IAEA Safety Standards

for protecting people and the environment

Application of the Concept of Clearance

General Safety Guide

No. GSG-18

IAEA

International Atomic Energy Agency

IAEA SAFETY STANDARDS AND RELATED PUBLICATIONS

IAEA SAFETY STANDARDS

Under the terms of Article III of its Statute, the IAEA is authorized to establish or adopt standards of safety for protection of health and minimization of danger to life and property, and to provide for the application of these standards.

The publications by means of which the IAEA establishes standards are issued in the **IAEA Safety Standards Series**. This series covers nuclear safety, radiation safety, transport safety and waste safety. The publication categories in the series are **Safety Fundamentals**, **Safety Requirements** and **Safety Guides**.

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The site provides the texts in English of published and draft safety standards. The texts of safety standards issued in Arabic, Chinese, French, Russian and Spanish, the IAEA Safety Glossary and a status report for safety standards under development are also available. For further information, please contact the IAEA at: Vienna International Centre, PO Box 100, 1400 Vienna, Austria.

All users of IAEA safety standards are invited to inform the IAEA of experience in their use (e.g. as a basis for national regulations, for safety reviews and for training courses) for the purpose of ensuring that they continue to meet users' needs. Information may be provided via the IAEA Internet site or by post, as above, or by email to Official.Mail@iaea.org.

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The **IAEA Nuclear Energy Series** comprises informational publications to encourage and assist research on, and the development and practical application of, nuclear energy for peaceful purposes. It includes reports and guides on the status of and advances in technology, and on experience, good practices and practical examples in the areas of nuclear power, the nuclear fuel cycle, radioactive waste management and decommissioning.

APPLICATION OF THE CONCEPT OF CLEARANCE

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The Agency's Statute was approved on 23 October 1956 by the Conference on the Statute of the IAEA held at United Nations Headquarters, New York; it entered into force on 29 July 1957. The Headquarters of the Agency are situated in Vienna. Its principal objective is "to accelerate and enlarge the contribution of atomic energy to peace, health and prosperity throughout the world''.

IAEA SAFETY STANDARDS SERIES No. GSG-18

APPLICATION OF THE CONCEPT OF CLEARANCE

GENERAL SAFETY GUIDE

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FOREWORD

by Rafael Mariano Grossi
Director General

The IAEA's Statute authorizes it to "establish…standards of safety for protection of health and minimization of danger to life and property". These are standards that the IAEA must apply to its own operations, and that States can apply through their national regulations.

The IAEA started its safety standards programme in 1958 and there have been many developments since. As Director General, I am committed to ensuring that the IAEA maintains and improves upon this integrated, comprehensive and consistent set of up to date, user friendly and fit for purpose safety standards of high quality. Their proper application in the use of nuclear science and technology should offer a high level of protection for people and the environment across the world and provide the confidence necessary to allow for the ongoing use of nuclear technology for the benefit of all.

Safety is a national responsibility underpinned by a number of international conventions. The IAEA safety standards form a basis for these legal instruments and serve as a global reference to help parties meet their obligations. While safety standards are not legally binding on Member States, they are widely applied. They have become an indispensable reference point and a common denominator for the vast majority of Member States that have adopted these standards for use in national regulations to enhance safety in nuclear power generation, research reactors and fuel cycle facilities as well as in nuclear applications in medicine, industry, agriculture and research.

The IAEA safety standards are based on the practical experience of its Member States and produced through international consensus. The involvement of the members of the Safety Standards Committees, the Nuclear Security Guidance Committee and the Commission on Safety Standards is particularly important, and I am grateful to all those who contribute their knowledge and expertise to this endeavour.

The IAEA also uses these safety standards when it assists Member States through its review missions and advisory services. This helps Member States in the application of the standards and enables valuable experience and insight to be shared. Feedback from these missions and services, and lessons identified from events and experience in the use and application of the safety standards, are taken into account during their periodic revision.

I believe the IAEA safety standards and their application make an invaluable contribution to ensuring a high level of safety in the use of nuclear technology. I encourage all Member States to promote and apply these standards, and to work with the IAEA to uphold their quality now and in the future.

THE IAEA SAFETY STANDARDS

BACKGROUND

Radioactivity is a natural phenomenon and natural sources of radiation are features of the environment. Radiation and radioactive substances have many beneficial applications, ranging from power generation to uses in medicine, industry and agriculture. The radiation risks to workers and the public and to the environment that may arise from these applications have to be assessed and, if necessary, controlled.

Activities such as the medical uses of radiation, the operation of nuclear installations, the production, transport and use of radioactive material, and the management of radioactive waste must therefore be subject to standards of safety.

Regulating safety is a national responsibility. However, radiation risks may transcend national borders, and international cooperation serves to promote and enhance safety globally by exchanging experience and by improving capabilities to control hazards, to prevent accidents, to respond to emergencies and to mitigate any harmful consequences.

States have an obligation of diligence and duty of care, and are expected to fulfil their national and international undertakings and obligations.

International safety standards provide support for States in meeting their obligations under general principles of international law, such as those relating to environmental protection. International safety standards also promote and assure confidence in safety and facilitate international commerce and trade.

A global nuclear safety regime is in place and is being continuously improved. IAEA safety standards, which support the implementation of binding international instruments and national safety infrastructures, are a cornerstone of this global regime. The IAEA safety standards constitute a useful tool for contracting parties to assess their performance under these international conventions.

THE IAEA SAFETY STANDARDS

The status of the IAEA safety standards derives from the IAEA's Statute, which authorizes the IAEA to establish or adopt, in consultation and, where appropriate, in collaboration with the competent organs of the United Nations and with the specialized agencies concerned, standards of safety for protection of health and minimization of danger to life and property, and to provide for their application.

With a view to ensuring the protection of people and the environment from harmful effects of ionizing radiation, the IAEA safety standards establish fundamental safety principles, requirements and measures to control the radiation exposure of people and the release of radioactive material to the environment, to restrict the likelihood of events that might lead to a loss of control over a nuclear reactor core, nuclear chain reaction, radioactive source or any other source of radiation, and to mitigate the consequences of such events if they were to occur. The standards apply to facilities and activities that give rise to radiation risks, including nuclear installations, the use of radiation and radioactive sources, the transport of radioactive material and the management of radioactive waste.

Safety measures and security measures[1] have in common the aim of protecting human life and health and the environment. Safety measures and security measures must be designed and implemented in an integrated manner so that security measures do not compromise safety and safety measures do not compromise security.

The IAEA safety standards reflect an international consensus on what constitutes a high level of safety for protecting people and the environment from harmful effects of ionizing radiation. They are issued in the IAEA Safety Standards Series, which has three categories (see Fig. 1).

Safety Fundamentals

Safety Fundamentals present the fundamental safety objective and principles of protection and safety, and provide the basis for the safety requirements.

Safety Requirements

An integrated and consistent set of Safety Requirements establishes the requirements that must be met to ensure the protection of people and the environment, both now and in the future. The requirements are governed by the objective and principles of the Safety Fundamentals. If the requirements are not met, measures must be taken to reach or restore the required level of safety. The format and style of the requirements facilitate their use for the establishment, in a harmonized manner, of a national regulatory framework. Requirements, including numbered ‘overarching’ requirements, are expressed as ‘shall’ statements. Many requirements are not addressed to a specific party, the implication being that the appropriate parties are responsible for fulfilling them.

Safety Guides

Safety Guides provide recommendations and guidance on how to comply with the safety requirements, indicating an international consensus that it

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[1] See also publications issued in the IAEA Nuclear Security Series.

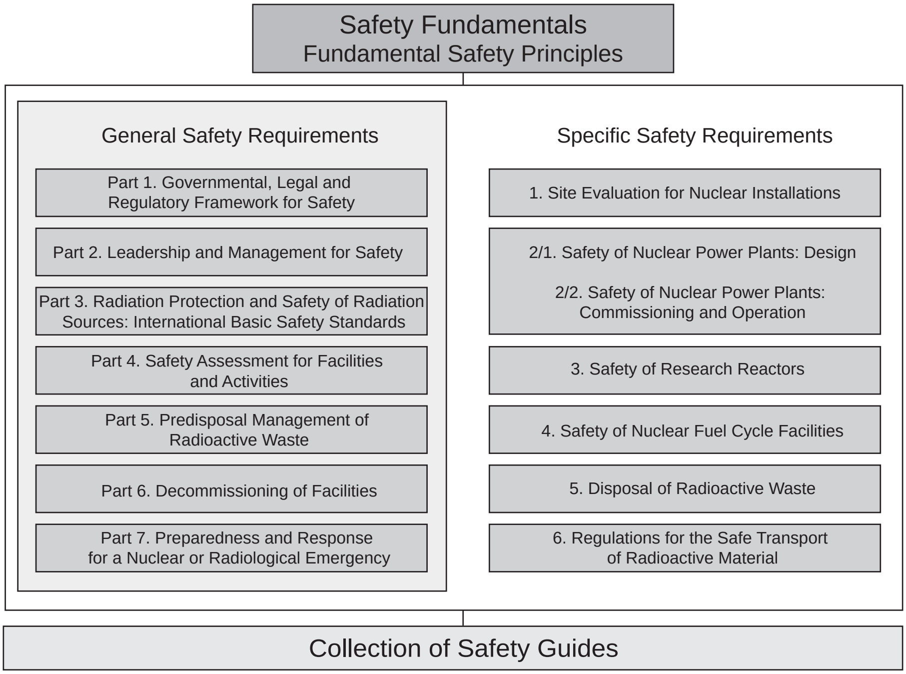

FIG. 1. The long term structure of the IAEA Safety Standards Series.

is necessary to take the measures recommended (or equivalent alternative measures). The Safety Guides present international good practices, and increasingly they reflect best practices, to help users striving to achieve high levels of safety. The recommendations provided in Safety Guides are expressed as ‘should’ statements.

APPLICATION OF THE IAEA SAFETY STANDARDS

The principal users of safety standards in IAEA Member States are regulatory bodies and other relevant national authorities. The IAEA safety standards are also used by co-sponsoring organizations and by many organizations that design, construct and operate nuclear facilities, as well as organizations involved in the use of radiation and radioactive sources.

The IAEA safety standards are applicable, as relevant, throughout the entire lifetime of all facilities and activities — existing and new — utilized for peaceful purposes and to protective actions to reduce existing radiation risks. They can be used by States as a reference for their national regulations in respect of facilities and activities.

The IAEA's Statute makes the safety standards binding on the IAEA in relation to its own operations and also on States in relation to IAEA assisted operations.

The IAEA safety standards also form the basis for the IAEA's safety review services, and they are used by the IAEA in support of competence building, including the development of educational curricula and training courses.

International conventions contain requirements similar to those in the IAEA safety standards and make them binding on contracting parties. The IAEA safety standards, supplemented by international conventions, industry standards and detailed national requirements, establish a consistent basis for protecting people and the environment. There will also be some special aspects of safety that need to be assessed at the national level. For example, many of the IAEA safety standards, in particular those addressing aspects of safety in planning or design, are intended to apply primarily to new facilities and activities. The requirements established in the IAEA safety standards might not be fully met at some existing facilities that were built to earlier standards. The way in which IAEA safety standards are to be applied to such facilities is a decision for individual States.

The scientific considerations underlying the IAEA safety standards provide an objective basis for decisions concerning safety; however, decision makers must also make informed judgements and must determine how best to balance the benefits of an action or an activity against the associated radiation risks and any other detrimental impacts to which it gives rise.

DEVELOPMENT PROCESS FOR THE IAEA SAFETY STANDARDS

The preparation and review of the safety standards involves the IAEA Secretariat and five Safety Standards Committees, for emergency preparedness and response (EPReSC) (as of 2016), nuclear safety (NUSSC), radiation safety (RASSC), the safety of radioactive waste (WASSC) and the safe transport of radioactive material (TRANSSC), and a Commission on Safety Standards (CSS) which oversees the IAEA safety standards programme (see Fig. 2).

All IAEA Member States may nominate experts for the Safety Standards Committees and may provide comments on draft standards. The membership of the Commission on Safety Standards is appointed by the Director General and includes senior governmental officials having responsibility for establishing national standards.

A management system has been established for the processes of planning, developing, reviewing, revising and establishing the IAEA safety standards.

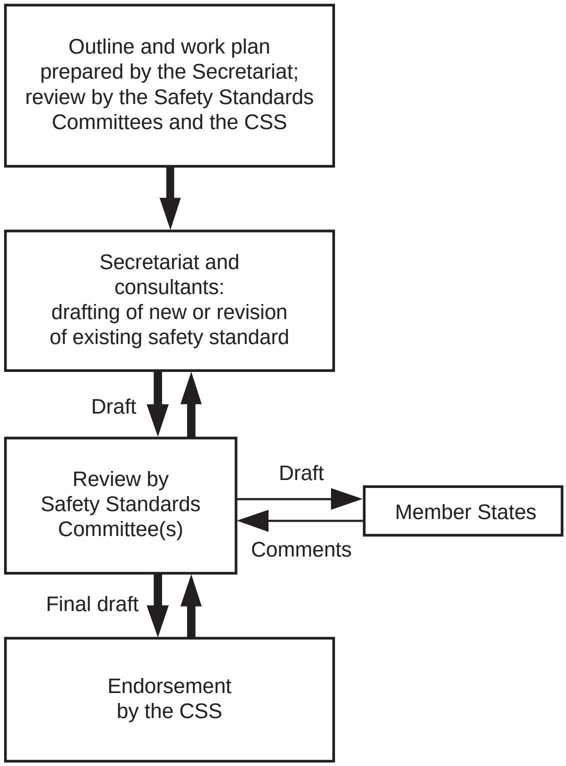

FIG. 2. The process for developing a new safety standard or revising an existing standard.

It articulates the mandate of the IAEA, the vision for the future application of the safety standards, policies and strategies, and corresponding functions and responsibilities.

INTERACTION WITH OTHER INTERNATIONAL ORGANIZATIONS

The findings of the United Nations Scientific Committee on the Effects of Atomic Radiation (UNSCEAR) and the recommendations of international expert bodies, notably the International Commission on Radiological Protection (ICRP), are taken into account in developing the IAEA safety standards. Some safety standards are developed in cooperation with other bodies in the United Nations system or other specialized agencies, including the Food and Agriculture Organization of the United Nations, the United Nations Environment Programme, the International Labour Organization, the OECD Nuclear Energy Agency, the Pan American Health Organization and the World Health Organization.

INTERPRETATION OF THE TEXT

Safety related terms are to be understood as defined in the IAEA Nuclear Safety and Security Glossary (see https://www.iaea.org/resources/publications/iaea-nuclear-safety-and-security-glossary). Otherwise, words are used with the spellings and meanings assigned to them in the latest edition of The Concise Oxford Dictionary. For Safety Guides, the English version of the text is the authoritative version.

The background and context of each standard in the IAEA Safety Standards Series and its objective, scope and structure are explained in Section 1, Introduction, of each publication.

Material for which there is no appropriate place in the body text (e.g. material that is subsidiary to or separate from the body text, is included in support of statements in the body text, or describes methods of calculation, procedures or limits and conditions) may be presented in appendices or annexes.

An appendix, if included, is considered to form an integral part of the safety standard. Material in an appendix has the same status as the body text, and the IAEA assumes authorship of it. Annexes and footnotes to the main text, if included, are used to provide practical examples or additional information or explanation. Annexes and footnotes are not integral parts of the main text. Annex material published by the IAEA is not necessarily issued under its authorship; material under other authorship may be presented in annexes to the safety standards. Extraneous material presented in annexes is excerpted and adapted as necessary to be generally useful.

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1. INTRODUCTION

BACKGROUND

1.1. As defined and explained in IAEA Safety Standards Series No. GSR Part 3, Radiation Protection and Safety of Radiation Sources: International Basic Safety Standards [1], and in Refs [2, 3], the concepts of exclusion, exemption and clearance are used to determine the extent of regulatory control in planned exposure situations. While exclusion and exemption are used as part of a process to determine the extent of application of the system of regulatory control, clearance is intended to establish which material under regulatory control can be removed from this control.

1.2. Figure 1 illustrates the concepts of exclusion, exemption and clearance in planned exposure situations and the application of screening values to decision making in existing exposure situations.

1.3. This Safety Guide is one of the publications supporting the application of Requirement 8 of GSR Part 3 [1] and addresses the concept of clearance. The concepts of exemption and exclusion are addressed in IAEA Safety Standards Series No. GSG-17, Application of the Concept of Exemption [4]. Together, these two Safety Guides supersede IAEA Safety Standards Series No. RS-G-1.7, Application of the Concepts of Exclusion, Exemption and Clearance, issued in 2004[1].

1.4. GSR Part 3 [1] provides values in terms of activity concentration (Bq/g) that can be used for clearance of bulk quantities of solid material. Values are provided for radionuclides of natural origin and for radionuclides of artificial origin. The models used in the calculations of individual doses for artificial radionuclides are described in Ref. [5]; these models are still valid, and therefore they are not repeated in this Safety Guide.

1.5. The values provided for artificial radionuclides were derived using a series of limiting (bounding) exposure scenarios. These scenarios are conservative; therefore, the recommendations in this Safety Guide aim to ensure safe and robust procedures of compliance with clearance levels in order to maintain a high

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[1] INTERNATIONAL ATOMIC ENERGY AGENCY, Application of the Concepts of Exclusion, Exemption and Clearance, IAEA Safety Standards Series No. RS-G-1.7, IAEA, Vienna (2004).

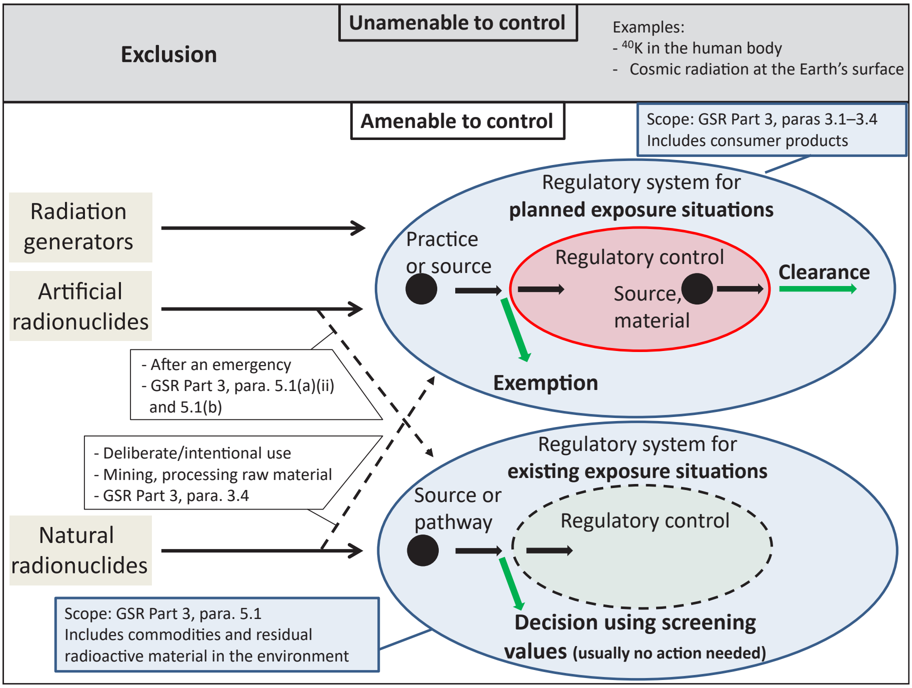

FIG. 1. The concepts of exclusion, exemption and clearance. Prior justification should be applied for sources or practices amenable to control.

level of confidence in receivers and users of cleared materials and to keep any radiation doses at a trivial level. The recommendations also reflect the use of a graded approach.

OBJECTIVE

1.6. The objective of this Safety Guide is to provide recommendations on the application of the concept of clearance for materials (including radioactive waste associated with planned activities), objects and buildings that are to be released from regulatory control in the framework of planned exposure situations, in accordance with Requirement 8 of GSR Part 3 [1]. This Safety Guide covers the regulatory framework for clearance; the clearance process; the derivation of clearance levels; the application of clearance to solid materials, liquids and gases; and generic clearance and specific clearance using activity concentration and surface contamination clearance levels. This Safety Guide also provides recommendations on the involvement of interested parties.

1.7. The application of screening values for recycling or disposal of materials and waste generated during remediation actions after a nuclear or a radiological emergency is also considered.

1.8. This Safety Guide is mainly intended for governments, regulatory bodies and operating organizations to assist them in the application of Requirement 8 of GSR Part 3 [1] in relation to the clearance of materials and objects from regulatory control. It will also be of interest to technical service providers in radiation protection.

SCOPE

1.9. The recommendations provided in this Safety Guide are applicable to facilities that use, manufacture, process or store radioactive material. The types of facility considered include nuclear power plants, research reactors, other nuclear fuel cycle facilities, facilities for the management of radioactive waste, industrial plants, medical facilities, research facilities, educational facilities and accelerators. The recommendations in this Safety Guide also apply to industries processing materials containing radionuclides of natural origin and to products from such industries (e.g. products containing uranium and/or thorium). Examples of industrial processes that use or generate naturally occurring radioactive material are production of oil and gas, manufacture of titanium dioxide pigments, extraction of rare earth elements and alloys, production of metals (e.g. aluminium, iron, steel) and use of thorium in gas mantles. The recommendations also apply to the management of material originating from remediation activities or from post-emergency situations.

1.10. This Safety Guide covers the following:

(a) Responsibilities of the operating organization (i.e. registrant or licensee) and the regulatory body;
(b) All relevant steps of the clearance process, including characterization, determination of the radionuclide composition where there is more than one radionuclide, sampling and measurement techniques, and monitoring and management of the clearance process;
(c) Activity concentration (Bq/g) and surface contamination (Bq/cm^2) clearance levels;
(d) Application of the concept of specific clearance;

(e) Derivation of specific clearance levels in terms of activity concentration and surface contamination;[2]
(f) A case by case approach to specific clearance, which can be used for situations where the generic clearance levels do not apply;
(g) Averaging masses and areas;
(h) Clearance of liquids;
(i) Clearance of gases;
(j) Scenarios underpinning the calculation of clearance levels and the implications for their application;
(k) Involvement of interested parties;
(l) Clearance of materials and waste associated with planned activities in an area affected by a nuclear or radiological emergency.

1.11. The recommendations provided in this Safety Guide are applicable during the operational phase of facilities and during the decommissioning of facilities. They support the minimization of radioactive waste and facilitate the application of the waste hierarchy[3] by maximizing reuse and recycling. The recommendations are also applicable to the clearance of sealed radioactive sources, if this is allowed in a State.

1.12. The application of the concept of exemption is outside the scope of this Safety Guide. Recommendations on exemption are provided in GSG-17 [4].

1.13. The trade of contaminated food and commodities is outside the scope of this Safety Guide.

1.14. Different concepts and criteria apply to the clearance of buildings and to the release of sites from regulatory control. The guidance provided in this Safety Guide is applicable to the clearance of buildings, for example buildings located on a nuclear site. The release of sites from regulatory control is outside the scope of this Safety Guide. This is addressed in IAEA Safety Standards Series No. WS-G-5.1, Release of Sites from Regulatory Control on Termination of Practices [6].

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[2] The actual values used will depend on the specific conditions applied. Consequently, no specific values are proposed in this Safety Guide.

[3] The concept of the ‘waste hierarchy’ is widely accepted to be fundamental to the sustainable management of all types of waste, including radioactive waste. The concept of the waste hierarchy has been widely adopted in national policies and has also been taken up internationally (e.g. by the European Union, the United Nations Environment Programme and the Organisation for Economic Co-operation and Development).

1.15. The management of radioactive waste in a nuclear or radiological emergency is outside the scope of this Safety Guide and is addressed in IAEA Safety Standards Series Nos GSR Part 7, Preparedness and Response for a Nuclear or Radiological Emergency [7], and GSG-11, Arrangements for the Termination of a Nuclear or Radiological Emergency [8].

1.16. The terms used in this Safety Guide are to be understood as defined and explained in GSR Part 3 [1] and the IAEA Nuclear Safety and Security Glossary [9].

STRUCTURE

1.17. Section 2 gives an overview of the regulatory framework for clearance, including the concepts of exclusion and clearance, generic clearance criteria, and the responsibilities of different parties. Section 3 provides recommendations on general aspects of clearance, such as the overall process and its management. Section 4 provides recommendations on the clearance of solid material, including activity concentration and surface contamination clearance levels, averaging masses and areas, implementation of clearance measurements, consideration of uncertainties, and mixing and dilution, as well as consideration of the conservatism applied in the derivation of clearance levels. Sections 5 and 6 provide recommendations on the clearance of liquids and gases, respectively. Recommendations on specific clearance are provided in Section 7. Section 8 provides recommendations on the involvement of interested parties and the enhancement of public understanding in relation to clearance.

1.18. The Appendix provides an example of the application of screening levels for the recycling or conventional landfill disposal of materials and waste generated in a post-emergency situation.

1.19. Annex I provides information on dosimetric models for the derivation of radionuclide specific values for clearance based on surface contamination measurements. Annex II provides examples of surface contamination values for generic clearance. Annex III provides examples of activity concentration values for specific clearance. Annexes IV and V provide examples of the application of clearance in small medical facilities and of a national approach to the clearance of scrap metal, respectively. Annexes VI and VII provide information on selecting significant radionuclides for clearance measurements and for dealing with the uncertainties associated with clearance measurements, respectively. Annex VIII provides information on the screening method applied after the accident at the Fukushima Daiichi nuclear power plant for recycling of material and disposal of waste to landfill. Annex IX considers the conservatism applied in the derivation of clearance levels and in the implementation of the clearance process as a whole.

2. REGULATORY FRAMEWORK FOR CLEARANCE

2.1. Clearance is defined as the removal of regulatory control, by the regulatory body, from radioactive material or radioactive objects within notified or authorized facilities and activities [9]. Removal of regulatory control in this context refers to regulatory control applied for radiation protection purposes. The two main options for clearance considered in this Safety Guide are the following:

(a) Generic clearance: clearance on the basis of the clearance levels provided in schedule I of GSR Part 3 [1] or of any set of values defined in the national regulations of Member States that are intended for clearance without any restrictions on the material's type, amount, further management, reuse, recycling or final destination.
(b) Specific clearance: clearance on the basis of any other clearance levels derived for specific situations, materials and destinations of the cleared material.

2.2. Requirement 8 of GSR Part 3 [1] states that "**The regulatory body shall approve which sources, including materials and objects, within notified practices or authorized practices may be cleared from regulatory control.**"

2.3. In GSR Part 3 [1], the term 'clearance' is used in relation to sources, including materials and objects. The term is also used in relation to waste [10] and, within the context of decommissioning, to buildings or parts of buildings (e.g. rooms or laboratories within a building) [11]. The term is generally not used in the context of release of sites from regulatory control [6], where different criteria are usually applied.

2.4. Paragraph I.10 of schedule I of GSR Part 3 [1] states:

"The general criteria for clearance are that:

(a) Radiation risks arising from the cleared material are sufficiently low as not to warrant regulatory control, and there is no appreciable likelihood of occurrence for scenarios that could lead to a failure to meet the general criterion for clearance; or

(b) Continued regulatory control of the material would yield no net benefit, in that no reasonable control measures would achieve a worthwhile return in terms of reduction of individual doses or reduction of health risks."

2.5. In accordance with para. I.11 of GSR Part 3 [1], material may be cleared without further consideration (i.e. generic clearance) provided that in reasonably foreseeable circumstances the effective dose expected to be incurred by any individual is of the order of 10 μSv in a year. The value of 10 μSv in a year takes into account the possibility of exposure to multiple cleared objects [5].

2.6. In addition to a dose criterion of the order of 10 μSv or less in a year, para. I.11 of GSR Part 3 [1] specifies that radioactive material may be cleared without further consideration (i.e. generic clearance) provided that the expected effective dose incurred by any individual from low probability scenarios does not exceed 1 mSv in a year. This concept of using two sets of scenarios (i.e. 'realistic' and 'low probability') is discussed further in Section 4.

2.7. Generic clearance levels in terms of activity concentration for solid material containing radionuclides of artificial origin are listed in table I.2 of GSR Part 3 [1]. They are based on the dose criteria provided in para. I.11 of GSR Part 3 [1] and were derived using generic models [5]. Clearance levels for material containing radionuclides of natural origin are listed in table I.3 of GSR Part 3 [1]. They were determined on the basis of the worldwide distribution of activity concentrations for such radionuclides.

2.8. Paragraph I.12 of GSR Part 3 [1] states (footnote omitted):

"Radioactive material within a notified practice or an authorized practice may be cleared without further consideration provided that:

…….

(c) For radionuclides of natural origin in residues that might be recycled into construction materials, or the disposal of which is liable to cause the contamination of drinking water supplies, the activity concentration in the residues does not exceed specific values derived so as to meet a dose criterion of the order of 1 mSv in a year, which is commensurate with typical doses due to natural background levels of radiation."

As such, the values in table I.3 of GSR Part 3 [1] for radionuclides of natural origin are not applicable, and the regulatory body will need to stipulate appropriate clearance values for construction materials. The approach to natural radioactivity in construction materials is considered in Refs [12, 13], and modelling of the drinking water exposure pathway for both drinking water and agriculture is addressed in Ref. [14].

2.9. Where compliance with generic clearance levels is not reasonable, different clearance levels that still correspond to the dose criteria in para. I.11 of GSR Part 3 [1] and to the qualitative criteria for clearance stated in para. I.10(a) and (b) of GSR Part 3 [1] may be derived using more specific models (i.e. that are less conservative), or specific materials may be cleared on the basis of specific circumstances. Paragraph I.13 of GSR Part 3 [1] states (footnote omitted):

> "Clearance may be granted by the regulatory body for specific situations… with account taken of the physical or chemical form of the radioactive material, and its use or the means of its disposal. Such clearance levels may be specified in terms of activity concentration per unit mass or activity concentration per unit surface area."

Values for such specific clearance can be derived by the operating organization and proposed to the regulatory body. In such cases, specific clearance values could be included in the authorization issued by the regulatory body. More commonly, specific clearance levels may be proposed by the regulatory body for specific situations. In either case, the regulatory body may attach certain conditions to the cleared material or object and its further management. Further recommendations on specific clearance are provided in Section 7 of this Safety Guide.

2.10. Paragraph 3.12 of GSR Part 3 [1] states:

> "The regulatory body shall approve which sources, including materials and objects, within notified or authorized practices may be cleared from regulatory control, using as the basis for such approval the criteria for clearance specified in Schedule I or any clearance levels specified by the regulatory body on the basis of these criteria. By means of this approval, the regulatory body shall ensure that sources that have been cleared from regulatory control do not again become subject to the requirements for notification, registration or licensing unless it so specifies."

Consequently, the provisions for clearance should be embedded into the regulatory framework, which should clearly specify that cleared materials are

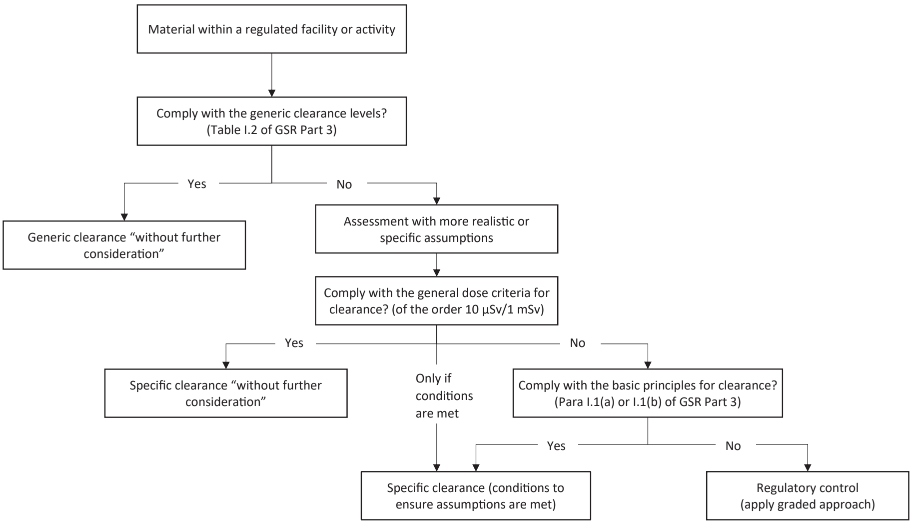

FIG. 2. Application of the concept of clearance based on schedule I of GSR Part 3 [1].

no longer under regulatory control, unless specified otherwise. This applies to materials released under either generic clearance or specific clearance.

2.11. Figure 2 illustrates the different options for clearance of material containing radionuclides of artificial origin, as described in GSR Part 3 [1].

2.12. Material from remediation activities or from post-emergency situations is required to be managed using a protection strategy based on reference levels for existing exposure situations, in accordance with section 5 of GSR Part 3 [1] (see also IAEA Safety Standards Series No. GSG-15, Remediation Strategy and Process for Areas Affected by Past Activities or Events [15]). The concept of clearance can also be applied to the management of material originating from remediation activities or from post-emergency situations. The same qualitative and quantitative criteria as for clearance of materials from planned exposure situations (see paras I.10–I.12 of GSR Part 3 [1]) can be used. Examples of the approaches to such materials are given in the Appendix.

THE CONCEPT OF EXCLUSION

2.13. In accordance with para. 1.42 of GSR Part 3 [1], the requirements of GSR Part 3 [1] apply to all situations involving radiation exposure that is amenable to control. Exposures deemed not to be amenable to control are excluded from the scope, and thereby from regulatory control, regardless of their magnitude. For example, it is not feasible to control exposure from ^{40}K in the human body or from cosmic radiation at the surface of the Earth (see footnote 8 of GSR Part 3 [1]). Other examples of excluded exposures include those from unmodified concentrations of radionuclides of natural origin in normal soil material, including those in high natural background radiation areas. Also excluded are exposures from unmodified concentrations of other primordial radionuclides (e.g. ^{87}Rb, ^{138}La, ^{147}Sm, ^{176}Lu) and fallout resulting from past atmospheric nuclear weapon tests. The concept of exclusion is addressed in more detail in GSG-17 [4].

RESPONSIBILITIES OF THE REGULATORY BODY IN RELATION TO CLEARANCE

2.14. To meet Requirement 8 of GSR Part 3 [1], the regulatory body should establish a framework for the clearance of material, including the clearance levels to be used (which should be in agreement with the clearance criteria defined in schedule I of GSR Part 3 [1]). As part of this framework, the regulatory body should establish requirements for the radiological characterization of materials and objects and should review and validate the results of the characterization programme implemented by operating organizations (see paras 3.9–3.24).

2.15. For the generic clearance of solid material, the regulatory body should refer to the derived clearance levels for radionuclides of artificial origin and for radionuclides of natural origin listed in table I.2 and table I.3 of GSR Part 3 [1], respectively. These values are applicable to clearance of moderate quantities of material (i.e. of the order of a tonne, at the most) as well as bulk quantities [5].

2.16. For clearance of surface contaminated objects, the regulatory body should enable the use of surface contamination clearance levels, derived using the same criteria as for activity concentration clearance levels. Further recommendations are provided in paras 4.17–4.22 and 7.15–7.22.

2.17. When establishing clearance levels, the regulatory body should take into account other regulatory requirements that might apply, such as those in relation to non-radiation-related hazards (e.g. chemical toxicity), and, to the extent possible, should harmonize these requirements.

2.18. If operating organizations are allowed to derive their own clearance levels for specific situations (i.e. specific clearance; see Section 7) on the basis of the clearance criteria in GSR Part 3 [1], the regulatory body should require that the operating organizations demonstrate that these levels will provide an equivalent level of protection and safety. In such cases, the regulatory body and the operating organization should explain the implications of the derived clearance levels to relevant interested parties (see Section 8). If conditions are specified for the type and amount of material to be cleared or for the material's destination, then these conditions should also be explained.

2.19. In addition to defining or approving clearance levels in terms of activity concentration or surface contamination, the regulatory body should specify averaging masses, volumes or areas of material to be monitored for clearance. When approving specific clearance levels proposed by an operating organization, the regulatory body should specify other relevant parameters, such as those relating to the characteristics of the material or its geometry. The regulatory body should also specify additional monitoring criteria to identify non-uniform distribution of activity and how to interpret the results for clearance purposes. Further recommendations are provided in paras 4.23–4.37.

2.20. The regulatory body should specify that deliberate dilution and/or mixing with non-radioactive material to meet clearance levels prior to release of the material from regulatory control is generally not an acceptable practice. However, in some specific exceptional cases, permission may be obtained from the regulatory body for such an action. Further recommendations are provided in paras 4.59–4.63.

2.21. Paragraph 3.37 of GSR Part 3 [1] states:

> "The regulatory body shall establish requirements that monitoring and measurements be performed to verify compliance with the requirements for protection and safety. The regulatory body shall be responsible for review and approval of the monitoring and measurement programmes of registrants and licensees."

The regulatory body should have the competence and resources to review and inspect the arrangements for clearance implemented by operating organizations, including the capability to make independent verification measurements.

2.22. The regulatory body should review the appropriateness of any monitoring undertaken by operating organizations to verify compliance with clearance levels. Based on the results of this monitoring, the operating organization should decide whether material complies with the clearance levels. If the national framework involves approval by the regulatory body as to whether a specific material is suitable for clearance, the approval should be based on these monitoring results and the regulatory body’s own verification programme. If this involves the use of statistically based methods (i.e. decision making on the basis of measurement of samples, where the measurement results are evaluated on the basis of statistical analysis) by the operating organization, the approach should be fully documented, defining the number and locations of samples and the statistical parameters to be met, and should be approved by the regulatory body prior to its implementation.

2.23. The quality management system implemented by the operating organization for clearance purposes should satisfy the requirements established by the regulatory body.

2.24. For specific clearance, the regulatory body should establish a mechanism to verify compliance with any conditions attached, such as on the destination for materials or objects and on their further processing or reuse (e.g. that metals will only go to a recycling facility and will be melted rather than reused directly). In addition, the regulatory body should clarify the responsibilities for the clearance process and the consequences of non-compliance.

2.25. Paragraph 2.35 of GSR Part 3 [1] states that “The regulatory body shall make provision for establishing, maintaining and retrieving adequate records relating to facilities and activities.” For the clearance of material, the regulatory body should define the content of key records and documentation needed to demonstrate compliance with regulatory requirements. In addition, the regulatory body should define the period of time that such records and documentation need to be kept (i.e. depending on the history, nature and characteristics of the material) after materials have been cleared.

2.26. Materials or objects that have been cleared could still be subject to regulatory controls for non-radiation-related purposes (see para. 2.17). Therefore, the regulatory body should coordinate its activities with other relevant regulatory authorities to facilitate management of the material after clearance. In the case of transboundary movement, this coordination should involve regulatory bodies from the relevant countries.

2.27. The regulatory body should establish requirements relating to the education and training of persons who have responsibilities for clearance, including qualified experts, radiation protection officers, workers and staff of the regulatory body.

2.28. The regulatory body should consult with interested parties in developing the regulatory framework for clearance, in particular to enhance public understanding (see Section 8).

RESPONSIBILITIES OF THE OPERATING ORGANIZATION FOR CLEARANCE

2.29. Paragraph 3.38 of GSR Part 3 states:

> "Registrants and licensees and employers shall ensure that:
>
> (a) Monitoring and measurements of parameters are performed as necessary for verification of compliance with the requirements of [GSR Part 3];
> (b) Suitable equipment is provided and procedures for verification are implemented;
> (c) Equipment is properly maintained, tested and calibrated at appropriate intervals with reference to standards traceable to national or international standards;
> (d) Records are maintained of the results of monitoring and verification of compliance, as required by the regulatory body, including records of the tests and calibrations carried out in accordance with [GSR Part 3];
> (e) The results of monitoring and verification of compliance are shared with the regulatory body as required."

2.30. The operating organization should perform all necessary steps for the clearance process, such as radiological characterization, any necessary treatment of the material or objects (e.g. decontamination), measurements to demonstrate compliance with the clearance criteria, (including selection of facilities and equipment for measurements and calibration of equipment), establishment of an organizational structure with clear responsibilities, hiring of competent people, training of staff, development of procedures and documentation, and liaison with the regulatory body and interested parties, in accordance with the national framework for clearance.

2.31. The process of clearance of material from regulatory control should be an integral part of the management system that the operating organization is required to establish and maintain (see Requirement 5 of GSR Part 3 [1]). The operating organization should develop and implement a quality management programme for monitoring of compliance with clearance levels, which should involve the development and use of controlled procedures and working instructions.

2.32. As part of the clearance process, the operating organization should perform a radiological characterization of the material or object to be cleared, comprising determination of the radionuclide composition (i.e. the radionuclides present, their activity concentration or surface contamination levels, and the spatial distribution of the activity), and should identify the relevant clearance option and clearance levels to be applied (i.e. generic, specific or derived on a case by case basis). The results should be submitted to the regulatory body, where this is required by the national framework for clearance.

2.33. The operating organization is responsible for the reliability of the results of its own monitoring programme for clearance. Any verification programme performed by the regulatory body should not be considered a substitute for the management system established by the operating organization.

2.34. The operating organization should communicate the results of its clearance monitoring programme to the regulatory body to obtain regulatory approval for the clearance of material, where this is required by the national framework for clearance.

2.35. The operating organization should retain key records from clearance monitoring to demonstrate that the monitoring has been adequately performed. These records should be produced and preserved in an appropriate format, as specified by the regulatory body. Such records should be stored for a defined period of time, as specified by the regulatory body.

2.36. The operating organization should liaise with receivers of waste and materials to ensure that they understand the clearance process. Other interested parties may include professional associations (e.g. a national association of metal recyclers) and non-governmental organizations. Further recommendations are provided in Section 8.

ORGANIZATION AND IMPLEMENTATION OF THE CLEARANCE PROCESS

2.37. The clearance process should consist of the following:

(a) Defining the roles and responsibilities of the operating organization and the regulatory body (and, where appropriate, technical support organizations and contractors) and establishing adequate human resources in terms of numbers and competence;
(b) Establishing an appropriate procedure for verifying compliance with the clearance criteria;
(c) Establishing an appropriate quality management programme (see para. 2.31);
(d) Making arrangements for the involvement of interested parties, including receivers of cleared materials and objects, prior to implementation of the process (see Section 8).

2.38. Clearance levels could either be defined by the regulatory body (for both generic clearance and specific clearance) or be proposed by the operating organization (for specific clearance of materials or objects). In either case, the clearance levels are required to be approved by the regulatory body (see Requirement 8 of GSR Part 3 [1]). In some cases, a combination of generic clearance levels for some radionuclides (as provided by the regulatory body) and clearance levels proposed by the operating organization for radionuclides not included in the generic clearance levels may be appropriate. A justification should be provided by the operating organization for the use of clearance levels other than the generic clearance levels specified by the regulatory body.

2.39. The clearance process should involve a structured approach both by the regulatory body and by operating organizations. The regulatory body should clearly define the different steps in the process and specify hold points if applicable. Arrangements should be established for timely discussions between the regulatory body and the operating organization as an important part of the clearance process.

APPLICATION OF A GRADED APPROACH TO CLEARANCE

2.40. Paragraph 2.31 of GSR Part 3 [1] states:

> "The regulatory body shall adopt a graded approach to the implementation of the system of protection and safety, such that the application of regulatory

requirements is commensurate with the radiation risks associated with the exposure situation.”

2.41. Requirement 6 of GSR Part 3 [1] states:

“**The application of the requirements of [GSR Part 3] in planned exposure situations shall be commensurate with the characteristics of the practice or the source within a practice, and with the likelihood and magnitude of exposures.**”

2.42. Clearance is an important means of applying a graded approach to the management of materials and objects, achieved by applying a level of regulatory control commensurate with the level of radiological risks. The application of a graded approach to the clearance process should take into account aspects such as the size and complexity of a facility (e.g. a nuclear power plant versus a small research laboratory, decommissioning versus operational activities), the amount of material or the number of objects to be cleared, the level of knowledge of the operational history, the national regulatory framework, and societal and economic factors.

2.43. Monitoring of materials and objects for clearance should be proportionate and sufficient to demonstrate that the requirements of the regulatory body are met. If the provenance and history of a material or object is well known and the levels of radioactivity (due to activation and/or contamination) can be reliably predicted, a less complex monitoring programme (e.g. in terms of the number of samples and measurements or the type) of analysis) may be appropriate. In some cases, it may be sufficient to state that the material has not been activated or contaminated; it may still, however, be necessary to perform limited measurements to confirm this. The process for declaring that material or objects are not radioactive should be documented and, where required by the national framework for clearance, should be subject to regulatory approval.

2.44. The level of effort devoted to quality management, documentation and record keeping should be commensurate with the scope and complexity of the monitoring programme.

3. GENERAL ASPECTS OF CLEARANCE

3.1. The clearance process results in a decision on whether regulatory control can be removed from materials or objects. Storage can be used to take advantage of radioactive decay in order to meet clearance levels (see para. 4.19 of IAEA Safety Standards Series No. GSR Part 5, Predisposal Management of Radioactive Waste [10], and the example provided in Annex IV to this Safety Guide).

3.2. The clearance process should consider the requirements of IAEA Safety Standards Series No. SSR-6 (Rev. 1), Regulations for the Safe Transport of Radioactive Material, 2018 Edition [16]. Table 2 of SSR-6 (Rev. 1) [16] specifies activity concentration limits for exempt material, which are higher than the generic clearance levels specified in tables I.2 and I.3 of GSR Part 3 [1]. Thus, any material that has been cleared on the basis of these generic activity concentration clearance levels will be exempt from the requirements of SSR-6 (Rev. 1) [16].

CONSIDERATION OF CLEARANCE FOR MATERIALS CONTAINING MORE THAN ONE RADIONUCLIDE

3.3. As part of the clearance process, the radionuclide composition of the material should be determined through a process of characterization. The result should be a list of the radionuclides present and their contribution to the total activity concentration or surface contamination level. The different processes that have contributed to the presence of radionuclides in the waste (e.g. nuclear fission, activation by particles, contamination) should also be identified.

3.4. The approach for materials containing more than one radionuclide of artificial origin is described in para. I.14 of GSR Part 3 [1], which states (equation number omitted):

> "For clearance of radioactive material containing more than one radionuclide of artificial origin, on the basis of the levels given in Table I.2 [of GSR Part 3]…the condition for clearance is that the sum of the activity concentrations for individual radionuclides is less than the derived clearance level for the mixture (X_m), determined as follows:

$$X_{\mathrm{m}} = \frac{1}{\sum_{i=1}^{n} \frac{f(i)}{X(i)}}$$

where

$f(i)$ is the fraction of activity concentration of radionuclide i in the mixture;

$X(i)$ is the applicable level for radionuclide i as given in Table I.2 [of GSR Part 3];

and n is the number of radionuclides present."

3.5. As an alternative to the equation in para. I.4 of GSR Part 3 [1], the following formula can be used (weighted summation rule):

$$\sum_{i=1}^{n} \frac{C_i}{CL_i} \leq 1 \tag{1}$$

where

C_i is the activity concentration (Bq/g) or surface contamination level (Bq/cm^2) of the ith radionuclide in the material;
CL_i, which is equivalent to $X(i)$ from para. I.4 of GSR Part 3 [1], is its corresponding derived clearance level in the material (Bq/g) or on the object (Bq/cm^2);

and n is the number of radionuclides present.

3.6. The clearance levels for radionuclides of artificial origin listed in table I.2 of GSR Part 3 [1] take into account dose contributions from relevant progeny radionuclides; thus, only the clearance level of the parent radionuclide needs to be considered (see footnote (a) to table I.2 in GSR Part 3 [1]).

3.7. Paragraph I.12(b) of GSR Part 3 [1] specifies that material containing radionuclides of natural origin can be cleared provided that the activity concentrations do not exceed the clearance levels given in table I.3 of GSR Part 3 [1]. These clearance levels apply to each individual radionuclide in the decay chains of ^{238}U and ^{232}Th, regardless of whether the decay chains are in secular equilibrium. Hence, the approach to mixtures of radionuclides described in para. I.14 of GSR Part 3 [1] is not appropriate for clearance of materials containing radionuclides of natural origin. Instead, the activity concentration of each radionuclide of natural origin should be compared with the clearance levels in table I.3 of GSR Part 3 [1], and if each one is less than or equal to the clearance level then the material can be cleared. For example, for a material containing radionuclides from the ^{238}U decay chain, the clearance level of 1 Bq/g would apply to each radionuclide present. Where the secular equilibrium is significantly disturbed (e.g. owing to thermal processes) or only parts of the decay chain are present, use of the clearance levels given in table I.3 of GSR Part 3 [1] might be too restrictive. In such cases, the regulatory body or operating organization might derive more appropriate values (i.e. specific clearance).

3.8. For clearance of solid bulk material containing a mixture of radionuclides of natural origin and radionuclides of artificial origin, the conditions given in paras I.12(b) and I.14 of GSR Part 3 [1] are required to be satisfied (see para. I.15 of GSR Part 3 [1]). The decision on clearance should therefore contain the following steps:

(1) The summation rule in para. I.14 of GSR Part 3 [1] (or in para. 3.5 of this Safety Guide) is applied to the radionuclides of artificial origin.
(2) The clearance levels in table I.3 of GSR Part 3 [1] are applied to each radionuclide of natural origin.
(3) If both the radionuclides of artificial origin and the radionuclides of natural origin meet the clearance criteria, then the material can be cleared. If the radionuclides of artificial origin or the radionuclides of natural origin fail to meet the clearance criteria, then the material cannot be cleared.

CHARACTERIZATION OF MATERIAL FOR CLEARANCE PURPOSES

3.9. The objective of radiological characterization of material to be cleared is to provide information on the radionuclides and their quantities, spatial distribution, physical states and chemical properties. The characterization results should be used by the operating organization to identify the material (or part thereof) to be cleared and to select the optimum monitoring programme for demonstration of compliance with the clearance levels. The characterization results should also be used to assess various options for the clearance of the material, for example (a) the use of batch monitoring tools and techniques; (b) the destinations for the cleared material; (c) the application of either generic clearance or specific clearance measures for the protection of workers, the public and the environment; and (d) economic factors. The level of detail and the implementation of the proposed steps should be proportionate to the complexity of the situation, in accordance with a graded approach.

3.10. Characterization involves a logical and systematic approach. A comprehensive characterization programme comprises the following steps [17]:

(1) Review of historical information, including process knowledge of the material;
(2) Activation and decay calculations, as appropriate;
(3) Preparation of the sampling and analysis plan, based on an appropriate statistical approach and taking into account the information from step (1);
(4) Performance of measurements, sampling and analyses;
(5) Review and evaluation of the data obtained from the monitoring programme;
(6) Comparison of calculated results and measured data.

3.11. Characterization should be considered an iterative process, taking into account possible alterations in the radionuclide composition, for example due to decontamination or dismantling activities.

3.12. As indicated in para. 3.9, a graded approach should be applied to the characterization of material for clearance purposes. For complex situations, the characterization process could involve collecting information on the following:

(a) The location and type of the originating facility or activity, the operational history (including incidents and post-incident remediation), the origin of the material within the facility or activity, and the radionuclides associated with operations (see also paras 3.14–3.17);
(b) The size, type and quantities (total and rate of production) of material;
(c) The radionuclides present in the material and the expected levels of contamination or activation;
(d) The nature of the contamination (i.e. fixed or non-fixed surface contamination, or bulk contamination);
(e) The distribution of contamination (including identification of hotspots on the surface or within the volume);
(f) Other hazards associated with the material;
(g) The time frame for the clearance process and the necessary clearance monitoring throughput.

Further information on characterization is provided in Ref. [18].

3.13. The characterization process could generate a large amount of data in different formats (e.g. paper records, drawings, digital information such as spectra, spreadsheets), and therefore the operating organization should have suitable records and a suitable data management system, which should be part of the overall management system. Examples of such systems to support decommissioning are described in Refs [19, 20].

Historical information on material for clearance purposes

3.14. Where appropriate (as indicated by the application of a graded approach), detailed information on the history of the material to be cleared should be collected as the first step in the characterization process. This information should be used to develop the other steps in the characterization process. Information should be obtained from various sources, such as historical records, knowledge of the types of process involving the material, experience gained elsewhere, public or institutional memory, and recollections from workers.

3.15. The historical information might include the following:

(a) A description of the facility and equipment, the processes or activities during the operation of the facility, and the type and form of the radioactive material used during operations;
(b) The location of controlled areas, supervised areas and undesignated areas, including their changes over time, and whether the radioactive material was kept within specific areas;
(c) Whether the material has been potentially activated by neutron exposure or by photonuclear reactions, and the time period over which this might have occurred;
(d) Whether the material has been contaminated as a consequence of an accident or spill, and when this might have occurred;
(e) Whether the facility or equipment has been refurbished or modified;
(f) Whether the facility, equipment and areas have been decontaminated;
(g) The results of any past characterization or monitoring of the material.

3.16. Establishing the historical information relevant to the material to be cleared may be straightforward for most facilities and activities. However, it might be more complicated for research facilities in which different activities, such as experiments and novel chemical processes, were performed or in facilities for which information on the plant history is lacking and for which no similar facilities can be used as reference. Where detailed historical information is not available, as is often the case for remediation activities or for old facilities, a greater emphasis should be placed on the characterization programme.

3.17. The information on the history of the material to be cleared should be used to determine an initial estimate of the radionuclide composition of the material, and this initial estimate should be used to implement steps (2), (3) and (4) described in para. 3.10 (activation and decay calculation; sampling and analysis plan; and measurements, sampling and analysis). Initial measurements (e.g. dose rate, radiation type, surface contamination) should be performed to provide additional information to guide the preparation of the sampling plan (see paras 3.18–3.20).

Sampling of material for clearance purposes

3.18. Steps (5) and (6) described in para. 3.10 (review and evaluation of the monitoring data, and comparison of calculated results and measured data) should be performed as early as possible and be used to provide feedback to the process of sampling and analysis. The characterization plans may change as a result of these ongoing assessments, for example where contamination is more (or less) extensive than originally anticipated or where trends in measurement results indicate that the original sampling plan will not provide the information necessary for clearance. The historical information may also need to be reviewed if additional radionuclides are identified in steps (2), (3) and (4) described in para. 3.10; consequently, the characterization process should be viewed as an iterative process. One of the important outputs from the characterization process is a credible radionuclide composition (or several possible compositions) for the material.

3.19. Two main types of measurement are relevant for the characterization of solid materials for clearance purposes: (a) measurements of surface contamination (fixed or removable) based on alpha, beta and/or gamma measurements and (b) bulk activity measurements, which are generally based on gamma spectrometry or total gamma measurements but can also include alpha and beta measurements (e.g. in cases where there is a uniform activity concentration). In each case, the methods of measurement should take into account the geometry, the surface conditions, and the nature, extent and distribution of the radioactive contaminants. It is unlikely that dose rate measurements alone will provide useful information for characterization for clearance, except in cases where a reliable relationship between dose rate and radionuclide composition has already been established. Further information on in situ measurement techniques is available in Refs [18, 21, 22].

3.20. As part of the sampling and analysis programme, representative samples should be taken from the material to be characterized. The application of sampling and analysis techniques should determine the radionuclides and their activity in selected locations. Further information on sampling and analysis techniques is available in Refs [17, 18, 21–23].

Establishing the radionuclide composition

3.21. Material for clearance may contain more than one radionuclide, and some of these radionuclides may be difficult to measure routinely during the clearance process. The information obtained from the historical review and the calculations can be used to determine an initial estimate of the radionuclides expected to be present and the ratios (also called scaling factors[4]) between the different radionuclides [23, 24]. Then, a limited number of detailed measurements can be used to determine whether difficult to measure radionuclides occur in a consistent ratio with easy to measure radionuclides. If this is the case, scaling factors can be used to estimate the activity of difficult to measure radionuclides based on the measurements of the easy to measure radionuclides. An example is the use of measured ^{60}Co activity to assess and monitor a wide range of difficult to measure radionuclides present in activated materials or in corrosion products associated with the operation of nuclear reactors.

3.22. Scaling factors for difficult to measure radionuclides should be used with caution and reviewed at an appropriate frequency. In some facilities, one set of scaling factors can be applicable over a large area, whereas in other facilities the radionuclide composition may vary considerably over space and time, and for different materials, particularly where chemical processes or decontamination procedures have taken place. Therefore, scaling factors should be based on information such as that described in para. 3.21, rather than applying values already determined (i.e. from other clearance batches or from the evaluation of low level radioactive waste). Radionuclide composition will also vary in cases where the radioactivity is generated by neutron activation of impurities in the material and the concentration of such impurities varies significantly (e.g. variations in the content of cobalt in steel).

3.23. The selection of the radionuclides that are significant enough to be evaluated for clearance is a screening process that should, in the first instance, be based on an

[4] Factors or parameters determined from sampling and analysis data and used in calculating the activity of difficult to measure radionuclides on the basis of measured radioactivity of easy to measure radionuclides.

initial estimate of the activity concentrations of the radionuclides in the material. If there are large uncertainties associated with this initial estimate, a larger number of radionuclides may initially be selected as potentially being significant. The number may subsequently be reduced when more reliable estimates of the activity concentrations are obtained, for example from the monitoring programme.

3.24. All radiation monitoring equipment has a response that depends on the type and energy of the radiation and on the detector–material geometry. The response of such equipment should be calculated for the radionuclide composition of the material. This involves the selection of key radionuclides to be measured based on the properties of their radiation emissions, the ease and efficiency with which they can be detected (particularly whether the necessary limit of detection can be achieved) and their contribution to the summation rule applied for clearance. Although it is preferable to select those radionuclides that have the biggest impact on the clearance of the material, in many cases it will be necessary to select other radionuclides because they are easier to measure. An example of a method for the selection of significant radionuclides for solid materials is provided in Annex VI.

MANAGEMENT OF THE CLEARANCE PROCESS

3.25. Paragraphs 3.26–3.35 provide recommendations on the management of the clearance process in situations where it is a regular process and the material throughput is substantial (e.g. during decommissioning of a nuclear power plant, where thousands of tonnes of material might be cleared). Some of the recommendations are still valid for smaller quantities of cleared material and should be applied in accordance with a graded approach.

3.26. The operating organization should ensure that staff implementing the clearance process are clearly identified and are suitably qualified, properly trained and competent for their roles. The number of staff should be commensurate with the quantities of material to be handled and the capacity of the associated monitoring programme.

3.27. Additional staff may be needed to record information on material undergoing the clearance process, for example by updating databases on the material and maintaining documentation. Staff may also be needed to ensure continued movement of the material through the steps of the clearance process and segregation of material that has been cleared.

3.28. The clearance process should be allocated suitable and sufficient equipment to implement the monitoring programme, as well as any equipment needed to handle the material. Additional equipment may be needed for determination of non-radiation-related characteristics of the material, for example the presence of free liquid and/or dust. The area where clearance measurements are being performed should be cleaned prior to use and should have a low radiation background, to the extent possible.

3.29. A prerequisite for the clearance process is the radiological characterization of the material (see paras 3.9–3.24). Sometimes, it is not possible to fully characterize a material or an object prior to the dismantling of a component. In that case, the characterization needs to be finalized after the dismantling and the results need to be available prior to the final measurements for compliance with clearance levels.

3.30. The results of the characterization should serve as the basis for defining appropriate batches of material in the clearance process. Processing batches of materials with similar characteristics enables the clearance process to be more efficient, for example because the settings and operation of monitoring equipment would also be similar.

3.31. A prerequisite for a high throughput clearance process is the availability of a database system to store information on the identification and location of materials and the results of the clearance measurements. Such a system should be kept updated to reflect the current situation.

3.32. The clearance process for materials is most effectively implemented if there are clearly assigned areas for material transfer, buffer storage, surface contamination measurements and activity concentration measurements, as well as staging areas where cleared material can be placed until it can be removed from the facility.

3.33. The following description refers to an idealized clearance process for solid materials, which might be appropriate, for example, for a large scale decommission operation. In other situations, individual steps can be omitted or performed in a different sequence:

(1) The material is transferred from its place of origin (e.g. an area in the facility where dismantling, segmentation and decontamination are taking place) to a buffer storage area. Material that has been segmented into pieces is usually moved in boxes.

(2) In the buffer storage area, the materials are sorted into batches depending on their origin and characteristics, in particular the part of a plant from which the material originates, the operation history, the material's radiological properties, and other characteristics. Batches entering the clearance process will then consist of material with similar characteristics. Sorting materials into batches should be done earlier (e.g. at the place of generation or in other dedicated areas) if radiological and other conditions allow. Sorting the material earlier (e.g. during dismantling) is essential to avoid cross-contamination and dilution (i.e. mixing of materials that can be cleared with materials that cannot be cleared).

(3) The surface contamination on accessible surfaces is measured, including, where possible, the inner surfaces that could become accessible in the subsequent use of the material following clearance. There should be a dedicated area for this purpose. The individual parts should be put on tables or racks where these surfaces can be accessed for contamination monitoring.

(4) The results of the surface contamination measurements are evaluated against clearance levels for surface contamination (if agreed with the regulatory body (see paras 4.17–4.22)), taking into account the averaging area, radionuclide composition and any other specifications of the process. If the surface contamination clearance levels are complied with, the material can be moved to the next station; if not, additional decontamination may be necessary, and the material is sent to a dedicated area for further treatment or for management as radioactive waste. In some cases, the results of the surface contamination measurement may be sufficient to demonstrate compliance with activity concentration clearance levels (in which case, steps (5) and (6) can be omitted).

(5) Following demonstration of compliance with the surface contamination clearance levels (if this step is included), the material is moved to the next buffer storage area, awaiting measurements for determination of the activity concentrations.

(6) The bulk activity is determined. In cases where the percentage of gamma emitting radionuclides is sufficiently high, bulk monitors based on gross gamma counting or drum monitors based on gamma spectrometric measurements can be used for this step. In other cases, the bulk activity is determined from the analysis of samples, from surface measurements or from other measurement methods.

(7) The results of these measurements are evaluated against the activity concentration clearance levels, taking into account the averaging mass, radionuclide composition and other specifications of the process. If the activity concentration clearance levels are complied with, the material has

successfully passed all measurements; if not, alternative waste management options should be considered.

(8) Before the material is released from the facility, verification measurements by or on behalf of the regulatory body may be necessary. In such cases, the material is brought to a further buffer storage area outside or at the border of the controlled area or supervised area, where these verification measurements are performed. If compliance with clearance levels is verified, the material is cleared. Otherwise, the material stays under regulatory control and other waste management options are considered.

(9) Cleared material is moved to a place where it can be handed over to a conventional waste management company (e.g. a scrap dealer, a recycler of building rubble) in accordance with any conditions that have been attached (i.e. for specific clearance).

(10) Before proceeding with the processing of the next batch of material, the buffer storage area is checked for the presence of any contamination.

(11) Once a batch of material has completed the clearance process, the database and the documentation are updated and archived accordingly.

3.34. Practical experience from decommissioning projects involving clearance of large amounts of material has shown that the following considerations are beneficial for effective implementation of the clearance process:

(a) Moving the material in suitable containers such as boxes (e.g. 1 m^3) or drums (e.g. 200 L), instead of as single items, ensures that material of similar origin is kept together, that the material can be traced easily via the identifier of the box and that the bulk measurements can be performed directly on these containers.

(b) Providing buffer storage areas of sufficient size between the various steps of the clearance process enables the flow of materials to be maintained even if there are delays (e.g. due to temporary unavailability of a monitoring instrument) during one of the steps.

(c) Having separate buffer storage areas between the individual steps avoids unintentional mixing of material or cross-contamination between steps and prevents material from missing a step in the process and being unintentionally cleared. The buffer storage areas also facilitate segregation of the material in accordance with its origin, material type, radionuclide composition and other criteria.

(d) Ensuring traceability of the material at all times and maintaining thorough documentation of the results of each step reduces the likelihood of an erroneous clearance decision being taken and ensures that the clearance decisions can be reviewed and understood, even many years later.

(e) Undertaking measurements in areas with a low background dose rate enables high quality measurements, resulting in decision thresholds that are appropriately below the clearance levels.

3.35. If a facility is too small to provide adequate space for the clearance process and if no area with sufficiently low background radiation is available, it might be better to construct an appropriately designed separate facility where the process can be implemented. The levels of radioactivity in material undergoing the clearance process are expected to be of a similar order to the clearance levels (this should be ensured by an adequate characterization performed prior to this step). Consequently, the clearance of material should pose a low radiological risk, even if some of the material does not actually comply with the clearance levels. Such separate facilities may be of simple design without extensive provisions for shielding or ventilation. The use of a separate facility for clearance can also contribute to reducing non-radiation-related risks by separating activities relating to clearance from the facility where other activities are performed.

4. CLEARANCE OF SOLID MATERIAL

4.1. Activity concentration clearance levels (Bq/g) are usually applied for clearance of solid materials that contain radionuclides throughout their volume, such as activated metal components, contaminated soil and building rubble. A surface contamination clearance level (Bq/cm^2) can additionally be applied to items contaminated on their surfaces. An example of a national approach to the clearance of scrap metal is presented in Annex V.

4.2. The characterization and management of the clearance process for solid materials should follow the recommendations provided in Section 3. This section provides recommendations on the following aspects that are specific to solid materials:

(a) The activity concentration clearance levels and surface contamination clearance levels that can be applied;
(b) The approach to averaging and to situations where mixing is part of the material management process after clearance;
(c) The implementation of clearance measurements and related uncertainties (see Annex VII).

ACTIVITY CONCENTRATION CRITERIA FOR GENERIC CLEARANCE

4.3. The activity concentration clearance levels specified in tables I.2 and I.3 of GSR Part 3 [1] apply to solid materials, for example contaminated or activated structures and components, or contaminated soils.

4.4. The methodology used to calculate the clearance levels for artificial radionuclides in table I.2 of GSR Part 3 [1] is described in Ref. [5]. For each radionuclide of artificial origin, the activity concentration clearance level was determined on the basis of a set of exposure scenarios, each of which considered external irradiation, dust inhalation and ingestion (direct and indirect, including ingestion of radionuclides via drinking water and water for agricultural purposes). The clearance levels listed in table I.2 of GSR Part 3 [1] are the lower of the values obtained from the following approaches:

(a) The use of ‘realistic’ scenarios, applying an effective dose criterion of 10 μSv in a year;
(b) The use of ‘low probability’ scenarios, applying an effective dose criterion of 1 mSv in a year and a skin equivalent dose limit of 50 mSv in a year.

The clearance levels derived from the calculations were then rounded to the nearest power of ten using a near logarithmic rounding approach [5]; the models and assumptions used do not justify higher precision. Consequently, demonstrating that the resulting dose will be of the order of 10 μSv in a year or less should be commensurate with this level of precision (i.e. with the logarithmically rounded values of the clearance level).

4.5. The clearance levels for artificial radionuclides in table I.2 of GSR Part 3 [1] are applicable for materials that may be incinerated, since scenarios relevant to incineration were taken into account when deriving these clearance levels [5]. The clearance levels specified in table I.2 of GSR Part 3 [1] also consider possible processes that could lead to increase of concentrations of radionuclides in the material.

4.6. The clearance levels for artificial radionuclides in table I.2 of GSR Part 3 [1] are not applicable to foodstuffs, drinking water, animal feed or any material intended for use in food or animal feed. In addition, in some cases, these clearance levels might not be appropriate for very large quantities of material. For example, in the case of excavated soil, the model used in Ref. [5] to derive these clearance levels assumes a dilution factor that might not be possible or permissible for very large quantities. In such cases, more specific models should be developed and specific clearance levels derived for application.

4.7. Activity concentration clearance levels are specified in table I.2 of GSR Part 3 [1] for over 250 radionuclides of artificial origin. Values for other radionuclides of artificial origin should be derived using the models for radionuclides of artificial origin described in Ref. [5]. Examples of values for other radionuclides can also be found in national regulations, for example Refs [25, 26].

4.8. A scenario based approach was not used in the case of material that contains radionuclides of natural origin. Instead, the activity concentration clearance levels specified in table I.3 of GSR Part 3 [1] were established using a pragmatic approach that involved consideration of the worldwide distribution of the concentration of radionuclides of natural origin present in material found in the environment. The values are applicable to all radionuclides of natural origin in the ^{238}U decay chain and the ^{232}Th decay chain and for ^{40}K. The same pragmatic approach should be used to determine the activity concentration clearance levels for other primordial radionuclides (e.g. ^{87}Rb, ^{138}La, ^{147}Sm, ^{176}Lu) in situations where these radionuclides are of interest for clearance.

4.9. The methodology in Ref. [5] focuses on the handling (i.e. transport, trade, use and disposal) of material outside the facilities in which it arises (e.g. nuclear reactors, particle accelerators, research laboratories). The scenarios used to derive clearance levels for radionuclides of artificial origin consider a decay time before the start of the exposure, which is assumed to be at least one day (or considerably longer for some scenarios). Therefore, the methodology used in Ref. [5] is not suitable for calculating activity concentration clearance levels for very short lived radionuclides (i.e. with half-lives of a few hours or less), unless scenarios without a significant decay time prior to start of the exposure are added. An alternative approach is described in para. 4.11. If direct handling could be avoided or decay storage for several days or weeks is provided before clearance of materials containing very short lived radionuclides, that may eliminate the need for such considerations.

4.10. When direct handling of moderate quantities of material after clearance is considered without significant decay, the exemption levels given in table I.1 of GSR Part 3 [1] are also applicable for clearance, since no decay or waiting time was assumed when deriving these exemption levels, and the same dose criteria were applied for both exemption and clearance [1].

4.11. For short lived radionuclides for which activity concentration exemption levels are specified in table I.1 of GSR Part 3 [1] but for which there are no clearance levels specified in table I.2 of GSR Part 3 [1], the following alternative approach could be taken:

(a) Use the methodology from Ref. [5] for radionuclides of artificial origin to obtain activity concentration values that meet the clearance criteria for direct handling;
(b) Identify the corresponding activity concentration exemption levels for moderate quantities of material from table I.1 of GSR Part 3 [1] that meet the clearance criteria;
(c) Take the lesser of the two results from (a) and (b) as the clearance level.

CONSERVATISM IN THE DERIVATION OF GENERIC ACTIVITY CONCENTRATION CLEARANCE LEVELS

4.12. The derivation of generic clearance levels, as described in Ref. [5], includes a number of conservative assumptions that were deliberately taken to encompass a large variety of exposure scenarios that could arise after clearance of any type of material. Many individual parameter values were chosen conservatively, for example the following:

(a) In many ‘low probability’ scenarios, completely bounding values were assumed for the following:
    (i) Exposure times (8760 hours for the full year, 1800 hours for the full working year);
    (ii) Dilution (a factor of 1 (i.e. no dilution));
    (iii) Decay time prior to and during the scenario (one day and no days, respectively, corresponding to virtually no decay at all);
    (iv) Unfavourable exposure conditions (as for the water pathway considerations in Ref. [5]).
(b) The groundwater model contains a number of conservative assumptions, such as the following:
    (i) The whole inventory of radionuclides in the material is available for migration.
    (ii) The K_d values have been selected conservatively from the values published in literature for different elements.
    (iii) The private well from which groundwater is abstracted for several uses is very close to the deposited material, thus significantly reducing the effect of radioactive decay.

(c) With regard to exposure from skin contamination, the dose coefficients were based on a skin surface weight of 4 mg/cm^2. In practice, contamination would predominantly occur on the hands, where the skin surface weight is significantly higher.

4.13. Methods have been applied to limit the overall degree of conservatism, as follows:

(a) Two sets of scenarios were used in parallel, one applying 'realistic' scenarios for an individual effective dose criterion of the order of 10 µSv in a year, and one applying 'low probability' scenarios for an individual effective dose criterion of 1 mSv in a year. In this way, parameter values for the 'realistic' scenarios could be less conservative. This approach meets the criteria for clearance in paras I.10 and I.11 of GSR Part 3 [1].
(b) The scenarios for workers and members of the public were formulated in such a way that exposure pathways that could occur simultaneously (e.g. external irradiation and inhalation) were analysed together and their dose contributions added. This allowed the necessary conservatism in the model to be applied over the sum of exposure pathways rather than applying it to each pathway individually, thus reducing the overall amount of conservatism in the model.

4.14. Notwithstanding the methods described in para. 4.13, there are other reasons why clearance levels are conservative, as follows:

(a) The application of the summation rule for cases where there is more than one radionuclide present is inherently a conservative approach since the exposure pathways for the representative person are not necessarily the same for each radionuclide, for example because of partitioning or separation of radionuclides by processes. A less conservative approach would be to sum the contributions of the radionuclides in the radionuclide mixture for each scenario and each exposure pathway first and then derive the activity concentration value.
(b) The dose contribution from progeny radionuclides is always included together with the parent radionuclide with a percentage that corresponds to the highest ingrowth within a time span of 100 years after clearance. This leads to a slight overestimation of the dose coefficient for the mixture of parent and progeny radionuclides.

4.15. The use of clearance levels in practice should take into account the conservatism that has already been applied in their derivation. It is recognized that the clearance levels have been derived on a sufficiently conservative basis while avoiding being overly conservative [5]. Their implementation in practice could take into account this model-intrinsic conservatism to avoid the imposition of further conservatisms commensurate with the degree of conservatism in the model. The particular case and the requirements of the national regulatory framework should also be considered.

4.16. That clearance levels have been derived conservatively should be considered in the implementation of the clearance process, for example by using larger averaging areas or averaging masses.

SURFACE CONTAMINATION CLEARANCE LEVELS

4.17. For surface contaminated items where the radioactivity is concentrated on surfaces, compliance with activity concentration clearance levels (Bq/g) might not be sufficient in all cases because there are additional considerations relating to the handling of the material. In such cases, surface contamination clearance levels could be specified, either for generic clearance (which may be specified by the regulatory body in accordance with para. 3.12 of GSR Part 3 [1]) or for specific clearance (which may be granted by the regulatory body in accordance with para. I.13 of GSR Part 3 [1]).

4.18. Surface contamination clearance levels could be derived by the operating organization and reviewed and approved by the regulatory body. Alternatively, surface contamination clearance levels could be specified by the regulatory body as part of the regulatory framework for clearance. The operating organization should then comply with these surface contamination clearance levels, in addition to complying with the generic activity concentration clearance levels[5].

4.19. Surface contamination clearance levels are intended to limit the contamination that is directly accessible and could be mobilized during handling of the material. They also limit direct exposure by external irradiation from handling surface contaminated items. The radioactivity inside and on the surface of the cleared material has to be appropriately limited to guarantee compliance with the dose criterion of the order of 10 μSv in a year stated in para. I.11 of GSR Part 3 [1]. An example of the application of surface contamination and activity concentration clearance levels is given in Table 1 [27].

[5] In some cases, compliance with activity concentration clearance levels can be inferred from measurements of surface activity (see para. 4.44).

TABLE 1. EXAMPLE OF APPLICATION OF SURFACE CONTAMINATION AND ACTIVITY CONCENTRATION CLEARANCE LEVELS TO A SURFACE CONTAMINATED ITEM
(adapted from Ref. [27])

Surface contamination clearance levels[a] (Bq/cm^2)	Activity concentration clearance levels[b] (Bq/g)	Recommended action
Average is below the clearance level	Average is below the clearance level	No need to undertake separation and segregation prior to clearing waste.
Average is above the clearance level	Average is below the clearance level	Separation and segregation should be undertaken unless a justification can be made that removal is not reasonably practicable, the expenditure (whether in time, trouble or money) is grossly disproportionate to the safety and environmental benefits gained, and the overall impact of disposal is less than of the order of 10 μSv in a year.
Average is below the clearance level	Average is above the clearance level	Unless commercial considerations (e.g. recycling or reuse options) for the surface layer are sufficient to justify the safety and environmental impacts of separation and segregation, it would be expected that articles or substances in this configuration would be managed as radioactive waste in accordance with the national strategy for management of radioactive waste.
Average is above the clearance level	Average is above the clearance level	Manage as radioactive waste in accordance with the national strategy for management of radioactive waste.

[a] For example, paint, laminate or region of increased radionuclide concentration.
[b] For example, brick, blockwork or metal structure.

4.20. A number of international studies have used models to establish a link between surface contamination and the resulting annual dose to an individual. Examples of such models are given in Annex I. References [28, 29] provide surface contamination clearance levels for metallic items both for direct reuse and for recycling by melting. Examples of surface contamination clearance levels applicable for general use are given in Annex II. In general, the derivation of these surface contamination clearance levels considers both the fixed and removable activity on the surface of items.

4.21. The values of surface contamination clearance levels for certain radionuclides, as derived by different international studies and recommendations, can differ. The differences are due to different assumptions in the models used to derive the clearance levels (e.g. in relation to the material, the size of an item, geometry, or exposure scenarios). Therefore, application of a set of surface contamination clearance levels (derived for a specific situation) to a different situation should be done with care, taking into account the adequacy of any assumptions, the characteristics of the material or items, and the exposure scenarios used.

4.22. If surface contamination clearance levels have not been specified, items with surface contamination might instead be considered in terms of their compliance with activity concentration clearance levels. This can be achieved by converting the total activity on the surface to an activity concentration (Bq/g), taking account of the total mass of the material below the surface (i.e. the activity concentration should not be calculated by just using the thickness of the layer of surface contamination). In doing this, considerations relating to the models used to derive the clearance levels need also be taken into account, particularly the averaging mass (see paras 4.23–4.37). For example, if a metal sheet with a thickness of 0.8 cm and a density of 7.8 g/cm^3 has a surface contamination level of 0.4 Bq/cm^2 of ^{60}Co on one side, the average activity concentration is calculated as 0.064 Bq/g. The relevant clearance level (as specified in table I.2 of GSR Part 3 [1]) is 0.1 Bq/g. Consequently, the material would comply with the activity concentration clearance level; however, the clearance level would not be met if both sides of the metal sheet were contaminated at this level.

AVERAGING MASSES AND AREAS FOR CLEARANCE PURPOSES

4.23. The generic clearance levels specified in table I.2 of GSR Part 3 [1] for artificial radionuclides were calculated using scenarios involving exposure to a large quantity of homogeneous material. For example, the transport scenario considers a truck containing 10 tonnes of material, and the landfill scenario considers even larger quantities [5]. When applying the clearance levels, it should be recognized that they were derived for these large amounts and that the averaging should be done accordingly, with due consideration of the exposure scenarios. Hence, very small averaging masses are not appropriate for large amounts of material.

4.24. The regulatory body should determine or approve appropriate averaging masses to be used in the clearance process, in particular for compliance measurements; the averaging procedures used by the operating organization should take these averaging masses into account. Examples of appropriate averaging masses are from a few hundred kilograms to the order of a tonne. The regulatory body should confirm that the averaging procedure is not used solely to allow material that contains radioactivity above the clearance levels to be cleared. The operating organization should ensure that averaging procedures, selected in accordance with the type of material, are an integral part of the clearance process. For small objects with a mass less than the specified averaging mass, a minimum default averaging mass (e.g. 1 kg) could be defined by the regulatory body, which in turn allows the specification of a maximum total activity for these objects for a specified radionuclide (e.g. 100 Bq for ^{60}Co) or radionuclide composition. It should be ensured that this does not result in large objects being divided into smaller ones solely as a means of achieving clearance. In the case of several small objects, an alternative method would be to monitor them together to replicate the averaging mass.

4.25. In the case of surface contamination clearance levels, the regulatory body should consider, depending on the type of material, and the nature and homogeneity of the contamination, averaging areas from several hundred square centimetres up to 1 m^2. In some cases, averaging areas for specific clearance could be higher, for example up to 10 m^2 for the clearance of buildings for demolition.

4.26. For non-accessible surfaces that might be expected to be contaminated to some degree, the operating organization should make a conservative assessment of the surface activity for comparison with the clearance levels. Similarly to the clearance of small and light objects on the basis of activity concentration clearance levels, the regulatory body could define a default minimum averaging surface area (e.g. 100 cm^2) for objects with surfaces below the default averaging area. If there are several of these small objects, an alternative method would be to combine measurements of the surfaces of individual objects to compare against clearance levels that are to be applied to a larger average surface area.

4.27. Averaging masses and areas for decision making on compliance with clearance levels should be distinguished from masses and areas used for actual measurements. For example, multiple samples of 100 g of soil could be used to determine whether a mass of a few tonnes complies with the clearance levels.

In any case, the masses or areas used in measurements should not exceed the averaging masses and areas for decision making on clearance.

4.28. In deciding on a measurement strategy, the operating organization should batch the material so that it is as homogeneous as possible in relation to both material and origin, and thus in relation to the radionuclide composition and activity level. Variations in the results of individual measurements within one averaging mass or area should be expected. For example, variations in individual results of up to a factor of ten (compared with the average value) are generally considered acceptable, whereas a greater variation would be acceptable if the overall average concentration or surface contamination level was a very small fraction of the clearance level.

4.29. The operating organization should make use of the maximum permitted averaging areas or masses when designing the monitoring programme for clearance, as this improves the efficiency of the clearance process. The monitoring programme may be constrained by the form and nature of the contamination; for example, the choice of equipment available for monitoring for beta activity inside a small pipe is likely to be limited. Nevertheless, using appropriate time integration in dynamic measurements (e.g. recording counts over a minute, rather than over a second) or numerically averaging over a number of single static measurements will enable a greater averaging area to be achieved.

4.30. Measurements to determine whether a material or object is in compliance with clearance levels will be based on a measurement unit defined by the monitoring method and the instrumentation used (e.g. contamination monitor, drum monitor, bulk monitor). The size of this measurement unit should be based on practical considerations that reflect the size of an object or sample of material (e.g. a drum of waste, an excavator bucket) and how the measurement will be made (e.g. the geometry of the measurement system).

4.31. The operating organization should select measurement units and should propose averaging masses and areas that are sufficiently representative of the material, taking into account homogeneity and the necessary detection levels and confidence intervals for the results of clearance measurements. In general, larger measurement units and averaging masses and areas are acceptable where the contamination is reasonably homogeneous. The averaging masses and areas should be agreed with the regulatory body and formally recorded by the operating organization as part of the clearance process. The regulatory body should also provide guidance and quantitative criteria on how the clearance process should address issues associated with inhomogeneity.

4.32. The averaging masses and areas will indirectly limit the activity of hotspots. The regulatory body should establish limiting criteria for such hotspots when defining the size of averaging masses and areas (see also paras 4.34–4.37).

4.33. If the results of samples taken from a material are subject to considerable variability, as described in appendix A to Ref. [30], then averaging over the whole material mass is unlikely to be acceptable without a properly documented consideration of the following:

(a) The practicability of separation and segregation of parts of the material;
(b) Suitable revision of the sampling and monitoring plan, including numbers of samples;
(c) Suitable reduction of the size of each measurement unit (mass or volume);
(d) The practicability of making further measurements to identify each volume or area containing significantly elevated levels of radioactivity;
(e) The practicability of removing or segregating small areas or volumes containing significantly elevated levels of radioactivity;
(f) The potential radiological significance of inhomogeneity.

The aspects listed should be considered when planning activities that give rise to the materials, such as decommissioning.

The effect of hotspots and the distribution of activity with depth and area on clearance measurements

4.34. One of the most challenging tasks in the removal of regulatory controls from materials and objects is to ensure that the presence of hotspots is taken into account in an appropriate manner. It is important to distinguish between hot particles and hotspots, where the latter are due to inhomogeneity. Hot particles are generally small items that are not part of the material in which they are found, for example small metal flakes with a high ^{60}Co activity or small pieces of spent nuclear fuel found in a cooling pond. Such hot particles can give rise to doses that can lead to deterministic effects and should be removed before the clearance process begins. The potential for hot particles should be considered and, if found to be possible, it should be ensured that the clearance process will identify their presence, rather than just considering them as contributors to the total activity of the averaging mass or area.

4.35. Local non-uniform distribution of activity that results in activity concentrations above clearance levels is to be expected. It is important that the variations of activity concentrations be reasonably restricted. Usually, variations of up to ten times are tolerated. Parts of the materials or surfaces with activity concentrations significantly higher than the clearance levels (i.e. hotspots) that were identified during characterization should be considered for removal (decontamination) prior to dismantling or demolition of structures or components. The regulatory body should approve or specify additional monitoring criteria to the existing averaging criteria in order to detect and manage any hotspots in or on materials considered for clearance.

4.36. In cases where compliance with surface contamination clearance levels is demonstrated using instruments with a much smaller detector surface area than the averaging area, information on homogeneity, and hence the presence of hotspots, can be derived from the variation in individual measurement results. The final value of surface contamination for comparison with clearance levels should be based on the averaging area. It may be possible to set an alarm on the monitoring instrument to help identify hotspots.

4.37. Many processes involving bulk measurement are based on scanning or multipoint measurements, both of which can be set up to identify elevated levels of activity. Another approach to demonstrating compliance with hotspot criteria [31] is to use measurement techniques sensitive enough to detect the 'worst case' activity levels. For example, if measurements are taken on the outside of a drum, calculations to demonstrate compliance could assume that all the contamination is located in the centre of the drum (surrounded by clean material), that is, furthest from the detectors and shielded by the clean contents. This will result in increased measurement resources (e.g. longer counting times, more measurements, more sensitive detectors), but the additional cost may be small compared with undertaking additional monitoring to demonstrate compliance with criteria for both the average mass and hotspots within the volume of the material. This approach works well for a drum where the density is low and the radionuclides emit high energy gamma radiation (e.g. ^{60}Co in concrete rubble) and where the activity concentration is well below the clearance level. This approach is not suitable if the material itself provides significant shielding and the gamma energy is lower (e.g. metal contaminated with ^{241}Am) and the activity concentration is close to the clearance level.

Monitoring programme and strategy for clearance

4.38. The monitoring programme to support the clearance process should be based on the results of the previously performed characterization, as described in paras 3.9–3.24. The monitoring programme should be managed in accordance with a material flow process that starts with well characterized material to be evaluated for clearance.

4.39. As recommended in paras 3.30, 3.33 and 4.28, material considered for clearance should be sorted into batches consisting of the same type of material, the same radionuclides and the same history. In the definition or selection of batches, a distinction should be made between bulk contaminated material and surface contaminated material.

4.40. Within the monitoring programme, a distinction should be made between the monitoring strategy and the monitoring technique. The monitoring strategy relates to the batch process itself, whereas the monitoring technique (i.e. surface measurement, bulk measurement or sample analysis) is the tool within the monitoring strategy to facilitate decision making on clearance of a batch. The monitoring strategy should take into account the input material in the batch process and the output options (i.e. whether the material is cleared or is selected for other waste management options). The optimal strategy should be based on occupational exposure, public exposure, protection of the environment and economic factors.

4.41. The monitoring strategy should determine which monitoring techniques are the most appropriate for a given batch. Depending on the material being considered, a combination of techniques can be used.

4.42. The choice of monitoring technique involves the selection of radiation measurement equipment. The response of the equipment will depend on the radiation type and energy and on the detector–source geometry. A good knowledge of the radionuclides to be measured should be obtained prior to establishment of the monitoring programme. Where appropriate, key radionuclides should be defined in the radionuclide mixture and the contribution of other radionuclides can be assessed by the use of scaling factors, as described in paras 3.21 and 3.22. On the basis of this information, an appropriate radiation measurement instrument should be selected for clearance monitoring, taking into account the value of the clearance level (in terms of activity concentration or surface contamination level) that has to be verified. Information on the selection of the instrument can be found in Ref. [18]. The personnel responsible for the selection of monitoring techniques and the associated equipment should have suitable qualifications, experience and knowledge.

4.43. The response of the measurement equipment, expressed in operational units (e.g. counts integrated over a period of time), should be converted into the same quantities as the clearance levels under consideration (i.e. Bq/g or Bq/cm^2).

Surface contamination measurements for clearance

4.44. If the radioactivity is limited to the surface, for example in the case of impermeable objects, surface contamination measurements could be performed as a means of demonstrating compliance with activity concentration clearance levels. For example, instead of performing activity concentration measurements directly, it might be possible to convert the results of surface measurements to activity concentrations and hence prove compliance also with activity concentration clearance levels. For this, the ratio between mass and surface area needs to be taken into account, for example by considering the thickness and density of the objects and the number of contaminated surfaces [31]. Alternatively, it may be possible to derive surface contamination clearance levels from activity concentration clearance levels. Conversely, if the material is permeable, then the contamination will penetrate some distance into the material, and in this case both surface and volumetric measurements may be needed to demonstrate compliance with clearance levels.

4.45. For the assessment of surface contamination, the principles and methods described in international standards (e.g. Ref. [32]) should be used for direct and indirect measurements and for the calibration of the associated instrumentation. If the use of surface contamination monitors in a ratemeter mode is not sufficiently reliable, reproducible or auditable for clearance purposes, then measurements based on integrated counts over a defined time should be used.

Measurement techniques for clearance

4.46. When undertaking direct measurements of surface contamination, attention should be given to the condition of the surface to be measured. The ideal surface is clean, dry and flat. Dust, grease, rust and moisture on a surface can absorb beta and (especially) alpha radiation. Thus, for clearance purposes, surfaces should be cleaned before measurement. This cleaning may also be a form of decontamination in cases of non-fixed contamination. In addition, analysis of the removed material (and measurements of the cleaning tool) could give information on the nature of the contamination.

4.47. An uneven surface will affect direct measurements of surface contamination owing to difficulties in ensuring the correct distance between the surface and the detector, which will affect the detector response.

4.48. For total gamma measurements, it may be possible to calibrate the monitoring instruments for a single radionuclide and derive the calibration factors for other radionuclides through calculations or on the basis of scaling factors (see paras 3.21 and 3.22) as determined from the radionuclide composition.

4.49. For in situ gamma spectrometry, the response to individual radionuclides depends on their distribution (i.e. on the surface and/or in the volume). Computer codes are available that allow calculations of calibration factors from a given radionuclide composition and spatial activity distribution (e.g. see Ref. [33]). Software for calibration is also provided by manufacturers of gamma spectrometry equipment.

4.50. Samples may be taken using wipe tests for removable surface contamination or by collecting a small fraction of the material itself. Wipe tests should be analysed using a surface contamination monitor or be subject to sample preparation and measurements in a laboratory (e.g. for tritium measurements). Only removable surface contamination can be quantified through wipe samples, and additional measurements may be needed to determine the fixed contamination. Samples of material should be analysed in laboratories with the necessary equipment. The laboratories should have a quality assurance system and should be accredited in accordance with national requirements or international standards (e.g. Ref. [34]).

4.51. When sampling is part of the clearance process, additional issues — such as sampling position, minimum sample size and number of samples — should be addressed. If the spatial distribution of the contamination is unknown or assumed to be uniform, a sample grid should be used, where the distance between two grid points is determined by the total area sampled and the number of samples needed. The position of the individual samples should be recorded. The sample measurements should provide information on the activity distribution in the material or object as a whole, to be compared with the clearance levels.

4.52. The minimum number of samples to be taken should be determined on the basis of a statistical compliance test. The decision on clearance of material should be based on a statistical test of the measured activity concentrations. Information on the selection of a statistical test is provided in Refs [21, 22]. According to these references, the number of samples should be increased if the results of the statistical test are not satisfactory with respect to median value and standard deviation.

4.53. An instrument used for monitoring a material for compliance with the clearance level for a specific radionuclide (or mixture of radionuclides) has to have a limit of detection sufficiently below the clearance level. The limit of detection should be determined in accordance with international standards (e.g. Ref. [35]). The limit of detection depends on the measurement technique; the measurement conditions, such as background and measurement time; and the accepted level of confidence in the measurement. In the case of sampling, the minimum sample size should be chosen to ensure that the limit of detection is well below the clearance level. Possible loss of material in the sample preparation process should be taken into account when calculating the minimum sample size.

4.54. When applying the concept of clearance, the background activity in the material (i.e. that existed prior to its contamination or activation during a practice) should be subtracted from the measured total activity. Activity from sources other than those in the practice itself, for example naturally occurring radionuclides (i.e. from the ^{238}U and ^{232}Th decay chains, and ^{40}K) in building material or fallout from nuclear weapon tests and nuclear accidents, should be subtracted from the results of clearance measurements. Similarly, cosmic radiation and naturally occurring levels of other primordial radionuclides should also be subtracted from the results.

4.55. When determining the approach to background subtraction for clearance measurements, normal variations in the background should be considered. The regulatory body should approve the process for determination of the background. This is especially important in the case of total gamma measurements. The distinction between different sources of activity can be significantly improved by using gamma spectrometry.

4.56. The necessary limit of detection for clearance measurements has an impact on the acceptable background conditions during the measurement. The background can be reduced by careful selection of the location for the measurements or by addition of shielding around the detector.

CONSIDERATION OF UNCERTAINTIES IN CLEARANCE MEASUREMENTS

4.57. The clearance process, in particular the measurement programme, involves a number of uncertainties that need to be taken into account to ensure that the final result used for making a decision about clearance is reliable. This result can be the activity concentration or surface contamination level of a single radionuclide or of multiple radionuclides present in or on a material or object.

4.58. The uncertainties to be considered involve two main types: statistical uncertainties associated with the counting process, and uncertainties relating to the measurement process that can be evaluated by means other than statistical analysis (e.g. based on experience or other information). The uncertainties that are most relevant for the clearance process are as follows:

(a) Statistical uncertainties associated with the counting process;
(b) State of the surface of the measured material;
(c) Fluctuation of the geometry and the self-shielding of the measured material;
(d) Fluctuation of the activity distribution in the measured material;
(e) Fluctuation of the background;
(f) Fluctuation of the contributions of individual radionuclides in the radionuclide composition;
(g) Fluctuation of the efficiency of surface radioactivity removal by wipe tests;
(h) Fluctuation of the content of natural radionuclides and other radionuclides to be disregarded in the material;
(i) Uncertainties associated with instrument calibration;
(j) Sampling uncertainty.

Further information on dealing with these uncertainties is provided in Annex VII.

MIXING AND DILUTION OF MATERIALS BEING CONSIDERED FOR CLEARANCE

4.59. Deliberate dilution of material to meet the clearance levels, as opposed to the dilution that occurs when radioactivity is not a consideration, should not be performed. In specific and exceptional cases, the regulatory body may consider approving deliberate dilution.

4.60. Some mixing of materials may be acceptable during decommissioning or for unavoidable material management purposes, as long as the purpose is not to deliberately dilute the concentration of radionuclides. For example, the use of an excavator to dig out a volume of contaminated soil may result in some mixing of soil with differing levels of contamination. In this case, the mixing happens as part of the normal material management process.

4.61. In cases where unavoidable mixing occurs, or where the distribution of radioactivity is inhomogeneous, care should be taken to ensure that any subsequent sampling or monitoring is suitably representative.

4.62. If it is necessary to reduce the uncertainty of the measurement result, it is acceptable to combine two or more samples of material (e.g. drums) after the initial characterization. This is not deliberate dilution, as the purpose is solely to reduce the measurement uncertainty, not to alter the characteristics of the material.

4.63. In the case of specific clearance (see Section 7), mixing with clean material could be a condition of this clearance. For example, it could be a condition that contaminated metal is melted in a general industrial melting facility. In this case, the destination of the cleared materials could also be a condition of the specific clearance, which might also include the average mixing ratio with clean materials to be applied, as considered in the model used to derive the specific clearance levels. The conditions could also specify that the destination of cleared materials is documented by the operating organization as part of the traceability of the clearance process for this material.

5. CLEARANCE OF LIQUIDS

DIFFERENCES BETWEEN THE DISCHARGE AND CLEARANCE OF LIQUIDS

5.1. Liquid effluents from facilities are usually treated as discharges that are managed in accordance with an authorization (see Requirement 31 of GSR Part 3 [1]). The dose constraint applied to liquid discharges is generally chosen in the range 0.1 to <1 mSv in a year (see paras 5.16 and A–2 of IAEA Safety Standards Series No. GSG-9, Regulatory Control of Radioactive Discharges to the Environment [36]). This is a fraction of the dose limit to members of the public. Using models such as those described in Ref. [37], this dose constraint is converted into authorized limits on discharges of radionuclides, usually expressed in becquerels per year (Bq/a). Further recommendations are provided in GSG-9 [36].

5.2. The authorized discharges of liquid effluents described in para. 5.1 are performed under the system of regulatory control. This is conceptually different from clearance, which is a process through which regulatory controls are removed. There are situations where the removal of regulatory control from a liquid may be more appropriate than releasing a material in accordance with an authorized discharge. For example, the clearance of liquids can be used in cases where small amounts of liquid containing low levels of radionuclides are produced, for which the management of a discharge regime is not warranted. There may also be cases where liquids constitute an asset and where there is a commercial interest in reuse or recycling, for example lubrication oils used in pumps, cooling liquids from transformers in nuclear power plants or acids from the manufacturing process of nuclear fuel. Alternatively, it may be beneficial to incinerate certain liquids in a conventional waste incineration plant because of their hazardous properties. In all such cases, clearance may be the best option. For clearance of liquids, the same basic principles given in Sections 2 and 3 of this Safety Guide apply as they do for solid materials. Clearance of liquids needs to be based on the same dose criterion as clearance of solid material (i.e. individual effective doses of the order of 10 µSv in a year).

5.3. There is a fundamental difference between the authorized discharge of liquid effluents and the granting of clearance to a liquid. In the case of discharges, once released, the radionuclides are dispersed in the environment and the possibility for reconcentration of activity in the environmental matrices is extremely low. In contrast, cleared liquids might not be dispersed, and after clearance the activity concentration might even increase (e.g. due to filtration, evaporation, distillation or fractionation). This difference should be taken into account in the derivation of clearance levels for liquids. The clearance of tritium (^{3}H) in the form of tritiated water does not usually necessitate such considerations because the concentration of this radionuclide is highly unlikely to be significantly increased by natural processes in liquids, sediments, plants or animals.

ASPECTS OF LIQUIDS RELEVANT TO CLEARANCE

5.4. Liquids have some properties that distinguish them from solid materials with respect to the application of the concept of clearance, as follows:

(a) Radionuclides in aqueous liquids or in organic liquids can be easily concentrated (e.g. by evaporation or distillation) so that the initial activity concentration can increase.
(b) If the liquid evaporates completely, the residual radionuclides effectively become a source of solid surface contamination.
(c) Radionuclides can be accumulated on filters during filtration processes.

5.5. Some types of liquid, including some oils, lubricants, antifreeze agents and other organic substances, often do not have the properties described in para. 5.4, or else they do to a much lesser extent. As a result, the activity concentration in such liquids is less likely to change during subsequent treatment. The main exception is incineration, which can lead to the concentration of certain radionuclides in residues such as ashes or slags. This should be considered in the model used for derivation of clearance levels, as was done in the model in Ref. [5] for solid materials.

5.6. Liquids containing contamination in the form of radionuclides bound to suspended particles can be decontaminated by filtration processes; however, the activity will then concentrate in the residue. A typical example is lubricants, in which abraded particles containing contamination accumulate.

5.7. Several States have chosen to limit clearance to liquids that have been filtered prior to clearance and for which the likelihood of any processes leading to an increase in activity concentration is very small or negligible. In such cases, therefore, the derivation of clearance levels might not need to consider processes that lead to such an increase.

NATURE AND SCOPE OF CLEARANCE FOR LIQUIDS

5.8. The clearance of non-aqueous liquids is an example of specific clearance, as described in Section 7. As such, additional conditions (e.g. with regard to the destination of the liquid) can be applied in accordance with para. I.13 of GSR Part 3 [1]. The following options can be considered:

(a) The liquids are cleared for any purpose; that is, they can be directly reused, recycled or further treated (e.g. by incineration). This may be the case for oil or lubricants after filtration, which can be directly reused, recycled (through conversion into fuel) or treated (by incineration in a waste incineration plant).
(b) The liquids are cleared for a specific process only, for example for treatment by incineration in a conventional waste incineration plant.
(c) The liquids are filtered before being cleared.
(d) The liquids are cleared for a specified use or for a specified recipient.
(e) The clearance of the liquids is subject to limitations, for example in terms of the total or annual quantities.

5.9. Case by case decisions are of considerable importance for the specific clearance of liquids, in particular when aqueous liquids such as diluted acids that have been used in certain processes in nuclear facilities (e.g. hydrogen fluoride in uranium fuel manufacturing) are to be cleared for further use in the chemical industry. Models describing such specific clearance need to consider possible processes of concentration (e.g. when instead of a diluted acid a strong acid is needed) and filtration, including those in water purification plants where many chemical elements are extracted from the water and concentrated in sewage sludge. The chemical toxicity of liquids should be taken into account when deciding if other regulatory controls need to remain after clearance.

5.10. Where the concept of clearance is applied to aqueous liquids, they may subsequently be discharged into a lake, river or sea. As the liquid has already been cleared, no authorization is needed for the discharge (although the approval of the water authorities may still be necessary). In such a case, the model used to derive the clearance levels needs to take into account all relevant pathways in the environment (i.e. dispersion of radionuclides in the water body, effects of sedimentation, and the uses of the water), as described in Ref. [37]. The clearance of liquids may involve similar exposure pathways as the clearance of solid materials (i.e. external irradiation, inhalation, direct ingestion and inadvertent ingestion). A model specifically designed for liquids from medical, industrial and research facilities and that covers all relevant exposure pathways and exposure scenarios is described in Ref. [30].

PRACTICAL APPLICATION OF THE CONCEPT OF CLEARANCE TO LIQUIDS

5.11. Reference [30] includes practical guidance on the application of the concept of clearance to liquids that are to be released to the environment. The values in table IV of Ref. [30] were derived with the intention of ensuring that annual doses to individual members of the public arising from any single practice will not exceed a dose of the order of 10 μSv. These values are expressed in Bq/a and can be converted into activity concentrations in Bq/m^3 or Bq/L, if the annual amount of liquids to be cleared is known. Information on specific activity values may be useful for making decisions on the choice of methods and equipment for monitoring the process of release of liquids into the environment. Compliance with these levels (or with similar levels derived on the basis of the clearance criteria from GSR Part 3 [1]) means that further monitoring (i.e. of the environment) is not necessary.

5.12. The clearance levels for solid materials provided in table I.2 of GSR Part 3 [1] may also serve as the basis for clearance of some liquids, provided that concentration or filtration processes will not occur after the liquids have been cleared. The model in Ref. [5] that underlies the clearance levels for solid materials covers various scenarios that would also be bounding for reuse, recycling or disposal of liquids (e.g. storage in a large tank giving rise to external gamma radiation; evaporation of the liquid leading to inhalation and ingestion of water sourced from contaminated groundwater). The only scenario not explicitly covered by the models described in Ref. [5] is the release of large quantities of liquids to the environment. As such, the clearance levels provided in table I.2 of GSR Part 3 [1] could be applied for clearance of non-aqueous liquids (e.g. oils, lubricants) for reuse, recycling or disposal by incineration. The clearance levels in table I.2 of GSR Part 3 [1] for solid materials are in Bq/g and should be converted into units that are suitable for liquids (e.g. Bq/L).

5.13. An example of regulations for the clearance of liquids is provided in Ref. [38], which covers ‘relevant liquids’: non-aqueous liquids and certain types of aqueous liquid with specific hazardous properties. The purpose of this definition is to allow clearance of such liquids on the basis of clearance levels for solid materials, as the exposure pathways considered in the derivation of clearance levels for solid materials encompass relevant exposure pathways for these liquids. An example of the practical application of the concept of clearance to liquids (usually via disposal in a waste incineration plant) is provided in Annex III to this Safety Guide.

5.14. The characterization of liquids for clearance purposes should follow the recommendations provided in paras 3.9–3.24. Special attention should be given to the homogeneity of the liquid and the possibility of deposition of sediments.

5.15. Clearance levels for liquids will usually be expressed in terms of activity concentration for each radionuclide (e.g. in Bq/L). In some cases, for example where liquids or their residues might accumulate in specific locations, it may be more appropriate to determine clearance levels in terms of total activity (i.e. of individual radionuclides or of groups of radionuclides (e.g. in Bq/a)), either in addition to, or instead of, activity concentration clearance levels. Examples of such an approach are provided in Refs [30, 39].

DILUTION OF LIQUIDS BEING CONSIDERED FOR CLEARANCE

5.16. The deliberate dilution of liquids (e.g. with clean water) to comply with clearance levels is generally not acceptable. However, in exceptional cases permission may be obtained from the regulatory body. For example, dilution may be necessary to manage non-radiation-related properties, such as pH or salt content, prior to discharge, and this should be taken into account when deciding about clearance. The possibility of subsequent concentration of radionuclides (e.g. in industrial uses or in the environment) should be considered. Dilution of the liquids after clearance will occur at many subsequent stages and should be taken into account in the models used to derive clearance levels.

5.17. A further aspect of dilution is relevant for the clearance of small volumes of liquids, for example residues of radiopharmaceuticals in vials to be disposed of by incineration. Such liquids cannot be readily emptied from the vials (or from other items such as gloves or syringes). In practice, when determining compliance with clearance levels, the total mass (i.e. of the liquid and the containers) should be considered.

BACKGROUND RADIATION IN THE CLEARANCE OF LIQUIDS

5.18. As with the clearance of solid materials (see para. 4.54), the clearance of liquids should be based on the radionuclides that originate from the practice in question, and background radioactivity should be disregarded. Examples of background radioactivity include radionuclides of the ^{238}U and ^{232}Th decay chains and ^{40}K (e.g. U or Th oxides and complexes, potassium iodide or iodate).

6. CLEARANCE OF GASES

DIFFERENCES BETWEEN THE DISCHARGE AND CLEARANCE OF GASES

6.1. Gaseous releases from facilities are usually treated as discharges that are managed in accordance with an authorization (see Requirement 31 of GSR Part 3 [1]). In specific cases where application of the concept of clearance is necessary, for example where reuse of the gaseous material is intended, clearance levels for gases should be calculated on the basis of individual effective doses of the order of 10 µSv in a year. Gases that meet such clearance levels can subsequently be released without an authorization, provided there are no restrictions due to their chemical or other hazardous properties. An example is clearance of nitrogen gas found in a nuclear facility (for use in gloveboxes where radioactive material is handled) for reuse outside the nuclear facility.

6.2. In contrast to liquids, it is unlikely that potentially radioactive gases originating from a facility will constitute an asset for which reuse or recycling could be envisaged. If, however, there is a need to apply the concept of clearance to gases that are to be reused or recycled, then the model used to derive clearance levels should consider the possibility that the concentration of radionuclides in the gas might change (e.g. owing to changes in the pressure or volume of the gas). The pressure of a gas may vary over orders of magnitude, depending on the volume in which it is contained. Exposure scenarios relevant to a compressed gas in a container may be fundamentally different from those for a gas under standard conditions. Examples of clearance levels for gases released from a vent at the side of a building are given in Ref. [30].

PRACTICAL APPLICATION OF CONCEPT OF CLEARANCE TO GASES

6.3. The application of the clearance levels provided in table I.2 of GSR Part 3 [1], or of any other clearance levels derived for solid or liquid materials, to the clearance of gases is not appropriate.

6.4. Practical guidance on the application of the concept of clearance to gases intended for release to the environment is provided in Ref. [30]. Once cleared, the gases should not be subject to any discharge authorization. The values in table III of Ref. [30] were derived with the intention of ensuring that annual doses to individual members of the public arising from any single practice will not exceed a dose of the order of 10 μSv. These values are expressed in Bq/a and can be converted into activity concentrations in Bq/m³ if the annual amount of gases to be cleared is known. Compliance with these levels (or with similar levels derived on the basis of the clearance criteria in GSR Part 3 [1]) means that further monitoring (e.g. of the environment) is not necessary.

6.5. The characterization of gases for clearance purposes should follow the recommendations provided in paras 3.9–3.24.

7. THE APPLICATION OF SPECIFIC CLEARANCE

7.1. Paragraph I.13 of GSR Part 3 [1] states:

> "Clearance may be granted by the regulatory body for specific situations, on the basis of the criteria of paras I.10 and I.11 [of GSR Part 3], with account taken of the physical or chemical form of the radioactive material, and its use or the means of its disposal[65]. Such clearance levels may be specified in terms of activity concentration per unit mass or activity concentration per unit surface area.
>
> "[65] For example, specific clearance levels may be developed for metals, for rubble from buildings and waste for disposal in landfill sites."

Hence, the radiological basis for specific clearance is the same as for generic clearance (i.e. as described in paras I.10 and I.11 of GSR Part 3 [1]). Examples of specific clearance that have been applied in individual States include scrap metal for recycling (melting), buildings for demolition and waste for disposal in landfill sites.

7.2. The models used to derive generic clearance levels [5] are deliberately conservative so as not to underestimate the possible exposures in all relevant circumstances. There are, however, situations where a generic approach is not suitable, either because a specific exposure scenario is not covered by the generic model or because key parameters describing a specific exposure scenario deviate significantly from the values used in the generic model. In such cases, specific clearance should be considered, in which a model is developed that specifically considers the relevant exposure scenarios and parameter values. Key parameters where significant deviations from generic values may occur include exposure times, distances on which dose rates from external irradiation are based, shielding geometries, concentrations of contaminated aerosols, quantities of materials to be cleared and the amount of material present.

7.3. The analysis of a specific situation may show that certain scenarios in the corresponding generic model are not relevant. These scenarios should then be excluded from consideration in the derivation of specific clearance levels.

7.4. The models used to derive specific clearance levels should take into account the same exposure pathways as the generic model described in Ref. [5]. Therefore, external irradiation, inhalation of contaminated aerosols, direct ingestion of small quantities of radionuclides, and ingestion of radionuclides via the food chain and skin contamination should be included. Although based on specific circumstances, the parameters used in the exposure scenarios should be chosen so as to take account of possible variations. For example, if exposure times are expected to vary between 240 and 480 h/a, it would be prudent to use a value of 500 h/a to take into account these variations.

7.5. The use of specific clearance allows specific circumstances within a State (e.g. industrial, environmental, climate related and regulatory requirements) to be taken into account and avoids the inappropriate use of generic clearance levels. Specific clearance levels can be established on a national basis (i.e. in legislation developed by the regulatory body) or in response to an application by one or more operating organizations. Specific clearance levels include conditions in relation to, for example, the type of material, the amount of material or the destination of the material.

7.6. With due consideration of the general criteria for clearance specified in para. I.10(b) of GSR Part 3 [1], the regulatory body may decide (where the national regulatory framework so allows) that the optimum regulatory option is to remove the material from regulatory control by setting or authorizing specific clearance levels. This might also be appropriate for specific materials containing radionuclides at activity concentrations that exceed the generic clearance levels (i.e. in table I.2 of GSR Part 3 [1]) in order to allow their further management as non-radioactive materials. In making such a decision, the regulatory body should consider the doses from ‘realistic’ and ‘low probability’ scenarios and the degree of conservatism in the dose estimates, as well as other factors. The mechanism for making such a decision will depend on the nature of the national regulatory framework. The decision will be made by the regulatory body on a case by case basis, in most cases following notification by the operating organization to the regulatory body.

7.7. In practice, specific clearance levels are usually expected to be less restrictive than generic clearance levels. This might, however, cause problems if a material cleared on the basis of specific clearance levels is exported to another State where only generic clearance levels (e.g. as provided in table I.2 of GSR Part 3 [1]) are applied. In such cases, the derivation of specific clearance levels should take into consideration the clearance levels applied in other States to which the material might be sent. Alternatively, specific arrangements should be made in advance with these other States for acceptance of the cleared material.

SPECIFIC CLEARANCE AS AN ADDITIONAL OPTION FOR MANAGEMENT OF MATERIAL

7.8. Specific clearance represents an additional option for the management of material (including waste) from authorized facilities and activities as well as from remediation activities after the termination of an emergency. It is part of the application of the graded approach to regulatory control and supports the application of the waste hierarchy, reducing the amount of material to be managed as radioactive waste and increasing the amounts to be reused or recycled. Specific clearance also provides an opportunity for disposal of waste as non-radioactive instead of radioactive waste.

7.9. When considering whether specific clearance is appropriate, the regulatory body should consider other factors, for example the need for measures at the facilities that receive the cleared material to ensure that exposures to persons are acceptable and protection of the environment is adequate, and whether these measures can be relied on without regulatory oversight. If regulatory oversight is considered necessary to ensure protection and safety, then clearance is not appropriate. In such cases, materials should only be transferred to appropriately authorized facilities.

DERIVATION AND USE OF SPECIFIC CLEARANCE LEVELS

7.10. In accordance with para. I.13 of GSR Part 3 [1], specific clearance levels may be derived in terms of activity concentration or surface contamination. Clearance levels may also be derived in terms of other appropriate quantities, for example total activity (e.g. in Bq or Bq/a). Specific clearance levels derived for a specific set of materials and/or destinations are not automatically applicable to other materials or destinations.

7.11. Specific clearance can take into account the destination and fate of the cleared material; consequently, the derived clearance levels can be less conservative than those used for generic clearance. However, in the derivation of specific clearance levels it should be assumed that the cleared material is handled in the same way as similar non-radioactive material; that is, clearance should not rely on special radiation protection measures being taken to meet the dose criteria.

7.12. Care should be taken in the derivation and use of specific clearance levels to ensure that cleared material (e.g. metals for melting) can be received at the specified destination (e.g. smelter) without the need for notification or authorization. One means of avoiding this might be to ensure that the specific clearance levels do not exceed the exemption levels for moderate quantities of material specified in table I.1 of GSR Part 3 [1]. In this way, the cleared material can be exempted from the requirement for notification[6].

7.13. It may also be necessary to limit quantities of specifically cleared material over a specified time period (e.g. in terms of kilograms or tonnes in a year) in order to meet the conditions embedded in the radiological model for derivation of the specific clearance levels (i.e. number of transports linked to the exposure to the driver) or not to exceed certain limits of the specified destination.

7.14. During processes such as metal smelting, certain radionuclides might concentrate in the dusts and slags, and the activity concentrations in these by-products might therefore exceed the activity concentrations in the metals. The model used to derive specific clearance levels should take this into account to ensure that the doses from exposure to such dusts and slags do not exceed of the order of 10 μSv in a year. Examples of such models can be found in Refs [5, 29, 40].

SURFACE CONTAMINATION CLEARANCE LEVELS FOR SPECIFIC CLEARANCE

7.15. Surface contamination clearance levels for generic clearance (see paras 4.17–4.22) need to be distinguished from those for specific clearance.

[6] If the specified destination is an authorized facility (e.g. a licensed smelter of metals), then specific clearance might not be needed for a delivery of the material to that facility.

Surface contamination levels for specific clearance may, for example, be derived for the following:

(a) Clearance of metals for melting;
(b) Clearance of buildings for reuse;
(c) Clearance of buildings for demolition.

7.16. In the various options for specific clearance, surface contamination clearance levels can fulfil different purposes. For example, the limitation of the surface activity on metallic items will protect persons handling these items prior to melting, and the limitation of activity on the surface of buildings will protect persons living or working in such buildings from direct irradiation and from inhalation of resuspended activity due to refurbishment activities. Examples of surface contamination clearance levels for specific clearance of scrap metal for melting are given in Ref. [29]; specific clearance levels for buildings for demolition are given in Ref. [41].

7.17. Clearance on the basis of surface contamination clearance levels generally applies only to surfaces where the contaminant can be detected by surface measurement techniques and the depth of the contaminant is such that the measurement technique can detect, to a reasonable degree, all the contamination. Surface contamination clearance levels are not suitable for materials such as excavated soil or building rubble[7].

7.18. For surfaces where the activity can penetrate into the volume (e.g. building surfaces and permeable ground), it should be specified whether the clearance levels apply only to the top layer (i.e. the actual surface) or to the surface plus part of the volume beneath the surface. Usually, it is a prudent approach to relate surface contamination clearance levels to both the contamination present directly on the surface and the contamination present immediately beneath the same surface area. Further details are given in Ref. [41].

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[7] Building rubble to be cleared is usually measured using bulk monitors that can measure several hundred kilograms of material at a time, applying activity concentration clearance levels such as in table I.2 of GSR Part 3 [1]. Buildings that are to be cleared in the form of the standing structure are most often cleared (e.g. for subsequent demolition) using surface related clearance levels like those given in Ref. [41]. The building rubble originating from demolition of these buildings does not have to be subjected to additional clearance measurements to demonstrate compliance with activity concentration clearance levels.

7.19. Careful distinction needs to be made between the surface contamination clearance levels and the other uses of surface contamination levels, such as in SSR-6 (Rev. 1) [16]. The surface contamination limits specified in para. 508 of SSR-6 (Rev. 1) [16] are derived for the transport of radioactive material under regulatory control, which is not relevant for clearance, so those values should not be used for clearance purposes.

7.20. Models used for the derivation of surface contamination clearance levels need to consider all exposure pathways associated with the presence of surface contamination. In particular, these include the following:

(a) External irradiation from the contaminated surface;
(b) Ingestion as a consequence of hand-to-mouth transfer of contamination when handling objects;
(c) Inhalation as a consequence of resuspension of contamination when handling or machining objects;
(d) Skin contamination.

7.21. The same criteria for clearance as specified in paras I.10 and I.11 of GSR Part 3 [1] should be applied when deriving surface contamination levels for specific clearance. In addition, an equivalent dose limit to the skin of 50 mSv in a year should be applied for low probability scenarios, as described in para. 4.4(b).

7.22. If a material has been activated (e.g. the concrete shield around a nuclear reactor or particle accelerator), surface contamination clearance levels such as those provided in Ref. [29] are not applicable.

MEETING THE CONDITIONS ATTACHED TO SPECIFIC CLEARANCE

7.23. Specific clearance is applied to a particular material, sometimes for a specified amount and for a particular fate and/or destination of that material: all these conditions are attached to the specific clearance. Such conditions need to be met in order to consider the clearance process complete. For example, scrap metal that was cleared on the condition that it would be melted needs to actually reach a furnace and be melted there, and not be reused in some other way before that point. If mixing with non-radiological metal is a condition of the specific clearance, the mixing ratio used in the derivation of the specific clearance levels should be respected. Similarly, a building that was cleared on the condition that it would be demolished should not be used in the meantime (e.g. as a temporary office or a workshop).

7.24. To ensure that the conditions attached to specific clearance are met, it may be necessary to make a formal arrangement between the operating organization that generates the material for clearance and the operating organization of the final destination. Such an arrangement should provide a high level of assurance that the material will not be diverted prior to completion of the clearance process and that the radiation risks are minimized. The practicalities of this should be agreed with the regulatory body. This could include overseeing the transport to the specified destination or requiring receipts to be sent to the facility in which the material was originally cleared that can be reviewed by the regulatory body. If the destination is in a different State and transboundary movement of the specifically cleared material is planned, the clearance process should take that into account so that the material can be accepted for transport and further management (e.g. recycling) in the destination State.

7.25. Specific clearance can therefore be considered a two-stage process. Stage 1 is the act of clearance when it is confirmed that (a) the material meets the specific clearance levels, (b) the fate or destination is agreed, and (c) a formal agreement exists for the transfer and treatment of the material. Stage 2 is a confirmation, which occurs when evidence is provided that the conditions attached to the specific clearance have been met.

7.26. In the case of specific clearance of scrap metal for melting, the process of dealing with scrap metal in the relevant States will need to be understood, so that the appropriate conditions can be identified. Scrap metal often goes to scrap dealers, who store metals until they have a sufficient quantity of a particular type of metal to sell on to a metal melting company, and there is significant international trade in scrap metal. This type of business is not appropriate for the specific clearance of scrap metal. Instead, arrangements should be made to ensure that scrap metal is sent directly to the specified melting facility.

7.27. In the case of specific clearance of material sent to a landfill, the specificities of the landfill have to be understood and included in the scenarios used to derive the specific clearance levels. The conditions attached should take into account the capacity of the receiving landfill, the activity concentration levels for specific clearance and the leachability of radionuclides from the cleared material into the environment. The post-closure period for the landfill should also be considered. Possible intrusion scenarios after the end of the institutional control period might be treated as low probability scenarios (i.e. subject to a dose criterion of 1 mSv in a year), in accordance with para. I.11 of GSR Part 3 [1].

7.28. In the case of generic clearance, the process may be considered complete once compliance with generic clearance levels has been demonstrated (which usually happens at the facility that generates the material). In contrast, specific clearance may involve conditions on the material reaching a certain destination or end state (e.g. metal cleared for melting has to reach the smelter, waste cleared for disposal has to reach the landfill, buildings cleared for demolition without prior reuse have to be demolished). It might therefore be necessary to consider at what point the removal of regulatory control is also complete (and whether handling of the material prior to completion of the specific clearance would require an authorization by the regulatory body).

7.29. The model used to derive specific clearance levels should explicitly take into account any transport of the material that will occur as part of the clearance process. Specific clearance is required to be based on the same dose criteria as for generic clearance (see para. I.13 of GSR Part 3 [1]). The same dose criteria were used to derive the values for exemption in SSR-6 (Rev. 1) [16] (see paras 402.3–402.7 of IAEA Safety Standards Series No. SSG-26 (Rev. 1), Advisory Material for the IAEA Regulations for the Safe Transport of Radioactive Material (2018 Edition) [42]). As such, the transport of material that meets specific clearance levels should normally not be subject to the requirements of SSR-6 (Rev. 1) [16]; this should be confirmed when specific clearance levels are derived. Similarly, the destination facility should not need an authorization for dealing with the cleared material: this should also be confirmed when deriving specific clearance levels.

8. ENGAGING INTERESTED PARTIES AND ENHANCING PUBLIC UNDERSTANDING OF CLEARANCE

8.1. Although material that is cleared arises within notified or authorized practices, it is likely to be subsequently processed or used by people who are not familiar with radiation protection and who do not understand that the dose criterion for clearance — of the order of 10 μSv in a year (see para. I.11 of GSR Part 3 [1]) — represents a trivial radiation risk. Consequently, it might not be understood why cleared materials can be used without taking any radiation protection measures.

8.2. In implementing the arrangements for clearance, operating organizations and regulatory bodies are required to consult with interested parties (see paras 2.30(f) and 2.43(d) of GSR Part 3 [1]). The aim should be to explain the concept of clearance, its basis, and how it is regulated and performed in practice. To build confidence in the clearance process and ensure its acceptance, this engagement should be performed using clear terminology, in a transparent manner, and should take different forms depending on the interested parties. Examples of different forms of communication that might be used where there is a high level of interest include formal consultation or communication on the national framework for clearance; discussions between different regulatory authorities, operating organizations and other (e.g. waste management) organizations; seminars and workshops with interested parties; public hearings; printed material, such as leaflets; and electronic media, such as web pages and social media.

8.3. The aim of the engagement is to understand the concerns of the interested parties, to address these concerns with respect and in a proportionate manner, and to explain the social, economic and environmental benefits associated with the clearance of materials (e.g. from recycling and sustainable use of resources).

8.4. Demonstrating clearance procedures and the associated monitoring programme can be effective in enhancing understanding and building confidence in the clearance process, as can making it clear that the process is overseen by the regulatory body. However, it is still important that conservativism is not used in the clearance process simply to gain public acceptance.

8.5. One way of enhancing understanding of the radiation risk from cleared materials is to compare it with that from natural background radiation. Alternatively, comparisons with commonly accepted radiation exposures — for example, from air travel or the presence of natural radionuclides in foodstuffs — are also useful communication tools. Relevant information for these comparisons can be found in IAEA posters and leaflets on radiation protection[8]. Another point to communicate is related to the economic savings (e.g. due to recycling and avoiding costs for treatment, conditioning and disposal) made possible through clearance of materials.

8.6. Communication tools developed to enhance public awareness of radiation risks in other situations may also be useful [43, 44].

[8] Available at https://www.iaea.org/resources/rpop/resources/posters-and-leaflets.

8.7. In the case of specific clearance, other interested parties (e.g. transport operators, other relevant regulatory authorities) should also be consulted. Since specific clearance levels are normally higher than generic clearance levels, the regulatory body and operating organizations should explain these differences to interested parties.

Appendix

SCREENING LEVELS FOR RECYCLING OR DISPOSAL IN LANDFILLS OF MATERIAL AND WASTE IN A POST-EMERGENCY SITUATION

A.1. A nuclear or radiological emergency and subsequent recovery operations may continue for a long time (weeks to potentially decades). After the early and intermediate phases of the emergency, a next phase will be to manage the recovery of the affected people and the area as an existing exposure situation.

A.2. In an existing exposure situation, the reference level for the optimization of protection of people living in the affected areas is selected from the band of 1–20 mSv in a year (see para. 1.26 of GSR Part 3 [1]). The reference level is required to be specified by the regulatory body or other relevant authority (see para. 5.4(b) of GSR Part 3 [1]). In addition, after an emergency, it may be necessary to establish a new regulatory framework for the management of materials and waste in the affected area (e.g. for disaster waste, rubbish after cleaning homes up, agricultural waste, and soil and waste generated from decontamination work). In accordance with the regulatory framework, some materials and waste may be put under the system of regulatory control for radioactive material and for radioactive waste, respectively.

A.3. Owing to radioactive decay, there is a possibility that the activity concentration of the material or waste may reduce to a level at which regulatory control is considered to be unnecessary. In such cases, the recycling of materials or the disposal of waste in landfills could be allowed. This is similar to the concept of specific clearance in a planned exposure situation.

RELATIONSHIP BETWEEN SCREENING LEVELS IN EXISTING EXPOSURE SITUATIONS AND CLEARANCE IN PLANNED EXPOSURE SITUATIONS

A.4. In existing exposure situations, the concept of reference levels should be used as part of a protection strategy in which protection and safety is optimized. Reference levels are tools for optimization and have a role in defining, selecting, analysing and benchmarking the protection strategy.

A.5. Recycling of material or disposal of waste in landfills in a post-emergency existing exposure situation often cannot use the same dose criteria as for clearance in a planned exposure situation. Instead, different dose criteria may be selected that are more appropriate and that take into account the specificities of the existing exposure situation. Consequently, the term ‘clearance level’ is not appropriate in the context of a post-emergency exposure situation. This appendix instead uses the term ‘screening level’ for operational use in measurements of activity concentrations in materials and waste.

A.6. If recycling of material or disposal of waste in landfills is necessary in a post-emergency situation, any screening levels should be based on individual effective dose criteria whose values are less than or equal to the selected reference level for the existing exposure situation under consideration. These dose criteria should relate to the subsequent exposures to people from the material or waste to be managed. In such cases, the dose criteria should be specified by the regulatory body or other relevant authority. As an example, a dose criterion for realistic scenarios in the later stage of recovery after an emergency could be of the order of 1 mSv or less in a year (e.g. for doses to workers and the public under normal operations, doses associated with recycling, and doses from groundwater migration following disposal in a landfill). Dose criteria for low probability scenarios, such as intrusion into a landfill site after its closure, should also be specified and would be expected to be greater than or equal to the dose criteria for realistic scenarios.

A.7. Hence, for practical applications to support decision making in an existing exposure situation, an approach similar to that of clearance using ‘screening levels’ (in Bq/g) derived from suitable dose criteria is recommended. The regulatory body or other relevant authority should also specify the actions to be taken in cases where screening levels are exceeded. An example of such an approach implemented after the Fukushima Daiichi accident is presented in Annex VIII.

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Annex I

DOSIMETRIC MODELLING FOR DERIVATION OF RADIONUCLIDE SPECIFIC VALUES FOR CLEARANCE BASED ON SURFACE CONTAMINATION MEASUREMENTS

I–1. Calculation of clearance levels for surface contamination has been performed in various national and international studies (see Refs [I–1 to I–11]).

I–2. The radiological models underlying these studies are presented briefly in this annex.

EUROPEAN COMMISSION: RECOMMENDATIONS ON CLEARANCE

Overview of the approach

I–3. The recommendations in Refs [I–1, I–2] on clearance, issued by the European Commission, contain radiological models for the derivation of surface contamination clearance levels. Reference [I–2] contains the detailed description of the model. While the surface contamination clearance levels were derived for scrap metal (i.e. steel, copper and aluminium), they can be regarded as meeting the criteria for generic clearance, as they cover scenarios for reuse and recycling. Because of the nature of these scenarios, especially those for reuse, these clearance levels are applicable not only to metals but also to other materials that are handled, treated and used (e.g. items made from plastics, wood or glass).

I–4. The surface contamination clearance levels recommended in Ref. [I–1] apply to the total surface contamination (i.e. fixed plus non-fixed contamination) and are intended as an average over moderate areas, which is stated as "several hundred square centimetres up to 1 square meter", "depending on the type of material, contamination and homogeneity of the contamination" [I–1]. It is further stated in Ref. [I–1] that "Surface contamination limits for scrap metal are largely independent of the metal type since the transport and handling are similar regardless of the metal."

I–5. An overview of the scenarios used for the derivation of surface contamination clearance levels for scrap metal for recycling or reuse is provided in Fig. I–1. Both sets of scenarios are very similar in structure. However, in Refs [I–1, I–2], the significance of surface contamination clearance levels is

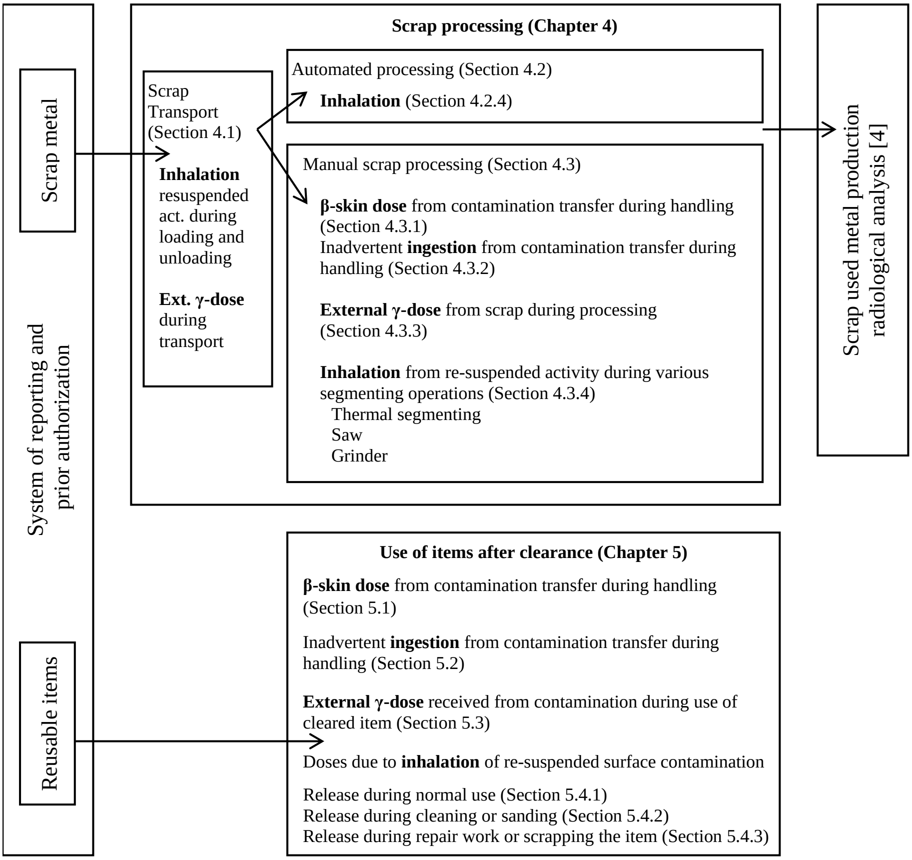

FIG. I–1. Overview of the scenarios for the derivation of clearance levels for metals for recycling and reuse from Ref. [I–2]. (The source, chapters and sections referred to in the figure are from Ref. [I–2].)

deemed different in the two areas of recycling and reuse of scrap metal: recycling is mainly governed by activity concentration clearance levels, while “The clearance criteria for direct reuse are primarily surface contamination limits since measurement of the bulk activity would in many cases mean destroying the equipment’s integrity” [I–1].

I–6. The scenarios developed in Ref. [I–2] are primarily of a deterministic nature and represent normal situations, during which contact with and exposure from the cleared metal can occur. The radiological model for surface contamination clearance levels developed in Ref. [I–2] is totally independent from that developed for activity concentration clearance levels in Ref. [I–12] for scrap metal. This means that the surface contamination clearance levels are not derived using a conversion factor to account for a mass to surface ratio.

I–7. In Refs [I–1, I–2], the clearance levels for radionuclides with radioactive progeny include the dose contributions from those progeny, which are accounted for by assuming they are in secular equilibrium with the parent radionuclide. Lists of these parent radionuclides and their progeny are provided in Refs [I–1, I–2].

I–8. Paragraphs I–9 through I–17 describe the scenarios used in Ref. [I–2] for the derivation of surface contamination clearance levels for generic clearance. Scenarios specific to recycling are not described.

Scenarios for the reuse of material

I–9. Reference [I–2] states:

> "The continued use of items after clearance from an authorized facility is termed reuse. The reuse of equipment and tools is a common practice in the nuclear industry and is economically preferable to disposal or scrapping the equipment."

I–10. Modelling reuse requires different scenarios than in the case of melting. Reference [I–2] states:

> "Unlike reuse, recycling scrap involves melting and reforming the scrap into new products. During this process the scrap is mixed with scrap from non-nuclear sources leading to a reduction in the mass specific activity of the product compared to the cleared scrap."

There is no such mixing for reuse; consequently, no dilution or modification of the material is assumed.

I–11. All relevant exposure pathways are taken into account in the models described in Ref. [I–2], as follows:

(a) External irradiation;
(b) Inadvertent ingestion of contamination;
(c) Inhalation of contamination from the resuspension of activity;
(d) Skin exposure from the transfer of contamination to the body.

Dose from external irradiation during the reuse of cleared equipment

I–12. In Ref. [I–2], it is acknowledged that there may be a large variety of exposure conditions to which a person using a cleared piece of equipment might be exposed. Therefore, an enveloping approach has been taken where a worker is exposed to a large item, in this case a tool cabinet that has a comparatively large overall surface: two panels (doors and back), six shelves, and two sides, and overall dimensions of 2 m in height, 1 m in width and 0.4 m in depth, leading to a total surface of 8 m^2. It is assumed that the worker is effectively exposed to 4 m^2, which represents the front and back of the cabinet. The exposure time is set to 1800 h/a, representing a full working year.

Dose from inadvertent ingestion of contamination during the reuse of cleared equipment

I–13. Exposure due to inadvertent ingestion during the reuse of a cleared item can occur when the contamination is transferred from the item to the mouth via the hands, for example while eating or smoking. This part of the model is similar to that used for derivation of surface contamination clearance levels for recycling. It is conservatively assumed that ingestion takes place during 200 h/a, with an ingestion rate of 1.2 cm^2/h and a transfer of 1% of the surface activity to the hand.

Dose from inhalation of contamination from resuspension of activity during the reuse of cleared equipment

I–14. Reference [I–2] considers four types of inhalation scenario, as follows:

(a) During normal use, the surface activity can be shaken loose and resuspended.
(b) The item can be cleaned or sanded, for example in preparation for painting, leading to resuspension of the surface activity.
(c) Repair work such as welding or thermal cutting can be performed, leading to resuspension of the surface activity.
(d) At the end of the item’s useful life it will be scrapped, which means it could be thermally segmented, leading to resuspension of the surface activity.

The last two scenarios are very similar and are considered together.

I–15. The normal use scenario assumes that exposure time is 1800 h/a and that 1% of the reused item is resuspended, the ambient dust concentration is 0.2 mg/m^3, and the breathing rate is 1.2 m^3/h.

Dose to the skin from the transfer of contamination to the body during the reuse of cleared equipment

I–16. During the reuse of a cleared item, the contamination can be transferred to the skin and cause a skin dose, especially from beta radiation. This scenario assumes that the contaminated area of the skin is 0.1 m^2, that the exposure time is 1800 h/a and that 1% of the contamination is transferred from the item to the skin. In this scenario, the skin dose is also converted to an effective dose by using the tissue weighting factor for the skin of 0.01.

Other scenarios for the derivation of surface contamination clearance levels

I–17. As shown in Fig. I–1, Ref. [I–2] contains a number of other scenarios that are also used for the derivation of surface contamination clearance levels. These include scenarios for automated scrap processing (mainly for the use of automated shear presses, shredders, hammer mills and scrap presses), for which external irradiation and ingestion pathways are analysed, and for manual scrap processing (mainly for manual cutting with thermal techniques), for which scenarios covering all exposure pathways as listed in para. I–10 are included, albeit with different parameter values. Manual scrap processing leads to the highest doses per unit surface contamination level, since the workers are in direct contact with the contaminated scrap.

I–18. A further analysis of the two most important exposure pathways — inhalation from manual processing of scrap and external gamma irradiation from using cleared items — was performed in Ref. [I–2] using dedicated stochastic models. These two exposure scenarios are deemed to be especially important, because they involve prolonged close contact with large quantities of scrap or large items, and because the radionuclides associated with contaminated scrap metal are often gamma emitters (e.g. ^{60}Co, ^{137}Cs) or are radionuclides with high inhalation dose coefficients (e.g. nuclides of uranium and plutonium). The results of this further analysis showed that the choice of parameters in the deterministic scenarios was conservative.

UNITED STATES OF AMERICA: NUREG-1640 AND ANSI/HPS N13.12

I–19. The model described in Ref. [I–3] for the derivation of surface contamination clearance levels is different from the one described in Ref. [I–2]. Reference [I–3] describes a complex model primarily aimed at deriving activity concentration clearance levels for the reuse, recycling and disposal of iron and steel scrap, scrap aluminium, scrap copper, and concrete rubble. These materials make up the bulk of components that would be potentially cleared from nuclear installations or other licensed facilities.

I–20. Surface contamination clearance levels are derived from the activity concentration values by a conversion factor describing the mass to surface ratio of the material. The most common values for these conversion factors are 5.1 g/cm^2 for steel and 280 g/cm^2 for concrete. This approximately 50-fold difference is why the clearance of contaminated steel or copper scrap yields the highest effective dose and, therefore, the most restrictive surface contamination clearance levels. A similar approach has been used in Ref. [I–4], where a similar conversion has been performed on the basis of the activity concentration clearance levels provided in IAEA Safety Standards Series No. GSR Part 3, Radiation Protection and Safety of Radiation Sources: International Basic Safety Standards [I–13]. The conversion factor in this case has simply been set to 1 g/cm^2, so that the values in Bq/cm^2 are numerically equal to those in Bq/g.

I–21. Reference [I–3] is only used by the regulatory body to assist with evaluating specific exposure scenarios and their relevant exposure pathways. The regulatory body does not use the clearance levels to make regulatory decisions.

I–22. The models described in Refs [I–3, I–4] do not directly derive surface contamination clearance levels. Nevertheless, the conversion of activity concentration clearance levels to surface contamination clearance levels may be a viable approach for cases where a dedicated model for derivation of surface contamination clearance levels would be too challenging.

UNITED STATES OF AMERICA: ARGONNE NATIONAL LABORATORY SURFACE CLEARANCE CRITERIA FOR WORKERS

I–23. Reference [I–5] contains an evaluation of the potential dose distribution resulting from surface contamination, using occupational exposure scenarios. The aim was to test a set of surface contamination clearance levels for their compliance with dose limits or dose constraints for workers.

I–24. Two scenarios were considered in calculating dose distributions for 13 selected radionuclides that most commonly occur in nuclear installations, as follows:

(a) The first scenario assumes the use of a contaminated building by workers. Two buildings — a large warehouse and a small office, with different building dimensions — were analysed. Contamination was assumed to exist on the floor and inside the four surrounding walls, with equal levels on all surfaces. A worker inside such a building was assumed to receive radiation doses through (i) external irradiation from the floor and interior walls; (ii) inhalation of contaminated particles resuspended from the contamination on the floor and interior walls; (iii) ingestion of deposited dust particles; (iv) external irradiation from submersion in contaminated air; and (v) external irradiation from deposited dust particles.

(b) The second scenario assumes use of a contaminated desk in an office setting. It was assumed that the top of a writing desk of typical size was uniformly contaminated and that the worker was sitting at a normal distance from the desk. The worker was assumed to receive radiation doses through (i) external irradiation from the top of the desk; (ii) inhalation of contaminated particles resuspended from the contamination on the desk; (iii) ingestion of deposited dust particles; (iv) external irradiation from submersion in contaminated air; and (v) external irradiation from deposited dust particles.

I–25. The analysis was performed assuming statistical distributions for each key parameter value, with a distribution type appropriate for the parameter and limited by reasonable boundaries. The analysis established a link between a given level of surface contamination and the distribution of the estimated doses, using the mean dose for the final assessment. The dose criteria against which the results were assessed were in the range of 50–100 μSv in a year. On this basis, the surface contamination clearance levels from which the analysis started were judged to be applicable.

JAPAN: GUIDELINE FOR REMOVING OBJECTS CONTAMINATED WITH RADIOACTIVE MATERIALS IN A PLANNED EXPOSURE SITUATION

I–26. With respect to the control of surface contaminated objects[1], guidelines have been developed by the Standardization Committee on Radiation Protection of the Japan Health Physics Society for planned exposure situations, emergency exposure situations and existing exposure situations [I–6]. Table I–1 summarizes the main points of the guidelines for planned exposure situations.

[1] Reference [I–6] uses the term ‘commodities’, which are defined as solid valuable goods (e.g. vehicles, equipment) whose reuse or recycling after removal from a controlled area has been justified. In this annex, ‘commodities’ is replaced by the term ‘object’.

TABLE I–1. SUMMARY OF JAPANESE GUIDELINES FOR REMOVAL OF OBJECTS CONTAMINATED WITH RADIOACTIVE MATERIALS IN PLANNED EXPOSURE SITUATIONS
(adapted from Ref. [I–6] with permission)

Guidelines	Criteria
Dose criteria (effective dose)	Order of 10 μSv/a or less
Concept	Clearance
Basic purpose	Removal of objects from controlled areas Clearance of many relatively small objects
Exposure scenarios	Handling of small packages [I–7] Handling of general objects [I–8]
Examples of readings of typical Geiger–Müller survey meter widely used in Japan	1000 cpm (10 Bq/cm^2 of Co-60) 2300 cpm (10 Bq/cm^2 of Cs-137)

I–27. This guideline is to be applied for removing objects from controlled areas. In general, there is no control on the use of objects after their removal, and therefore the removal is similar to the concept of clearance. Clearance is normally based on an assumption of handling a large amount of materials, such as dismantling waste from a nuclear installation, while the removal of objects from controlled areas in planned exposure situations more often involves the handling of many relatively small objects.

I–28. The surface contamination levels on small objects being removed from controlled areas that would correspond to the clearance dose criterion of an annual effective dose of the order of 10 μSv or less (for realistic exposure scenarios) were calculated. It was concluded that continued control of objects that have been removed from controlled areas is not warranted from a radiation protection perspective. The applicability of the concept of clearance has, therefore, been demonstrated and included in the guideline.

I–29. Two exposure scenarios were used for derivation of surface contamination levels for the removal of objects in planned exposure situations. These are handling of small packages [I–7] and handling of general objects [I–8] (including manually handled objects with a surface area of 0.1 m^2, closely handled objects with a surface area of 1 m^2 and remotely handled objects with a surface area of

10 m^2). In both scenarios, the surface contamination levels for ^{60}Co and ^{137}Cs that correspond to the dose criterion of an annual effective dose of 10 μSv were calculated to be 10 Bq/cm^2 for both radionuclides. These surface contamination levels for ^{60}Co and ^{137}Cs correspond to readings of 1000 cpm and 2300 cpm, respectively, using a typical Geiger–Müller surface contamination survey meter with a 20 cm^2 window, based on Ref. [I–14].

I–30. For high energy gamma emitters, such as ^{60}Co and ^{137}Cs, the surface contamination level that corresponds to the dose criterion of an annual effective dose of 10 μSv significantly depends on the assumed size of the contaminated surface. The same radionuclides may also be key radionuclides for surface contamination measurements of beta radiation.

I–31. In developing the guideline, it became clear that surface contamination clearance levels derived for large objects would be too conservative for small objects routinely removed from controlled areas. Instead, separate surface contamination levels are applied for routine control of small objects removed from controlled areas, in accordance with the dimensions of the surface likely to be contaminated.

UNITED KINGDOM: NUCLEAR INDUSTRY GUIDE TO CLEARANCE AND RADIOLOGICAL SENTENCING

I–32. Appendix F to Ref. [I–9] contains a derivation of surface contamination clearance levels for contaminated items. In particular, it contains a calculation of the maximum alpha, beta and total activity levels for the reuse of metallic equipment from dismantling of nuclear installations. The model used in Ref. [I–9] is the same as that used in Ref. [I–2], leading to the same derived surface contamination clearance levels.

THE SUDOQU MODEL

I–33. Reference [I–10] describes a model intended to evaluate the annual effective dose to members of the public resulting from exposure to surface contaminated objects, for example from nuclear accidents, taking into account all relevant exposure pathways (i.e. external irradiation, inhalation, indirect ingestion and skin contamination).

I–34. The applicability of this model to the calculation of surface contamination clearance levels has been evaluated [I–11]. The model calculations were applied to a number of deterministic scenarios for calculating the annual effective dose resulting from exposure to a typical office item (i.e. a bookcase), considering different scenarios of use and different radionuclides. The scenarios were then used to calculate surface contamination levels that would correspond to an annual effective dose of 10 µSv in a year.

I–35. The results of these calculations were then compared with the results of Ref. [I–2]. Differences due to different assumptions for parameters and exposure scenarios were observed. One of the main differences is that the SUDOQU model in Ref. [I–10] considers reduction of the surface activity with time not only through radioactive decay but also through resuspension and transfer of activity to the hands. Similarly, the resuspended activity contributes to the increase in airborne activity concentration and can, in turn, partly redeposit onto the object surface.

I–36. It was concluded in Ref. [I–11] that the SUDOQU model for dose assessments was suitable for the clearance of objects from nuclear installations. Further development of the model allowed for detailed parameter sensitivity analyses and probabilistic dose evaluations. Derived surface contamination clearance levels have been accepted by the Belgian Federal Agency for Nuclear Control for regulatory use [I–15].

REFERENCES TO ANNEX I

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Annex II

EXAMPLES OF SURFACE CONTAMINATION VALUES FOR GENERIC CLEARANCE

II–1. IAEA Safety Standards Series No. GSR Part 3, Radiation Protection and Safety of Radiation Sources: International Basic Safety Standards [II–1], does not specify values for clearance levels in terms of surface contamination; however, there are a number of international and national recommendations and guidelines containing such levels. This annex provides a short overview of selected recommendations and studies in which surface contamination clearance levels have been derived. Examples of the derived values (in Bq/cm^2) for a small selection of radionuclides are presented in Table II–1.

II–2. This annex is limited to examples of surface contamination clearance levels intended for use in generic clearance. Models aimed at providing surface contamination clearance levels for specific clearance contain too many differences in terms of assumptions, exposure scenarios and parameter values to enable a meaningful comparison of the derived clearance levels.

II–3. IAEA Safety Standards Series No. SSR-6 (Rev. 1), Regulations for the Safe Transport of Radioactive Material, 2018 Edition [II–2], is also considered. These regulations do not contain clearance levels, but they do contain surface contamination values, which have been frequently misused as clearance levels.

EUROPEAN COMMISSION

II–4. Surface contamination clearance levels have been derived for metals arising from the dismantling of nuclear installations and intended for recycling or reuse [II–3, II–4]. As described in Annex I to this Safety Guide, deterministic scenarios were used to calculate radionuclide specific results in terms of μSv per year for a unit surface contamination level of 1 Bq/cm^2. The surface contamination level for each scenario that would lead to a dose of 10 μSv per year was then derived. The smallest derived value was then used as the surface contamination clearance level for each radionuclide. In most cases, this value is based on the reuse scenario.

GERMANY

II–5. Reference [II–5] describes a study conducted in the course of the preparation of the German Radiation Protection Ordinance in 1998 and 1999. Much of this work was performed in parallel to work undertaken by the European Commission described in para. II–4. The main aim of the study described in Ref. [II–5] was to derive a single set of surface contamination clearance levels for both reuse and recycling. This makes these clearance levels applicable also for generic clearance.

UNITED STATES OF AMERICA

II–6. Reference [II–6] describes a comprehensive study on clearance of scrap metal in the United States of America, in which activity concentration clearance levels and surface contamination clearance levels were derived. Reference [II–6] provides a description of the calculations and the estimated annual doses to an individual following the clearance of scrap iron and steel, copper, aluminium, and concrete rubble from licensed nuclear installations. The estimated doses are calculated probabilistically to take into account a large number of possible variations in each of 86 exposure scenarios. These scenarios encompass the full range of realistic situations likely to yield the highest doses. Each scenario was analysed with the 115 radionuclides considered most likely to be associated with materials from licensed nuclear installations. The aim of the analyses was to realistically model current processes, to identify critical groups on a radionuclide by radionuclide basis, and to enable the conversion of a dose criterion into clearance levels in terms of activity concentration or surface contamination.

COMPARISON OF DERIVED VALUES OF SURFACE CONTAMINATION FOR GENERIC CLEARANCE

II–7. A selection of surface contamination clearance levels from the studies and recommendations described in paras II–4 to II–6 are presented in Table II–1. Table II–1 contains only clearance levels that can be reasonably associated with generic clearance (i.e. they are not specific to buildings or land). All the clearance levels in Table II–1 were derived on the basis of an individual effective dose of 10 μSv in a year.

II–8. Table II–1 shows that for strong gamma emitters (e.g. ^{60}Co, ^{137}Cs, ^{154}Eu) and for alpha emitters (e.g. ^{242}Pu, ^{241}Am), there is generally good agreement between the surface contamination clearance levels derived by the different models, indicating that the models for external irradiation (gamma emitters) and for inhalation of resuspended surface contamination (alpha emitters) are based on similar assumptions. The agreement between the clearance levels for strong beta emitters (e.g. ^{90}Sr) can also be considered to be fairly good, indicating that ingestion pathways (direct and secondary ingestion) are generally based on similar assumptions. The surface contamination clearance levels for weak beta emitters and electron capture emitters (e.g. ^{3}H, ^{14}C, ^{36}Cl, ^{55}Fe) differ more, indicating that the scenarios underlying the derivation of these values are significantly different

TABLE II–1. COMPARISON OF SURFACE CONTAMINATION CLEARANCE LEVELS FOR GENERIC CLEARANCE

Radionuclide	European Commission [II–3, II–4] (Bq/cm^2)		Germany [II–5] (Bq/cm^2)	United States of America [II–6][a] (Bq/cm^2)	
	Unrounded	Rounded	Rounded	Mean	95th percentile
H-3	25 000	10 000	100	1500	700
C-14	770	1000	100	1600	1100
Cl-36	130	100	100	29	7
Fe-55	1500	1000	100	110 000	30 000
Co-60	1	1	1	1	0.3
Sr-90	8.5	10	1	83	34
Cs-137	3.7	10	1	3.1	1.0
Eu-154	1.8	1	1	2.3	0.6
U-234	0.49	1	1	3.7	1.2
Pu-242	0.11	0.1	0.1	1.6	0.5
Am-241	0.12	0.1	0.1	1.1	0.3

[a] Reference [II–6] was only released in draft form and has never been incorporated into the official US policy on clearance.

with respect to assumptions about secondary ingestion and skin contamination. This is an area for further work on harmonization of approaches and surface contamination clearance levels used internationally.

THE APPLICATION OF VALUES FOR SURFACE CONTAMINATION IN THE IAEA REGULATIONS FOR THE SAFE TRANSPORT OF RADIOACTIVE MATERIAL

II–9. SSR-6 (Rev. 1) [II–2] specifies surface contamination values of 0.4 Bq/cm^2 for beta and gamma emitters and low toxicity alpha emitters and of 0.04 Bq/cm^2 for all other alpha emitters in the definition of contamination (for fixed and non-fixed contamination). It also specifies values of 4 Bq/cm^2 and 0.4 Bq/cm^2, respectively, for the limit of surface contaminated objects (SCO-I) and surface contamination on packages and conveyances, relating to non-fixed contamination only. These limits are applicable when averaged over an area of 300 cm^2 on any part of the surface.

II–10. The values for surface contamination specified in SSR-6 (Rev. 1) [II–2] were originally derived using a very simple model [II–7]. A review of this model, together with the proposal of new modelling approaches for limiting the surface contamination on packages and conveyances, is presented in Ref. [II–8], from which it is clear that the values of surface contamination specified in SSR-6 (Rev. 1) [II–2] are not applicable to clearance.

REFERENCES TO ANNEX II

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Annex III

EXAMPLES OF ACTIVITY CONCENTRATION VALUES FOR SPECIFIC CLEARANCE

III–1. Examples of specific clearance levels in terms of activity concentration that have been derived include those for scrap metal for melting [III–1] and those for buildings for demolition or reuse [III–2, III–3]. Specific clearance levels for disposal of waste in landfill sites have been adopted in Belgium, Germany and the United Kingdom, as described in this annex. This annex also includes an IAEA approach to specific clearance for landfill disposal and an example from the United Kingdom of the clearance of certain liquids.

EXAMPLE FROM THE UNITED KINGDOM OF SPECIFIC CLEARANCE OF SOLID WASTE

III–2. The conditions relating to specific clearance[1] of very low level waste for disposal in landfills in the United Kingdom [III–4] are given Table III–1.

TABLE III–1. SPECIFIC CLEARANCE OF VERY LOW LEVEL WASTE FOR DISPOSAL IN LANDFILL SITES IN THE UNITED KINGDOM [III–4]

Type of radioactive waste	Maximum concentration of radionuclides	Maximum quantity of waste to be disposed of per calendar year (Bq/a)
Solid radioactive waste, with no single item > 4×10^4 Bq	4×10^5 Bq per 0.1 m^3 for the sum of all radionuclides	2×10^8
Solid radioactive waste containing tritium and C-14 only, with no single item > 4×10^5 Bq	4×10^6 Bq of tritium and C-14 per 0.1 m^3	2×10^9

[1] In the United Kingdom, specific clearance is called ‘specific (conditional) exemption’ for reasons of continuity of regulatory terminology.

EXAMPLE FROM BELGIUM OF SPECIFIC CLEARANCE OF SOLID WASTE

III–3. FBFC International is a nuclear installation situated in Dessel, Belgium. From 1960 until 2012, it produced fuel assemblies of uranium for nuclear power plants. In 2011, it was decided to shut down the installation for economic reasons.

III–4. During operation, water was used in contaminated zones as part of the production process and for personnel utilities (washrooms). This contaminated water circulated through underground pipes and was collected in the water treatment building, where it was treated before discharge to the environment.

III–5. An initial decommissioning survey identified several leaks in the underground pipes (mostly at the joints between pipes), resulting in deposition of small amounts of uranium in the surrounding soil (mainly sand). In addition, slightly contaminated sand was found in canals outside the facility site due to sedimentation. This sand was brought on the site and was part of the contaminated soil to be disposed of. The total volume of material was estimated to be 8300 m^3, with a total mass of 12 000 tonnes.

III–6. For generic clearance of soil, a level of 1 Bq/g for U_{tot} (which is the sum of all radioisotopes of uranium) is specified by the Belgian nuclear regulatory body. Measurements of soil samples gave results slightly above this activity concentration.

III–7. In accordance with the Belgian regulations, specific clearance is possible based on a licence from the regulatory body. In the licence application, the operating organization has to propose a specific clearance level below the generic exemption level and include a radiological impact study demonstrating that the individual dose criterion of the order of 10 µSv in a year will not be exceeded and that the collective dose will be below 1 person-Sv in a year.

III–8. The operating organization decided to apply for a specific clearance licence and to radiologically sort the sand into three categories, with the final destination based on the activity concentration, as follows:

(a) < 1 Bq/g: generic clearance.
(b) 1 Bq/g to 10 Bq/g: disposal in conventional landfill.
(c) > 10 Bq/g: radioactive waste to be transferred to the Belgian waste management agency.

Impact study for disposal in conventional landfill

III–9. The selected landfill site for disposal of the uranium contaminated soil is situated in the province of Antwerp and is intended for hazardous waste. It also accepts waste containing naturally occurring radioactive material. Specific zones of the disposal site will be used for the low level contaminated soil from FBFC International.

III–10. An internal impact study [III–5] considered the exposure scenarios for handling 8300 m^3 of soil as a worst case, assuming a contamination level of 10 Bq/g U_{tot}. This amount is equal to the total volume of the soil from FBFC International, and the expected volume for disposal in the landfill site will be significantly lower.

III–11. The results of the impact study are summarized in Table III–2 for workers who might be affected by the specific clearance process.

III–12. Table III–2 shows that the most exposed workers will be those involved in the unloading of the soil on the landfill site. It is assumed that workers perform only one of the listed tasks and that the job of unloading would be shared by two workers. Therefore, it was concluded that an activity concentration level of 10 Bq/g U_{tot} in soil would not give an individual annual effective dose in excess of 10 μSv to any of the workers considered.

III–13. A similar analysis was performed for members of the public of all age categories living in the vicinity of the landfill, cultivating a garden and walking on the landfill. This analysis led to a similar conclusion.

III–14. A dose calculation was also performed to estimate the impact of on-site sorting of sand on workers. The result was also found to be below 10 μSv in a year, if workers used the protective equipment typical for such work. Nevertheless, these workers are considered as occupationally exposed workers by the operating organization.

III–15. On the basis of this study, a licence for specific clearance up to an activity concentration of 10 Bq/g U_{tot} in sand was granted by the regulatory authority for removal of a maximum of 12 450 tonnes of waste to a conventional landfill for hazardous waste.

III–16. Since the activity concentration remains below 10 Bq/g U_{tot}, no licence for the transport of the soil to the landfill site is needed.

TABLE III–2. RESULTS OF THE IMPACT STUDY FOR DISPOSAL IN CONVENTIONAL LANDFILL

Type of worker	Type of exposure	Annual dose (μSv)
Transporter (drivers) of soil	External	1.5
	Inhalation	Negligible
	Ingestion	Negligible
	Total	1.5
Workers on landfill during unloading	External	7.3
	Inhalation	7.4
	Ingestion	0.2
	Total	14.9
Workers on landfill during disposal	External	3.3
	Inhalation	1.6
	Ingestion	0.1
	Total	5.0
Other workers on landfill	External	2.8
	Inhalation	Negligible
	Ingestion	Negligible
	Total	2.8

Traceability of the conditionally cleared soil

III–17. The information about the amount and the location of the cleared soil will be preserved in two sets of documents per transported container:

(a) Departure document prepared by FBFC International containing the container identifier, type and amount of packages, radionuclide and activity content, total mass, and the date of pick-up.

(b) Reception document prepared by the landfill operating organization containing the container identifier, time of delivery, total mass, and location of on-site disposal.

The documents have to be kept by the operating organization (FBFC International) for 30 years. At the time of licence termination, the documents will be transferred to the regulatory body.

EXAMPLE FROM GERMANY OF SPECIFIC CLEARANCE OF SOLID WASTE

III–18. Clearance levels for material to be disposed of in landfills or for incineration in waste incinerator plants have been derived in Germany by the German Commission on Radiological Protection on behalf of the Federal Ministry for the Environment, Nature Conservation and Nuclear Safety. The values are provided in Ref. [III–6] and were included in the German Radiation Protection Ordinance in 2011. The clearance is meant for conventional landfill disposal sites and for conventional waste incineration plants that are also used for ordinary refuse (i.e. not for landfills or incineration plants with a radiation protection licence of any kind).

III–19. The clearance levels were derived on the basis of a complex radiological model that took into account all relevant exposure scenarios and exposure pathways, from the point of clearance until the material reaches its final destination (i.e. emplacement in the landfill site or burning in the waste incineration plant). The structure of this model is shown in Fig. III–1.

III–20. The model in Fig. III–1 takes into account scenarios for the workers transporting the material to the landfill site or the waste incineration plant and for the general public. Scenarios describing the gradual release of radionuclides via environmental pathways and subsequently entering into the human food chain include (a) airborne dust and its deposition on the ground and (b) leaching due to precipitation of radionuclides from the waste to surface waters and then (after a few hundred years) to groundwater, and the use of the water for drinking, irrigation and preparation of food.

III–21. The model distinguishes between ‘small’ quantities (i.e. up to 100 tonnes per year) and ‘large’ quantities (i.e. up to 1000 tonnes per year) to landfill or to an incineration facility. This would enable facilities generating smaller quantities of waste (e.g. medical, research and industrial facilities) to be treated differently

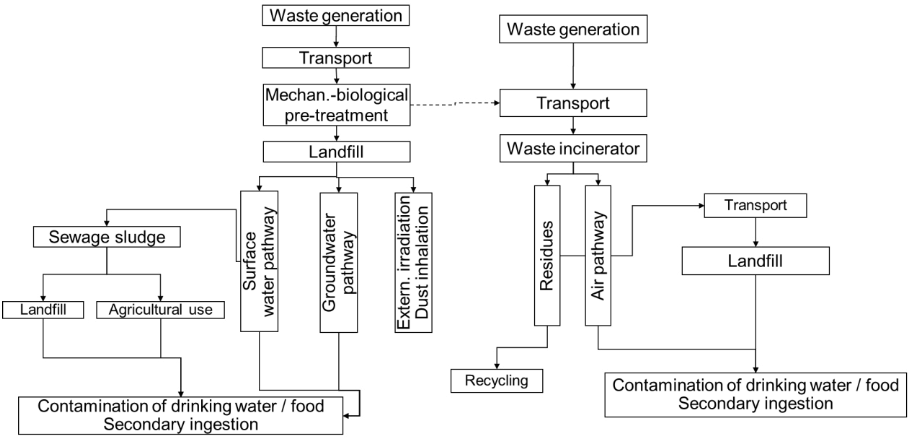

FIG. III–1. Radiological model used in Germany for calculation of clearance levels for disposal in landfills and for incineration [III–4].

TABLE III–3. EXAMPLE OF CLEARANCE LEVELS FOR SPECIFIC CLEARANCE IN GERMANY

Radionuclide	Clearance for landfill disposal(Bq/g)		Clearance for incineration(Bq/g)	
	Up to 100 t/a	Up to 1000 t/a	Up to 100 t/a	Up to 1000 t/a
Co-60	6	2	7	2
Sr-90	6	0.6	40	4
Cs-137	9	3	9	1

from facilities generating large quantities of waste, such as nuclear power plants undergoing decommissioning.

III–22. Clearance levels were derived for a large number of radionuclides. Examples are given in Table III–3 for ^{60}Co, ^{90}Sr and ^{137}Cs.

III–23. The clearance levels are applied at several landfills and incineration plants. Their use involves normal administrative procedure for waste disposal under the jurisdiction of the waste authorities in addition to the clearance process under the jurisdiction of the radiation protection authorities.

EXAMPLE FROM IAEA PUBLICATION ON SPECIFIC CLEARANCE OF SOLID WASTE FOR DISPOSAL ON LANDFILL SITES

III–24. Reference [III–7] provides information on a study of the derivation of specific clearance levels for disposal of material in a landfill. Large amounts of solid material with a low level of radioactivity are encountered in decommissioning projects; consequently, the study started with a focus on the specific clearance of decommissioning waste for disposal in landfills.

III–25. For the purpose of the study, a new tool, called ‘Clearance Tool’ was developed for the derivation of specific clearance levels for different types of landfill and, ultimately, for the reuse and recycling of materials from decommissioning projects. The dose criteria and scenarios for the derivation of these clearance levels are based on Ref. [III–8] and IAEA Safety Standards Series No. GSR Part 3, Radiation Protection and Safety of Radiation Sources: International Basic Safety Standards [III–9], namely 10 μSv in a year for realistic scenarios and 1 mSv in a year for low probability scenarios.

III–26. The derivation of the specific clearance levels focused on radionuclides relevant to nuclear power plants. This annex presents the results for a subset of the radionuclides considered: ^{90}Sr, ^{99}Tc, ^{106}Ru, ^{131}I, ^{134}Cs, ^{137}Cs, ^{144}Ce, ^{239}Pu, ^{241}Pu and ^{241}Am.

III–27. A basic set of exposure scenarios was used to describe the disposal of material in an ordinary landfill without any special radiation protection arrangements. The scenarios took into account the exposure of workers that might arise from transport of the material to the site, handling of the material at the landfill, and releases of radionuclides to the atmosphere in the case of a landfill fire. The exposure of members of the public living close to the landfill was also considered. The scenarios included consideration of the consequences of a controlled or uncontrolled release of leachates to groundwater and surface water.

III–28. The lifetime of the landfill is divided into two phases — the operational phase and the post-operational phase — in which a distinction is made between the period during and after institutional control. For the post-operational phase of a landfill site, a recreational use scenario was considered, including the possibility of small excavations being performed in the landfill. In addition, an intrusion scenario was considered in which houses are built on the site of the original landfill. In this scenario, only the exposure to people living in these houses was considered, as the exposure time would be longer and there would be additional exposure pathways (e.g. ingestion of contaminated garden products), leading to substantially higher exposures than those to construction workers building the houses.

III–29. A resident living on the landfill after closure and after the end of institutional control was treated as an unlikely scenario. Therefore, for this scenario a dose criterion of 1 mSv in a year was used.

III–30. The study considered three generic landfill types:

(a) Landfill for inert waste;
(b) Landfill for municipal non-hazardous waste;
(c) Landfill for hazardous waste.

The different types of landfill were assumed to have different properties in terms of the bottom liner, leachate collection system and top cover.

III–31. The calculation tool developed as part of this study[2] may be used to calculate specific clearance levels that take specific site features into account. The parameter values used in the calculation for food ingestion were updated in accordance with Ref. [III–10].

III–32. Both deterministic and probabilistic calculations were performed in the study. The results of the deterministic calculations for the selected subset of radionuclides are shown in Table III–4. Logarithmic rounding has not been applied to the calculated results, which needs to be taken into account when comparing the results with the generic clearance levels from GSR Part 3 [III–9].

EXAMPLE FROM THE UNITED KINGDOM OF CLEARANCE OF LIQUIDS

III–33. Two studies on clearance levels for liquids were conducted in the United Kingdom: one for aqueous liquids and one for non-aqueous liquids. The study on non-aqueous liquids [III–11] demonstrated that the clearance levels for solids specified in Ref. [III–12] are suitable for use for generic clearance of non-aqueous liquids for most radionuclides. Some exceptions are ^{32}P, ^{33}P, ^{35}S, ^{65}Zn and ^{99}Tc: for these radionuclides, it may be necessary to proceed to specific clearance by, for example, restricting the activity concentration or applying disposal conditions. Further information is given in Ref. [III–11].

[2] ECOLEGO, available at http://ecolego.facilia.se/ecolego/show/Ecolego%20player.

TABLE III–4. RESULTS OF DETERMINISTIC CALCULATIONS OF ACTIVITY CONCENTRATION LEVELS FOR SPECIFIC CLEARANCE FOR DISPOSAL OF WASTE IN CONVENTIONAL LANDFILLS [III–7]

Radionuclide	Generic clearance level (from GSR Part 3 [III–9]) (Bq/g)	Activity concentration level for disposal in landfills (Bq/g)		
		Inert waste landfill	Municipal non-hazardous waste landfill	Hazardous waste landfill
Sr-90	1	6.3	12	46
Tc-99	1	2.8	2.8	5.3
Ru-106	0.1	11	11	14
I-131	10	87	87	110
Cs-134	0.1	0.75	0.75	1.0
Cs-137	0.1	1.7	1.7	2.2
Ce-144	10	45	45	48
Pu-239	0.1	2.7	2.7	4.2
Pu-241	10	64	64	97
Am-241	0.1	3.2	3.2	40

III–34. A similar study performed for aqueous liquids [III–13] and based on a dose criterion of 10 μSv in a year produced clearance levels ranging from 10^{-4} Bq/L to 10^{3} Bq/L, with 80% of the values being in the range 0.01–1 Bq/L. The study recommended that the volume of liquids containing radionuclides at these activity concentrations that could be disposed of to a sewer be restricted to 3000 m^3/a. The study also identified the difficulties in making laboratory measurements to demonstrate compliance with these clearance levels for some radionuclides.

REFERENCES TO ANNEX III

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[III–3] EUROPEAN COMMISSION, Definition of Clearance Levels for the Release of Radioactively Contaminated Buildings and Building Rubble, Radiation Protection 114, Office for Official Publications of the European Communities, Luxemburg (1999).

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[III–9] EUROPEAN COMMISSION, FOOD AND AGRICULTURE ORGANIZATION OF THE UNITED NATIONS, INTERNATIONAL ATOMIC ENERGY AGENCY, INTERNATIONAL LABOUR ORGANIZATION, OECD NUCLEAR ENERGY AGENCY, PAN AMERICAN HEALTH ORGANIZATION, UNITED NATIONS ENVIRONMENT PROGRAMME, WORLD HEALTH ORGANIZATION, Radiation Protection and Safety of Radiation Sources: International Basic Safety Standards, IAEA Safety Standards Series No. GSR Part 3, IAEA, Vienna (2014).

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[III–11] MOBBS, S.F., Investigation of Possible Exemption or Exclusion Levels for Non-aqueous Liquids to Support the RSA93 Exemption Order Review, HPA-CRCE-006, Health Protection Agency, Chilton, Oxfordshire (2010).

[III–12] EUROPEAN COMMISSION, Practical Use of the Concepts of Clearance and Exemption: Part II — Application of the Concepts of Exemption and Clearance to Natural Radiation Sources, Radiation Protection 122, Office for Official Publications of the European Communities, Luxemburg (1999).

[III–13] EWERS, L.W., MOBBS, S.F., Derivation of Liquid Exclusion or Exemption Levels to Support the RSA93 Exemption Order Review, HPA-CRCE-005, Health Protection Agency, Chilton, Oxfordshire (2010).

Annex IV

EXAMPLE OF THE APPLICATION OF THE CLEARANCE CONCEPT IN A SMALL NUCLEAR MEDICINE FACILITY

IV–1. This annex is based on the Practical Guide of the Ibero-American Forum of Radiological and Nuclear Regulatory Agencies, developed through the project "Implementation of the Clearance Concept and Criteria for Small Nuclear Installations Handling Radioactive Waste" [IV–1].

IV–2. Certain facilities conducting practices with unsealed or sealed radioactive sources use radionuclides with short and very short half-lives (less than 100 days). Examples of such facilities are small research laboratories, medical departments and industrial applications in which such radioactive sources are used, processed or stored. The activity of the radionuclides used in such facilities varies in accordance with the practice. For example, for medical purposes, the activity used can vary from less than 1 MBq up to 100 GBq depending on whether the facility is conducting medical research, clinical therapy or diagnostic procedures. Information on unsealed sources and sealed sources and their range of activity per practice can be found in Refs [IV–1, IV–2]. In such facilities, moderate amounts of radioactive waste[1] are generated, and this waste needs to be managed to ensure the protection of people and the environment. With a proper methodology, the best option for management of a significant volume of these radioactive wastes could be clearance.

IV–3. The facilities considered in this annex are those that have standardized procedures for the safe use of radioactive sources. A standardized methodology for the clearance of material within these practices is a useful means of ensuring the safe management of the radioactive waste by the operating organizations and, at the same time, facilitating the regulatory process, including record keeping, regulatory inspections and verification of compliance with the relevant standards and regulations.

IV–4. This annex describes, as an example, a methodology applicable for the solid radioactive waste generated by a nuclear medicine facility. This methodology could assist operating organizations and regulatory bodies to protect people and the environment effectively and efficiently by using the concept of clearance in a practical way.

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[1] Moderate amounts of radioactive waste means less than 3 tonnes per year per facility [IV–3].

IV–5. Solid radioactive waste in a nuclear medicine facility may be generated in the form of paper, plastics, contaminated materials, discarded radiopharmaceutical containers, bandages, protective clothing, plastic sheets and bags, gloves, masks, filters, overshoes, paper wipes, towels, metal and glass, hand tools and discarded contaminated equipment [IV–1]. Liquid radioactive waste generated in a nuclear medicine facility may include contaminated water and other effluents, waste arising from chemical processing and decontamination solutions, blood or other body fluids, discarded liquid radiopharmaceuticals, wound or oral discharges, and urine [IV–1]. Such waste needs special consideration by the treatment systems in the facility, making it difficult to provide a generic example.

IV–6. This annex specifically considers a nuclear medicine facility that is authorized to use the following techniques:

(a) Gamma radiography studies for diagnostics and follow-up with ^{99m}Tc;
(b) Thyroid function tests and treatment of thyroid cancer with ^{131}I.

The maximum activity of each radionuclide and the number of patients per week authorized in the facility for these techniques are shown in Table IV–1.[2]

IV–7. The following is a non-exhaustive list of the types of solid radioactive waste that may be generated from the use of ^{99m}Tc and ^{131}I [IV–2]:

(a) Solid compactable waste (e.g. papers, cottons, chiffon gloves);
(b) Metals (syringe needles);
(c) Glass (vials).

TABLE IV–1. MAXIMUM ACTIVITIES AND NUMBER OF PATIENTS FOR THE EXAMPLE NUCLEAR MEDICINE FACILITY

Practice	Radionuclide	Half-life	Type of emitter	Patients per week	Maximum activity per week (GBq)
Diagnostic	Mo–Tc-99m generator	6.03 hours	Gamma	70	40
Diagnostic and therapy	I-131	8.04 days	Gamma	45	74

[2] This example is taken from a real nuclear medicine facility. All information presented in the table is based on a real case.

METHODOLOGY FOR THE CLEARANCE OF WASTE IN SMALL FACILITIES AND ACTIVITIES

IV–8. The practical methodology presented in this annex for the clearance of waste arising in small facilities and activities consists of the following main steps:

(1) Collection and segregation;
(2) Measurement or estimation of the activity concentration of the waste;
(3) Management options (i.e. storage, decay, clearance, disposal);
(4) Record keeping.

Step 1: Collection and segregation

IV–9. Appropriate collection and segregation of residual radioactive material is an important step of the methodology and is necessary in order to minimize hazards associated with the waste and to facilitate subsequent management of the waste. It is normally better that the waste collection and segregation be performed at the time and place where the waste is generated. This process needs to be suitable for the radionuclide and its half-life and for the physical and chemical form and other properties of the waste, such as pathogenic or physical hazards (e.g. sharp objects).

IV–10. Only one radionuclide is used in each medical procedure; this makes it easy to segregate waste by individual radionuclides [IV–1]. However, any significant presence of other radionuclides resulting from the production process, either as impurities or as decay products, needs to be properly taken into account.

IV–11. In some cases, it may be convenient to segregate wastes in accordance with their half-life, for example wastes with a half-life of about 10 hours or less, wastes with a half-life of less than 10 days and wastes with a half-life of less than 100 days.

IV–12. In other cases, the solid wastes can be segregated according to their physical characteristics, such as compactible or non-compactible, and incinerable or non-incinerable.

IV–13. To ensure adequate collection and segregation, the nuclear medicine facility needs to be provided with suitably labelled containers and bags. Further information on segregation and labelling of wastes in a nuclear medicine facility is provided in Ref. [IV–1].

Step 2: Measurement or estimation of activity concentration of the waste

IV–14. The methodology to measure or estimate the activity concentration of the solid waste in a nuclear medicine facility needs to be practical and simple. Such an approach is sufficient for the purposes of clearance in this type of facility owing to the low activities and the short half-lives involved.

IV–15. Once the waste is collected and segregated as described in step 1, a radiological characterization is performed to determine the initial activity concentration or total activity for each waste stream. Different methods for the measurement or estimation of initial activity concentration or initial total activity in the wastes are used depending on the geometry of the waste containers and the properties of the materials or items in the waste. Consequently, each nuclear medicine facility needs to establish its own measurement or estimation procedure relevant to its own circumstances and the technical properties of any equipment used to perform measurements. Further information is provided in Ref. [IV–1], and a method for clearance measurements for medical waste is provided in Ref. [IV–4].

IV–16. The radionuclide and activity involved in each medical procedure, as well as the total activity authorized, are expected to be known with precision. Hence, the residual activity in the waste can be estimated by means of a simple balance of activity and corrections for radioactive decay. However, if impurities or decay products are present in addition to the radionuclides used in the procedure, their contribution needs to be taken into account, as their relative importance can increase significantly over time.

Step 3: Management options (storage, decay, clearance, disposal)

IV–17. Once the measurement or estimation of the activity concentration of the waste has been completed in step 2, the waste management options need to be chosen in accordance with the process shown in Fig. IV–1.

IV–18. The result of the measurement or the estimation of the activity concentration of the waste from Step 2 should be compared with the relevant clearance level for the radionuclide involved, as specified in IAEA Safety Standards Series No. GSR Part 3, Radiation Protection and Safety of Radiation Sources: International Basic Safety Standards [IV–5].

IV–19. As shown in Fig. IV–1, if the activity concentration of the waste is above the clearance level and the half-life of the radionuclide is below 100

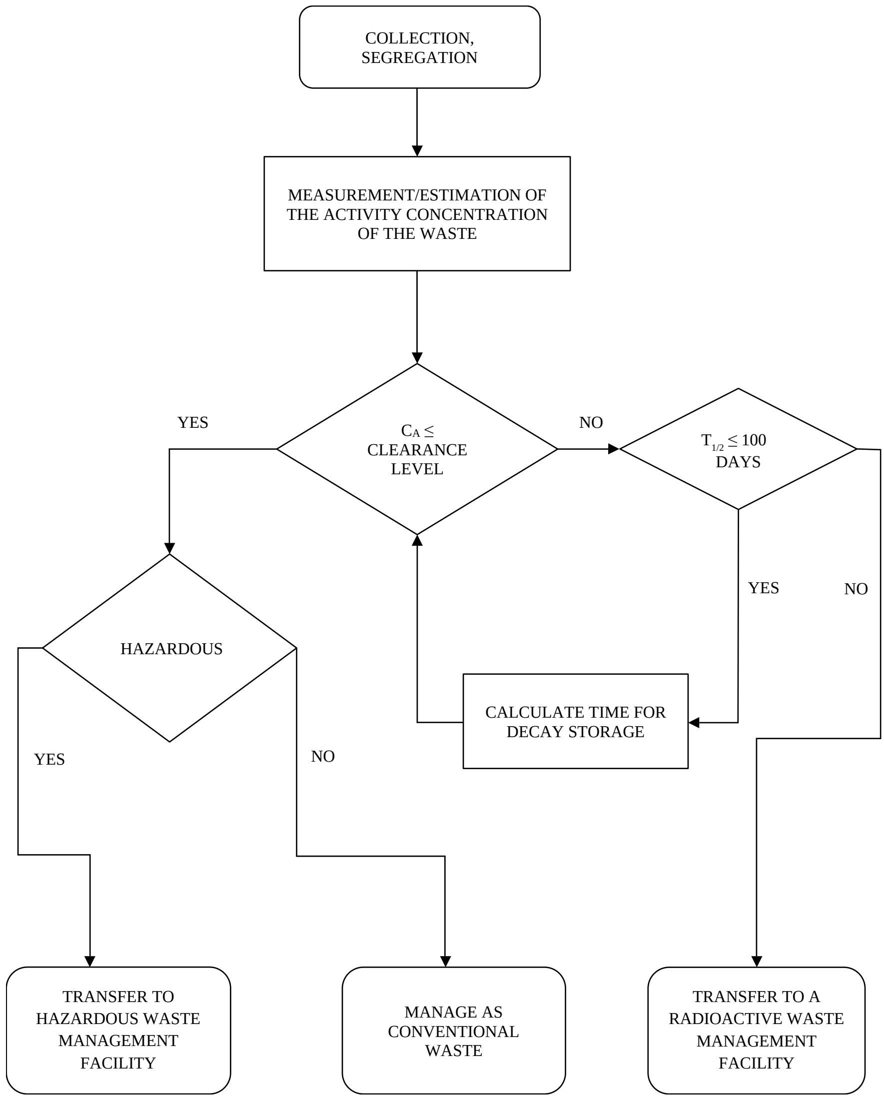

FIG. IV–1. Process for selection of the optimal waste management option.

days, the waste could be stored to allow radioactive decay until the authorized clearance levels are met. In some cases, the activity concentration may be so high that it would need long term storage to achieve this. In such cases, it may be better to transfer the waste to a radioactive waste management facility for adequate treatment or disposal, in accordance with national regulations.

IV–20 For the example of the nuclear medicine facility, the storage decay time for each solid waste stream is shown in Table IV–2.

IV–21 Once the calculation of the storage decay time until the clearance level is met has been performed, the wastes are transferred to the radioactive waste storage room for temporary storage. In addition, the wastes are labelled, and the label should include the radionuclide, activity concentration, date of storage and probable clearance date.

IV–22. From Table IV–2, it can be seen that some of the waste need only be stored for a few days before reaching the clearance levels. Other waste needs to be stored for several weeks: the waste stream that needs the longest period of decay storage is that generated by clinical therapy, where the highest activity levels are used.

IV–23. After the necessary decay storage time has expired, a further measurement may be performed to confirm that the clearance levels have been met. In the case of a nuclear medicine facility, this determination could be based on gamma radiation measurements around the outside of waste containers. Further information is provided in Ref. [IV–1].

IV–24. As shown in Fig. IV–1, cleared waste is managed as conventional waste or as hazardous waste, as appropriate. Conventional waste can be disposed of in municipal landfills with household waste without any further consideration. Hazardous waste will normally be sent to a hazardous material landfill.

TABLE IV–2. STORAGE DECAY TIME FOR SOLID WASTES

Waste bag	Radionuclide	T ½ (days)	Initial activity concentration (kBq/g)	Clearance level[a] (Bq/g)	Storage decay time (days)
1223	Tc-99m	0.25	90.4	100	2.46
1224	Tc-99m	0.25	195.3	100	2.73
3220	I-131	8.04	42.4	100	70.17
3221	I-131	8.04	44.9	100	70.85

[a] From table I.1 of GSR Part 3 [IV–5].

IV–25. Once compliance with the clearance levels is verified, it is important to remove any labels containing radioactive warning symbols from the waste packages before proceeding to dispose of them as conventional or hazardous waste.

Step 4: Record keeping

IV–26. Nuclear medicine facilities need to implement an adequate record keeping system to demonstrate that the clearance procedure has been performed within the framework of a quality management programme and that wastes are traceable from the time they are generated to the time of final disposal. The records are important to both operating organizations and the regulatory body, and the aim of the system is to track waste at each step in the waste management process. Records of solid waste might include the following information:

(a) Identification numbers for individual waste containers;
(b) Radionuclide(s);
(c) Weight of waste in each container;
(d) Results of initial measurements and date;
(e) Activity or activity concentration, as determined by measurements or estimation;
(f) Decay time needed to meet clearance levels;
(g) Estimated date of clearance;
(h) Result of final measurements;
(i) Actual date of clearance and destination.

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Annex V

EXAMPLE OF A NATIONAL APPROACH TO THE CLEARANCE OF SCRAP METAL

V–1. The implementation of a clearance process for solid materials will depend on many details, such as the type of material (e.g. laboratory waste, concrete rubble, scrap metal), the origin of the material (e.g. a hospital, a nuclear power plant), the way in which the radiological characterization is performed (e.g. immediately prior to dismantling of a component or after the dismantling) and whether the material will be processed. As an example, Fig. V–1 (which is from a national standard for the clearance of scrap metal [V–1]) provides an overview of the clearance process using three approaches, as follows:

(a) Approach 1: Facility-wide radiological characterization in advance, prior to dismantling.
(b) Approach 2: Characterization by system just before dismantling.
(c) Approach 3: Characterization on the basis of sampling during decontamination.

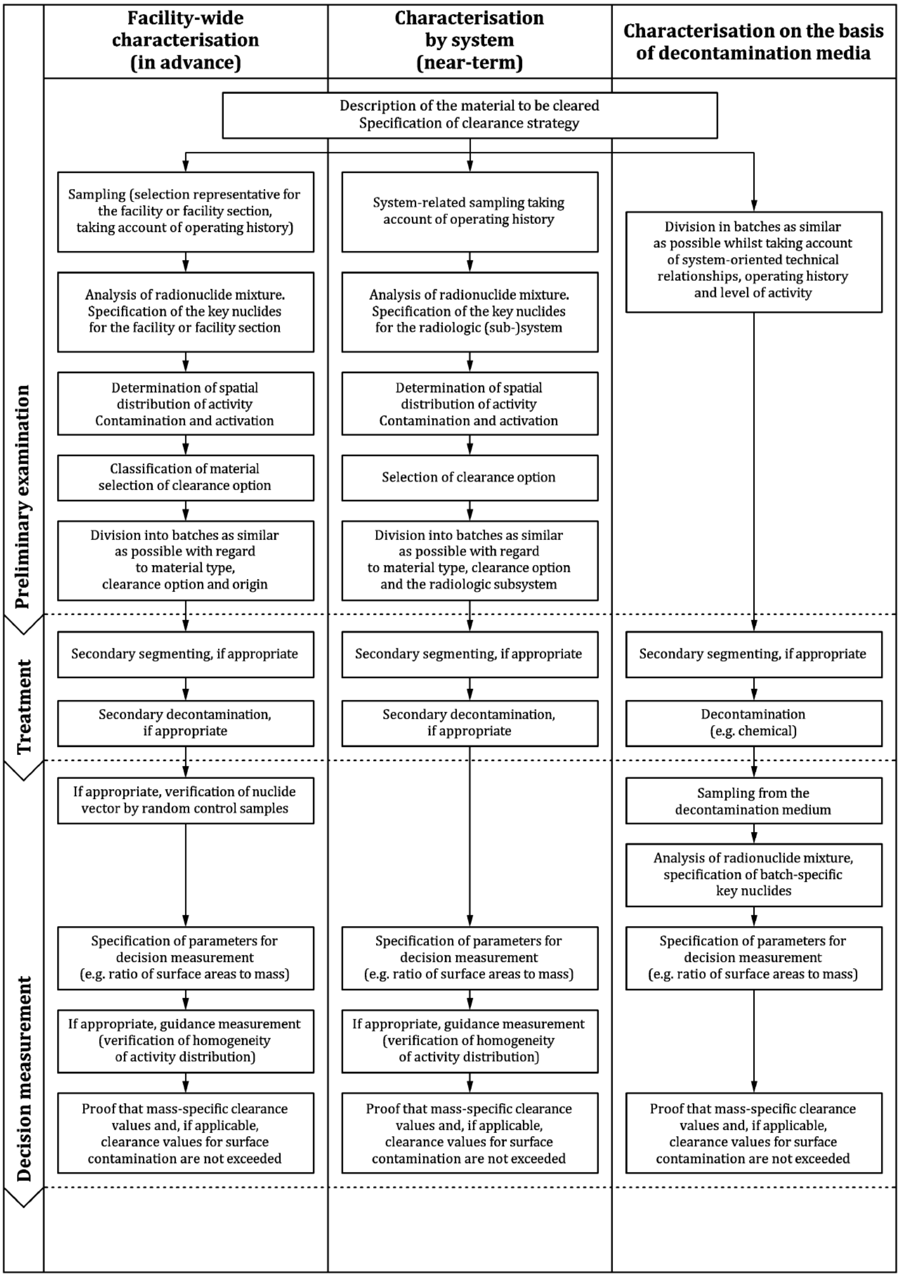

FIG. V–1. An example of the clearance process for scrap metal [V–1].

REFERENCE TO ANNEX V

[V–1] GERMAN INSTITUTE FOR STANDARDIZATION, Activity Measurement Methods in the Clearance of Radioactive Substances and Components of Nuclear Facilities — Part 4: Contaminated and Activated Metal Scrap, DIN 25457-4, DIN, Berlin (2013).

Annex VI

EXAMPLE METHOD FOR THE SELECTION OF SIGNIFICANT RADIONUCLIDES FOR CLEARANCE PURPOSES

VI–1. An example of a method for selection of significant radionuclides for solid materials is provided in Fig. VI–1 [VI–1] and contains two steps, as follows:

(a) The first step determines which radionuclides are included in the overall evaluation. A key radionuclide is selected among the easy to measure radionuclides, which gives a relatively high value of C/CL, where C is the evaluated radionuclide concentration and CL is the clearance level for radionuclides. Then a measure of significance is introduced as the relative ratio $(C_j/CL_j)/(C/CL)_{\text{key}}$, where $(C/CL)_{\text{key}}$ is the ratio for the key radionuclide, and an initial set of significant radionuclides is selected so that all the radionuclides, j (see para. VI–2), satisfy the condition $(C_j/CL_j)/(C/CL)_{\text{key}} > 0.01$.

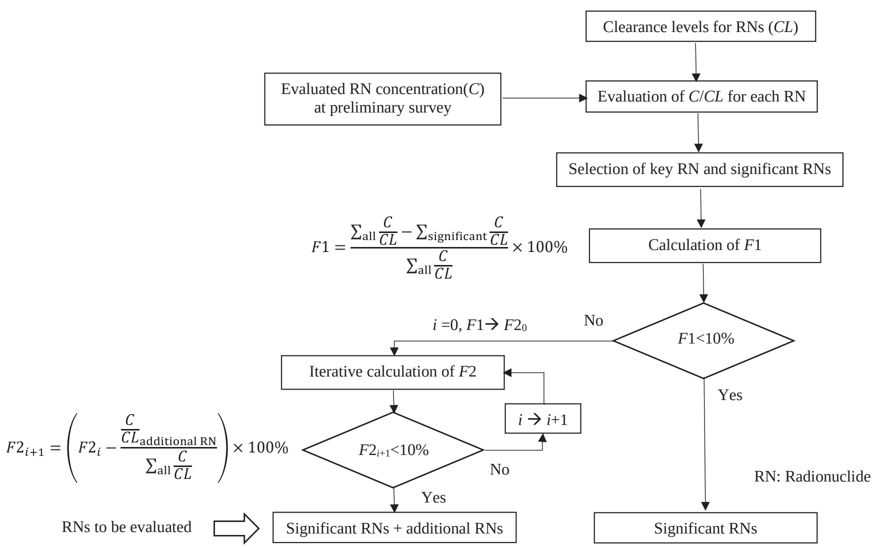

FIG. VI–1. An approach to selection of the significant radionuclides to be evaluated (based on Ref. [VI–1]).

(b) In the second step, a subgroup of the radionuclides is selected so that with the smallest possible number of selected radionuclides, the sum of *C*/*CL* for that selection is more than 90% of the sum of *C*/*CL* of all radionuclides. The key radionuclide is always included in the selection. This is done in an iterative process. The sums of *C*/*CL* for all radionuclides and for selected significant radionuclides are calculated. If the relative difference between the two sums, *F*1, is less than 10% of the sum of *C*/*CL* for all radionuclides, the selection of radionuclides for evaluation is completed. If *F*1 is more than 10%, one additional radionuclide (with the highest *C*/*CL* among the remaining radionuclides) is added to the selection and a new value for *F*2 is calculated. The process continues until the difference, *F*2, is less than 10%.

VI–2. The following is an example of a selection method specified in a national regulatory standard for the evaluation of solid materials for clearance [VI–2]. This national regulatory standard specifies that *m* significant radionuclides are selected from *n* listed radionuclides so as to satisfy the following formulas:

$$\frac{\sum_{j=1}^{m}\frac{C_j}{CL_j}}{\sum_{k=1}^{n}\frac{C_k}{CL_k}} \geq 0.9,\ \frac{C_1}{CL_1} \geq \frac{C_2}{CL_2} \geq \ldots \geq \frac{C_m}{CL_m} \geq \ldots \geq \frac{C_n}{CL_n} \qquad \text{(VI–1)}$$

where

k is the assigned number for the radionuclide listed;
j is the assigned number for the selected radionuclide with high C_j/CL_j from the list for the evaluation;
C_k is the activity concentration (Bq/g) of the *k*th radionuclide in the material;
CL_k is the clearance level (Bq/g) of the *k*th radionuclide;
C_j is the activity concentration (Bq/g) of the *j*th radionuclide for the evaluation;
CL_j is the clearance level (Bq/g) of the *j*th radionuclide for the evaluation;
n is the total number of all listed radionuclides whose activity concentration limits are derived;

and *m* is the total number of the selected radionuclides for the evaluation.

VI–3. Once the significant radionuclides to be evaluated have been selected, the response of the monitoring system can be calculated in terms of the known radionuclide composition. This approach can also allow calculation of the likely variation in the response of contamination monitoring equipment for measurement of surface contamination. If the equipment, for example, has a good response over a wide range of beta energies, then the response will change quite rapidly with the degree of self-absorption. Sometimes, a correction factor needs to be introduced, particularly if a significant proportion of the emissions are of low energy. Alternatively, the equipment can be modified to shield the low energy emissions, so that the variations are reduced.

VI–4. The radionuclide composition and the scaling factors need to be re-evaluated as monitoring of material proceeds, particularly for materials from old, complicated facilities that cannot be characterized in detail before the clearance process begins. Simple means can sometimes be employed to check the constancy of the radionuclide composition, for example the ratio of the count rates from two different counting windows on a monitor or the influence of an absorber placed between the contaminated surface and the monitor. Gamma spectrometry is also a relatively simple process that can be employed to check the photon emitting component. A combination of gamma spectrometry and gross beta measurement can be used in cases where the main contaminants are beta and gamma emitters, for example ^{137}Cs and ^{137m}Ba.

VI–5. The following example demonstrates how to identify the clearance level for the key radionuclide that can be used for compliance measurements. In the example, the matrix has a mixture of two radionuclides, ^{14}C (clearance level = 1 Bq/g) and ^{60}Co (clearance level = 0.1 Bq/g), contributing 75% and 25% to the total activity, respectively. The derived clearance level (CL_{eff}) for a mixture of radionuclides in this example is as follows:[1]

$$\frac{1}{CL_{\text{eff}}} = \frac{0.75}{1\,\text{Bq/g}} + \frac{0.25}{0.1\,\text{Bq/g}} \qquad \text{(VI–2)}$$

$$CL_{\text{eff}} = 0.31\ \text{Bq/g} \qquad \text{(VI–3)}$$

VI–6. To demonstrate compliance with this effective clearance level for a mixture of radionuclides, one easy to measure radionuclide needs to be selected for measurements. In the example above, ^{60}Co is selected as the key radionuclide. The level to be used for compliance measurements, associated with this key radionuclide in this given mixture, is then calculated by multiplying the CL_{eff}

[1] CL_{eff} is used in this context to represent X_m from equation I.2 of IAEA Safety Standards Series No. GSR Part 3, Radiation Protection and Safety of Radiation Sources: International Basic Safety Standards [VI–3].

by the activity fraction of this key radionuclide. In the example given above, that level is 0.25 × 0.31 Bq/g. Hence, material with a ^{60}Co activity concentration below this level can be cleared.

VI–7. Following characterization, the clearance levels that are to be applied during the clearance process are selected. Sampling and monitoring for compliance with these clearance levels might identify additional radionuclides or changes in the scaling factors between different radionuclides. This will then feed back into additional characterization work, followed by a revised monitoring scheme for the clearance process.

REFERENCES TO ANNEX VI

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Annex VII

DEALING WITH UNCERTAINTIES IN CLEARANCE MEASUREMENTS

VII–1. When considering the clearance of materials, due account should be taken of the measurement uncertainties. The upper confidence level of the measurement result should take into account all significant sources of uncertainty and should be below the clearance level. Examples of this approach are provided in Refs [VII–1 to VII–3]. Examples of linking the measurement uncertainty to the detection limit are provided in sections 5.1–5.3 of Ref. [VII–4]. However, noting the overall conservatisms inherent in clearance levels (see paras 4.12–4.16 of this Safety Guide), care should be taken, whenever possible, not to introduce significant additional conservatisms through this mechanism (i.e. by choosing suitable measurements techniques and measurement setups, by selecting appropriate measurement times, etc.).

STATISTICAL UNCERTAINTIES ASSOCIATED WITH THE COUNTING PROCESS

VII–2. Radiation measurements for clearance involve counting events (e.g. detection of photons, beta particles or alpha particles) in instruments that either count the total number of events (e.g. total gamma measurements) or that have spectrometric capabilities. Examples of such instruments are contamination monitors (e.g. with proportional counters or scintillation detectors), bulk material monitors, in situ gamma spectrometers used on moderate or bulk quantities or on surfaces, laboratory gamma spectrometers and liquid scintillation counters used for analysing samples. When these instruments count events for a certain period of time, the result will form part of a distribution (usually Poisson or normal) around a best estimate. This difference between a single counting result and best estimate is a purely statistical effect and gives rise to an unavoidable measurement uncertainty.

UNCERTAINTIES ASSOCIATED WITH THE STATE OF THE SURFACE OF THE MEASURED MATERIAL

VII–3. The state of the surface of a material influences the efficiency of emission of alpha and beta radiation (see para. 4.46). The emission efficiency with respect to alpha contamination is strongly influenced by the thickness of the contaminated layer. A deeper penetration of activity is likely with porous materials, for example concrete and wood. Up to a small depth, the effect can be taken into account by adjusting the surface emission efficiency. However, if the thickness of the layer is significant, measurements of alpha and beta radiation are likely to be unreliable.

VII–4. This uncertainty is relevant for all surface measurements of alpha and beta emitters and needs to be included in the analysis of uncertainties. It is of minor relevance for measurements of surface contamination due to gamma radiation and for the use of in situ gamma spectrometers or measurement of samples.

UNCERTAINTIES ASSOCIATED WITH THE GEOMETRY AND THE SELF-SHIELDING OF THE MEASURED MATERIAL

VII–5. When performing clearance measurements, the monitoring instruments are normally calibrated for certain geometries of the material being monitored, and the calibration includes assumptions on self-shielding. Planar sources with a certain distance between the surface and the window of the contamination monitor (e.g. a few millimetres) are normally used for calibration purposes. In real measurement situations, the surface may be curved or uneven, or the distance to the instrument may need to be higher because of surface roughness. These differences can be taken into account by using correction factors or by using multiple calibration geometries that cover all likely situations.

VII–6. When undertaking bulk measurements of a large quantity of material, the effect of self-shielding by the material needs to be taken into account in the calibration process. In real measurements, there may still be deviations from the calibration conditions, for example because the material is more densely packed. This effect can be evaluated, for example by numerical simulations.

UNCERTAINTIES ASSOCIATED WITH THE ACTIVITY DISTRIBUTION IN THE MATERIAL

VII–7. During the calibration of instruments used for direct monitoring (i.e. surface measurements or bulk measurements) of materials for clearance, certain assumptions have to be made with respect to the spatial distribution of activity, either on the surface or in the volume of the material. The calibration of surface measurement instruments is often performed with homogeneous thin layer sources of known activity and surface emission rate; in contrast, real surfaces often have inhomogeneous or localized contamination. Similarly, the calibration of bulk monitors is often performed using calibration materials containing a homogeneous distribution of activity. In contrast, boxes containing scrap metal or building rubble typically have one or more (often many) areas of localized contamination. In either case, the measurement result needs to be corrected for the difference in spatial activity distribution between calibration and measurements.

UNCERTAINTIES ASSOCIATED WITH BACKGROUND RADIATION

VII–8. Background radiation needs to be measured separately so that it can be subtracted from the results of clearance measurements, as well as being needed for determination of the limit of detection. The background has to be regularly measured (e.g. before and after a measurement campaign during the working day). Even then, variations of the background during the measurement campaign can occur. The possible variation of the background radiation level therefore needs to be determined and included in the analysis of uncertainties.

UNCERTAINTIES ASSOCIATED WITH THE RADIONUCLIDE COMPOSITION

VII–9. The uncertainties associated with materials containing a mixture of radionuclides include those associated with the determination of scaling factors (see paras 3.21 and 3.22) between difficult to measure radionuclides and easy to measure radionuclides. The uncertainty in the determination of a scaling factor is associated with variations in the activity ratios from which it is derived (e.g. as a mean value together with a standard deviation). Usually, scaling factors for key radionuclides will be derived on a conservative basis so that the activity of the difficult to measure radionuclides will not be underestimated, taking into consideration the difference between the mass of samples used for determination of the scaling factors and the total mass of the material to be cleared.

VII–10. The uncertainty in the determination of the radionuclide composition or scaling factors needs to be taken into account in the analysis of uncertainties. The way in which uncertainties in the derivation of the radionuclide composition and the associated scaling factors are treated can give rise to a high degree of conservatism in the clearance process. For example, if a scaling factor is derived from a series of activity measurements of difficult to measure radionuclides and key radionuclides, it may be better to use an appropriate upper confidence level of the estimated mean value of the activity ratio as the scaling factor, rather than the maximum measured ratio.

UNCERTAINTIES ASSOCIATED WITH THE WIPING EFFICIENCY OF INDIRECT SURFACE ACTIVITY MEASUREMENTS

VII–11. When surface contamination levels are determined by wipe tests, assumptions on the wiping efficiency need to be made to estimate the level of removable surface contamination. Often, it is conservatively assumed that 10% of the removable activity is transferred to the wipe [VII–5] to take into account that the actual wiping efficiency is hard to determine and will depend on many factors. Even if the wiping efficiency is determined under well defined conditions, the chemical and physical boundary conditions in real measurement environments during the taking of wipe tests may deviate from the idealized conditions. Possible differences between the idealized and real wiping efficiency need to be included in the analysis of uncertainties.

UNCERTAINTIES ASSOCIATED WITH THE PRESENCE OF RADIONUCLIDES OF NATURAL ORIGIN AND OTHER RADIONUCLIDES TO BE DISREGARDED IN CLEARANCE MEASUREMENTS

VII–12. Radionuclides of natural origin can be present in materials being considered for clearance, in particular in building rubble, where radionuclides of the ^{238}U and ^{232}Th decay chains, as well as ^{40}K, may contribute to the measurement result, in particular for gross gamma measurements (performed using bulk monitors) and for measurements with surface contamination monitors. This is likely to be less important for measurements with in situ or laboratory based gamma spectrometry. Radionuclides of natural origin that were not part of the practice giving rise to the material to be cleared can be disregarded, and therefore their contribution can be subtracted from the gross measurement results.

VII–13. The activity concentration of radionuclides of natural origin will need to be determined in advance from an adequate set of samples. However, the activity concentration of these radionuclides in the materials being monitored might differ from this previously determined value. Hence, this difference has to be determined and needs to be included in the analysis of uncertainties for gross gamma measurements and surface contamination monitor measurements on building rubble or on other materials where radionuclides of natural origin are expected to be present. This also applies to other radionuclides in the material that are to be disregarded, for example ^{137}Cs from the fallout resulting from past atmospheric nuclear weapon tests.

UNCERTAINTIES ASSOCIATED WITH THE CALIBRATION OF MONITORING EQUIPMENT

VII–14. The calibration of monitoring equipment will also have associated uncertainties. Examples are uncertainties in the activity of calibration standards, in the readout of instruments and in other parameters, such as calibration distances. In most cases, these uncertainties are much less significant than those described in paras VII–2 to VII–13.

OTHER UNCERTAINTIES ASSOCIATED WITH THE CLEARANCE OF MATERIALS AND OBJECTS

VII–15. In addition to the uncertainties described in paras VII–2 to VII–14, there may be other uncertainties that need to be taken into account in specific situations. Reference [VII–1] provides practical guidance and examples relating to the treatment of other uncertainties for decisions on clearance, such as those relating to sampling (e.g. uncertainties relating to the selection of samples and the samples' size and homogeneity). Uncertainties relating to sampling can be greater than the uncertainties associated with measurements. Guidance on treatment of uncertainties relating to sampling is given in Ref. [VII–6].

REFERENCES TO ANNEX VII

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Annex VIII

SCREENING METHOD APPLIED AFTER THE FUKUSHIMA DAIICHI ACCIDENT FOR RECYCLING OF MATERIAL AND DISPOSAL OF WASTE TO LANDFILL

VIII–1. After the Fukushima Daiichi accident, a distinction was made between on-site and off-site (in relation to the nuclear power plant) in terms of the original location of materials and waste and the target location for recycling or disposal to landfill. Consequently, the possible ways to recycle or dispose to landfill were categorized into three types: (1) from off-site to off-site, (2) from on-site to on-site, and (3) from on-site to off-site. Category 3 has not been undertaken yet in Japan.

VIII–2. For category 1, in 2016 the Ministry of the Environment of Japan, which is responsible for regulating off-site contamination, developed values for activity concentrations for recycling of the removed soil. This is an example of recycling of material generated off the site, implemented under Ref. [VIII–1]. The Ministry of the Environment of Japan defined waste with activity concentrations above 8000 Bq/kg of ^{134}Cs and ^{137}Cs as ‘designated waste’, for which the national government is responsible for treatment under Ref. [VIII–1]. The Ministry of the Environment also established a procedure for cancelling this designation when the activity concentration of the designated waste is reduced to 8000 Bq/kg or less owing to radioactive decay. The cancellation of the designation allows for disposal of the waste in landfills. This is not the same as clearance, because the waste is disposed of under the regulatory controls specified in Ref. [VIII–2].

VIII–3. For category 2 (on-site to on-site), in 2017 the Japan Atomic Energy Agency examined the activity concentration of waste generated in the nuclear power plant for recycling on the site, which is under regulatory oversight by the Nuclear Regulation Authority of Japan. This is also not an example of clearance, because the recycled material is still under regulatory control.

VIII–4. Categories 1 and 2 are described in more detail in the remainder of this annex.

CATEGORY 1: FROM OFF-SITE TO OFF-SITE

Example of off-site recycling of soil removed from off-site locations during decontamination works

VIII–5. The Ministry of the Environment of Japan established the Technology Development Strategy for Volume Reduction and Recycling of Removed Soil in April 2016, towards the final disposal of removed soil outside the Fukushima Prefecture. As part of this strategy, the Ministry of the Environment established a basic concept for the safe use of recycled soil, including protection of recovery workers who handle the soil as well as of the public, in June 2016.

VIII–6. In the basic concept, it is clarified that the use of the recycled soil is limited to uses that are intended to last for a long time period and are part of public projects managed by the public authority (e.g. for basic structural material of banking for coastal levees, for disaster prevention purposes on beaches, for road construction). The recycled soil has to be used by the appropriate organization in accordance with the criteria established in Ref. [VIII–1], with exposures from recycled soil being further restricted by shielding using soil covering to ensure the additional exposure from the material is below 1 mSv in a year for workers and the public. The safety assessment by the Ministry of the Environment considered various exposure scenarios to ensure that the exposures to workers and public were 1 mSv in a year or less. Later, the concept was extended to the management of other materials, when facilities were constructed using the recycled materials. In such cases, the appropriate thickness of shielding was ensured in order to meet the same dose criterion.

VIII–7. Various activity concentration criteria below 8000 Bq/kg have been derived (e.g. 7000, 6000, 5000 and 4000 Bq/kg), in accordance with the purpose of the recycling, the shielding conditions and the annual working time for the use of the recycled soil. The value of 8000 Bq/kg is the same as the concentration level given in Ref. [VIII–1] as a screening level for decision making on exemption from regulatory requirements in an existing exposure situation (see annex II to IAEA Safety Standards Series No. GSG-17, Application of the Concept of Exemption [VIII–3]).

Example of disposal of material and waste generated off the site on off-site landfills

VIII–8. The Ministry of the Environment of Japan categorizes waste with an activity concentration over 8000 Bq/kg as designated waste. If the activity concentration exceeds this value, the exposure of workers and the public could exceed 1 mSv in a year, based on the exposure scenarios and dose calculations provided by the Ministry of the Environment. The government is responsible for treating the designated waste since special control is necessary. If the activity concentration of the waste does not exceed 8000 Bq/kg, it can be safely managed using normal treatment methods, as any additional exposures are expected to be 1 mSv in a year or less for both workers and the public.

VIII–9. The Ministry of the Environment of Japan also established a procedure to cancel the designation of the waste as follows:

(a) If the activity concentration of designated waste reduces to 8000 Bq/kg or less owing to radioactive decay, the Minister of the Environment can cancel the designation after consultation with the person or entity storing the designated waste and the person or entity who would become responsible for the management of the material, including transport and disposal in landfills, after cancellation of the designation. The cancellation of the designation is not performed without their acceptance.
(b) After the designation of the waste has been cancelled, the waste is treated by local municipalities or business operators in accordance with the treatment standards in Ref. [VIII–2]. The Ministry of the Environment provides technical and financial support for the treatment as necessary, including explaining that the treatment of the waste is safe, in order to facilitate the disposal of the waste after the designation has been cancelled.

CATEGORY 2: FROM ON-SITE TO ON-SITE

Example of on-site recycling of waste generated on the site

VIII–10. Tokyo Electric Power Company has proposed that contaminated rubble with a surface dose rate of less than 5 μSv/h, which is stored outdoors on the site of the Fukushima Daiichi nuclear power plant, will be recycled for restricted use only within the site. Consequently, the Japan Atomic Energy Agency has started a study on the recycling of rubble on the site. If the rubble can be recycled into construction materials without creating additional effective doses for workers on the site or for the public off the site, that will help reduce the amount of radioactive waste in the future, because clean materials will not need to be brought from off the site.

VIII–11. The activity concentration of material for recycling for restricted use on the site is estimated using a case by case approach, within the context of decommissioning activities at the site as part of an existing exposure situation. This approach is based on the following basic concepts:

(a) The recycling of material on the site should not lead to an undue increase in the effective dose to persons on the site, nor should it prevent future decommissioning activities.
(b) The recycling of material should not lead to an undue radiation risk off the site, taking into account a hypothetical member of the public immediately outside the site boundary.

VIII–12. Figure VIII–1 shows the procedure for estimating the activity concentration of material for recycling on the site. As a first step, the activity concentration in material to be recycled that would give rise to an additional dose rate of 1 μSv/h due to the recycling is determined. A dose rate of 1 μSv/h corresponds to the minimum dose rate on the site, as measured in air at a height of 1 m from the ground surface. The additional effective doses for workers closest to recycled material are not to exceed 2 mSv in a year (10% of the dose limit for workers). There are also two criteria for the protection of the public outside the site boundary. The first criterion is an effective dose along the boundary of less than 1 mSv in a year from all radiation sources on the site after the recycling. The second criterion is that the activity concentrations in groundwater for radionuclides migrated from recycling material do not exceed the operational target value for the boundary between the site and the ocean.

VIII–13. In accordance with the procedure described in para. VIII–12, the Japan Atomic Energy Agency calculated the activity concentrations for recycling material for restricted use for road construction and for the base of concrete buildings on the site [VIII–4, VIII–5]. The results of these calculations for caesium with an activity concentration ratio (^{134}Cs to ^{137}Cs) of 0.209 as of March 2016 are shown in Table VIII–1.

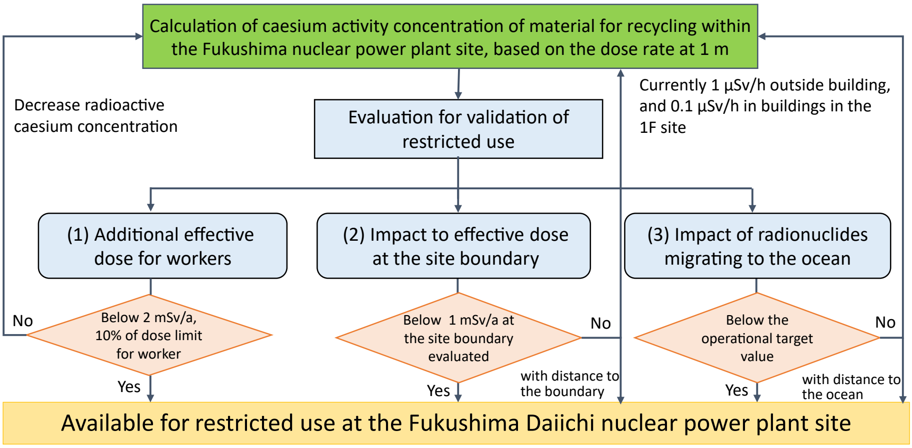

FIG. VIII–1. Procedure for estimating the activity concentration of material for recycling within the Fukushima Daiichi nuclear power plant site.

TABLE VIII–1. ESTIMATED ACTIVITY CONCENTRATIONS FOR RECYCLING CONCRETE FOR RESTRICTED USE: ROAD CONSTRUCTION AND THE BASE OF CONCRETE BUILDINGS ON THE SITE

Application		Activity concentration (Bq/kg)	Shielding provided
Asphalt road	Road bed	13 000	Pavement thickness 5 cm
	Pavement	7400	No shielding
Concrete road	Road bed	100 000	Pavement thickness 15 cm
	Pavement	8100	No shielding
Building concrete	Base	16 000[a]	Floor slab thickness 20 cm

[a] Restricted use in the building based on an effective dose rate of 0.1 µSv/h in the building (scaled from a value of 160 000 Bq/kg, corresponding to 1 µSv/h, calculated by the Japan Atomic Energy Agency).

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Annex IX

ILLUSTRATION OF CONSERVATISM IN THE CLEARANCE PROCESS

IX–1. One of the principal criteria for clearance in relation to radionuclides of artificial origin is that in reasonably foreseeable circumstances the effective dose expected to be incurred by any individual due to the cleared material is of the order of 10 µSv or less in a year (see para. I.11 of IAEA Safety Standards Series No. GSR Part 3, Radiation Protection and Safety of Radiation Sources: International Basic Safety Standards [IX–1]).

IX–2. There are five components to the practical application of clearance, each of which includes a degree of conservatism, as follows:

(a) Application of the dose criterion for clearance to an individual practice;
(b) Conversion from the dose criterion to activity concentration (Bq/g);
(c) Margins associated with practical clearance measurements;
(d) Taking account of multiple radionuclides — radionuclide composition and summation rules;
(e) Activity distribution in cleared material.

APPLICATION OF THE DOSE CRITERION FOR CLEARANCE TO AN INDIVIDUAL PRACTICE

IX–3. The phrase ‘of the order of 10 µSv or less in a year’ is intended to be considered as a trivial dose. In this context, International Commission on Radiological Protection Publication 104 [IX–2] uses the phrase “some tens of microsieverts per year”[1]. A lower boundary value of 10 µSv in a year was used for the derivation of generic clearance levels, since an individual could be exposed to more than one cleared material. Given the wide range of clearance practices and scenarios, with many different representative persons, and the low probability that individuals will be exposed to multiple cleared materials, this approach is likely to be conservative.

[1] This is intended to cover the range 10–100 µSv in a year (see para. 67 of Ref. [IX–2]).

CONVERSION FROM THE DOSE CRITERION TO ACTIVITY CONCENTRATION

IX–4. Scenarios are used to derive activity concentration values for each radionuclide that correspond to the dose criterion for clearance. Many scenarios are considered, with different representative persons. The model used for these scenarios includes two principal groups of parameters: parameters representative of specific scenarios (e.g. duration of exposure, source–person geometry, resuspension factors) and generic parameters (e.g. environmental transfer factors, dose per unit intake). The number of parameters used can vary from four up to a maximum of around 12 (when environmental transfers are involved). The parameter set has to be chosen carefully to avoid overconservativism.

IX–5. It is important to ensure that the parameters chosen are relevant for the representative person; for example, parameters for an adult might not be sufficiently protective for non-adults.

MARGINS ASSOCIATED WITH PRACTICAL CLEARANCE MEASUREMENTS

IX–6. The derived activity concentration values are usually established as legally binding values (clearance levels) in national regulations. To ensure compliance, operating organizations will usually incorporate a degree of conservatism in the programme of clearance measurements.

TAKING ACCOUNT OF MULTIPLE RADIONUCLIDES — RADIONUCLIDE COMPOSITION AND SUMMATION RULES

IX–7. There may be several radionuclides within a material identified for clearance, accounted for by the summation rules described in para. I.14 of GSR Part 3 [IX–1]. In deriving clearance levels, different radionuclides may have different representative persons to consider, so the exposures might not be strictly additive. Hence, the application of the summation rule is conservative.

ACTIVITY DISTRIBUTION IN CLEARED MATERIAL

IX–8. In deriving clearance levels, it is typically assumed that all cleared material contains activity at the derived concentration value. In practice, there is a range of activity concentrations in cleared material, ranging from virtually zero up to a value of the clearance level. Experience shows that it is virtually impossible to have a consistently uniform waste stream at the maximum allowed activity concentration, and the average activity concentration is usually significantly below the maximum allowed.

CUMULATIVE IMPACT OF CONSERVATISMS

IX–9. The conservatisms in the five components of the clearance process accumulate in a multiplicative manner. In broad terms, the overall level of conservatism in some cases could exceed one order of magnitude.

IX–10. Given the level of overall conservatism in the clearance process, as described in this annex, and noting that clearance is an important means of applying a graded approach to protection, it is important that reasonable steps be taken to reduce the degree of conservatism. In practice, for generic clearance, the main ways in which overconservatism can be avoided are in the clearance measurement programme and the approach to the activity distribution in cleared materials. Specific clearance, by definition, is also a means of avoiding overconservatism in the application of dose criteria to individual practices and in the conversion to activity concentration clearance levels.

REFERENCES TO ANNEX IX

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