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Definition of Clearance Levels for the Release of Radioactively Contaminated Buildings and Building Rubble

European Commission

Radiation protection 114

Definition of Clearance Levels for the Release of Radioactively Contaminated Buildings and Building Rubble

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Dr. A. Deckert
Dr. S. Thierfeldt
Dr. E. Kugeler
Dipl.-Ing. I. Neuhaus

Brenk Systemplanung, Aachen

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Foreword

The European Commission has published guidance of the Group of Experts established under Article 31 of the Euratom Treaty on the clearance of buildings and building rubble arising from the dismantling of nuclear installations (Radiation Protection N° 113). The technical basis for these recommended criteria was provided by a consultant (Brenk Systemplanung, contract C1/ETU/970040) as input to the discussions of a working party of the Group of Experts.

The present publication of the consultants report is important in view of ensuring full transparency of the approach and traceability of the values. It will in addition facilitate similar work carried out in Member States or outside the European Union and contribute to the harmonisation of clearance levels proposed internationally.

S. Kaiser

Abstract

The technical basis for the establishment of clearance levels for buildings and building rubble arising from the dismantling of nuclear installations is given. This includes an overview of the exposure scenarios for the different considered release options, the resulting individual and collective doses and the corresponding clearance levels. The scenarios are translated in formula for the calculation of doses and a comprehensive data set is provided with all chosen parameters values as well as all radionuclides specific data.

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1. INTRODUCTION

Radiation protection requirements pertaining to the operation of nuclear installations in the Member States of the European Union (EU) are established at a national level, whereby national legislation is bound by the Euratom Treaty to comply with the general EU standards: "The Basic Safety Standards for the Health Protection of the General Public and Workers against the Dangers of Ionizing Radiation" (BSS). A revised Basic Safety Standards Directive was adopted in May 1996 and must be implemented in national legislation by May of the year 2000 [CEU 96].

One of the requirements in the new Standards is that the disposal, recycling and reuse of material containing radioactive substances is subject to prior authorisation by national competent authorities. It is stated, however, that the authorities may specify clearance levels below which such materials are no longer subject to the requirements of the Standards. Clearance levels shall be established on the basis of the general criteria for exemption laid down in Annex 1 of the Directive, and take into account technical guidance provided by the Community (e.g. [EUR 98]). According to Article 5 it is the responsibility of the competent national authorities to define the numerical values for clearance. Thus upon decommissioning and dismantling of nuclear installations, regulatory control may be relinquished for part of the premises or materials arising from dismantling. For example, there are currently more than a hundred nuclear reactors operating in the EU and around 40, many of which are research reactors, which have been shut down and are being decommissioned. This represents a large potential for "waste" material under regulatory control, the largest portion being building material, which is to a large degree not or only very slightly radioactive. Recycling or conventional disposal of the rubble resulting from the demolition of the buildings or the non-nuclear use of the buildings would avoid unjustified allocation of resources for the radioactive disposal of this low activity waste and save valuable natural resources.

The European Commission has already published technical guidance for the clearance of items, equipment and metal scrap in "Recommended radiological protection criteria for the recycling of metals from dismantling of nuclear installations" (RP 89) [EUR 98]. The present report serves as the basis for developing a similar recommendation for technical guidance for the clearance of buildings and building material belonging to practices subject to reporting and prior authorisation as set out in Article 2 para. 1 of the BSS. The emphasis is placed on the material from dismantling large facilities, of which nuclear power plants will give rise to the largest quantity in the EU, but the general nuclide specific criteria developed here are also valid for nuclear fuel cycle installations and facilities such as isotope laboratories. The term "building" is used here in a broader sense to mean not only buildings but also rooms, sections of buildings and building structures.

The criteria being developed in the present study only apply to practices and the material within these practices which fall under the scope of Article 2 para. 1 of the BSS. In particular, the radiological analysis in this study applies to all buildings which are part of an authorized practice. The criteria do not apply to naturally occurring radionuclides unless they are present as a result of the authorised practice. Furthermore the criteria do not apply to materials relating to past practices as defined in Title IX of the BSS or work activities as defined in Title VII and Title IX of the BSS.

During the process of decommissioning nuclear facilities and to a lesser degree during the operational phase, buildings and building rubble arise which from a radiological standpoint do not need to be kept under regulatory control. This study endeavors to develop clearance criteria based on Article 5 in conjunction with Annex I of the BSS which the competent national authorities can apply when making clearance decisions for buildings and building rubble. The analysis is carried out according to the following steps:

- The analysis covers buildings of nuclear installations which will be reused or will be dismantled after clearance as well as building rubble that arises from the dismantling of parts of the buildings of these installations prior to their clearance.
- In section 2, the potential quantity of buidling material and its development with time is estimated for single countries and for the entire European Union in order not to underestimate quantities arising simultaneously within a certain region. In addition, relevant aspects of . These data are incorporated in the scenarios and also serve for estimating collective doses.
- Section 3 contains basic information which influence the dose calculations, like the relevant radionuclides and considerations to the dose coefficients, to averaging areas and masses as well as to the relationship between surface and bulk activity.
- Section 4 provides the definition of the radiological scenarios and discusses the parameters used for these calculations. Scenarios are evaluated for clearance of buildings for reuse, clearance of buildings for demolition and for building rubble.
- The results of the dose calculations are presented in section 5 which provides list of nuclide specific clearance levels for the three cases of clearance.
- The appendices contain additional radionuclide specific data.

The results of this work served as a basis for a recommendation which was issued by the European Commission on clearance levels for buildings and building rubble from nuclear installations. These clearance levels have to be regarded in the context of the BSS: The scheme in figure 1–1 illustrates the decision making process prescribed by Title III of the BSS. It should be noted that the scope of the BSS is defined in terms of practices and only indirectly in terms of radioactive substances. Any practice involving radioactivity requires justification. If the use is deemed justifiable it must be decided whether the practice should be put under the system of reporting and prior authorisation as prescribed by the BSS. Practices which do not fall under this system are called exempt practices. Some practices are explicitly put under the regulatory system due to their potential risks, for example all nuclear fuel cycle installations. Other practices can be exempt if the associated risks are sufficiently low. Radionuclide quantities and activities per unit mass giving rise to trivial risks are called exemption levels and have been derived for the BSS [EUR 93]. It is understood that no reporting need be required for practices which are not a priori subject to regulation and which involve radioactive substances below either one of the exemption levels, i.e., these practices may be exempt from the regulatory requirements.

Work activities which involve “natural radioactivity” but where the radioactive properties are not used do not fall under this system of reporting and prior authorisation and therefore the criteria developed here do not apply to the material involved in these activities. The BSS gives basic guidance for regulating such activities in Title VII and the Commission has published a specific recommendation [EUR 97].

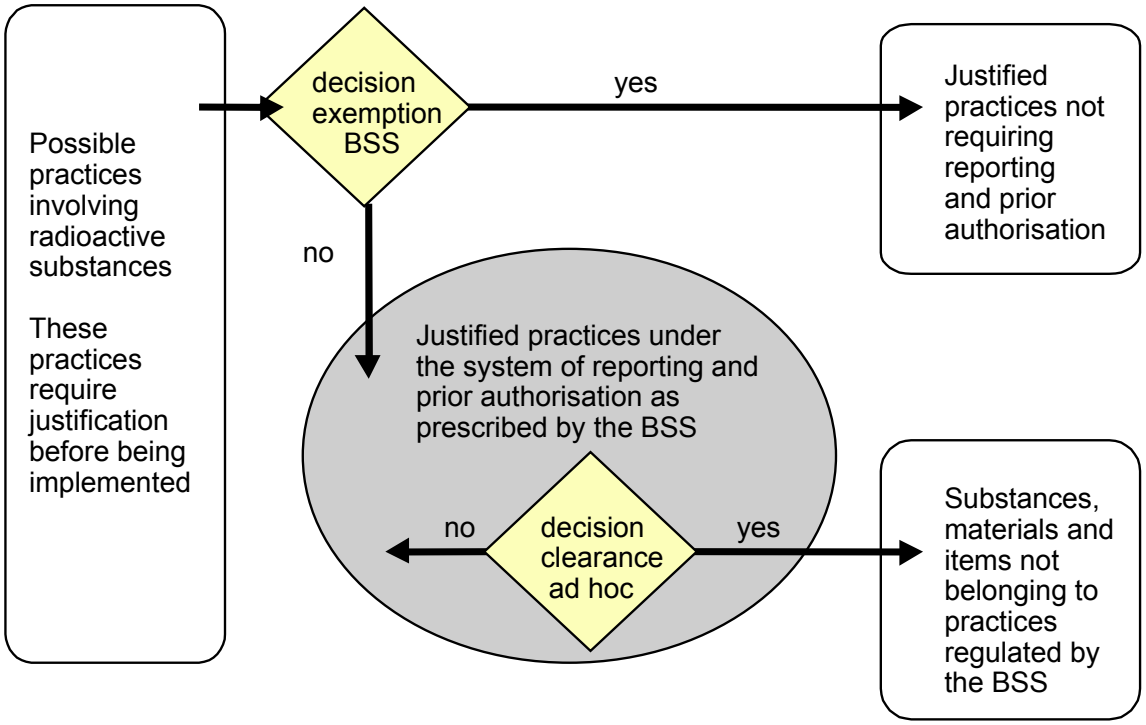


Figure 1–1 Schematic diagram illustrating the implementation of the European Union's Basic Safety Standards (BSS).

2. BUILDINGS AND BUILDING MATERIAL FROM NUCLEAR INSTALLATIONS

Very important parameters pertaining to the derivation of clearance levels for building rubble and buildings are the number of installations that are and will be decommissioned now and in the future and the quantity of material arising from these installations as well as from other sources. Therefore, this section gives an overview of the number and capacity of nuclear installations in various European countries and provides an estimate of the material quantities that will probably arise from dismantling of the buildings. In addition, this section outlines the recycling and disposal options for building rubble and buildings which have to be taken into account for the development of radiological scenarios.

2.1 Potential Quantity of Buildings and Building Material

2.1.1 Installations in Germany

Building rubble can result from refurbishing and revision work during the operational period of a nuclear installation, but the quantities arising during the decommissioning phase represent by far the most significant source. Over the last years experience with decommissioning to green field conditions has been gained, for example in Germany the successful completion of the dismantling of the nuclear power plant Niederaichbach (KKN) [ORW 95] and the *Heißdampfreaktor* (HDR) in Karlstein [KÜH 98] to green-field conditions.

The quantities of building material arising from a nuclear facility depend on the type of facility and in the case of power stations their power rating. For example, the concrete mass from the research reactor MZFR in Germany with 58 MW_e which is currently under decommissioning amounts to about 42,000 Mg, the larger decommissioned power reactor KKN with 106 MW_e had about 70,000 Mg, and the building mass of an average nuclear power plant with a 1000 MW_e light water reactor (LWR) will amount to 150,000 to 200,000 Mg (see [DEC 96] [JOH 94]). Besides nuclear power plants, facilities for fuel element construction or reprocessing plants are decommissioned which will also result in clearable buildings and building rubble. Eurochemic reprocessing plant in Dessel, Belgium [TEU 95] and the HOBEG in Hanau, Germany where buildings have been cleared are examples of such facilities. The size of the buildings and the amount of resulting rubble from such facilities will typically be significantly less than for a commercial power reactor (see [JOH 94]). In addition, the number of power reactors outnumbers all other facilities together and therefore most of the cleared building rubble in the future will come from the demolition of commercial power reactors.

Under the assumption that no new nuclear power plants will be built in the near future and that the existing power reactors will be decommissioned after their planned operational period, the amount of building rubble from nuclear power plants in Germany was estimated [JOH 94]. This estimate is shown graphically in figure 2–1. The total expected quantities of rubble from other sources was shown to be insignificant compared to the total rubble from all the nuclear power plants. Nevertheless for several nuclear fuel cycle facilities the quantity arising from their demolition is comparable to the quantity from the demolition of a single nuclear power plant, for example the reprocessing plant at Karlsruhe Germany, WAK [JOH 94] (the same applies for installations in other countries like the gaseous diffusion plant at Capenhurst UK [CLE 94]).

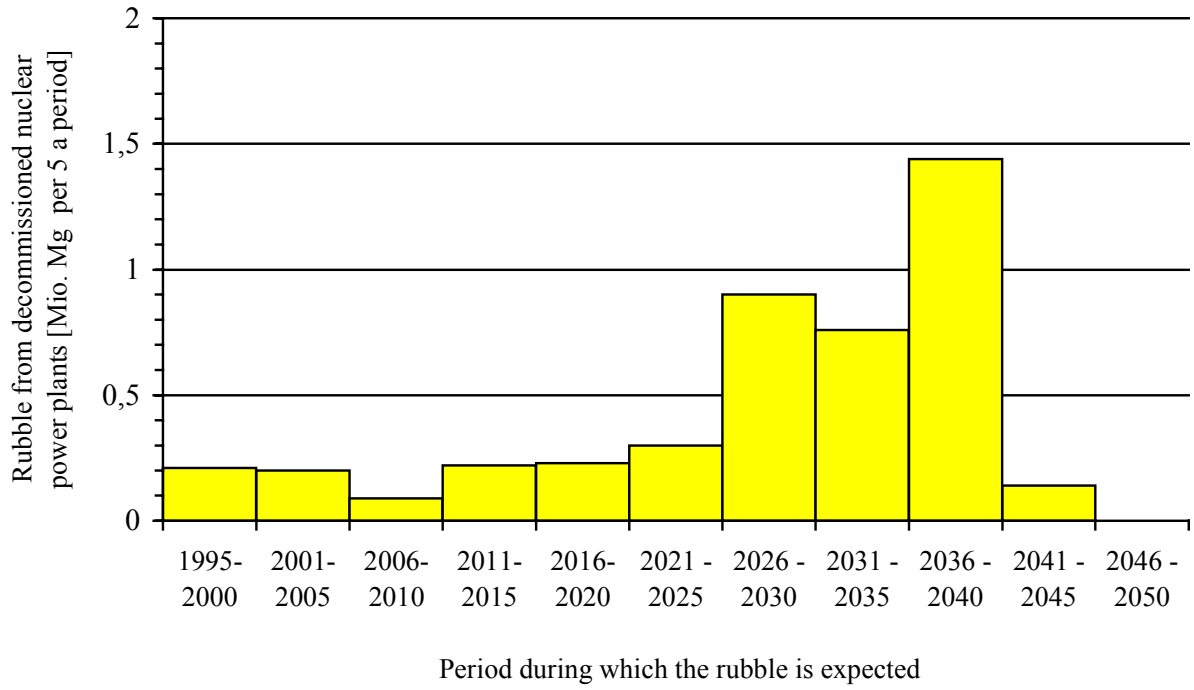


Figure 2–1	Estimated quantity of building rubble from decommissioned nuclear power plants in Germany under the assumption that no new plants will be built in the near future and existing plants will be operated for the planned period [JOH 94]	BS

During operation and decommissioning, building rubble also arises prior to the demolition of the building itself. An example is the rubble arising from the removal of walls in order to reach reactor components which could otherwise not be dismantled. In figure 2–2 a schematic diagram of the estimated amount of rubble occurring during the operation and decommissioning of a nuclear power plant is shown along with the expected time scale during which this rubble will arise [JOH 94]. It can be seen that the quantities during the decommissioning phase exceed by far those of the operational phase. In the course of component removal and decontamination up to around 1000 Mg/a of rubble can be expected. If the biological shield is dismantled during this phase an additional 100 Mg to several 100 Mg of rubble can be expected. After all the components in the building have been removed, the buildings can be demolished which will give rise to around 100,000 to 200,000 Mg of rubble. In general the demolition will take at least about 1 to 2 years to complete.

A complete list of nuclear installations from which building rubble will arise can be found in tables A3–1 to A3–4 in the appendix.

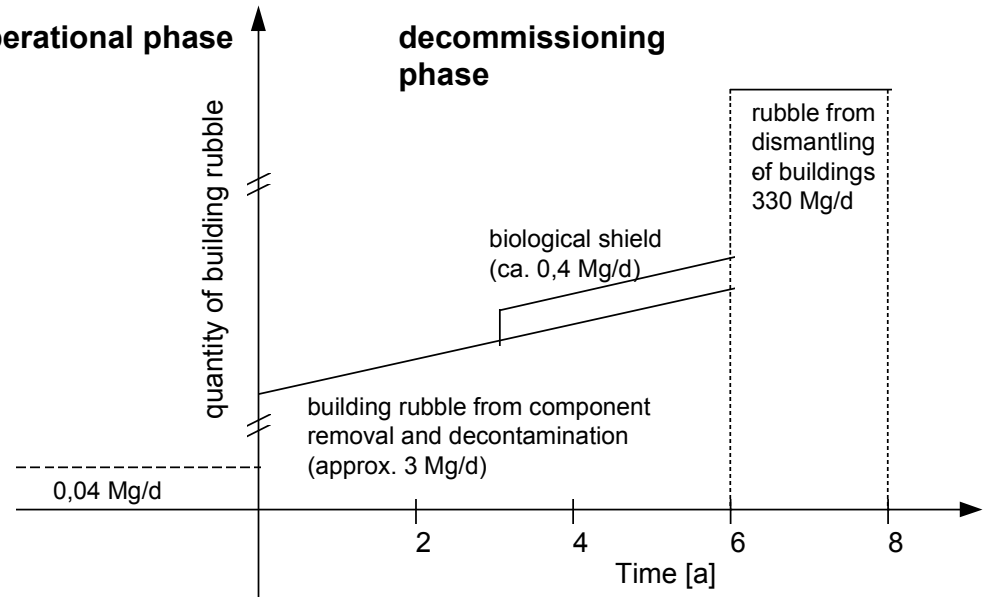


Figure 2–2	Schematic presentation of the rubble arising during the operation and decommissioning of a nuclear power plant (taken from [JOH 94])	BS

2.1.2 Installations in France

2.1.2.1 Overview of nuclear facilities

The masses of rubble deriving from the decommissioning of nuclear facilities in France are estimated in [ONE 98]. In general the situation is similar to the German conditions. Most of the rubble is expected from nuclear power reactors which are or have been operated in France. 34 PWRs of 900 MWe, 20 PWRs of 1300 MWe, 4 PWRs of 1450 MWe and one fast breeder of 230 MWe are in operation or become operable before the year 2000. Facilities which are already shut down are listed in table 2–1.

Table 2–1: Shut down nuclear power plants in France

Nr.	Name	Type	Power	Dismantling progress
1	Brennilis	Heavy Water	75 MWe	Level 2 dismantling started in 1997
2	Chooz	PWR	320 MWe	Partial dismantling started 1993
3	Chinon A1	GMGCR	83 MWe	Complete dismantling planned for 2027
4	Chinon A2	GMGCR	210 MWe	Complete dismantling planned for 2039
5	Chinon A3	GMGCR	375 MWe	Complete dismantling planned for 2044
6	Saint Laurent A1	GMGCR	405 MWe	Complete dismantling planned for 2044
7	Saint Laurent A2	GMGCR	465 MWe	Complete dismantling planned for 2046
8	Bugey 1	GMGCR	555 MWe	Stopped in 1994
9	Super Phenix	FBR	1242 MWe	Stopped in 1997

In addition to the power plants some rubble is expected from fuel cycle, military and nuclear research facilities. The total amount is almost one order of magnitude lower than that expected from the power plants. In only a few cases the masses expected from single fuel cycle, military and nuclear research facilities will reach the same amount as the masses of a nuclear power plant. The following list presents on overview over the facilities (without power reactors) from which building rubble is expected:

- Reprocessing plant UP1 in Marcoule
- MELOX plant in Marcoule
- Enrichment Plants in Pierrelatte
- five Uranium chemical treatment plants at different locations
- Laboratories in Cadarache
- 13 sites of the CEA
- Harbours for nuclear submarines and aircraft carrier
- Military sites
- Experimental reactor in Strasbourg
- CENTRACO plant in Marcoule
- CERN
- GANIL
- Decontamination plant SOCATRI in Pierrelatte
- ESRF in Grenoble
- Conventional plants, which are contaminated with radionuclides.

Experience in France for the complete dismantling of nuclear facilities (i.e. dismantling to IAEA level 3) is based on six experimental reactors and six nuclear laboratories. In addition, the gas-cooled reactors G1, G2 and G3 in Marcoule are dismantled to IAEA level 2 conditions.

2.1.2.2 Waste quantities

In total (power plants and the facilities listed above) the expected amount of waste with a specific activity below 1 Bq/g during the next century is about 13 Mio Mg. The part of the power reactors amounts to 10.1 Mio Mg (78%). The concrete structures of the old diffusion plant at Pierrelatte have a mass of 0.5 Mio Mg. The amount of very low level waste (VLLW) with activities between 1 Bq/g and 100 Bq/g is expected to be about 1 Mio Mg. This waste will mainly arise before 2030

(17,000 Mg/y from 2000 to 2010; 25,000 Mg/y between 2010 and 2020; 35,000 Mg/y between 2020 and 2030). In table 2–2 the expected concrete masses are listed according to their specific activity and the dismantling level.

Table 2–2: Expected masses for level 2 and 3 dismantling of nuclear power plants separated according to their specific activity A (in Mio Mg).

Rubble activity	A < 1 Bq/g	1 Bq/g < A < 100 Bq/g	100 Bq/g < A
Level 2 dismantling	2.4	0.16	0.013
Level 3 dismantling	7.7	0.075	0.031

In addition to the waste quantities listed above, some waste amounts are already fixed by ANDRA. ANDRA annually updates an inventory of all radioactive waste present in France:

- Cherbourg (National Defence): rubble resulting from the recasting and dismantling of installations are stored in containers, accounting for 1 GBq with mainly Co 60.
- Bauzot (mine): a few hundreds of barrels with soil and rubble coming from a fuel processing factory in Fontenay (dismantled 1959) with very low activity content.
- Gueugnom (mine): 40,484 Mg (activity 0.1 TBq) resulting of scouring the factory ground
- Brennilis (nuclear power plant): 180 m³ of concrete have a specific activity above 100 Bq/g (including 18 m³ contaminated with H 3) and 10,000 m³ of concrete having a specific activity between 1 Bq/g and 100 Bq/g (VLLW).
- Chinon A1 (nuclear power plant): 1000 Mg of concrete were contaminated or activated (LLW).
- Chinon A2 (nuclear power plant) – decree in 1985: the estimated weight of concrete contaminated or activated is 1000 Mg (LLW).
- Chinon A3 (nuclear power plant) – decree in 1990: 1500 Mg of concrete were contaminated or activated (LLW).
- Saint Laurent des Eaux A1 (nuclear power plant): the estimated concrete contaminated or activated amounts to 1500 Mg (LLW).
- Saint Laurent des Eaux A2 (nuclear power plant) : the estimated weight of concrete contaminated or activated amounts to 1500 Mg (LLW).
- Bourges (National defence): 18 barrels (200 l each) of rubble contaminated with U 238, resulting from firing tests.
- Chooz A (nuclear power plant) stopped in 1991: 1000 Mg of contaminated or activated concrete.
- Saint Nicolas at Aliermont (non nuclear industry): due to handling of luminescent paintings, several points of the factory are contaminated with Ra 226.
- Serquigny (non nuclear industry): the factory for thorium nitrate production was demolished and 100 m³ (250 Mg) of Th 232 contaminated rubble with an activity of 87,5 GBq are stored.
- Chilly-Mazarin A87: the decontamination of the old factory of Le Bouchet produced 2200 m³ of soil and rubble resulting from decontamination of the radium plant.
- Le Bouchet: the amount from the old factory was 8,000 Mg of rubble (74 GBq Ra 226).
- Montboucher: the decontamination of the old factory produced 36,765 Mg of soil and rubble with activities on the order of 1 Bq/g (U and Th)

- Saclay (CEA): wastes arising from decontamination and dismantling stored on site in a tumulus (1050 m³ of rubble contaminated with Cs 137 and Eu 152 with 15.6 GBq). 3127 waste containers (concrete blocks) were reused as structure of various buildings.
- Saclay: old residues (150 m³ of concrete activated with Eu 152, having a specific activity below 16 Bq/g) are stored, waiting for decision for evacuation.
- Fontenay-Aux-Roses: reactor ZOE represents approximately 500 Mg of concrete to dismantle (contamination with Co 60 of about 37 GBq).
- Fontenay-Aux-Roses: RM2, the laboratory for irradiated fuels (containing Pu) will produce 8 m³ of concrete with specific activities above 100 Bq/g, 300 m³ of concrete with specific activity between 1 Bq/g and 100 Bq/g, and 2000 m³ of concrete with specific activity below 1 Bq/g
- Arcueil (non nuclear industry): the old site of radioelements (radium), transformed into research laboratory must be decontaminated or demolished (contamination Th 230 and U 238).
- Bellegarde: dismantling of the chemical unit of Uranium enrichment will give rise to 60 m³ of soil and rubble (37 GBq of U).
- Marcoule (reprocessing plant): approximately 14,000 m³ of rubble are stored in CDS (alpha contamination: 2 GBq, beta and gamma: 49.7 GBq) which result from the operation and maintenance of the installations.
- Brugeaud (mine): the mine is used as a storage for 16,790 Mg of soil and rubble coming from dismantling the Le Bouchet factory (70 GBq of Ra 226, Th and U).
- Bugey 1 (old nuclear power plant): the estimated concrete mass contaminated or activated is estimated to 1,500 Mg for total dismantling.
- Grenoble: for the Melusine reactor which was shut down in 1988, the concrete mass estimated (except civil structure) for dismantling of the installation to IAEA level 3 is 8 m³ concrete with a specific activity above 100 Bq/g (20 TBq of Co 60) and 3 m³ concrete having a specific activity below 100 Bq/g.
- Pierrelatte (Cogema): the site is used as surface storage for the radioactive waste waiting for evacuation. 331 barrels of rubble resulting from the gas diffusion and the U chemistry are contaminated with U 234, U 235 and U 238.

2.1.2.3 Management options for the buildings and building rubbles

After the final shut-down of a nuclear power plant, the fuel and operating fluids are removed. A partial dismantling (for 7 years) will transform the plant into a temporary storage for its own equipment. Then the installation will stay under surveillance during 40 years. The last step is the total dismantling for unrestricted use. Research facilities where nuclides with long half-lives have been used will be dismantled directly after closure, because the radiological benefit from several devades decay time is very low. For some facilities (e.g. the UP1) dismantling for unrestricted use has not yet been planned.

Today, the clearance strategy is to divide between “radioactive wastes areas” (RWA) and “non nuclear wastes areas” (NNWA). All the wastes coming from RWA are managed as radioactive wastes, and sorted into VLLW and LLW (or MLW) according to their specific activity. The original intention had been that wastes coming from the NNWA could be released as non nuclear wastes without radiological measurement. Now it is more and more advised to check the specific activity before release. All the wastes are to be released or disposed of according to identified pathways. A report justifying the division into RWA and NNWA and describing and justifying the waste pathways is to be accepted by authorities (DSIN) before starting the dismantling. Today about 30% of concrete

rubble from the conventional sector is being recycled in France while no recycling takes place from nuclear power plants. It is still open whether this might change in the future.

2.1.3 Installations in Europe in general

Meanwhile, a considerable amount of data on other decommissioning projects in Europe which are very advanced or nearing completion, like the Windscale Advanced Gas-Cooled Reactor (WAGR) in the UK, are available and are supplementing the information and data gained from earlier decommissioning projects [NEA 96]. Further information characterizing decommissioning waste can be found in many publications like for example [WAT 87], [SMI 85], [IAE 95], [RUO 94], and [HAR 95].

In order to estimate the total concrete masses – and the time of their generation – it is advisable to choose a generic approach on a very broad basis. It was pointed out in the sections above, that most of the rubble is produced by level 3 dismantling of nuclear power plants. Therefore, the following estimation is more elaborate for power plants than for other nuclear facilities. As the data available for concrete masses in power plants is limited, reference data for one plant of each plant type is used. A linear extrapolation of the concrete masses in relation to the power output for smaller and larger units of the same type is assumed in order to generate the data for units with different power ratings.

In addition, it is necessary to specifiy the time for reaching level 3 dismantling conditions. For each country in the EU and each reactor type, a typical operating time and a typical time of the post operational phase and of safe enclosure is assumed. With these data the time when level 3 dismantling commences can be estimated from the date when the plant started operation. For plants which have already been shut down, only the time for the post operational phase and the safe enclosure is relevant. In table A3–1 and table A3–2 in the appendix, the list of all power plants which are included in the calculation are shown with their specific data.

For research reactors and fuel cycle facilities it is not feasible to use the same assumptions as for power reactors. On the one hand, the variety in types of such facilities is much larger and the data on concrete masses in such facilities is even more scarce. On the other hand, the total waste masses are small in comparison to those of power reactors so that larger uncertainties in these fields do not have large effects to the estimation of the total mass. Therefore, only one typical value for each facility (mass of the concrete per power) is used. In addition, for each country a typical time span is assumed between the date of final shut-down and level 3 dismantling. The shut-down date for operating facilities has been estimated for each plant on the basis of typical plant lifetimes. Tables A3–3 and A3–4 in the appendix provide an overview of all research reactors and all fuel cycle facilities which are included in this estimation.

Table 2–3: Assumptions for the estimation of waste masses in Europe (for detailed information see appendix A)

Country	Number of NPPs	Planned Operating time [a]	Decay time [a]	Building rubble per power [Mg/MW_e]
Nuclear Power Plants				
France, Italy	71 4	40	30	PWR 111,5; FBR, GGR and GCHWR 145; Magnox 111
Germany	34	40	10	PWR 111,5; BWR 237,5; HTR 250; FBR 180, VVER 111,5
United Kingdom	46	GGR 25; AGR 30; PWR 40	130	GGR / AGR 145; DWR 111,5; FBR 145; HTGR, SGHWR / Magnox 111
Belgium, Finland, Netherlands Spain, Sweden	8 4 2 14 12	40 (WWER 25)	10	PWR 111,5; BWR 237,5; GGR 145, VVER 111,5
Research Reactors				
UK	7	Individual	20	All: 100 Mg/MW_{th}
Germany Belgium Netherlands Denmark Italy France others	14 3 2 2 2 15 8		20	
Fuel Cycle Facilities				
France	11	Individual	30	Enrichment 0,4 [Mg/(swu y)] Fuel Fabrication 40 [Mg/(Mg_{UO} y)] Interim Storage 20 [Mg/(Mg_{HM} y)] Reprocessing 30 [Mg/(Mg_{HM} y)]
UK	21		130	
All Others	19		10	

Abbreviations: MWe = electrical power output, PWR = Pressurized Water Reactor, BWR = Boiling Water Reactor, FBR = Fast Breeder Reactor, GGR = Gas-Graphite Reactor, GCHWR = Gas cooled Heavy Water Reactor, HTR = High Temperature Reactor, AGR = Advanced Gas-cooled Reactor, SGHWR = Steam Generating Heavy Water Reactor, HTGR = High Temperature Gas-cooled Reactor, VVER = Pressurized Water Reactor (Russian Type)

Table 2–3 shows the general data for all facilities (nuclear power plants, research reactors and fuel cycle facilities) which were used. The first and second column show the country for which installations of a specific type (nuclear power plants, research reactors, fuel cycle facilities) have been included and the number of these installations. The third column presents the assumptions on the normal plant life-time (i.e. the average total operating time of a plant type, not the time from now until start of decommissioning). The fourth column indicates the time that is normally allowed for decay after plant closure which may be a specific safe enclosure period or simply the time before dismantling of the buildings can commence. The last column lists the assumptions on the specific mass (in Mg) of the entire plant per power (in MW_e or MW_{th}) or per capacity (in appropriate units, like per Mg heavy metal production per year). It should be carefully noted that the assumptions listed in table 2–3 are only an estimate which is, however, accurate enough for the present analysis.

It should further be noted that this estimation does not include any new nuclear installations that might be built in the future, any nuclear installations in countries that might become member states of the European Union in the future, and any accelerators.

The results are presented in figure 2–3. The mass as a function of time shows two distinct peaks in the range between 2020 and 2040 as well as between 2070 and 2090. The first peak is caused by nuclear power plants that will be dismantled soon after their final shut-down, the second peak corresponds to those installations for which a safe enclosure of several decades is foreseen prior to final dismantling. It can be seen that in this model building rubble will arise also in the time after the year 2100. This corresponds to installations mainly in the UK where a long term safe enclosure with an enclosure period of around 130 years is envisaged. With respect to the quantities, figure 2–3 clearly shows that the largest quantity of building rubble will arise from those nuclear power plants which are currently operating.

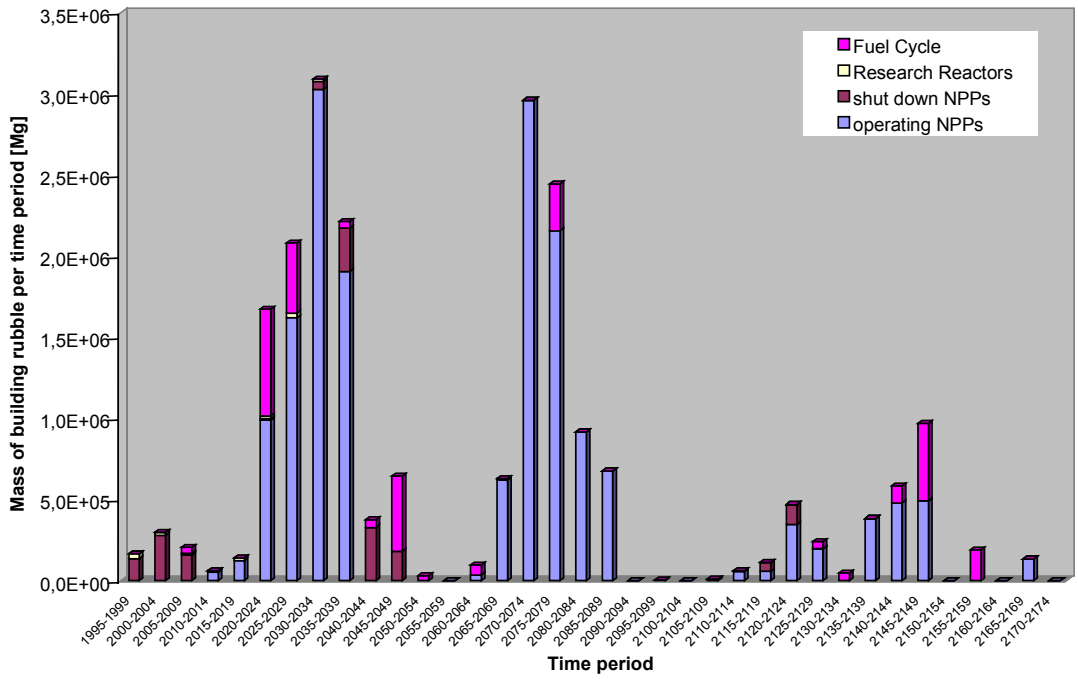


Figure 2–3	Total mass of building rubble from all types of nuclear facilities which is expected to arise in Europe per 5a period (only presently existing installations included)	BS

2.2 Recycling and Disposal Options for Building Rubble and Buildings

Not only the quantities of rubble or the number of installations are important for the radiological assessment which is carried out in section 4, even more important are the material pathways, the further use of the material or the buildings, and details of possible exposure scenarious which apply for the various steps of handling or disposing of the material or the buildings. After a building or building rubble has been cleared, many options are available which are shown schematically in figure 2–4. The steps along the route depicted in this figure are subsequently transformed into radiological scenarios.

Upon clearance of the building, the first important decision is whether the building shall be left standing for reuse for a non-nuclear purpose or whether it shall be demolished without previous reuse. If the building is left intact, scenarios have to be included that describe the renovation of the building and its later occupation or industrial use.

The other possibility is that the building will be demolished without previously reusing it for a non-nuclear purpose, which has proven to be the more likely case for buildings of nuclear power plants and fuel cycle installations. This is depicted at the right branch in figure 2–4. The resulting building rubble can either be disposed of at a landfill (the very right branch in the figure) which only makes it necessary to take transport and disposal scenarios into account in the radiological analysis. The other main option is to recycle the building rubble.

Generally, the rubble must first be processed (including crushing) and then sorted according to grain sizes depending on the later use (see [COR 94]). The material can be used in civil engineering for road construction or as an additive for manufacturing of new concrete, which is shown at the bottom in the left sub-branch in figure 2–4. In these cases, a number of scenarios have to be taken into account, like living in a house which is manufactured by using this concrete or working on a cover which is made by this material (like e.g. the ground cover of a gas station or a parking place), but also the scenarios describing the construction and recycling processes itself, like construction of road surfaces, building foundations or surface covers.

Rubble can also be used in foundations, to backfill holes or in recultivation and landscaping projects for which the rubble does not necessarily need to be processed. This is shown in the sub-branch in the middle of figure 2–4. Because in these cases the material does not form part of a new concrete matrix, the radiological assessment has to concentrate also on washout processes by which the nuclides might migrate to groundwater and from there to the food chain. In its radiological relevance, this option resembles the disposal option described above, although the material is of course recycled and not directly disposed of.

Both recycling options with processing as well as without have been studied in detail (see e.g. [HAR 95], [JOH 94] among other studies).

Building rubble from nuclear installations could, however, also be used for other purposes where the contact to the general population is significantly reduced. Backfilling of underground mines, manufacturing of concrete overpacks for waste disposal or in grouting waste packs are examples of alternative management options [JOH 93]. Such options must be reviewed in connection with the national regulations and could after a radiological impact study be implemented for material which has an activity level above the clearance level. This type of clearance is generally referred to as conditional clearance (see [IAE 96]). Such options are, however, not dealt with in this study.

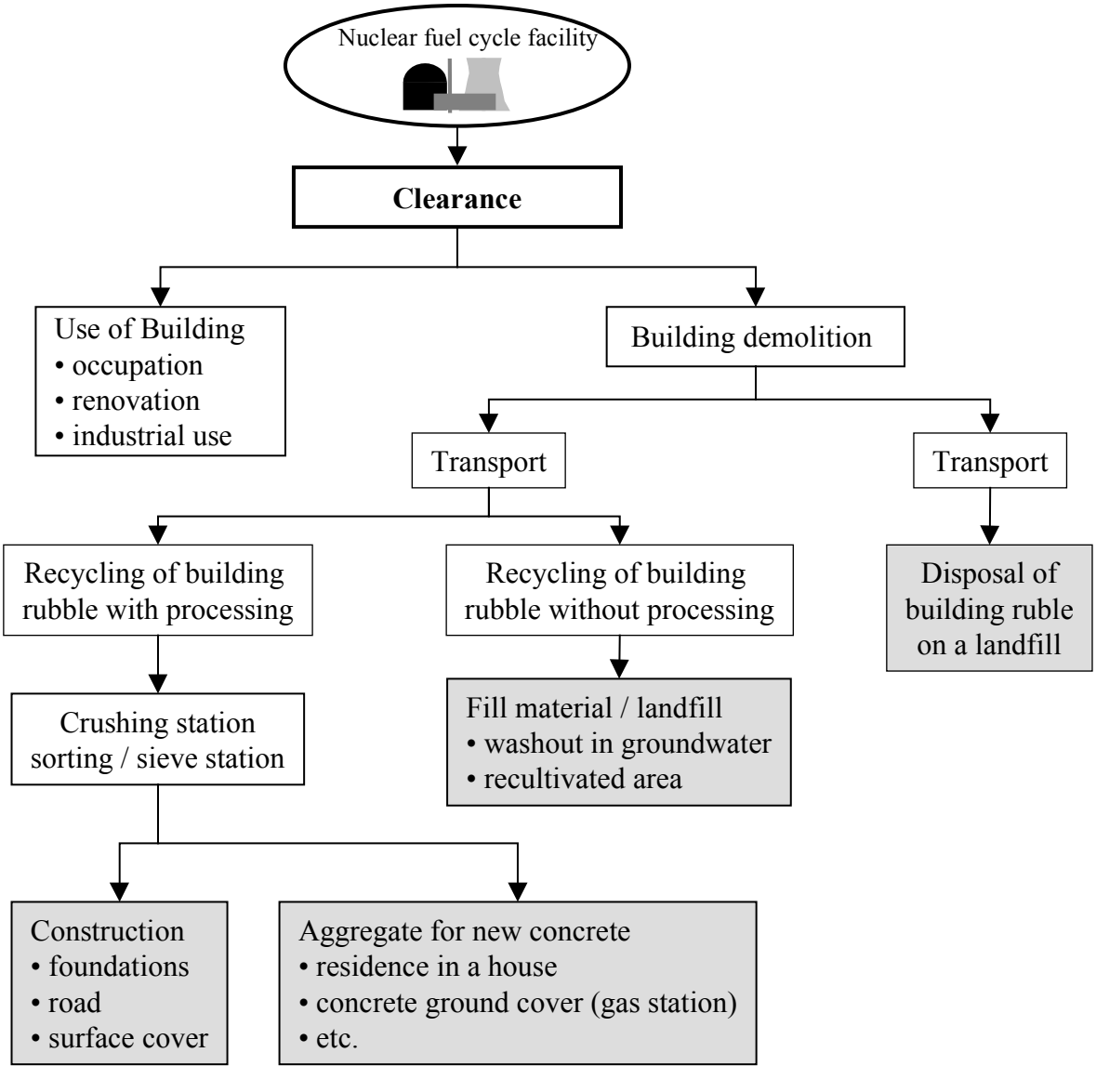


Figure 2–4 Recycle, reuse and disposal options for cleared buildings and building rubble

3. ASPECTS INFLUENCING THE DOSE CALCULATIONS

3.1 General Aspects

Contaminated buildings can originate from many different types of facilities. Each facility will have a typical set of radionuclides in its contamination vector. The reuse or recycling options are dependent on the type of facility as well. For example, a nuclear power plant uses high quality concrete in its construction so that the recycling of the building rubble is generally economic. A decommissioned isotope laboratory in a hospital or research center will most likely be reused after clearance. The goal of creating clearance levels which are valid for this wide range of radionuclides, facility types and uses, means that a certain amount of pessimism must be included in the selection of the parameters.

Each facility will have its own set of characteristics such as material quantity, preferred management option, activity level, relevant radionuclides, etc. As the objective of this study is to develop generic clearance levels which apply to all types of facilities, it has been decided to calculate radionuclide specific clearance levels which can be applied to material containing a radionuclide mixture by using a summation formula. While this approach has a number of advantages, the main one being ease of application for the competent authorities, there also are disadvantages as briefly outlined in the following.

The radiological analysis is in general based on large amounts of material, in particular from nuclear power plants. A significant number of the radionuclides considered are not present in any significant quantity in the typical radionuclide mixes coming from such facilities and hence the cleared quantities are over-estimated for such radionuclides. The authorities should be aware that these clearance levels may therefore be overly restrictive in particular when the facilities are small like research laboratories. On the other hand the authorities should also be aware that multiple facilities at the same site are likely to use the same waste management strategy which could lead to larger quantities being treated in the same way. The clearance levels developed here will ensure that Article 5 in conjunction with Annex I of the Euratom Directive is complied with also for large quantities.

Article 5 in conjunction with Annex I reflects the main aspects of the IAEA recommendation Safety Series 89 [EUR 98] in Community legislation. The IAEA recommendation, laid down in Safety Series 89, refers to an individual dose of “some tens of microsieverts per year” (μSv/y) as being trivial and therefore a basis for clearance. Furthermore, the IAEA suggests that in order to take account of exposures of individuals from more than one exempt practice, “each exempt practice should utilize only a part of that criterion, and it may be reasonable for national authorities to apportion a fraction of that upper bound to each practice. This fractionation could lead to individual doses to the critical group of the order of 10 μSv in a year from each exempt practice” [EUR 98]. For comparison 10 μSv/y corresponds roughly to around 0.5 % of the average natural background. In addition, the IAEA recommends that for each practice a study of available options be made by the regulating authorities in order to optimise radiation protection. If the study “indicates that the collective dose commitment resulting from one year of the unregulated practice will be less than about 1 manSv ... it may be concluded that the total detriment is low enough to permit exemption without a more detailed examination of other options”. The general international consensus for the basic criteria for exemption is reflected by their inclusion in both the IAEA BSS [IAE 94] and Eu-

ratom BSS. In addition, the work leading to the exemption values in the BSS limited the skin dose to 50 mSv/y [EUR 93].

In order to relate the dose received by individuals to a specific practice, or to the levels of radioactivity involved in a practice, a set of exposure scenarios, which relates the activity content to an individual dose is constructed. Clearance levels can finally be derived by dividing the dose criterion (10 µSv/y or a skin dose of 50 mSv/y) by the calculated dose (per Bq/g or Bq/cm²) from the most critical scenario.

3.2 Radionuclides

Radioactivity in nuclear installations originates from the nuclear fuel, including fission products and neutron capture products (e.g. Sr 90, Cs 137, U 235, U 238, Pu 239, etc.) and from radionuclides created by neutron flux, activation products (Fe 55, Co 60, Ni 63, etc.). A differentiation is made between radioactivity that is transported for example by air or water to an item (which is termed "contamination"), and radioactivity within an item created by neutron flux (which is termed "activation"). Activation products are created in reactors and are transported throughout the reactor and its systems as contamination. Fission products are also found in the contamination spectra of many nuclear facilities. The radioactivity used in medicine, research and industry are typically created in reactors using neutron flux.

It is not possible to give a standard radionuclide spectrum for each type of nuclear facility. The nuclide spectra in nuclear power plants for example depend on the type of fuel, if fuel was reprocessed, the core geometry, the structural and building material, etc. A significant amount of literature exists which investigates nuclide spectra and how they change over time [SMI 85] [DEC 97] [ELE 92] [KIS 94]. Such data is important for deciding on decommissioning strategies, but is less important within the context of nuclide specific clearance levels. Nevertheless, the radionuclides which normally are found in the various facilities are known (cf. e.g. [DEC 97]). The radioactivity in cleared building rubble from nuclear power plants for example will generally be dominated by Co 60 and Cs 137, in nuclear fuel cycle facilities by uranium isotopes (predominantly by U 234) along with fission products and actinides and in research and medical facilities by the radioisotopes handled there which typically include among other radionuclides H 3, C 14, S 35, P 32 and I 125 [IAE 98] [DEC 97].

Because of the variation of nuclide spectra for the various types of installations and even within a single installation, clearance levels are calculated for all the radionuclides for which exemption levels in the Euratom Directive [CEU 96] exist and which have a half life longer than 60 days, with the exception of the noble gases. Recommended clearance levels for metal items and scrap for this same set of radionuclides can be found in [EUR 98]. The short lived radionuclides generally play only a minor role in clearance, especially for the large quantities coming from decommissioning, since the clearance generally occurs many months after the shut down of the facility. They are therefore not included in this study.

A number of those radionuclides which are further considered hiere decay into unstable short lived radioactive products. The doses from the decay products with a half life shorter than 60 days are accounted for by assuming that they are in secular equilibrium with the mother nuclide and adding their doses to the dose calculated for the mother nuclide. In table 3–1 the progeny, which were considered to be in secular equilibrium are explicitly given.

Table 3–1 List of radionuclides with short-lived progeny assumed to be in equilibrium

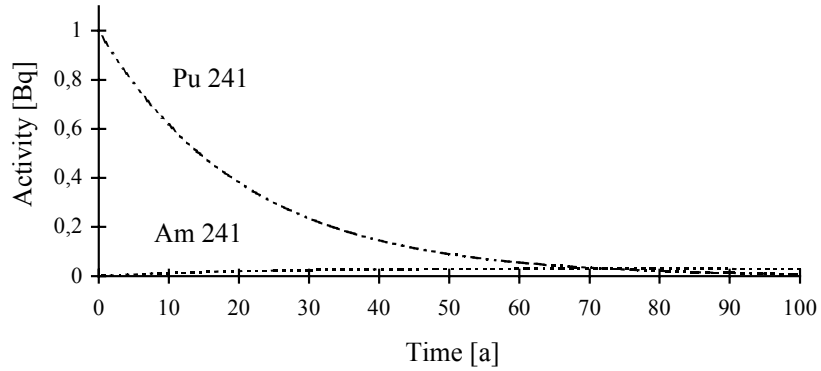
Parent	Progeny included in secular equilibrium
Sr 90	Y 90
Zr 95	Nb 95, Nb 95m
Ru 106	Rh 106
Pd 103	Rh 103m
Ag 108m	Ag 108
Ag 110m	Ag 110
Cd 109	Ag 109m
Sn 113	In 113m
Sb 125	Te 125m
Te 127m	Te 127
Cs 137	Ba 137m
Ce 144	Pr 144, Pr 144m
Pb 210	Bi 210
Ra 226	Rn 222, Po 218, Pb 214, Bi 214, Po 214
Ra 228	Ac 228
Th 228	Ra 224, Rn 220, Po 216, Pb 212, Bi 212, Tl 208, Po 212
Th 229	Ra 225, Ac 225, Fr 221, At 217, Bi 213, Tl 209, Po 213, Pb 209
U 235	Th 231
U 238	Th 234, Pa 234m, Pa 234
Np 237	Pa 233
Pu 244	U 240, Np 240m, Np 240
Am 242m	Np 238, Am 242
Am 243	Np 239
Cm 247	Pu 243
Es 254	Bk 250

3.3 Dose Coefficients

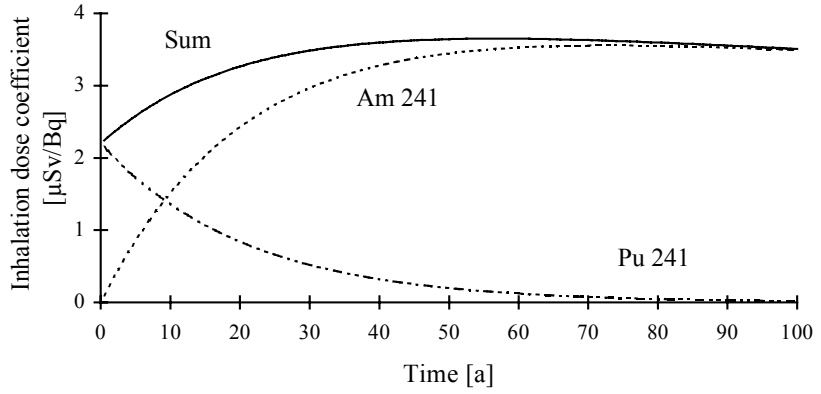
The incorporation dose coefficients (inhalation and ingestion) are taken from the Euratom Directive [CEU 96]. The β-skin doses are calculated using the doses coefficients in [KOC 87] for a skin depth of 7 mg/cm² as recommended in ICRP 26. The exposures from external γ irradiation from a semi-infinite volume are calculated using the tabulated dose rate versus photon energy values from [CHE 91], the exposure inside a room is calculated using a spherical geometry and the fluence to dose conversion factors from ICRP 51. The nuclide specific dose coefficients are listed in the appendix.

After material is cleared the radionuclides decay. Some of these nuclides decay into other long lived radionuclides which may be more radiotoxic than the mother nuclide. An example is Pu 241 which decays into Am 241. In figure 3–1 (a) the activity as a function of time is shown for an initial quantity of 1 Bq Pu 241. The activity maximum of the daughter nuclide Am 241 occurs at about 70 years at which time the total activity represents only a fraction of the initial activity. In figure 3–1 (b) the inhalation dose coefficient is plotted for material in which the initial activity of Pu 241 is 1 Bq. In contrast to the activity, the dose coefficient increases over time reaching a maximum at around 60 years although at this time the total activity has decreased to less than 0.1 Bq. This demonstrates

that if the cleared material remains together for a prolonged period of time, the scenarios occurring many years after clearance can lead to higher doses than those calculated for the first year after clearance. For buildings and building rubble it can in many cases be expected that the material will stay together for such a prolonged period of time. Cleared buildings, for example, can be put to another use and might only be demolished many years later. For rubble which is used in a single project like road construction, the exposure to the workers who renew the road many years later can thus exceed the exposure which would have resulted from the same material directly after clearance. It is also conceivable that a landfill be reused after closure so that the post-closure exposures could exceed those at the time of disposal for such radionuclides (see [RUO 94] [IAE 87] [SMI 85]).



(a)



(b)

Figure 3–1 Activity and dose coefficient as a function of time for Pu 241 and the radioactive daughter Am 241

To account for these possibilities without explicitly developing scenarios for the above discussed exposure possibilities, the maximum dose coefficients within the first 100 years after clearance were calculated and are used when calculating the radionuclide specific doses. The 19 radionuclides for which the progeny lead to higher dose coefficients during the first 100 years after clearance are

listed in the appendix in table A2–2. After 100 years it can in general be assumed that the cleared material will no longer be together. This mixing over time will lead to a reduction in the expected doses.

3.4 Relationship between Surface and Bulk Activity

For nearly all building structures, with the exception of the biological shield in a reactor, and the shielding walls in accelerators, the contamination will come from deposits on the surfaces of the buildings. From the surface the activity can then migrate into the structure, although, with the exception of tritium, it will typically not penetrate more than a few centimeters into the wall. In many instances the building's surfaces will be treated (e.g. coatings on surfaces which allow effective decontamination), or the contamination mechanism will be such that the penetration of the radionuclides into the wall will be significantly less than one centimeter. This is one of the reasons why the most efficient way to limit the cleared activity is to carry out the clearance measurements on the standing structure. Here, a distinction between removable, fixed and total surface activity is only meaningful on undamaged decontamination coatings on walls and floors. Any surface specific clearance levels should therefore apply to the total (sum of removable and fixed) activity on and in the structure.

The recommendation for the clearance of buildings and building rubble from the German Commission on Radiological Protection (SSK) [SSK 95] requires that the clearance measurements be carried out on the standing structure before demolition which has proven to be very effective in some German decommissioning projects (cf. section 2.1.1). The building rubble which results from the demolition work (which will take place only *after* the clearance has been authorized by the competent authority) can then be dealt with as conventional building rubble without any further restrictions. This approach has the advantage of making the measurements where the activity is located, which increases the chances of finding contamination and removing it before clearance and thereby optimizing radiation protection. After demolition, it is no longer possible to easily locate the activity since it is mixed up with the non-contaminated interior of the walls. Regardless whether the clearance measurements are carried out on the surface of the building or after demolition on the rubble, the total cleared activity must be limited, since this is the radiologically relevant quantity.

To adequately implement the radiological protection goals using only surface specific limits requires that the activity which has penetrated into the structure be counted as surface activity. This is shown schematically in figure 3–2. For radionuclides which emit high energy gamma rays this does not represent a significant restriction since the origin of the gamma ray is not determinable. Therefore this simply means that the total gamma ray emissions should be used in the clearance measurement. The actual measurement, though, will in many cases not be trivial since natural background, the penetration depth of the radionuclides, etc. can make their measurement difficult (see [TÜV 96]).

Including β and α emitting radionuclides which have penetrated into the bulk structure can pose a difficult measurement problem. Here it may be necessary to take surface samples to determine the average penetration depth, which can then be used to calculate a correction factor for the surface measurements. It may also be possible to use correlation factors with easily measurable radionuclides thereby avoiding the necessity of extensive laboratory analyses. For particularly difficult to measure radionuclide mixtures, it may be necessary to use a statistical surface sampling scheme to calculate the average surface activity. In any case the new Euratom Directive [CEU 96] requires

that the clearance be authorized by the competent authorities, so that part of their responsibility should be to ensure that the selected clearance measurement procedure adequately accounts for the total cleared radioactivity, thereby ensuring that the radiation protection goals are met.

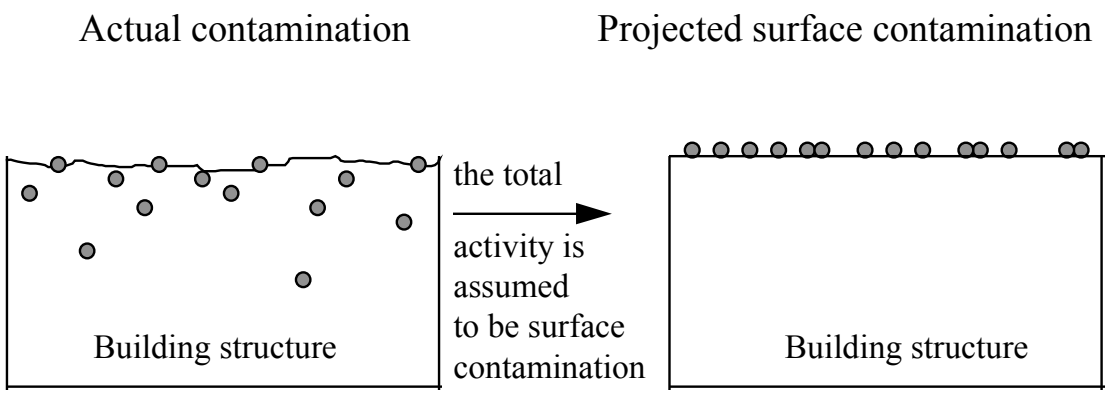


Figure 3–2 Schematic diagram demonstrating the projected surface contamination which is used to calculate the exposures

3.5 Averaging Areas and Masses

The contamination in building structures is typically not homogeneously distributed throughout the building. When applying clearance levels it will therefore be necessary to define an area over which the contamination can be averaged.

In general, for a given set of numerical clearance levels, a more restrictive (i.e. smaller) averaging area will lead to clearance of less activity. This can be seen from the following example: if the activity located on an area of 1000 cm² in an otherwise uncontaminated 10 m² area exceeded the clearance level by a factor of 100, then if averaging over 10 m² was allowed this hot spot could be cleared. If, however, in this example the averaging area is less than 10 m² (e.g. 1 m²) then it will be necessary to decontaminate this hot spot which will lead to a reduction in the amount of cleared activity, although the clearance level would be the same in both cases.

From a radiological point of view an averaging area for buildings of about 1 m² is reasonable, since typically the exposure distance from the walls or the floor in a room is on the order of at least 1 m. An inhomogeneous distribution on a scale less than 1 m will not lead to a significant variation in the dose rate in a room. If on the other hand averaging over an entire room was allowed then an area of 1 m² could exceed the clearance level significantly and lead to an elevated dose rate near this area. If this area was part of a work place after clearance then the dose to a person working there could exceed the dose constraint of 10 µSv/a. Such a scenario should not be considered as unlikely since if a laboratory was cleared and continued to be used as a laboratory after clearance then the areas

where the radioactive preparations were handled and where the highest contamination is expected, would most likely be the same areas where the work is carried out after clearance. It is therefore prudent, especially for building reuse, to limit the averaging area to about 1 m².

In the scenarios developed here (section 4) it is assumed that a large building or a large quantity of building rubble is cleared. If an averaging area of 1 m² is applied to the building and all contaminated areas which exceed the clearance level are decontaminated, then the total remaining activity, averaged over the entire building structure, will be significantly below the clearance level. For example it can be expected that in general the ceilings will be only slightly or not at all contaminated. This fact is therefore taken into account in the dose calculations by including a factor of 0.33 which describes the degree to which the clearance level is reached on the average.

For rubble resulting from renovation or refurbishing, like for example the removal of walls, it will not be possible to carry out surface specific measurements. Should the clearance *after* demolition be favored for a single building or the entire facility, then for this rubble as well mass specific clearance levels would be required since no well defined surface exists after demolition. The clearance of very large quantities of building rubble using mass specific measurements is, however, not trivial. If the rubble is measured in 100 kg batches then 10^6 measurements would be required to clear 100,000 Mg of rubble. On the other hand if significantly larger batches, for example 1 to 10 Mg, are allowed, it will be difficult to guarantee, using a single measurement, that the activity characterization is representative. Therefore, multiple samples from each batch will probably have to be measured which means that once again on the order of 10^6 measurements will be required. This should be compared to approximately 10^5 surface measurements, under the assumption that every 1 m² is measured once. Typically about 1 Mg of building material will be located below an averaging area of 1 m² for modern nuclear fuel cycle facilities so that an averaging mass of around 1 Mg would be consistent with the averaging area of 1 m².

4. RADIOLOGICAL ANALYSIS

4.1 General Approach to Calculation of Clearance Levels for Buildings

After termination of authorized practices at a nuclear facility the buildings must be dealt with in a responsible way. Since typically the buildings are only slightly radioactive it should, in most cases, be possible to clear them after decontamination. The criteria for clearing buildings or the resulting building rubble must ensure that the 10 μSv concept, as laid down in Art. 5 of the Basic Safety Standards [CEU 96] is complied with. This is typically achieved by setting down numerical clearance levels for the maximum allowed activity, as well as additional clearance criteria, such as averaging quantities, which define how the clearance levels are to be applied. To guarantee that the criteria are met the possible uses of the building or the building rubble must be identified and the radiological consequences analyzed (cf. figure 2–4 and section 2.2).

The scenario leading to the largest exposure depends on what happens to the building after clearance. Two basic possibilities exist, either the building can be left standing and put to another use or it can be torn down. If the demolition option is chosen the resulting rubble can either be disposed of at a landfill or recycled. These options have been analyzed in detail and the radiologically critical

scenarios identified [SMI 85] [ASS 95] [ELE 92] [RUO 94] [JOH 94] [HAR 95] [DEC 93] [DEC 96] [RAU 95]. Typically the critical scenarios involve large amounts of material which can accumulate at one location and remain there during a significant period of time. Therefore particular attention must be paid to the decommissioning of nuclear power stations which represent one of the largest sources of potentially contaminated building material in Europe (see section 2.1).

The *simple* application of a deterministic scenario to the quantity of material expected from the decommissioning of a nuclear power plant (approx. 10^5 Mg) results in clearance levels which are not practicable. The following straightforward calculation for the external exposure from Co 60 for a landfill worker who is assumed to process the material in one year exemplifies this problem. The dose equation can be written as

$$H_{ext} = A \cdot D_{ext} \cdot Q/P_r$$

where

H_{ext}	2000 [µSv/a] individual dose received by the landfill worker,
D_{ext}	1 [(µSv/h)/(Bq/g)] Co 60 dose rate for a semi infinite geometry [CHE 91],
A	1 [Bq/g] activity concentration in the material,
P_r	50 [Mg/h] processing rate [DEC 93] [HAR 95] and
Q	100.000 [Mg/a] quantity of building rubble.

The individual dose received by the landfill worker would amount to 2 mSv per 1 Bq/g activity content in the material, which in this simple calculation corresponds to a clearance level for Co 60 of 0.005 Bq/g (i.e. a dose of 10 µSv/a). Such a small clearance value is totally impractical to apply in decommissioning. The problem is that the scenario assumes a *homogeneous* contamination of the entire material which does not occur in reality. Nevertheless the scenario shows that averaged over the entire 100,000 Mg, an activity level of about 0.005 Bq/g is necessary in order to ensure compliance with the 10 µSv-concept.

In [HAR 95] the problem was circumvented, but not solved, by assuming that only 2700 Mg of building rubble from a single facility is contaminated. As a result a clearance level for Co 60 of 1.56 Bq/g was recommended. However, without appropriate conditions for the application of such a value it would be possible e.g. to mix the biological shield into the rubble, since the total contamination on and in the building material of a nuclear power plant is estimated at about 10^{10} Bq [SMI 85] and averaging over the entire 10^5 Mg would allow $1.5 \cdot 10^{11}$ Bq (1.5 Bq/g $\cdot 10^{11}$ g) to be cleared. Clearing the biological shield in this way is from a radiological point of view certainly not desirable and would cause doses which significantly exceed 10 µSv/a.

On the other extreme, in the study [ELE 92] a source term of 150,000 Mg of homogeneously contaminated building rubble was used in the radiological evaluations, which resulted in clearance levels on the order of 0.005 Bq/g for Co 60 as outlined above. This study was discussed during the work which lead to defining the unconditional clearance levels for TECDOC 855 [IAE 96] but the values were not used, since they were not considered realistic or practical.

Making clearance measurements on the surface of the standing structure, as recommended by the German Commission on Radiological Protection [SSK 95], will avoid these problems and guarantee that the 10 µSv/a dose constraint is complied with. Alternatively a larger mass specific clearance level (e.g. 1.56 $Bq_{Co\ 60}$/g from [HAR 95]) could be supplemented with an annual release limit which would then guarantee that, averaged over the entire material from the demolished structure, the very

restrictive mass specific activity (e.g. 0.005 $Bq_{Co\,60}/g$) is not exceeded and thereby guarantying compliance with the 10 μSv/a dose constraint.

4.2 Clearance of Buildings for Continued Use as a Non-Nuclear Facility

A cleared building can be left standing and used as a factory, storage facility, workshop, museum etc. Scenarios for evaluating the radiological impact of clearing buildings for continued use are developed in this section. The strategy pursued here is to identify the most restrictive scenario for each of the exposure pathways: external irradiation, inhalation, ingestion and skin contamination, and then use these scenarios to derive the clearance levels by setting the level such that the resulting dose equals 10 μSv/a. The results of the calculations are given in section 5.

Since the building is to be reused the clearance measurements must be carried out on the surface of the structure. The clearance levels developed here are therefore expressed in terms of Bq/cm². Of course the doses are also caused by the activity which has migrated into the structure, for this reason the radiological analysis assumes that the surface specific activity includes the activity which has migrated into the structure (see figure 3–2). No mass specific clearance levels are developed for the reuse of buildings although mass specific measurements of surface samples may be necessary to show compliance with the surface specific clearance levels when hard to measure radionuclides are present. In this case mass specific values can be calculated from surface specific values by using the migration depth and material density. The use of mass specific values together with the migration depth would be equivalent to using surface specific levels.

It must be emphasized that the radiological analysis for building reuse as outlined in this section also takes into account the scenarios which are developed for clearance of buildings for demolition, as outlined in section 4.3. This is necessary because of the fact that a building, once cleared for reuse, may all the same be demolished shortly after clearance resulting in same situation as in the case of clearance for demolition only. In addition, if fuel cycle facilities where long-lived nuclides dominate the nuclide vector are to be cleared for building reuse, activity and dose reduction from the radioactive decay will generally be so small that even decades after clearance the radiological situation concerning building demolition is unchanged.

4.2.1 External γ-dose

A factory worker in a cleared building or a researcher in a cleared laboratory represent realistic scenarios which can be expected to occur. These scenarios are used to evaluate the radiological consequences of clearing buildings. The dose factor is based on a point kernel integration with spherical geometry. To account for the photon reflection from the walls of the room, a calculation for Co 60 using the photon transport code MORSE [EMM 75] was conducted and compared to the point kernel approximation for the radionuclide Co 60 [DEC 96]. The ratio between MORSE and the point kernel calculation f_r is approximately 1.5 which is used for all radionuclides to account for the reflection of the γ-emissions inside the room. As was discussed in section 3.5, a factor f_d is introduced into the dose equation to account for the average contamination on surfaces much larger than the averaging area. Furthermore, the decay of the radionuclide in question during one year time of the scenario is accounted for by a decay factor.

The external γ dose in a surface contaminated sphere is calculated using the following formula,

$$H_{ext} = D_{ext} \cdot t_e \cdot f_d \cdot f_{dec} \qquad \text{eq. 4-1}$$

where,

H_{ext} [(µSv/a)/(Bq/cm²)] individual annual dose from external irradiation per Bq/cm²,
D_{ext} [(µSv/h)/(Bq/cm²)] average dose rate during the exposure year,
t_e [h/a] exposure time,
f_d [-] factor to account for the average contamination in the building, and
f_{dec} [-] factor to account for the decay during 1 year the scenario with

$$f_{dec} = \frac{1 - e^{-\lambda \cdot 1a}}{\lambda \cdot 1a}$$

The dose rate D_{ext} in equation 4-1 is calculated using the following formula,

$$D_{ext} = f_r \int_0^a dt \cdot \varepsilon_t \cdot \sum_i F(E_i) \cdot w_i \cdot \phi \qquad \text{eq. 4-2}$$

where the parameters have the following meaning:

f_r [-] factor to account for the reflection from the walls,
$\int_0^a dt$ integral over exposure year,
ε_t 1/8760, one hour of the year during which the person is exposed,
i photon index,
$F(E_i)$ [(Sv/h)/(photon/cm²/s)] effective dose equivalent per unit fluence (ICRP 51),
E_i [MeV] energy of i^{th} photon,
w_i [-] emission probability of the i^{th} photon and
ϕ [-] geometric photon flux factor.

The photon flux factor, ϕ, for a spherical geometry is equal to one. The attenuation and build up in air is negligible for the distances inside a building and therefore are ignored in the calculations. The fluence to dose conversions were made using the tabulated values for rotational orientation in ICRP 51. The parameters which are radionuclide independent are given in the table 4–1.

Table 4–1: Building reuse, parameters for the external exposure scenario

Parameter description	Symbol	Value	Units
Exposure time	t_e	1800	h/a
Factor to account for the average contamination on very large surfaces	f_d	0.33	--
Reflection factor	f_r	1.5	--

A number of very unusual and unrealistic scenarios could be constructed which would lead to higher doses. For example, using the nuclear facility as a prison could lead to exposure times on the order of 8760 h/a, likewise turning the facility into a senior citizens home. In the first place, such scenarios can be considered extremely unlikely since the facilities were not designed for such purposes and therefore significant renovation work, and costs, would be required before they could be used in these ways. Secondly, even if the cleared facility should be used in such a way the resulting doses would still be less than 50 µSv/a (8760 h/a is approx. 5 · 1800 h/a) and due to the long exposure times it would not be possible that the occupants be exposed to a second source. Therefore, the

dose from all practices would remain in the trivial dose range (see IAEA Safety Series 89 [IAE 88]). These very pessimistic scenarios are for the above reasons not used to derive clearance levels.

4.2.2 Inhalation dose

In general it is not expected that any significant inhalation doses will occur during building occupancy. Inhalation doses can be expected when the surface is manipulated, like drilling holes or removing the surface layer in preparation for a new surface finish. Therefore, a renovation scenario has been chosen as representative for the critical inhalation exposures. Since it can be expected that large scale renovation work will be carried out by workers, the dose coefficients for workers from table C in the Euratom Directive are used. The mass median aerodynamic diameter of the dust particles created during renovation are typically in the range of several micrometers [JOH 94] [HAR 95]. Therefore the most restrictive dose coefficient for a AMAD of 5 µm was chosen for the dose calculations.

The exposure time (see table 4–2) is taken from the assumption that the abrasive work proceeds at around 10 to 40 m²/h [JOH 94] and that several 1000 m² are treated. The α-emitting radionuclides are generally limited by inhalation scenarios and nuclear fuel fabrication facilities, where significant α-activity is expected, can have contaminated surfaces on this order of magnitude. Typically not the entire surface of a building will be treated in this way and the work will generally not be carried out by one person. Furthermore, before clearance during the decontamination of the building the most contaminated surfaces will be treated using abrasive means or other decontamination techniques until the contamination level is below the clearance level. These surfaces will then be filled in after clearance, immobilizing the remaining activity which will most likely not be released during a subsequent renovation.

Based on measurements of the dust production during abrasive treatment of concrete surfaces [EBE 89] [REG 89], the effective dust concentration is assumed to be 5 mg/m³. This dust concentration in the scenario is only valid during the time that the abrasive treatment is being carried out and does not represent a year long average. It should be noted that during renovation work the workers will carry out various tasks, many of which will create only minor amounts of dust. The assumed breathing rate is typical for physical excursion which can be expected during such work and the dose calculations are carried out assuming that the workers are not wearing protective masks. Since radioactivity has a higher affinity for the small cement particles, which are responsible for the dust, the mass specific activity concentration in the fine dust will be higher than in the surface material [BAK 90] [COR 94]. This activity concentration effect in the fine dust fraction is accounted for in the dose calculations by the factor f_k.

Because of the short duration of the scenario (exposure time 100 h/a) and because of the fact that no significant delay between clearance of the building and start of the renovation work would be necessary, no radioactive decay is assumed.

The dose calculations are made using the following equation,

$$H_{inh} = D_{inh} \cdot t_e \cdot f_d \cdot f_k \cdot C_{dust} \cdot B_r / \delta / \rho \qquad \text{eq. 4-3}$$

where,

H_{inh} [(µSv/a)/(Bq/cm²)] individual dose from inhalation during building renovation,
D_{inh} [µSv/Bq] dose coefficient for inhalation (table C Euratom Directive, 5 µm),

t_e [h/a] exposure time,
f_d [-] factor to account for the average contamination in the building,
f_k [-] concentration factor for the activity in the inhalable dust fraction,
C_{dust} [g/m³] effective dust concentration during abrasive treatment of surface,
B_r [m³/h] breathing rate,
δ [cm] depth of surface layer which is removed during surface treatment and
ρ [g/cm³] density of surface layer.

Table 4–2: Building reuse, parameters for the inhalation scenario

Parameter description	Symbol	Value	Units
Exposure time	t_e	100	h/a
Factor to account for the average contamination on very large surfaces	f_d	0.33	--
Concentration factor for inhalable dust fraction	f_k	3	--
Effective dust concentration during abrasive treatment	C_{dust}	5	mg/m³
Breathing rate	B_r	1.5	m³/h
Removal depth	δ	0.3	cm
Density of surface material	ρ	2.3	g/cm³

4.2.3 Dose from secondary ingestion

Like the inhalation pathway it can also be expected that no significant ingestion of contaminated building material will occur during the normal occupancy of a building. Here again an incorporation can occur when the surface layer is manipulated causing a release of the radionuclides. To estimate the radiological impact due to ingestion it is assumed that during renovation work the workers ingests 1 g of dust. The following equation is used to calculate the doses,

$$H_{ing} = D_{ing} \cdot M \cdot f_d \cdot f_k / \delta / \rho \qquad \text{eq. 4-4}$$

where
H_{ing} [µSv/a] individual dose from ingestion during building renovation,
D_{ing} [µSv/Bq] dose coefficient for ingestion (table C Euratom Directive),
M [g/a] ingested quantity,
f_d [-] factor to account for the average contamination in the building,
f_k [-] concentration factor for the activity in the ingested dust fraction,
δ [cm] depth of surface layer which is removed during surface treatment and
ρ [g/cm³] density of surface layer.

The parameter values with the exception of the ingested quantity are taken from the inhalation scenario and given in table 4–3. An inadvertently ingested quantity used to estimate ingestion doses is often taken to be approximately 20 g/a [SSK 93] [STU 94] [IAE 92] [EUR 98] [SSK 98]. Since the dust generating renovation work will last far less than a full year, the ingested quantity is assumed to be 1 g/a in this scenario. As in the inhalation scenario (section 4.2.2) no radioactive decay is assumed because of the short period during which the material could be ingested.

Table 4–3: Building reuse, parameters for the ingestion scenario

Parameter description	Symbol	Value	Units
Ingested quantity	M	1	g/a
Factor to account for the average contamination on very large surfaces	f_d	0.33	--
Concentration factor for the ingested dust fraction	f_k	3	--
Removal depth	δ	0.3	cm
Density of surface material	ρ	2.3	g/cm³

4.2.4 Skin dose

During the renovation work contaminated dust can accumulate on the skin causing a β-skin dose. To estimate the doses from this pathway the dust loading factor L_{dust} = 0.05 g/cm² for the exposed skin is assumed, which is chosen such that the dust is visible but would not bother a worker. The activity in the dust is calculated from the surface activity by assuming that the surface is removed to a depth of 0.3 cm and that this material has a density of 2.3 g/cm³. Like in the incorporation scenarios, a concentration factor f_k = 3 accounts for the affinity of the activity for the fine fraction of the dust. The exposure time is assumed to be longer than the time required for the abrasive treatment of the surface (see section 4.2.2) since it is assumed that the dust is not washed off immediately. As in the inhalation scenario (section 4.2.2) no radioactive decay is assumed because of the short duration of the scenario.

In order to calculate the skin dose, a contaminated skin surface area must also be chosen. It is assumed that both hands and both forearms are contaminated which leads to a skin area of about 0.17 m², since a hand has approximately 400 cm² (front plus back) and the skin area of one forearm is approximately the same (see ICRP 23). In addition, the total skin area of a person is needed, which is taken to be 1.7 m² (reference man in ICRP 26). Choosing this large contaminated area will guarantee that the skin dose is less than the skin dose limit of 50 mSv/a on any square centimeter as required by the Euratom Directive, when the effective individual dose is less than 10 µSv/a.

The following equation is used to calculate the skin dose,

$$H_{skin} = D_{skin} \cdot w_{skin} \cdot k_f \cdot t_e \cdot L_{dust} \cdot f_d \cdot f_k / \delta / \rho \qquad \text{eq. 4-5}$$

where

H_{skin} [µSv/a] effective individual dose from skin contaminated with β-emitters,
D_{skin} [µSv/h)/(Bq/cm²)] skin dose coefficient for 7 mg/cm² [KOC 87],
w_{skin} [-] skin weighting factor according to ICRP 60,
k_f [-] fraction of skin which is contaminated,
t_e [h/a] time during which the skin is contaminated,
L_{dust} [g/cm²] effective dust loading of contaminated skin,
f_d [-] factor to account for the average contamination in the building,
f_k [-] concentration factor for the fine dust fraction,
δ [cm] depth of surface layer which is removed during surface treatment and
ρ [g/cm³] density of surface layer.

Table 4–4: Building reuse, parameters for the skin dose scenario

Parameter description	Symbol	Value	Units
Skin weighting factor	w_{skin}	0.01	--
Fraction of skin which is contaminated	k_f	0.1	--
Time during which the skin is contaminated	t_e	150	h/a
Dust loading on the skin	L_{dust}	0.05	g/cm²
Factor to account for the average contamination on very large surfaces	f_d	0.33	--
Concentration factor for the fine dust fraction	f_k	3	--
Removal depth	δ	0.3	cm
Density of surface material	ρ	2.3	g/cm³

4.2.5 Exposure to Radon

A building contaminated with radium isotopes will release radon isotopes into the air of the room. The radon progeny can then cause inhalation doses to the occupants. An estimation is made here to predict the Rn 222 concentration in a room where the walls are contaminated with Ra 226. The simplified model assumes that the Ra 226 is located near the surface and that 50% of the Rn 222 production is emanated into the room [GAD 95]. The ratio of surface area to volume of a room can be characterized by 6/L where L is the characteristic dimension of the room, for offices L can be taken as 3 m. With this information and an estimate of the air exchange in the room, the Rn 222 concentration can be estimated with the following equation,

$$C_{Rn\,222} = r \cdot f_d \cdot f_E \cdot k_f \cdot \frac{1}{E_{air}} \cdot 6 / L \qquad \text{eq. 4-6}$$

where

$C_{Rn\ 222}$ 50 [($Bq_{Rn\ 222}$/m³)/($Bq_{Ra\ 226}$/cm²)] Rn 222 concentration in the room per Ra 226 surface activity,
r 75 [($Bq_{Rn\ 222}$/m²/h)/($Bq_{Ra\ 226}$/cm²)] production rate of Rn 222,
f_d 0.33 [-] factor to account for the average contamination in the building,
f_E 0.5 [-] emanation fraction (50% into the wall and 50% into the room)
E_{air} 0.5 [h^{-1}] air exchange rate in the room and
L 3 [m] characteristic dimension of the room.

The production rate for Rn 222 is determined by the decay constants of the Ra 226 decay chain. This result shows that if the average surface contamination after clearance is about 0.33 of the clearance level (see other scenarios) then the Rn 222 concentration in a room which was cleared using a clearance level for Ra 226 of 1 Bq/cm² will be around 50 Bq/m³. This value is well below typically recommended values for intervention but can lead to doses in excess of 10 µSv/a, see for example ICRP 65 [IEA 94] [SSK 91] [SSK 94] [CEU 96]. Since the Rn 222 concentration depends strongly on the use of the room, the type of finish put on the walls, the naturally occurring Ra 226 in the building structure as well as the Rn 222 in the ground under the building and on other factors, it does not seem prudent to use a radon scenario to define the clearance levels for Ra 226. Nevertheless if the building is to be reused after clearance and a significant fraction of the activity in the radionuclide mixture is Ra 226 it may be advisable to make direct measurements of the radon in the

room air. If the concentration should be deemed too high it may be possible with simple means, like painting the walls, to reduce the concentration significantly.

4.3 Clearance of Building Rubble and Buildings for Demolition

Besides the possibility of using the standing structure, after clearance a building could also be torn down and the resulting building rubble either recycled or disposed of at a landfill. This, however, will not allow for easy measurement of the radionuclide contents. In addition, it should be the goal when clearing material from authorized facilities to keep the total cleared radioactivity to a minimum. By carrying out the measurements on the standing structure it is possible to measure directly where the activity is expected, i.e. on the surfaces of the structure, and thereby reach maximum effectiveness in localizing and removing the activity. Once the building has been demolished. however, it is no longer a trivial task to identify those parts of the rubble with an elevated activity level since it is mixed up with non-contaminated rubble. Furthermore, if the building is to be demolished before being cleared it would be necessary to do this under radiological supervision. The problem of containing the contaminated dust created during the demolition is not trivial. For these reasons the German Commission on Radiological Protection recommends that the building be cleared *before* being demolished. It is nevertheless conceivable that under certain circumstances clearance *after* demolition is preferable to clearing the standing structure for which mass specific clearance levels would be required.

The scenarios developed in the following sections account both for clearance of buildings for dismantling only (i.e. with no previous reuse), for which surface specific clearance criteria are required, and for clearance of rubble for recycling or disposal, for which mass specific criteria are required. Rubble arises not only during the demolition of the decommissioned buildings but also during the removal of the components and decontamination of the building or during refurbishing and renovation work. Rubble does not have a well defined surface and can therefore not be cleared using surface contamination criteria. For these reasons, in the following sections mass *and* surface specific clearance levels for building rubble will be developed.

The strategy is – like in the case of the radiological assessment of building reuse (section 4.2) – to develop a small set of simple scenarios which represent average exposure situations to the critical groups. Two basic options must be covered by the enveloping scenarios, disposal at a landfill and recycle. Four exposure pathways are considered for recycling and disposal of the building rubble: external exposure to gamma emissions, inhalation of dust containing radioactivity, ingestion of activity and β-skin dose. Nuclide specific doses are calculated for a unit activity of 1 Bq/g for rubble and 1 Bq/cm² for the clearance of the standing structure before demolition. Because of the similarity of the scenarios, clearance of buildings for demolition (with subsequent recycling or disposal of the rubble) and clearance of rubble for recycling or disposal are developed in parallel. The results of the radionuclide specific calculations are presented in section 5.

4.3.1 Source term and radiological considerations

In order to derive the clearance levels, a generic source term must be developed. The largest source of potentially contaminated building rubble comes from the decommissioning work on large nuclear power plants where up to 200,000 Mg of building material can be found in the controlled areas. Typically such material will consist mainly of concrete which can be processed or disposed of at a

rate of around 20 to 150 Mg/h [DEC 93] [JOH 94] [HAR 95] [ELE 92]. Nuclear power plants and modern nuclear fuel reprocessing facilities typically have surface to mass ratios of around 1 m²/Mg [JOH 94]. Smaller facilities have ratios around 4 m²/Mg, but they also have a much smaller total amount of building rubble.

In order to carry out the dose calculations it is assumed that the building to be cleared has 100,000 Mg of material with a surface to mass ratio of 1 m²/Mg and that the clearance and demolition will take one year to complete. This mass is also used for those scenarios which depend on a mass input rather than on surface specific activity. Thus it is ensured that the results of the dose calculations (and therefore the derived mass specific clearance levels) provided in section 5 are valid also for the case that an entire facility is cleared using mass specific levels. The dose calculations are performed with a normalized mass specific activity value of 1 Bq/g so that it is subsequently possible to specify the activity that will lead to 10 µSv/a.

If the averaging mass is kept to 1 Mg or less it will not be possible to clear larger batches of e.g. 1000 Mg/a at the clearance level. This means that the activity equivalent of 1000 Mg/a of rubble *at the clearance level* will in reality be reached for significantly *larger* quantities, typically on the order of 3000 to 10,000 Mg/a. If it is forbidden to mix highly contaminated surfaces or rubble (e.g. the biological shield) with the uncontaminated bulk of the structure and if the averaging quantities are kept at approx. 1 Mg then the mass specific activity averaged over the total quantity of building rubble (10^5 Mg) will be around one order of magnitude less than the clearance level. A reasonable assumption is that on the average 20% of the clearance level is reached. Additionally, if it is assumed that processing (or, likewise, disposal) of rubble takes places at a rate of 100 Mg/h at a specific site, it will take about 1000 h (approx. 50% of a working year) to process 10^5 Mg.

Therefore some mixing with rubble from other sources can be assumed. To account for both the fact that it is impossible to actually reach the clearance level over the entire amount of material (20%) and the fact that mixing with some uncontaminated material is necessary after clearance during processing or disposal to reach the capacity (50%), a dilution factor of $0.2 \cdot 0.5 = 0.1$ is introduced in the relevant scenarios.

Since in general the doses are not strongly dependent on the quantity of material but instead on the *total* activity cleared, it makes more sense to limit, if necessary, the total annually cleared activity and not the rubble quantity. In order to avoid clearing highly contaminated rubble, e.g. the biological shield (see discussion above), by mixing it with non-contaminated material, an additional limit should be implemented, especially when the authorities decide to authorize the clearance of the building rubble after demolition, in which case the clearance measurements would be carried out on the rubble and not on the standing structure. The derived surface clearance levels include the activity in the structure under the measurement area as well as the activity directly on the surface (see figure 3–2) and here no additional annual activity limits are necessary in order to meet the 10 µSv/a criterion.

Typically, the largest doses can be expected to occur when the largest amounts of material are cleared. In the case of surface clearance levels smaller facilities with surface to mass ratios around 4 m²/Mg will lead to mass specific activity levels in the rubble which can be up to 4 times higher than for buildings with a ratio of 1 m²/Mg. Since such facilities typically contain significantly less building material, say 25,000 Mg, the amount of activity which can be cleared will be approximately equivalent to the larger facilities and therefore the expected doses should remain in the same trivial range.

During the removal of components and the decontamination of the building and to a lesser degree during operation of a facility, building rubble which must be cleared using mass specific levels arises. This material will generally be cleared before the building rubble from the demolition is removed from the site. Therefore it can be assumed that a person will not be exposed to both sources at the same time. Furthermore, since building rubble is not especially valuable it is not expected that it will be transported over great distances, unlike scrap metal. This means that it is extremely unlikely that a person will be exposed to the building rubble from two different sites in the same year. Should a large facility be cleared, say 100,000 Mg/a and the processing or disposal of the resulting material be carried out at a rate of 50 Mg/h at a single workplace, 2000 h/a would be required to process the rubble, making it nearly impossible for one person to be exposed to the rubble from two large facilities.

4.3.2 External γ-dose

For external exposure many different scenarios are possible. For example: the transport of the rubble, a landfill worker, living in a house made from recycled rubble, highway construction using recycled rubble, exposure during processing, exposure to stock piles of building rubble and construction of concrete building elements in a factory (for example pre-stressed I-beams). Such scenarios have been proposed and calculated in other studies [JOH 94] [HAR 95] and are therefore not recalculated here but instead one scenario is chosen to represent the average dose to the critical group.

The most restrictive *geometry* is living in a building since the irradiation will come from all sides. This is, however, not the most restrictive *scenario* since it is not possible to build a building out of 100% recycled concrete. For structural reasons, typically only around 10% of the material in a building can come from recycled concrete [JOH 94]. Therefore, in the context of recycling, the most restrictive geometry is the semi-infinite volume made of 100% recycled concrete. This represents the exposure to the landfill worker or the construction worker building a road. The exposure time is chosen to be a full working year (i.e. 1800 h/a), since it could take this long to process 100,000 Mg of material. The decay during a 1 year period of the scenario is taken into account. No decay time is, however, assumed between clearance of the rubble or of the buildings for demolition and the start of the scenario because the material might be used or disposed off immediately.

The mass specific calculations are carried out also for 100,000 Mg assuming a dilution of 1:10 (cf. section 4.3.1) and the same exposure time. The semi-infinite volume source has the advantage that the dose rate is not dependent on the density.

The mass specific activity can be calculated from the surface specific activity using the surface to mass ratio 1 m²/Mg typical for nuclear power plants. Since not all the surface will be contaminated and those which are will be decontaminated to some extent, the surface activity averaged over the entire structure will typically be significantly below the clearance level. The actual cleared activity will strongly depend on the averaging area selected for the clearance measurements. In this scenario it is assumed that the average surface activity over the total structure is equivalent to 1/3 of the clearance level. Under these assumptions a surface specific clearance level of 3 Bq/cm² results in an average mass specific activity of 0,01 Bq/g in the 100,000 Mg of rubble, with which the scenario is calculated (see table 4–5). The equation used to calculate the external irradiation dose is,

$$H_{ext} = D_{ext} \cdot C_f \cdot t_e \cdot f_{dec} \qquad \text{eq. 4-7}$$

where

H_{ext} [(μSv/a)/(Bq/g)] or [(μSv/a)/(Bq/cm²)] individual dose from external irradiation,
D_{ext} [(μSv/h)/(Bq/g)] average dose rate during the exposure year [CHE 91],
C_f factor defining the actual activity in the rubble
[(Bq/g)/(Bq/cm²)] for the case of buildings, or
[-] for rubble,
t_e [h/a] exposure time, and
f_{dec} [-] factor to account for the decay during 1 year of the scenario with

$$f_{dec} = \frac{1 - e^{-\lambda \cdot 1a}}{\lambda \cdot 1a}$$

The external γ-dose received by the worker standing on a semi-infinite volume is calculated using the tabulated energy dependent dose rate from [CHE 91] for a photon flux at 1 m above semi-infinite volume of soil. The dose rate was calculated by [CHE 91] using a Monte-Carlo simulation with ICRP 51 rotational orientation conversion factors for the fluence to dose and is shown in figure 4–1. The nuclide specific dose rates are listed in the appendix and were calculated using a 3rd order spline fit interpolation [PRE 87] to calculate the dose from each photon and then summing over all the emitted photons, multiplied by their emission probability. The parameter C_f is the conversion factor to calculate mass specific ($C_f = 0.01$) values or surface specific values,

$$C_f = R \cdot f_d \qquad \text{eq. 4-8}$$

where

C_f [(Bq/g)/(Bq/cm²)] conversion factor, mass specific to surface specific activity,
R [cm²/g] surface to mass ratio and
f_d [-] factor to account for the average contamination in the building.

Table 4–5: Buildings for demolition and building rubble: parameters for the external exposure scenario

Parameter description	**Symbol**	**Rubble**	**Buildings**	**Units**
Exposure time	t_e	1800	1800	h/a
Dilution factor for mass specific calculations	C_f	0.1	n.a.	--
Conversion factor, surface specific to mass specific	C_f	n.a.	0.0033	(Bq/g)/(Bq/cm²)
Factor to account for the average contamination on very large surfaces	f_d	n.a.	0.33	--
Surface to mass ratio	R	n.a.	0.01	cm²/g
Column “rubble”: parameter values applying to clearance of rubble; Column “buildings”: the same for buildings; n.a.: not applicable to this clearance option				

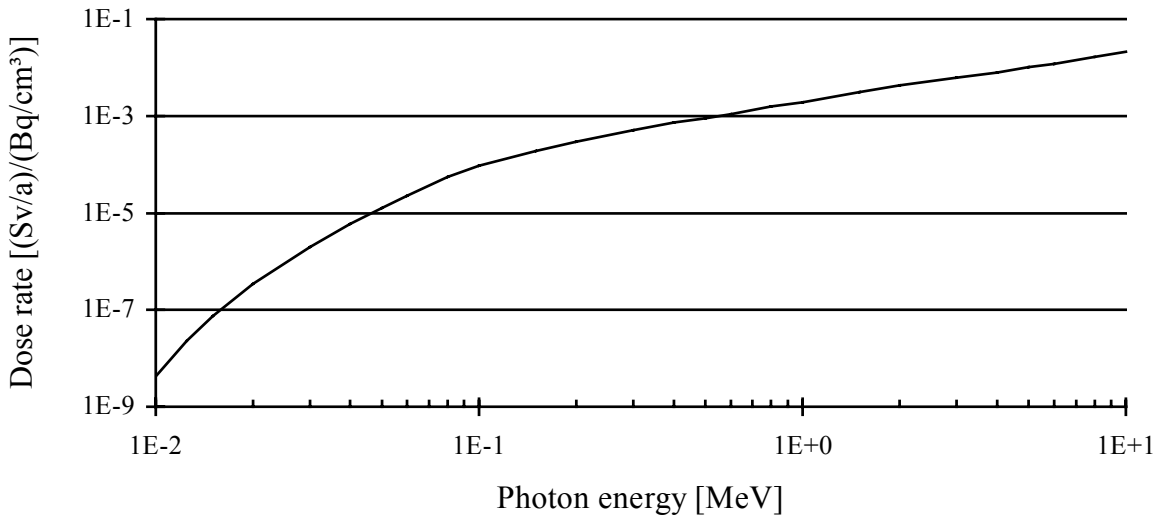


Figure 4–1	Dose rate as a function of the photon energy for a point 1 m over a semi-infinite volume of soil [CHE 91]	

4.3.3 Inhalation dose

An inhalation dose can only occur when the activity in the building rubble is re-suspended into the air and someone breathes this dust loaded air. Such a re-suspension will occur when the rubble is broken up and processed or dumped at a landfill. For projects like road construction, where water is used to help compact the rubble, or when the recycled rubble is used as an additive in new concrete it can be expected that the amount of dust created is significantly smaller compared to processing the concrete which takes place in dry conditions. For the inhalation pathway the critical scenario is the operation of the processing plant for rubble where the rubble is processed by crushing and separation of the fractions according to grain size [JOH 94] [HAR 95]. During each of these steps, crushing, sieving and sorting, large amounts of air born dust are created.

The dust concentration for the critical inhalation scenario, processing the rubble, is based on measurements taken near a processing machine during operation. The dust fraction that does not come from crushing the rubble can be ignored since background dust concentrations are seldom above 0.1 mg/m³ for prolonged periods, while the dust created by the crushing machine ranges from around 1 mg/m³ to peak concentrations of several 10 mg/m³. An annual average dust concentration of 5 mg/m³ is chosen for the scenario which is significantly higher than typical dust concentrations at landfills (see [DEC 97] [ASS 95]) The fraction of the processed rubble which comes from the decommissioning project can be estimated by applying typical processing rates, which lie around 50

to 150 Mg/h, and the amount of rubble to be recycled, which has been assumed to be 10^5 Mg/a. Under these conditions a person could be exposed for a full working year to the cleared rubble.

The decay during a 1 year period of the scenario is taken into account. No decay time is assumed, however, between clearance of the building for demolition or the rubble and the begin of the scenario, because processing could in both cases start immediately after clearance.

The mass specific activity can be calculated from the surface specific activity using the same assumptions as for the external exposure scenario (see eq. 4-8). The breathing rate is assumed to be 1.5 m³/h which is based on light exertion (see ICRP 23). The mass specific concentration of the dust will be higher than for the rubble itself, since the activity preferably binds to the small cement particles causing the activity to concentrate in the fine dust fraction [COR 94] [BAK 90]. A factor of 3 is used to account for the affinity of the activity to the finer cement particles (see [DEC 93] and [JOH 94]). The inhalation dose coefficients from table C in the Euratom Directive for an AMAD of 5 µm are used and no correction is made for the particle size distribution. The following equation is used to calculate the inhalation doses,

$$H_{inh} = D_{inh} \cdot t_e \cdot B_r \cdot f_k \cdot C_{dust} \cdot C_f \cdot f_{dec} \qquad \text{eq. 4-9}$$

where

H_{inh} [(µSv/a)/(Bq/g)] or [(µSv/a)/(Bq/cm²)] individual dose from inhalation during recycling of building rubble,
D_{inh} [µSv/Bq] dose coefficient (table C Euratom Directive, 5 µm),
t_e [h/a] exposure time,
B_r [m³/h] breathing rate,
f_k [-] concentration factor for the activity in the inhalable dust fraction,
C_{dust} [g/m³] effective dust concentration during recycling and
C_f factor defining the actual activity in the rubble
[(Bq/g)/(Bq/cm²)] for the case of buildings, or
[-] for rubble,
f_{dec} [-] factor to account for the decay during 1 year of the scenario (cf. section 4.3.2).

Table 4–6: Buildings for demolition and building rubble: parameters for the external exposure scenario

Parameter description	Symbol	Rubble	Buildings	Units
Exposure time	t_e	1800	1800	h/a
Breathing rate	B_r	1.5	1.5	m³/h
Effective dust concentration during recycling	C_{dust}	5	5	mg/m³
Concentration factor for inhalable dust fraction	f_k	3	3	--
Dilution factor for mass specific calculations	C_f	0.1	n.a.	--
Conversion factor, surface to mass specific (eq. 4-8)	C_f	n.a.	0.0033	(Bq/g)/(Bq/cm²)
Column “rubble”: parameter values applying to clearance of rubble; Column “buildings”: the same for buildings; n.a.: not applicable to this clearance option				

4.3.4 Ingestion dose

4.3.4.1 Direct ingesiton of material

Normally building rubble will not be ingested, although dust which builds up on the skin can be inadvertently ingested, for example while smoking a cigarette or eating during a break. A scenario to describe this exposure might assume that the worker consumes 20 g/a of contaminated material [SSK 98] [IAE 92] [EUR 98]. Since the rubble could also be used for recultivation or ground cover it is conceivable that small children could play on the ground and swallow some of the contaminated rubble. Therefore the enveloping scenario for the inadvertent ingestion of building rubble assumes that a small child (age class 1 to 2 years old) consumes 100 g/a of contaminated building rubble [STU 94] [SSK 93].

The decay during a 1 year period of the scenario is taken into account. In addition, a decay time between clearance of the rubble or of the building for demolition and the start of the ingestion is assumed. This time is set to 1 year for the case that the material is already in the form of rubble and 2 years for clearance of buildings because in this case additional time has to be allowed for building demolition.

The ingestion dose is calculated as follows,

$$H_{ing} = D_{ing} \cdot M \cdot C_f \cdot e^{-\lambda \cdot t_{init}} \cdot f_{dec} \qquad \text{eq. 4-10}$$

where

H_{ing} [(µSv/a)/(Bq/g)] or [(µSv/a)/(Bq/cm²)] individual dose from inadvertent ingestion of building rubble,
D_{ing} [µSv/Bq] dose coefficient (table A Euratom Directive for ages 1 to 2 years old),
M [g/a] ingested quantity,
C_f factor defining the actual activity in the rubble
[(Bq/g)/(Bq/cm²)] for the case of buildings, or
[-] for rubble,
t_{init} [a] initial time between clearance and start of scenario
f_{dec} [-] factor to account for the decay during 1 year of the scenario (cf. section 4.3.2).

The parameter values are given in the table 4–7.

4.3.4.2 Ingestion of contaminated water (groundwater pathway)

If the building rubble is disposed of in a landfill or used for backfilling pits etc., it is possible that the activity be washed out of the rubble and transported via the seep water into a water supply and could in this way lead to an ingestion dose. Two different scenarios for the use of contaminated water are described in [DEC 93] (also see [POS 95] [DEC 97] [ASS 95] [IAE 87] [ELE 92] [NEA 89]): The infiltrated seepage leaks through the landfill into the ground water where it is transported to a nearby well, or the contaminated runoff from the landfill plus the collected untreated seepage run into a stream which is used for irrigation. The well scenario is the more restrictive of the two scenarios and is therefore used here to estimate the radiological impact via migration into the biosphere of the radionuclides from rubble used as fill material or dumped at a landfill site. The radioactive decay is taken into account in C_w (cf. eq. 4-12).

To simplify the dose calculations, the migration scenario assumes that the well water is only used for drinking purposes. A difference is made between children (ages 1 to 2 years) and adults. The equation describing this scenario is,

$$H_{ing} = D_{ing} \cdot Q_w \cdot C_w \qquad \text{eq. 4-11}$$

where

H_{ing} [(µSv/a)/(Bq/g)] or [(µSv/a)/(Bq/cm²)] individual dose from ingestion of contaminated drinking water,
D_{ing} [µSv/Bq] dose coefficient (table A Euratom Directive),
Q_w [l/a] annual amount of drinking water consumed and
C_w [(Bq/l)/(Bq/cm²)] or [(Bq/l)/(Bq/g)] radionuclide concentration in well water.

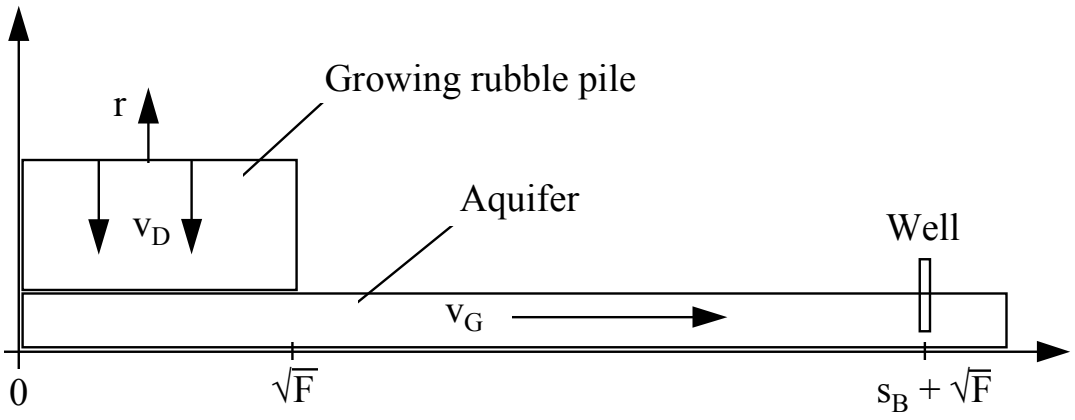


Figure 4–2 Schematic diagram of washout model
(See table 4–7 for parameter description, $v_G = k_f \cdot dl/dh$)

To calculate the radionuclide concentration in the well water C_w, a solution to the transport equation in one dimension for a growing rubble pile is used (see appendix). Furthermore it is assumed that the radionuclide concentration in the fluid phase is in equilibrium with the radionuclides bound to the solid phase (K_d-concept). The radionuclide specific equilibrium constants K_d are the values given for sandy soil in [SHE 90] and are listed for ease of reference in the appendix. The radionuclides are washed out of the rubble with the seep water into a water saturated unconfined aquifer. A well is located 500 m from the rubble pile (see figure 4–2) from which drinking water is drawn. The following equation is used to calculate the concentration in the well water,

$$C_w = \frac{A}{h\,F\,\lambda\,(n_G + \rho_G\,K_d)}\,\mathrm{e}^{-\lambda t_o}\left(\mathrm{e}^{\lambda\frac{\sqrt{F}\,|v_{ND}|}{v_{NG}\,(r+|v_{ND}|)}} - 1\right) \qquad \text{eq. 4-12}$$

where

A [(Bq/a)/(Bq/g)] or [(Bq/a)/(Bq/cm²)] total activity,

h [m] aquifer depth,
F [m²] area occupied by the rubble pile,
λ [1/a] radioactive decay constant,
n_G [-] fraction of the aquifer volume through which the fluid flows,
ρ_G [g/cm³] density of the aquifer (without water),
K_d [cm³/g] radionuclide specific distribution constant,
v_{ND} [m/a] migration velocity in the rubble pile,
v_{NG} [m/a] migration velocity in the aquifer,
r [m/a] growth rate of the rubble pile,
$t_o = x_o / v_{NG}$ [a] time at which the concentration is at a maximum and
$x_o = \sqrt{F} + s_B$ [m] position of the well (see figure 4–2).

The radionuclide migration velocities in the rubble pile v_{ND} and in the aquifer v_{NG} are calculated as,

$$v_{ND} = \frac{v_D}{n_D + \rho_D K_d} \qquad \text{eq. 4-13}$$

and

$$v_{NG} = \frac{k_f \cdot dh / dl}{n_G + \rho_G K_d} \qquad \text{eq. 4-14}$$

The meaning of the parameters and their values are given in table 4–7. The normalized total activity A is calculated by multiplying the quantity of rubble, 10^{11} g with the conversion / dilution factor C_f in table 4–7.

4.3.4.3 Ingestion of vegetables grown in soil mixed with rubble

A recultivated site, where the building rubble was used as fill material, or the utilization of a landfill where the rubble was disposed of after closure, could lead to a contamination of food products grown on this site. It is, however, very unlikely that the crops would incorporate a significant amount of activity from the rubble, since plants cannot be grown directly in rubble. Therefore, a mixture of soil and rubble has to be assumed to account for exposure situations in which activity is transferred from the rubble to plants. Such a scenario is possible when e.g. an old landfill is covered with soil which is subsequently ploughed so that soil and rubble are mixed. However, it can only be assumed that a small fraction of the soil can be replaced by rubble for which 3% is a reasonable assumption. To account for the fact that such a scenario can only occur a significant time after closure of the landfill, a period of 10 a is assumed before vegetables are grown. The activity fraction that is transferred from the soil to the plants is described by transfer factors.

The following equation is used for the dose calculation,

$$H_{ing} = D_{ing} \cdot C_f \cdot f_{soil} \cdot Q_{veg} \cdot T_{sp} \cdot e^{-\lambda t_{veg}} \qquad \text{eq. 4-15}$$

where
H_{ing} [(µSv/a)/(Bq/g)] or [(µSv/a)/(Bq/cm²)] individual dose from ingestion of vegetables grown in soil contaminated by recycled rubble,
D_{ing} [µSv/Bq] dose coefficient (max. value from table A Euratom Directive for all ages),
f_{soil} [-] fraction of soil that consists of contaminated rubble,

T_{sp} [-] transfer factor for activity transfer from soil to plant,
C_f factor defining the actual activity in the rubble
[(Bq/g)/(Bq/cm²)] for the case of buildings, or
[-] for rubble,
t_{veg} [a] time between clearance and start of cultivation of vegetables, and
Q_{veg} [kg/a] annual amount of vegetables consumed.

Table 4–7: Buildings for demolition and building rubble: parameters for the ingestion scenarios

Parameter description	**Symbol**	**Rubble**	**Buildings**	**Units**
Ingested quantity of rubble (child, age 1 to 2 years)	M	100	100	g/a
Time betw. clearance and start of ingestion scenario	t_{init}	1	2	a
Annual drinking water, adult	Q_w	500	500	l/a
Annual drinking water, child	Q_w	200	200	l/a
Ingested quantity of vegetables	Q_{veg}	100	100	kg/a
Dilution factor for mass specific calculations	C_f	0.1	n.a.	--
Conversion factor, surface to mass specific (eq. 4-8)	C_f	n.a.	0.0033	(Bq/g)/(Bq/cm²)
Reference amount of activity per Bq/g or Bq/cm²	A	20		GBq/a
Landfill area	F	10^4		m²
Depth of aquifer	h	5		m
Growth rate of landfill	r	6,25		m/a
Distance between landfill and well	s_B	500		m
Hydraulic conductivity	k_f	0,003		m/s
Hydraulic gradient	dh/dl	0,02		--
Usable pore fraction in aquifer	n_G	0,2		--
Density of aquifer	ρ_G	1,86		g/cm³ or Mg/m³
Seep water velocity (Darcy)	v_D	0,2		m/a
Usable pore fraction in landfill	n_D	0,2		--
Mass of rubble	-	10^5		Mg
Fraction of soil made up of recycled rubble	f_{soil}	0,03		--
Density of rubble	ρ_D	1,6		g/cm³ or Mg/m³
Time before vegetables can be grown	t_{veg}	10		a
Column “rubble”: parameter values applying to clearance of rubble; Column “buildings”: the same for buildings; n.a.: not applicable to this clearance option				

4.3.5 Skin dose

During the time the rubble is being processed (1800 h/a) dust can accumulate on the skin and cause a dose due to skin contamination. This scenario is modeled using the same parameters as in the renovation scenario (see section 4.2.4) with the exception of the exposure time which is set to a full working year. The equation is,

$$H_{skin} = D_{skin} \cdot w_{skin} \cdot k_f \cdot t_e \cdot f_k \cdot L_{dust} \cdot C_f \cdot f_{dec} \qquad \text{eq. 4-16}$$

where

H_{skin} [(µSv/a)/(Bq/g)] or [(µSv/a)/(Bq/cm²)] effective individual dose from skin contaminated with β-emitters,

D_{skin} [µSv/h)/(Bq/cm²)] skin dose coefficient for 7 mg/cm² [KOC 87],

w_{skin} [-] skin weighting factor according to ICRP 60,

k_f [-] fraction of skin which is contaminated,

t_e [h/a] time during which the skin is contaminated,

f_k [-] concentration factor for the fine dust fraction,

L_{dust} [g/cm²] effective dust loading of contaminated skin,

C_f factor defining the actual activity in the rubble
[(Bq/g)/(Bq/cm²)] for the case of buildings, or
[-] for rubble,

f_{dec} [-] factor to account for the decay during 1 year of the scenario (cf. section 4.3.2).

The conversion factor C_f is calculated the same way as in the other scenarios (see eq. 4-8).

Table 4–8: Buildings for demolition and building rubble: parameters for the skin dose scenario

Parameter description	Symbol	Rubble	Buildings	Units
Skin weighting factor	w_{skin}	0.01	0.01	--
Fraction of skin which is contaminated	k_f	0.1	0.1	--
Time during which the skin is contaminated	t_e	1800	1800	h/a
Concentration factor for the fine dust fraction	f_k	3	3	--
Dust loading on the skin	L_{dust}	0.05	0.05	g/cm²
Dilution factor for mass specific calculations	C_f	0.1	n.a.	--
Conversion factor, surface specific to mass specific	C_f		0.0033	(Bq/g)/(Bq/cm²)
Factor to account for the average contamination on very large surfaces	f_d	n.a.	0.33	--
Surface to mass ratio	R	n.a.	0.01	cm²/g
Column "rubble": parameter values applying to clearance of rubble; Column "buildings": the same for buildings; n.a.: not applicable to this clearance option				

4.4 Considerations to the Collective Dose

Besides limiting the individual dose, Annex 1 of the BSS [CEU 96] requires an optimisation if a collective dose in excess of 1 manSv per year of practice can be expected. In the following the collective doses are estimated for building reuse as well as the collective dose from recycling and disposal of cleared rubble.

When assessing collective doses, it is important to start from actual exposure situations. It is therefore meaningless to calculate collective doses for each nuclide. Instead, collective doses must be assessed for the most important exposure situations which are:

- reuse of large buildings and
- recycling of building rubble from nuclear power plants,

because nuclear power plants provide the largest contribution to the total mass of building rubble from nuclear installations (cf. section 2). This in turn makes Co 60 and Cs 137 the most important nuclides as they dominate nuclide vectors in nuclear power plants, leading to external irradiation as the most important exposure pathway. As a consequence, the collective doses are assessed here based on generic assumptions.

4.4.1 Collective dose from the reuse of buildings

The collective dose from building reuse is based on the assumption that people will be working in the building full time. This is consistent with the scenarios used in section 4.2. For the purpose of calculating the collective dose it is conservatively assumed that 1 person per 10 m² of floor space is exposed and that in one year buildings with 10,000 m² floor space are cleared for reuse (if these buildings were demolished they would result in about 10^5 Mg of building rubble). This would lead to 1000 persons being exposed. Since the buildings were cleared such that the external exposure in one year will not exceed 10 µSv this would lead to a collective dose of 0.01 manSv/a (corresponding to the first year after clearance, see below).

Significant doses from ingestion, inhalation or skin contamination only occur during renovation work, which is not carried out every year. Furthermore, the number of persons involved in renovation work is significantly less than 1 per 10 m² of floor space. Therefore, the collective doses from these paths will be significantly lower than the collective dose from external exposure during occupancy.

These above rough estimates of the collective dose from reuse of buildings show that it will be significantly below the 1 manSv criterion of Annex 1 of the BSS and therefore a detailed optimisation is not necessary. However, as the use of the buildings continues over the years and as it must be assumed that further buildings are cleared for reuse each year, it must be assessed how this will influence the *overall* collective dose. It is clear that only the external irradiation can lead to a continuous exposure to people working in the building because nuclides that are relevant for inhalation and ingestion scenarios are fixed in the building structure until the following renovation. This makes Co 60 the most important nuclide (half life 5.27 years), which makes up most of the contamination in facilities involved with nuclear power generation. To account for a prolonged use of the buildings, e.g. for 30 years, it would now be a considerable over-estimation to simply use the product of 30 a · 0.01 manSv/a as the collective dose. Because of the radioactive decay, the integral for Co 60 would be equal to a continuous exposure of 7.6 a resulting in a collective dose estimate on the order of 0.1 manSv.

4.4.2 Collective dose from cleared building rubble

A preliminary review of the nuclear power plants in the EU and an estimate of the time at which they will be decommissioned shows that a maximum of around $3 \cdot 10^6$ Mg in a 5 year period can be expected in the future (cf. section 2.1). Assuming a density of 2 Mg/m³ and an averaged thickness of a layer of building rubble of about 0.5 m, a surface area of approx. 1 km² could be covered with this material annually. The number of people spending time on this surface could then be estimated from the average population density (assumed to be 200 persons/km²) and a factor 10 to account for higher population densities in cities where it is assumed that most of the recycled concrete will be used. This results in 2000 persons being exposed. It is very unlikely that people would spend more than their full working time (i.e. 1800 h/a) on the area covered with the material (for which no cover

has been assumed). So the results of the scenarios which lead to the clearance levels provided in section 4 can be directly used. Assuming that the individual dose is 10 µSv/a for each of these persons, then the collective dose can be estimated as 10 µSv/a · 200 persons/km²·10, resulting in 0.02 manSv/a which can again be corrected for the continuous exposure by Co 60 (the most relevant nuclide for the external irradiation) by multiplying with a factor of 7.6 (see section 4.4.1). By the same reasoning as described above for building reuse, it is impossible that inhalation and ingestion scenarios could become as important as the ones for external irradiation, because the number of fuel cycle installations is far smaller compared to nuclear power plants.

This demonstrates that in spite of the very pessimistic assumptions, the 1 manSv criterion of Annex 1 of the BSS is clearly met. Of course, most of the rubble will be used in foundations and road construction or be disposed of in landfills so that in reality the dose in the years following clearance will be considerably smaller.

5. DOSE CALCULATIONS AND CLEARANCE LEVELS

5.1 Results of the Dose Calculations

In this section the results of the dose calculations are presented:

- In table 5–1 the results of the normalized radionuclide specific dose calculations for the use of cleared buildings are shown. The dose calculations were carried out assuming that the contamination represents the entire activity on and in the wall, i.e. the surface activity is the sum of the fixed and removable activity and the activity which has migrated into the structure (see figure 3–2 and section 3.4). The scenarios are those given in section 4.2.
- In table 5–2 the doses caused by the recycling or disposal of building rubble which comes from a building which has been cleared with a unit contamination of 1 Bq/cm² before being demolished are presented. Here again like for the reuse the surface activity limit includes the total activity in and on the structure (see figure 3–2 and section 3.4). The scenarios are those given in section 4.3.
- Doses from clearing building rubble for recycling or disposal with a unit mass specific activity of 1 Bq/g are shown in table 5–3. These calculations assume that a mass of 100,000 Mg/a of rubble is cleared in which in the average the activity is below the clearance level. The scenarios are those given in section 4.3.

Table 5–1: Results of the dose calculations for the reuse scenarios for buildings

Nuclide	External exposure (µSv/a)/(Bq/cm²)	Inhalation (worker 5µm) (µSv/a)/(Bq/cm²)	Ingestion (worker) (µSv/a)/(Bq/cm²)	ß-skin dose (7 mg/cm²) (µSv/a)/(Bq/cm²)	Most restrictive scenario
H 3	0.0E+0	4.4E-5	6.0E-5	0.0E+0	ingestion
C 14	0.0E+0	6.2E-4	8.3E-4	3.6E-3	ß-skin
Na 22	2.3E+1	2.2E-3	4.6E-3	2.1E-2	external
S 35	0.0E+0	1.2E-3	1.1E-3	3.8E-3	ß-skin
Cl 36	0.0E+0	5.5E-3	1.3E-3	2.1E-2	ß-skin
K 40	1.8E+0	3.2E-3	8.9E-3	2.2E-2	external
Ca 45	2.0E-9	2.5E-3	1.1E-3	9.5E-3	ß-skin
Sc 46	7.6E+0	5.2E-3	2.2E-3	1.5E-2	external
Mn 53	0.0E+0	3.9E-5	4.3E-5	0.0E+0	ingestion
Mn 54	6.7E+0	1.3E-3	1.0E-3	0.0E+0	external
Fe 55	0.0E+0	9.9E-4	4.7E-4	0.0E+0	inhalation
Co 56	1.2E+1	5.3E-3	3.6E-3	1.2E-2	external
Co 57	8.7E-1	6.5E-4	3.0E-4	8.5E-4	external
Co 58	3.1E+0	1.8E-3	1.1E-3	3.4E-3	external
Co 60	2.8E+1	1.8E-2	4.9E-3	1.2E-2	external
Ni 59	0.0E+0	2.4E-4	9.0E-5	0.0E+0	inhalation
Ni 63	0.0E+0	5.6E-4	2.2E-4	0.0E+0	inhalation
Zn 65	4.3E+0	3.0E-3	5.6E-3	2.8E-4	external
As 73	2.5E-2	7.0E-4	3.7E-4	6.0E-3	external
Se 75	1.9E+0	1.8E-3	3.7E-3	1.0E-3	external
Sr 85	1.6E+0	6.9E-4	8.0E-4	1.6E-4	external
Sr 90	0.0E+0	8.5E-2	4.4E-2	4.5E-2	inhalation
Y 91	9.6E-3	6.6E-3	3.4E-3	2.5E-2	ß-skin
Zr 93	2.0E-2	3.2E-2	5.7E-4	0.0E+0	inhalation
Zr 95	5.6E+0	6.7E-3	2.6E-3	2.0E-2	external
Nb 93m	2.0E-2	9.3E-4	1.7E-4	0.0E+0	external
Nb 94	1.9E+1	2.7E-2	2.4E-3	1.7E-2	external
Mo 93	1.3E-1	2.4E-3	3.9E-3	0.0E+0	external
Tc 97	1.2E-1	1.7E-4	1.2E-4	5.5E-4	external
Tc 97m	3.5E-2	2.9E-3	9.5E-4	8.7E-3	external
Tc 99	6.0E-6	3.4E-3	1.1E-3	1.1E-2	ß-skin
Ru 106	1.8E+0	3.8E-2	1.0E-2	2.7E-2	external
Ag 108m	2.0E+1	2.0E-2	3.3E-3	2.6E-3	external
Ag 110m	2.1E+1	7.9E-3	4.0E-3	6.1E-3	external
Cd 109	2.5E-1	1.0E-2	2.9E-3	5.5E-3	external
Sn 113	1.4E+0	2.0E-3	1.1E-3	9.1E-3	external
Sb 124	5.2E+0	5.1E-3	3.6E-3	2.0E-2	external
Sb 125	4.8E+0	4.3E-3	1.9E-3	1.2E-2	external
Te 123m	7.1E-1	3.7E-3	2.0E-3	9.6E-3	external
Te 127m	8.0E-2	6.9E-3	3.6E-3	2.5E-2	external
I 125	1.3E-1	7.9E-3	2.2E-2	0.0E+0	external

Table 5–1: Results of the dose calculations for the reuse scenarios for buildings

Nuclide	External exposure (µSv/a)/(Bq/cm²)	Inhalation (worker 5µm) (µSv/a)/(Bq/cm²)	Ingestion (worker) (µSv/a)/(Bq/cm²)	ß-skin dose (7 mg/cm²) (µSv/a)/(Bq/cm²)	Most restrictive scenario
I 129	3.3E-1	5.5E-2	1.6E-1	2.3E-3	external
Cs 134	1.6E+1	1.0E-2	2.7E-2	1.5E-2	external
Cs 135	0.0E+0	1.1E-3	2.9E-3	5.5E-3	ß-skin
Cs 137	6.9E+0	7.2E-3	1.9E-2	2.0E-2	external
Ce 139	8.6E-1	1.5E-3	3.7E-4	3.6E-3	external
Ce 144	3.8E-1	3.1E-2	7.6E-3	3.8E-2	external
Pm 147	3.4E-5	3.8E-3	3.7E-4	6.6E-3	ß-skin
Sm 151	7.9E-5	2.8E-3	1.4E-4	6.4E-6	inhalation
Eu 152	1.3E+1	2.9E-2	2.0E-3	9.7E-3	external
Eu 154	1.5E+1	3.8E-2	2.9E-3	2.2E-2	external
Eu 155	6.7E-1	5.1E-3	4.6E-4	3.6E-3	external
Gd 153	8.3E-1	2.7E-3	3.9E-4	1.4E-3	external
Tb 160	3.5E+0	5.8E-3	2.3E-3	2.2E-2	external
Tm 170	2.7E-2	5.6E-3	1.9E-3	2.3E-2	external
Tm 171	6.9E-3	9.8E-4	1.6E-4	2.6E-3	external
Ta 182	6.1E+0	8.0E-3	2.2E-3	2.3E-2	external
W 181	2.0E-1	4.6E-5	1.1E-4	1.1E-3	external
W 185	8.6E-5	2.4E-4	6.3E-4	1.2E-2	ß-skin
Os 185	3.0E+0	1.5E-3	7.3E-4	3.9E-4	external
Ir 192	2.7E+0	5.3E-3	2.0E-3	2.6E-2	external
Tl 204	1.1E-2	6.7E-4	1.9E-3	2.1E-2	ß-skin
Pb 210	4.5E-2	3.3E+0	1.2E+0	2.3E-2	inhalation
Bi 207	1.9E+1	3.4E-3	1.9E-3	1.2E-2	external
Po 210	4.9E-5	2.4E+0	3.4E-1	0.0E+0	inhalation
Ra 226	2.0E+1	5.6E+0	1.6E+0	1.2E-1	external
Ra 228	1.6E+1	2.3E+1	9.6E-1	4.4E-2	inhalation
Th 228	1.4E+1	3.7E+1	2.0E-1	4.9E-2	inhalation
Th 229	4.0E+0	8.6E+1	8.6E-1	6.4E-2	inhalation
Th 230	8.7E-1	3.0E+1	3.6E-1	4.9E-3	inhalation
Th 232	2.8E+1	7.0E+1	1.4E+0	7.4E-2	inhalation
Pa 231	5.1E+0	7.5E+2	2.7E+0	5.5E-2	inhalation
U 232	1.5E+1	5.9E+1	6.2E-1	4.4E-2	inhalation
U 233	4.5E-2	8.3E+0	8.0E-2	6.0E-4	inhalation
U 234	1.3E-2	7.3E+0	7.0E-2	2.8E-5	inhalation
U 235	2.0E+0	7.7E+0	7.2E-2	1.2E-2	inhalation
U 236	1.2E-2	6.8E+0	6.6E-2	2.3E-5	inhalation
U 238	2.9E-1	6.1E+0	6.7E-2	2.9E-2	inhalation
Np 237	2.9E+0	1.6E+1	1.6E-1	2.1E-2	inhalation
Pu 236	5.1E-1	1.4E+1	1.2E-1	1.7E-3	inhalation
Pu 238	1.4E-2	3.2E+1	3.3E-1	5.3E-9	inhalation
Pu 239	6.0E-3	3.4E+1	3.6E-1	1.1E-9	inhalation

Table 5–1: Results of the dose calculations for the reuse scenarios for buildings

Nuclide	External exposure (µSv/a)/(Bq/cm²)	Inhalation (worker 5µm) (µSv/a)/(Bq/cm²)	Ingestion (worker) (µSv/a)/(Bq/cm²)	ß-skin dose (7 mg/cm²) (µSv/a)/(Bq/cm²)	Most restrictive scenario
Pu 240	1.3E-2	3.4E+1	3.6E-1	4.7E-10	inhalation
Pu 241	9.6E-3	8.9E-1	8.9E-3	8.1E-7	inhalation
Pu 242	1.2E-2	3.3E+1	3.4E-1	4.5E-10	inhalation
Pu 244	4.1E+0	3.2E+1	3.4E-1	3.8E-2	inhalation
Am 241	3.3E-1	2.9E+1	2.9E-1	2.7E-5	inhalation
Am 242m	2.8E-1	3.1E+1	3.0E-1	1.6E-2	inhalation
Am 243	2.7E+0	2.9E+1	2.9E-1	2.8E-2	inhalation
Cm 242	7.7E-3	4.0E+0	1.7E-2	2.7E-11	inhalation
Cm 243	1.5E+0	2.2E+1	2.2E-1	1.4E-2	inhalation
Cm 244	1.3E-2	1.8E+1	1.7E-1	9.7E-13	inhalation
Cm 245	8.8E-1	3.3E+1	3.4E-1	7.1E-4	inhalation
Cm 246	1.2E-2	2.9E+1	3.0E-1	4.2E-14	inhalation
Cm 247	4.2E+0	2.7E+1	2.7E-1	1.8E-2	inhalation
Cm 248	9.8E-3	1.0E+2	1.1E+0	2.9E-8	inhalation
Bk 249	6.8E-3	1.2E-1	1.4E-3	1.0E-3	inhalation
Cf 248	8.1E-3	6.6E+0	4.0E-2	5.9E-4	inhalation
Cf 249	4.1E+0	4.8E+1	5.0E-1	2.3E-3	inhalation
Cf 250	1.1E-2	2.4E+1	2.3E-1	5.8E-6	inhalation
Cf 251	1.4E+0	5.0E+1	5.2E-1	2.0E-2	inhalation
Cf 252	9.4E-3	1.4E+1	1.3E-1	5.9E-6	inhalation
Cf 254	8.4E-8	2.4E+1	5.7E-1	5.1E-1	inhalation
Es 254	7.0E+0	6.5E+0	4.0E-2	7.0E-3	external

Table 5–2: Results of the surface specific dose calculations for building demolition with recycling of the building rubble

Nuclide	Ext. expos. landfill	Inhalation worker. 5µ	Ingestion child	Contam. drinking water adult	Contam. drinking water child	Vegetable consumpt.	ß-skin dose (7 mg/cm²)	Most restrictive scenario
	(µSv/a)/(Bq/cm²)							
H 3	0.0E+0	5.3E-6	3.4E-5	2.3E-3	2.6E-3	0.0E+0	0.0E+0	water child
C 14	0.0E+0	7.8E-5	5.3E-4	1.6E-3	1.7E-3	0.0E+0	2.9E-10	water child
Na 22	2.9E+0	2.3E-4	2.5E-3	2.2E-7	4.1E-7	4.1E-3	1.5E-9	landfill
S 35	0.0E+0	4.8E-5	1.8E-6	0.0E+0	0.0E+0	1.4E-14	1.0E-10	ing. worker
Cl 36	0.0E+0	6.8E-4	2.1E-3	5.3E-2	1.4E-1	3.1E-1	1.7E-9	vegetable
K 40	2.3E-1	4.0E-4	1.4E-2	5.7E-3	1.5E-2	4.2E-1	1.8E-9	vegetable
Ca 45	1.4E-12	1.6E-4	3.8E-5	1.5E-11	4.3E-11	5.9E-10	4.0E-10	inhalation
Sc 46	9.5E-1	2.0E-4	2.0E-6	0.0E+0	0.0E+0	1.3E-15	3.9E-10	landfill
Mn 53	0.0E+0	4.8E-6	7.3E-5	8.3E-6	2.4E-5	4.4E-4	0.0E+0	vegetable
Mn 54	8.6E-1	1.1E-4	1.4E-4	8.4E-46	1.5E-45	1.9E-6	0.0E+0	landfill
Fe 55	0.0E+0	1.1E-4	4.2E-4	2.9E-63	8.3E-63	9.1E-6	0.0E+0	ing. child
Co 56	1.6E+0	2.0E-4	2.8E-6	6.0E-198	1.4E-197	6.2E-17	3.1E-10	landfill
Co 57	7.8E-2	5.2E-5	5.3E-5	1.0E-62	3.2E-62	2.7E-8	4.6E-11	landfill
Co 58	3.9E-1	6.1E-5	2.6E-7	4.2E-229	9.9E-229	1.2E-19	7.6E-11	landfill
Co 60	3.5E+0	2.1E-3	6.4E-3	7.0E-12	2.2E-11	1.4E-3	9.4E-10	landfill
Ni 59	0.0E+0	2.9E-5	1.1E-4	2.2E-6	4.7E-6	6.7E-5	0.0E+0	ing. child
Ni 63	0.0E+0	6.9E-5	2.7E-4	5.8E-9	1.3E-8	1.5E-4	0.0E+0	ing. child
Zn 65	5.4E-1	2.3E-4	4.2E-4	2.8E-216	4.5E-216	1.5E-6	1.5E-11	landfill
As 73	4.8E-4	2.6E-5	3.5E-7	2.0E-150	6.0E-150	7.8E-19	1.5E-10	landfill
Se 75	2.1E-1	9.5E-5	2.7E-5	0.0E+0	0.0E+0	4.9E-11	3.6E-11	landfill
Sr 85	1.9E-1	2.2E-5	1.2E-7	1.1E-63	2.5E-63	2.3E-19	3.4E-12	landfill
Sr 90	0.0E+0	1.0E-2	2.9E-2	1.2E-2	1.5E-2	2.9E-1	3.7E-9	vegetable
Y 91	0.0E+0	1.9E-4	2.5E-7	0.0E+0	0.0E+0	1.1E-22	4.7E-10	inhalation
Zr 93	3.1E-5	4.0E-3	5.6E-4	2.8E-5	1.6E-5	5.0E-5	0.0E+0	inhalation
Zr 95	6.8E-1	2.0E-4	2.3E-7	0.0E+0	0.0E+0	5.8E-22	3.9E-10	landfill
Nb 93m	3.1E-5	1.1E-4	2.7E-4	8.0E-14	2.4E-13	5.5E-5	0.0E+0	ing. child
Nb 94	2.3E+0	3.3E-3	3.2E-3	1.5E-4	3.3E-4	9.6E-4	1.4E-9	landfill
Mo 93	2.0E-4	2.9E-4	2.5E-3	4.4E-3	4.2E-3	3.8E-3	0.0E+0	water adult
Tc 97	2.2E-4	2.1E-5	1.6E-4	3.9E-3	1.1E-2	1.5E-2	4.6E-11	vegetable
Tc 97m	1.5E-4	1.2E-4	1.4E-6	6.4E-3	1.9E-2	3.5E-14	2.4E-10	water child
Tc 99	3.8E-7	4.3E-4	1.6E-3	3.6E-2	1.1E-1	1.4E-1	8.7E-10	vegetable
Ru 106	2.2E-1	3.4E-3	2.9E-3	1.3E-42	3.7E-42	4.7E-6	1.6E-9	landfill
Ag 108m	2.4E+0	2.5E-3	3.6E-3	1.1E-4	2.2E-4	2.1E-2	2.1E-10	landfill
Ag 110m	2.6E+0	6.2E-4	3.9E-4	8.5E-97	1.7E-96	1.2E-6	3.2E-10	landfill
Cd 109	2.5E-3	9.9E-4	8.4E-4	6.2E-48	1.2E-47	1.8E-4	3.5E-10	landfill
Sn 113	1.5E-1	1.0E-4	7.8E-6	9.7E-303	2.7E-302	2.0E-12	3.0E-10	landfill
Sb 124	0.0E+0	1.5E-4	3.7E-7	3.0E-192	7.6E-192	6.2E-21	3.9E-10	inhalation
Sb 125	5.4E-1	4.7E-4	1.4E-3	1.5E-15	3.4E-15	1.3E-4	8.9E-10	landfill
Te 123m	6.4E-2	1.9E-4	1.8E-5	1.2E-273	3.0E-273	1.3E-10	3.3E-10	landfill
Te 127m	3.0E-3	3.3E-4	2.4E-5	2.6E-300	8.0E-300	3.5E-11	7.9E-10	landfill

Table 5–2: Results of the surface specific dose calculations for building demolition with recycling of the building rubble

Nuclide	Ext. expos. landfill	Inhalation worker. 5µ	Ingestion child	Contam. drinking water adult	Contam. drinking water child	Vegetable consumpt.	ß-skin dose (7 mg/cm²)	Most restrictive scenario
	(µSv/a)/(Bq/cm²)							
I 125	0.0E+0	2.4E-4	1.3E-6	3.4E-6	5.2E-6	2.2E-20	0.0E+0	ing. worker
I 129	2.0E-3	6.8E-3	7.3E-2	1.3E+0	1.1E+0	4.4E-2	1.9E-10	water adult
Cs 134	2.0E+0	1.1E-3	2.3E-3	2.0E-98	6.8E-99	3.5E-4	1.0E-9	landfill
Cs 135	0.0E+0	1.3E-4	7.6E-4	1.0E-4	4.6E-5	1.1E-3	4.6E-10	vegetable
Cs 137	8.3E-1	8.9E-4	3.7E-3	1.5E-10	5.6E-11	5.1E-3	1.6E-9	landfill
Ce 139	7.3E-2	8.6E-5	6.3E-6	0.0E+0	0.0E+0	1.7E-12	1.4E-10	landfill
Ce 144	4.2E-2	2.6E-3	1.4E-3	0.0E+0	0.0E+0	4.8E-7	2.1E-9	landfill
Pm 147	3.0E-6	4.1E-4	3.2E-4	2.8E-5	8.2E-5	3.9E-6	4.8E-10	inhalation
Sm 151	2.0E-7	3.5E-4	2.1E-4	6.5E-8	1.7E-7	1.8E-5	5.3E-13	inhalation
Eu 152	1.6E+0	3.5E-3	2.1E-3	1.9E-109	4.0E-109	1.3E-4	7.8E-10	landfill
Eu 154	1.8E+0	4.5E-3	3.3E-3	8.4E-159	2.0E-158	1.6E-4	1.8E-9	landfill
Eu 155	3.8E-2	5.9E-4	5.1E-4	4.6E-276	1.3E-275	1.6E-5	2.8E-10	landfill
Gd 153	3.4E-2	2.1E-4	4.5E-5	0.0E+0	0.0E+0	1.5E-9	6.9E-11	landfill
Tb 160	4.4E-1	2.0E-4	9.0E-7	0.0E+0	0.0E+0	2.6E-19	5.1E-10	landfill
Tm 170	1.1E-3	3.0E-4	2.7E-5	0.0E+0	0.0E+0	7.3E-13	8.4E-10	landfill
Tm 171	1.7E-4	1.0E-4	1.0E-4	1.7E-97	4.8E-97	6.0E-7	1.8E-10	landfill
Ta 182	7.4E-1	4.0E-4	1.4E-5	0.0E+0	0.0E+0	1.3E-13	7.7E-10	landfill
W 181	5.7E-3	2.4E-6	9.7E-7	1.7E-26	4.2E-26	7.0E-14	3.7E-11	landfill
W 185	7.8E-6	8.6E-6	4.3E-7	3.4E-38	1.0E-37	3.0E-18	3.0E-10	ing. worker
Os 185	3.5E-1	6.5E-5	1.4E-6	4.1E-155	8.5E-155	3.4E-15	1.1E-11	landfill
Ir 192	3.2E-1	1.8E-4	7.8E-7	1.3E-217	3.3E-217	1.5E-18	6.0E-10	landfill
Tl 204	5.1E-4	7.6E-5	1.8E-3	1.4E-12	3.9E-12	4.1E-3	1.6E-9	vegetable
Pb 210	4.1E-4	4.1E-1	3.7E+0	1.8E-10	4.8E-10	6.9E+0	1.9E-9	vegetable
Bi 207	2.2E+0	4.2E-4	2.2E-3	2.4E-6	5.3E-6	1.2E-2	9.6E-10	landfill
Po 210	5.8E-6	1.4E-1	3.5E-2	4.5E-282	1.3E-281	9.4E-9	0.0E+0	inhalation
Ra 226	2.6E+0	6.9E-1	4.0E+0	3.3E-2	8.0E-2	1.1E+1	9.8E-9	vegetable
Ra 228	2.0E+0	2.6E+0	1.4E+0	1.2E-63	4.0E-63	1.5E+0	3.5E-9	inhalation
Th 228	1.9E+0	3.8E+0	1.5E-1	0.0E+0	0.0E+0	1.4E-3	3.4E-9	inhalation
Th 229	3.9E-1	1.1E+1	7.9E-1	1.3E-3	2.0E-3	1.2E-1	5.3E-9	inhalation
Th 230	1.1E-1	3.7E+0	2.7E-1	1.1E-3	1.3E-3	4.0E-2	4.1E-10	inhalation
Th 232	3.6E+0	8.7E+0	2.4E+0	4.8E-3	1.3E-2	3.6E-1	6.1E-9	inhalation
Pa 231	5.2E-1	9.4E+1	1.8E+0	4.7E-2	5.3E-2	1.6E-1	4.6E-9	inhalation
U 232	2.0E+0	7.3E+0	5.5E-1	7.7E-2	1.2E-1	7.6E-2	3.6E-9	inhalation
U 233	3.9E-3	1.0E+0	5.3E-2	2.3E-2	2.5E-2	7.9E-3	5.0E-11	inhalation
U 234	1.3E-4	9.1E-1	4.3E-2	1.9E-2	2.1E-2	6.4E-3	2.3E-12	inhalation
U 235	1.8E-1	9.6E-1	4.6E-2	2.0E-2	2.2E-2	6.9E-3	9.6E-10	inhalation
U 236	4.5E-5	8.4E-1	4.3E-2	1.9E-2	2.1E-2	6.4E-3	1.9E-12	inhalation
U 238	2.8E-2	7.6E-1	4.9E-2	1.9E-2	2.4E-2	7.4E-3	2.4E-9	inhalation
Np 237	2.9E-1	2.0E+0	7.3E-2	3.0E-1	2.4E-1	4.4E-2	1.7E-9	inhalation
Pu 236	6.8E-2	1.5E+0	3.9E-2	8.0E-143	8.1E-143	7.3E-5	1.3E-10	inhalation

Table 5–2: Results of the surface specific dose calculations for building demolition with recycling of the building rubble

Nuclide	Ext. expos. landfill	Inhalation worker. 5µ	Ingestion child	Contam. drinking water adult	Contam. drinking water child	Vegetable consumpt.	ß-skin dose (7 mg/cm²)	Most restrictive scenario
	(µSv/a)/(Bq/cm²)							
Pu 238	2.1E-5	4.0E+0	1.3E-1	2.1E-7	1.5E-7	1.5E-3	4.4E-16	inhalation
Pu 239	5.5E-5	4.3E+0	1.4E-1	6.1E-3	4.1E-3	1.7E-3	9.4E-17	inhalation
Pu 240	2.1E-5	4.3E+0	1.4E-1	5.5E-3	3.7E-3	1.7E-3	3.9E-17	inhalation
Pu 241	2.3E-4	1.1E-1	3.2E-3	2.0E-32	1.4E-32	2.7E-5	6.5E-14	inhalation
Pu 242	1.8E-5	4.1E+0	1.3E-1	6.1E-3	4.0E-3	1.6E-3	3.8E-17	inhalation
Pu 244	4.9E-1	4.0E+0	1.4E-1	6.1E-3	4.3E-3	1.7E-3	3.2E-9	inhalation
Am 241	7.8E-3	3.6E+0	1.2E-1	1.1E-6	7.9E-7	1.1E-3	2.2E-12	inhalation
Am 242m	1.6E-2	3.9E+0	1.3E-1	1.6E-12	1.2E-12	1.1E-3	1.3E-9	inhalation
Am 243	2.1E-1	3.6E+0	1.3E-1	9.6E-4	7.3E-4	1.1E-3	2.3E-9	inhalation
Cm 242	1.2E-5	2.5E-1	5.9E-4	0.0E+0	0.0E+0	4.6E-11	1.1E-18	inhalation
Cm 243	1.4E-1	2.6E+0	1.0E-1	6.7E-102	5.9E-102	7.7E-4	1.1E-9	inhalation
Cm 244	1.9E-5	2.2E+0	8.7E-2	8.0E-162	7.7E-162	5.9E-4	7.9E-20	inhalation
Cm 245	6.5E-2	4.1E+0	1.4E-1	3.9E-4	2.7E-4	1.2E-3	5.9E-11	inhalation
Cm 246	1.5E-5	3.6E+0	1.2E-1	1.8E-4	1.3E-4	1.1E-3	3.5E-21	inhalation
Cm 247	4.7E-1	3.3E+0	1.2E-1	6.6E-4	4.9E-4	1.0E-3	1.5E-9	inhalation
Cm 248	1.5E-5	1.3E+1	4.6E-1	2.6E-3	1.9E-3	4.2E-3	2.4E-15	inhalation
Bk 249	7.9E-4	1.0E-2	1.4E-4	0.0E+0	0.0E+0	3.3E-8	5.8E-11	inhalation
Cf 248	1.0E-5	5.7E-1	8.2E-3	0.0E+0	0.0E+0	2.5E-6	3.4E-11	inhalation
Cf 249	4.6E-1	6.0E+0	2.9E-1	6.8E-4	6.8E-4	2.5E-2	1.9E-10	inhalation
Cf 250	2.7E-5	2.9E+0	1.6E-1	3.8E-33	5.2E-33	9.6E-3	4.7E-13	inhalation
Cf 251	1.2E-1	6.1E+0	2.9E-1	3.4E-3	3.3E-3	2.6E-2	1.6E-9	inhalation
Cf 252	1.9E-5	1.5E+0	8.7E-2	1.6E-153	3.5E-153	1.1E-3	4.3E-13	inhalation
Cf 254	0.0E+0	7.1E-1	5.9E-5	0.0E+0	0.0E+0	1.5E-19	1.0E-8	inhalation
Es 254	8.7E-1	5.3E-1	5.6E-3	0.0E+0	0.0E+0	5.2E-7	3.8E-10	landfill

Table 5–3: Results of the mass specific dose calculations for recycling building rubble

Nuclide	Ext. expos. landfill	Inhalation worker. 5µ	Ingestion child	Contam. drinking water adult	Contam. drinking water child	Vegetable consumpt.	ß-skin dose (7 mg/cm²)	Most restrictive scenario
	(µSv/a)/(Bq/g)							
H 3	0.0E+0	1.7E-4	1.1E-3	1.4E-1	1.6E-1	1.0E-1	0.0E+0	water child
C 14	0.0E+0	2.3E-3	1.6E-2	9.5E-2	1.0E-1	9.6E-1	8.9E-3	vegetable
Na 22	9.8E+1	8.1E-3	1.0E-1	1.3E-5	2.5E-5	1.3E-1	5.2E-2	landfill
S 35	0.0E+0	4.5E-3	9.8E-4	0.0E+0	0.0E+0	2.8E-13	9.6E-3	ß-skin
Cl 36	0.0E+0	2.1E-2	6.3E-2	3.2E+0	8.7E+0	9.4E+0	5.2E-2	vegetable
K 40	7.0E+0	1.2E-2	4.2E-1	3.5E-1	9.4E-1	1.3E+1	5.5E-2	vegetable
Ca 45	8.5E-11	9.3E-3	5.4E-3	9.4E-10	2.6E-9	1.8E-8	2.4E-2	ß-skin
Sc 46	9.1E+1	1.9E-2	1.2E-3	0.0E+0	0.0E+0	3.9E-14	3.7E-2	landfill
Mn 53	0.0E+0	1.5E-4	2.2E-3	5.1E-4	1.5E-3	6.6E-3	0.0E+0	vegetable
Mn 54	3.8E+1	4.9E-3	9.5E-3	5.1E-44	8.9E-44	2.9E-5	0.0E+0	landfill
Fe 55	0.0E+0	3.7E-3	1.6E-2	1.7E-61	5.1E-61	1.7E-5	0.0E+0	ing. child
Co 56	1.6E+2	2.0E-2	2.0E-3	3.6E-196	8.7E-196	9.3E-16	3.1E-2	landfill
Co 57	3.7E+0	2.4E-3	4.1E-3	6.3E-61	1.9E-60	4.1E-7	2.1E-3	landfill
Co 58	4.4E+1	6.9E-3	3.1E-4	2.5E-227	6.0E-227	1.9E-18	8.6E-3	landfill
Co 60	1.1E+2	6.9E-2	2.2E-1	4.2E-10	1.3E-9	2.2E-2	3.1E-2	landfill
Ni 59	0.0E+0	8.9E-4	3.4E-3	1.3E-4	2.8E-4	1.0E-3	0.0E+0	ing. child
Ni 63	0.0E+0	2.1E-3	8.3E-3	3.5E-7	7.8E-7	2.3E-3	0.0E+0	ing. child
Zn 65	2.6E+1	1.1E-2	3.5E-2	1.7E-214	2.7E-214	1.5E-4	7.1E-4	landfill
As 73	4.8E-2	2.6E-3	2.5E-4	1.2E-148	3.6E-148	2.4E-17	1.5E-2	landfill
Se 75	1.5E+1	6.9E-3	6.6E-3	0.0E+0	0.0E+0	2.9E-9	2.6E-3	landfill
Sr 85	2.3E+1	2.6E-3	1.7E-4	6.7E-62	1.5E-61	5.3E-18	4.0E-4	landfill
Sr 90	0.0E+0	3.2E-1	9.0E-1	7.4E-1	8.8E-1	6.6E+0	1.1E-1	vegetable
Y 91	0.0E+0	2.5E-2	5.6E-4	0.0E+0	0.0E+0	1.1E-20	6.2E-2	ß-skin
Zr 93	9.4E-4	1.2E-1	1.7E-2	1.7E-3	9.6E-4	5.1E-5	0.0E+0	inhalation
Zr 95	8.5E+1	2.5E-2	4.1E-4	0.0E+0	0.0E+0	5.9E-22	4.9E-2	landfill
Nb 93m	9.5E-4	3.5E-3	8.4E-3	4.8E-12	1.5E-11	1.7E-3	0.0E+0	ing. child
Nb 94	7.1E+1	1.0E-1	9.7E-2	8.9E-3	2.0E-2	2.9E-2	4.3E-2	landfill
Mo 93	6.1E-3	8.9E-3	7.7E-2	2.7E-1	2.6E-1	2.3E-1	0.0E+0	water adult
Tc 97	6.8E-3	6.5E-4	4.9E-3	2.3E-1	6.8E-1	7.3E-1	1.4E-3	vegetable
Tc 97m	1.4E-2	1.1E-2	7.5E-4	3.9E-1	1.2E+0	1.8E-12	2.2E-2	water child
Tc 99	1.2E-5	1.3E-2	4.8E-2	2.2E+0	6.6E+0	7.2E+0	2.7E-2	vegetable
Ru 106	9.3E+0	1.4E-1	1.8E-1	8.0E-41	2.2E-40	1.4E-4	6.8E-2	landfill
Ag 108m	7.2E+1	7.7E-2	1.1E-1	6.9E-3	1.3E-2	6.3E-1	6.5E-3	landfill
Ag 110m	1.2E+2	3.0E-2	3.2E-2	5.2E-95	1.0E-94	3.6E-5	1.5E-2	landfill
Cd 109	9.6E-2	3.9E-2	4.3E-2	3.8E-46	7.2E-46	5.5E-3	1.4E-2	landfill
Sn 113	1.1E+1	7.7E-3	2.2E-3	5.9E-301	1.6E-300	3.0E-11	2.3E-2	landfill
Sb 124	0.0E+0	1.9E-2	6.5E-4	1.8E-190	4.6E-190	9.4E-20	4.9E-2	ß-skin
Sb 125	1.9E+1	1.6E-2	5.3E-2	8.8E-14	2.1E-13	1.9E-3	3.1E-2	landfill
Te 123m	4.7E+0	1.4E-2	4.5E-3	7.3E-272	1.8E-271	6.0E-12	2.4E-2	landfill
Te 127m	2.3E-1	2.6E-2	7.3E-3	1.6E-298	4.8E-298	1.6E-12	6.2E-2	landfill
I 125	0.0E+0	3.0E-2	2.3E-3	2.1E-4	3.1E-4	6.7E-19	0.0E+0	ing. worker

Table 5–3: Results of the mass specific dose calculations for recycling building rubble

Nuclide	Ext. expos. landfill	Inhalation worker. 5µ	Ingestion child	Contam. drinking water adult	Contam. drinking water child	Vegetable consumpt.	ß-skin dose (7 mg/cm²)	Most restrictive scenario
	(µSv/a)/(Bq/g)							
I 129	6.1E-2	2.1E-1	2.2E+0	8.1E+1	6.5E+1	1.3E+0	5.9E-3	water adult
Cs 134	7.0E+1	3.9E-2	9.8E-2	1.2E-96	4.1E-97	1.5E-3	3.7E-2	landfill
Cs 135	0.0E+0	4.0E-3	2.3E-2	6.0E-3	2.8E-3	4.8E-3	1.4E-2	ing. child
Cs 137	2.5E+1	2.7E-2	1.2E-1	9.3E-9	3.4E-9	2.2E-2	4.9E-2	landfill
Ce 139	4.8E+0	5.7E-3	1.2E-3	0.0E+0	0.0E+0	5.7E-12	8.9E-3	landfill
Ce 144	1.9E+0	1.2E-1	1.1E-1	0.0E+0	0.0E+0	1.6E-6	9.6E-2	landfill
Pm 147	1.0E-4	1.4E-2	1.3E-2	1.7E-3	5.0E-3	1.2E-4	1.7E-2	ß-skin
Sm 151	6.1E-6	1.1E-2	6.3E-3	3.9E-6	1.0E-5	3.6E-4	1.6E-5	inhalation
Eu 152	5.0E+1	1.1E-1	6.8E-2	1.1E-107	2.4E-107	3.9E-3	2.4E-2	landfill
Eu 154	5.5E+1	1.4E-1	1.1E-1	5.1E-157	1.2E-156	4.9E-3	5.5E-2	landfill
Eu 155	1.2E+0	1.9E-2	1.8E-2	2.8E-274	7.7E-274	5.0E-4	8.9E-3	landfill
Gd 153	1.7E+0	1.0E-2	3.9E-3	0.0E+0	0.0E+0	4.5E-8	3.4E-3	landfill
Tb 160	4.8E+1	2.2E-2	8.7E-4	0.0E+0	0.0E+0	8.0E-18	5.5E-2	landfill
Tm 170	7.7E-2	2.1E-2	5.9E-3	0.0E+0	0.0E+0	2.2E-11	5.9E-2	landfill
Tm 171	6.3E-3	3.7E-3	4.5E-3	1.0E-95	2.9E-95	1.8E-5	6.5E-3	ß-skin
Ta 182	5.6E+1	3.0E-2	4.0E-3	0.0E+0	0.0E+0	3.8E-12	5.9E-2	landfill
W 181	4.2E-1	1.7E-4	2.4E-4	1.0E-24	2.5E-24	2.1E-12	2.7E-3	landfill
W 185	8.1E-4	8.9E-4	3.5E-4	2.1E-36	6.3E-36	9.2E-17	3.1E-2	ß-skin
Os 185	3.0E+1	5.7E-3	6.3E-4	2.5E-153	5.1E-153	1.0E-13	9.9E-4	landfill
Ir 192	3.5E+1	2.0E-2	7.6E-4	8.1E-216	2.0E-215	4.6E-17	6.5E-2	landfill
Tl 204	1.7E-2	2.5E-3	6.5E-2	8.4E-11	2.4E-10	1.2E-1	5.2E-2	vegetable
Pb 210	1.2E-2	1.3E+1	1.1E+2	1.1E-8	2.9E-8	2.6E+1	5.9E-2	ing. child
Bi 207	6.8E+1	1.3E-2	6.9E-2	1.5E-4	3.2E-4	3.5E-1	2.9E-2	landfill
Po 210	3.8E-4	8.9E+0	6.5E+0	2.7E-280	8.0E-280	6.3E-9	0.0E+0	inhalation
Ra 226	7.8E+1	2.1E+1	1.2E+2	2.0E+0	4.9E+0	3.6E+1	3.0E-1	ing. child
Ra 228	6.5E+1	8.5E+1	4.8E+1	7.4E-62	2.4E-61	5.2E+0	1.1E-1	inhalation
Th 228	6.9E+1	1.4E+2	6.4E+0	0.0E+0	0.0E+0	4.3E-3	1.2E-1	inhalation
Th 229	1.2E+1	3.2E+2	2.4E+1	7.9E-2	1.2E-1	3.6E-1	1.6E-1	inhalation
Th 230	3.3E+0	1.1E+2	8.1E+0	6.9E-2	8.0E-2	1.2E-1	1.2E-2	inhalation
Th 232	1.1E+2	2.6E+2	7.2E+1	2.9E-1	7.6E-1	1.1E+0	1.8E-1	inhalation
Pa 231	1.6E+1	2.8E+3	5.4E+1	2.8E+0	3.2E+0	6.5E+1	1.4E-1	inhalation
U 232	6.2E+1	2.2E+2	1.7E+1	4.7E+0	7.4E+0	4.6E-1	1.1E-1	inhalation
U 233	1.2E-1	3.1E+1	1.6E+0	1.4E+0	1.5E+0	4.8E-2	1.5E-3	inhalation
U 234	3.8E-3	2.8E+1	1.3E+0	1.2E+0	1.2E+0	3.9E-2	7.1E-5	inhalation
U 235	5.6E+0	2.9E+1	1.4E+0	1.2E+0	1.3E+0	4.2E-2	2.9E-2	inhalation
U 236	1.4E-3	2.6E+1	1.3E+0	1.1E+0	1.3E+0	3.9E-2	5.9E-5	inhalation
U 238	8.4E-1	2.3E+1	1.5E+0	1.2E+0	1.4E+0	4.5E-2	7.4E-2	inhalation
Np 237	8.7E+0	6.1E+1	2.2E+0	1.8E+1	1.4E+1	1.3E-1	5.2E-2	inhalation
Pu 236	2.3E+0	5.3E+1	1.5E+0	4.8E-141	4.9E-141	2.8E-4	4.3E-3	inhalation
Pu 238	6.4E-4	1.2E+2	4.0E+0	1.3E-5	9.0E-6	5.5E-3	1.3E-8	inhalation
Pu 239	1.7E-3	1.3E+2	4.2E+0	3.7E-1	2.5E-1	6.3E-3	2.8E-9	inhalation

Table 5–3: Results of the mass specific dose calculations for recycling building rubble

Nuclide	Ext. expos. landfill	Inhalation worker. 5µ	Ingestion child	Contam. drinking water adult	Contam. drinking water child	Vegetable consumpt.	ß-skin dose (7 mg/cm²)	Most restrictive scenario
	(µSv/a)/(Bq/g)							
Pu 240	6.3E-4	1.3E+2	4.2E+0	3.3E-1	2.2E-1	6.3E-3	1.2E-9	inhalation
Pu 241	7.0E-3	3.4E+0	1.0E-1	1.2E-30	8.7E-31	1.0E-4	2.0E-6	inhalation
Pu 242	5.6E-4	1.3E+2	4.0E+0	3.7E-1	2.5E-1	6.0E-3	1.1E-9	inhalation
Pu 244	1.5E+1	1.2E+2	4.2E+0	3.7E-1	2.6E-1	6.3E-3	9.6E-2	inhalation
Am 241	2.4E-1	1.1E+2	3.7E+0	6.5E-5	4.8E-5	8.7E-3	6.8E-5	inhalation
Am 242m	4.8E-1	1.2E+2	3.9E+0	9.6E-11	7.1E-11	8.9E-3	4.0E-2	inhalation
Am 243	6.4E+0	1.1E+2	3.8E+0	5.8E-2	4.4E-2	9.1E-3	7.1E-2	inhalation
Cm 242	7.4E-4	1.5E+1	8.3E-2	0.0E+0	0.0E+0	2.3E-10	6.8E-11	inhalation
Cm 243	4.3E+0	8.1E+1	3.2E+0	4.0E-100	3.6E-100	3.9E-3	3.4E-2	inhalation
Cm 244	6.0E-4	6.9E+1	2.7E+0	4.9E-160	4.7E-160	3.0E-3	2.4E-12	inhalation
Cm 245	2.0E+0	1.3E+2	4.2E+0	2.3E-2	1.6E-2	6.3E-3	1.8E-3	inhalation
Cm 246	4.6E-4	1.1E+2	3.7E+0	1.1E-2	7.8E-3	5.5E-3	1.0E-13	inhalation
Cm 247	1.4E+1	1.0E+2	3.5E+0	4.0E-2	3.0E-2	5.2E-3	4.6E-2	inhalation
Cm 248	4.6E-4	3.8E+2	1.4E+1	1.6E-1	1.2E-1	2.1E-2	7.4E-8	inhalation
Bk 249	3.4E-2	4.5E-1	9.1E-3	0.0E+0	0.0E+0	9.9E-7	2.5E-3	inhalation
Cf 248	4.5E-4	2.5E+1	5.3E-1	0.0E+0	0.0E+0	7.7E-5	1.5E-3	inhalation
Cf 249	1.4E+1	1.8E+2	8.7E+0	4.1E-2	4.1E-2	7.7E-1	5.9E-3	inhalation
Cf 250	8.4E-4	8.9E+1	5.1E+0	2.3E-31	3.1E-31	2.9E-1	1.4E-5	inhalation
Cf 251	3.7E+0	1.9E+2	8.8E+0	2.0E-1	2.0E-1	7.9E-1	4.9E-2	inhalation
Cf 252	6.4E-4	5.3E+1	3.4E+0	9.4E-152	2.1E-151	3.2E-2	1.5E-5	inhalation
Cf 254	0.0E+0	8.9E+1	1.1E-1	0.0E+0	0.0E+0	4.6E-18	1.3E+0	inhalation
Es 254	4.0E+1	2.4E+1	4.2E-1	0.0E+0	0.0E+0	1.6E-5	1.8E-2	landfill

5.2 Derivation of Clearance Levels

Regarding the act of clearance, three main groups for buildings and building rubble, due to the further purpose of usage or handling, are considered:

- clearance for buildings for any purpose (reuse or demolition);
- clearance for buildings for demolition only;
- clearance for building rubble.

Corresponding to these clearance options, three sets of clearance levels for buildings and building rubble are developed from the results of table 5–1, table 5–2 and table 5–3. Those clearance levels are presented in table 5–4 and are set such that the most restrictive scenarios result in an individual dose of 10 μSv/a as required by the Euratom Directive [CEU 96] (also see [IAE 88]). The possible exposure paths are due to external γ-dose ("external"), inhalation, ingestion ("water child", "water adult", "vegetable", "landfill") and β-skin-dose (cf. section 4 for a detailed description of the scenarios).

It should be kept in mind that the radionuclides investigated here are those with half-lives longer than 60 days for which exemption values in the BSS exist, with the exception of the noble gases. Although this selection of nuclides will cover the vast majority of clearance situations, it may occur that in special cases additional radionuclides are relevant. For those nuclides, the dose calculations can be performed as outlined in section 4.

The following considerations apply to the clearance levels:

- The surface specific clearance levels for **reuse of buildings** are the most restrictive of the values from table 5–1 and table 5–2 since, after clearance, the buildings can be reused or demolished.
- The most restrictive values from table 5–2 are used to derive the surface specific clearance levels for **buildings** for which it is guaranteed that after clearance they will be **demolished immediately**. For most of the relevant radionuclides these values are about a factor 10 less restrictive than for reuse.
- Finally, mass specific clearance levels for **rubble** are derived from the most restrictive scenarios in table 5–3. The clearance levels are designed in such a way that they apply also for large amounts of material. It must, however, be ensured by good radiation protection practice that parts of the facility with high activities, like e.g. the inner parts of the biological shield in nuclear power plants, are not mixed with the uncontaminated parts of the building and being cleared.
- The calculations of the clearance levels were not based on any assumptions that would require to apply restrictions on the total mass or total activity.

5.2.1 Surface specific clearance levels

The recommended clearance levels in the second and third column of table 5–4 represent the total activity in the structure per unit surface area. After clearance the building can be used as indicated,

i.e. when complying with the clearance levels in the second column of table 5–4, it may be used for non-nuclear purposes or demolished, when complying with the clearance levels in the third column, it must be demolished without prior non-nuclear use. The clearance levels have been derived on the basis of the following considerations:

1. The surface specific clearance levels in table 5–4 apply to the total activity under the surface to be measured divided by its area. The total activity is the sum of the fixed and non-fixed activity on the surface plus the activity which has penetrated into the bulk.
2. In nearly all practical cases more than one radionuclide is involved. To determine if a mixture of radionuclides is below the clearance level a summation formula can be used:

$$\sum_{i=1}^{n} \frac{c_i}{c_{li}} < 1.0$$

where
c_i is the total activity (fixed, non-fixed and penetrated) in the structure per unit surface area of radionuclide i (Bq/cm²),
c_{li} is the clearance level of radionuclide i (Bq/cm²),
n is the number of radionuclides in the mixture.

In the above expression, the ratio of the concentration of each radionuclide to the clearance level is summed over all radionuclides in the mixture. If this sum is less than one the material complies with the clearance requirements.

3. The short lived progeny of the radionuclides do not need to be included in the summation formula (cf. table 3–1).
4. The activity from naturally occurring radionuclides which does not arise from the regulated practice itself does not need to be taken into account.

5.2.2 Mass specific clearance levels

The mass specific clearance levels in the fourth column of table 5–4 apply for building rubble which has already been generated. When complying with these clearance levels, the building rubble may be used for any purpose (recycling in roads, houses, etc., disposal in landfills etc.). The clearance levels have been derived on the basis of the following considerations:

1. The mass specific clearance levels in table 5–4 shall not be exceeded. The mass over which averaging is allowed should in general not exceed 1 Mg.
2. In nearly all practical cases more than one radionuclide is involved. To determine if a mixture of radionuclides is below the clearance level a summation formula can be used:

$$\sum_{i=1}^{n} \frac{c_i}{c_{li}} < 1.0$$

where
c_i is the mass specific activity of radionuclide i (Bq/g),
c_{li} is the clearance level of radionuclide i (Bq/g),
n is the number of radionuclides in the mixture.

In the above expression, the ratio of the concentration of each radionuclide to the clearance level is summed over all radionuclides in the mixture. If this sum is less than one the material complies with the clearance requirements.

3. The short lived progeny of the radionuclides do not need to be included in the summation formula (cf. table 3–1).
4. The activity from naturally occurring radionuclides which does not arise from the regulated practice itself does not need to be taken into account.

The mass specific clearance levels in table 5–4 are valid for any quantity of rubble, typically on the order of one nuclear power plant. For quantities of rubble not exceeding about 100 Mg/a from one site, it is possible to apply relaxed clearance levels. For such quantities mass specific clearance levels a factor 10 higher would usually be radiologically acceptable.

5.2.3 Applicability of the clearance levels to other types of installations

The discussion of potential quantities of contaminated rubble and source term used in the scenarios (10^5 Mg), shows that this approach leads to nuclide specific levels which are also appropriate for further installations, e.g. isotope labs, etc. In these cases, the quantities are smaller while the use of the cleared material is the same. Differences in the nuclide vectors are accounted for by the fact that the clearance levels are expressed as nuclide specific values.

The radiological analysis which lead to the clearance levels evaluated the non-nuclear use of buildings and the recycle or disposal of large quantities of building rubble coming from the dismantling of nuclear facilities, in particular nuclear power plants. A number of the radionuclides in table 5–4 are not present in any significant quantity in the typical radionuclide mixes coming from such facilities and hence the cleared quantities are probably over-estimated for such radionuclides. In some cases these clearance levels may therefore be overly restrictive.

The scope of this recommendation does not include particle accelerators because the scenarios used to derive the clearance values are not applicable to the conditions characteristic for accelerators, especially high and deep activations in the building walls.

5.2.4 Clearance levels in relation to exemption values

Problems could occur if the clearance criteria were so high that the cleared material would still require reporting upon receipt for further use or processing. The mass specific clearance level should therefore not exceed the corresponding exemption values in the BSS [CEU 96].

The derived clearance levels in table 5–4 were compared to the exemption values and in no case were the exemption values more restrictive than the clearance levels. Under these circumstances the radionuclide concentration in the cleared material will be below the mass specific exemption value and therefore exempt from reporting or authorisation. It should be noted, however, that during certain processes radionuclides can concentrate in certain fractions of the cleared material, for example in the dusts generated during the processing of building rubble, so that the activity concentration in these by-products may exceed the exemption values. The radiological analysis has accounted for such phenomena in the scenarios.

Table 5–4: Derived clearance levels for buildings and building rubble

Radio-nuclide	Surface specific clearance levels for buildings		Clearance levels for rubble
	Reuse or Demoliton	Direct Demolition	Mass specific
	Bq/cm²	Bq/cm²	Bq/g
H 3	3.8E+3	3.8E+3	6.2E+1
C 14	2.8E+3	5.8E+3	1.0E+1
Na 22	4.4E-1	3.5E+0	1.0E-1
S 35	2.6E+3	2.0E+5	1.0E+3
Cl 36	3.2E+1	3.2E+1	1.1E+0
K 40	5.6E+0	2.4E+1	7.9E-1
Ca 45	1.1E+3	6.4E+4	4.2E+2
Sc 46	1.3E+0	1.1E+1	1.1E-1
Mn 53	2.3E+4	2.3E+4	1.5E+3
Mn 54	1.5E+0	1.2E+1	2.6E-1
Fe 55	1.0E+4	2.4E+4	6.1E+2
Co 56	8.2E-1	6.1E+0	6.2E-2
Co 57	1.2E+1	1.3E+2	2.7E+0
Co 58	3.2E+0	2.6E+1	2.3E-1
Co 60	3.6E-1	2.9E+0	8.9E-2
Ni 59	4.2E+4	8.9E+4	2.9E+3
Ni 63	1.8E+4	3.7E+4	1.2E+3
Zn 65	2.3E+0	1.9E+1	3.8E-1
As 73	4.0E+2	2.1E+4	2.1E+2
Se 75	5.2E+0	4.9E+1	6.7E-1
Sr 85	6.2E+0	5.2E+1	4.4E-1
Sr 90	3.4E+1	3.4E+1	1.5E+0
Y 91	4.1E+2	5.4E+4	1.6E+2
Zr 93	3.1E+2	2.5E+3	8.2E+1
Zr 95	1.8E+0	1.5E+1	1.2E-1
Nb 93m	5.0E+2	3.8E+4	1.2E+3
Nb 94	5.3E-1	4.3E+0	1.4E-1
Mo 93	7.5E+1	2.3E+3	3.8E+1
Tc 97	8.0E+1	6.9E+2	1.4E+1
Tc 97m	2.9E+2	5.2E+2	8.6E+0
Tc 99	7.0E+1	7.0E+1	1.4E+0
Ru 106	5.6E+0	4.5E+1	1.1E+0
Ag 108m	5.1E-1	4.2E+0	1.4E-1
Ag 110m	4.8E-1	3.9E+0	8.1E-2
Cd 109	4.0E+1	4.1E+3	1.0E+2
Sn 113	7.2E+0	6.7E+1	8.9E-1
Sb 124	1.9E+0	1.5E+1	2.0E+2
Sb 125	2.1E+0	1.8E+1	5.4E-1
Te 123m	1.4E+1	1.6E+2	2.1E+0
Te 127m	1.3E+2	3.3E+3	4.3E+1
I 125	7.5E+1	1.4E+4	1.1E+2

Table 5–4: Derived clearance levels for buildings and building rubble

Radio-nuclide	Surface specific clearance levels for buildings		Clearance levels for rubble
	Reuse or Demoliton	Direct Demolition	Mass specific
	Bq/cm²	Bq/cm²	Bq/g
I 129	7.5E+0	7.5E+0	1.2E-1
Cs 134	6.3E-1	5.1E+0	1.4E-1
Cs 135	1.8E+3	8.8E+3	4.3E+2
Cs 137	1.5E+0	1.2E+1	4.0E-1
Ce 139	1.2E+1	1.4E+2	2.1E+0
Ce 144	2.6E+1	2.4E+2	5.2E+0
Pm 147	1.5E+3	2.4E+4	6.0E+2
Sm 151	3.6E+3	2.9E+4	9.5E+2
Eu 152	7.7E-1	6.2E+0	2.0E-1
Eu 154	6.9E-1	5.7E+0	1.8E-1
Eu 155	1.5E+1	2.6E+2	8.1E+0
Gd 153	1.2E+1	2.9E+2	6.0E+0
Tb 160	2.9E+0	2.3E+1	2.1E-1
Tm 170	3.7E+2	9.0E+3	1.3E+2
Tm 171	1.5E+3	5.8E+4	1.5E+3
Ta 182	1.7E+0	1.4E+1	1.8E-1
W 181	5.1E+1	1.7E+3	2.4E+1
W 185	8.1E+2	3.9E+5	3.2E+2
Os 185	3.3E+0	2.9E+1	3.3E-1
Ir 192	3.7E+0	3.1E+1	2.9E-1
Tl 204	4.8E+2	2.5E+3	8.1E+1
Pb 210	1.4E+0	1.4E+0	8.7E-2
Bi 207	5.4E-1	4.5E+0	1.5E-1
Po 210	4.2E+0	7.4E+1	1.1E+0
Ra 226	4.9E-1	9.4E-1	8.3E-2
Ra 228	4.4E-1	3.8E+0	1.2E-1
Th 228	2.7E-1	2.6E+0	7.3E-2
Th 229	1.2E-1	9.4E-1	3.1E-2
Th 230	3.3E-1	2.7E+0	8.8E-2
Th 232	1.4E-1	1.2E+0	3.8E-2
Pa 231	1.3E-2	1.1E-1	3.5E-3
U 232	1.7E-1	1.4E+0	4.5E-2
U 233	1.2E+0	9.7E+0	3.2E-1
U 234	1.4E+0	1.1E+1	3.6E-1
U 235	1.3E+0	1.0E+1	3.4E-1
U 236	1.5E+0	1.2E+1	3.9E-1
U 238	1.6E+0	1.3E+1	4.3E-1
Np 237	6.2E-1	5.0E+0	1.6E-1
Pu 236	7.1E-1	6.5E+0	1.9E-1
Pu 238	3.1E-1	2.5E+0	8.2E-2
Pu 239	2.9E-1	2.3E+0	7.7E-2

Table 5–4: Derived clearance levels for buildings and building rubble

Radio-nuclide	Surface specific clearance levels for buildings		Clearance levels for rubble
	Reuse or Demoliton	Direct Demolition	Mass specific
	Bq/cm²	Bq/cm²	Bq/g
Pu 240	2.9E-1	2.3E+0	7.7E-2
Pu 241	1.1E+1	9.2E+1	3.0E+0
Pu 242	3.0E-1	2.4E+0	8.0E-2
Pu 244	3.1E-1	2.5E+0	8.2E-2
Am 241	3.4E-1	2.8E+0	9.1E-2
Am 242m	3.2E-1	2.6E+0	8.5E-2
Am 243	3.4E-1	2.8E+0	9.1E-2
Cm 242	2.5E+0	4.0E+1	6.7E-1
Cm 243	4.6E-1	3.8E+0	1.2E-1
Cm 244	5.5E-1	4.5E+0	1.5E-1
Cm 245	3.0E-1	2.4E+0	8.0E-2
Cm 246	3.4E-1	2.8E+0	9.1E-2
Cm 247	3.7E-1	3.0E+0	9.9E-2
Cm 248	9.8E-2	7.9E-1	2.6E-2
Bk 249	8.4E+1	9.8E+2	2.2E+1
Cf 248	1.5E+0	1.7E+1	4.0E-1
Cf 249	2.1E-1	1.7E+0	5.5E-2
Cf 250	4.2E-1	3.5E+0	1.1E-1
Cf 251	2.0E-1	1.6E+0	5.4E-2
Cf 252	7.1E-1	6.6E+0	1.9E-1
Cf 254	4.2E-1	1.4E+1	1.1E-1
Es 254	1.4E+0	1.2E+1	2.5E-1

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APPENDIX

1. DERIVATION OF THE FORMULA FOR THE RADIONUCLIDE CONCENTRATION IN DRINKING WATER

The activity in the building rubble is washed out into an aquifer from which drinking water is drawn. Figure 4–2 shows a schematic diagram of the model. The radionuclide concentration in the ground water is calculated in section 4.3.4 using the equation 4-12, which is derived here starting with the transport equation,

$$(\partial_t + \vec{v}\cdot\vec{\nabla} - \vec{\nabla}\cdot(\underline{\underline{D}}\cdot\vec{\nabla}) + \lambda)\, C(\vec{x},t) = Q(\vec{x},t) \qquad \text{eq. A1}$$

where,

- $\vec{x}$ position vector, $\vec{x} = (x,y,z)$,
- t time,
- ∂_i partial derivative, i ∈ {t, x, y, z},
- $\vec{\nabla}$ 3-dimensional differential operator, $\vec{\nabla} = (\partial_x, \partial_y, \partial_z)$,
- $\vec{v}$ fluid velocity, $\vec{v} = (v_x, v_y, v_z)$,
- $\underline{\underline{D}}$ dispersion tensor,
- λ radioactive decay constant,
- $C(\vec{x},t)$ concentration and
- $Q(\vec{x},t)$ source term.

Although using the transport equation to calculate the migration of substances in ground water only applies to very ideal situations it has nevertheless been used in many radiological impact studies to estimate the doses arising from the use of contaminated water (see for example, [IAE 87], [SSK 88], [DOM 90], [WIL 93]) and is used here.

The concentration vector can be expressed as the sum of the activity bound to the solid medium and that found in solution,

$$C(\vec{x},t) = f\,\varepsilon\, C_l(\vec{x},t) + (1-\varepsilon)\,\rho_s\, C_s(\vec{x},t)\,, \qquad \text{eq. A2}$$

where

- ρ_s : the material density of the solid medium (the global density without fluid can be expressed as $\rho = (1\text{-}\varepsilon)\,\rho_s$),
- ε : fraction of the volume occupied by pores and
- f : fraction of pore volume which is filled with fluid (usable pore fraction $n = f\,\varepsilon$).

To describe the sorption processes it is assumed that the activity fraction bound to the solid medium is at all times in equilibrium with the fraction in solution and that this equilibrium can be described by a constant, which is called K_d (see table a2–1),

$$C_s(\vec{x},t) = K_d\, C_l(\vec{x},t) \quad . \qquad \text{eq. A3}$$

Besides expressing all the chemical reactions by this one simple factor, K_d, a number of further simplifications are made:

- there is no diffusion or dispersion during the transport ($\underline{\underline{D}} = \underline{\underline{0}}$)
- the fluid flow is laminar stationary and one dimensional,
- the aquifer is unconfined,
- the activity is found only in the fluid and solid phases (not in the gas phase),
- the medium through which the activity is transported is homogeneous.

The first assumption is valid for one dimensional calculations when the distance between the source and observation point is the same order of magnitude as the dimension of the source and the observation point lies on the line of maximum concentration. The laminar stationary flow and homogeneous medium assumptions will in general not be valid, but if only long time averages and large areas are considered the assumptions will often be sufficiently fulfilled to make the calculations meaningful. In general activity will not be found in the gas phase. Exceptions to this are H 3 and C 14 which under the right circumstances are found extensively in the gas phase. With the help of equations A2 and A3 the transport equation A1 can be rewritten using the simplifying assumptions as,

$$\left(\partial_t + \frac{n v_x}{n+\rho K_d}\partial_x + \lambda\right) C_l(x,t) = \frac{Q(x,t)}{n+\rho\, K_d} \quad . \qquad \text{eq. A4}$$

The Darcy velocity $v_{Darcy} = n\, v_x$ for the fluid and the radionuclide migration velocity $v_N = n\, v_x / (n+\rho K_d)$ show up naturally in the derivation equation A4.

Equation A4 is solved here using the method of Green's functions. For ease of reading the equation is rewritten using the operator L,

$$L = \partial_t + u\partial_x + \lambda \quad ,$$

where $u = n\, v_x/(n+\rho K_d)$ with which the equation can be expressed as,

$$L\Psi(x,t) = q(x,t) \quad .$$

Letting the source term $q(x,t) = Q(x,t)/(n+\rho K_d)$ a solution to this equation is,

$$\Psi(x,t) = \iint dx'\, dt'\, \Phi(x,x',t,t')\, q(x',t') \quad , \qquad \text{eq. A5}$$

under the condition that

$$L\Phi(x,x',t,t') = \delta(x-x')\,\delta(t-t') \qquad \text{eq. A6}$$

where δ is the Dirac delta function. The solution to equation A4 can now be reduced to searching for a solution to equation A6 which is the Green's function for the problem at hand. To solve this equation the spatial coordinate, x is Fourier transformed[1]

$$(\partial_t - iuk + \lambda)\tilde{\Phi}(k,x',t,t') = \mathrm{e}^{ikx'}\,\delta(t-t')$$

and the differential equation is now solved in Fourier space,

$$\tilde{\Phi}(k,x',t,t') = \mathrm{e}^{(iuk-\lambda)t}\int_{-\infty}^{t} dt''\,\mathrm{e}^{(\lambda-iu)t''}\delta(t''-t')\mathrm{e}^{ikx'} = \theta(t-t')\mathrm{e}^{(iuk-\lambda)(t-t')}\,\mathrm{e}^{ikx'},$$

where θ is the Heaviside function ($\theta(x) = 0$ when $x < 0$ otherwise $\theta(x) = 1$). The solution is now transformed back which results in the desired solution to equation A6,

$$\Phi(x,x',t,t') = \theta(t-t')\mathrm{e}^{-\lambda(t-t')}\,\delta(x-x'-u(t-t'))\,. \qquad \text{eq. A7}$$

The following source term assumes that the pile of rubble grows at a constant rate r during the time period t_b and that the activity is homogeneously distributed through out the rubble,

$$\frac{Q(y,t)}{n_D + \rho_D K_d^D} = \alpha\,\delta(y-rt)\theta(t_b - t), \qquad \text{eq. A8}$$

where

$$\alpha = \frac{A}{F t_b (n_D + \rho_D K_d^D)}$$

A : total activity in the rubble,
F : area over which the rubble is distributed,
n_D : usable fraction of the pore volume,
ρ_D : global density without fluid,
K_d^D : equilibrium constant for the rubble seep water system (see table a2–1).

The solution to equation A4 for this source term is arrived at by integrating the product of the source term A8 and the Green's function A7 according to equation A5 over space and time and is given by,

$$C_l(y,t) = \frac{\alpha}{|r - v_{ND}|}\,\theta\left(t - \frac{y - v_{ND}\,t}{r - v_{ND}}\right)\theta\left(t_b - \frac{y - v_{ND}\,t}{r - v_{ND}}\right)\mathrm{e}^{-\lambda\left(t - \frac{y - v_{ND}\,t}{r - v_{ND}}\right)}, \qquad \text{eq. A9}$$

[1] Fourier transformation

$$g(x) = \tfrac{1}{2\pi}\int dk\,\tilde{g}(k)\,\mathrm{e}^{-ikx}$$

where $v_{ND} = n_D v_D / (n_D + \rho_D K_d^D)$ is the migration velocity of the radionuclides through the rubble. Noting that the growth of the rubble pile and the direction of the radionuclide migration are in opposite directions, the concentration of the seep water exiting at the base of the rubble pile (y = 0) can be expressed as,

$$C_l(0,t) = \frac{A}{F t_b (r + |v_{ND}|)(n_D + \rho_D K_d^D)} \, e^{-\lambda t \left(\frac{r}{r + |v_{ND}|} \right)} \, \theta(t_n - t) \quad , \qquad \text{eq. A10}$$

where

$|v_{ND}|$: the magnitude of the radionuclide migration vector and

t_n : $t_b \frac{(r + |v_{ND}|)}{|v_{ND}|}$ is the time period during which the activity is washed out of the rubble.

Equation A10 now serves as the source term for the activity in the aquifer. It is assumed that the activity enters the aquifer perpendicular to the flow between $x = 0$ and $x = \sqrt{F}$ (see figure 4–2). Under these circumstances the source term for the contamination in the aquifer can be written as,

$$\frac{Q(x,t)}{n_G + \rho_G K_d^G} = \alpha \, f(t) \, \theta(x) \, \theta(\sqrt{F} - x) \, \theta(t_n - t) \quad , \qquad \text{eq. A11}$$

where

$$f(t) = \frac{A}{F t_b (r + |v_{ND}|)(n_D + \rho_D K_d^D)} \, e^{-\lambda t \left(\frac{r}{r + |v_{ND}|} \right)}$$

$$\alpha = \frac{|n_D v_D|}{h (n_G + \rho_G K_d^G)}$$

$|n_D v_D|$: magnitude of the Darcy velocity in the rubble pile,

h : depth of the aquifer,

n_G : usable pore volume fraction in the aquifer,

ρ_G : global density of the aquifer without fluid and

K_d^G : equilibrium constant for the aquifer water system (see table a2–1).

It is assumed that the equilibrium constants in table a2–1 are valid for both the aquifer and the rubble, that is $K_d^G = K_d^D$. Equation A5 in conjunction with the Green's function A7 and the source term A11 are used to derive the concentration function for the ground water,

$$C_l(x,t) = \int_0^\infty dt' \, \alpha f(t') e^{-\lambda (t - t')} \, \theta(t_n - t') \theta(t - t') \theta(x - v_{NG}(t - t')) \theta(\sqrt{F} - x + v_{NG}(t - t')) \quad , \text{ eq. A12}$$

where $v_{NG} = n_G v_G / (n_G + \rho_G K_d^G)$ is the migration velocity of the radionuclides through the aquifer. The solution given by equation A12 can be solved for a particular set of parameters. Letting

x_o $= \sqrt{F} + s_B$ position of the well (see figure 4–2),
t_o $= x_o / v_{NG}$ the time at which the water is drawn from the well and
v_{NG} $> \sqrt{F}/t_n$ the magnitude of the radionuclide migration velocity in the aquifer

the maximum concentration in the drinking water can be expressed as,

$$C_l(x = x_o, t = t_o) = \frac{A}{h F t_b \lambda (n_G + \rho_G K_d^G)} \, e^{-\lambda t_o} \left(e^{\lambda \frac{\sqrt{F}|v_{ND}|}{v_{NG}(r+|v_{ND}|)}} - 1 \right) . \qquad \text{eq. A13}$$

The condition on the migration velocity guaranties that the nuclide washout time is longer than the time required for the nuclide to be transported from the back side to the front side of the rubble pile (from x = 0 to x = $\sqrt{F}$, see figure 4–2) since equation A13 is only valid when $v_{NG} > v_{ND}\sqrt{F}/(t_b(v_{ND} + r))$. The hydraulic conductivity and gradient are chosen such that this condition is satisfied and equation A13 is valid (see table 4–7).

2. DOSE COEFFICIENTS USED IN THE CALCULATIONS

In table a2–3 the dose coefficients used in the dose calculations are presented. In table a2–2 the radionuclides whose decay products are radioactive and cause an increase in the dose coefficients during the first 100 years are shown. For these radionuclides the dose coefficients, including the radionuclides in secular equilibrium, at the time of clearance CD(t=0) are given in the first row. The second row gives the time t_{max} at which the dose coefficient reaches its maximum and the last row the maximum dose coefficient CD($t=t_{max}$) used for the calculations and found in table a2–3. The list contains 19 radionuclides for which this correction leads to an increase in the dose coefficients. If the time was set significantly longer, like on the order of 10^4 to 10^6 years, then of course radionuclides like U 238 would also have to be corrected. During the first 100 years though the build up of U 234 from the decay of U 238 is negligible and therefore the dose coefficient remains essentially constant during this time.

Table A2–1: Equilibrium constants (K_d) from [SHE 90] for sandy soil

Radio-nuclide	K_d cm³/g	Comments	Radio-nuclide	K_d cm³/g	Comments	Radio-nuclide	K_d cm³/g	Comments
H 3	0,1	smallest value (Tc)	Sn 113	130		Th 232	3200	
C 14	5		Sb 124	45		Pa 231	550	
Na 22	15	same as K	Sb 125	45		U 232	35	
S 35	150	same as Se	Te 123m	125		U 233	35	
Cl 36	0,1	same as Tc	Te 127m	125		U 234	35	
K 40	15		I 125	1		U 235	35	
Ca 45	5		I 129	1		U 236	35	
Sc 46	170	same as Y	Cs 134	280		U 238	35	
Mn 53	50		Cs 135	280		Np 237	5	
Mn 54	50		Cs 137	280		Pu 236	550	
Fe 55	220		Ce 139	500		Pu 238	550	
Co 56	60		Ce 144	500		Pu 239	550	
Co 57	60		Pm 147	5	same as Np	Pu 240	550	
Co 58	60		Sm 151	245		Pu 241	550	
Co 60	60		Eu 152	1900	same as Am	Pu 242	550	
Ni 59	400		Eu 154	1900	same as Am	Pu 244	550	
Ni 63	400		Eu 155	1900	same as Am	Am 241	1900	
Zn 65	200		Gd 153	4000	same as Cm	Am 242m	1900	
As 73	45	same as Sb	Tb 160	245	same as Sm	Am 243	1900	
Se 75	150		Tm 170	245	same as Sm	Cm 242	4000	
Sr 85	15		Tm 171	245	same as Sm	Cm 243	4000	
Sr 90	15		Ta 182	220		Cm 244	4000	
Y 91	170		W 181	10	same as Mo	Cm 245	4000	
Zr 93	600		W 185	10	same as Mo	Cm 246	4000	
Zr 95	600		Os 185	55	same as Ru	Cm 247	4000	
Nb 93m	160		Ir 192	60	same as Co	Cm 248	4000	
Nb 94	160		Tl 204	45	same as Sb	Bk 249	550	same as Pu
Mo 93	10		Pb 210	270		Cf 248	550	same as Pu
Tc 97	0,1		Bi 207	100		Cf 249	550	same as Pu
Tc 97m	0,1		Po 210	150		Cf 250	550	same as Pu
Tc 99	0,1		Ra 226	500		Cf 251	550	same as Pu
Ru 106	55		Ra 228	500		Cf 252	550	same as Pu
Ag 108m	90		Th 228	3200		Cf 254	550	same as Pu
Ag 110m	90		Th 229	3200		Es 254	550	same as Pu
Cd 109	80		Th 230	3200				

Table A2–2: The dose coefficients (DC) including nuclides in secular equilibrium at the time of clearance (t=0), the time (t_{max}) within the first 100 years at which the dose coefficients reach a maximum due to the radioactive progeny and the dose coefficient at this time DC($t=t_{max}$)

Radio-nuclides	External gamma dose rates		Inhalation coefficients	Ingestion coefficients			β-Skin dose coefficients (7 mg/cm²)
	Sphere (µSv/h)/ (Bq/cm²)	Semi-inf. vol. (µSv/h)/ (Bq/g)	Worker (5 µm) Sv/ Bq	Child (1-2 a) Sv/ Bq	Adult (>17a) Sv/ Bq	Worker Sv/ Bq	(Sv/a)/ (Bq/cm²)
Zr 93							
DC(t=0)	0	0	2.90E-08	7.60E-10	1.10E-09	2.80E-10	0
t_{max}	100	100	100	100	100	100	0
DC($t=t_{max}$)	2.20E-05	1.90E-06	3.00E-08	1.70E-09	1.20E-09	4.00E-10	0
Mo 93							
DC(t=0)	1.30E-04	1.10E-05	1.40E-09	6.90E-09	3.10E-09	2.60E-09	0
t_{max}	76	76	96	71	46	50	0
DC($t=t_{max}$)	1.50E-04	1.20E-05	2.20E-09	7.70E-09	3.20E-09	2.70E-09	0
Pb 210							
DC(t=0)	5.10E-05	4.70E-05	1.20E-06	3.60E-06	6.90E-07	6.80E-07	1.90E-02
t_{max}	0	1	2	2	2	2	0
DC($t=t_{max}$)	5.10E-05	4.70E-05	3.10E-06	1.20E-05	1.80E-06	8.60E-07	1.90E-02
Ra 226							
DC(t=0)	2.30E-02	3.90E-01	2.20E-06	9.60E-07	2.80E-07	2.80E-07	8.10E-02
t_{max}	0	0	100	100	100	100	85
DC($t=t_{max}$)	2.30E-02	3.90E-01	5.20E-06	1.20E-05	2.00E-06	1.10E-06	9.60E-02
Ra 228							
DC(t=0)	1.20E-02	2.00E-01	1.70E-06	5.70E-06	6.90E-07	6.70E-07	2.00E-02
t_{max}	3	3	4	0	0	0	3
DC($t=t_{max}$)	1.90E-02	3.30E-01	2.10E-05	5.70E-06	6.90E-07	6.70E-07	3.60E-02
Th 230							
DC(t=0)	1.50E-05	3.20E-05	2.80E-05	4.10E-07	2.10E-07	2.10E-07	0
t_{max}	100	100	100	100	100	100	100
DC($t=t_{max}$)	9.80E-04	1.70E-02	2.80E-05	8.10E-07	2.80E-07	2.50E-07	4.00E-03
Th 232							
DC(t=0)	1.20E-05	1.20E-05	2.90E-05	4.50E-07	2.30E-07	2.20E-07	1.80E-05
t_{max}	100	100	100	100	100	100	100
DC($t=t_{max}$)	3.10E-02	5.50E-01	6.50E-05	7.20E-06	1.10E-06	1.00E-06	6.00E-02
Pa 231							
DC(t=0)	4.90E-04	5.40E-03	8.90E-05	1.30E-06	7.10E-07	7.10E-07	6.60E-04
t_{max}	100	100	100	100	100	100	100
DC($t=t_{max}$)	5.70E-03	7.20E-02	7.00E-04	5.40E-06	1.90E-06	1.90E-06	4.50E-02
U 232							
DC(t=0)	1.80E-05	2.00E-05	2.60E-05	8.20E-07	3.30E-07	3.30E-07	3.00E-05
t_{max}	10	10	9	9	7	7	10
DC($t=t_{max}$)	1.70E-02	3.20E-01	5.50E-05	1.70E-06	4.30E-07	4.30E-07	3.60E-02
U 233							
DC(t=0)	7.60E-06	2.50E-05	6.90E-06	1.40E-07	5.10E-08	5.00E-08	6.80E-06
t_{max}	100	100	100	100	100	100	100
DC($t=t_{max}$)	5.00E-05	5.40E-04	7.70E-06	1.60E-07	5.70E-08	5.60E-08	4.90E-04

Table A2–2: The dose coefficients (DC) including nuclides in secular equilibrium at the time of clearance (t=0), the time (t_{max}) within the first 100 years at which the dose coefficients reach a maximum due to the radioactive progeny and the dose coefficient at this time DC($t=t_{max}$)

Radio-nuclides	External gamma dose rates		Inhalation coefficients	Ingestion coefficients			β-Skin dose coefficients (7 mg/cm²)
	Sphere (µSv/h)/ (Bq/cm²)	Semi-inf. vol. (µSv/h)/ (Bq/g)	Worker (5 µm) Sv/ Bq	Child (1-2 a) Sv/ Bq	Adult (>17a) Sv/ Bq	Worker Sv/ Bq	(Sv/a)/ (Bq/cm²)
U 234							
DC(t=0)	1.50E-05	8.60E-06	6.80E-06	1.30E-07	4.90E-08	4.90E-08	2.10E-05
t_{max}	100	100	100	100	100	100	100
DC($t=t_{max}$)	1.50E-05	1.60E-05	6.80E-06	1.30E-07	4.90E-08	4.90E-08	2.30E-05
U 235							
DC(t=0)	2.20E-03	2.40E-02	6.10E-06	1.30E-07	4.70E-08	4.60E-08	9.30E-03
t_{max}	100	100	100	100	100	100	100
DC($t=t_{max}$)	2.30E-03	2.40E-02	7.20E-06	1.40E-07	5.10E-08	5.00E-08	9.40E-03
Pu 236							
DC(t=0)	1.80E-05	2.80E-06	1.30E-05	2.20E-07	8.70E-08	8.60E-08	0
t_{max}	18	18	0	0	0	0	18
DC($t=t_{max}$)	6.40E-04	1.20E-02	1.30E-05	2.20E-07	8.70E-08	8.60E-08	1.40E-03
Pu 241							
DC(t=0)	0	0	5.80E-07	5.70E-09	4.80E-09	4.70E-09	0
t_{max}	73	73	52	60	47	48	77
DC($t=t_{max}$)	1.10E-05	2.70E-05	8.30E-07	1.10E-08	6.20E-09	6.20E-09	6.60E-07
Pu 244							
DC(t=0)	9.80E-05	2.60E-04	3.00E-05	4.20E-07	2.40E-07	2.40E-07	3.10E-02
t_{max}	100	1	100	100	100	100	0
DC($t=t_{max}$)	4.60E-03	7.00E-02	3.00E-05	4.20E-07	2.40E-07	2.40E-07	3.10E-02
Am 242m							
DC(t=0)	3.00E-04	2.10E-03	2.40E-05	3.00E-07	1.90E-07	1.90E-07	1.30E-02
t_{max}	2	0	52	51	55	55	0
DC($t=t_{max}$)	3.10E-04	2.10E-03	2.90E-05	3.90E-07	2.10E-07	2.10E-07	1.30E-02
Cm 245							
DC(t=0)	9.60E-04	8.30E-03	2.70E-05	3.70E-07	2.10E-07	2.10E-07	5.80E-04
t_{max}	100	100	100	100	100	100	0
DC($t=t_{max}$)	9.90E-04	8.40E-03	3.10E-05	4.20E-07	2.40E-07	2.40E-07	5.80E-04
Bk 249							
DC(t=0)	0	0	1.00E-07	2.90E-09	9.70E-10	9.70E-10	8.20E-04
t_{max}	8	8	5	0	0	0	0
DC($t=t_{max}$)	1.10E-05	1.60E-04	1.10E-07	2.90E-09	9.70E-10	9.70E-10	8.20E-04
Cf 254							
DC(t=0)	3.20E-10	4.30E-10	2.20E-05	2.60E-06	4.00E-07	4.00E-07	4.10E-01
t_{max}	2	3	0	0	0	0	0
DC($t=t_{max}$)	3.90E-10	6.30E-09	2.20E-05	2.60E-06	4.00E-07	4.00E-07	4.10E-01

Table A2–3: Dose coefficients used in the calculations, which include the radionuclides in secular equilibrium and the progeny within the first 100 years after clearance

Radio-nuclides	External gamma dose rates		Inhalation coefficients	Ingestion coefficients			β-Skin dose coefficients (7 mg/cm²)
	Sphere (µSv/h)/ (Bq/cm²)	Semi-inf. vol. (µSv/h)/ (Bq/g)	Worker (5 µm) Sv/ Bq	Child (1-2 a) Sv/ Bq	Adult (>17a) Sv/ Bq	Worker Sv/ Bq	(Sv/a)/ (Bq/cm²)
H 3	0.00E+00	0.00E+00	4.10E-11	1.20E-10	4.20E-11	4.20E-11	0.00E+00
C 14	0.00E+00	0.00E+00	5.80E-10	1.60E-09	5.80E-10	5.80E-10	2.90E-03
Na 22	2.90E-02	4.80E-01	2.00E-09	1.50E-08	3.20E-09	3.20E-09	1.70E-02
S 35	0.00E+00	0.00E+00	1.10E-09	5.40E-09	7.70E-10	7.70E-10	3.10E-03
Cl 36	0.00E+00	0.00E+00	5.10E-09	6.30E-09	9.30E-10	9.30E-10	1.70E-02
K 40	2.00E-03	3.70E-02	3.00E-09	4.20E-08	6.20E-09	6.20E-09	1.80E-02
Ca 45	4.40E-12	9.10E-14	2.30E-09	4.90E-09	7.10E-10	7.60E-10	7.70E-03
Sc 46	2.70E-02	4.50E-01	4.80E-09	7.90E-09	1.50E-09	1.50E-09	1.20E-02
Mn 53	0.00E+00	0.00E+00	3.60E-11	2.20E-10	3.00E-11	3.00E-11	0.00E+00
Mn 54	1.10E-02	1.90E-01	1.20E-09	3.10E-09	7.10E-10	7.10E-10	0.00E+00
Fe 55	0.00E+00	0.00E+00	9.20E-10	2.40E-09	3.30E-10	3.30E-10	0.00E+00
Co 56	4.50E-02	8.30E-01	4.90E-09	1.50E-08	2.50E-09	2.50E-09	1.00E-02
Co 57	1.50E-03	1.60E-02	6.00E-10	1.60E-09	2.10E-10	2.10E-10	6.90E-04
Co 58	1.30E-02	2.20E-01	1.70E-09	4.40E-09	7.40E-10	7.40E-10	2.80E-03
Co 60	3.30E-02	5.70E-01	1.70E-08	2.70E-08	3.40E-09	3.40E-09	9.90E-03
Ni 59	0.00E+00	0.00E+00	2.20E-10	3.40E-10	6.30E-11	6.30E-11	0.00E+00
Ni 63	0.00E+00	0.00E+00	5.20E-10	8.40E-10	1.50E-10	1.50E-10	0.00E+00
Zn 65	7.70E-03	1.30E-01	2.80E-09	1.60E-08	3.90E-09	3.90E-09	2.30E-04
As 73	9.20E-05	1.80E-04	6.50E-10	1.90E-09	2.60E-10	2.60E-10	4.90E-03
Se 75	5.20E-03	6.80E-02	1.70E-09	1.30E-08	2.60E-09	2.60E-09	8.40E-04
Sr 85	7.10E-03	1.00E-01	6.40E-10	3.10E-09	5.60E-10	5.60E-10	1.30E-04
Sr 90	0.00E+00	0.00E+00	7.90E-08	9.30E-08	3.10E-08	3.10E-08	3.70E-02
Y 91	4.70E-05	8.10E-04	6.10E-09	1.80E-08	2.40E-09	2.40E-09	2.00E-02
Zr 93	2.20E-05	1.90E-06	3.00E-08	1.70E-09	1.20E-09	4.00E-10	0.00E+00
Zr 95	2.60E-02	4.20E-01	6.20E-09	1.00E-08	1.80E-09	1.80E-09	1.60E-02
Nb 93m	2.30E-05	1.90E-06	8.60E-10	9.10E-10	1.20E-10	1.20E-10	0.00E+00
Nb 94	2.10E-02	3.50E-01	2.50E-08	9.70E-09	1.70E-09	1.70E-09	1.40E-02
Mo 93	1.50E-04	1.20E-05	2.20E-09	7.70E-09	3.20E-09	2.70E-09	0.00E+00
Tc 97	1.40E-04	1.50E-05	1.60E-10	4.90E-10	6.80E-11	8.30E-11	4.50E-04
Tc 97m	1.20E-04	4.60E-05	2.70E-09	4.10E-09	5.50E-10	6.60E-10	7.10E-03
Tc 99	6.70E-09	4.80E-08	3.20E-09	4.80E-09	6.40E-10	7.80E-10	8.60E-03
Ru 106	2.80E-03	4.40E-02	3.50E-08	4.90E-08	7.00E-09	7.00E-09	2.20E-02
Ag 108m	2.20E-02	3.40E-01	1.90E-08	1.10E-08	2.30E-09	2.30E-09	2.10E-03
Ag 110m	3.70E-02	6.10E-01	7.30E-09	1.40E-08	2.80E-09	2.80E-09	5.00E-03
Cd 109	3.60E-04	3.70E-04	9.60E-09	9.50E-09	2.00E-09	2.00E-09	4.50E-03
Sn 113	3.90E-03	5.40E-02	1.90E-09	5.20E-09	7.60E-10	7.60E-10	7.40E-03
Sb 124	2.40E-02	4.30E-01	4.70E-09	1.60E-08	2.50E-09	2.50E-09	1.60E-02
Sb 125	6.10E-03	8.70E-02	4.00E-09	7.60E-09	1.30E-09	1.30E-09	9.90E-03
Te 123m	1.90E-03	2.00E-02	3.40E-09	8.80E-09	1.40E-09	1.40E-09	7.80E-03

Table A2–3: Dose coefficients used in the calculations, which include the radionuclides in secular equilibrium and the progeny within the first 100 years after clearance

Radio-nuclides	External gamma dose rates		Inhalation coefficients	Ingestion coefficients			β-Skin dose coefficients (7 mg/cm²)
	Sphere (µSv/h)/ (Bq/cm²)	Semi-inf. vol. (µSv/h)/ (Bq/g)	Worker (5 µm) Sv/ Bq	Child (1-2 a) Sv/ Bq	Adult (>17a) Sv/ Bq	Worker Sv/ Bq	(Sv/a)/ (Bq/cm²)
Te 127m	2.30E-04	1.10E-03	6.40E-09	1.90E-08	2.50E-09	2.50E-09	2.00E-02
I 125	6.20E-04	2.70E-04	7.30E-09	5.70E-08	1.50E-08	1.50E-08	0.00E+00
I 129	3.70E-04	2.20E-04	5.10E-08	2.20E-07	1.10E-07	1.10E-07	1.90E-03
Cs 134	2.10E-02	3.40E-01	9.60E-09	1.60E-08	1.90E-08	1.90E-08	1.20E-02
Cs 135	0.00E+00	0.00E+00	9.90E-10	2.30E-09	2.00E-09	2.00E-09	4.50E-03
Cs 137	7.80E-03	1.20E-01	6.70E-09	1.20E-08	1.30E-08	1.30E-08	1.60E-02
Ce 139	2.10E-03	2.10E-02	1.40E-09	1.60E-09	2.60E-10	2.60E-10	2.90E-03
Ce 144	6.50E-04	9.60E-03	2.90E-08	3.90E-08	5.30E-09	5.30E-09	3.10E-02
Pm 147	4.40E-08	4.40E-07	3.50E-09	1.90E-09	2.60E-10	2.60E-10	5.40E-03
Sm 151	8.90E-08	1.60E-08	2.60E-09	6.40E-10	9.80E-11	9.80E-11	5.20E-06
Eu 152	1.50E-02	2.50E-01	2.70E-08	7.40E-09	1.40E-09	1.40E-09	7.90E-03
Eu 154	1.70E-02	2.70E-01	3.50E-08	1.20E-08	2.00E-09	2.00E-09	1.80E-02
Eu 155	8.00E-04	5.10E-03	4.70E-09	2.20E-09	3.20E-10	3.20E-10	2.90E-03
Gd 153	1.50E-03	6.90E-03	2.50E-09	1.80E-09	2.70E-10	2.70E-10	1.10E-03
Tb 160	1.40E-02	2.30E-01	5.40E-09	1.00E-08	1.60E-09	1.60E-09	1.80E-02
Tm 170	7.00E-05	3.20E-04	5.20E-09	9.80E-09	1.30E-09	1.30E-09	1.90E-02
Tm 171	9.20E-06	2.40E-05	9.10E-10	7.80E-10	1.10E-10	1.10E-10	2.10E-03
Ta 182	1.70E-02	2.80E-01	7.40E-09	9.40E-09	1.50E-09	1.50E-09	1.90E-02
W 181	5.30E-04	1.60E-03	4.30E-11	4.70E-10	7.60E-11	7.60E-11	8.80E-04
W 185	3.30E-07	3.40E-06	2.20E-10	3.30E-09	4.40E-10	4.40E-10	1.00E-02
Os 185	9.60E-03	1.40E-01	1.40E-09	2.60E-09	5.10E-10	5.10E-10	3.20E-04
Ir 192	1.10E-02	1.60E-01	4.90E-09	8.70E-09	1.40E-09	1.40E-09	2.10E-02
Tl 204	1.40E-05	7.00E-05	6.20E-10	8.50E-09	1.20E-09	1.30E-09	1.70E-02
Pb 210	5.10E-05	4.70E-05	3.10E-06	1.20E-05	1.80E-06	8.60E-07	1.90E-02
Bi 207	2.10E-02	3.20E-01	3.20E-09	7.10E-09	1.30E-09	1.30E-09	9.50E-03
Po 210	1.20E-07	1.90E-06	2.20E-06	8.80E-06	1.20E-06	2.40E-07	0.00E+00
Ra 226	2.30E-02	3.90E-01	5.20E-06	1.20E-05	2.00E-06	1.10E-06	9.60E-02
Ra 228	1.90E-02	3.30E-01	2.10E-05	5.70E-06	6.90E-07	6.70E-07	3.60E-02
Th 228	1.90E-02	3.50E-01	3.40E-05	1.10E-06	1.40E-07	1.40E-07	4.00E-02
Th 229	4.50E-03	5.50E-02	8.00E-05	2.40E-06	6.10E-07	6.00E-07	5.20E-02
Th 230	9.80E-04	1.70E-02	2.80E-05	8.10E-07	2.80E-07	2.50E-07	4.00E-03
Th 232	3.10E-02	5.50E-01	6.50E-05	7.20E-06	1.10E-06	1.00E-06	6.00E-02
Pa 231	5.70E-03	7.20E-02	7.00E-04	5.40E-06	1.90E-06	1.90E-06	4.50E-02
U 232	1.70E-02	3.20E-01	5.50E-05	1.70E-06	4.30E-07	4.30E-07	3.60E-02
U 233	5.00E-05	5.40E-04	7.70E-06	1.60E-07	5.70E-08	5.60E-08	4.90E-04
U 234	1.50E-05	1.60E-05	6.80E-06	1.30E-07	4.90E-08	4.90E-08	2.30E-05
U 235	2.30E-03	2.40E-02	7.20E-06	1.40E-07	5.10E-08	5.00E-08	9.40E-03
U 236	1.40E-05	4.70E-06	6.30E-06	1.30E-07	4.70E-08	4.60E-08	1.90E-05
U 238	3.30E-04	3.90E-03	5.70E-06	1.50E-07	4.80E-08	4.70E-08	2.40E-02

Table A2–3: Dose coefficients used in the calculations, which include the radionuclides in secular equilibrium and the progeny within the first 100 years after clearance

Radio-nuclides	External gamma dose rates		Inhalation coefficients	Ingestion coefficients			β-Skin dose coefficients (7 mg/cm²)
	Sphere (µSv/h)/ (Bq/cm²)	Semi-inf. vol. (µSv/h)/ (Bq/g)	Worker (5 µm) Sv/ Bq	Child (1-2 a) Sv/ Bq	Adult (>17a) Sv/ Bq	Worker Sv/ Bq	(Sv/a)/ (Bq/cm²)
Np 237	3.30E-03	4.00E-02	1.50E-05	2.20E-07	1.10E-07	1.10E-07	1.70E-02
Pu 236	6.40E-04	1.20E-02	1.30E-05	2.20E-07	8.70E-08	8.60E-08	1.40E-03
Pu 238	1.60E-05	1.50E-06	3.00E-05	4.00E-07	2.30E-07	2.30E-07	4.30E-09
Pu 239	6.70E-06	6.70E-06	3.20E-05	4.20E-07	2.50E-07	2.50E-07	9.20E-10
Pu 240	1.50E-05	1.50E-06	3.20E-05	4.20E-07	2.50E-07	2.50E-07	3.80E-10
Pu 241	1.10E-05	2.70E-05	8.30E-07	1.10E-08	6.20E-09	6.20E-09	6.60E-07
Pu 242	1.30E-05	1.40E-06	3.10E-05	4.00E-07	2.40E-07	2.40E-07	3.70E-10
Pu 244	4.60E-03	7.00E-02	3.00E-05	4.20E-07	2.40E-07	2.40E-07	3.10E-02
Am 241	3.70E-04	9.20E-04	2.70E-05	3.70E-07	2.00E-07	2.00E-07	2.20E-05
Am 242m	3.10E-04	2.10E-03	2.90E-05	3.90E-07	2.10E-07	2.10E-07	1.30E-02
Am 243	3.00E-03	2.80E-02	2.70E-05	3.80E-07	2.00E-07	2.00E-07	2.30E-02
Cm 242	1.70E-05	1.70E-06	3.70E-06	7.60E-08	1.20E-08	1.20E-08	2.20E-11
Cm 243	1.70E-03	1.90E-02	2.00E-05	3.30E-07	1.50E-07	1.50E-07	1.10E-02
Cm 244	1.50E-05	1.30E-06	1.70E-05	2.90E-07	1.20E-07	1.20E-07	7.90E-13
Cm 245	9.90E-04	8.40E-03	3.10E-05	4.20E-07	2.40E-07	2.40E-07	5.80E-04
Cm 246	1.40E-05	8.60E-07	2.70E-05	3.70E-07	2.10E-07	2.10E-07	3.40E-14
Cm 247	4.70E-03	6.70E-02	2.50E-05	3.50E-07	1.90E-07	1.90E-07	1.50E-02
Cm 248	1.10E-05	1.10E-06	9.50E-05	1.40E-06	7.70E-07	7.70E-07	2.40E-08
Bk 249	1.10E-05	1.60E-04	1.10E-07	2.90E-09	9.70E-10	9.70E-10	8.20E-04
Cf 248	1.30E-05	8.10E-07	6.10E-06	1.60E-07	2.80E-08	2.80E-08	4.80E-04
Cf 249	4.60E-03	6.60E-02	4.50E-05	8.70E-07	3.50E-07	3.50E-07	1.90E-03
Cf 250	1.30E-05	2.50E-06	2.20E-05	5.50E-07	1.60E-07	1.60E-07	4.70E-06
Cf 251	1.60E-03	1.60E-02	4.60E-05	8.80E-07	3.60E-07	3.60E-07	1.60E-02
Cf 252	1.20E-05	1.70E-06	1.30E-05	5.10E-07	9.00E-08	9.00E-08	4.80E-06
Cf 254	3.90E-10	6.30E-09	2.20E-05	2.60E-06	4.00E-07	4.00E-07	4.13E-01
Es 254	1.20E-02	1.90E-01	6.00E-06	1.60E-07	2.80E-08	2.80E-08	5.70E-03

3. DATA FOR THE NUCLEAR INSTALLATIONS

This section lists the nuclear installations in Europe that were used for deriving the estimates for the arisings of building rubble. Table A3–1 shows the nuclear power plants that currently (1998) possess a valid operating licence. Table A3–2 shows all the plants which have already been shut down up to the year 1998, tables A3–3 and A3–4 present all research reactors and facilities of the fuel cycle, respectively.

An explanation of the data is given in section 2.1 of the main part of this report. Concerning the following tables it must be emphasized that the dates given for the shut-down for operating installations and for level 3 dismantling (i.e. total dismantling completed) as well as the value for the mass per unit power and for the total waste masses are estimations only which are only used as a data basis for the calculations presented in the main part of this report. Masses and mass ratings are shown unrounded because they are directly taken from the calculations. Although the data were collected with great care, there might be errors in the assumptions for a certain plant (operating and dismantling time, mass rating etc.) which are, however, insignificant for the overall result.

Table A3–1: Operating nuclear power plants

Name	Type	Operating Time [a]	Decay Time [a]	Power [MWe]	Start operation in	Level 3 dismantling	Mass/power [Mg/MWe]	Total waste masses [Mg]
Germany								
Obrigheim	PWR	40	10	357	1968	2018	112	39819
Stade	PWR	40	10	672	1972	2022	112	74954
Biblis A	PWR	40	10	1204	1974	2024	112	134292
Biblis B	PWR	40	10	1300	1976	2026	112	145000
Neckarwesth.1	PWR	40	10	855	1976	2026	112	95365
Unterweser	PWR	40	10	1300	1978	2028	112	145000
Grafenrheinfeld	PWR	40	10	1300	1981	2031	112	145000
Grohnde	PWR	40	10	1365	1984	2034	112	152250
Philippsburg-2	PWR	40	10	1349	1984	2034	112	150465
Brokdorf	PWR	40	10	1383	1986	2036	112	154258
Mülheim-Kärl.	PWR	40	10	1302	1986	2036	112	145223
Emsland	PWR	40	10	1341	1988	2038	112	149573
Isar-2	PWR	40	10	1400	1988	2038	112	156154
Neckarwesth.2	PWR	40	10	1365	1989	2039	112	152250
Brunsbüttel	BWR	40	10	806	1976	2026	238	191425
Isar-1	BWR	40	10	907	1977	2027	238	215413
Philippsburg-1	BWR	40	10	900	1979	2029	238	213750
Krümmel	BWR	40	10	1316	1983	2033	238	312550
Gundremm. B	BWR	40	10	1310	1984	2034	238	311125
Gundremm. C	BWR	40	10	1310	1984	2034	238	311125

Table A3–1: Operating nuclear power plants

Name	Type	Operating Time [a]	Decay Time [a]	Power [MWe]	Start operation in	Level 3 dismantling	Mass/power [Mg/MWe]	Total waste masses [Mg]
France								
Belleville-1	PWR	40	50	1363	1987	2077	112	152027
Belleville-2	PWR	40	50	1363	1988	2078	112	152027
Blayais-1	PWR	40	50	951	1981	2071	112	106073
Blayais-2	PWR	40	50	951	1982	2072	112	106073
Blayais-3	PWR	40	50	951	1983	2073	112	106073
Blayais-4	PWR	40	50	951	1983	2073	112	106073
Bugey-2	PWR	40	50	945	1978	2068	112	105404
Bugey-3	PWR	40	50	945	1978	2068	112	105404
Bugey-4	PWR	40	50	917	1979	2069	112	102281
Bugey-5	PWR	40	50	917	1979	2069	112	102281
Cattenom-1	PWR	40	50	1362	1986	2076	112	151915
Cattenom-2	PWR	40	50	1362	1987	2077	112	151915
Cattenom-3	PWR	40	50	1362	1990	2080	112	151915
Cattenom-4	PWR	40	50	1362	1991	2081	112	151915
Chinon B-1	PWR	40	50	954	1982	2072	112	106408
Chinon B-2	PWR	40	50	954	1983	2073	112	106408
Chinon B-3	PWR	40	50	954	1986	2076	112	106408
Chinon B-4	PWR	40	50	954	1987	2077	112	106408
Chooz B-1	PWR	40	50	1516	1996	2086	112	169092
Chooz B-2	PWR	40	50	1516	1997	2087	112	169092
Cruas-1	PWR	40	50	956	1983	2073	112	106631
Cruas-2	PWR	40	50	956	1984	2074	112	106631
Cruas-3	PWR	40	50	956	1984	2074	112	106631
Cruas-4	PWR	40	50	956	1984	2074	112	106631
Dampierre-1	PWR	40	50	937	1980	2070	112	104512
Dampierre-2	PWR	40	50	937	1980	2070	112	104512
Dampierre-3	PWR	40	50	937	1981	2071	112	104512
Dampierre-4	PWR	40	50	937	1981	2071	112	104512
Fressenheim-1	PWR	40	50	920	1977	2067	112	102615
Fressenheim-2	PWR	40	50	920	1977	2067	112	102615
Flamanville-1	PWR	40	50	1382	1985	2075	112	154146
Flamanville-2	PWR	40	50	1382	1986	2076	112	154146
Golfech-1	PWR	40	50	1363	1990	2080	112	152027
Golfech-2	PWR	40	50	1363	1993	2083	112	152027
Gravelines B-1	PWR	40	50	951	1980	2070	112	106073
Gravelines B-2	PWR	40	50	951	1980	2070	112	106073
Gravelines B-3	PWR	40	50	951	1980	2070	112	106073
Gravelines B-4	PWR	40	50	951	1981	2071	112	106073
Gravelines C-5	PWR	40	50	951	1984	2074	112	106073
Gravelines C-6	PWR	40	50	951	1985	2075	112	106073

Table A3–1: Operating nuclear power plants

Name	Type	Operating Time [a]	Decay Time [a]	Power [MWe]	Start operation in	Level 3 dismantling	Mass/power [Mg/MWe]	Total waste masses [Mg]
Nogent-1	PWR	40	50	1363	1987	2077	112	152027
Nogent-2	PWR	40	50	1363	1988	2078	112	152027
Paluel-1	PWR	40	50	1382	1984	2074	112	154146
Paluel-2	PWR	40	50	1382	1984	2074	112	154146
Paluel-3	PWR	40	50	1382	1985	2075	112	154146
Paluel-4	PWR	40	50	1382	1986	2076	112	154146
Penly-1	PWR	40	50	1382	1990	2080	112	154146
Penly-2	PWR	40	50	1382	1992	2082	112	154146
St. Alban-1	PWR	40	50	1381	1985	2075	112	154035
St. Alban-2	PWR	40	50	1381	1986	2076	112	154035
St, Laurent B-1	PWR	40	50	956	1981	2071	112	106631
St, Laurent B-2	PWR	40	50	956	1981	2071	112	106631
Tricastin-1	PWR	40	50	955	1980	2070	112	106519
Tricastin-2	PWR	40	50	955	1980	2070	112	106519
Tricastin-3	PWR	40	50	955	1981	2071	112	106519
Tricastin-4	PWR	40	50	955	1981	2071	112	106519
Civaux-1	PWR	40	50	1516	1997	2087	112	169092
Civaux-2	PWR	40	50	1516	1998	2088	112	169092
Phenix	FBR	40	50	250	1973	2063	145	36250
UK								
Bradwell-1	GGR	25	130	166	1962	2117	145	24070
Bradwell-2	GGR	25	130	166	1962	2117	145	24070
Calder Hall-1	GGR	25	130	55	1956	2111	145	7975
Calder Hall-2	GGR	25	130	55	1957	2112	145	7975
Calder Hall-3	GGR	25	130	55	1959	2114	145	7975
Calder Hall-4	GGR	25	130	55	1959	2114	145	7975
Chapelcross-1	GGR	25	130	55	1959	2114	145	7975
Chapelcross-2	GGR	25	130	55	1959	2114	145	7975
Chapelcross-3	GGR	25	130	55	1959	2114	145	7975
Chapelcross-4	GGR	25	130	55	1960	2115	145	7975
Dungeness A-1	GGR	25	130	285	1965	2120	145	41325
Dungeness A-2	GGR	25	130	285	1965	2120	145	41325
Hinkley Pnt A-1	GGR	25	130	270	1965	2120	145	39150
Hinkley Pnt A-2	GGR	25	130	270	1965	2120	145	39150
Oldbury-1	GGR	25	130	313	1967	2122	145	45385
Oldbury-2	GGR	25	130	313	1968	2123	145	45385
Sizewell A-1	GGR	25	130	325	1966	2121	145	47125
Sizewell A-2	GGR	25	130	325	1966	2121	145	47125
Wylfa-1	GGR	25	130	670	1971	2126	145	97150
Wylfa-2	GGR	25	130	670	1971	2126	145	97150

Table A3–1: Operating nuclear power plants

Name	Type	Operating Time [a]	Decay Time [a]	Power [MWe]	Start operation in	Level 3 dismantling	Mass/power [Mg/MWe]	Total waste masses [Mg]
Dungeness B-1	AGR	30	130	660	1983	2143	145	95700
Dungeness B-2	AGR	30	130	660	1985	2145	145	95700
Hartlepool-1	AGR	30	130	660	1983	2143	145	95700
Hartlepool-2	AGR	30	130	660	1984	2144	145	95700
Heysham I-1	AGR	30	130	660	1983	2143	145	95700
Heysham I-2	AGR	30	130	660	1984	2144	145	95700
Heysham II-1	AGR	30	130	660	1988	2148	145	95700
Heysham II-2	AGR	30	130	660	1988	2148	145	95700
Hinkley Pnt B-1	AGR	30	130	660	1976	2136	145	95700
Hinkley Pnt B-2	AGR	30	130	660	1976	2136	145	95700
Hunterston B-1	AGR	30	130	660	1976	2136	145	95700
Hunterston B-2	AGR	30	130	660	1977	2137	145	95700
Torness Point-1	AGR	30	130	701	1988	2148	145	101645
Torness Point-2	AGR	30	130	701	1988	2148	145	101645
Sizewell B	PWR	40	130	1200	1995	2165	111	133200
Sweden								
Ringhals-2	PWR	40	10	915	1974	2024	112	102058
Ringhals-3	PWR	40	10	960	1982	2032	112	107077
Ringhals-4	PWR	40	10	960	1982	2032	112	107077
Barsebeck-1	BWR	40	10	615	1975	2025	238	146063
Barsebeck-2	BWR	40	10	615	1977	2027	238	146063
Forsmark-1	BWR	40	10	1006	1980	2030	238	238925
Forsmark-2	BWR	40	10	1006	1981	2031	238	238925
Forsmark-3	BWR	40	10	1200	1985	2035	238	285000
Oskarshamm-1	BWR	40	10	465	1971	2021	238	110438
Oskarshamm-2	BWR	40	10	630	1974	2024	238	149625
Oskarshamm-3	BWR	40	10	1205	1985	2035	238	286188
Ringhals-1	BWR	40	10	875	1974	2024	238	207813
Belgium								
Doel-1	PWR	40	10	412	1974	2024	112	45954
Doel-2	PWR	40	10	412	1975	2025	112	45954
Doel-3	PWR	40	10	1020	1982	2032	112	113769
Doel-4	PWR	40	10	1056	1985	2035	112	117785
Tihange-1	PWR	40	10	908	1975	2025	112	101277
Tihange-2	PWR	40	10	934	1982	2032	112	104177
Tihange-3	PWR	40	10	1065	1985	2035	112	118788

Table A3–1: Operating nuclear power plants

Name	Type	Operating Time [a]	Decay Time [a]	Power [MWe]	Start operation in	Level 3 dismantling	Mass/power [Mg/MWe]	Total waste masses [Mg]
Netherlands								
Borsele	PWR	40	10	480	1973	2023	112	53538
Dodewaard	BWR	40	10	58	1968	2018	238	13775
Spain								
Almaraz-1	PWR	40	10	973	1981	2031	112	108527
Almaraz-2	PWR	40	10	975	1983	2033	112	108750
Asco-1	PWR	40	10	973	1983	2033	112	108527
Asco-2	PWR	40	10	966	1985	2035	112	107746
Trillo	PWR	40	10	1066	1988	2038	112	118900
Vandellos-2	PWR	40	10	1009	1987	2037	112	112542
Zorita	PWR	40	10	160	1968	2018	112	17846
Cofrentes	BWR	40	10	990	1984	2034	238	235125
St.M.d. Garona	BWR	40	10	460	1971	2021	238	109250
Finland								
Loviisa-1	WWER	25	10	465	1977	2012	112	51865
Loviisa-2	WWER	25	10	465	1980	2015	112	51865
Olkiluoto-1	BWR	40	10	735	1978	2028	238	174563
Olkiluoto-2	BWR	40	10	735	1980	2030	238	174563

Table A3–2: Shut down nuclear power plants

Name	Type	Shut down	Decay time [a]	Power [MWe]	Start operation in	Level 3 dismantl.	Mass/power [Mg/MWe]	Total waste masses [Mg]
Germany								
Greifswald 1	WWER	1990	10	440	1973	2000	112	49077
Greifswald 2	WWER	1990	10	440	1974	2000	112	49077
Greifswald 3	WWER	1990	10	440	1977	2000	112	49077
Greifswald 4	WWER	1990	10	440	1979	2000	112	49077
Greifswald 5	WWER	1989	10	440	1989	1999	112	49077
HDR Großwelzheim	BWR	1971	10	27	1969	1981	238	6413
Gundremmingen A	BWR	1980	10	250	1966	1990	238	59375
Hamm	HTR	1988	10	308	1985	1998	250	77000
Jülich, AVR	HTR	1988	10	15	1967	1998	250	3750
Kahl	BWR	1985	10	16	1961	1995	238	3800
Kompakt KNK II	FBR	1991	10	20	1978	2001	180	3600
Lingen	BWR	1977	10	268	1968	1987	238	63650
Rheinsberg	PWR	1990	10	75	1966	2000	112	8365
Würgassen	BWR	1995	10	670	1971	2005	238	159125

Table A3–2: Shut down nuclear power plants

Name	Type	Shut down	Decay time [a]	Power [MWe]	Start operation in	Level 3 dismantl.	Mass/power [Mg/MWe]	Total waste masses [Mg]
France								
Bugey 1	Gas-graphite	1994	50	555	1972	2044	145	80475
Chinon A1	Gas-graphite	1973	50	83	1962	2023	145	12035
Chinon A2	Gas-graphite	1985	50	210	1965	2035	145	30450
Chinon A3	Gas-graphite	1990	50	375	1965	2040	145	54375
Chooz	PWR	1991	50	320	1967	2041	112	35692
Marcoule G1	Gas-graphite	1968	50	2	1956	2018	145	290
Marcoule G2	Gas-graphite	1980	50	42	1958	2030	145	6090
Marcoule G3	Gas-graphite	1984	50	42	1960	2034	145	6090
Monts d'Arree	GCHWR	1985	50	75	1967	2035	145	10875
St Laurent A1	Gas-graphite	1990	50	405	1969	2040	145	58725
St Laurent A2	Gas-graphite	1990	50	465	1971	2040	145	67425
Super-Phenix	FBR	1997	50	1242	1986	2047	145	180090
U.K.								
Winscale AGR	AGR	1981	130	36	1963	2111	145	5220
Dounreay DFR	FBR	1977	130	15	1962	2107	145	2175
Dounreay PFR	FBR	1994	130	270	1975	2124	145	39150
Winfrith Dragon	HTGR	1976	130	20 (thermal)	1966	2106	111	< 2200
Berkeley 1	Magnox	1989	130	167	1962	2119	111	18537
Berkeley 2	Magnox	1988	130	167	1962	2118	111	18537
Hunterston A1	Magnox	1990	130	169	1964	2120	111	18759
Hunterston A2	Magnox	1989	130	169	1964	2119	111	18759
Trawsfynydd 1	Magnox	1993	130	236	1965	2123	111	26196
Trawsfynydd 2	Magnox	1993	130	236	1965	2123	111	26196
Winfrith SGHWR	SGHWR	1990	130	100	1967	2120	111	11100
Belgium								
Mol	PWR	1987	10	12	1962	1997	112	1338
Spain								
Valdecaballeros 1	BWR	2000	10	975	2000	2010	238	232050
Valdecaballeros 2	BWR	2000	10	975	2000	2010	238	232050
Lemoniz 1	PWR	2000	10	930	2000	2010	112	104160
Lemoniz 2	PWR	2000	10	930	2000	2010	112	104160
Vandellos 1	Gas-graphite	1990	10	496	1972	2000	145	71920
Italy								
Caorso	BWR	1986	50	882	1978	2036	238	209475
Garigliano	BWR	1982	50	160	1964	2032	238	38000
Latina	Magnox	1987	50	160	1964	2037	111	17760
Trino Vercellese	PWR	1990	50	270	1965	2040	112	30115

Table A3–3: Research reactors

Name	Type	Shut down	Decay time [a]	Power [MWth]	Start operation in	Level 3 dismantl.	Mass/power [Mg/MWth]	Tot. waste mass [Mg]
Germany								
FRM	Pool	2000	20	4	1957	2020	100	400
FRG-1	Pool	2008	20	5	1958	2028	100	500
FRJ-2	Heavy Water	2005	20	23	1962	2025	100	2300
BER-2	Pool	2010	20	10	1973	2030	100	1000
FR-2	Tank	1981	20	44	1961	2001	100	4400
FRJ-1	Pool	1985	20	5	1962	2005	100	500
RFR	Tank	1991	20	10	1957	2011	100	1000
FRG-2	Tank	1995	20	8	1963	2015	100	800
MZFR	PHWR	1984	20	58	1965	2004	100	5800
FMRB	Pool	1995	20	1	1967	2015	100	100
NS Otto-Hahn	PWR	1979	20	38	1968	1999	100	3800
KNK II	FBR	1991	20	58	1971	2011	100	5800
FRN TRIGA III	TRIGA	1982	20	1	1972	2002	100	100
FRF-2	TRIGA	1983	20	1	1971	2003	100	100
France								
EL 3	Heavy Water	1979	10	18	1957	1989	100	1800
Melusine	Pool	1988	10	8	1958	1998	100	800
Triton	Pool	1982	10	6,5	1959	1992	100	650
Siloe	Pool	1997	10	35	1963	2007	100	3500
EL 4	Heavy Water	1985	10	267	1966	1995	100	26700
Rapsodie	FBR	1983	10	40	1967	1993	100	4000
Osiris	Pool	2006	10	70	1966	2016	100	7000
HFR	Heavy Water	2011	10	57	1971	2021	100	5700
Prototype ADF Boiler	BWR	2015	10	120	1975	2025	100	12000
Cabri	Pool	2003	10	25	1963	2013	100	2500
Phebus	Pool	2018	10	40	1978	2028	100	4000
Orphee	Pool	2020	10	14	1980	2030	100	1400
Scarabee	Pool	2022	10	100	1982	2032	100	10000
EL 2	Tank	1965	10	2	1952	1975	100	200
Pegase	Tank	1974	10	30	1963	1984	100	3000
U.K.								
DIDO	Heavy Water	1990	130	26	1956	2120	100	2600
PLUTO	Heavy Water	1995	130	26	1957	2125	100	2600
Dounreay Fast Reactor	FBR	1977	130	65	1959	2107	100	6500
Herald	Pool	1988	130	5	1960	2118	100	500
Merlin	Pool	1976	130	5	1959	2106	100	500
Dounreay MTR	Heavy Water	1969	130	22,5	1958	2099	100	2250
Bepo	Graphite	1968	130	6,5	1962	2098	100	650
Sweden								
R-2	Tank	2007	10	50	1960	2017	100	5000
R2-0	Pool	2007	10	1	1960	2017	100	100

Table A3–3: Research reactors

Name	Type	Shut down	Decay time [a]	Power [MWth]	Start operation in	Level 3 dismantl.	Mass/power [Mg/MWth]	Tot. waste mass [Mg]
Belgium								
BR-1	Graphite	2004	20	4	1956	2024	100	400
BR-2	Tank	2005	20	100	1961	2025	100	10000
BR-3	PWR	1987	20	40,9	1962	2007	100	4090
Netherlands								
HOR	Pool	2003	20	20	1963	2023	100	2000
KSTR	AQU Breeder	1977	20	1	1974	1997	100	100
Spain								
JEN-1 MOD	Pool	1987	20	3	1958	2007	100	300
Italy								
Galileo Galilei RTS-1	Pool	1980	20	5	1963	2000	100	500
Triga RC-1	TRIGA	2000	20	1	1960	2020	100	100
Greece								
Demokritos GRR-1	Pool	2001	20	5	1961	2021	100	500
Austria								
Astra	Pool	2000	20	10	1960	2020	100	1000
Denmark								
DR 2	Pool	1975	20	5	1958	1995	100	500
DR 3	Heavy Water	2000	20	10	1960	2020	100	1000
Norway								
HBWR	Heavy Water	1999	20	25	1959	2019	100	2500
JEEP II	Tank	2001	20	2	1966	2021	100	200
Portuguese								
RPI	Pool	2001	20	1	1961	2021	100	100
EU								
High Flux Reactor HFR	Tank	2001	20	45	1961	2021	100	4500
Essor Nuclear Plant	Heavy Water	1983	20	43	1967	2003	100	4300

Table A3–4: Fuel cycle facilities

Name	Type	Shut down	Decay time[a]	Power [MWth]	Start operation in	Level 3 dismantl.	Capacity*	Total waste mass [Mg]
Germany								
Gronau	Enrichment	2015	10	800000	1985	2025	0,4	320000
U-Facility Hanau	Fuel-Element	1995	10	700	1969	2005	40,0	28000
Hanau MOX Alt	Fuel-Element	1994	10	25	1963	2004	40,0	1000
Ahaus	Interim Storage	2050	10	1500	1990	2060	20,0	30000
Gorleben	Interim Storage	2050	10	1500	1984	2060	20,0	30000
Karlsruhe	Interim Storage	2030	10	55	1971	2040	20,0	1100
WAK	RPP	1990	10	40	1971	2000	30,0	1200

Table A3–4: Fuel cycle facilities

Name	Type	Shut down	Decay time[a]	Power [MWth]	Start operation in	Level 3 dismantl.	Capacity*	Total waste mass [Mg]
France								
Georges Besse Plant	Enrichment	2019	30	1080000	1979	2049	0,4	432000
Melox, Marcoule	Fuel-Element	2035	30	120	1995	2065	40,0	4800
Cadarache	Fuel-Element	2009	30	15	1969	2039	40,0	600
Romans	Fuel-Element	2020	30	750	?	2050	40,0	30000
Pierrelatte	Fuel-Element	2015	30	400	?	2045	40,0	16000
Cascad	Interim Storage	2037	30	180	1987	2067	20,0	3600
La Hague	Interim Storage	2045	30	14500	?	2075	20,0	290000
La Hague UP2 800	RPP	2014	30	800	1994	2044	30,0	24000
La Hague UP2 400	RPP	1994	30	400	1976	2024	30,0	12000
La Hague UP 3	RPP	2010	30	800	1990	2040	30,0	24000
Marcoule	RPP	1996	30	400	1958	2026	30,0	12000
U.K.								
Urenco	Enrichment	2016	130	1100000	1976	2146	0,4	440000
Dounreay RR-Fuel 3	Fuel-Element	2020	130	50	1999	2150	40,0	2000
NOFC Springfields (AGR)	Fuel-Element	2015	130	290	1995	2145	40,0	11600
NOFC Springfields (LWR)	Fuel-Element	2016	130	330	1996	2146	40,0	13200
Sellafield Mox Plant	Fuel-Element	2027	130	120	1997	2157	40,0	4800
Dounreay RR-Fuel 1	Fuel-Element	1999	130	50	1959	2129	40,0	2000
Sellafield MDF	Fuel-Element	2013	130	8	1993	2143	40,0	320
Springfields (AGR)	Fuel-Element	1996	130	300	1995	2126	40,0	12000
Springfields (Magnox)	Fuel-Element	2010	130	1300	1995	2140	40,0	52000
BNFL B27	Interim Storage	2004	130	2300	1964	2134	20,0	46000
BNFL Fuel Handling Plant 2	Interim Storage	2026	130	1800	1986	2156	20,0	36000
BNFL Pond 4	Interim Storage	2026	130	1445	1986	2156	20,0	28900
BNFL Thorp RT&ST 1	Interim Storage	2028	130	1900	1988	2158	20,0	38000
BNFL Thorp RT&ST 2	Interim Storage	2029	130	1900	1989	2159	20,0	38000
Sellafield Oxide Plant 2	Interim Storage	2021	130	3000	1981	2151	20,0	60000
Spent Fuel Handling Plant 1	Interim Storage	2026	130	850	1986	2156	20,0	17000
Wylfa	Interim Storage	2019	130	700	1979	2149	20,0	14000
BNFL B30	Interim Storage	1995	130	1500	1962	2125	20,0	30000
Dounreay RPP	RPP	2015	130	8	1980	2145	30,0	240
Sellafield (Magnox)	RPP	2014	130	1500	1964	2144	30,0	45000
Sellafield (Thorp)	RPP	2025	130	850	1994	2155	30,0	25500
Sweden								
Vasteras	Fuel-Element	2011	10	400	1971	2021	40,0	16000
CLAB	Interim Storage	2015	10	5000	1985	2025	20,0	100000

Table A3–4: Fuel cycle facilities

Name	Type	Shut down	Decay time[a]	Power [MWth]	Start operation in	Level 3 dismantl.	Capacity*	Total waste mass [Mg]
Belgium								
Dessel, belgonucleare	Fuel-Element	2025	10	35	1985	2035	40,0	1400
Dessel, FBFC	Fuel-Element	2037	10	400	1997	2047	40,0	16000
Mol	Interim Storage	1997	10	370	?	2007	20,0	7400
Mol	RPP	1997	10	100	1974	2007	30,0	3000
Netherlands								
Almelo SP3, SP4	Enrichment	2013	10	1550000	1973	2023	0,4	620000
Spain								
Juzbado	Fuel-Element	2025	10	250	1985	2035	40,0	10000
Finland								
KPA Store, Olkiluoto	Interim Storage	2027	10	1270	1987	2037	20,0	25400
Italy								
Bosko-Marengo	Fuel-Element	2014	10	200	1974	2024	40,0	8000
Saluggia	Fuel-Element	2026	10	60	1986	2036	40,0	2400
Saluggia	RPP	2023	10	10	1983	2033	30,0	300
*) Capacity is given in [swu/y] or [tHM/y] or [tHM]								