Authority for Nuclear Safety and
Radiation Protection

ANVS Guide on Level 3 PSA

Contents

Preface

Licensees and licence applicants for a nuclear facility in The Netherlands are under the obligation to perform a risk analysis. That risk analysis is a Probabilistic Safety Assessment, or PSA, which is customarily subdivided into three levels, i.e. a level 1, level 2, and level 3 PSA. A level 1 PSA calculates the probability of core melt. A level 2 PSA calculates the probability of releases following core melt and the associated characteristics of those releases. A level 3 PSA, the subject of this Guide, determines the external consequences on the basis of the accident releases determined in the level 2 PSA.

The IAEA has published guides for level 1 PSA (SSG-3 [1]) and level 2 PSA (SSG-4 [2]) in 2010. No guide is available for level 3 PSA (the IAEA no longer supports the IAEA Safety Series 50-P-12 [3] published in 1996), and the Dutch Level 3 PSA Guidelines dates from 1993 [4]. Therefore it was decided to update the 1993 Level 3 PSA Guidelines to take account of the latest insights and to incorporate the latest developments in legislation on radiological requirements for nuclear facilities.

This Level 3 PSA Guide replaces the Level 3 PSA Guidelines published in 1993. The changes concern mainly: the adjustment to the prevailing dose limit values, adjustment to the prevailing risk factor, explanation of a number of steps in the computational methods, a discussion of the atmospheric dispersion model SRM-3 (part of the New National Model, NNM [5]), and a discussion of the implications of the development of the Guidelines on the Safe Design and Operation of Nuclear Reactors (usually referred to as Dutch Safety Requirements, DSR Guide [6]).

This document is a guide. It does not impose any binding requirements on performing a level 3 PSA, but includes recommendations for meeting the existing legal requirements when performing a level 3 PSA. It is not intended to pre-empt the use of equivalent new or alternative methods. This guide has a similar function as the above mentioned IAEA SSG-3 and SSG-4 and other guides from the IAEA safety standards series.

Legal framework and policy documents

In this Guide on level 3 PSA, frequently reference is made to the existing legal framework and policy instruments developed for the assessment of nuclear licenses:

Status	Reference
Act	
Nuclear Energy Act	Kew [7]
Decrees	
Decree on Nuclear Facilities, Fissionable Materials and Ores	Bkse Decree [8]
Decree on basic safety standards in radiation protection	Bbs Decree [9]
Regulation	
ANVS Regulation on basic safety standards in radiation protection	Vbs Regulation [10]
Guidelines	
Calculation of the dose in the environment on granting an ionizing radiation license	
A. Releases to the atmosphere and water	DOVIS-A [11]
B. External Radiation	DOVIS-B [12]
New National Model – Model for the dispersion of air pollution from sources over short distances	NNM [5]
Guide	
Guidelines on the Safe Design and Operation of Nuclear Reactors	DSR Guide [6]

1 Introduction

1.1 Legal framework

The Dutch Nuclear Energy Act (Kew, [7]) prohibits the following without a licence: the construction, commissioning, operation, alteration or decommissioning of a facility where nuclear energy can be released, fissile materials can be produced, treated or processed, or where fissile materials are stored or the dismantling of a facility where nuclear energy can be released, fissile materials can be produced, treated or processed or where fissile materials are stored[1].

The Decree on Nuclear Facilities, Fissionable Materials and Ores (Dutch: Besluit Kerninstallaties, Splijtstoffen en Ertsen, Bkse) [8], in Chapter II, § 3, Article 6, paragraph 11, under a, lays down more detailed specifications of the information that shall be included in applications for a licence. Several articles prescribe that a risk analysis shall be included in the application. Moreover, a risk analysis of beyond design basis accidents is prescribed for specific types of nuclear facilities, such as nuclear power plants (Article 6) and facilities in which fissile materials (containing plutonium or enriched uranium) or spent fuel can be produced, treated or processed (Article 7) or stored (Article 8) (risk analysis of beyond design basis accidents as referred to in Article 6, paragraph 1, under i). Article 18, paragraph 3, of the Bkse Decree stipulates that a licence for a nuclear facility may be refused if the results from the risk analysis of beyond design basis accidents do not meet the prescribed individual risk (a) and group risk (b) limit values:

a. a probability of 10^{-6} per year that a person residing permanently and unprotected outside the relevant facility will die as a result of a beyond design basis accident;
b. a probability of 10^{-5} per year that a group of at least ten persons present outside the relevant facility will be direct fatalities of a beyond design basis accident, or a n^2 times smaller probability for n times more direct fatalities.

A risk analysis of beyond design basis accidents in which the results shall be tested against the limit values prescribed above is performed on the basis of the level 3 PSA specified in this Guide.

The Bkse Decree also stipulates that a licence shall be refused when the dose criteria for design basis accidents are not met. Although that calculation is not, strictly speaking, a level 3 PSA, the calculation

[1] The Dutch Nuclear Energy Act also prohibits the preparation, transportation, possession, application, disposal, imports into or exports out of The Netherlands or causing such imports into or exports out of The Netherlands of defined radioactive materials without a licence. In practice, this relates to hundreds of actions and equipment each year for which licences are arranged via the Bbs Decree [9]. Nuclear facilities and other facilities with fissile materials and ores must also comply with the requirements imposed by the Bkse Decree [8].

method and computer codes are virtually identical. For that reason, the present Guide also considers dose calculations for design basis accidents.

Article 3a (paragraph 2) of the Bkse Decree stipulates that the government can lay down rules for the manner in which a risk analysis (such as a PSA) shall be performed. For the determination of the necessary environmental dose equivalents and the equivalent and effective doses, the ANVS Regulation on Basic Safety Standards for Radiation Protection (Vbs) [10] applies the *Computation rules for analysis of the consequences of ionizing radiation* (AGIS) for what is referred to as the 'closer examination' (Dutch: nadere analyse) laid down in Annex 10 accompanying Article 4.37-4.39 of the ANVS Regulation (Vbs) [10][2].

The ANVS Regulation (Vbs) refers for the 'closer examination' of releases to the atmosphere and water to the report *Calculation of the dose in the environment on granting an ionizing radiation licence – Part A: releases to the atmosphere and water* (DOVIS-A, [11]). That report discusses the calculation of the dispersion and deposition of radioactive materials in the environment and the calculation of the resulting doses. DOVIS-A prescribes the hour-by-hour method of the New National Model (NNM, [5]) as the standard for calculation of the dispersion of radioactive material in the atmosphere. The NNM can be used to model the dispersion of air pollution, for example within the context of environmental permits. The present Level 3 PSA Guide is in line with the models used in DOVIS-A and, for releases to the atmosphere, the NNM.

The use of the computation rules for the 'closer examination' of routine emissions is required by law. The use of those computation rules is not prescribed for the calculation of doses and probabilities for testing against the Bkse Decree criteria, in part because some of those rules are not suitable for application in accident analyses. This received a great deal of attention during the preparations of the present Guide, and where necessary computation rules for level 3 PSA have been formulated that fit in as closely as possible with the original computation rules. These computation rules are discussed in depth in Chapter 5, detailed description of models.

In 2015 the Dutch government published the DSR Guide [6]. The DSR Guide sets out safety requirements and conditions for the safe design and operation of light water-cooled nuclear reactors. The DSR Guide provides insight into the best available technology[3] for the safety of new nuclear reactors available in 2015. The DSR Guide also offers a reference framework for existing reactors within the context of the latest relevant nuclear safety developments and insights. Insight into the best available technology is also important for new reactors, since pursuant to Article 15b, second paragraph, of the Dutch Nuclear Energy Act, licence applications for the construction of a nuclear reactor of an outdated type can be refused, even when all other requirements are met. This insight is

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[2] The Vbs Regulation [10] abbreviates the term 'ionizing radiation' to 'radiation'.

[3] The 'best available technology' concept is explained in more detail in the definitions of the *Guide on continual improvement of nuclear safety* (Handreiking continu verbeteren van de nucleaire veiligheid, see www.anvs.nl).

of importance to holders of an existing licence as they are required to regularly test their facility against the best available technology. The ‘as far as reasonably possible’ criterion is then applicable to the implementation of any necessary measures.

Most of the DSR Guide [6] addresses specific requirements on the hardware of nuclear facilities, such as the redundancy of core cooling systems and the building’s ability to withstand external influences.

The present DSR Guide devotes more attention to the design basis for new reactors than before: the design basis now includes postulated initial events with multiple failure and postulated core-melt accidents, which were included as beyond design basis accidents in the Bkse Decree. The DSR Guide also imposes more stringent requirements on the radiological objectives than the dose limits stipulated in Article 18, paragraph 2, of the Bkse Decree. The DSR Guide does not include further details on beyond design basis accidents and the associated residual and other risk analyses, such as PSA and level 3 PSA: these details are available in this Level 3 PSA Guide and the aforementioned IAEA guides ([1], [2]).

Although the DSR Guide is not a Ministerial Regulation and, consequently, does not prescribe any legally binding requirements, assessments of licence applications will be based on the safety requirements and conditions prescribed by the DSR Guide. This is because the DSR Guide provides insight into the best technology currently available for the design and operation of new reactors to achieve the best possible safety level. More information about the DSR requirements is enclosed in Section 3.7, design basis accidents.

1.2 Risk analysis and level 3 PSA

The legal framework (Bkse Decree [8]) prescribes how risk analyses for specific types of nuclear facilities have to be performed. That risk analysis can be carried out in the form of a full Probabilistic Safety Assessment, PSA. A distinction can be made between three PSA levels.

1. Level l PSA: analyses initiating event frequencies and the reliability of the systems to determine the total probability of the accident scenarios that result in the materialization of a core damage scenario. This part of the PSA assesses mainly the functional safety of the nuclear power plant’s safety systems.
2. Level 2 PSA: determines the probability of the failure of the reactor containment building and the characteristics of the associated accident releases (the source term). Analyses of thermohydraulic and chemical processes play an important role in a level 2 PSA.

3. Level 3 PSA: determines the external consequences of the spectrum of accident releases. This level is the subject of the present Guide[4].

Table 1-1 PSA levels 1, 2 and 3.

Level 1 PSA	Level 2 PSA	Level 3 PSA
Identification of scenarios that result in core damage	Progression of the various system states during an accident	Radiological consequences of accident emissions
Result: description of the system states at onset of core damage & the probability of those states **(Core Damage Frequency)**	**Result:** extent of system damage & amount of released radioactive material from various system components & probabilities **(Source Term Spectrum)**	**Result:** Doses and risks

1.3 Safety analysis and design basis accidents

The legal framework (Bkse Decree [8]) requires that an application for a licence shall include a safety report containing a description of the measures and countermeasures that the applicant will implement to prevent damage or mitigate the risk of damage, including measures to prevent damage outside the facility during normal operations and countermeasures to prevent damage resulting from the postulated initiating events mentioned in that description, as well as a risk analysis of the damage outside the facility caused by those events. An application for a licence shall also include a risk analysis of the damage outside the facility as a consequence of beyond design basis accidents.

The analysis of the postulated initiating events is often referred to as a 'Design Basis Accidents analysis' or 'deterministic analysis', in contrast to a PSA that is referred to as a 'probabilistic analysis'.

The aforementioned safety report shall demonstrate that the dose limit values prescribed in Article 18, paragraph 2, of the Bkse Decree are not exceeded. The present Guide also addresses dose calculation for design basis accidents.

[4] The official IAEA definition of level 3 PSA encompasses the entirety of the above descriptions of level 1, 2, and 3 PSA. That definition is seldom used in practice, and rarely results in confusion.

Table 1-2 Dose limit values for design basis accidents pursuant to Article 18, paragraph 2, of the Bkse Decree [8]*

Event frequency F per year	Maximum effective dose allowed		Maximum thyroid dose allowed
	persons 16 years and over	persons under the age of 16	
$F \geq 10^{-1}$	0.1 mSv	0.04 mSv	500 mSv
$10^{-1} > F \geq 10^{-2}$	1 mSv	0.4 mSv	500 mSv
10-2 > $F \geq 10^{-4}$	10 mSv	4 mSv	500 mSv
$F < 10^{-4}$	100 mSv	40 mSv	500 mSv

* The government proposes even more stringent dose limit values in the DSR Guide [6], see **Table 3-7**.

1.4 Substantiating studies

The following approach was adopted in the preparation of the present Guide:

1. The structure, text, models, and data are taken from the 1993 Level 3 PSA Guidelines [4] whenever possible.
2. The text and criteria are amended in line with the prevailing Bkse Decree [8] and other government guidelines, such as the DSR Guide [6].
3. The definitions, models, and data are brought in line with the rules laid down in the Vbs Regulation [10], DOVIS-A [11] and NNM [5].
4. The new Guide remains in line with the international approach and the internationally accepted computational tools.

Adopting this approach required a number of substantiating studies, also referred to as 'topical studies'.

Topical study 1: Application of the NNM to accident emissions

The most extensive study was topical study 1: Application of the NNM SRM-3 model for accident emissions [13]. The topical report of that study reviewed the backgrounds of the models adopted for the NNM and, in particular, the manner in which those models need to be applied for level 3 PSA. As the NNM does not always offer sufficient clarity, additional directions have been included for the application of the models for level 3 PSA.

Topical study 2: Application of DOVIS-A to accident emissions

The topical report of study 2: Application of DOVIS-A [11] to accident emissions [14] mapped the implications of the use of DOVIS-A. As DOVIS-A is focused on continuous releases and limited to calculations of the effective dose based on the adult male focus group, the topical report supplemented the DOVIS-A model parameters with some data and suitable references required for level 3 PSA. These relate, in particular, to additional information for the calculation of organ doses for deterministic effects and for the calculation of doses and consequences for children as focus group.

Topical study 3: Application of international level 3 PSA models

The COYSMA and MACCS2 computer programs were (almost completely) compatible with the Level 3 PSA Guidelines published in 1993 [4] by default. Topical study 3 [15] documented the extent to which the models used in COSYMA and MACCS are in agreement with the standard level 3 PSA models adopted for the present Guide.

Topical study 4: Bkse Decree and DSR design basis accidents

Both the Bkse Decree [8] and DSR Guide [6] state dose limit values that may not be exceeded in design basis accidents. The level 3 PSA calculation method yields a statistical distribution of the doses due to an accident. Topical study 4 examined how this statistical distribution can be used to demonstrate that the dose limit values are not exceeded. The results are included in the present Guide.

Topical study 5: International application of level 3 PSA and development of standards

Under the auspices of the IAEA, in 2014 an initiative was launched to develop a guide on performance of level 3 PSA along the same lines as the previously developed guides for level 1 [1] and level 2 [2] PSA. Also a Dutch expert[5] took part in the initiative for a new level 3 guide, an update of [3] in the form of a new Specific Safety Guide (SSG) 'Development and Application of Level 3 Probabilistic Safety Assessment for Nuclear Power Plants'. However, the project was put on hold in 2015, and the IAEA has no further funds available for continuation of that project.

A number of elements of the incomplete draft of the SSG have been included in the present Guide, e.g. the summary of a level 3 PSA, the motivation for performing a level 3 PSA, and descriptions of the models on a conceptual level.

[5] J.B. Grupa of NRG, on IAEA's invitation.

1.5 The present Guide

1.5.1 Objective of the present Guide

This Guide has been drawn up to promote uniformity in performance of a level 3 PSA by licensees and licence applicants, enabling more effective comparisons of the results. The objective of this Guide is also to specify the terms, conditions, standards, and legal requirements for the calculation of the dose consequences that are needed to promote equal treatment of the various licensees and licence applicants.

The actual calculations are performed using computer codes that incorporate the dispersion models. The 1993 Guidelines [4] devoted specific attention to the internationally available COSYMA [16] and MACCS [17] computer codes and the manner in which they can be used in the level 3 PSA of a nuclear facility in The Netherlands. Meanwhile, the computational models determined by the Vbs Regulation [10], DOVIS-A [11] and NNM [5] have been incorporated in the NUDOS2 [18] computer code. That computer code can be used to calculate the dose from routine airborne emissions as well as the dose and risks resulting from accident emissions with a – presently still – simple emission progression.

Although the emphasis is mainly on nuclear power plants, this Guide can also be used to estimate the risks of accidents at other nuclear facilities with different kinds of source terms. Such source terms may differ, for example, in radionuclide composition, in physical and chemical properties of their released material, or particle size distribution.

1.5.2 Scope of the present Guide

The International Nuclear Event Scale *(INES)* has been adopted to classify incidents and accidents at nuclear facilities. INES consists of seven incident/accident levels, ranging from level 1 (anomaly, deviations beyond the facility's permissible operating range) via level 4 (accident with consequences mainly within the installation and on its site, with public exposure levels comparable to the prescribed limits) to level 7 (major accident, with large-scale contamination and health consequences). Events without nuclear safety significance are classified in INES as 'Below scale/Level 0'. Accidents related to conventional industrial safety, i.e. not related to the nuclear aspects of operating a nuclear power plant, are referred to as 'Outside the scale'.

- The present Guide addresses level 5 through 7 accidents, the larger accidents or *severe accidents*.
- The approach to the analyses of accidents in levels 1 through 4, including *design basis accidents*, shows many similarities with the approach to the analyses of severe accidents. § 3.7 explains how the analyses of accidents of that kind should be performed.

The Standards, Recommendations and Requirements in this Guide are applicable solely to the estimation of radiological risks resulting from accidents that can be classified in INES. These are the risks of exposure to *direct radiation* from the reactor buildings (the buildings as the source of radiation) and the *radiation hazard* resulting from the release of radionuclides to the environment. Risks posed by, for example, the release of chemotoxic materials or explosions fall outside the scope of this Guide. This Guide also focuses on the quantification of group risk and individual risk. Deterministic calculations in so-called 'real time' applications, for example in the context of accident management, have specific characteristics that partly require a different approach.

A level 3 PSA assesses a broad spectrum of source terms. Such a source term spectrum ranges from small source terms involving a minor fraction of the core inventory to large source terms for which a major fraction of the core inventory is released. Each source term type imposes specific requirements on the calculations. Some source terms have a larger contribution to the total risk of a nuclear power plant than others. For that reason, specific aspects of a calculation that are only relevant to less important source terms are covered in less detail.

Much of the experience and knowledge regarding the importance of specific source terms and exposure pathways, and the associated choices made in the development of computer codes in the past, is based on standard sets of source terms (for example, in WASH-1400 [19]). Level 3 PSA computer codes are designed to cater for a set of source terms of that nature.

Experience has revealed that the risks of nuclear power plants are primarily determined by releases to the atmosphere. Releases to the ground and surface waters contribute relatively little to the total risk to the public and are therefore not addressed by the present Guide.

Other choices are directly related to the Dutch approach to risks. By the way in which group risk for severe accidents is limited, that approach places emphasis on low probability/high effect accidents. Consequently, that approach imposes stringent requirements on the models, in particular the models used to assess the deterministic consequences in the immediate surroundings of the nuclear power plant. It also implies that considerations in other studies that led to the conclusion that specific improvements have little impact on the endpoints of a level 3 PSA have been re-examined. The end results that are tested in international PSA codes and the associated principles are not always of relevance for the Netherlands. The studies carried out for the present Guide [20], [13], [15], [14] examined this aspect of the various models for the performance of a level 3 PSA.

1.5.3 Principles and design of this Guide

Principle 1

The Level 3 PSA Guide dovetails as closely as possible with the dispersion models, computation methods, and parameters for the *closer examination'* of routine emissions in the context of the Vbs Regulation [10], DOVIS-A [11] and the NNM [5].

Principle 2

The Guide is focused on the provision of good estimates of the group risk, maximum individual risk, and the dose for design basis accidents. That choices made for those endpoints may have an optimistic or a conservative effect on the determination of *other* endpoints must be taken for granted.

In addition to the group risk and maximum individual risk endpoints referred to above, further endpoints can be set for an analysis, depending on its purpose. These include the collective dose and the contaminated ground area size, as well as the effect of countermeasures.

Principle 3

In The Netherlands also internationally available level 3 PSA computer codes are applied. As those computer codes cannot readily be customized, this Guide needs to exercise restraint in prescribing stringent requirements for the implementation of specific Dutch models.

Although there have been all kinds of model developments in the Netherlands that also play a role in level 3 PSA, in The Netherlands no computer code has been nor is being developed with the advanced level 3 PSA functionality offered by MACCS2 [17] or COSYMA [16]. The strength of those computer codes lies in their capability to analyse a large number of endpoints at the same time. As the design of those computer codes was based on principles that ensured that justice was done to all their endpoints, no priority has been given to the group risk or maximum individual risk criteria.

Users can exercise their discretion in accepting any overestimation of an endpoint resulting from ‘international practice’, or can opt to follow the advice given in the present Guide. However, when ‘international practice’ results in an underestimation of the group risk and maximum individual risk endpoints then users should comply with the regulations prescribed in this Guide.

Principle 4

Any developments in other Dutch policy areas of relevance to level 3 PSA should be taken into account when using level 3 PSA computer codes.

Design of the present Guide

The information given in this Guide refers to *standards*, *refinements*, and *requirements*. The use of those terms is in line with the definitions given in DOVIS-A [11].

Standard: the *standard* is expected to yield a realistic but conservative estimate of the dose and the risk in most licence situations. The *standard* is in line with the customary national and international methods and models. The *standard* is also adopted when insufficient knowledge is available. The *standard* is expected to be adequate for most calculations.

Refinement: a refinement may be used when (the licence applicant can demonstrate that) the *standard* will result in a relevant overestimation of the calculated dose. This is, for example, the case when the *standard* yields conservative results for the group risk or maximum individual risk. This is also the case when the *standard* does not provide any models, such as for deterministic effects.

Requirement: the dose calculation should, irrespective of the actual situation, meet the preconditions prescribed by the requirements.

This Guide refers to existing legislation and requirements for parameter values and limits whenever possible. As a result, this Guide will remain up to date, even when such values and limits are amended.

2 Outlines of a level 3 PSA

2.1 Reasons for performing a level 3 PSA

A level 3 PSA can be performed for many reasons. The most important reason is to test the level 3 PSA results against the Dutch standards for group risk and maximum individual risk. Results of such an analysis are recorded in the facility's Safety Report.

The operator of a nuclear facility can also maintain a *living PSA*. The PSA addressed in the Safety Report is an envelope that encompasses all the expected future operational states of the facility. In a *living PSA,* the operator takes the current situation of all systems into account, and makes the necessary adjustments following changes in that situation. This *living PSA* reveals the safety consequences in case a safety system fails or is temporary shut down for maintenance. The *living PSA* can also be used for work planning to ensure that safety remains as high as possible.

The *living PSA* is of particular value when failure probabilities of (auxiliary) systems change, as it offers the ability to almost immediately determine – without complex, lengthy calculations – the influence of such changes on the current core damage frequency and the individual and group risks.

Also, level 3 PSAs are often used to estimate the economic loss and damages that may be caused by severe accidents and for the preparation of emergency response plans.

2.2 Accident releases and exposure pathways

Descriptions of the manner how and the conditions in which radionuclides enter the environment, both in routine emissions and as a result of accidents, are referred to as 'discharges', 'releases', or 'emissions'. In accident analyses, these are also referred to as 'source terms'.

Releases

Consistent with the 'closer examination' (Vbs: 'nadere analyse') of regular emissions pursuant to DOVIS-A, releases that can lead to exposure of individuals to radioactive material are the following:

- accident releases of radioactive material to the atmosphere
- accident releases of radioactive material to the surface water and discharges into the sewers
- accident releases of radioactive material to the ground
- increased radiation level around a source caused by an accident and the resultant direct radiation to the surroundings ('external radiation').

Exposure pathways

Exposure to radioactive materials in the environment can cause health detriment. That radiological hazard is quantified by means of the 'effective dose', in this Guide hereinafter referred to as 'dose'. Consistent with the 'closer examination' of routine emissions pursuant to DOVIS-A, the exposure pathways to be assessed are the following:

- Direct γ and β radiation from the cloud or plume passing overhead. The radiological hazard to individuals caused by this radiation is referred to as the 'cloud dose' (§ 3.4.4).
- Direct γ and β radiation from ground contaminated with radioactive materials. The radiological hazard to individuals caused by this radiation is referred to as the 'ground dose' (§ 3.4.5).
- Internal radiation exposure resulting from the inhalation of radioactive materials while the cloud passes overhead. The radiological hazard to individuals resulting from this exposure is referred to as the 'inhalation dose' (§ 3.4.6).
- Internal radiation exposure resulting from the ingestion of food contaminated with radioactive material (including drinking water). The radiological hazards to individuals resulting from this exposure is referred to as the 'ingestion dose' (§ 3.4.10).
- Depending on the source terms being assessed, a further radiological hazard can be caused by the inhalation of radionuclides from contaminated ground that become resuspended in air, the 'resuspension dose' (§ 3.4.9).
- External γ and β radiation from contaminated clothing and skin can result in a radiological hazard in the form of a 'skin dose' (§ 3.4.8).
- Direct exposure to radiation from radioactive material that remains contained in a power plant or facility results in a radiological hazard in the form of an external radiation dose.

An exposure pathway encompasses both[6] the dispersion of radioactivity in the environment, as the way of exposure to it. The following subsections review the exposure pathways associated with the various forms of release.

2.2.1 Exposure caused by the release of radioactive material to the atmosphere

Accident releases to the atmosphere occur either from the ventilation stack or other release point that may be fitted with a filter unit, either as gradual release of radioactive gases and aerosols from a building in which an accident has occurred, or as release of radioactive gases and aerosols entrained in smoke from a blazing fire.

[6] This is in line with the internationally accepted meaning of 'exposure pathway'. DOVIS-A [11] uses 'exposure path' only for the last part of the exposure chain, the way in which an individual is exposed to radioactivity in the environment. In practice, this rarely leads to confusion.

The next chapters of this Guide provide a comprehensive description of the release data required to perform the level 3 PSA (the 'source term').

2.2.2 Exposure caused by release of radioactive material to surface waters or sewers

Releases to surface waters occur either via sewers and waste water treatment plants or directly to surface waters. Routine releases to surface waters or sewers are permitted under strict licence conditions, usually limited per source of release.

In a PSA, accident releases to surface waters are rarely assessed, the present the level 3 PSA guide is restricted to accident releases to the atmosphere. The reason for this is that a release of radioactive material to surface waters does not pose a direct threat to individuals, in contrast to a release to the atmosphere (for example, by inhalation of radioactive materials in the residential area). In principle, indirect exposure is possible, for example by the consumption of contaminated drinking water or food (either directly, by the consumption or fish, or indirectly, by the use of sludge as soil improver or water for irrigation purposes), although appropriate countermeasures can limit this form of exposure simply and effectively. Implementation of countermeasures of this nature can be expensive though, hence the damage resulting from releases of radioactive material to surface waters is expected to be primarily economic.

In case of a level 3 PSA focused on the assessment of the individual risk and group risk, it should be examined whether releases to the surface waters or the sewers (contaminated fire extinguishing water) would relevantly contribute to the risk: when accident releases to the atmosphere are also feasible then those will usually dominate the risk.

2.2.3 Exposure caused by the release of radioactive material to the ground

Contaminated water leaking from the facility that is not contained for processing in a controlled manner will seep into the ground. At a few metres depth, the contaminated water will then mix with groundwater. Groundwater could also become contaminated by radioactive material leaching from landfill sites. Contaminated groundwater will slowly flow to a ditch, canal, brook or river and, ultimately, to the sea.

Currently, routine releases to the soil are not permitted and not even licensable, except for very small quantities (Bbs Decree [9], Article 10.5). Facilities must be designed in such a manner that leakages are precluded. Landfill sites are governed by a zero-emission regime: underneath a landfill a drainage systems is installed that intercepts any leached material. The water collected in the drainage system is cleansed and discharged in a controlled manner. Hence no computation rules exist for this exposure pathway.

Accidents in which contaminated water seeps into the ground cannot be excluded. These concern not only contaminated process water but also contaminated fire extinguishing water. This will not result in risks that pose a direct and immediate threat to individuals. However, in the longer term radioactive material may reach the immediate environment of individuals through groundwater flow. Following an accident involving leakages of contaminated water both the soil and groundwater must be cleansed.

2.2.4 Exposure to external radiation from a facility

External radiation from a facility potentially can cause a substantial radiation exposure. During an accident the containment of a radioactive source can be breached (as a result of fire, mechanical impact, earthquake, or human failure, for example). Moreover, following an accident in a facility a large amount of the radioactive material can be deposited on walls, ceilings, and floors, or remain in the filter house. The radiation from those radioactive sources will then need to be taken into account in the doses and risks for local residents as well as temporarily present people. For a ‘closer examination’’ of routine exposure to ionizing radiation, the Vbs Regulation [10] requires the application of the DOVIS-B [12] computation rules.

2.3 Conceptual level 3 PSA models

The 1993 Level 3 PSA Guidelines discussed actually only the modelling of accident releases to the atmosphere. This subsection discusses the conceptual models on which that is based.

For completeness, this discussion is supplemented with a description of the conceptual models for dispersion in water (aquatic exposure pathway) and for external radiation. Those conceptual models need to be discussed here, since in some specific situations those exposure pathways cannot be neglected in a level 3 PSA. As standard, a reference to DOVIS-A and DOVIS-B suffices for the calculation of the dose for those hazard paths. Except for the present section, as in the 1993 Level 3 PSA Guidelines, this Guide will not discuss the precise modelling of the aquatic pathway and external radiation in any detail.

2.3.1 Accident releases of radioactive material to the atmosphere

The following figure shows the processes to be modelled in a level 3 PSA assessment of accident releases to the atmosphere.

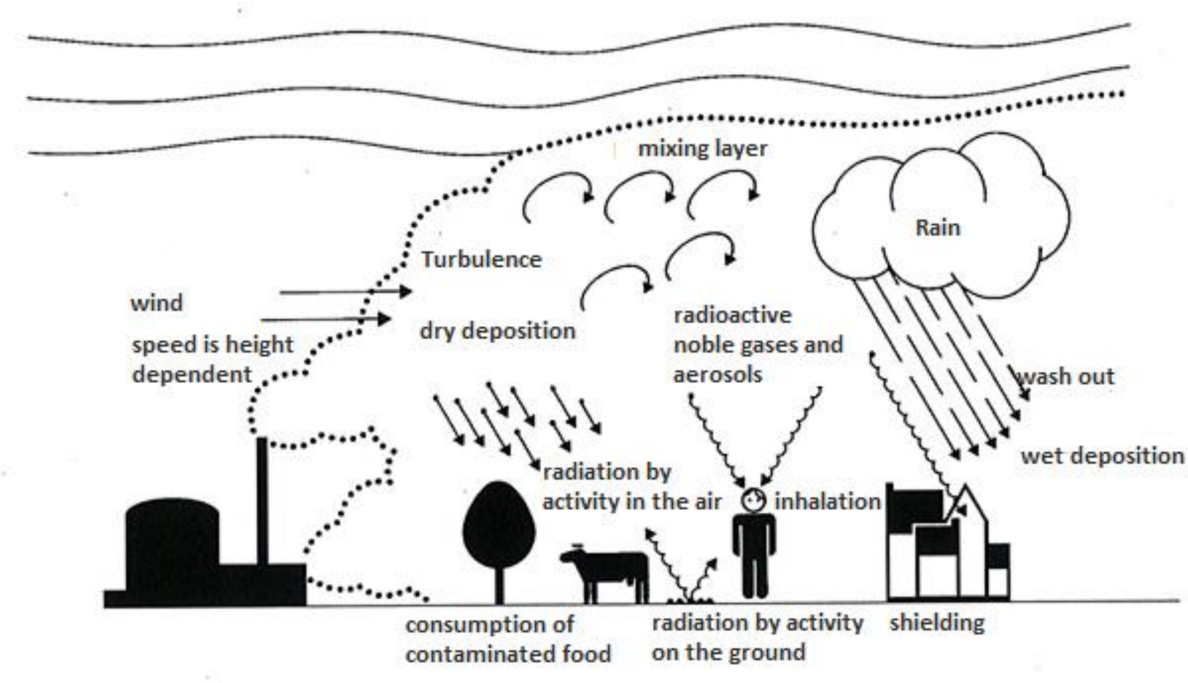

Figure 2-1 Processes to be modelled for accident releases to the atmosphere

The model for dispersion in the atmosphere is a Gaussian plume model. For emissions from a point source a widening plume is formed. This plume will not remain in a steady position due to the influence of changing wind directions during an emission. The average concentrations during the period of the release can be approximated with a Gaussian geometry, the Gaussian plume, in which the average concentrations at the plume edges will be lower than at the plume axis. Due to the statistical nature of the widening of the plume with distance from the source and wind direction fluctuations, the plume has the shape of a normal distribution, also called the Gaussian distribution, both vertically and horizontally. This is illustrated in the following figure.

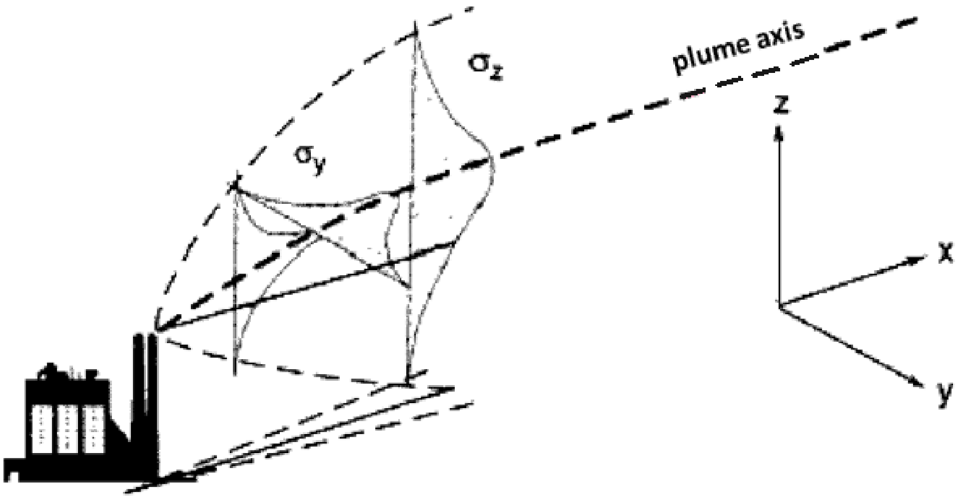

Figure 2-2 The basis of the Gaussian plume model (based on Infomil, 1998)

The dispersion calculation ultimately results in concentration statistics (in the atmosphere and on the ground) in the vicinity of the source. An annual average can be derived from these statistics, as well as percentiles and probabilities of exceedance.

When the concentrations of a radioactive material in the atmosphere and on the ground are known, the effective dose can be determined. The effective dose is a measure of the radiation-induced health detriment inflicted on an individual. This detriment is adjusted for the various biological effects of the different types of ionizing radiation. The effective dose is expressed in sievert.

The effective dose is not only determined by the concentrations of radioactive materials in the vicinity of the individual, but also by the biological characteristics of the individual (such as age and gender) and the individual's living habits.

Many of the parameters concerning exposure are specified in regulations. These concern for instance the choice of diet, the respiratory rate, consumption of drinking water, the region of origin of food products and the dose conversion coefficients to be applied. Sometimes, small children are considered as focus group, in particular for risk calculations. This is because at a given dose, children are at greater risk.

2.3.2 Accident releases of radioactive material to water and/or the ground

The aquatic exposure pathway is the route by which radionuclides released to water can reach an individual and cause exposure. The consumption of meat and milk (contaminated via sewage sludge) and freshwater fish (contaminated following direct releases to surface waters) usually dominates exposure via the aquatic exposure pathway.

This aquatic exposure pathway via surface water can be of relevance in some situations, for example a 'leakage from a tank of liquid waste' design basis accident. The dose resulting from an accident release to water can be calculated using the method laid down in DOVIS-A [11].

Contaminated water could also leak to groundwater. DOVIS-A does not include a model for this form of release, as this is not a permitted routine release path. For this kind of release, a groundwater model is needed for the calculation of the dispersion. For groundwater flow in The Netherlands, Deltares did develop the so-called NHI model. The very simple ERB1b model of the IAEA [21] can be used as a screening model to assess the dose/risk relevance. This (extremely) conservative screening model assumes that all nuclides released in an accident enter a groundwater flow of 10,000 m^3/year and that exposure is caused by a per capita consumption of 1.2 m^3/year of that groundwater.

2.3.3 Exposure to external radiation from a facility

Surfaces contaminated in an accident are sources of ionizing radiation. DOVIS-A refers to DOVIS-B [12] for the calculation of the resulting dose. A contaminated surface can, in terms of DOVIS-B, be assessed as an 'open source'. Another type of accident is an accident in which the source containment is breached, after which a release of radioactivity during normal operations cannot be excluded. The dose from external exposure to both contained and open sources at the site boundary is almost always only due to γ radiation.

The dose rate as a function of distance to the source depends on the spatial distribution of the radioactivity and the geometry of the radiation field. That geometry is dependent on the shape and dimensions of the source and can be approximated with one of the following five models:

1. Point source (spherical symmetrical field)
2. Linear source (cylindrical symmetrical field)
3. Surface source (uniform, homogeneous field)
4. Bundle (radiation field within a limited angle of aperture)
5. Volume source (source of large dimensions and usually non-symmetrical field)

These 'sources' are mathematical models for the radiation field of a physical radiation source and describe the dose rate as a simple function of the position (the distance) of any given point from the source.

Shielding

The dose rate at the site boundary from a source of ionizing radiation on the site is influenced by objects between the source and the nearest site boundary, which scatter and provide shielding from the radiation. These are, for example, buildings between the source and the nearest site boundary, or the wall of the structure containing the source. Account may also be taken of the shielding provided by air when sources are at a large distance from the site boundary.

The shielding from radiation provided by an object, such as a wall, depends on the material of the wall and its thickness, the type of radiation, and the radiation energy.

When assessing the shielding offered by a wall also the scattering of radiation by air must be taken into account. An individual behind a shielding wall can be exposed to radiation from the source due to radiation scattered from the air above (skyshine). A ceiling above a source can have a similar effect, as it can act as a 'reflector' of radiation.

The standard level 3 PSA computer codes cannot be used to calculate doses due to direct radiation from sources. DOVIS-B computation rules can be used to calculate the dose for simple geometries, for calculations involving complex geometries special computer codes such as MicroShield and MCNP are available. Some research institutes have developed applicable in-house computer codes as well.

The dose due to direct radiation should be added to the dose (inhalation and ingestion, etc.) calculated with the level 3 PSA computer code and then tested against the dose standards. The dose due to direct radiation should be taken into account in the risk calculations carried out in risk evaluations.

2.4 Endpoints of a level 3 PSA

The computer codes used to conduct a level 3 PSA can yield one or more of the following results:

- per source term, the contamination of the ground in the form of a statistical distribution due to the various weather conditions that may be encountered during the release.
- per source term, the dose an individual outside the facility may receive, in the form of a statistical distribution due to the various weather conditions that may be encountered during the release.
- per source term, the collective dose that a group of individuals may receive, in the form of a statistical distribution due to with the various weather conditions that may be encountered during the release.
- per source term, an increased probability, or severity, of health effects for an individual outside the facility, in the form of a statistical distribution due to the various weather conditions that may be encountered during the release.
- per source term, the risk incurred by an individual outside the facility if an accident occurs, expressed as the (location-specific) conditional individual risk and conditional group risk.
- for the full source term spectrum, the risk incurred by an individual outside the facility, expressed as the (location-specific) individual risk and group risk.

- per source term and weather condition, the size of areas in which the intervention levels for implementation of protective measures is exceeded (such as sheltering, evacuation, decontamination, relocation, and food control).
- the effect of planned countermeasures on the risks.
- per source term and weather condition, the cost of health effects, and countermeasures.

It should be noted that the various computer codes frequently make different choices with respect to the presentation of the statistical distributions. These differences do not relate so much to the actual presentation – which is always in the form of averages, percentiles and/or a CCDF, see the glossary – as to the underlying unit. The weather-condition-dependent statistical distribution of a dose from one source can, for example, relate to either the dose at a grid point, the average dose in an area or the maximum dose in an area.

2.5 Test against the legal framework

The results that are tested against the criteria stipulated in the Bkse Decree [8] are:

1. For the full source term spectrum of all beyond design basis accidents, the risk incurred by individuals outside the facility, expressed as the (location-specific) individual risk and group risk.
2. Per source term per design basis accident, the maximum dose that an individual at any point outside the facility's site boundary may receive, in the form of a statistical distribution of these maxima due to the various weather conditions that may be encountered during the release.

3 Models and model parameters

This chapter discusses the models and the model parameters to be used for the various subjects. The standards do not always prescribe the use of the most advanced models. The selection of a simple or more advanced model depends upon the endpoints that are being assessed (the deterministic or stochastic consequences, or ground contamination) and the sensitivity of these endpoints to a specific model. The existing computer codes include a considerable degree of conservatism, which has a great influence on specifically the endpoints used for testing against the Dutch group risk criterion.

As explained in Chapter 1, this Guide refers to three categories of requirements, namely Standards, Refinements and Requirements.

3.1 Components of a level 3 PSA

Figure 3-1 shows the elements of a level 3 PSA.

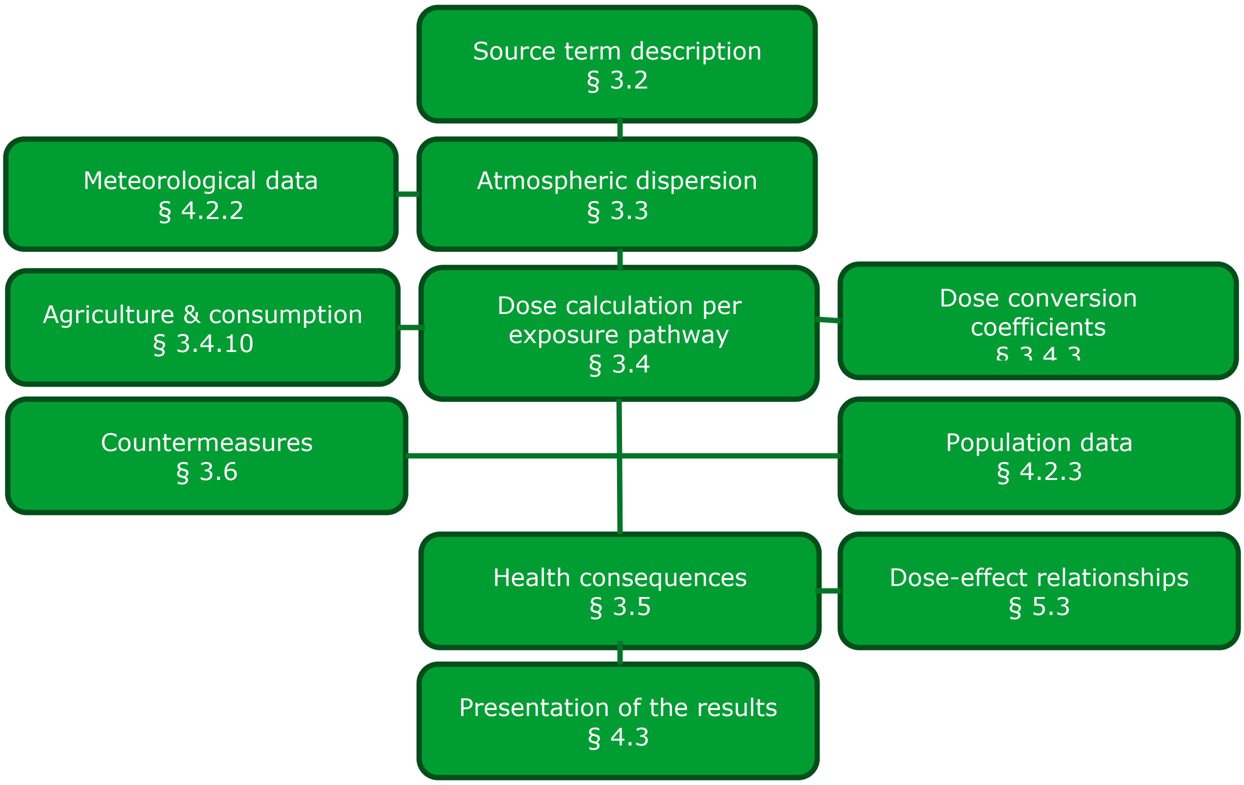

Figure 3-1 Elements of a level 3 PSA

3.2 The source term spectrum

The manner and conditions in which radionuclides enter the environment are described in terms of the 'source term' or the 'source term spectrum'. 'Source term' is used in the context of assessments of a release in a specific accident scenario. 'Source term spectrum' is used in the context of assessments of all the source terms of relevance to a facility. A source term encompasses the following elements:

1. Release point: the ventilation stack, via openings in the buildings, via cooling water to surface water and directly to the ground.
2. Size and composition of the release.
3. Chemical form of the released radionuclides.
4. Particle size distribution of the radionuclides that are released.
5. Thermal heat content of the release.
6. Humidity of the plume.
7. Time progression of the release.
   and
8. Likelihood of a specific source term.

Although 'source term' is often used in the sense of 'the release of radioactive material', a 'source term' is not a unit as such. The elements that should be specified in a 'source term' are determined by the degree of detailing of the level 3 PSA. Depending on the objective of the level 3 PSA (see also Chapter 4), more or less of the above facility-specific parameters will be required. In case of future developments in level 3 PSA models and methods, the above list will have to be supplemented with the parameters required to accommodate those developments. Such parameters could relate, for example, to the heat content of the plume, further detailing of its chemical composition or a more detailed specification of the dimensions and location of the various buildings. The following subsections discuss the most important source term data and state the step in a level 3 PSA calculation in which that data plays a relevant role.

3.2.1 Release point

The height of the release point relative to the height of the buildings plays an important role. Elevated releases are usually from ventilation stacks, where the source is regarded as a point source that is not influenced by lower buildings in the vicinity. Releases from buildings undergo an initial dilution that is caused by the buildings themselves as well as an extra initial widening of the emission plume. As a result, the concentrations in the plume at short distances from the source are lower than those of a release from a point source at the same height while the area covered by the plume is larger. The release point data is required for the analysis of the atmospheric dispersion (§ 3.3) and the influence of the buildings on the dispersion (§ 3.3.8).

3.2.2 Magnitude and composition of the release

The data on the magnitude and composition of the release result from a level 2 PSA.

The magnitude of the release determines whether only stochastic effects are to be expected or whether deterministic effects could also arise. Aside from a possible reduction factor (*DDREF*) for low LET radiation at low doses or low dose rates, the stochastic effects increase linearly with the dose and, consequently, with the magnitude of the release.

The severity of deterministic health effects as a function of dose generally increases sharply over a relatively limited dose range. That dose range is characterized by the LD_{50}, the dose that is lethal for half of the exposed population. Doses below that dose range (below the threshold dose) have virtually no effect, whilst exposure to doses above that dose range is almost always fatal. As a result, the number of deterministic effects is highly sensitive to the magnitude of the release. The LD_{50} and threshold dose also strongly dependent on the dose rate.

The composition of the release largely determines which exposure pathways will play a relevant role. In general, the magnitude of a release is specified as a fraction of the nuclide inventory present just before the accident. This implies that the nuclide inventory must be known.

3.2.3 The chemical form of the released radionuclides

The chemical form in which the nuclides are released affects the deposition processes and *dose conversion coefficients*. For purpose of deposition calculations most computer codes distinguish only a limited number of nuclide-groups, where the chemical properties play a role in the classification. The chemical form also influences nuclide behaviour in the soil after their deposition. When insufficient information about the chemical form is available usually only the various forms of iodine are distinguished, namely organic iodine, elemental iodine, and iodine bound to aerosol.

The chemical form of inhaled aerosols determines the time required to clear the lungs of that aerosol. Pursuant to the prevailing respiratory tract model [22], the various chemical forms of the radionuclides are classified into three classes of clearance time, namely F, 'fast', M, 'medium', and S, 'slow'. The Vbs Regulation [10] refers to ICRP Publication 119 [23], which states the inhalation class for most chemical elements, depending on the chemical compound in which the element is released.

3.2.4 Particle size distribution of the radionuclides being released

The particle size distribution of the released aerosol influences amongst others its deposition (§ 3.3.9) and the dose per unit intake (§ 3.4.6). *Standard* analyses are based on middle-fine aerosol [11].

3.2.5 Thermal heat content of the release

The plume rise of a release from a ventilation stack with a temperature above ambient temperature is taken into account in a simple way (§ 3.3.5). For releases between buildings plume rise will usually not occur, not even when its temperature is above ambient temperature. Only when the temperature of the release is much higher than the ambient temperature, the plume will rise partly or completely (§ 3.3.6). Aside in the event of fire, such a scenario will only occur in case of very large source terms.

It is usually assumed that a rising plume will not break through the top of the mixing layer, although that cannot be excluded when the plume's heat content is very high (§ 3.3.7).

The initial impetus of the release will, in principle, also play a role in plume rise. However, as the direction of the impetus for releases between buildings is not known, generally that contribution is not taken into account.

3.2.6 Humidity of the plume

Little is currently known about the assessment of the humidity of a plume within the context of level 3 PSA. Water droplets and water vapour in a release plume can have a dual effect: water droplets in a hot plume can evaporate and, as result, reduce the effective plume rise, whilst the cooling of the plume can result in condensation of water vapour to form droplets that then cause extra contamination close to the release point (wet deposition) and reduce the amount of radioactive material in the plume. The influence of those effects on the analysis endpoints is complex. As for those processes no models for use in level 3 PSA are available this Guide does not take plume humidity into account.

3.2.7 Time progression of the release

The time progression of a release influences the total released amount of nuclides, due to radioactive decay. The duration of the release also influences the area covered by the release plume. Moreover, the release's time progression also influences the implementation and effectiveness of countermeasures.

The duration of accident releases from nuclear power plants varies from less than one hour to many days. This has major consequences for risk calculations. Most computer codes are developed for source terms resulting from relatively short releases (up to some 10 hours), whereby it is assumed that changes in wind direction will be relatively small. For most source terms, the release rate of the various radionuclides and radionuclides groups is not constant throughout the release period. That variation is usually approximated by dividing the source term into a number of phases, each with a constant release rate. This partitioning into release phases is part of the interface between a level 2 and level 3 PSA. As the way of distributing the release has a great influence on the endpoints of a level 3 PSA – in particular, for deterministic effects – a number of directions for that distribution are given below.

Generally, parameter values for delay time, warning time, and release duration have to be entered in the computer code.

Delay time

The calculated core inventory (which is usually calculated at end of cycle, the actual inventory is usually lower) is assumed to be applicable at reactor shutdown, the on-set of the accident. As a release will usually begin at a later time, the specified core inventory will need to be corrected for radioactive decay and in-growth of daughter radionuclides during that delay time.

Warning time

The warning time is of importance when the analysis takes account of the implementation of countermeasures. The warning time is the time between the first warning of a pending release and the start of the release. The warning time is of relevance for the procedures in the emergency response plans.

Release duration

The duration of the release depends on the conditions in the reactor containment and the condition of containment itself. During the release, the amount of remaining radionuclides will decrease due to radioactive decay. Changes in wind direction will also increase the dispersion area of the released radionuclides, as a result of which the maximum time-integrated concentrations of identical release fractions will be higher with shorter release periods than with longer release periods. This is directly influenced by the ‘timing’ of the source term, i.e. the fraction released during each release phase. The modelling of this timing needs to be tailored to the capabilities of the computer code.

3.2.8 Likelihood of the source term

For each source term, the likelihood per reactor year is needed to combine the results for the various source terms in the eventual risk calculation.

3.3 Atmospheric dispersion and deposition

The NNM model is the standard model to be used for atmospheric dispersion and deposition in level 3 PSAs. The NNM models are suitable for consequence analyses of accident releases of radioactive materials virtually without adaptation. As the NNM has primarily been developed for semi-continuous sources, its modelling has to be adapted at some points for the assessment of the releases in level 3

PSAs. The NNM model also lacks some specific information for the performance of dispersion analyses for accident releases. That is elaborated in more detail in Chapter 5.

3.3.1 Dispersion model

The model for atmospheric dispersion that is applied in most consequence analyses is the Gaussian plume model with reflection at the ground and the top of the mixing layer. This model is relatively simple and, for low sources, yields results close to those derived by more complex models, in particular for dispersion over relatively flat terrain.

Standard: the bi-Gaussian plume model should be used to calculate atmospheric concentrations. This is the standard model in the NNM reference model. The σ parameterization prescribed by the NNM should be used when working with that model. Other internationally accepted methods and models may be used when it has been made sufficiently plausible that they yield realistic though conservative estimates of both dose and risk for the specific licence situation.

The NNM adopts a straight-line approximation of the plume. Other models include wind shifts during transport, which is of particular importance to forecasts in support of accident response. In level 3 PSA assessments the wind shift effect disappears almost completely, as cases in which wind shifts would occur causing the cloud to miss a population centre are about as often as cases in which a population centre would be reached as result of a wind shift. Although account needs to be taken of the possibility that wind shifts occurring shortly after each other could result in a cloud passing twice over the same area, weather conditions of this nature occur on an average of at most once a year and then almost always in combination with still air (information about how to deal with that situation is enclosed in § 3.3.3).

3.3.2 Time progression of the release

The NNM model yields hourly average results for the concentration and the deposition, whereby it is assumed that the release during each hour is relatively constant. However, accident releases can be of a very short duration, for example 10 minutes, or of a much longer duration, for example a number of days.

Standard: the release should be divided into hourly release phases: the level 3 PSA computational tool should then calculate the concentration and deposition in each release phase and cumulate the results for the entire release period.

Refinement: for small source terms, generally the risk is proportional to the size and the probability of the occurrence of the source term, and is insensitive to the precise progression of the release. Consequently, for small source terms (minor accidents or design basis accidents at nuclear power plants, as well as accidents at installations with a small inventory) a complex release or release with a lengthy time progression can be approximated by a release with a duration of one hour and a constant release rate.

Refinement: the lateral width of a plume from a very short release will be smaller than the one-hour average used in the NNM. The plume width can then be corrected with [24]:

$$\sigma_y(t_{release\ duration}) = \sigma_{y,NNM}(1\ uur)\left(\frac{\max(t_{release\ duration}, 10\ minutes)}{1\ hour}\right)^{0.35}$$

Refinement: for large source terms, threshold effects (group risk) and the associated countermeasures can play a role. It will then be necessary, although the computation time will be longer, to utilize the options offered by the computer code to define as many release phases as possible. The following points need to be taken into account when partitioning the source term into release phases:

- The minimum release duration of a phase should not be shorter than the averaging time adopted for the determination of the dispersion parameters (one hour with the NNM).
- The initial partitioning into phases is determined by any sudden changes in the source term release behaviour.
- The duration of a release with a predominantly constant release rate should be divided into release phases of equal length, whilst a release with a predominantly exponentially decreasing release rate should be divided into equal release phases using a logarithmic partitioning.
- The 'effective' release duration of a specific release phase with an exponentially decreasing release rate is equal to twice the time in which the first 50% of the total release in that phase is released. The effective release duration is then shorter than the actual release duration. Most computer codes assume a constant release rate in each of the release phases during the release. The use of the effective release duration for an exponentially decreasing release yields a better approximation of the instantaneous concentrations and dose rates at the beginning of the release phase that mainly influences the occurrence of deterministic effects.
- With a constant release rate, the release duration is equal to the time needed for 100% of the release in that phase to occur.

3.3.3 Wind

Standard: the *wind speed profile,* the wind speed as a function of height, is dependent on the *stability* pursuant to the NNM. Calculations using a Gaussian plume model with distance-dependent σ parameters overestimate concentrations at very low wind speeds *(*< 0.5 m/s). For that reason, a minimum wind speed of 0.5 m/s should be adopted[7].

[7] More information about the NNM approach to low wind speeds and their influence is available in [5], § 9.1, page 53, under '*onberekenbare uren'* (incalculable hours). Practice has revealed that these incalculable hours have

The lower limit of wind speeds at a height of 10 metres accepted by the NNM reference model is 1.0 m/s. In practice, for wind speeds of 0.5 – 1.0 m/s a wind speed of 1.0 m/s can be assumed: until 2002, the NNM calculations did not take account of lower wind speeds. The meteorological data is now presented in the form of hour-averaged wind speed values whereby the values in wind calm hours are replaced by the last real value.

Refinement: wind speed profile data of a higher quality may be used when available.

Standard: the effect of changes in wind direction on concentrations during the release should be taken into account by hourly changes in stability, wind direction, and wind speed. The NNM's hour-to-hour data is available for this purpose.

The effect of the release duration on concentrations is then taken into account adapting the dispersion in the specified release phases for changes in stability, wind speed, and wind direction.

3.3.4 Mixing layer height

Standard: the method for the calculation of the mixing layer height for use in the dispersion model is specified in the NNM reference model. The influence of the mixing layer height on dispersion should be taken into account by making adequate corrections to the bi-Gaussian plume model. Other internationally accepted methods and models may be used when it has been made sufficiently plausible that they yield realistic though conservative estimates of both dose and risk for the specific licence situation.

3.3.5 Plume rise

Standard: the formulation of the NNM reference model should be used for plume rise from a point source release (i.e. without building wake). Other internationally accepted methods and models may be used when it has been made sufficiently plausible that they yield realistic though conservative estimates of both dose and risk for the specific licence situation.

3.3.6 Plume rise of releases between buildings (*lift-off*)

Standard: in case of building influences, it should be assumed that plume rise does not occur.

Refinement: in case of a fire, the standard may yield an excessively conservative result for flue gases with very high heat content. Hot flue gases from a fire may break out of the building wake zone. The minimum plume rise required to break out, Δh_v, can be calculated using the following equation:

only minor influence on both the average and the percentiles. Although this might differ for high sources, other studies have not revealed any indications of a greater influence.

$$\Delta h_v = D/(2 \cdot \beta_e)$$

where

D diameter of the surface area of the fire

β_e Briggs' entrainment constant: $\beta_e \approx 0.6$

When the calculated plume rise is larger than Δh_v, it may be assumed that the calculated plume rise can be used in the calculation. When the calculated plume rise is smaller than Δh_v then it should be assumed that plume rise does not occur.

3.3.7 Plume breakout of the mixing layer

Standard: calculations of atmospheric concentrations should assess plume breakout of the mixing layer. This should be carried out using the NNM modelling. Other internationally accepted methods and models may be used when it has been made sufficiently plausible that they yield realistic though conservative estimates of both dose and risk for the specific licence situation.

3.3.8 Building influences

Standard: the influence of buildings in the vicinity of the release point is modelled as specified in the NNM or with a comparable method such as the ASHRAE recommendation [25], which is often used to calculate the required stack height.

Standard: when plume entrainment by the building wake is an issue and the model for building influences does not include a recirculation zone model then only the results calculated from a minimum distance from the building of about 100 metres may be used. When the distance between the site boundary and the building with the aforementioned recirculation zone is less than 100 m then a refined analysis should be used.

Refinement: the refined analysis should encompass at least a qualitative specification of the influence of the recirculation zone on the calculated doses and risks at the site boundary and, when the doses and risks are influenced by the recirculation zone, state at least a substantiated multiplication factor for the increase in the concentrations, doses, and risks in the recirculation zone as compared to the results from the calculation without modelling of the recirculation zone.

Refinement: in the case of a freestanding building, the NNM building influence model or a comparable model may always be used.

3.3.9 Dry deposition

Standard: the modelling of the deposition of aerosols should be based on the NNM reference model. DOVIS-A states the following as the standard for the aerosol size distribution: the NNM calculation of the deposition of radioactivity on particles is based on the middle-fine aerosol category unless this is demonstrably incorrect. Other internationally accepted methods and models may be used when it has been made sufficiently plausible that they yield realistic though conservative estimates of both dose and risk for the specific licence situation.

Standard: no deposition calculations are required for noble gases.

Standard: when no further information is available then a low surface resistance (R_c) of 50 s/m should be used for elemental iodine. A surface resistance of 5000 s/m should be used for organic iodine compounds.

For elemental iodine and organic iodine compounds this is equivalent to a deposition rate of 0.015-0.06 m/s (depending on the thickness of the surface layer z_0 and the friction speed u*).

Standard: no calculations are required for C-14 and H-3 deposition on the ground for the purposes of dose calculations: those radionuclides do not contribute to the ground dose and the transfer of material to the food chain is calculated based on the nuclide concentration in air.

3.3.10 Wet deposition

Standard: the NNM *washout* reference model should be used for the modelling of the wet deposition of aerosols and gases, both for matter within the mixing layer and – when plumes break through the top of the mixing layer – for matter above the mixing layer. Other internationally accepted methods and models may be used when it has been made sufficiently plausible that they yield realistic though conservative estimates of both dose and risk for the specific licence situation.

Standard: the NNM washout reference model should be used for the wet deposition of iodine. When no further information is available then, pursuant to DOVIS-A, $D_g = 0.2\ cm^2\ s^{-1}$ should be used for the calculation of the washout coefficient of elemental iodine and $D_g = 0.05\ cm^2\ s^{-1}$ for organic iodine compounds.

3.3.11 Mist

Standard: the influence of mist on deposition does not need to be modelled until good models and data are available in the Netherlands.

3.4 Exposure pathways and dose calculations

The exposure pathways to be assessed in level 3 PSAs are delineated in § 3.4.1. These are: exposure to direct radiation from radionuclides in the atmosphere, on the ground and on the skin and clothes, the inhalation of radionuclides in air, and the ingestion of radionuclides in contaminated food. § 3.4.2

lists the general principles for dose calculations. This is followed by information about the computation method to be used for each exposure pathway.

3.4.1 Relevant exposure pathways for level 3 PSAs

Level 3 PSAs assess those exposure pathways along which the population could receive a dose. The exposure pathways that are of relevance are determined solely by the source term. Exposure to radiation from a release of radioactive noble gases, for example, will be solely via direct radiation from the plume passing overhead. Exposure to radiation from a release of actinides (for example, accident release from a radioactive waste storage facility) will be primarily via inhalation and ingestion. Exposure to radiation from a release of volatile nuclides (such as I and Cs) will, depending on the exposure period, be via direct radiation (also from the ground), inhalation, and ingestion.

Standard: level 3 PSAs (risk analysis) of accident releases should assess the following types of accident releases and exposure pathways to the surroundings:

- accident releases of radioactive material to the atmosphere
- accident releases of radioactive material to surface waters and discharges into sewers
- accident releases of radioactive material to the ground
- increased radiation levels around a source caused by an accident and the resultant direct radiation to the surroundings

Experience has shown that the risks of nuclear power plants are primarily determined by releases to the atmosphere. Releases to the ground and surface waters play a minor role in the total risk to the population [20], as exposure via those pathways can readily be controlled by the implementation of appropriate countermeasures. The consequences of releases of that nature are primarily of economic nature, which fall outside the scope of the present Guide[8].

Experience gained from consequence analyses of nuclear power plant accidents reveals that six exposure pathways play a greater or lesser role in exposure to radiation from releases to the atmosphere. These are:

Relevant exposure pathways:

- Direct γ radiation and β radiation from the cloud or plume passing overhead. The dose resulting from that radiation is referred to as 'cloud dose' (§ 3.4.4).
- Direct γ radiation and β radiation from ground contaminated with radioactive material. The dose resulting from that radiation is referred to as 'ground dose' (§ 3.4.5).

[8] When the potential releases from a facility are limited to releases to the ground and to water, the appropriate DOVIS-A models may be used to calculate potential doses and to estimate the potential economic loss.

- Internal radiation exposure resulting from the inhalation of radioactive material while the cloud passes overhead. The dose resulting from that radiation is referred to as 'inhalation dose' (§ 3.4.6).
- Internal radiation exposure resulting from the ingestion of food contaminated with radioactive material. The dose resulting from that radiation is referred to as 'ingestion dose' (§ 3.4.10).
- The inhalation of resuspended radionuclides deposited on the ground can, depending on the source terms that are being assessed, contribute to the total dose (§ 3.4.9). Internal radiation caused by consumption of crops contaminated with resuspended radioactive material is not assessed.
- *External* γ and β radiation from contaminated clothing and skin (§ 3.4.8).

This Guide addresses the six aforementioned exposure pathways. Figure 3-2 shows the relationship between a release to the atmosphere and the assessed exposure pathways.

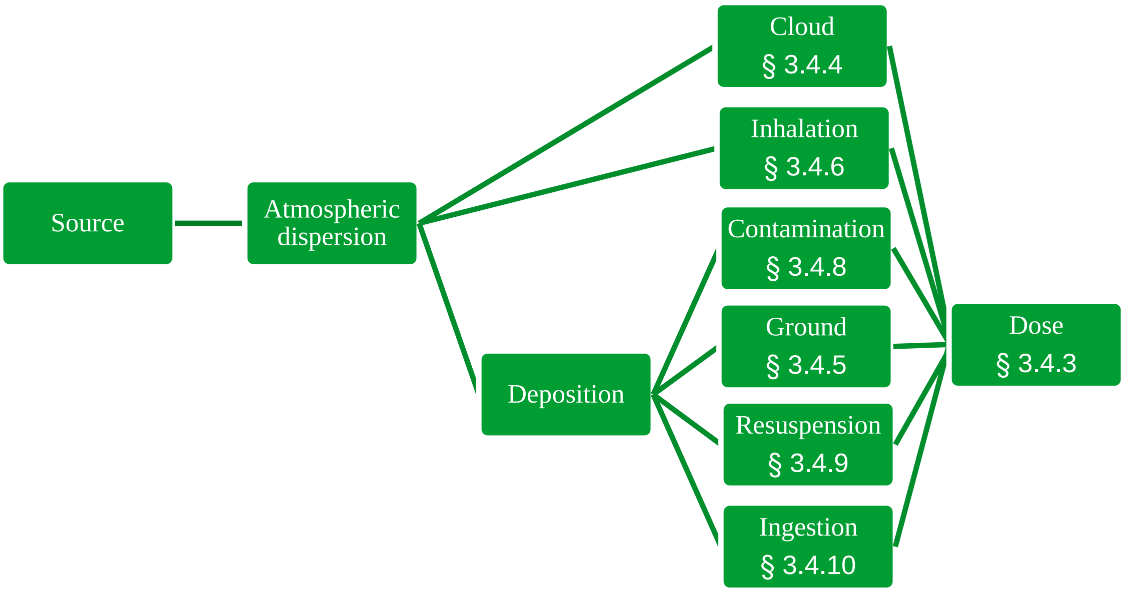

Figure 3-2 Relationship between the assessed exposure pathways

When an accident results in a large release from the primary system to the reactor containment but is not followed by a release to the surroundings then external radiation exposure of individuals in the immediate surroundings should also be assessed. Information about models and the computer codes that should be used for the calculation of this external radiation exposure (which is not discussed further in this Guide) is available in DOVIS-B [12].

The relative importance of the exposure pathways referred to before is dependent on:

- The composition of the source term;
- The effect that is being assessed (deterministic or stochastic);
- The countermeasures that are implemented to lower the dose.

Exposure other than through the aforementioned exposure pathways can also contribute to the total dose received. However, as those pathways give relatively small contributions to the total dose from a severe nuclear power plant accident they do not usually need to be taken into consideration (see Section 2.2). However, when the circumstances of the accident being assessed deviate greatly from the ‘customary’ circumstances of nuclear power plant accidents (that result in the ‘source term spectrum’, the set of standard source terms referred to in Chapter 2) then such neglected exposure pathways <u>may</u> need to be taken into consideration. The following is a non-exhaustive list of exposure pathways that may then be of relevance:

The following exposure pathways are not assessed in standard level 3 PSAs:

- Contamination of seawater and surface water by:
  - direct deposition of radioactive material dispersed in the atmosphere,
  - activity washed from contaminated ground by rain,
  - groundwater contaminated by activity leached from the surface,
  - groundwater contaminated by radioactive material following a meltdown,
  - direct release.

  Surface water and seawater contaminated in this manner can result in a dose via:
  - Consumption of fish and shellfish,
  - Drinking water,
  - Watering of fields,
  - Swimming.

- Consumption of agricultural produce grown on fields contaminated via groundwater contaminated with radioactive material,
- Consumption of agricultural produce grown on fields fertilized with sewage sludge contaminated with radioactive material,
- Contamination of food by resuspension of prior deposited radioactive material.

The models required for (for nuclear power plant accidents) non-relevant exposure pathways are not discussed in this Guide. However, when a source term of a specific accident necessitates consideration of one or more of such exposure pathways, either a method from DOVIS-A [11] should be selected or a substantiated ad hoc approach should be adopted.

3.4.2 General principles for dose calculations

Many of the principles that must be adopted for dose calculations are stipulated in DOVIS-A [11]. The most important general principles are summarized in this subsection. Other relevant principles stipulated by DOVIS-A are addressed in subsequent subsections.

Focus groups and the representative person

The prevailing Dutch radiation protection policy assumes that protection must be provided to the groups of persons who, in comparison to the other members of the population, can be on average at greater risk from exposure to radiation from a given source term and/or source of radioactive matter. These theoretically identifiable groups of persons are referred to as ***focus groups*** for the source, see Vbs [10] Annex 10, Section 6.3. Within this context, 'theoretically identifiable' should be understood as a group identifiable from shared characteristics that have a determinative influence on the risks posed to members of such group. These characteristics can relate to age, gender, and cultural aspects, such as dietary habits.

The ***representative person***[9] is an individual receiving a dose representative of the more highly exposed individuals in the population, excluding those with extreme or rare habits. The representative person is an average member of the focus group receiving the highest dose summed over all exposure pathways.

Requirement: both for the calculation of dose and risk under the specific circumstances for which a calculation applies, it must be checked which focus groups might be present. *For accident releases resulting in exposure to radiation of all age groups in the population in general the group of one-year-old children must be assessed as focus group.*

A further principle of the aforementioned policy stipulates that protection must be offered to the vast majority of the population (about 95%). It is conceivable only for some groups of people who are not 'theoretically identifiable' as meant above (such as leisure activities or the consumption of food caught or collected in the wild) that they may be at higher risk than the focus group.

Requirement: when the total number of individuals in the aforementioned groups amounts to more than 5% of the population then the group on the 95th percentile of the population is regarded as focus group for the assessment, too.

The greater risk to a focus group can be either due to physiological factors that are taken into account in the dose conversion coefficients or to behavioural factors that have an impact on the exposure pathways that are to be assessed. This means that the calculation of the *individual dose* cannot be

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[9] According to ICRP 101, the representative person is equivalent to, and replaces, the average member of the critical group recommended previously by the Commission. ICRP has no equivalent term for 'focus groups', these are generally referred to as 'groups (of people)'.

based solely on a dose conversion coefficient for a member of the population of average age: the age dependency of this coefficient will need to be taken into account for the relevant exposure pathways.

Requirement: dose calculations should be based on dose conversion coefficients for various age groups, including the focus group of one-year-old children. The procedure that should be used is explained in more detail in the following subsections.

Irradiation periods

Requirement: in accident assessments, all doses resulting from an accident, including those received in later years (such as radiation from the ground, resuspension, and ingestion) shall be attributed to the year in which the accident occurred. The period considered for the committed dose is 70 years for children and 50 years for adults.

3.4.3 Dose conversion coefficients

Most computer codes use pre-calculated sets of dose conversion coefficients (DCCs) for the calculation of the dose from the various exposure pathways. A DCC expresses the relationship between the intake of a radionuclide (expressed in becquerel (Bq)) or the exposure to a radionuclide concentration (Bq/m^2 or Bq/m^3) and the resultant dose integrated over a specific period. DCCs relate to specific organs and are, in particular for internal exposure pathways, age dependent.

Standard: age-dependent organ dose and effective dose conversion coefficients calculated from the latest ICRP publications should be used whenever available. Vbs Regulation [10] refers for these ICRP DCC values to ICRP Publication 119 [23]. Alternative sources of DCCs, such as FGR-12 [26], may be used when they supplement the ICRP DCCs or when it is plausible that they will yield highly comparable results.

Standard: calculations of the individual dose for the determination of the maximum individual risk should use the DCCs for the focus group. This means that the DCCs used for the exposure pathways external radiation from the cloud and from the ground should be multiplied by age dependent correction factors listed in **table 3-1**. Internal exposure pathways calculations should use dose conversion coefficients for the specific focus group. The set of dose conversion coefficients for skin contamination is age independent.

Table 3-1 Age dependent correction factors for cloud and ground doses [27]

Age (years)	*cloud dose*	*ground dose*
0 to 1	1.5	1.8
1 to 4	1.5	1.8
5 to 9	1.4	1.5
10 to 14	1.3	1.4
15 to 19	1.15	1.15
20^{+}	1.0	1.0

3.4.4 External γ radiation from the cloud

Standard: dose calculations for this exposure pathway use nuclide-specific DCCs (Sv/s per Bq/m^3) for submersion in a semi-infinite cloud of a homogeneous concentration. As the plume shape differs from a semi-infinite cloud, certainly near the release point (over 10 km in stable atmospheric conditions), supplementary to DOVIS-A a correction must be applied to take account of the plume geometry and the distance to the plume axis. The elaboration of this supplement is given in Subsection 5.2.2.

Standard: when the cloud contributes less than 25% of the total dose, which will usually be the case for source terms following a serious accident, one γ energy may suffice to model the energy dependence of the set of plume correction factors during the total release.

3.4.5 External γ radiation from the ground

Standard: dose calculations for this exposure pathway use nuclide-specific DCCs (Sv/s per Bq/m^2) for exposure to a homogeneously contaminated infinite flat surface. The calculation is based on the local concentration on the ground. A correction factor for the dose of γ radiation in the field should be used to correct for the roughness of the ground surface. Moreover, next to radioactive decay also the time dependency of the dose rate due to activity transport to deeper layers of the soil should be modelled.

Over the years, part of the radionuclide contamination will be transported into deeper soil, causing reduction of the dose rate. This is referred to as ‘weathering’.

There are a number of weathering processes: the most important are leaching and the shifting of sand and clay particles in the top soil (the top centimetres) caused by wind and rain.

DOVIS-A assesses solely leaching from the surface layer. The DOVIS-A model is based on the net rainfall in the Netherlands of about 40 cm a year that drains into the soil (the remaining fraction evaporates). In essence, the DOVIS-A compartment model is based on the assumption that water brings materials 100 cm deeper into the soil each year. Once these materials have reached a depth of several metres below ground level they are carried by the groundwater to ditches, brooks and rivers, and, ultimately, transported to the sea.

Many deposited radionuclides are in oxidized form and thus relatively hard water soluble. Moreover, the oxides often bind to sand and clay particles. As a result, in practice these oxides are not carried by water: the DOVIS-A model, for example, assumes that the leaching velocity of Cs oxide is less than 0.1 mm per year.

Sand and clay particles in the surface layer (the top centimetres) – to which the oxidized radionuclides are bound – are shifted by wind and rain. The weathering model of Gale *et al*. (1964) is the most frequently used model

$$C(t) = 0{,}5\ e^{-1.39t} + 0{,}5\ e^{-0.0077t}$$

with t in years.

This weathering has often been observed, for example by Golikov *et al*. [28], who carried out measurements in Bryansk (about 500 km north-east of Chernobyl). However, much less weathering has been observed in heath and forest soil (soddy-podzolic soils [29]). The shifting of sand and clay particles will be much stronger in inhabited areas.

It should be noted that DOVIS-A does use Gale's weathering model for the calculation of the resuspension-inhalation dose caused by radionuclides blown up from the ground surface.

The following standards have been formulated on the basis of the aforementioned considerations:

Standard: the dose rate calculated based on a homogeneously contaminated flat surface is reduced by a factor of 2 to take account of the shielding effect of irregularities in the surface.

Standard: the modelling of the dispersion of radionuclides in the ground is calculated on the basis of accumulation in and removal from an assumed homogeneous top soil layer of 0.2 m for fields and 0.1 m for grassland. Alongside radioactive decay, account may also be taken of leaching in accordance with DOVIS-A.

Standard: the dose rate is reduced by mechanical weathering in accordance with the aforementioned model based on Gale's studies.

3.4.6 Internal exposure by inhalation of radionuclides

Individuals residing within a plume as it passes by receive a dose via this exposure pathway. The exposure period, as is the case with cloud radiation, is relatively short – the time the individual is within the plume or cloud. However, as radionuclides remain in the body after the cloud has passed the *irradiation period* is longer than the exposure period (the committed dose).

Standard: the models used for the calculation of the inhalation dose are based on the product of the time-integrated concentration of radionuclides in the air, the *volumetric inhalation rate*, and the appropriate dose conversion coefficient. Both latter parameters are age dependent.

Although the particle size of the inhaled nuclides has influence on the inhalation dose, for aerosols an AMAD of 1µm may be assumed.

The lung absorption types (and the factors f_1 for intake in the intestine) should take account of the chemical form of the element. In general, in the absence of information on those parameters the most conservative value should be adopted as published in ICRP Publication 119 [23], as referred by the Vbs Regulation [10].

3.4.7 Shielding factors for external radiation and inhalation

The calculation of the collective and individual dose resulting from an accident should not assume any countermeasures and the dose should be integrated over the expected lifespan of the population group being assessed. This implies that normal behaviour of the population or representative behaviour of the focus group(s) is assumed. For that reason, it is necessary to state the extent to which the dose from external radiation from the plume and from the ground, inhalation in the plume and inhalation from resuspension is influenced by the time spent indoors and outdoors. This means in fact, that it has to be established for which part of the population, for which part of the day, for which part of the release duration and for which type of dwelling the dose calculation must be corrected for indoor residence. Consequently, in principle a wide range of shielding factors is conceivable for the calculation of the dose from the various exposure pathways in combination with the corresponding population subgroups. However, in view of the large number of variables the calculation codes often do not allow different shielding factors for the various starting times, for example in the day and in the night.

DOVIS-A states that shielding factors can be used for external radiation. The Vbs Regulation [10] prescribes (translated from the original Dutch text): ‘The calculation of the external exposure dose at a specific location is based on individuals in a focus group that live at the relevant location and spend 24 hours a day throughout their entire life inside or in the vicinity of their home which provides some degree of shielding.’ and ‘On the basis of a rough estimation of the shielding that a standard home provides to gamma radiation of different energy levels and different sources, the shielding factor is set at 0.25.’

The following table lists the shielding factors that are to be used.

Table 3-2 Shielding factors to be used for the various exposure pathways

	1993 Level 3 PSA Guidelines	Vbs Regulation	DOVIS-A	Level 3 PSA Guide Revision 2020
inhalation	1	1		1
cloud dose: shielding by home	0.3	1	-	1
ground dose: shielding by home	0.2	0.25	-	0.25
ground dose: shielding by surface roughness	0.5	-	1 (not assessed)	0.5
ground dose: shielding by transport to deeper layers of the soil	weathering model of [30] (empirical)	-	model for homogeneous leaching (K_d-model) for the top soil layer of 20 cm	both weathering and leaching are considered (hybrid model).

3.4.8 External exposure to radiation from deposition on skin

This exposure pathway results in a skin dose in particular by β emitters deposited on skin and clothing. Exposure occurs during the period spent within the cloud until the skin is washed to remove the contamination. Level 3 PSA computer codes often use models such as the models drawn up by Henrichs [31], [32]. Those models assume that the amount of radioactive material deposited on skin and clothing is proportional to the amount deposited on the ground. The skin dose is calculated using dose rate conversion coefficients for each of the nuclides (calculated by GSF [32]) and the time integral from the time of contamination to the time of decontamination by washing.

Requirement: this exposure pathway should certainly be assessed. The deposition on skin should be assumed to be equal to the deposition on the ground. The duration of contamination should be assumed to be one day (24 hours).

Refinement: Since rain will wash away much of the contamination of skin and clothing, assessment of this exposure pathway may be limited to dry conditions.

3.4.9 Resuspension

Experience with consequence analyses of light-water reactor accidents has shown that inhalation of resuspended material is a relatively unimportant exposure pathway. However, this conclusion may not necessarily be valid for other facilities such as reprocessing plants. When the source term primarily consists of actinides the relative dose contribution via resuspension will be higher.

Standard: the calculation of the resuspension dose requires an assessment of weathering in accordance with the model of [30] and the data of [33], as well as of the effect of leaching as specified in DOVIS-A (hybrid model).

Refinement: the calculation of the resuspension dose may also be calculated using the refinement suggested in DOVIS-A as an alternative to the standard model (see § 5.2.5).

3.4.10 Internal exposure to radiation via ingestion

The contribution from this exposure pathway can readily be limited by restricting the consumption of contaminated food, in particular of vegetable crops directly contaminated with radioactive material. Although for risk calculations in the Netherlands no countermeasures may be assumed, the calculations for that pathway may be based on food contaminated up to the permitted maximum level. It is assumed then that the part of the population that (partly) provides in their own food needs, will stop using self-grown food after a warning before or very soon after a release.

Standard: calculations should be based on the assumption of local production and consumption of food. The calculations should be based on the assumption that the consumed food is less contaminated than the intervention levels specified by the EU. Contaminated food categories exceeding the intervention level for that category are withdrawn from the market. For that reason, the individual dose via ingestion is always limited to a maximum.

The models in this Guide address solely releases to the atmosphere. DOVIS-A should be consulted for all other accident releases. For releases to the atmosphere, only the consumption of contaminated agricultural produce needs to be assessed.

Standard: pursuant to DOVIS-A it should be assumed that all consumed produce is grown in the immediate surroundings, i.e. the area within a radius of 25 km from the source. It is assumed that half of the leafy vegetables consumed by the representative person are grown in its – assumed – own vegetable garden. The remainder of the produce is contaminated with the average contamination level in the aforementioned immediate surroundings.

Standard: calculate the concentrations of radionuclides in the various food products in accordance with DOVIS-A and the supplements enclosed in Subsection 5.2.6 of this Guide.

3.5 Dose effects

The deterministic and stochastic effects that should be determined are death from deterministic effects and increase in risk of fatal cancers.

3.5.1 Stochastic effects

The UNSCEAR 2012 report [34] gives the following estimates for the age and gender-averaged risk of death from solid cancers and from leukaemia caused by exposure to ionizing radiation:

Table 3-3 Risk of death from solid cancers and from leukaemia caused by exposure to ionizing radiation (UNSCEAR 2012)

	immediate dose of 0.1 Gy	immediate dose of 1 Gy
Increase in risk of death from solid cancers	0.36 - 0.77%	4.3 - 7.2%
Increase in risk of death from leukaemia	0.03 - 0.05%	0.6 - 1%
Risk factor	3.9 - 8.2% per Gy	4.9 - 8.2% per Gy

UNSCEAR 2012 further states that the uncertainty in these statistics does not exclude the possibility that the risk factor for very small doses is much lower or even zero, as the low-dose risk factor is masked by the natural occurrence of solid cancers and leukaemia. The statistics exclude a much larger risk factor than the risk factors listed above with a high degree of certainty, as a large risk factor will not be masked by the natural occurrence of solid cancers and leukaemia.

ICRP-103 [35] states that the estimate of this risk factor remained unchanged since ICRP-60 [36] at about 5% per Sv. According to ICRP, at low doses (less than 100 mSv) or at lower dose rates (less than 3 mSv per hour) the risk of solid cancers decreases by a factor of 2. This, using the UNSCEAR estimates, yields a risk factor of between 2.1% and 4.85% per Sv at low doses or low dose rates.

Standard: it can be assumed that for level 3 PSAs the risk factor for adults is 5% per Sv and that the risk factor for children is 15% per Sv.

UNSCEAR 2012 does discuss the ICRP's proposed reduction factor (DDREF) for lower doses or low dose rates. The uncertainty at low doses seems too large for a proper substantiation of the DDREF value. This would appear to be of limited importance to level 3 PSAs, as the dose level at which the risk for beyond design basis accidents approaches the risk limit of 1E-6 per year is around 100 mSv and above. The standard risk factor (5% per Sv) is also close to the upper limit of the stated range of between 2.1% and 4.85% per Sv at low doses or low dose rate as based on UNSCEAR data.

Standard: DDREF is not used in the determination of the individual risks in level 3 PSAs.

Refinement: the DDREF recommended by the ICRP may be used in the determination of the stochastic risk.

3.5.2 Deterministic effects

Damage to the following organs and tissues caused by radiation can be fatal: bone marrow, the lungs, gastrointestinal tract, and skin. A high radiation dose to the brain can also be fatal. The organ or tissue subjected to the largest radiation dose depends on the relevant exposure pathways and the radionuclide composition of the plume.

UNSCEAR 2012 and ICRP-60 state bandwidths for the LD_{50} (dose that is lethal for half of the exposed individuals) and threshold dose T (the dose below which the fatal effect does not occur) for these fatal symptoms.

The dose is specified in Gy equivalents, whereby instead of the ICRP-60 radiation weighting factor w_R, the quality factor D_n for the concerned organ symptom and radiation type should be used.

Table 3-4 Lethal Dose (LD_{50}) for deterministic effects of brief exposure to penetrating low LET radiation

Symptom/organ	LD_{50} (Gy-Eq.)	Threshold T (Gy-Eq.)	Time until death
Red bone marrow syndrome	3 - 5*	~ 1*	30 - 60 days
Gastrointestinal tract (small intestine)	5 - 15*	~ 6*	6 - 9 days
Lungs	5 - 15	7 - 8	1 - 7 months
Nervous system	> 15	(> 10)	1 - 5 days
Skin**	(Not stated)	5 - 10	2 - 3 weeks**

* without medical care
** serious burns of a large area of skin caused by radiation can be fatal, although this is uncommon

The following choices have been made to arrive at a calculation model suitable for use in level 3 PSAs:

- With minimum medical care (treatment of inflammation, antibiotics), the LD_{50} and threshold dose T for the red bone marrow syndrome and gastrointestinal tract can be increased by a factor of about 1.5.
- Little data is available for the specification of the mathematical form of the dose-effect relationship. For pragmatic reasons it has been decided to select the following relationship between the mortality risk *r*, short term absorbed low LET dose *D*, shape factor *v* and LD_{50}:

$$r = 1 - e^{-\ln(2)\left(D/LD_{50}\right)^{v}}$$

- When selecting an LD_{50} and a T from the aforementioned ranges the value of the shape factor *v* is calculated such that the calculated risk of a syndrome that results in fatality at dose T is precisely 1%. The selected values for LD_{50} and T are in line with the values reported by ICRP and UNSCEAR.
- To assess the biological effect of other radiation types, RBE factors (D_n) should be taken into account. The internationally accepted RBE factors (ICRP-60: w_R) used to calculate the equivalent organ dose are based on stochastic effects. The RBE factors (D_n) for fatal deterministic effects are smaller than the RBE factors (w_R) for stochastic effects. For that reason, the equivalent organ dose for stochastic effects can be used conservatively in the calculation of the deterministic effects.
- When the exposure period exceeds a day, the effect of exposure will be less than when the same dose is received in one day or less. That effect can be taken into account by adopting a higher value for the LD_{50} for a longer exposure period. The degree of uncertainty of the increase in the LD_{50} is large. At a constant dose rate, the LD_{50} for red bone marrow for a period of 1 day, 7 days, 14 days, and 30 days is 4 - 4.5 Gy-Eq, 6.4 - 9 Gy-Eq, 7.6 - 9 Gy-Eq, and 9.7 to 18 Gy-Eq, respectively. The LD_{50} for the other organs increases so rapidly that, for level 3 PSAs, the dose received after the first 24 hours, at either a constant or decreasing dose rate, hardly contributes to the risk. The LD_{50} for lung syndrome, for example, at a constant dose rate for a period of 1 day, 7 days, 14 days, and 30 days is 15 Gy-Eq, 80 - 240 Gy-Eq, 100 - 240 Gy-Eq, and 150 - 500 Gy-Eq, respectively.

The dose and dose rates for accidents assessed in level 3 PSA are the highest during the first day. Deposited material will thereafter deliver a dose at a declining ground dose rate. When the ground dose rate is so high that deterministic effects could occur then the area will be evacuated within one day after the accident.

Red bone marrow syndrome will dominate the deterministic risk of the most frequent level 3 PSA source terms. In some weather conditions also exposure of the skin can make a substantial contribution to the risk. The gastrointestinal tract and lung syndrome will only make a substantial contribution to the risk when unusual source terms and exposure pathways are involved.

Standard: for the dose-effect relationship for deterministic effects, for exposure during one day, the parameter values in **table 3-5** can be used:

Table 3-5 Standard parameter values for the dose-effect relationship for the deterministic effects

Organ	LD_{50} (Gy-Eq)	shape factor v	Threshold T (Gy-Eq)	Consequence
Red bone marrow*	4	5	1.75	fatality
Lungs	10	7	5.5	fatality
Gastrointestinal tract*	14	5	6	fatality
Skin	20 (D_{50})	5	8.5	life threatening burns**

* with limited medical care
** it should be assumed that 5% of the individuals with life threatening burns will die.

For longer exposure periods, the LD_{50} can be increased. The computational methods for this are discussed in Subsection 5.3.1.

Requirement: when it is not possible to determine in advance which injury results in death, level 3 PSAs should analyse the effects to bone marrow, lungs, gastrointestinal tract, and skin.

Requirement: it should be assumed in the Netherlands that the victims of an accident will receive supportive medical care, at least blood transfusion and antibiotic treatment.

3.6 Countermeasures

An accident in a nuclear facility resulting in a release of radioactive material will trigger the facility's emergency plan. Each municipality with an operational nuclear facility has an emergency plan in place. The execution of the emergency plan will, depending on the accident's severity, be managed at a local, regional or national level. The Responseplan National Crisisplan Radiationincidents (2017, [37]) provides the framework for these municipal emergency plans, the regional and national nuclear emergency management and response organization, and the general approach to emergency management, such as the intervention levels for countermeasures.

The objective of countermeasures implemented in response to a radiation accident is to protect both humans and livestock and restore the situation as much as possible to its original state prior to the accident. The countermeasures should be tailored to the various types of exposure pathways and relate to direct and indirect exposure to radiation or to other consequences for humans and society. Protective measures can be classified into direct and indirect countermeasures.

Direct countermeasures

Direct countermeasures are focused on reduction of the direct exposure of humans to radioactive materials and radiation, for example from a radioactive cloud. Examples include sheltering, evacuation (immediate, early, or late), iodine prophylaxis, and skin decontamination within 24 hours. The execution of these countermeasures is complex and appropriate coordination of their implementation is required.

Indirect countermeasures

Indirect countermeasures are focused on the indirect exposure pathways from a radioactive release, such as the consumption of contaminated food. Examples of indirect countermeasures are access control, agricultural countermeasures to prevent contamination of the food chain, medical care, and psychosocial assistance. Indirect countermeasures can be announced and implemented either immediately after an accident or at a later date.

Standard: calculations of risks to be tested against regulatory limits should not take account of dose reductions due to implementation of direct countermeasures. This relates to immediate, first-day, or late evacuation, iodine prophylaxis, sheltering, and skin decontamination within 24 hours [38]. However, countermeasures to prevent ingestion of contaminated food may be taken into account in the risk calculation. Furthermore, it may be assumed that the other countermeasures will be implemented after one or more days when intervention levels are exceeded.

This takes account of the complexity of the implementation and coordination of direct countermeasures. Consequently, it is not clear in advance whether direct countermeasures can be implemented effectively and in due time and whether all individuals at risk can be reached in time. For that reason, the individual risk is determined by the group that cannot be reached when possibly countermeasures will be implemented. Also for the group risk it is impossible to determine in advance how effective countermeasures can be implemented.

In addition, the calculated group risk could be reduced considerably when direct countermeasures such as evacuation would be taken into account. This could then result in a situation in which a risk analysis shows that a minor accident, in which intervention levels for the implementation of countermeasures are not exceeded, could make a larger contribution to the group risk than a major accident in which direct countermeasures are taken into account. Finally, group risk should be also useful as measure for social disruption, and implementation of the early countermeasure evacuation would result in social disruption too.

3.7 Design basis accidents

The effectiveness of safety systems and other facilities designed to provide adequate assurance for the safety of local residents and employees should be assessed by means of a number of accident scenarios.

Each component of the facility – i.e. each process, each storage area, etc. – should be assessed on the basis of a list of postulated initiating events to determine which events could lead to an accident involving the component (failure scenario) and the response of the safety system to that accident. A distinction can then be made between the following two categories:

Design basis accidents

A design basis accident is an accident that has been taken into account in the design. A design basis accident relates to the failure of one or more safety systems due to a postulated initiating event with an internal or external cause, while the other systems then remain intact. It shall be demonstrated that the remaining operational systems adequately mitigate the consequences of that accident.

Beyond design basis accidents

These are accidents so improbable that the facility does not need to be designed as to fully prevent these accidents to happen or fully mitigate their consequences. However, the design of the facility does need to ensure that the overall risk of all beyond design basis accidents is below the legal limit. Identification and analyses of beyond design basis accidents is the subject of level 1, level 2, and level 3 PSAs.

In Article 6, under h, the Bkse Decree [8] requires:

> a safety report containing a description of the measures and countermeasures that will be taken by or on behalf of the applicant to prevent damage or mitigate the risk of damage, including the measures to prevent damage outside the facility during normal operations, and countermeasures to prevent damage resulting from the postulated initiating events mentioned in that description, as well as a risk analysis of the damage outside the facility as a consequence of those events;

and under i:

> a risk assessment of the damage outside the facility as a consequence of beyond-design-basis accidents;

The safety report contains, under the 'design basis accident' heading, a risk analysis of the damage outside the facility as a consequence of the postulated initiating events. The adequacy of the measures incorporated in the design to prevent damage outside the facility (often a defence in depth system) can be tested against the *dose* limits as specified in Article 18, paragraph 2, of the Bkse Decree [8].

Table 3-6 Dose limits for design basis accidents pursuant to Article 18, paragraph 2, of the Bkse Decree [8].

Event frequency F per year	Maximum effective dose allowed		Maximum thyroid dose allowed
	persons aged 16 years and over	persons under the age of 16 years	
$F \geq 10^{-1}$	0.1 mSv	0.04 mSv	500 mSv
$10^{-1} > F \geq 10^{-2}$	1 mSv	0.4 mSv	500 mSv
10-2 > F ≥ 10^{-4}	10 mSv	4 mSv	500 mSv
$F < 10^{-4}$	100 mSv	40 mSv	500 mSv

It should be noted that pursuant to the Bkse Decree the residual risk of design basis accidents does not need to be tested against the risk limits, as the dose limits for design basis accidents ([8] Article

18, paragraph 2) are so stringent that the (residual) risk limit requirements ([8] Article 18, paragraph 3) will automatically be met.

The DSR Guide [6] imposes more stringent preconditions for radiological objectives in view of the options available following technological developments and the WENRA recommendations. In addition, as referred to above the new DSR devotes more attention to the design basis for new reactors than in the past: the design basis now includes postulated initial events with multiple failure and postulated core-melt accidents that were previously classified as beyond design basis accidents.

The assessment of the safety systems to determine whether they provide adequate safety is then based on the dose limits specified in the Bkse Decree [8] and the supplementary dose limits in the DSR Guide [6].

Table 3-7 Bkse Decree [8] and DSR Guide [6] dose limits for the children focus group

Design basis accidents	Event frequency (per year)	Bkse decree (70 year committed dose)	DSR Guide (70 year committed dose)
Bkse: postulated initial events. **DSR: accidents without core melt, including anticipated operational occurrences (DSR safety levels 2, 3a, and 3b).**	> 1E-1	0.04 mSv	0.1 mSv
	1E-1 - 1E-2	0.4 mSv	0.1 mSv
	1E-2 - 1E-3	4 mSv	1 mSv
	1E-3 - 1E-4	4 mSv	10 mSv
	< 1E-4	40 mSv	10 mSv
			Countermeasure zones*
Postulated core melt accidents (DSR safety level 4)	Beyond design basis accident		> 3 km from site boundary: no evacuation necessary > 5 km from site boundary: no sheltering or iodine prophylaxis necessary

* The zones serve as design requirements in combination with the Dutch intervention levels. The following intervention levels (see [37]) apply to the calculated potential dose to be received within 48 hours (for evacuation and sheltering) or 7 days (for iodine prophylaxis) after start of the release: for evacuation, the intervention level is a potential effective dose of E ≥ 100 mSv; for sheltering, the intervention level is a potential effective dose of E ≥ 10 mSv; and for the distribution of iodine prophylaxis the intervention level for children is a potential thyroid dose of $H_{\text{thyroid},<18\text{ yr}} \geq 50$ mSv.

3.7.1 Computation methods for design basis accidents

The computation method for level 3 PSA source terms can also be used for the analysis of the dose resulting from design basis accidents. The computation method yields a statistical distribution of the doses outside the site boundary.

Requirement: the analysis of the dose from design basis accidents should be performed ‘probabilistically’. This means that the distribution of the maximum dose received in the various potential weather conditions should be determined for each weather scenario. The 95th percentile of the calculated distribution is tested against the dose limits specified in the Bkse Decree [8] and, when applicable, the DSR Guide [6].

For design basis accidents, the following 95th percentiles of the maximum doses for each weather scenario should be tested:

- the effective dose for adults with 50-year committed dose (Bkse Decree)
- the effective dose for children with 70-year committed dose (Bkse Decree, DSR Guide)
- the thyroid dose for adults (Bkse Decree)
- the thyroid dose for children (Bkse Decree)
- the effective dose for children more than three 3 km and 5 km from the site boundary, with a 48-hour committed dose (DSR Guide, safety level 4b)

When a statement on where a limit applies is missing it should be assumed that the focus group is present everywhere outside the site boundary. The location of the site boundary is determined by the nuclear licence, pursuant to Article 15, under b.

3.7.2 Differences from a level 3 PSA

The following differences between a level 3 PSA and the analysis of design accidents stand out:

- The design basis accident source terms are much smaller than those of beyond design basis accidents (PSA). This means, among other things, that for design accidents no deterministic consequences will occur.
- Beyond design basis accidents are tested against the risk criteria, whilst design basis accidents are tested against a dose criterion. Consequently, an analysis of the dose *effects* is not carried out for design basis accidents.
- Beyond design basis accidents that do not result in deterministic effects are tested against the maximum individual risk. This individual risk is proportional to the meteorological conditions averaged dose. For a similar design accident, the 95th percentile of the dose distribution is presented and tested. Consequently, the dose results from an analysis of a design basis accident (95th percentile dose) cannot be converted to a risk or conditional risk (average dose x risk factor) without data about the complete dose distribution.
- The risk from an external cloud dose usually makes only a relatively small contribution to the total risk in level 3 PSAs. However, that exposure pathway can be the dominant pathway for certain design basis accidents, in particular for accidents resulting in pure noble gases releases.

3.8 Uncertainty analyses

An estimate of the accuracy – also referred to as the ‘uncertainty’ – of the presented results may be required for the interpretation of the results from level 3 PSAs. The uncertainty of the results of a PSA is a combination of the uncertainties of the results of the level 1, level 2, and level 3 PSA. Consequently, the uncertainty of the level 1 and level 2 PSA (source term frequency, size, and nuclide composition) serves as input for the uncertainty analysis of the results of level 3 PSA.

An uncertainty analysis of a consequence analysis should be carried out as an examination of the outcome of parameter variability. In order to also consider model uncertainty in the analysis, its uncertainty is attributed to a parameter relevant to that model. Uncertainty analyses focus on determining the confidence limits of the consequences and the sensitivity of the consequences to changes in individual parameters.

When a level 3 PSA is carried out using the models and parameter values specified in the Vbs Regulation, DOVIS-A, NNM, and the present Guide, there is no procedural uncertainty regarding the level 3 PSA results to be tested. The question is then to what extent the doses and risks calculated using the specified models and parameter values may differ from the actual doses and risks.

The literature reveals that there is a high degree of uncertainty of the dispersion, dose, and consequence calculations for a specific accident and specific weather conditions. The measured contamination and dose, and the risks determined from epidemiologic studies can vary by one to two orders of magnitude from the predicted levels calculated with the customary models. This is largely due to the much greater local variation in air dispersion, land use, way of life, and the physiological characteristics of individuals than is taken into account in the calculation models.

Level 3 PSAs assess a wide range of weather conditions and, usually, several source terms. A statistical method is then used to pool the effects on various groups of individuals in various circumstances. It is plausible that this statistical pooling largely flattens out the effect of the large local variations by source term and weather condition. The effect on a group of individuals that is twice as sensitive to radiation as assumed in the calculation is, for example, covered by the situation in which the wind speed is halved. It is expected that the ultimate uncertainty due the stochastic character of the level 3 PSA results will be much smaller, probably less than one order of magnitude.

In this process attention should be devoted to systematic deviations. A model or parameter value that systematically underestimates an effect can yield results that underestimate the doses, probability of health effects, and risks. Nevertheless, even that can be compensated by the use of essentially conservative models and parameter values. This yields results with a certain degree of robustness against any systematic underestimations that might be identified after the performance of the calculations.

4 Performance and presentation

The previous chapter discussed the models that should be incorporated in a computer code to conduct probabilistic consequence analyses. The present chapter addresses the actual performance of a probabilistic consequence analysis in practice. The chapter first discusses the collection of the input data followed by the presentation of the results.

4.1 Selection of computer code

Between the mid-eighties and mid-nineties, a range of national and international level 3 PSA computer codes were developed. Many tens of man-years were required to develop each computer code. At that time, also benchmarks were carried out, such as the NEA benchmark [39], which compared the COSYMA (EC), Arano (Finland), Lena (Sweden), Condor (UK), MACCS (USA), MECA2 (Spain), and OSCAAR (Japan) computer codes. Between 1993-2000, the US Nuclear Regulatory Commission (USNRC) and the European Commission (EC) organized a project involving the very wide scale international application of COSYMA and MACCS, to obtain the best possible insight into the accuracy of the predictions of level 3 PSA computer codes.

Other internationally available computer models are RODOS (which began as a spin-off of the COSYMA team and now has been developed into an emergency management support product, not suitable for level 3 PSAs anymore) and HotSpot [40] (primarily intended for emergency management support and with limited level 3 PSA applicability). Only the MACCS, RODOS, and HotSpot computer codes have been developed further in the past 10 years.

The COSYMA and MACCS2 source codes are freely available in the EU and associated states. These computer codes are used by about 50 organizations in various Member States. In view of the wide international acceptance of COSYMA and MACCS2, the wealth of experience regarding their application and the available experience in the comparison of the codes and use in uncertainty studies, this Guide exercises restraint, whenever feasible, in prescribing requirements for the implementation of modifications to the international computer models.

Nevertheless, some developments in the Netherlands that are of relevance to level 3 PSAs should be incorporated in level 3 PSA calculations. For that reason, this Guide adopts the following principle: *Users can exercise their discretion in accepting any <u>overestimation</u> of an endpoint resulting from 'international practice', or can opt to follow the advice given in this Guide. However, when 'international practice' results in an <u>underestimation</u> of the group risk and maximum individual risk endpoints then users should comply with the regulations prescribed in this Guide.*

The general form of the COSYMA and MACCS2 models is in line with the description in this Guide. However, there are differences at detail level and in the selection of parameter values. One can state – subject to certain conditions – that the results of COSYMA/MACCS2 calculations sufficiently agree with the results from calculations carried out in accordance with the models as described in this Guide. This has been investigated in more detail in a topic study for this Guide [15].

In the Netherlands, a precise implementation of the computational models of the Vbs Regulation [10], DOVIS-A [11], and NNM [5] has been incorporated in the NUDOS2 [18] computer code. That computer code can be used to calculate the dose from routine airborne emissions as well as the doses and risks resulting from accident emissions with a clear phased emission progression.

Section 5.4 discusses the conditions under which computer codes like COSYMA and MACCS2 can be used for level 3 PSAs as well as for analysis of design basis accidents.

4.2 Input data

For a probabilistic consequence analysis, very many input data have to be collected. As the computer code user can manipulate the input data and exert a great influence on the results of the analysis it is necessary to use an appropriately structured and documented process to collect the input data. The following subsections specify the input data that must be available and should be qualified and quantified prior to the analysis. The input data can be roughly classified into the following classes:

- Model parameters
- Material constants such as the half-life by radioactive decay, the energy of emitted γ radiation, and the produced daughter nuclides
- Nuclide-specific data, such as dose conversion coefficients. This data results from calculations by other, often highly complex, computer codes.
- Meteorological data. A distinction is made between the dispersion model parameters (such as the sigma values of the Gaussian distribution) and the hourly weather data input for the probabilistic analysis.
- Population and land use data.

Most of the parameters for level 3 PSAs are laid down in or determined by the Vbs Regulation [10], DOVIS-A [11], and NNM [5]. Other parameters are specific to the computer code that is used.

The following subsections discuss the data for the source term, atmospheric dispersion and deposition, dose calculation and health effects, population data and countermeasures.

4.2.1 Source term

All consequence analyses begin with the source term. The definition of ‘source term’ encompasses the composition of the release, the manner in which the release occurs and the frequency/likelihood of the accident release. This relates to data about the release height and building dimensions, the released amount, chemical composition, particle size distribution, heat content, and humidity of the release, the delay time, the release duration, and, finally, the probability of the source term release. These have already been reviewed in detail in § 3.2 and are not discussed further in this section.

4.2.2 Meteorological data

The NNM contains a database and an algorithm to generate a set of representative meteorological data for any location in the Netherlands, sufficiently accurate for predictions for the dispersion calculation.

For reporting, it is sufficient to provide a brief summary of the most important meteorological statistics for the location, such as the wind speed, wind direction, rainfall, and a stability and height indication of the mixing layer.

4.2.3 Population data and land use

The latest population data and land use information available should be used for the group risk calculation. The land use also describes the temporary population, such as people working at an industrial park in the vicinity or being present in a nearby recreational area. Accurate specifications of the site boundary are needed for the calculation of the individual risk and – for design basis accidents – the doses. In the report, this information can be presented by means of maps and summary tables.

4.3 Presentation of the results

The results of a probabilistic consequence analysis are also the final results of a complete – level 1 to 3 – probabilistic risk analysis. The usual presentation of probabilistic results concerning sizes (e.g. number, area size) determined for the entire calculation grid is a complementary cumulative distribution function (CCDF) curve. A CCDF curve can also be used to present the distribution of results at a particular point on the grid, such as the individual dose. A CCDF curve is constructed from the results of a large number of repeated calculations in which the following variables are varied:

- The source term,
- Weather sequences (stability, wind speed, wind direction, and rain intensity)

For example, for each combination of a source term and weather sequence the number of fatalities due to deterministic and stochastic effects can be calculated. This number, together with the combined frequency of occurrence of the source term and the weather sequence, contributes to the CCDF curve. These calculation results can not only be used to construct a CCDF curve, but can also be used to determine the – population independent – individual-risk contours.

In addition to the 'deterministic and stochastic fatalities' endpoints most frequently used, a large number of other endpoints are also conceivable, for example:

- deterministic effects such as illness and non-fatal tumours,
- number of individuals in each dose band,
- hereditary effects,
- size of non-accessible contaminated area,
- financial loss,
- numbers of individuals for early or late evacuation.

The primary objective of level 3 PSAs as discussed in this Guide is to provide the results to be tested against the risk criteria. This means that at least the maximum individual risk and the group risk should be presented. For the group risk only deterministic effects have to be assessed, for the individual risk assessment of both deterministic and stochastic effects is required. In addition to these testable results, the report will often need to present the collective dose and large-scale ground contamination, as well as the influence of countermeasures on those endpoints. Alongside the end results discussed in the following paragraphs, the report should also state the data and intermediate results required to perform calculation checks.

4.3.1 Individual risk

The endpoint to be tested against the requirements is the maximum individual risk. This risk is comprised of the contributions from all combinations of source terms and weather sequences and encompasses the mortality risk from both deterministic and stochastic effects.

Requirement: the maximum individual risk to be presented and tested against the risk criteria is a single number. The location of this risk should be stated. This requirement cannot be met with the existing level 3 PSA computer codes: modifications are required.

Refinements:

- It is recommended that individual risk contours are presented to provide insight into the distribution of the individual risk in the vicinity of the nuclear power plant. The maximum individual risk can be indicated in these contours.

- As the occurrence of deterministic and stochastic effects is determined by various processes that differ in sensitivity to countermeasures, it is recommended that the maximum individual risk and the risk contours are presented separately for deterministic and stochastic effects.
- Further insight can be provided by presenting the contribution by exposure pathway and by nuclide.
- Feedback to the level 2 PSA can be created by presenting the conditional individual risk by source term for at least the location of the maximum individual risk. This may be presented in the form of a table or CCDF curve. This form of presentation provides insight into the risk contribution determined by the probability of the source term and that of the radiological consequences of that source term. This presentation can also be supplemented with separate presentations of the deterministic and stochastic effects contribution.

4.3.2 Group risk

Requirement: the required endpoint is the CCDF curve of the number of fatalities due to deterministic effects. This risk comprises the contributions from all combinations of source term scenarios and weather sequences.

Refinements:

- The presentation can include the CCDF curves for each source term, providing insight regarding the contribution of the various source terms to the group risk.
- When countermeasures are taken into account, their impact on the group risk can be included in the presentation, showing the influence of each separate countermeasure and that of the combination of all countermeasures.
- The usual form for presentation of the group risk (CCDF curves) can also be used to present the extent of the countermeasures (number of individuals evacuated, size of contaminated area, etc.) for the various source terms.

4.3.3 Collective dose

The collective dose is the integral of all individual doses to members of the population and, consequently, consists of a single figure for each source term and weather scenario. In the Netherlands, there is no test criterion for the assessment of collective dose.

- The collective dose can be presented in the same manner as the group risk, using CCDF curves. The collective dose – alongside the group risk for the assessment of the extent of the deterministic effects resulting from severe accidents – primarily gives insight into the extent of the stochastic effects resulting from both minor and severe accidents.
- A presentation of the relative contributions of the individual dose bands and the exposure pathways can be included to provide for an evaluation of countermeasures.

4.3.4 Ground contamination

In the Netherlands, no test criteria for the assessment of the size of contaminated areas apply. The size of the contaminated area is a measure of the potential economic loss and serves as a supplement to the group risk and collective dose when the area around the nuclear power plant is not populated. The size of the contaminated area is a single figure for each source term, weather scenario and contamination level.

The ground contamination can be presented in the same manner as the group risk, using CCDF curves.

4.4 Reporting

The report should supplement the presentation of the numerical results specifying the framework within which the analysis was performed. This framework is determined by the objective of the analysis, the methodology, and the input data used for the analysis.

4.4.1 Objective of the analysis

The report should begin with an explanation of the reasons for performing the analysis. The results from a level 3 PSA can be used in the following applications:

- Quantitative risk analysis
- Comparison of sites
- As part of an environmental impact report (Dutch: MER)
- Support in the development of regulations
- Evaluation of design modifications and procedures
- Development and evaluation of countermeasures
- Testing design basis accidents for the purposes of the safety report

‘PSA’ is technically not the correct term for some of the aforementioned applications. The calculations will not always be probabilistic calculations (such as in the evaluation of countermeasures) or the endpoint will not be the mortality risk (for design basis accidents, calculations are probabilistic but the analysis ends with the dose calculation).

When results are to be tested against quantitative risk criteria then the report should state the criteria against which the results will be tested and the testing method. In addition to testing the extent of specific consequences (in the Netherlands, only fatalities from deterministic and stochastic effects) against quantitative standards, it is desirable to provide insight into other consequences, such as the size of the contaminated area and the economic loss, as well as the influence of countermeasures. For a proper assessment of the final results, it is desirable to indicate, in advance, how those supplementary results will be assessed.

4.4.2 Principles and methods

Requirements:

- Analysts performing a level 3 PSA should provide insight into the principles adopted for the analysis, including the computer code that was used and the options that were selected from the many via its input available options, together with an explanation why those options were selected.
- Detailed information should be enclosed about any modifications to the internationally available computer code that the analyst held to be necessary, together with the reasons for each modification. This should be accompanied by a statement of the influence of the modifications on the results from the calculations.
- The report should also include a description of any computer code used other than generally available level 3 PSA computer codes such as MACCS2 and COSYMA. That description shall be given in the detail required to test the methods and models against this Guide.

4.4.3 Input data

The input data that analysts performing level 3 PSAs select and the manner in which they process the data can have a great influence on the results of the analysis. For this reason, the report should state the method used to prepare the input data. The description of the MACCS2 input data in [41], prepared in the context of "Computer Code Application Guidance for Documented Safety Analysis", is a good example of a systematic description of the process involved in the collection of input data.

Requirement: analysts performing a level 3 PSA should give a full description of the input data, including:

- the origin of the data, with references,
- the manner in which the data has been processed,
- when only limited data is available, the manner in which supplemental input has been provided,
- a summary of all input data that has been used.

Analyses that are to be tested against the Dutch risk approach should be accompanied by a description of at least the following input data:

- the source terms,
- the meteorological data,
- a description of the location,
- population data.

Additional data may also be required, depending on the code being used. When the presentation of the results includes a presentation of the influence of countermeasures on the health risk or the financial risk, also a description of the following input data should be given:

- countermeasure scenarios and location-specific evacuation data,
- economic data.

4.4.4 Results and testing the results

The results should be presented in the manner laid down in Subsection 4.3. This can be followed by a presentation of the testing of the results against the standards, see Subsection 2.5, and a qualitative discussion of the uncertainties.

5 Detailed description of the models

The models and data are described in detail in the Vbs Regulation [10], DOVIS-A [11], and NNM [5]. For this reason, this chapter only describes the modifications and supplements to those models required for performing of level 3 PSAs.

5.1 Atmospheric dispersion and deposition

A comprehensive description of the atmospheric dispersion and deposition models of the NNM [5] and alternatives for the NNM models' limitations for use in accident releases is available in [13]. In summary, the aforementioned NNM models can be used for consequence analyses of accident releases of radioactive materials virtually without adaptation. As the NNM has been developed primarily for semi-continuous sources, this Guide deviates on some points from the NNM modelling for the finite releases that are assessed in level 3 PSAs. The NNM also lacks some information specifically required for performing dispersion analyses for accident releases from nuclear reactors. This comprises the following:

- Virtual source model for increase in σ parameters
- Partitioning of the plume in case of partial breakout of the mixing layer due to plume rise
- Wet deposition of aerosols and gases due solely to washout
- Deposition of gaseous iodine

5.1.1 Virtual source model for increase in σ parameters

The NNM equation-set for the calculation of the dispersion parameters is based on a bi-Gaussian plume from a point source with a fixed transport velocity. However, actual sources are of finite dimensions. Hence the NNM [5] gives instructions for the influence of plume rise (§ 3.7.2) and building wake (§ 4.5) on dispersion on the basis of initial values of σ_y and σ_z. However, there is no provision in the NNM on how to combine those initial values with the rules for determining the growth in $\sigma_y(t)$ and $\sigma_z(t)$ for point sources. In fact, due to the link between the transport velocity of the bi-Gaussian plume and the centre of gravity the NNM models cannot be used even for point sources without modification.

The following determines that the customary 'virtual source' model [20] is to be used for the determination of the dispersion parameters. This is based on the modelling of a virtual source that is located at a position upwind of the actual source such that the plume dimensions of the bi-Gaussian plume from the virtual source at the location of the actual source are exactly equal to the initial dimensions of the source (see Figure 5-1).

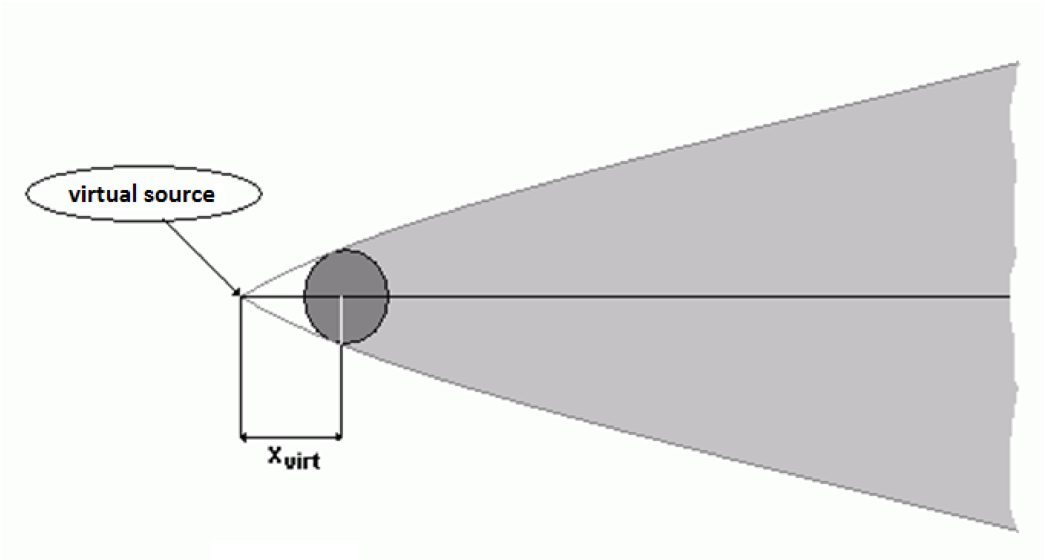


Figure 5-1 'Virtual source' model

Explanation of the 'virtual source' model:

- Given the dimensions $\sigma'_{y,z}(x)$.
- Based on the transport velocity u and turbulence conditions at transport height the distance $x_{virt;y,z} = u \cdot \tau_{y,z}$ to a virtual source is selected such that $\sigma'_{y,z}(x) = \sigma(\tau_{y,z})$: in principle, $\tau_y \neq \tau_z$.
- $\tau_{y,z}$ is used to calculate $\sigma_{y,z}$ at distance $dx = u \cdot dt$: $\sigma'_{y,z}(x + dx) = \sigma_{y,z}(\tau_{y,z} + dt)$

5.1.2 Partitioning of the plume on partial breakout from the mixing layer

The plume may rise when the temperature of the release is higher than ambient temperature. The extent of the plume rise, which is dependent on the heat content and on factors like stability and wind speed, is calculated using the procedure laid down in the NNM. A plume with sufficient heat content can partially or entirely break through the mixing layer. According to the NNM ([5], § 3.9), the fraction P that remains within the mixing layer when the plume partially breaks through the mixing layer can be calculated with the equation:

$$P = \max[\ 0\ ,\ \min\ \{\ 1\ ,\ (z_i - H_s - \Delta h/2) / \Delta h\ \}\ ]$$

where:

z_i	height of the mixing layer
H_s	height of the chimney / ventilation stack
Δh	the calculated plume rise

A plume that partially breaks through the mixing layer (i.e. $P \neq 0$ or 1) is divided into a sub-plume *within* the mixing layer and a sub-plume *above* the mixing layer. However, the NNM does not yield a complete description of both sub-plumes. That description is therefore specified below.

The NNM ([5], § 3.7.2, equation (50)) yields initial values for σ_y and σ_z due to plume rise: $\sigma_y^0 = \sigma_z^0 = \Delta h/3.5$ at plume axis height H_{axis} (= $H_s + \Delta h$). This specifies the initial concentration profile to be used in calculating the two sub-plumes. To harmonize the combined concentration profile of both sub-plumes:

- the plume axis height of both sub-plumes is taken to be:

  H^{-}_{axis} = min (H_{axis} , z_i) plume axis height of the sub-plume *within* the mixing layer
  H^{+}_{axis} = max (H_{axis} , z_i) plume axis height of the sub-plume *above* the mixing layer

- the initial value of σ_y of both sub-plumes is taken to be equal to: $\sigma_y^0 = \Delta h/3.5$ (pursuant to the NNM instructions)
- the initial value of σ_z of the sub-plume with plume axis <u>unequal</u> to z_i is taken to be equal to the value pursuant to the NNM instructions: $\sigma_z^0 = \Delta h/3.5$
- the initial value of σ_z of the sub-plume with the plume axis <u>equal</u> to z_i is calculated from the initial concentration profile:

$$\sigma_z^{-0^2} = \int_0^{z_i} (z - z_i)^2\, C(z; H_{as}; \sigma_z^0) dz, \text{ when } H^{-}_{axis} = z_i,$$
$$\sigma_z^{+0^2} = \int_{z_i}^{\infty} (z - z_i)^2\, C(z; H_{as}; \sigma_z^0) dz, \text{ when } H^{+}_{axis} = z_i.$$

  with C(z; H_{axis}; σ_z^0), the crosswise integrated bi-Gaussian plume around H_{axis}, with the initial σ_z^0 prescribed by the NNM
- an extra factor 2 is inserted in the Gaussian equation for the sub-plume with the plume axis equal to z_i to ensure that the amount of material in this sub-plume (between 0 and z_i, or, between z_i and infinity) is equal to the amount of material in the mixing layer.
- an extra factor $C_{ls} \geq 1$ is inserted in the Gaussian equation for the sub-plume with the plume axis unequal to z_i which, when $\sigma_z > \sigma_z^0$, compensates for the 'numeric' loss of material in the zone of the other sub-plume.
- the transport velocity and growth in the σ parameters for both sub-plumes are determined following the NNM instructions.

5.1.3 Wet deposition of aerosols and gases

The NNM deposition models for aerosols and gases are largely based on the Dutch National Institute for Public Health and the Environment (RIVM) OPS dispersion program [42]. That program has primarily been developed for analyses of long-distance dispersion of air pollution, with specific attention to acidifying pollutants such as NO_x, SO_2 and NH_3 and their secondary products. These gases disperse over great distances, which results in a fairly uniform concentration profile both within and above the mixing layer.

OPS – as NNM – distinguishes between two wet deposition processes for aerosols and gasses:

- Washout, when material in the atmosphere is entrained in falling raindrops, and

- Rainout, when the concentration in the cloud droplets is in equilibrium with the concentration in the atmosphere.

The latter deposition process is the dominant process for pollutants assessed in OPS, in particular because the long-distance transport offers sufficient time for the concentrations in the atmosphere and cloud water to reach the equilibrium. However, the short distances involved in consequence analyses of accident releases (distances of up to about 10 km) do not offer sufficient time for the development of such a uniform concentration profile up till and above the mixing layer or for the attainment of that equilibrium. This is also the case for the sub-plume above the mixing layer when part of the plume rises and breaks through the mixing layer (see § 5.1.2).

For that reason, wet deposition following an accident release is modelled slightly different but analogous to the NNM approach:

$$F(x,y) = \Lambda_{washout}\left[\frac{\sigma_z^-}{(\sigma_z^- + 15)}\frac{Q^-(x)\,e^{-\frac{y^2}{2\sigma_y^{-2}}}}{u^-\sqrt{2\,\pi\,\sigma_y^-}} + \frac{\sigma_z^+}{(\sigma_z^+ + 15)}\frac{Q^+(x)\,e^{-\frac{y^2}{2\sigma_y^{+2}}}}{u^+\sqrt{2\,\pi\,\sigma_y^+}}\right],$$

where:

u^+, u^-	the wind speed at centre of gravity height above and within the mixing layer
Q^+, Q^-	the distance-corrected source strengths above and within the mixing layer
σ_y^+, σ_y^-	the horizontal plume dilution above and within the mixing layer
σ_z^+, σ_z^-	the vertical plume dilution above and within the mixing layer
$\Lambda_{washout}$	the washout coefficient for aerosols or gases according to the NNM.

5.1.4 Deposition of gaseous iodine

The NNM states parameter values solely for the deposition of gases of relevance to air pollution, such as SO_2 and NO_x. However, in accident releases of radioactive material, gaseous iodine can also be of importance, for which the NNM does not provide all necessary parameter values.

The NNM can be used to determine the resistance values R_a and R_b for dry deposition of gaseous iodine. R_a, the resistance to turbulent transport to the semi-laminar layer is independent of the type of gas. R_b, the resistance to diffusive transport in the semi-laminar layer demarcated by z_0, is proportional to the cube root of the molar mass (OPS [42], equation (5.8)) and can be calculated from $R_b(SO_2) = 7.2225/u^*$ (on the basis of [5], equation (81), Table 1 and adjacent text).

For elemental and organic iodine gas, DOVIS-A gives standard values for the surface resistance R_c for determination of the deposition rate for dry deposition and the diffusion coefficient D_g for determination of the washout coefficient for wet deposition. Consequently, for gaseous iodine the following turbulence-dependent deposition rates and washout coefficients should be used:

Table 5-1 Deposition rates and washout coefficients for gaseous iodine

[R in mm/h, D_g in cm^2/s]	Elemental iodine	Organic iodine
DOVIS-A standard values	R_c = 50 s/m, D_g = 0.2 cm^2/s	R_c = 5000 s/m, D_g = 0.05 cm^2/s
$v_d = (R_a + R_b + R_c)^{-1}$	$R_b = R_b(SO_2) * \left(\frac{M(I_2)}{M(SO_2)}\right)^{1/3} = \frac{11.43}{u^*}$ $v_d = \left(R_a + \frac{11.43}{u^*} + 50\right)^{-1}$ m/s	$R_b = R_b(SO_2) * \left(\frac{M(CH_3I)}{M(SO_2)}\right)^{1/3} = \frac{9.41}{u^*}$ $v_d = \left(R_a + \frac{9.41}{u^*} + 5000\right)^{-1}$ m/s
$\Lambda_{washout} = 1.14 \cdot D_g^{0.74} \cdot R^{0.64}$	$\Lambda_{washout} = 1.14 \cdot 0.2^{0.74} \cdot R^{0.64}\ h^{-1}$	$\Lambda_{washout} = 1.14 \cdot 0.05^{0.74} \cdot R^{0.64}\ h^{-1}$

5.2 Dose calculation

Dose calculations are carried out using the DOVIS-A [11] models and parameter values. As DOVIS-A has been developed to calculate the radiological effects of routine releases, it is limited to the 'effective dose' measure for the focus group 'adult male'. However, besides calculation of effective dose for the performance of level 3 PSAs and design basis accident analyses also calculations of organ doses are required for testing against the group risk resp. thyroid dose criteria. The assessment should be carried out for the 'adult' focus group and at least supplemented with the 'one-year-old children' focus group. The parameter values required in addition to the DOVIS-A parameter values are specified below.

5.2.1 External exposure by γ radiation from the cloud

Individuals within or close to a passing plume receive a direct radiation dose from this exposure pathway (cloudshine, hereinafter referred to as 'cloud dose'). Due to the nature of the exposure this dose is received within a short period of time, which means that this dose is of particular importance for the calculation of deterministic effects or the size of countermeasure zones.

The dose models for this exposure pathway are usually based on submersion, whereby the dose received by each organ is calculated by multiplying the time-integrated concentration of each radionuclide in the air by a dose conversion coefficient for the specific organ. These predetermined dose conversion coefficients are based on calculations that assume a semi-infinite cloud with uniform concentration (the 'submersion dose').

A correction factor still has to be applied in the calculation of the γ dose in the field to take account of the plume geometry and the distance from the plume axis. These predetermined correction factors are of particular importance to calculations for distances of less than about 20 km from the release point, when the plume has still limited dimensions.

Furthermore, in principle, a shielding factor can be applied to account for the shielding effect provided by buildings for that part of the population that is indoors.

The individual cloud dose received by organ *o* is then:

$$D_{Cld,o} = F_{cld,ext} \sum_{k} TIC_k PCF_k DCC_{Cld,o,k}$$

where:

$D_{Cld,o}$	Cloud dose received by organ *o*	(Sv)
$F_{Cld,ext}$	Shielding factor for being indoors	(-)
TIC_k	Time-integrated concentration of nuclide *k* (at centre of gravity height)	(Bq.s.m^{-3})
PCF_k	Plume shape correction factor for nuclide *k*	(-)
$DCC_{Cld,o,k}$	Submersion dose coefficient of nuclide *k* for organ *o*	(Sv/s) / (Bq/m^3)
k	Index for nuclide	

The parameter values for this calculation are specified in the Vbs Regulation and DOVIS-A. The determination of the plume shape correction factors is discussed below.

Standard: pursuant to DOVIS-A: The difficulty in calculating the precise concentration at each height complicates the accurate calculation of the cloud dose. For that reason, the standard method assumes that the cloud is already in contact with the ground and that the individual exposed to radiation is exposed to a 'semi-infinite cloud' containing the relevant released radionuclide. This means that a homogeneous concentration is assumed.

Refraining from the use of the plume correction factors results, for the first hundreds of metres of the cloud, in an overestimation of the cloud dose rate by at least a factor of 2 to 5. Outside of the cloud, the DOVIS-A model may yield much greater overestimations. This will be the case, for example, with radiation from a plume from an elevated release where the concentration in air at ground level is virtually zero: pursuant to DOVIS-A, the submersion dose should be based on the concentration at the plume's axis.

Refinement: a cloud shape correction model can be used to compensate for the standard method's occasionally great overestimation of the cloud dose. This is already incorporated in most level 3 PSA models.

The most transparent model is described in NUREG/CR-4691 [33], which is written for γ energies from between 50 keV to about 0.7 MeV.

Table 5-2 Plume correction factors, NUREG/CR-4691

$\sqrt{\sigma_y \sigma_z}$ (m)	*Distance to plume axis in units of effective plume size* $\sqrt{\frac{y^2+z^2}{\sigma_y \sigma_z}}$					
	0	1	2	3	4	5
3	0.020	0.018	0.011	0.007	0.005	0.004
10	0.074	0.060	0.036	0.020	0.015	0.011
20	0.150	0.120	0.065	0.035	0.024	0.016
30	0.220	0.170	0.088	0.046	0.029	0.017
50	0.350	0.250	0.130	0.054	0.028	0.013
100	0.560	0.380	0.150	0.045	0.016	0.004
200	0.760	0.511	0.150	0.024	0.004	0.001
400	0.899	0.600	0.140	0.014	0.001	0.001
1000	0.951	0.600	0.130	0.011	0.001	0.001

When the cloud dose contributes less than 25% to the total dose no modification of this model is required. Topical study [14], *Application of DOVIS A to accident emissions,* describes a variant to this approach that can be used for all relevant γ energies (from 50 keV to 10 MeV).

5.2.2 External exposure by γ radiation from the ground

Individuals present on ground contaminated with radioactive material deposited from the passing cloud receive a radiation dose from this exposure pathway (groundshine, hereinafter referred to as 'ground dose'). The dose models developed for this exposure pathway are based on exposure to radiation from a uniformly contaminated infinite flat surface. A correction factor for the calculation of the dose by γ radiation in the field should be used to take account of the natural roughness of the surface. Alongside radioactive decay, also the time dependency of the dose rate on transport of activity to deeper layers of the soil should be modelled. Both corrections have either already been incorporated in the pre-calculated set of dose conversion coefficients (for example, COSYMA) or need to be entered separately (for example, MACCS2).

In the latter case it is necessary to adopt a nuclide-independent surface roughness correction factor of 0.5, and the time-dependent correction factor C(t) for transport to deeper layers of the soil should be calculated using the equation:

$$C(t) = 0.5\, e^{-1.39t} + 0.5\, e^{-0.0077t}$$

with t in years. This model was drawn up by Gale [30]. The values are from [43].

In addition, in the calculation of the ground dose a shielding factor may be applied to take account of the shielding effect that buildings provide to individuals who are indoors . The individual dose received by organ *o* is then given by:

$$D_{Gr,o,T} = F_{Gext} \sum_{k} C_{t_0 k} DCC_{Gr,o,T,k}$$

where:

$D_{Gr,o,T}$	Ground dose for organ *o*	(Sv)
F_{Gext}	Shielding factor for staying indoors	(-)
$C_{t0,k}$	Initial deposition of nuclide *k*	(Bq.s.m^{-2})
$DCC_{Gr,o,T,k}$	Ground dose conversion coefficient of nuclide *k* for organ *o* during period *T*	(Sv/s) / (Bq/m^2)
k	Index for nuclide	

The contribution of β radiation to the ground dose may be neglected.

5.2.3 Internal exposure by inhalation of radionuclides from the cloud

This exposure pathway leads to a dose for individuals within a passing plume. The exposure period, as is the case with the cloud dose, is very short, namely the time the individual is in the cloud. However, as radionuclides remain in the body after the cloud has passed, the irradiation period is longer than the exposure period (committed dose).

The models used for the calculation of the inhalation dose are based on the product of the time-integrated concentration of radionuclides in the air, the volumetric inhalation rate, and the dose conversion coefficient. Both latter parameters are age dependent.

The formula used to calculate the committed dose received by organ *o* resulting from inhalation from the cloud is:

$$D_{Inh,o} = B\, F_{F,Inh} \sum_{k} TIC_k DCC_{Inh,o,k}$$

where:

$D_{Inh,o}$	Inhalation dose received by organ *o*	(Sv)
B	Volumetric inhalation rate	(m^3 s^{-1})
$F_{F,Inh}$	Filter factor whilst indoors	(-)
TIC_k	Time-integrated concentration of nuclide *k* (at ground level)	(Bq.s.m^{-3})
$DCC_{Inh,o,k}$	Inhalation dose conversion coefficient for nuclide *k* for organ *o*	(Sv/s) / (Bq/m^3)
k	Index for nuclide	

Depending on the period for which the committed dose is to be calculated, the correct pre-calculated DCCs should be selected. The calculation does not take any account of the filtration effect of homes (filter factor $F_{F,Inh}$ = 1). However, in calculations concerning the countermeasure sheltering a filter factor should be included. The standard filter factor is 0.5. It is recommended to divide the population into groups by type of behaviour and associated shielding level.

The available sets of dose conversion coefficients are calculated assuming that the nuclides are present as specific chemical compounds (usually oxides), usually as aerosols. Some computer codes also take account of the particle size distribution of the aerosol: sets of dose conversion coefficients are available for various particle size distributions. However, as data on the particle size distribution of a source term is usually not available most dose conversion coefficients are calculated for an aerosol AMAD of 1 µm.

5.2.4 External exposure to radiation due to deposition on skin

This exposure pathway results, in particular, in a β radiation skin dose from radionuclides deposited on skin and clothing. Exposure occurs during the period spent within the cloud until the skin is washed to remove the contamination. Level 3 PSA computer codes often use models such as the models drawn up by Henrichs [32] or Eckerman [26]. Those models assume that the amount of radioactive material deposited on skin and clothing is proportional to the amount deposited on the ground. The skin dose is calculated using dose rate conversion factors and the time integral from the time of contamination to the time of decontamination by washing.

This can be one of the most important exposure pathways for radiation resulting in deterministic effects. The dose is highly dependent on the deposition processes.

The equation for the calculation of the skin dose is:

$$D_{Skin,T_{skin}} = P_{Skin}\, F_{Skin,ext} \sum_k C_{g,k} c_k DCC_{Skin,k}\, U_{T_{skin},k}$$

with

$$U_{T_{skin},k} = \frac{T_{eff}}{\ln(2)}\left(1 - e^{-\frac{\ln(2)}{T_{eff}}T_{skin}}\right)$$

where:

$D_{Skin,\ Tskin}$	Skin dose resulting from deposition on skin and clothes	(Sv)
P_{Skin}	Fraction of the skin that is contaminated	(-)
$F_{Skin,ext}$	Shielding factor for being indoors	(-)
$C_{g,k}$	Concentration of nuclide k on the ground due to dry and wet deposition	(Bq.m^{-2})
$DCC_{Tskin,k}$	Skin dose conversion coefficient for nuclide *k*	(Sv/s) / (Bq/m^2)
$U_{Tskin,k}$	Time integral for nuclide *k*	(s)
T_{eff}	Effective half-life = $(1/T_k, + 1/T_b)^{-1}$	(s)
T_k	Physical half-life	(s)
T_b	Biological half-life	(s)
T_{skin}	Period between contamination and decontamination of the skin	(s)

k Index for nuclide

$F_{Skin,ext}$ can, depending on the dose period, be used to apply a shielding factor for the period spent indoors. The local skin dose is of relevance for the calculation of deterministic effects: P_{Skin} in the aforementioned general equation should then be set to 1. For stochastic effects, the lifetime dose is has to be calculated: in those calculations no local skin doses are used, but average skin doses, which are obtained by multiplication of the unprotected skin doses with the fraction P_{Skin} of contaminated skin.

5.2.5 Internal exposure by inhalation of resuspended radionuclides

The calculation of the resuspension dose may also be calculated using the refinement suggested in DOVIS-A as an alternative to the standard model.

$$C_{i,res}(t) = r_0 \, e^{-\lambda_{res}\, t} C_{i,0}(t) + r_\infty \, C_{i,g}(t)$$

with:

$C_{i,res}(t)$	resuspended activity concentration of nuclide i	($Bq.m^{-3}$)
$C_{i,0}(t)$	activity concentration <u>on</u> the ground (with ground penetration correction, § 5.2.2)	($Bq.m^{-2}$)
$C_{i,g}(t)$	activity concentration <u>in</u> the upper soil compartment (root layer)	($Bq.m^{-3}$)
r_0	resuspension coefficient at time t = 0	10^{-6} m^{-1}
r_∞	resuspension coefficient at time $t = \infty$	$3 \cdot 10^{-11}$
λ_{res}	resuspension decline constant	ln(2) $year^{-1}$

5.2.6 Internal exposure by ingestion of radionuclides

DOVIS-A specifies the representative model for the food chain that should be used to calculate the ingestion dose.

Standard: supplementing DOVIS-A take account of the following feed pattern for animals:

Table 5-3 Feed pattern for animals in the calculation of meat contamination

Animal	Food	kg/day or l/day	Storage time (days)
Cows	Hay, grass, silage feed	7	60
	Fresh grass	35	0
Calf	Milk/artificial milk	8	60
Pigs	Corn	6	60
Poultry	Corn	0.1	60

N.B. For other types of meat (for example, mutton) assume the same contamination as for beef

The ingestion dose is calculated by calculating the contamination of food products and subsequently the radioactivity intake based on the consumption pattern of the representative person of each focus group. The dose can then be calculated using the dose conversion coefficients.

Standard: another approach needs to be adopted for the calculation of the ^{3}H and ^{14}C dose, where the concentration in the body is a function of the water and stable C content.

DOVIS-A gives the consumption pattern for adults. The following data can be used to determine the consumption pattern of other age groups.

Table 5-4 Average annual consumption of age cohorts relative to the average adult Dutch male (*Handleiding beleidsuitvoering stralingsbescherming* (implementation of radiation protection policy guidelines), draft 1993)

Age cohort	Milk	Bread	Potatoes	Fish	Remainder
0 – 6 months	0.5 – 1 l/day	-	-	-	-
1/2 – 7 years	4/3 adult	1/2 adult	1/2 adult	1/4 adult	1/2 adult
7 – 12 years	4/3 adult	3/4 adult	3/4 adult	1/4 adult	3/4 adult
12 – 17 years	4/3 adult	= adult	= adult	1/4 adult	= adult
17 – 20 years	4/3 adult	4/3 adult	4/3 adult	1/4 adult	4/3 adult

5.3 Dose effects

5.3.1 Deterministic effects

The relationship between dose and effect (the fraction of the exposed individuals who exhibit the effect) is referred to as the 'hazard function'. This hazard function can be expressed in terms of a 2-parameter Weibull distribution:

$$H = \ln(2)\left(\frac{D}{D_{50}}\right)^{v} = \ln(2)\,X^{v}, D > D_D$$

where:

H	cumulative effect	
D	organ dose	(Sv)
D_{50}	dose that results in 50% of the exposed individuals exhibiting the effect assuming receipt of regular medical care	(Sv)
v	shape parameter that determines the shape of the dose-effect curve	
X	normalized dose, the relationship between the absorbed dose and the D_{50}	
D_D	organ threshold dose	(Sv)

When mortality is the biological effect then D_{50} is LD_{50}, the median lethal dose. The mortality probability is:

$$R = 1 - e^{-H}.$$

When the simultaneous effects on a number of organs contribute to the mortality risk, then the sum of the hazards calculated for each of the organs should be taken into account in the aforementioned equation ($H \rightarrow \sum_o H_o$).

D_{50} and the dose rate

The deterministic effects of a dose received from exposure to radiation depend both on the total dose as well as the exposure time in which the dose is received.

Table 5-5 Increase van LD_{50} en D_{50} as function of the exposure time (data from [44])

	LD_{50} or D_{50}, (Sv) for exposure time in days							
	1 day	**7 d.**	**10 d.**	**14 d.**	**21 d.**	**30 d.**	**200 d.**	**365 d.**
Red bone marrow	3.8			7.6		15		
Lungs	10			160			370	920
Gastrointestinal tract	15	35						
Skin	20		80					

This increase in LD_{50} can also be expressed as a functional relationship:

$$LD_{50}(\dot{D}) = \frac{\theta}{\dot{D}} + LD_{50}^{\infty}$$

where:

$LD_{50}(\dot{D})$	Dose dependent LD_{50}	(Sv)
LD_{50}^{∞}	Asymptotic value of LD_{50} at high dose rate	(Sv)
θ	Model parameter	(Sv^2 / hour)

Table 5-6 LD_{50} calculated with the exposure time functional relationship

	θ	LD_{50} or D_{50} (Sv) for exposure time in days*							
	(Sv^2 / hr)	**Instantaneous**	**1 day**	**7 d.**	**10 d.**	**14 d.**	**21 d.**	**30 d.**	**200 d.**
Red bone marrow	0.1	4	(4.5)	6.5		8.1		10.7	
Lungs	30	10	(32)	(76)		105			385
Gastro-intestinal tract	4.35	14	(19)	35					
Skin	20	20	(34)	69	80				

* values between (brackets) are unusable interpolations, see text

The LD_{50} for an exposure duration of 1 day is set to the instantaneous exposure dose: consequently, the calculated LD_{50} values for that exposure time are too large. **Table 5-6** also shows that these functional relationships can yield unusable interpolations for short exposure times. The calculated LD_{50} lung dose, for example, for an exposure duration of 7 days is 76 Sv. That is more than 10 Sv/day and, consequently, higher than the LD_{50} for an exposure duration of one day.

Therefore, discrete exposure durations of 1 day, 7 days, 30 days, and 365 days are applied in the functional relationship for the LD_{50}.

The use of the functional relationship for the LD_{50} also influences the shape of the hazard function (see Subsection 3.5.2). This is revealed by the following example of red bone marrow exposure to radiation for 7 days:

1. The LD_{50} is (not rounded off) **6.55 Sv**: the dose rate is then 0.039 Sv/hour. θ = 0.1 Sv^2 / hour with LD_{50}^{∞} = 4 Sv does indeed yield LD_{50} = 0.1 / 0.039 + 4 = 6.55 Sv.
2. LD_{50} = 6.55 Sv, v = 5 and a risk r of 1% of the threshold dose T yields (with the hazard function): T = 2.8 Sv.
3. However, for a dose of 2.8 Sv in 7 days the dose rate is 0.017 Sv/hour, where the LD_{50} at this dose rate amounts to almost 10 Sv. With an LD_{50} of 10 Sv and v = 5, T amounts to 4.27 Sv.
4. Further calculations yield a threshold dose of **3.67 Sv** (The dose rate is then 0.022 Sv/hour, the LD_{50} 8.6 Sv, and the threshold dose 3.67 Sv is equivalent to a risk of 1%).
5. An LD_{50} of **6.5 Sv** and a threshold dose of **3.6 Sv** strongly resembles a hazard function with shape factor ***v* = 7** - whilst with a short exposure duration ***v* = 5**. Consequently, the shape of the hazard function would appear to change with longer exposure times.

In fact, too little is known about the shape of the hazard function, especially at long exposure times. However, it is clear that with longer exposure times the dose rate will need to be very high (more than about 0.5 Sv/day) to exceed the threshold dose.

5.3.2 Stochastic effects

The risk factor is strongly age dependent, as is revealed in the following table from UNSCEAR 2006 [45].

Table 5-7 Age-dependent excess mortality risks ([45]; Annex A, Table 61: UK)

Age group (years)	Mortality risk factor (% per Sv)
0 to 9	12.8
10 to 19	10.6
20 to 29	8.7
30 to 39	6.7
40 to 49	4.9
50 to 59	3.2
60 to 69	1.8
70 to 100	0.6

Weighting these age-dependent risk factors for the demography (in 1990) yields an average risk factor of approximately 6.5%. Projecting the bandwidth of the uncertainty in the average risk factor stated by UNSCEAR on the risk factor for the age group between 0 to 9 years yields a bandwidth of 8% to 16% around the stated 12.8%. Within this group, the risk factor for children of 0 to 1 year will be higher than the risk factor for children of 8 to 9 years.

On the basis of the above, the standard risk factor (see the *standard* in Subsection 3.5.1) for young children is set to 15% per Sv. The *standard* for adults and average members of the population is set at the ICRP standard of 5% per Sv.

5.4 Calculation models

The general form of the COSYMA [16] and MACCS2 [17] models is in line with the description in this guide. There are differences at detail level and in the selection of parameter values. For that reason, it can – subject to certain conditions – be concluded that the results from COSYMA/MACCS2 calculations are in adequate agreement with the results from calculations carried out in accordance with the models described in this Guide.

The following subsections discuss the differences in the models, the choice of parameter values, and the required approach in this. The NUDOS2 [18] computer code is not discussed in this Guide because it contains the precise implementation of the NNM and DOVIS-A.

5.4.1 Dispersion models

Plume growth for low elevation sources

Results can be obtained with the COSYMA and MACCS2 models (Pasquill-Gifford or PG models) that are comparable with the results obtained with the model in this Guide (NNM model). It is not possible to configure PG models in such a way that one can be sure that the dose percentiles and associated risk measures will be conservatively calculated. For that reason, attention should be devoted to the accuracy of the results from PG models in each assessment. It may then be necessary to enter new meteorological statistics (stability classes and mixing layer heights) for each assessment and to calibrate the dose percentiles against the statistical distribution obtained with an NNM model.

Plume growth with high elevation sources and plume rise

The NNM report states that the NNM and PG models yield different results for elevated sources (higher than 20 m) and in case of plume rise. For that reason, the use of PG models with high elevation sources or for sources with substantial plume rise is not recommended. However, it is important to note that most level 3 PSA source terms are low elevation sources (as defined in the NNM terminology).

A conservative approximation for high elevation sources or sources with plume rise can always be obtained by assuming a low elevation source and refraining from modelling plume rise.

Dry deposition

Computer codes such as COSYMA and MACCS2 give a good approximation of fine aerosol and elemental iodine deposition with a simple deposition model for areas characterized by a roughness length of 0.1 m. A higher deposition rate should be selected for areas larger than 1 km^2 with a larger roughness length. Although spatial variation in z_0 could be of potential importance, this is not included in either COSYMA/MACCS2 or the NNM.

NNM/DOVIS-A computer codes assume higher elemental/organic iodine deposition than the 1993 level 3 PSA standard. The COSYMA default value for elemental iodine is just below the NNM value: the COSYMA default value for organic iodine is larger by a factor of 2.5. However, the effect on the results is small enough that it will be best to use the default values of the computer code.

Wet deposition

The COSYMA/MACCS2 power-law scavenging model yields results that are a good approximation of those of the NNM washout model.

For the washout of organic iodine, the NNM/DOVIS-A computer codes yield higher organic iodine deposition than COSYMA (default values) and computer codes in accordance with the Level 3 PSA Guidelines of 1993. However, the effect on the calculated results is small enough that it will be best to use the default values of the computer code.

The NNM rainout model yields extremely conservative results and is not applicable to situations assessed in level 3 PSAs. The use of a scavenging coefficient that is linked by a power law to the rain intensity – as in COSYMA and MACCS2 – is a more realistic approach for level 3 PSAs than the extremely conservative NNM rainout model.

Building wake

COSYMA and MACCS2 yield practicable results outside the recirculation zone. They take adequate account of the influence of the building on the plume dimensions and height. A release of material in the recirculation zone will at least require a qualitative assessment of the consequences. That qualitative assessment can be substantiated further with a SCREEN analysis or the NNM building module as far as the geometry of the building can be approximated by a standalone building.

5.4.2 Dose calculation

The COSYMA and MACCS2 dose calculation models are well in line with this Guide. The parameter values (such as for food consumption) can be set to the appropriate Dutch values. The computer codes include a code-specific library containing thousands of dose conversion coefficients. Modifying the data in those libraries is a complex and error-sensitive process.

The MACCS2 dose conversion coefficients are based on US regulations and differ slightly from the European/Dutch values. COSYMA uses the same source as the Dutch/European values (ICRP), i.e. the data from ICRP Publication 119 [23].

External dose

The COSYMA data is identical to the data in ICRP Publication 119 [23]. MACCS2 uses the FGR-12 [26] data, which is in good agreement for most nuclides. Large differences arise for nuclides that in decay emit a beta particle, as the COSYMA data does not take *Bremsstrahlung* (braking radiation) into account, and with MACCS2 for nuclides that may decay by spontaneous fission, as FGR-12 does not take that process into account. However, as these differences relate to only a few individual nuclides differences for source terms with a broad nuclide vector will be minor.

Internal dose

The data published in ICRP Publication 119 [23], also used in COSYMA, differs from the MACCS data that is based on FGR-13 [46]. Both datasets are derived from the same biological model as described by the ICRP. However, in FGR-13 the dose in tissue caused by a nuclide due to its decay is calculated using the FGR-12 data, so that the aforementioned differences in the FGR-12 database are also present in FGR-13.

Future developments

It is expected that the new weighting factors published in ICRP-103 [35] will, in the longer term, result in changes in the dose conversion coefficients used in effective dose calculations. A comparable change arose on the transition from ICRP-26 to ICRP-60. In practice, the DC values for most nuclides remain virtually unchanged whereas for a few nuclides changes range between 10% and 20%. The values based on ICRP-103 will lie between the ICRP-26 and ICRP-60 values. A 20% decrease can be expected for iodine isotopes due to the lower thyroid weighting factor in ICRP-103.

6 Gaps in current knowledge

At the time that the 1993 Level 3 PSA Guidelines were published, many developments were in progress in the aftermath of the Chernobyl accident. Shortly afterwards, the focus shifted to accident management support based on the models developed for level 3 PSA. The methodology on which level 3 PSA is based has changed very little in the past decades. The termination of the development of an IAEA guideline indicates that there is little international interest in level 3 PSAs. This is in part because performing level 3 PSAs is not a legal requirement in many states.

The NNM dispersion model is continuously being developed. However, the core of the model – the dispersion and deposition calculations – has remained unchanged since its publication in 1998. Although a number of changes have been made to some model parameter value details and the NNM has been expanded, these are not of relevance to level 3 PSAs.

Deposition patterns yielded by dispersion calculations are often much flatter than those found in practice. This is caused by local small-scale factors, such as hillocks and clumps of trees, etc. Weathering of the topsoil is also highly dependent on quite local factors.

The dose calculation as described in DOVIS-A (2002) is largely up to date. Some parameter values regarding life patterns, food chain, and ingestion pathway may in time be updated.

It is expected that the new weighting factors published in ICRP-103 will, in time, result in changes in the dose conversion coefficients to be used in effective dose calculations. A comparable change arose on the transition from ICRP-26 to ICRP-60. In practice, the DC values for most nuclides remain virtually unchanged and for a few nuclides they change by some 10% to 20%.

It is difficult to determine the probabilities of health effects of low doses or low dose rates due to the lack of statistically distinguishable data and quantifiable biological models. However, there is broad consensus on the conservatism of the models and parameters currently being used.

7 Glossary

The majority of the following definitions are based on [47].

Actinides The group of elements with atomic numbers 89 through 103: actinium, thorium, protactinium, uranium, neptunium, plutonium, americium, curium, berkelium, californium, einsteinium, fermium, mendelevium, nobelium, and lawrencium. The first four elements occur in nature and the others are transuranium elements (also known as transuranic elements) and are produced artificially.

Activity The number of spontaneous *atomic nucleus transformations* in a quantity of radioactive material per time unit. The unit of activity is the becquerel (Bq). One becquerel is the activity from the decay of one *nucleus* per second. The previously common unit of activity was the curie (Ci):
1 Ci = 37 x 10^9 Bq.

Aerosol Very small droplets or solid particles in air. The droplets/particles have a diameter of between 0.001 and 100 µm.

Alpha decay Radioactive transmutation where an alpha particle is emitted. During alpha decay the number of protons (= the atomic number) is reduced by two units and the number of nucleons (= the mass number) by four. For example, the alpha decay of U-238 (with 92 protons) forms Th-234 (with 90 protons).

α radiation See alpha radiation

Alpha radiation A positively charged particle that is released by decay of some radioactive materials. An alpha particle consists of two protons and two neutrons and is identical to the nucleus of a helium atom. The rest mass of an alpha particle is 6.64424×10^{-27} kg. Alpha radiation is the radiation with the lowest penetration potential of the three types of radiation (alpha, beta, and gamma radiation) that occur in nature. Alpha radiation is stopped by just a sheet of paper and, consequently, is only harmful when particles are ingested, inhaled or enter wounds.

AMAD Abbreviation of Activity Median Aerodynamic Diameter. This is the diameter of the aerosol at which half of the activity is associated with particles smaller than the AMAD and half with particles larger than the AMAD.

Atmospheric dispersion The process of the transport and dilution of material in the atmosphere under influence of turbulent motion in the atmosphere. The Gaussian plume model is a simple model for atmospheric dispersion.

Atomic transformation The transformation of a nucleus of one nuclide into a nucleus of another nuclide. 'Atomic transformation' replaces 'disintegration' that used to be in common usage and was also officially introduced to extend the definition to include changes in the nucleus such as electron capture.

Becquerel (Bq) See activity

Bq See becquerel

Best available technology According to article 15(2) of the EU Industrial Emissions Directive, emission limit values and the equivalent parameters and technical measures in permits shall be based on the best available techniques, without prescribing the use of any technique or specific technology. The directive includes a definition of best available techniques in article 3(10): 'best available techniques' means the most effective and advanced stage in the development of activities and their methods of operation which indicates the practical suitability of particular techniques for providing the basis for emission limit values and other permit conditions designed to prevent and, where that is not practicable, to reduce emissions and the impact on the environment as a whole:

- 'techniques' includes both the technology used and the way in which the installation is designed, built, maintained, operated and decommissioned;
- 'available' means those developed on a scale which allows implementation in the relevant industrial sector, under economically and technically viable conditions, taking into consideration the costs and advantages, whether or not the techniques are used or produced inside the Member State in question, as long as they are reasonably accessible to the operator;
- 'best' means most effective in achieving a high general level of protection of the environment as a whole.

Beta decay Radioactive conversion on the emission of an electron or positron from the nucleus. Beta radiation in the form of the emission of electrons is referred to as beta-minus radiation (β^- radiation), in which a neutron in the nucleus transforms into a proton. Beta radiation in the form of the emission of positrons (positive electrons) is referred to as beta-plus radiation (β^+ radiation) in which a proton in the nucleus transforms into a neutron.

Beta radiation Beta radiation is the term used for the emission of electrons or positrons during the radioactive decay process (see beta decay). Beta radiation has an energy continuum, but is usually characterized by a statement of the maximum energy. Beta radiation is stopped by just thin layers of material (for example, 2 cm of plastic or 1 cm of aluminium).

Building wake Influence of buildings on atmospheric dispersion of releases in the vicinity of those buildings.

CCDF Abbreviation of Cumulative Complementary Distribution Function. CCDF curves are the customary form of presentation of large quantities of probabilistic results that quantify a specific parameter (for example, the number of deterministic effects or the size of the contaminated area). These results are obtained from a large number of calculations, whereby a range of values for the following variables is selected for a level 3 PSA:

- The source term
- The weather sequences: stability, wind speed and wind direction, etc.

Calculations can, for example, be carried out for each scenario (the combination of a source term and a weather sequence, with the probability of its occurrence) to determine the number of deterministic effects. Totalling the probabilities of the scenarios resulting in effects in excess of a specific value yields a point on the CCDF curve. Repeating this procedure for all values between the minimum effect and maximum effect yields a complete CCDF curve.

Cloud dose Dose received from exposure to direct radiation from the plume.

Committed dose Dose received by an organ or tissue after the one-off intake of radionuclides in the body during the period following the intake. The length of this period should always be specified.

Consequence analysis Assessment of the consequences (magnitude of the injuries/damage/loss) of a given source term.

Conservative A subjective qualification that indicates that the assumptions (often simplifications) made when carrying out a calculation will yield a result that is 'less favourable' than the result that would be required on the basis of current insights.

Core inventory Total amount of radionuclides in the reactor core. The core inventory is determined using computer codes such as ORIGEN [48]. The input data for the computer code includes the nominal power of the reactor, the fuel burnup, and the power output as a function of time. The calculations are usually based on the standard inventory of a BWR or PWR. It is assumed that the inventory is linearly proportional to the reactor power capacity. These standard inventories may not be used for other types of reactor: an inventory will then need to be calculated for the specific reactor.

A reactor core contains a very large number – several hundreds – of nuclides. These nuclides make different contributions to the radiological effects on the population in the area. In practice, the many calculations that would be required are reduced by selecting the important nuclides in the core inventory to be assessed on the basis of factors including the amount of the nuclide in the core (Bq), the release fraction, the type of and energy of the radiation, the chemical properties, the radiological effects, and the half-life. For that reason, care should be taken when using a standard inventory to ensure that the reduction criteria result in the selection of nuclides that are representative for the specific source term. The inventory in, for example, the WASH- 1400 study was drawn up on the basis of the assumption of a delay time of at least 30 minutes. This implies that the contribution of shortlived nuclides in source terms with a shorter delay time could be unjustifiably neglected. In those cases it will be necessary to repeat the selection of nuclides in the core inventory using adjusted reduction criteria.

Core melt scenario Failure of the core cooling system – for example due to a major leak in the cooling water system that results in the simultaneous failure of the emergency cooling water system – will result in the reactor core being heated by the residual heat from the decay of the fission products in the fissionable material. This can in certain situations result in the temperature of fissionable material rising to melting point. If the fissionable material melts, then the core support structure will no longer be able to support the core. The molten mass will then drop into the somewhat hemispherical bottom of the reactor pressure vessel. It is then assumed that the heat released by the molten core will also result in the core melting through the reactor pressure vessel. The leak-tightness of the containment structure will then play an important role in restricting the amount of radioactive material released to the surroundings following a core melt accident.

Countermeasures Planned actions initiated after an accident to mitigate the consequences of the accident for individuals in the vicinity.

Daughter nuclide A nuclide formed in a radioactive decay series on the decay of a radioactive (parent) nuclide that in turn decays into a granddaughter nuclide. An example of a radioactive decay series is Radium-266 (parent) ➔ Radon-222 (daughter) ➔ Polonium-218 (granddaughter) ➔ Lead-214 (great-granddaughter) ➔ Bismuth-214 (great-great-granddaughter).

DDREF Abbreviation of Dose Rate Effectiveness Factor. The risk factors recommended by the ICRP are derived from extrapolations of the data on population groups exposed to relatively high doses and dose rates (such as survivors of the atomic bombs in Japan and radiotherapy and diagnostic patients). For that reason, the risk factor for relatively low doses and dose rates is reduced by multiplication with the DDREF.

Delay time Period between the time at which the nuclear reaction stops and the release begins.

Depletion Term for the decline on the content of a release plume due to deposition.

Deposition Process whereby material from a release plume settles on or is brought to surfaces on the ground and becomes bound to objects on the surface (ground, plants, buildings, and clothes, etc.).

Deposition rate Model parameter that specifies the relationship between the concentration in the air just above the surface (for example, about 1 m) and the amount of material that is deposited.

Design basis accident Accident conditions relating to operational occurrences and accidents (such as pipe failure) and external influences (such as earthquakes and floods) that have been taken into account in the design of the facility and are taken into account during the operation of the facility. A traditional or deterministic safety analysis shall demonstrate that it will then be possible to shut down and cool the nuclear power plant's core reactor in accordance with stringent radiological requirements.

Deterministic effect Acute radiation injury to individuals who have received a very large irradiation dose to the body. A dose greater than 0.3 Gy causes symptoms such as lethargy, nausea, vomiting, and headache. Doses greater than 8 Gy are lethal: see Lethal dose.

Direct radiation Radiation emitted from a source that, where relevant, is attenuated by intermediate obstacles, reaches an exposed individual via the shortest possible route. A distinction is made between direct and scattered radiation. The latter refers to radiation that is scatted by intermediate media and indirectly reaches the observation point. This Guide understands ‘direct radiation’ as radiation from outside the exposed individual’s body, such as from a plume or from contaminated ground.

Direct victims Local residents who die shortly after a reactor accident (within several weeks).

Dispersion parameters Input data for the atmospheric dispersion model. This data is usually partially or wholly linked to the stability.

Dose Physical quantity that specifies the absorbed energy per mass. The SI unit is J/kg, or Gy (gray). ‘Dose’ is used solely for ionizing radiation. ‘Dose’ is often used in radiation hygiene (and this Guide) as a simplified designation of some form of dose equivalent.

Dose conversion coefficient Coefficient for the calculation of the dose received by organs and the whole body from radioactive material inside or outside the body. The dose conversion coefficients are dependent on the radionuclide, the exposure pathway (cloud/ground/inhalation/ingestion), the chemical form of the radionuclide (soluble/insoluble) and the age of the individual.

Dose-effect relationship Relationship between the dose or dose equivalent received by an organ, part of the body, or the entire body, and the resultant effect of the radiation.

Dose equivalent The dose equivalent, H, is the product of the dose (in Gy) and the radiation quality factor. The radiation quality factor reflects the biological effect of the various types of radiation. The benchmark is the effect of X rays/gamma radiation. As the quality factor of this type of radiation is 1, and as this form of radiation results in the greatest exposure, in practice the terms ‘dose’ (in Gy) and dose equivalent (in Sv) are used as synonyms. This Guide also uses the term ‘dose’ for ‘dose equivalent’. In radiation hygiene, the dose equivalent H can only be used for values up to several tens of mSv. The unit of dose equivalent is the sievert (Sv), with the formal dimension J/kg. In its latest recommendations, the ICRP has recommended that ‘dose equivalent’ be replaced by ‘equivalent dose’. Rem is an old unit for dose equivalent (= 0.01 Sv).

Dose rate Dose received in a given time period divided by the duration of that time period. The dose rate is usually expressed in units of mSv/h and μSv/h.

Endpoint The various calculated outputs and results from a level 3 PSA.

Expected value Average value of the endpoint for all conceivable weather conditions. Although this is an adequate term for routine releases, 'average' is preferable for accidents.

Exposure Every exposure of individuals to ***ionizing radiation.*** A distinction is made between:

- external exposure: exposure due to a source outside the body;
- internal exposure: exposure due to a source inside the body;
- total exposure: combination of external and internal exposure.

Official provisions often use the term 'exposure' instead of 'irradiation', as the latter term is often used in reference to the intentional use of radiation for specific purposes.

Exposure pathway The entire train of dispersion of radioactivity in the environment and the way in which an individual is exposed to it.

Exposure period The time an individual spends in an area where the individual can be exposed to radiation.

External radiation, dose Radiation or dose from radiation emitted by radionuclides outside the body. See also plume dose, ground dose, and direct radiation.

Filter factor Factor specifying the relationship between the air concentrations inside and outside buildings.

Focus group A (hypothetical) group of people to which attention is focused in connection with a possible higher dose due to exposure to ionizing radiation than the average population. For such a group conservative but realistic assumptions are made with regard to dose-relevant behaviour – living, working, dietary habits, etc. – and physiological parameters. Generally, different focus groups will be considered for different source term types and exposure pathways.

Frequency distribution Classification of many years' meteorological observations into a limited number of categories (for example, 12 wind direction, 6 stability, and 3 wind speed categories). The number of observations is specified for each category.

Frequency model Calculation model that uses a frequency distribution to take account of meteorological statistics (stability, wind direction, and wind speed). The model uses these statistics to yield expected values.

γ radiation See gamma radiation

Gamma radiation High energy electromagnetic radiation of a very short wavelength emitted by many types of nuclei. The energy level of gamma radiation is usually between 0.01 and 10 MeV. The energy level of X Rays also lies in this range. However, X Rays do not originate from the nucleus, but from electron transitions in the electron cloud and from the retardation of electrons in materials (retardation radiation). Alpha and beta decay is usually accompanied by gamma radiation: fission is always accompanied by gamma radiation. Gamma radiation is highly penetrating and can only be stopped by heavy materials such as lead.

Gray Unit of dose. Symbol: Gy. 'Gray' was selected in memory of Lois Harold Gray (1905-1965), who made a contribution to fundamental insights into radiation dosimetry. Definition: 1 Gy = 1 J/kg.

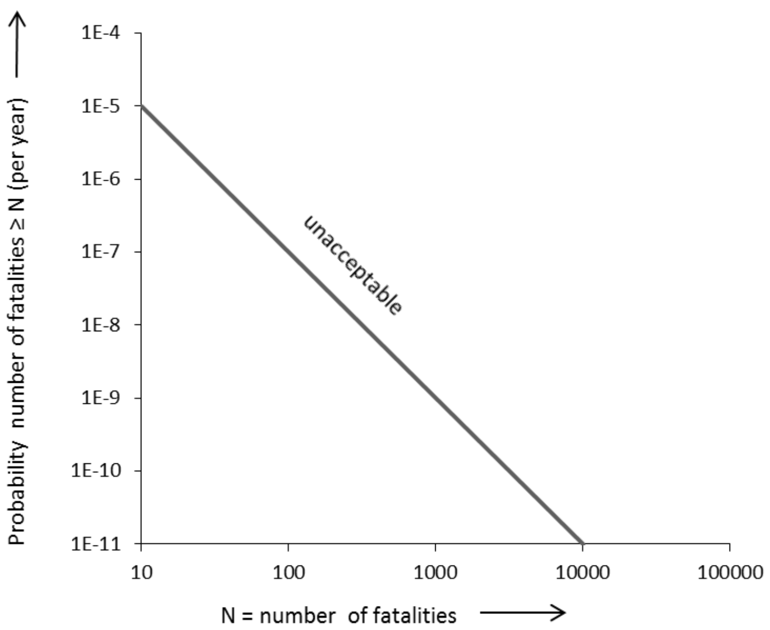
Ground dose External dose received during the time spent on ground contaminated with radionuclides.

Ground contamination The contamination of the top layer of the ground with radionuclides.

Ground roughness This term is assigned two meanings within the context of this Guide:

1. As a parameter (roughness length) with an influence on vertical dispersion during atmospheric dispersion.
2. In the calculation of the ground dose, where the roughness has an influence on the shielding from γ radiation provided by the ground.

Group risk Annual probability that a group of a specific minimum number will become the victim of an accident. The assessment criterion for the group risk is shown in the following figure. This assessment assigns a heavier weighting to accidents with many victims (quadratic) than to accidents with fewer victims.



Ground contamination intervention levels The amount of radionuclides present in the topsoil above which restrictions are imposed on the use (housing or agricultural) of the land.

Gy see gray

Half-life The time required for half of a radionuclide to decay. The half-life of radionuclides varies greatly, for example from 1.5×10^{24} years for Tellurium-128 to 2×10^{-16} seconds for Beryllium-8. The relationships between the half-life T, decay constant λ (lambda) and average life τ (tau) are as follows:

$$T = \ln 2 / \lambda = 0.693 / \lambda$$
$$\lambda = \ln 2 / T = 0.693 / T$$
$$\tau = 1 / \lambda = 1.44\,T$$

'Half-life' is also used with other removal processes.

ICRP Abbreviation of International Commission on Radiological Protection. The ICRP regularly publishes radiation hygiene recommendations and guidelines in its journal (Annals of the ICRP). The ICRP publications are identified by the number assigned to them.

Incident Reactor accident without serious impact. See International Nuclear Event Scale.

Individual dose Dose of radiation received by an individual.

Individual risk The annual probability that an individual will suffer an effect (usually death) as a result of an activity taken by the individual or a third party.

INES See International Nuclear Event Scale

Ingestion The intake of substances in the body via the oesophagus. In the context of radiation hygiene: the intake of radioactive materials in the body on the consumption of foods and liquids contaminated with the radioactive material.

Ingestion dose The committed dose received on the intake of radioactive materials by ingestion.

Inhalation The intake of substances in the body via the trachea. In the context of radiation hygiene: the intake of radioactive materials in the body by the inhalation of radioactive or radioactively contaminated vapours or gases.

Inhalation dose The committed dose received on the intake of radioactive substances by inhalation.

Initiating event The first disturbance that, in the absence of adequate actions by reactor operators and/or safety systems, can lead to an accident.

Intake The amount of radioactive materials that enters the body via the mouth or nose (inhalation or ingestion) or the intact or injured skin.

International Nuclear Event Scale Scale developed in 1990 by the International Atomic Energy Agency and the Organization for Economic Co-operation and Development to classify the severity of nuclear events (incidents and accidents). This scale, which classifies events into a level from 0 through 7 on the basis of their severity, is intended for the prompt communication of the safety significance to the media and the public. The scale takes account of safety aspects off the site of the nuclear facility and on the site, as well of failures in safety provisions.

The scale is as follows:

Level 0: event with no safety significance;
Level 1: anomaly;
Level 2: incident;
Level 3: serious incident;
Level 4: accident without significant off-site risk;
Level 5: accident with off-site risk;
Level 6: serious accident;
Level 7: major accident.

The accidents with the nuclear power plants at Three Mile Island (1979), Chernobyl (1986), St. Petersburg (1991) and Fukushima (2011) were classified into levels 5, 7, 3, and 7 respectively of this scale**.**

The various levels can be characterized as follows:

Level 1 Irregularities in the functioning or operation of the facility that do not pose risks but are indicative of inadequate safety provisions. This may be due to equipment failure, human error or procedural inadequacies.

Level 2 Technical incidents or irregularities that do not have a direct effect on the safety of the facility but do give cause to a re-evaluation of the safety provisions.

Level 3 External release of radioactivity above the permitted limits whereby the individual off the site exposed to the greatest amount of radiation receives a dose equivalent of several tenths of a millisievert.

Level 4 External release of radioactivity whereby the individual off the site exposed to the greatest amount of radiation receives a dose equivalent of several millisieverts. Dose equivalents received by radiological workers can result in acute health effects (of the order of 1 sievert).

Level 5 External release of fission products (in quantities radiologically equivalent to the order of hundreds to thousands of terabecquerels of iodine-131). Partial implementation of emergency plans required in some cases.

Level 6 External release of fission products (in quantities radiologically equivalent to the order of thousands to tens of thousands of terabecquerels of iodine-131). Full implementation of local emergency plans.

Level 7 External release of a large fraction of the inventory of the *reactor core*, typically involving a mixture of short and long-lived radioactive fission products (in quantities radiologically equivalent to more than tens of thousands of terabecquerels of iodine-131). Such a release would result in the possibility of acute health effects; delayed health effects over a wide area, possibly involving more than one country; long-term environmental consequences.

Intervention level Effective dose equivalent from exposure to radiation following the release of radioactive material at which the emergency plan prescribes that countermeasures should be implemented.

Ionizing radiation Radiation that on interaction with material causes ionization. The most well-known types of ionizing radiation are alpha, beta, gamma, and neutron radiation.

Iodine prophylaxis Administration of a relatively large amount of stable iodine (iodine tablets) to prevent the accumulation of iodine in the thyroid as a protective measure in the event of a severe reactor accident. An accident of this nature is accompanied by the release of radioactive iodine to the atmosphere that on inhalation will accumulate in the thyroid. When the thyroid is already saturated with iodine the excess will be excreted in a natural manner.

Irradiation Exposure to radiation resulting from the decay of radionuclides outside or inside the body that delivers a dose. 'Irradiation' is generally used with respect to the intentional use of radiation for specific purposes.

Irradiation period The period in which an individual receives a dose from internal or external irradiation. With external irradiation, the maximal duration of the irradiation period is equal to the exposure period. With internal irradiation, the irradiation period is equal to the exposure period multiplied by the integration time to be used for the calculation of the committed dose.

Level 3 PSA See PSA

Lethal dose Dose that results in death due to acute radiation injury. The lethal dose median (LD_{50}) is the dose that results in the death of half of the individuals exposed to the dose in the same manner. The most recent estimates and calculations for the dose received by bone marrow, the most sensitive organ, indicate a LD_1 of 2.5 Gy, LD_{50} of 5 Gy, and LD_{99} of 8 Gy. LD_1 is then the dose fatal to 1%, LD_{50} to 50% and LD_{99} to 99% of the exposed individuals.

Lift-off Situation in which a cloud released at ground level that has a temperature above ambient temperature rises from the ground.

Light water reactor Collective term for all nuclear reactors moderated and cooled by H_2O: boiling water reactor, pressurized water reactor (H_2O = 'light' water, as opposed to D_2O, 'heavy' water). The reactor core, consisting of fuel rods and control rods, is enclosed in a pressurized vessel filled with light water. The heat produced by fission is transferred to the water. The heated water in a boiling water reactor forms steam in the pressurized vessel, whilst the heated water in a pressurized water reactor forms steam in a steam generator in a secondary circulation system. The energy in the steam is then used to drive a steam turbine that in turn drives a generator to produce electricity. The steam leaving the turbine is passed through a condenser and the condensed water is then returned to the pressurized vessel or steam generator. The water needed to cool the condenser is pumped from a local river. The hot water discharged from the condenser is returned to the river or pumped through a cooling tower where some of the water evaporates to the atmosphere and cools the remainder of the water.

Long term Period after an accident during which stochastic effects are to be expected. This Guide specifies the duration of this period as a maximum of 70 years.

Maximum individual risk The risk associated with a given industrial activity varies with location and between individuals. The highest risk identified is the maximum individual risk.

Meteorological data Weather observations that serve as input for the atmospheric dispersion model. This data usually relates to the wind speed and direction (hourly and 10-minute averages), rain intensity, and mixing layer height.

Meteorological sampling It is not practical, in view of the relatively long calculation times with current level 3 PSA computer codes, to use virtually all the available meteorological data (at least 6% of the annual hourly values) in the analysis. For that reason, a sampling scheme is used to make a selection from the available data.

Mixing layer The planetary mixing layer is covered by a stable stratum of air with a strong positive temperature coefficient. Turbulence is greatly reduced in this layer, which then forms an effective boundary for vertical atmospheric dispersion. The height of this layer is referred to as the 'mixing layer height' and the layer itself as the 'mixing layer'.

Nuclear facility Nuclear reactors, except for those used to propel a means of transport, plants for the manufacture or treatment of nuclear material, plants for the separation of isotopes from fissile materials, plants for the reprocessing of irradiated fissile materials and facilities for the storage of nuclear materials other than facilities for the storage of nuclear materials pending the transport of those materials.

Nucleus The positively charged core of an atom. The diameter is some ten trillionths of a centimetre (10^{-13} cm), about 1/100000 of the diameter of an atom. The nucleus contains almost the total mass of the atom and – with the exception of the nucleus of hydrogen atoms, which consists only of one proton – consists of protons and neutrons. The number of protons determines the atomic number, Z: the number of protons plus the number of neutrons determines the number of nucleons, A, of the nucleus.

Optimistic Subjective qualification that indicates that the assumptions (often simplifications) made when carrying out a calculation will yield a result that is 'more favourable' than the result that would be more realistic on the basis of current insights. Opposite of 'conservative'.

Particle size distribution Classification of source term aerosols on the basis of the size of their droplets/particles.

Percentile Always stated in combination with a number with a suffix. When a set of observations is accompanied by a statement that the 95th percentile is equal to X then this means that the value of 5% of the observations is greater than X and, consequently, the value of 95% of the observations is smaller than or equal to X.

Plume Cloud of released material being dispersed in the atmosphere.

Plume Releases over a longer period of time are dispersed in the atmosphere and carried by the wind and develop the form of a plume.

Plume correction factor The external dose received from radiation in the plume is usually calculated using dose conversion coefficients determined for submersion in air that is homogeneously contaminated with radionuclides. Plume correction factors are required to take account of the finite dimensions of plumes.

Plume rise Upward motion of the plume when the temperature of the plume is higher than the temperature of the air around the plume.

Point source Release source with a plume of negligible dimensions at the location of the release.

Potential dose Dose that could be received when no limits are imposed on the exposure (shielding when spending time indoors, exposure duration).

Primary system The part of the nuclear power plant that contains the coolant that transports the heat from the reactor core.

Probability Number between zero and one that indicates the likelihood that a particular event will occur. Risk analyses always specify the probability within a given time: as a result, the 'frequency' is often used as a synonym.

PSA Abbreviation of Probabilistic Safety Assessment. This assessment is focused on the determination of the following:

- the total probability of the accident scenarios that result in core melt as determined on the basis of the frequencies of the initiating events and a reliability analysis of the various systems (level 1 PSA);
- the probability of the failure of the reactor containment (on the basis of the core melt scenarios) and the characteristics of the associated accident release, or source term (level 2 PSA);
- the consequences for humans and the environment in terms of harm to health, encompassing both deterministic (severity, short term) and stochastic (probability, long term) effects and, where relevant, economic effects (level 3 PSA).

PSAs assess accidents that are not taken into account in the design, 'beyond design basis accidents'. Probabilistic refers to the calculation of probabilities. All safety analyses used to be carried out on the basis of deterministic calculations. The deterministic approach assigns an initiating event a probability of 1 or 0. See also Accident analysis.

Radiation Emission or propagation of energy in the form or particles or rays (for example, alpha, beta, gamma, and neutron radiation).

Radiation hazard Equivalent to 'exposure to radiation'

Radiation hygiene Entirety of measures implemented to achieve maximum possible safety when working with radioactive materials and with equipment emitting ionizing radiation.

Radioactivity Property of certain nuclides to spontaneously emit, without external influence, a characteristic radiation and at the same time to transform into another nuclide. Becquerel discovered the radioactivity of uranium in 1896. Radioactivity from nuclides – or, to be more precise, radionuclides – that occur in nature is referred to as natural radioactivity: radioactivity from radionuclides produced by nucleus transmutation in nuclear reactors or accelerators is referred to as artificial or man-made radioactivity. Some 2200 radionuclides have been identified to date. Each radionuclide is characterized by its half-life, the time after which half of the nuclei have transmutated. Radionuclides have half-times varying from several billion years (Uranium-238) to a millionth of a second (Po-212). Radionuclides are also characterized by the radiation they emit on decay and the energy of the radiation: for example, Radium-226 emits alpha radiation on decay and Iodine-131 emits beta radiation.

Radioactive decay Spontaneous transformation of a radionuclide to another radionuclide or to a lower energy state. Every decay process is characterized by a specific half-life.

Radiological effects Effects of exposure to radioactive materials on human health.

Radionuclide Unstable nuclide that spontaneously, without any external influence, decays and emits radiation. More than 2200 natural and man-made radionuclides are known.

Reactor containment building Outermost air-tight structure around a nuclear reactor designed to contain radioactive material that escapes during an accident and prevent release to the surroundings. The containment building is the fourth and final barrier to radioactive release to the surroundings. The containment building encloses only the nuclear components of the reactor and is designed to withstand the pressure and temperature increases caused by the escape of steam in a reactor accident. These nuclear components include the quick-close valves in the outgoing pipes and the air locks for persons and goods. The air-tight steel containment structure, with a diameter of about 50 metres and a thickness of about 30mm, is usually enclosed in a reinforced concrete missile shield of about 2 metres thick that is strong enough to withstand the impact of an aircraft and prevent flooding.

Reactor core Part of the reactor in which the fission chain reaction takes place.

Reactor year The nuclear power plant risk is usually expressed per reactor year. This indicates that the annual shutdown for maintenance and replacement of the fuel rods is taken into account in the determination of the risk.

Release The generation of electricity with conventional and nuclear power plants is accompanied by the release of undesirable materials to water, the atmosphere and, after deposition, to the ground. Conventional power plants release nitrogen oxides (NO_x), sulphur dioxide (SO_2), and carbon dioxide (CO_2), the last of which is the most important greenhouse gas. Nuclear power plants release solely radioactive materials. A distinction is made between routine releases and accident releases. Routine releases, which are permitted by law, occur during normal operations and do not pose any hazard to individuals in the vicinity of the power plant. Accident releases are releases that occur during an accident. An accident release may result in exposure to:

- radiation from radioactive material in a plume carried by the wind
- radiation from the ground surface from radioactive material deposited from the plume, whereby a distinction is made between dry and wet deposition. Wet deposition (rainout) results in much more severe ground contamination than dry deposition.
- radiation from ingested food contaminated with radioactive material after deposition.

Exposure to radioactive materials released to water rather than to the atmosphere will be almost entirely restricted to exposure to radiation via the food chain (consumption of fish or shrimps).

Release height Height of the release above ground level at the location of the release.

Release point Location of the release from the reactor buildings, i.e. the ventilation stack, the reactor dome or other building.

Representative person An individual that is exposed to a dose representative for the population group most exposed to ionizing radiation, excluding individuals with extreme or rare habits. The representative person is an average member of that focus group receiving the highest individual dose as a result of all the emission types considered together.

Risk Risk is, in particular for the purposes of quantitative risk assessments, defined as the product of the effect (the extent of the consequences of materialization) and the probability (the frequency of materialization). The risk of the use of a technology in which accidents are frequent but the effects of those accidents are relatively minor (such as travelling by road) can then be greater than the risk of the use of a technology in which accidents are not frequent but the effects of those accidents are great (such as travelling by air). Risk defined in this manner is a measure used to estimate the potential consequences of a technology and the compare the potential consequences of various technologies. The government has adopted risk standards that take account of the fact that frequent occurrences that each cause a small number of casualties (such as road accidents) are more readily acceptable than rarer occurrences that each cause a large number of casualties (such as air accidents), even though the annual number of casualties caused by the first type of occurrences can be much higher than the second, see also group risk.

Risk analysis See PSA.

Risk criteria The risks posed by the construction and operation of nuclear power plants are tested against the risk approach specified in the government's environmental policy. This policy imposes limits on the risks to which the population may be exposed. Normal operations and accidents are assessed against the individual risk criterion. Severe accidents are assessed against the group risk criterion. The Minister of Infrastructure and the Environment is bears the responsibility for the risk approach specified in the government's environmental policy.

Risk factor The probability of the occurrence of a stochastic effect (cancer, hereditary effect) per sievert dose.

Roughness length A measure of the 'roughness' of the surface of the Earth that has an influence on atmospheric dispersion. A distinction is usually made between level ground, arable land, cultivated land, residential areas, and urban areas.

Routine release Also referred to as 'continuous release'. Nuclear facilities in normal operation release radioactive materials via the ventilation system and/or cooling water. A licence is required for these releases. The exposure to radiation from these radioactive materials is below detection levels but can nevertheless be assessed with computer models. The maximum individual annual dose, for example, from releases to the atmosphere by the Borssele nuclear power plant varies between 0.02 and 0.05 microsievert. The maximum individual annual dose from releases to surface water calculated on the basis of the consumption of various fish products from various locations are a further one order of magnitude lower. Consequently, the maximal individual annual dose from releases by Dutch nuclear power plants in die Netherlands is less than 0.05% of the average dose from natural background radiation.

Sampling scheme The categories and criteria used to classify meteorological data.

Severe accidents Large-scale accidents, also referred to as 'beyond design basis accidents' (see design basis accidents) or accidents resulting in a 'large release'. It has been proposed that a release that gives cause to the implementation of countermeasures pursuant to the National Nuclear Emergency Management and Response Plan be referred to as a 'large release'.

Shielding factor The dose received by an individual exposed to a source of radiation will be lower in when material, such as structural materials, is present between the individual and the source. This reduction of the dose is taken into account by using the shielding factor.

Short term Period immediately following an accident (usually of several weeks) during which deterministic effects are to be expected.

Sievert Name assigned to the dose equivalent unit. Symbol: Sv. Named after the Swedish scientist Sievert, who contributed to the introduction and development of radiation protection.
Definition: 1 Sv = 1 J/kg (γ radiation).

Site boundary Also referred to as the perimeter. The demarcation between the nuclear facility's site and the surrounding area accessible to third parties. The location of the site boundary is determined by the licence, pursuant to Article 15, under b.

Source depletion A modelling approach to the process in which radioactive material in the release cloud is lost on deposition.

Source term The entirety of the factors that describe the composition of, amount of, and the manner in which radioactive material is released to the biosphere in the event of a serious reactor accident.

Source term phases Source terms that can be divided into a number of phases on the basis of the release rate and/or composition of the release over the release duration.

Source term spectrum The set of source terms (including the frequency at which they occur) that is representative for every conceivable accident at the nuclear facility that is being assessed.

Stochastic effect Stochastic effects are assessed on the basis of the probability of the occurrence of the effect rather than on the severity of the effect as a function of the dose, whereby it is assumed that there is no threshold value. The severity of deterministic effects, in contrast, does vary with the dose and, consequently, a threshold value of dose does exist. Hereditary effects resulting from doses in the range of relevance to radiation protection are regarded as stochastic effects. Some somatic effects are also regarded as stochastic effects: the development of cancer is considered to be the most important somatic risk from low doses of radiation.

Straight-line Gaussian plume model Simple dispersion model in which part or the entire plume is carried by the wind in the wind direction at the time that the release began.

Submersion Term used in radiation hygiene to refer to the assumption that an individual is located in a semi-infinite space with a homogeneous concentration of radioactive material.

Surface depletion Modelling approach to the process in which radioactive material in the plume is lost on deposition. Alternative for 'source depletion'.

Surface source Release source with horizontal dimensions that have a noticeable influence on atmospheric dispersion (as compared to a point source) that can be detected at a distance of a number of kilometres.

Threshold dose Smallest amount of absorbed energy or the lowest dose that has a specific effect.

Threshold effect Health effect that only occurs when the dose received is above the threshold dose for that effect. Level 3 PSA endpoints that are determined by a threshold effect are often very sensitive to small variations in the input parameters.

Virtual release point The influence of buildings (building wake) on atmospheric dispersion is often modelled using a point source at a certain location (depending on the stability class) behind the actual release point, the 'virtual release point'.

Volumetric inhalation rate The average volume of air an individual (from the focus group being assessed) inhales and exhales per time unit.

Warning time Period between the time at which the severity of the accident is appreciated and the time at which the release begins.

Weather sequences Start times in the meteorological data are determined using meteorological sampling of the meteorological database. The successive average meteorological values during an hour, or other time interval, after a given start time is referred to as a 'weather sequence'.

Weather scenario See weather sequence.

Wind speed profile The wind speed is a function of altitude. This function – the wind speed profile – is dependent on the stability.

8 References

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This report is a publictaion of the

Authority for Nuclear Safety and Radiation Protection ANVS

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March 2020