{\phi}=\frac{10^8}{\pi\cdot 40^2\cdot 100\cdot 2\cdot 9{,}687x10^{-2}}\begin{pmatrix}68{,}606\cdot[0{,}1106-0{,}9285\cdot 0{,}0970]- \\ 67{,}606\cdot[0{,}1046-0{,}9285\cdot 0{,}0914]\end{pmatrix}\approx 76{,}7 \text{ photons cm}^{-2}\text{ s}^{-1}.$$

Note: The above expression is not a ‘real’ flux since the buildup factor is only valid for dose!

$$\dot{D}=5{,}77x10^{-4}\cdot 0{,}033\cdot(0{,}513)\cdot 76{,}7\approx 0{,}75\ \mu\text{Sv h}^{-1}$$

At P_2, on side

Use Eq. (III-16).

$$F([1+\alpha_1]b_1,\mu_s,R,h,a)=F(1.203{,}9.687x10^{-2}{,}40{,}100{,}400)\approx 0{,}0065.$$

$$F([1+\alpha_2]b_1,\mu_s,R,h,a)=F(1.243{,}9.687x10^{-2}{,}40{,}100{,}400)\approx 0{,}0063.$$

$$\hat{\phi}=\frac{10^8}{\pi\cdot 40^2\cdot 100\cdot 2\cdot\pi\cdot 9{,}687\cdot 10^{-2}}\cdot[68{,}606\cdot 0{,}0065-67{,}606\cdot 0{,}0063]\approx 6{,}54 \text{ photons cm}^{-2}\text{ s}^{-1}.$$

Note: The above expression is not a ‘real’ flux since the buildup factor is only valid for dose!

$$\dot{D}=5{,}77x10^{-4}\cdot 0{,}033\cdot(0{,}513)\cdot 6{,}54\approx 0{,}064\ \mu\text{Sv h}^{-1}.$$

III-6.3.2. Without shield

At P_1, on end

Use Eq. (III-13).

$$E_2(\mu_s h)=E_2(9{,}687x10^{-2}\cdot 100)=E_2(9{,}687)\approx 5{,}3817x10^{-6}.$$

$$E_2\left(\mu_s h\sqrt{1+\frac{R^2}{a^2}}\right)=E_2\left(9{,}687\cdot\sqrt{1+\frac{40^2}{200^2}}\right)=E_2(9{,}8788)\approx 4{,}3689x10^{-6}.$$

$$\phi^0 = \frac{10^8}{\pi \cdot 40^2 \cdot 100 \cdot 2 \cdot 9{,}687x10^{-2}}\left(1 - 5{,}3817x10^{-6} + \frac{1}{\sqrt{1+\frac{40^2}{200^2}}}\left[4{,}3689x10^{-6} - 1\right]\right) \approx 19{,}9$$

ph. cm^{-2} s^{-1}.

$\dot{D} = 5{,}77x10^{-4} \cdot 0{,}033 \cdot (0{,}513) \cdot 19{,}9 \approx 0{,}19$ μ Sv h^{-1}.

At P_2, on side:

Use Eq. (III-17):

$$F(0, \mu_s, R, h, a) = F(0{,}9.687x10^{-2}{,}40{,}100{,}400) \approx 2{,}20 \cdot 10^{-2}.$$

$$\phi^0 = \frac{10^8}{\pi \cdot 40^2 \cdot 100 \cdot 2 \cdot \pi \cdot 9{,}687 \cdot 10^{-2}} \cdot 2.2 \cdot 10^{-2} \approx 7{,}19 \text{ photons cm}^{-2}\text{ s}^{-1}.$$

$\dot{D} = 5{,}77x10^{-4} \cdot 0{,}033 \cdot (0{,}513) \cdot 7{,}19 \approx 0{,}070$ μ Sv h^{-1}.

III-6.4. Cube

Calculate the dose rate for 100 MBq of Sr-85 in a cube filled with water. With and without a slab steel shield of thickness 10 cm (t). Cube side 100 cm ($L = 100$ cm). Measuring point on end of drum axis P_1 200 cm away from drum end ($a = 200$ cm).

Solution:

From [III-3]:

Sr-85: $E_{\gamma 1}$ 0,513 MeV (100%)

From [III-1]:

$\frac{\mu_s}{\rho} = 9{,}687x10^{-2}\, cm \cdot g^{-1}$ Water at 0,5 MeV, $\rho = 1\, g \cdot cm^{-3}$, $\frac{\mu_a}{\rho} = 3{,}299x10^{-2}\, cm \cdot g^{-1}$.

$\frac{\mu}{\rho} = 8{,}414x10^{-2}\, cm \cdot g^{-1}$ Steel (Fe) material at 0,5 MeV, $\rho = 7{,}87 g \cdot cm^{-3}$.

$$R = \frac{\sqrt{3} \cdot 100}{2} = \sqrt{3} \cdot 50$$

III-6.4.1. With shield

Taylor dose buildup parameters for steel (Fe) material at 0,5 MeV: A_1=31,379 , A_1=1- A_2, α_1 = -0,0684, α_2 =-0,0374 (data taken from [III-3]).

At P_1

Use Eq. (III-20) (neglecting source buildup).

$$\mu_s R = 9{,}7x10^{-2} \cdot \sqrt{3} \cdot 50 \approx 8{,}4$$

$$\varphi_0 = \arcsin\left(\frac{\sqrt{3} \cdot 50}{200 + \sqrt{3} \cdot 50} \right) \approx 17{,}6^\circ = 0{,}307$$

$$E_2([1+\alpha_1]b_1) = E_2(0{,}9316 \cdot 8{,}414x10^{-2} \cdot 7{,}87 \cdot 10) = E_2(6{,}169) \approx 2{,}630x10^{-4}.$$

$$\cos\varphi_0 \cdot E_2([1+\alpha_1]b_1 \sec\varphi_0) = \cos\varphi_0 \cdot E_2(6{,}169 \cdot 1{,}05) = 0{,}9533 \cdot 1{,}8706x10^{-4} \approx 1{,}783x10^{-4}.$$

$$E_2([1+\alpha_2]b_1) = E_2(0{,}9626 \cdot 8{,}414x10^{-2} \cdot 7{,}87 \cdot 10) = E_2(6{,}374) \approx 2{,}087x10^{-4}.$$

$$\cos\varphi_0 \cdot E_2([1+\alpha_2]b_1 \sec\varphi_0) = \cos\varphi_0 \cdot E_2(6{,}374 \cdot 1{,}05) = 0{,}9533 \cdot 1{,}4692x10^{-4} \approx 1{,}401x10^{-4}.$$

$$m_s = \left[2/(1-\cos\varphi_0) + \left(\frac{200}{\sqrt{3} \cdot 50} \right)^3 \right]^{1/3} - \frac{200}{\sqrt{3} \cdot 50} \approx 1{,}4959.$$

$$\hat{\phi} = \frac{10^8}{100^3 \cdot 2 \cdot 9{,}7x10^{-2}} \cdot \left(1 - e^{-1{,}49599{,}7x10^{-2} \cdot \sqrt{3} \cdot 50}\right).$$
$$\left[31{,}379 \cdot \left(2{,}630x10^{-4} - 1{,}783x10^{-4}\right) - 30{,}379 \cdot \left(2{,}087x10^{-4} - 1{,}401x10^{-4}\right)\right]$$

$\approx 0{,}295$ photons cm^{-2} s^{-1}.

Note: The above expression is not a 'real' flux since the buildup factor is only valid for dose!

$$\dot{D} = 5{,}77x10^{-4} \cdot 0{,}033 \cdot 0{,}513 \cdot 0{,}295 \approx 2{,}9x10^{-3} \ \mu \text{Sv h}^{-1}.$$

III-6.4.2. Without shield

At P_1

From Eq. (III-21):

$$\phi^0 = \frac{10^8}{100^3 \cdot 2 \cdot 9{,}7x10^{-2}} \cdot \left(1 - e^{-1{,}49599{,}7x10^{-2} \cdot \sqrt{3} \cdot 50}\right) \cdot (1 - 0{,}9533) \approx 24{,}1 \text{ photons cm}^{-2} \text{ s}^{-1}.$$

$$\dot{D} = 5{,}77x10^{-4} \cdot 0{,}033 \cdot 0{,}513 \cdot 24{,}1 \approx 0{,}24 \ \mu \text{Sv h}^{-1}.$$

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